NRC Exam Legend

Times are HH:MM:SS [00:00:00 – 23:59:59]

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LIT	**DARK**
	

LIGHT INDICATIONS:

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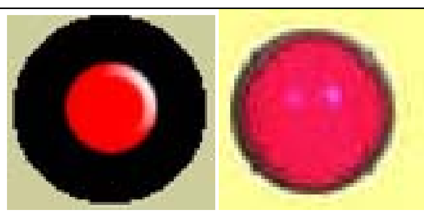
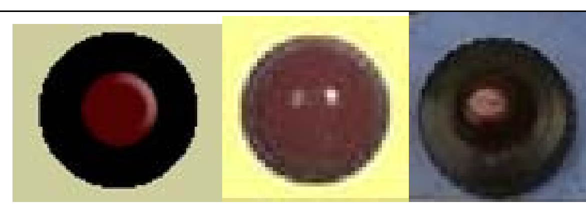
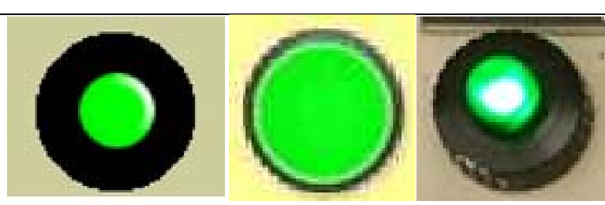
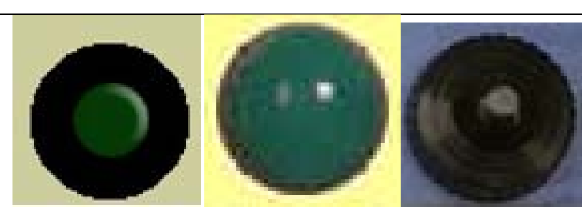
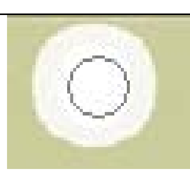
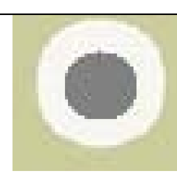
RED		
GREEN		
WHITE		

1.

In accordance with 1-E-0, Reactor Trip or Safety Injection, which ONE of the following describes how the OAC ENSURES the turbine is tripped and the required action(s) if the OAC cannot ENSURE the Turbine Trip?

The OAC is required to ENSURE Turbine Trip using ____(1)____.

If the OAC cannot ENSURE Turbine Trip, 1-E-0 directs them to ____(2)____.

A. (1) indication that all turbine stop valves are CLOSED
   (2) TRIP turbine locally

B. (1) indication that all turbine stop valves are CLOSED
   (2) CLOSE MSIVs and bypasses

C. (1) a **GREEN** light indication at 1-HS-47-24, Turbine Trip
   (2) TRIP turbine locally

D. (1) a **GREEN** light indication at 1-HS-47-24, Turbine Trip
   (2) CLOSE MSIVs and bypasses

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: While it is correct that the OAC should check that the stop valves are closed, it is not correct that 1-E-0 directs that the turbine be tripped locally.

B. Correct: 1-E-0 does require that the OAC check that all turbine stop valves are closed. Also, 1-E-0 does direct that the MSIVs and bypasses be closed given that the turbine failed to trip.

C. Incorrect: As mentioned in the distractor analysis for A, it is inappropriate to regard the indication on 1-HS-47-24 as the indication for turbine trip. Additionally, 1-E-0 does not mention tripping the turbine locally. It is plausible for an applicant to believe that it would be as this is the response not obtained for an immediate action in 1-ECA-0.0. Such procedure dictates: IF turbine can **NOT** be tripped, THEN: TRIP turbine locally: 1) TRIP from front standard. Again, this is a fairly commonly observed mistake in which either an ILT student or a LOR participant will during the immediate actions of 1-E-0, dispatch an AUO to trip the turbine upon the failure of a turbine trip.

D. Incorrect: In accordance with 1-E-0, step 2, the OAC is required to ENSURE Turbine Trip: All turbine stop valves CLOSED. 1-E-0 does not mention checking 1-HS-47-24. It is plausible for an applicant to believe that 1-HS-47-24 is required to be checked as a green light on such is a prime indication of a turbine trip. Additionally, it is a common bad practice for both ILT students as well as LOR participants to rely on the green light on 1-HS-47-24 as an indication of a turbine trip. Often times, persons being evaluated will miss the fact that the turbine did **NOT** trip because they simply look for the green light indication. The distractor is also plausible because 1-E-0 does require (as a RNO for step 2) the OAC to RUNBACK turbine manually OR CLOSE MSIVs and bypasses.

Question Number: 1

Tier: 1 Group: 1

K/A: 007 Reactor Trip
EA1 Ability to operate and monitor the following as they apply to a reactor trip:
EA1.01 T/G controls

Importance Rating: 3.7 3.4

10 CFR Part 55: (CFR 41.7 / 45.5 / 45.6)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to demonstrate that they can correctly monitor the T/G controls during a reactor trip and subsequently operate those controls during the reactor trip response.

Technical Reference: 1-E-0, Reactor Trip and Safety Injection
1-ECA-0.0, Loss of Shutdown Power
.JPG picture of 1-HS-74-24

Proposed references to be provided: None

Learning Objective: 3-OT-EOP0000, Emergency Instructions Reactor Trip Or Safety Injection, 1-E-0, ES-0.1, 1-ES-0.2, ES-0.3 and ES-0.4
23. CITE the immediate action steps including RNO for 1-E-0.

Cognitive Level:
- Higher ____
- Lower X

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Step	Action/Expected Response	Response Not Obtained

3.0 OPERATOR ACTIONS

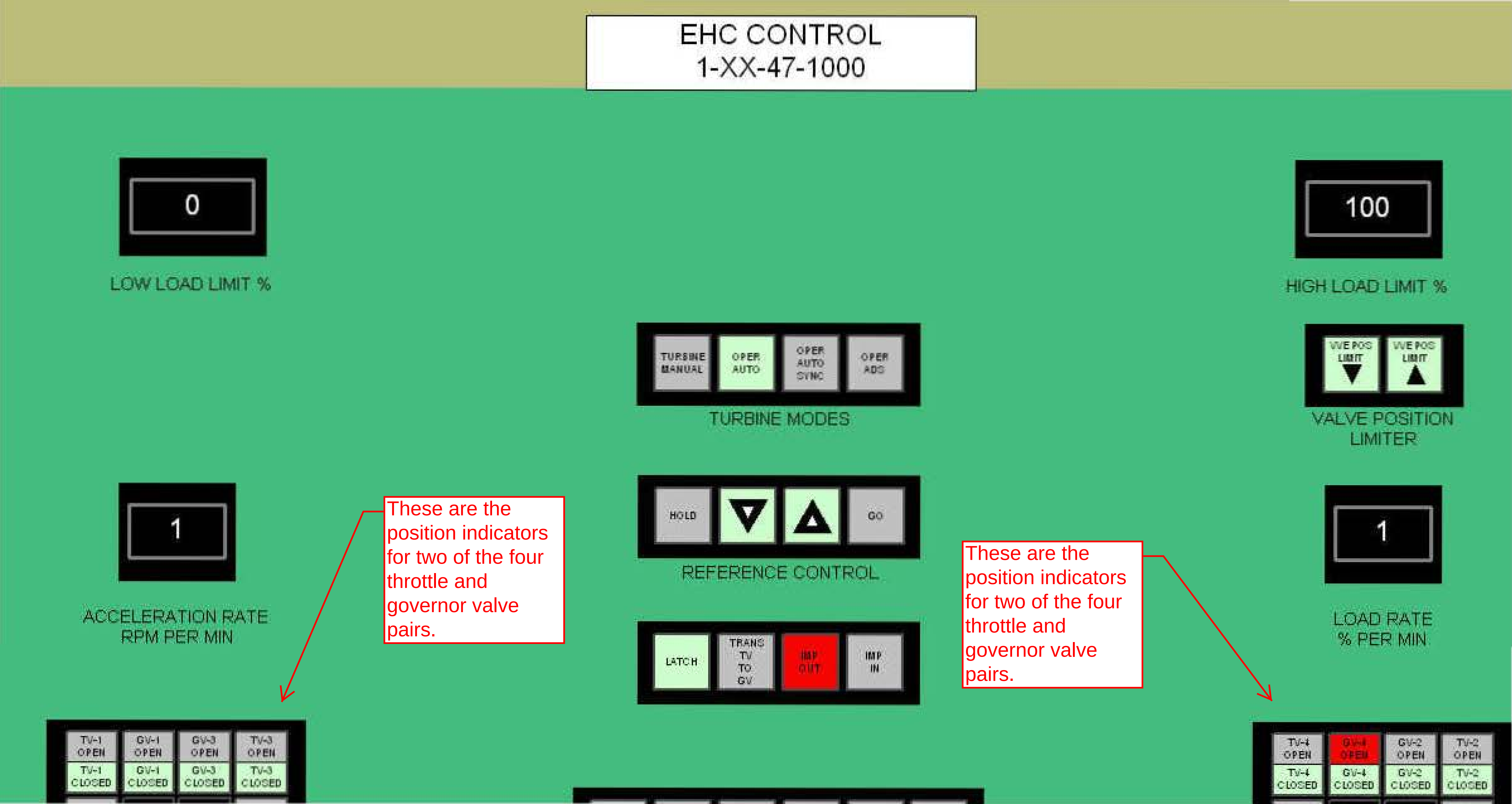
NOTE

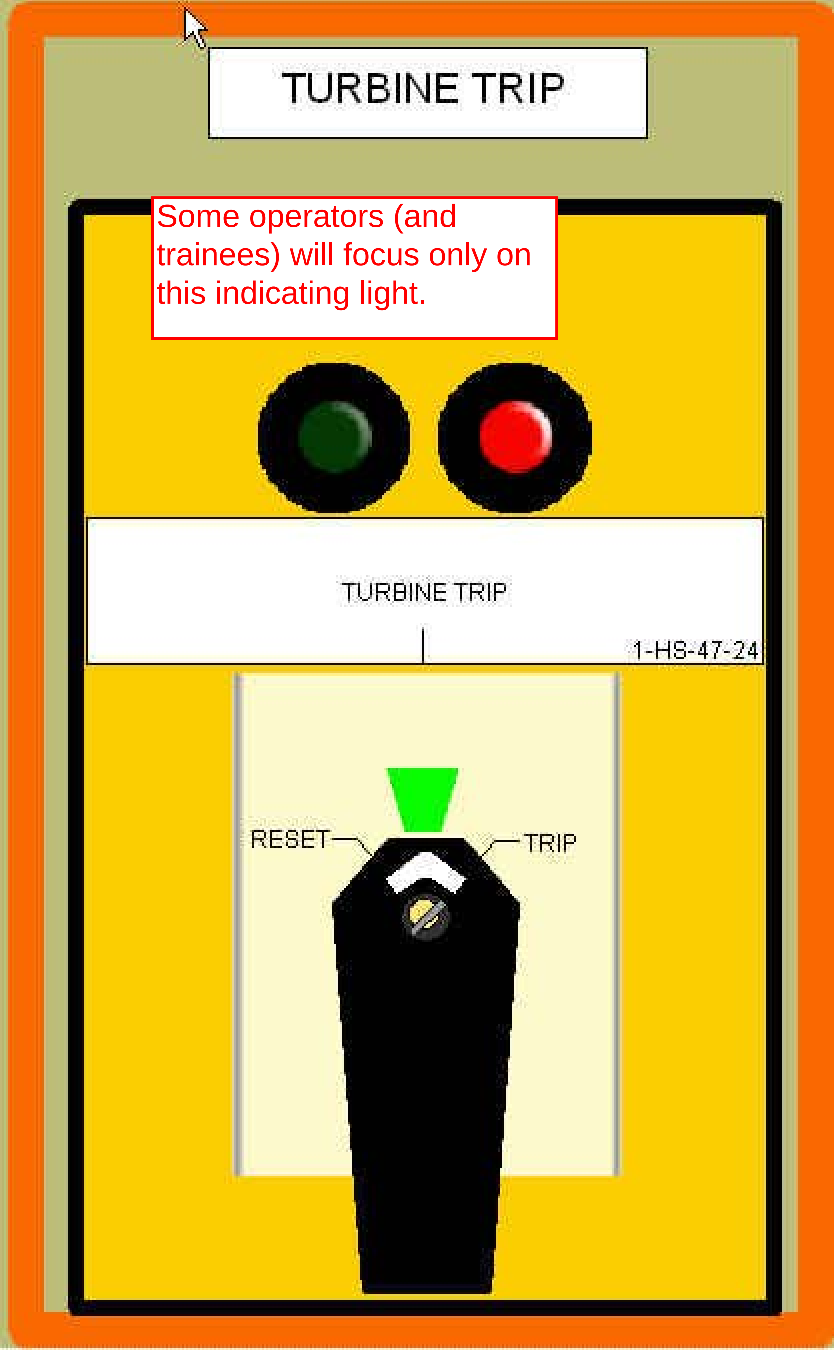
- Steps 1 thru 4 are **IMMEDIATE ACTION STEPS**
- Status Trees / SPDS should be monitored when transitioned to another instruction.

Step	Action/Expected Response	Response Not Obtained
1.	**ENSURE** reactor trip: • Reactor trip and bypass breakers OPEN. • RPIs at bottom of scale. • Neutron flux DROPPING.	Manually **TRIP** reactor. **IF** reactor will **NOT** trip, **THEN** ** **GO TO** 1-FR-S.1, Nuclear Power Generation / ATWS.
2.	**ENSURE** Turbine Trip: • All turbine stop valves CLOSED.	Manually **TRIP** turbine. **IF** turbine will **NOT** trip, **THEN** • **RUNBACK** turbine manually **OR** • **CLOSE** MSIVs and bypasses

This verification is done from the EHC panel. See the attached pictures.

Note that there is nothing about tripping the turbine locally. Any manual RUNBACK is conducted from the MCR.


EHC CONTROL
1-XX-47-1000
0
LOW LOAD LIMIT %
100
HIGH LOAD LIMIT %
TURBINE MANUAL
OPER AUTO
OPER AUTO SYNC
OPER ADS
TURBINE MODES
VALVE POSITION LIMITER
1
ACCELERATION RATE RPM PER MIN
HOLD
GO
REFERENCE CONTROL
1
LOAD RATE % PER MIN
These are the position indicators for two of the four throttle and governor valve pairs.
LATCH
TRANS TV TO GV
IMP OUT
IMP IN
These are the position indicators for two of the four throttle and governor valve pairs.
TV-1 OPEN
GV-1 OPEN
GV-3 OPEN
TV-3 OPEN
TV-1 CLOSED
GV-1 CLOSED
GV-3 CLOSED
TV-3 CLOSED
TV-4 OPEN
GV-4 OPEN
GV-2 OPEN
TV-2 OPEN
TV-4 CLOSED
GV-4 CLOSED
GV-2 CLOSED
TV-2 CLOSED


TURBINE TRIP
Some operators (and trainees) will focus only on this indicating light.
TURBINE TRIP
1-HS-47-24
RESET
TRIP

Step	Action/Expected Response	Response Not Obtained
2.	**ENSURE** turbine trip: • All turbine stop valves CLOSED.	Manually **TRIP** turbine. **IF** turbine can **NOT** be tripped, **THEN**: a. **CLOSE** turbine valves, **OR** b. **CLOSE** MSIVs and bypasses, **OR** c. **TRIP** turbine locally: 1) **TRIP** from front standard. 2) **STOP** and **PULL TO LOCK** both EHC pumps. [Turb Bldg T4J/745]
3.	**ENSURE** RCPs STOPPED, **AND** Momentarily **PLACE** Handswitches in STOP to break seal-in.	

This does not appear in 1-E-0

2.

Given the following conditions:

- Unit 2 is at 100% power.

Subsequently:

- PZR pressure is rapidly LOWERING ⇓.
- The Reactor was MANUALLY tripped.
- PZR level is rapidly RISING ⇑.

Which ONE of the following describes the accident in progress?

A. Ejected control rod

B. SBLOCA on a RCS cold-leg

C. Feedline break inside of containment

D. PZR Level Detector Condensing Pot Rupture

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: Plausible if candidate thought that the LOCA on the top of Rx vessel would cause PZR level to rise, however for all RCS LOCAs, with the exception of PZR vapor space LOCAs, PZR level would lower during the initial phase of the accident.

B. Incorrect: Plausible since SBLOCA will cause RCS pressure to decrease while not initially affecting RCS temperature right after a Rx trip. However PZR level would be rapidly LOWERING⇓ not RISING⇑.

C. Incorrect: Plausible since a Feed line break would cause a rapidly LOWERING⇓ PZR pressure and after SG blowdown PZR level would rise but not initially while water in the SG is blowing down. Also RCS temperature would be rapidly LOWERING⇓ during initial phase of accident.

D. Correct: PZR vapor space LOCAs are the only LOCAs that cause the indicated PZR level to rise rapidly while RCS pressure continues to rapidly decrease.

Question Number: 2

Tier: 1 Group: 1

K/A: 008 Pressurizer Vapor Space Accident
AK2.02 Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Sensors and Detectors

Importance Rating: 2.7*/2.7

10 CFR Part 55: (CFR 41.7)

10CFR55.43.b: Not applicable

K/A Match: This question matches the K/A by asking the candidate how the level detectors would respond to a Pressurizer Vapor Space accident.

Technical Reference: 1-E-0, Reactor Trip and Safety Injection
1-ECA-0.0, Loss of Shutdown Power
.JPG picture of 1-HS-74-24

Proposed references to be provided: None

Learning Objective: 3-OT-TAA013, LOSS OF COOLANT ACCIDENTS
1. Describe the dynamic behavior of the reactor, from a thermodynamic and hydraulic point of view, following a loss of coolant accident (LOCA) for the following categories:
e. Pressurizer vapor space break.

Cognitive Level:
- Higher X
- Lower ____

Question Source:
- New ____
- Modified Bank ____
- Bank X

Question History: Bank question from the WBN Sep 2010 exam.

Comments:

3.

Given the following conditions:

- Unit 1 has experienced a SBLOCA.
- RCS pressure is 1700 psig and STABLE ⇔**.**
- Containment pressure peaked at 1.5 psig.
- Safety Injection Pump 1A-A failed to AUTOMATICALLY start.

Which ONE of the following completes the statements listed below?

In accordance with 1-E-1, Loss of Reactor or Secondary Coolant, the control room staff will control SG NR levels between ____(1)____ and 50%.

Taking **ONLY** the action to ____(2)____ will **RAISE** ⇑ ECCS injection flow rate.

A. (1) 29%

(2) LOWER ⇓ the #2 SG PORV setpoint

B. (1) 39%

(2) LOWER ⇓ the #2 SG PORV setpoint

C. (1) 29%

(2) MANUALLY start the Safety Injection Pump 1A-A

D. (1) 39%

(2) MANUALLY start the Safety Injection Pump 1A-A

CORRECT ANSWER: A

DISTRACTOR ANALYSIS:

A. Correct: In accordance with step 5 of 1-E-1, the operators will CONTROL intact SG levels between 29% and 50% [39% and 50% ADV]. As seen in the WOG background document for 1-E-1 on page 58 (step 3 in the WOG generic E-1 is step 5 in WBNP's 1-E-1), the purpose of maintaining adequate SG level is To ensure adequate feed flow or SG inventory to ensure a secondary heat sink for small and intermediate size LOCAs and secondary break accidents. Lowering a SG PORV setpoint will cause the PORV to open more to control SG pressure at a lower value. Because the S/Gs remain coupled to the RCS (as this is a SBLOCA), the depressurization of the SG will yield an RCS cooldown and thus depressurization. Because of this, injection flow from the centrifugal charging pumps will increase. RCS pressure is stated to be at 1700 psig. The greatest pressure at which the safety injection pumps could have any forward flow is that of the Maximum Composite Curve which is found in WBN-SDD-N3-63-4001, Safety Injection System. This curve shows that the SIPs reach shutoff head at 3560 feet/H20 or 1566 psig. An item which must be considered is the fact that 1-E-0, Reactor Trip or Safety Injection contains the step: "7. **MONITOR** ECCS operation: f. RCS pressure greater than 1650 psig." The response not obtained for such step is to "f. **ENSURE** SI pump flow." Therefore, the question places the applicant not only above the design (as tested) shutoff head of the SIP but also above the Westinghouse EOP setpoint value used in 1-E-0 as well (within the readability of the MCR wide range pressure meter which would be used during an accident).
If the depressurization of the S/Gs continued, the running B train SIP would begin to inject below that value.

B. Incorrect: While it is correct that LOWERING⇓ the #2 SG would increase the injection flow of the ECCS, it is not correct to maintain a SG level band of between 39% and 50%. Such band would only be required if adverse setpoints were warranted on account of a ΦB (containment pressure of 2.8 psig) It is plausible to believe that. ΦB occurred at 1.5 psig-the automatic setpoint of safety injection. It is also plausible to believe that containment pressure rose above 2.8 psig, caused the ΦB and then dropped back to 1.5 psig. However, the stem stipulates that pressure peaked at 1.5 psig.

C. Incorrect: No further ECCS injection would result as RCS pressure is above the shutoff head of the SIP. It is plausible to believe that starting the SIP would increase ECCS injection because doing so would certainly cause such result if done below the shutoff head of the pump.

D. Incorrect: As previously mentioned, it is both Incorrect but plausible to believe that SG level should be maintained between 39% and 50% and that starting the A SIP would increase ECCS flow.

Question Number: 3

Tier: 1 Group: 1

K/A: 009 Small Break LOCA
EK2 Knowledge of the interrelations between the small break LOCA and the following:
EK2.03 SGs

Importance Rating: 3.0 3.3*

10 CFR Part 55: (CFR 41.7 / 45.7)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to demonstrate understanding of the interrelations between the SBLOCA and the SG in both halves of the question. Firstly, the applicant must understand that the SBLOCA requires that a certain level band be maintained in the S/G. Secondly, the applicant must be able to understand the effect that changing pressure in the SG has on the primary side parameters.

Technical Reference: 1-E-1, Loss of Reactor or Secondary Coolant
WBN-SDD-N3-63-4001, Safety Injection System
WOG background document for 1-E-1

Proposed references to be provided: None

Learning Objective: 3-OT-EOP0100, 1-E-1, Loss Of Reactor Or Secondary Coolant
5. EXPLAIN the purpose and basis of each step of 1-E-1, 1-ES-1.1, ES-1.2, ES-1.3 and ES-1.4.
8. Given a set of plant conditions IMPLEMENT the Action Steps, RNOs, Foldout Pages, Notes, and Cautions of 1-E-1, 1-ES-1.1, ES-1.2, ES-1.3, and ES-1.4

Cognitive Level:
Higher X
Lower ____

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam
Comments:

Step	Action/Expected Response	Response Not Obtained

5. **MAINTAIN** Intact S/G NR levels:

a. **MONITOR** levels greater than 29% [39% ADV].

a. **MAINTAIN** total feed flow greater than 410 gpm UNTIL level is greater than 29% [39% ADV] in at least one S/G.

Containment pressure peaked at a value less than phase B. Therefore, normal setpoints are in effect.

b. **CONTROL** intact S/G levels between 29% and 50% [39% and 50% ADV].

b. **IF** level in any intact S/G continues to rise without feed flow,
THEN

** **GO TO** 1-E-3, Steam Generator Tube Rupture.

6. **CHECK** secondary radiation:

- S/G discharge monitors NORMAL.
- Condenser vacuum exhaust rad monitors NORMAL.
- S/G blowdown rad monitor recorders NORMAL trend prior to isolation.

IF rad monitors **NOT** available,
THEN

a. **NOTIFY** RADPROT to survey main steamlines and S/G blowdown lines.

b. **NOTIFY** Chemistry to sample S/G activity.

IF radiation is high,
THEN

** **GO TO** 1-E-3, Steam Generator Tube Rupture.

STEP: Check Intact SG Levels

PURPOSE:

- o To ensure adequate feed flow or SG inventory to ensure a secondary heat sink for small and intermediate size LOCAs and secondary break accidents
- o To provide a positive static head of water to prevent primary to secondary leakage for a large LOCA

The stem of the Q lists that RCS pressure is stable at 1700 psig. This indicates that a small size LOCA exists. Therefore, the steam generators continue to remove core decay heat because they remain coupled to the RCS.

BASIS:

The minimum feed flow requirement satisfies the feed flow requirem Heat Sink Status Tree until level in at least one SG is restored i narrow range. Narrow range level is reestablished in all SGs to m symmetric cooling of the RCS. The control range ensures adequate with level readings on span. The transition to E-3, STEAM GENERAT RUPTURE, responds to an increasing level which would be observed f SGTR.

Even though a secondary heat sink is not required for large LOCAs, it is beneficial to maintain SG narrow range levels on span to ensure a positive static head of water between the secondary and primary sides of the SG since these SGs will eventually be depressurized (either later in this guideline or by natural heat losses). Water level in the SGs will prevent (or minimize) primary to secondary leakages from the intact SGs which may have leakage paths that are within the limits of the Technical Specifications.

ACTIONS:

- o Determine if SG narrow range level is greater than (M.02)% [(M.03)% for adverse containment]
- o Determine if SG narrow range level in any SG increases in an uncontrolled manner
- o Maintain total feed flow greater than (S.02) gpm until narrow range level greater than (M.02)% [(M.03)% for adverse containment] in at least one SG
- o Control feed flow to maintain narrow range level between (M.02)% [(M.03)% for adverse containment] and 50%
- o Transfer to E-3, STEAM GENERATOR TUBE RUPTURE, step 1

INSTRUMENTATION:

- o SG narrow range level indication
- o Total feed flow indication
- o Feed flow control valves position indication

Appendix A
(Page 3 of 11)

Equipment Verification

Step	Action/Expected Response	Response Not Obtained
7.	**MONITOR** ECCS operation:	
	a. Charging pumps RUNNING.	a. Manually **START** charging pumps.
	b. Charging pump alignment: • RWST outlets 1-LCV-62-135 and 1-LCV-62-136 OPEN. • VCT outlets 1-LCV-62-132 and 1-LCV-62-133 CLOSED. • Charging 1-FCV-62-90 and 1-FCV-62-91 CLOSED.	b. **ENSURE** at least one valve in each set aligned.
	c. RHR pumps RUNNING.	c. Manually **START** RHR pumps.
	d. SI pumps RUNNING.	d. Manually **START** SI pumps.
	e. BIT alignment: • Outlets 1-FCV-63-25 and 1-FCV-63-26 OPEN. • Flow thru BIT.	e. **ENSURE** at least one valve aligned, and flow thru BIT.
	f. RCS pressure greater than 1650 psig.	f. **ENSURE** SI pump flow. **IF** RCS press drops to less than 150 psig, **THEN** **ENSURE** RHR pump flow.

This is an oddity which must be accounted for; otherwise, someone could claim that the SIP could inject less than this value.

8.2 Figure 2 - SIP Composite Curve

The "best" case for the shutoff head for a SIP is 1566 psig.

	Maximum Composite Curve	FSAR	Test Acceptance Curve
Flow (gpm)	Head (ft)	Head (ft)	Head (ft)
0	3560	3345	3428
50	3505	3300	3383
100	3495	3290	3373
150	3460	3240	3323
200	3415	3200	3283
250	3355	3150	3233
300	3250	3050	3133
350	3105	2905	2988
400	2990	2750	2833
450	2795	2550	2633
500	2600	2380	2463
550	2395	2175	2258
600	2190	1980	2063
650	1990	1725	1808

Note: Based on a 3 % head (2750 ft X 3 %) margin between the Test Acceptance Curve and the FSAR Curve.

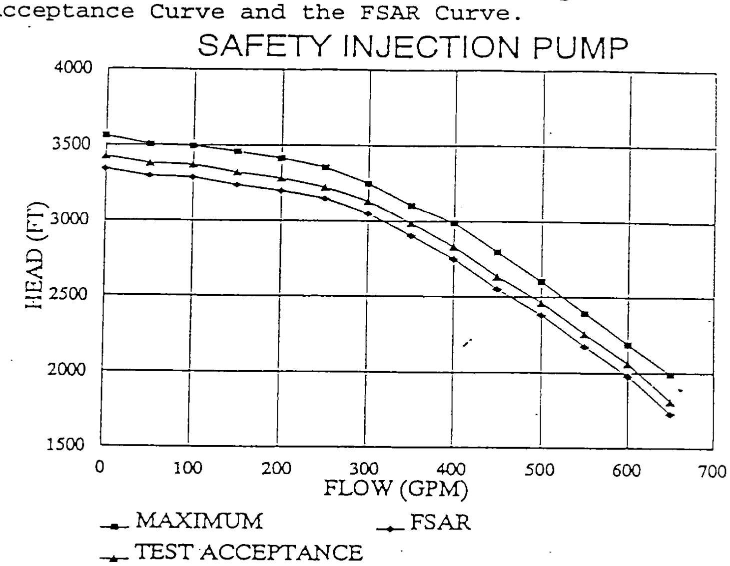


FIGURE 2 SIP COMPOSITE CURVE
(REF 7.5.84)

4.

Which ONE of the following describes the method used to remove decay heat 5 minutes after a LBLOCA has occurred?

5 minutes after the LBLOCA occurs; decay heat is removed by ________.

A. Reflux boiling

B. RCS Hot Leg recirculation

C. Natural circulation and SG steaming

D. ECCS injection and inventory loss from the break

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: As seen in the Westinghouse E-1 background document, reflux boiling is credited for heat removal during a LOCA of a size greater than 1” in diameter to less than 13 ½” (1 square foot) in diameter. For LOCAs in excess of these dimensions (i.e. LBLOCAs), the secondary plant is not credited for any heat removal.

B. Incorrect: While it is true that during a LBLOCA decay heat removal is afforded by the ECCS; the use of Hot leg recirculation will not be made until 3 hours after the event has occurred.

C. Incorrect: As seen in the Westinghouse E-1 background document, the coupling of the RCS to the S/Gs is lost most probably before the 15 second point for a Large Break LOCA. Therefore, it is incorrect that natural circulation would be possible during a large break LOCA.

D. Correct: in accordance with the WOG background document for 1-E-1, during a LBLOCA, decay heat is removed by a continuous supply of water from the ECCS

Question Number: 4

Tier: 1 Group: 1

K/A: Knowledge of the operational implications of the following concepts as the apply to the Large Break LOCA:

Natural circulation and cooling, including reflux boiling.

Importance Rating: 3.0 3.3*

10 CFR Part 55: (CFR 41.7 / 45.7)

10CFR55.43.b: Not applicable

K/A Match: The K/A is matched because the applicant must understand the applicability of secondary plant heat removal (natural circulation and reflux boiling) to a Large Break LOCA.

Technical Reference: 1-E-1, Loss of Reactor or Secondary Coolant
WBN-SDD-N3-63-4001, Safety Injection System
WOG background document for 1-E-1

Proposed references to be provided: None

Learning Objective: 3-OT-TAA013, LOSS OF COOLANT ACCIDENTS
1. Describe the dynamic behavior of the reactor, from a thermodynamic and hydraulic point of view, following a loss of coolant accident (LOCA) for the following categories:
f. Large-break LOCA, >1 square foot and up to double ended guillotine shear.

Cognitive Level:
Higher ____
Lower X

Question Source:
New ____
Modified Bank ____
Bank X

Question History: Bank Question imported from San Onofre Nuclear Generating Station

Comments:

Breaks ~ 1" < diameter <~ 13-1/2" (1FT^2)

For break sizes of one to two-inch in equivalent diameter, the RCS will rapidly depressurize early in the transient, and an automatic reactor trip and safety injection signal will be generated based on low pressurizer pressure. During the early stages of the depressurization, when the system is still full of two-phase liquid, the break flow, which also will be mostly liquid, is not capable of removing all the decay heat. Therefore, the early depressurization is limited by energy removal considerations, and the RCS pressure will temporarily hang up above the steam generator safety valve set pressure, assuming no steam dump is available. The RCS pressure stays at this level in order to provide a temperature difference from primary to secondary so that core heat may be removed by the steam generator. At this energy-balance controlled pressure, however, pumped safety injection flow is less than th break flow, and there is a net loss of mass in the RCS. Voiding throughou the primary side occurs and eventually the RCS begins to drain, starting f the top of the steam generator tubes. The rate of RCS drain is determined the net loss of liquid inventory, a function of both SI flow and break siz

Note that for breaks up to a large break LOCA (>13 1/2"), decay heat is removed by the secondary, first by natural circulation and then reflux.

Prior to the occurrence of draining, heat is removed from the steam genera through continuous two-phase natural circulation, with two-phase mixture flowing over the top of the steam generator tubes. As the draining continues, the natural circulation mode of heat removal as just defined ceases, and core heat is removed through condensation of steam in the steam generator. This method of heat removal is called reflux and is discussed in Reference 2.

The condensation mode of heat removal is almost as efficient as continuous two-phase natural circulation in removing heat. However, condensation heat transfer coefficients may be lower than continuous two-phase natural circulation heat transfer coefficients. Thus, as the steam generator tubes drain, a slight increase in primary system pressure occurs to give a greater ΔT from primary to secondary in order to remove all the decay heat. The steam generator secondary side pressurizes to the safety valve set pressure would correspond to an event which is approaching the design basis assumptions. It should be noted that more probable delivery rates for the safety injection system (e.g., maximum safeguards or minimum safeguards with no spilling line) will yield less core uncovery or no core uncovery.

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

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

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

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

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

Nothing in here about secondary heat removal.

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

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

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

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

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

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

As seen later, this is at about the 1 minute point

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

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

TABLE 1

TIME SEQUENCE OF EVENTS FOR LARGE BREAK*

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

Note that at the one minute point, all heat removal is via the ECCS. The secondary cannot provide any heat removal as the S/G tube inventory is lost (most probably at the < 15 sec point).

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

5.

Given the following conditions:

- Unit 1 is at 100% power.
- 1-FCV-62-93, CHARGING HEADER FLOW is in AUTOMATIC.
- Letdown flow is 75 gpm.

Subsequently:

- A failure causes 1-HIC-62-93A, CHARGING FLOW PZR LEVEL CONTROL to output a FULLY CLOSED signal.

Which ONE of the following describes the **MINIMUM** indicated CHARGING FLOW AND the response of Annunciator CHARGING FLOW HI/LO (108-A)?

The **MINIMUM** which 1-FI-62-93A, CHARGING FLOW will read is ____(1)____ gpm.

Annunciator 108-A, will alarm at ____(2)____ value as it would **IF** letdown flow **HAD BEEN** 45 gpm.

A. (1) 0

(2) the SAME

B. (1) 0

(2) a DIFFERENT

C. (1) 33.5

(2) the SAME

D. (1) 33.5

(2) a DIFFERENT

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Correct: Considering SSD-1-LPF-62-93A, one will observe that the instrument air to 1-FCV-62-93 passes through a pneumatic relay – 1-FM-62-93E. This relay outputs an air signal to provide a minimum flow of 33.5 gpm. It does this by ensuring that a minimum air pressure is always applied to the FAIL OPEN 1-FCV-62-93. Additionally, 1-HIC-62-93A possesses software restraints which preclude it from outputting a signal less than 25.6%. This provides a minimum charging flow of 55 gpm when the controller is in automatic. It is plausible to believe that if a full closed signal were applied to 1-FCV-62-93 that it would drive full closed. This is the normal occurrence with other controllers on the main control boards. Also, in accordance with the ARI for annunciator 108-A, one may witness that three setpoints actuate such alarm. A high charging flow with a setpoint of 150 gpm exists and a low charging flow (driven by the software bistable in DCS – 1-FS-62-93B). The low charging flow receives input from the position of the letdown orifices valves. If either of the 75 gpm orifices is open, then the DCS software will utilize a 55 gpm setpoint. If both of the 75 gpm orifices are closed, then the DCS software will use a 47 gpm setpoint. Therefore, it is true that the 108-A will alarm at different values for letdown flows of 75 and 45 gpm respectively. It is plausible to believe that the LOW charging alarm functions in a similar fashion to the high charging alarm; namely, that it possesses a single setpoint.

B. Incorrect: While it is true that two setpoints exist which drive the 108-A annunciator, it is not true that charging flow would indicate 0 gpm.

C. Incorrect: While it is correct that charging flow would indicate a minimum of 33.5 gpm. It is not true that only one setpoint exists for annunciator window 108-A.
Note that for this question the value of 33.5 gpm was chosen. It is possible for the software of DCS to fail in such a manner that the restricting charging flow to greater than 55 gpm while in automatic is not operable.

D. Incorrect: As mentioned two setpoints exists for the window of concern and charging flow would indicate 33.5 gpm

Question Number: 5

Tier: 1 Group: 1

K/A: 022 Loss of Reactor Coolant Makeup
AA1. Ability to operate and / or monitor the following as they apply to the Loss of Reactor Coolant Makeup:
AA1.02 CVCS charging low flow alarm, sensor, and indicator

Importance Rating: 3.0 2.9

10 CFR Part 55: (CFR 41.7 / 45.5 / 45.6)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to monitor both the CVCS charging low flow alarm and the indicator (charging flow) during a loss of reactor coolant makeup. The failure of the 1-FCV-62-93 valve as described in the stem does precipitate a loss of makeup as charging flow is insufficient to maintain letdown.

Technical Reference: SSD-1-LPF-62-93A
1-ARI-102-108, HVAC & CVCS

Proposed references to be provided: None

Learning Objective: 3-OT-STG-062A, CHEMICAL AND VOLUME CONTROL SYSTEM
3. Given plant conditions, DETERMINE if any of the following CVCS alarms would be present and actions required by the ARI :
c. [1-XA-55-5C 108A] CHARGING FLOW HI/LO

Cognitive Level:
- Higher ____
- Lower X

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

SCALING AND SETPOINT DOCUMENT
LOOP COVER SHEET

Loop Name: Charging Header Flow

Applicable Tech Specs:

Loop Components	Location
1-FC-62-93A	1-R-18
1-FCV-62-93	692, A5-U
1-FI-62-93A	1-M-5
1-FI-62-93B	1-L-112
1-FM-62-93A	1-R-18
1-FM-62-93B	1-L-112
1-FM-62-93D	1-R-18
1-FS-62-93A/B	1-R-18
1-FS-62-93B/A	1-R-18
1-FT-62-93A	1-L-61
1-HIC-62-93A	1-M-5
1-HIC-62-93B	1-L-112
1-HIC-62-93C	1-L-10
1-HIC-62-93D	1-R-18
1-PX-62-93A	1-R-18
Computer Pt F0128A	MCR
1-PI-62-93	1-L-112
1-FM-62-93E	1-L-112

Even if one drives 1-HIC-62-93 to the fully closed position, one will have a minimum charging flow (33.5 gpm).

This pneumatic relay (or stop) was installed to prevent an appendix R fire from destroying seal flow.

The only time that this pneumatic relay is bypassed is during outages (for solid plant control).

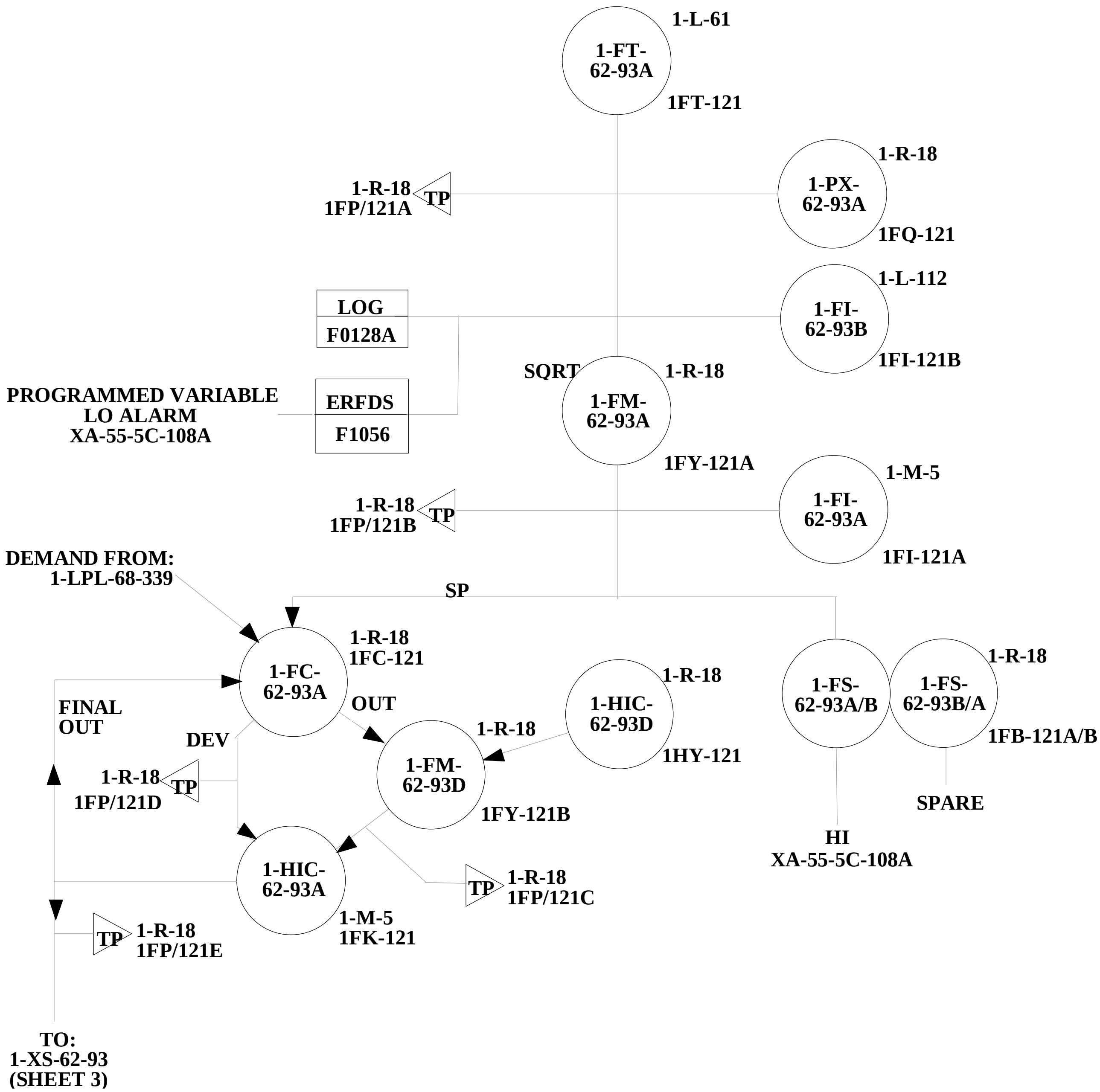
Associated Drawings: 45W600-62; 45W1635-52 & 103; Foxboro Dwg 1-47043, CD 18; Westinghouse Dwg 108D408 SH 27; 47W600-118

Remarks:
(1) In automatic operation, the output of 1-FC-62-93A is effectively limited by the action of 1-FM-62-93D and 1-HIC-62-93D. This provides a minimum charging flow of 55 GPM. Pneumatic relay 1-FM-62-93E output provides a minimum flow of 33.5 GPM for Appendix R.
(2) DCN M-33222-A (F-38016-A) provided pneumatic relay (1-FM-62-93E) and setpoint, bypass valve and pressure indicator (1-PI-62-93).
(3) DCN W-36890-A removed the mechanical travel stop previously added by DCN M-17644-A.
(4) DCN W-39199-A & F-39315-A disconnected output of 1-FS-62-93B/A and added a programmed variable low alarm from ERFDS. The low alarm will have a setpoint of 55 GPM if a 75 GPM orifice is in use OR will have a setpoint of 47 GPM if the 45 GPM orifice (PD Pump) is being used.
(5) DCN 51036 replaced obsolete Foxboro E13DH transmitter with Rosemount transmitter.
(6) DCN 50918 and PIC 51462 revised NE SSD Allowable Value.
(7) DCN 51812 changed 1-PREG-62-93 to 48 psig -0/+2 psig.
(8) DCN 51934 replaced actuator body and trim for 1-FCV-62-93 and revised 1-PREG-62-93 and 1-FM-62-93E setpoints.

Reviewed by: Mike Johnson Approved by: R. S. Henderson Date: 02/28/2011

SCALING AND SETPOINT DOCUMENT
LOOP DRAWING

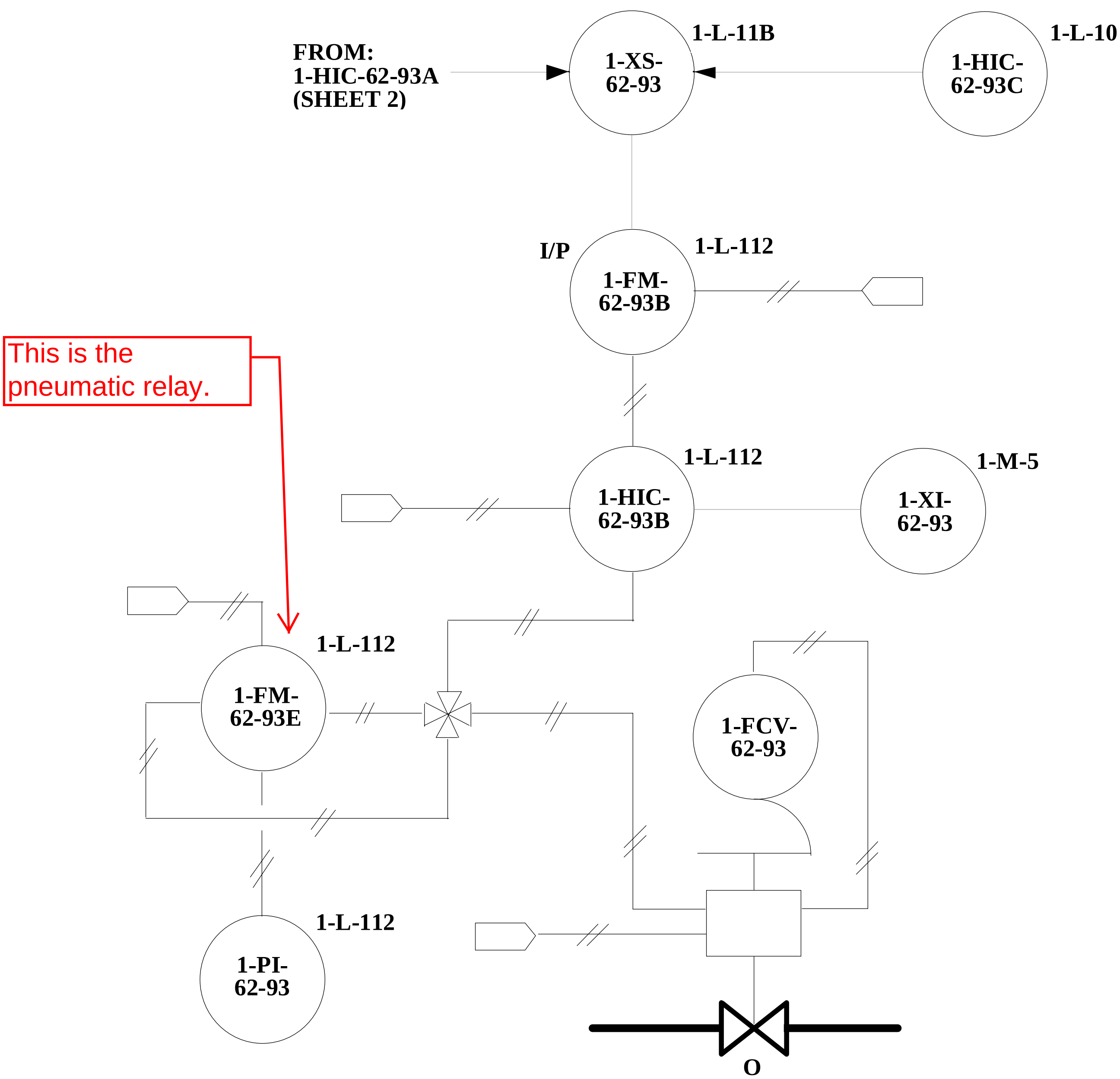
LOOP NAME: CHARGING HEADER FLOW



Reviewed by: Mike Johnson Approved by: R. S. Henderson Date: 02/28/2011

SCALING AND SETPOINT DOCUMENT
LOOP DRAWING

LOOP NAME: CHARGING HEADER FLOW



Reviewed by: Mike Johnson Approved by: R. S. Henderson Date: 02/28/2011

108-A

CHARGING FLOW HI/LO

(Page 1 of 1)

Source	**Setpoint**
HI: 1-FS-62-93A (DCS)	150 gpm
LO: 1-FS-62-93B (DCS)	47 gpm with BOTH 75 gpm orifice valves CLOSED OR 55 gpm with EITHER 75 gpm orifice valve OPEN

Note that this would be the setpoint with 45 gpm letdown

This is the setpoint for 75 gpm letdown

Probable Cause:

A. System pipe break
B. Charging pump tripped
C. Malfunction of Pressurizer Level Control System

Corrective Action:

[1] **IF** ALL the following conditions exist:
- Any RCP Thermal Barrier Out-of-Service
- In-Service Charging pump trips
- RCP seal injection flow required

THEN
IMMEDIATELY START available charging pump to restore seal flow.

[2] **CHECK** 1-FI-62-93A, CHARGING FLOW [1-M-5] to determine if flow is high or low.
[3] **CHECK** 1-LI-68-320, -335A, and -339A, PZR LEVEL [1-M-4].
[4] **IF** PZR Level Control System malfunctions, **THEN**
GO TO 1-AOI-20.
[5] **IF** charging flow is low, **THEN**
CHECK letdown temperature **AND**
EVALUATE increasing charging flow **OR**
ISOLATE letdown.
[6] **IF** charging is lost, **THEN**
IMMEDIATELY ISOLATE letdown **AND**
REFER to 1-AOI-20.

[7] **DETERMINE** cause of problem **AND**
INITIATE corrective action, as necessary.
[8] **REFER TO** 1-SOI-62.01 for CVCS system operation.

References:

1-47W610-62-2
1-47W809-1
1-AOI-20
1-SOI-62.01
08F734235-FD-1605

6.

Given the following conditions:

- Tavg is 150°F following a refueling outage.
- Neither 1-PCV-68-334 nor 1-PCV-68-340 will OPEN.
- Unit 1 RCS pressure is 250 psig and RISING⇑ uncontrollably.

In accordance with N3-74-4001, Residual Heat Removal System description, which ONE of the following completes the statement below?

System design will limit RCS pressure to _____(1)_____ psig.

A. 370

B. 405

C. 450

D. 600

CORRECT ANSWER: C

DISTRACTOR ANALYSIS:

A. Incorrect: The stem of the question places the Unit at 150°F (MODE 5) and thus implicitly requires that RHR be in service. The stem then states that pressure is 250 psig and rising uncontrollably. This indicates that a Loss of RHR is occurring (as seen in section 3.4 of 1-AOI-14, Loss of RHR Shutdown Cooling which has an entry diagnostic of RCS High Pressure during RHR Shutdown Cooling).

It is plausible to believe the 370 psig is correct because this value is the setpoint for a contact of a SOR pressure switch which provides an interlock for the opening of the RHR suction valves (and thus provides protection from overpressure conditions). It is not correct because this interlock does nothing but prevent the RHR suction valves from opening. If as in this case, the valves are already open and the system is in service, then the system will not be further protected from an overpressure condition (i.e. the main suction valves will not automatically close).

B. Incorrect: The value of 405 psig is the setpoint for the pressurizer PORV 1-PCV-68-334 for an RCS temperature of 150°F. If the COMS system had performed its function successfully, then this would be the highest pressure which the RCS experienced. However, as the stem of the question related, neither PORV operated and as such this value is not the limiting RCS pressure observed.

C. Correct: As seen on print 1-47W810-1, the 1-RFV-74-505 has a lift setpoint of 450 psig. As seen in system description N3-74-4001, “RESIDUAL HEAT REMOVAL SYSTEM,” this valve protects the RHR system from RCS over pressurization while the RHR system is in operation.

D. Incorrect: As seen on print 1-47W811-1, the discharge reliefs for the RHR pumps have a lift setpoint of 600 psig. If 1-RFV-74-505 failed to operate, RCS pressure would be controlled by these relief valves at their lift setpoint.

Question Number: 6

Tier: 1 Group: 1

K/A: 025 Loss of Residual Heat Removal System (RHRS)
AA2. Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System:
AA2.06 Existence of proper RHR overpressure protection

Importance Rating: 3.2* 3.4*

10 CFR Part 55: (CFR: 43.5 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: The K/A is matched because the applicant must ascertain the correct pressure that the RCS will stabilize at (an thus verify the existence of RHR overpressure protection) given that section 3.4 of 1-AOI-14, Loss of RHR Shutdown Cooling is in effect.

Technical Reference: 1-AOI-14, Loss of RHR Shutdown Cooling
1-47W810-1
1-47W811-1
1-ARI-109-115
N3-74-4001

Proposed references to be provided: None

Learning Objective: 3-OT-STG-074, OPERATIONS THE RESIDUAL HEAT REMOVAL SYSTEM
1. DESCRIBE the design criteria, purpose and/or functions of the Residual Heat
Removal System (RHR) and the major system components listed below: (IER
11-3: Having a solid understanding of plant design, engineering principles,
and sciences):
g. RHR suction piping relief valve, 74-505

Cognitive Level:
- Higher X
- Lower ____

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

3.0 OPERATOR ACTIONS

3.1 Diagnostics

IF	**GO TO** Subsection
RHR Pump Trip During Midloop Operation.	3.2
RHR Pump Cavitation During Midloop Operation.	3.3
RCS High Pressure during RHR Shutdown Cooling.	3.4
RHR Pump 1A-A trip.	3.5
RHR Pump 1B-B trip.	3.6
RHR System Outleakage or LOCA While On Shutdown Cooling.	3.7
Loss of CCS to RHR System.	3.8
RCS Alternate Cooling Method With the Rx Vessel Head Installed	3.9
RCS Alternate Cooling Method With the Rx Vessel Head Off	3.10
In the event of an extended loss of AC power while in Mode 5 or 6, the use of portable FLEX equipment should be considered.	

Source

1-PS-68-63B (PB406BX)

1-PS-68-64B (PB407BX)

Setpoint

See Probable Cause

This alarm is driven by one contact of a SOR pressure switch. At 375 psig the pressure switch will cause this alarm (given the valves are open). At 370 psig (off of another contact on the same pressure switch) the valves will be prevented from opening if they are not already.

113-E

RHR SUCT
FCV-74-1, 2, 8, 9
OPEN &
HI PRESS

(Page 1 of 1)

NOTE

1-FCV-74-1, -2, -8 and -9 will **NOT** auto close on HI PRESS. If **NOT** open, these valves are prevented from being opened.

Probable Cause:

A. 1-FCV-74-1, -2, -8 and -9 **NOT** fully closed with RCS pressure 375 psig or more

B. Pressure Switch deenergization

CAUTION

RHR Pump suction relief valve lifts at 450 psig.

Corrective Action:

[1] **IF** RCS water solid, **THEN**
STOP Charging pump(s).

[2] **IF** RHR operating and RCS pressure reaches 450 psig, **THEN**
GO TO 1-AOI-14, LOSS OF RHR SHUTDOWN COOLING.

[3] **REDUCE** pressure below 345 psig:

- CONTROL Charging & Letdown
- CONTROL PZR Heaters
- CONTROL RHR Letdown

NOTE

Step [4] is **N/A** if RHR suction relief valve satisfies TS 3.4.12, Cold Overpressure Mitigation System.

[4] **IF** RHR is **NOT** operating, **THEN**
ENSURE the following valves CLOSED:

- 1-FCV-74-1, and -2, LOOP 4 HL TO RHR SUCTION
- 1-FCV-74-8 and -9, RHR SYSTEM ISOLATION BYPASS

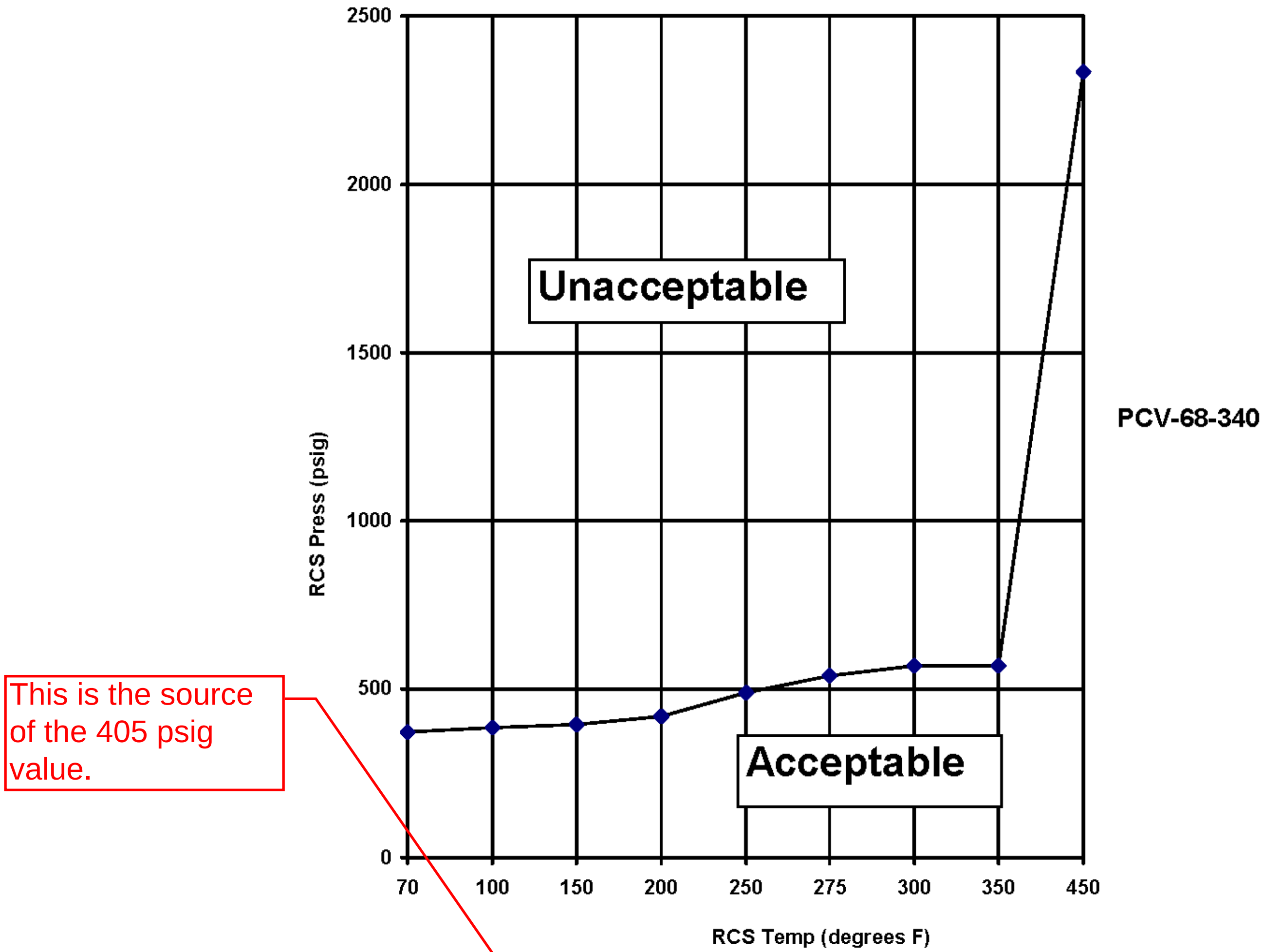
[5] **VERIFY** STATUS of 1-PS-68-63B and 1-PS-68-64B.

References:

1-45W600-57-17
1-45W760-74-2, -4
1-AOI-14

Appendix D
(Page 1 of 1)

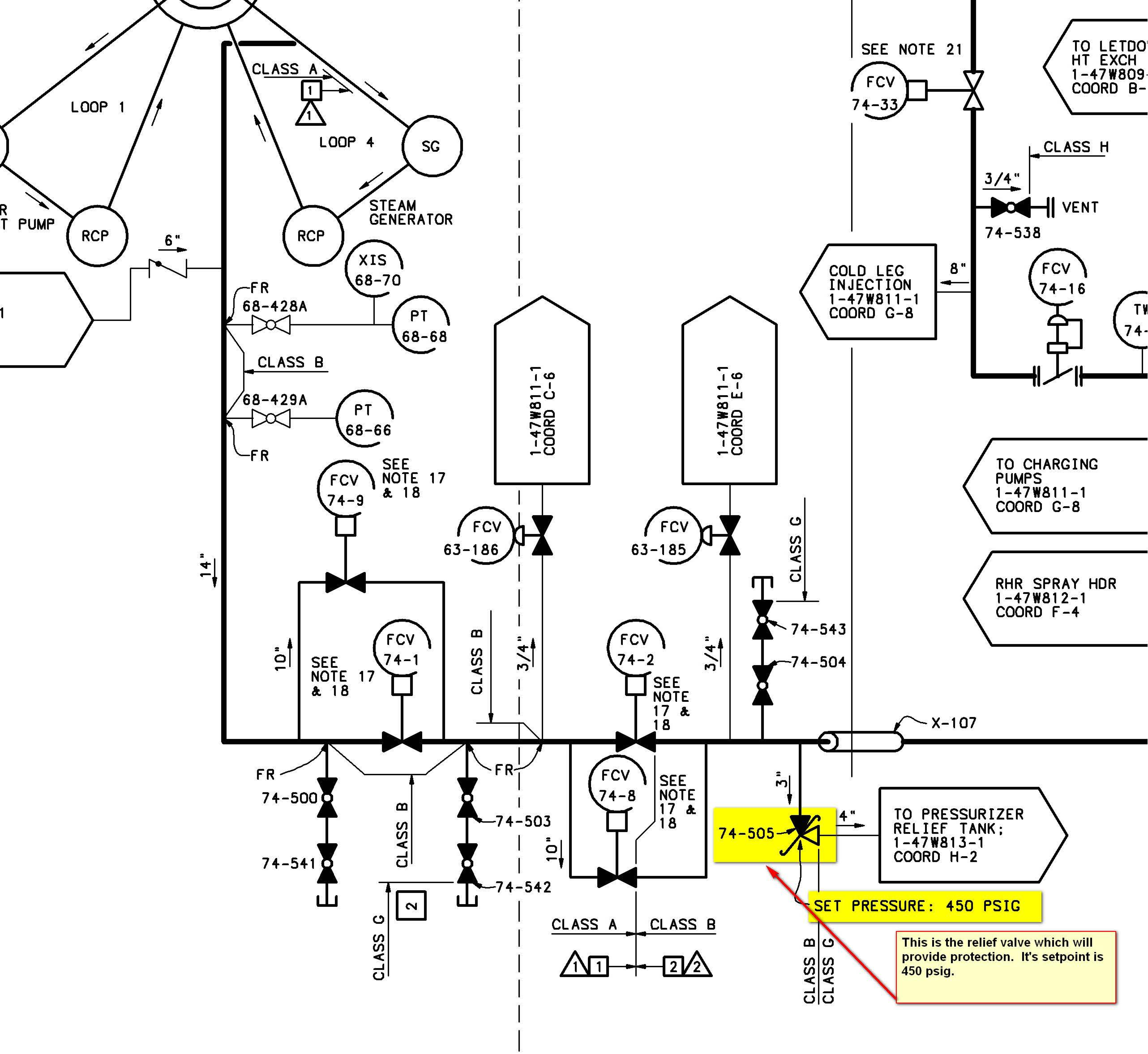
Cold Overpressure Limit Curve



NOTE: The values at right are for ≤ 4 RCPs in operation with RCS temp. ≥ 105°F. The most conservative values (lowest pressure, first to open) for PCV-68-340 were used to generate the curve above. RCS press should not approach RCS press setpoints marked with an asterisk due to #1 seal ΔP concerns.

Temp °F	PCV 334 Setpoint, psig	PCV 340 Setpoint, psig
70	383*	373*
100	386	376*
150	405	395
200	430	420
250	510	490
275	560	540
300	590	570
350	590	570
450	2335	2335

* Setpoint violates pump seal limit.



LOOP 1
CLASS A
LOOP 4
SG
STEAM GENERATOR
RCP
RCP
T PUMP
6"
XIS 68-70
FR 68-428A
PT 68-68
CLASS B
68-429A
PT 68-66
FR
14"
FCV 74-9
SEE NOTE 17 & 18
1-47W811-1 COORD C-6
1-47W811-1 COORD E-6
FCV 63-186
FCV 63-185
10"
SEE NOTE 17 & 18
FCV 74-1
CLASS B
3/4"
FCV 74-2
SEE NOTE 17 & 18
3/4"
CLASS G
74-543
74-504
X-107
FR
74-500
74-541
CLASS B
74-503
74-542
FR
CLASS G
FCV 74-8
SEE NOTE 17 & 18
10"
CLASS A
CLASS B
3"
74-505
4"
TO PRESSURIZER RELIEF TANK; 1-47W813-1 COORD H-2
SET PRESSURE: 450 PSIG
This is the relief valve which will provide protection. It's setpoint is 450 psig.
CLASS B
CLASS G
SEE NOTE 21
FCV 74-33
TO LETDOWN HT EXCH 1-47W809- COORD B-
CLASS H
3/4"
VENT
74-538
COLD LEG INJECTION 1-47W811-1 COORD G-8
8"
FCV 74-16
TO CHARGING PUMPS 1-47W811-1 COORD G-8
RHR SPRAY HDR 1-47W812-1 COORD F-4

3.2.3 RHR Valves (continued)

5. RHR HXs Bypass FCV-74-32 (W 618):

   This non-trained, air-operated, fail open, 8" butterfly valve is located in the HXs' common bypass line. A steel plate is installed as a protection barrier for the positioner linkage of FCV-74-32 but does not affect testing. This valve has controllers located in the MCR and ACR. By adjusting this valve in conjunction with FCV-74-16 and -28, total system flow can be held constant while changing flow through the HX(s). The line containing this valve is isolated from the RHRPs and HXs by HCV-74-36 and -37. These manual valves are opened prior to initiating RHR normal operations.

6. RHRP Mini-Flow FCV-74-12, -24 (W 610, 611):

   The RHRP mini-flow MOVs are safety-related, 3" globe valves located in each pump's mini-flow loop. They are controlled by a flow indicating switch and the RHRP breaker.

7. RHR Crosstie FCV-74-33, -35 (W 8716A, 8616B):

   These MOVs are safety-related, 8" gate valves located in the RHR piping crosstie downstream of the HXs. These valves must be open during normal operation of the RHRS. ECCS function is addressed in Ref. 7.2.17.

   During Modes 5 and 6, these valves may be closed. Because of the low flow required during Modes 5 and 6, the pump-to-pump interaction could result in back pressure from one pump being strong enough to limit the weaker pump's injection flow or forward flow. Only a small break LOCA is required to be postulated during Mode 4. Per system description N3-63-4001 (Ref. 7.2.17), Section 3.1.3, the RHR pumps are not required for the injection phase of a Mode 4 LOCA. ECCS operation following a Mode 4 LOCA is bounding for a Mode 5 LOCA. Therefore, isolation of the crosstie valves does not adversely impact ECCS operation for a Mode 5 LOCA. DBA LOCA events are not required to be postulated in Mode 6 per WB-DC-40-64 (Ref. 7.2.32) and automatic SI is not required in Modes 5 and 6 per the Technical Specification bases B3.3.2.

   To prevent pressure locking of these valves and/or liquid entrapment a 1/4" dia. hole is located on the upstream side of the valve disc per Ref. 7.5.63.

B. Relief Valves and Manual Valves

1. RHR Suction Header Relief RFV-74-505 (W 8708):

   This relief valve protects the RHRS from RCS over pressurization while the system is in operation (i.e., FCV-74-1 and -2 open). The valve's nozzle has a 3" inlet diameter, 4" outlet diameter and is austenitic stainless steel.

The question stem places the Unit at 150 degrees. This would be MODE 5 and thus require RHR to be in operation.

3.2.3 RHR Valves (continued)

The valve has a relief capacity of 900 gpm and a required flow rate of 480 gpm at 350 degrees F and 690 gpm at 200 degrees F. The setpoint for the valve is 450 psig (Ref. 7.5.45, 7.5.48, 7.5.49, 7.5.50). The valve discharges to the Pressurizer Relief Tank (PRT) and its discharge flow capacity is analyzed in Ref. 7.4.7. The valve cannot protect the RHRS from overpressure during modes other than startup or cooldown if FCV-74-1 and -2 are inadvertently open when RCS pressure is above RHRS design. Interlocks, alarms and administrative controls are used on FCV-74-1 and -2 to prevent the event from occurring (see Section 3.3 and 4.0).

This discusses the "plausible" 370 psig distractor.

2. Valves HCV-74-34, -36, -37 (W 8735, 8726A, 8726B):

   These valves are 8" manual gate valves.

   HCV-74-34: This normally closed RHR-to-RWST Return Line Isolation Valve is throttled open when pumping primary coolant to the RWST following refueling or when reducing or lowering RCS inventory or level. At all other times, this valve is locked closed (see Section 4.0).

   HCV-74-36, -37: The normally closed HX Bypass Isolation Valves are opened when bypass flow through FCV-74-32 is required for temperature control during startup, cooldown, and refueling.

3. RHRP Check Valves CKV-74-514, -515 (W 8730A, 8730B):

   These valves are 8" swing-type check valves located downstream of RHRPs. The valves will protect against reverse flow through an RHRP when it is not operating to ensure the operating pump meets system flow requirements.

4. RHRP and HX Isolation ISV-74-520, -521, -524, -525 (W 8728A, 8728B, 8724A, 8724B):

   These valves are 8" gate valves located between the RHRPs and the respective HXs A and B. These valves provide RHRP and RHR HX isolation for maintenance.

5. Letdown Isolation to CVCS ISV-74-530, -531 (W 8734A, 8734B):

   These valves are 2" Kerotest globe valves located in RHR cross-tie piping. They are opened when RHRS letdown to the CVCS (Ref. 7.2.35) is necessary (see Section 4.0).

6. Sample Root Valves SMV-74-522, -523, -532, -533 (W 8725A, 8725B, NA, NA):

   These valves are 3/4" Kerotest diaphragm globe valves. SMV-74-532, -533 are on mini-flow return lines A-A and B-B. SMV-74-522, -523 are in the RHR flowpath downstream of the RHRPs. These normally open valves provide for RHR sampling when required for water chemistry control prior to and during operations. Frequency and duration of sampling is controlled by the Sampling and Water Quality System (System 43).

63-637
SET PRESS.
600 PSI
TO RHR SPRAY CONT
ON 1-47W812-1, E-4
63-309A
FT
63-173
SET PRESS.
600 PSI
63-627
63-310A
3/4" DR
CONT ON 47W813-1,
H-5, AT PEN. X-24
FE
63-92
FM PEN X30 RELIEF CONT
ON 1-47W811-1, D-6
VTV-63-16
VTV-63-17
1/2"V
63-506
1" DR
FCV
63-7
63-502
63-503
3/4" TC
RESIDUAL
HEAT
REMOVAL
PUMPS
63-340A
63-339A
FT
63-92
FLOW
RESTRICTOR
OR-63-92
63-629
63-693
63-696
3/4" TV
63-628
SET
PRESS.
600 PSI
63-626
The RHR pump's
discharge relief has a
setpoint of 600 psig.
FLOW
RESTRICTOR
OR-63-91
1B-B
B
RESIDUAL
HEAT
EXCHANGERS
RHR CONT ON
1-47W810-1
8 PLACES
LETDOWN HX (RHR)
1-47W809-1, B-8
This is the 1A-A
RHR pump. The
highlighted path
would be for
ECCS injection.
1A-A
A
63-338A
63-337A
FT
63-91
63-370A
PI
63-57B
63-665
3/4" TC
MANWAY
ISV-63-73
K-14
63-729
63-730
63-728
63-727
K-15
FCV
63-73
63-596
TO RHR SPRAY CONT ON
1-47W812-1, F-4
SEE NOTE 20
63-505
3/4" TC
FCV
63-8
CONT ON 47W810-1, E-9
63-368A
PI

7.

Given the following conditions:

- Unit 1 is at 100% power.
- 1-FCV-62-84, AUX SPRAY TO PZR has stuck OPEN.
- 1-AOI-18, Malfunction of Pressurizer Pressure Control System is in progress.
- The crew is placing excess letdown in service in accordance with 1-AOI-18 Attachment 2, Excess Letdown.

Which ONE of the following describes the basis for the excess letdown flowrate limit?

The excess letdown flowrate is limited to prevent ________.

A. high letdown and CCS temperatures

B. RISING ⇑ dose rates in the auxiliary building

C. RCP #1 seal leakoff flow RISING ⇑ above the limit

D. damaging the liquid radioactive waste pumps if excess letdown is diverted

CORRECT ANSWER: A

DISTRACTOR ANALYSIS:

A. Correct: 1-AOI-18 contains the following caution in Attachment 2: Excess letdown design flow is 20 gpm but can go as high as 50 gpm which could cause high letdown and CCS temperatures.

B. Incorrect: Attachment 2 of 1-AOI-18 contains step 1.0 D: **NOT**IFY Radiological Protection Shift Supervisor Excess Letdown is in service. The reason for this step is that the Excess Letdown heat exchanger is located on elevation 737' of the Aux Building. Unlike the regenerative and non-regenerative heat exchangers (which are located within a shielded room inside of containment), the excess letdown heat exchanger exists in a location which can affect the dose to nuclear plant personnel. It is plausible to believe that an applicant would believe that given the location of the excess letdown heat exchanger, which the basis for the flow limit would be to limit personnel dose.

C. Incorrect: Attachment 2 of 1-AOI-18 contains a caution which states: As excess letdown is placed in service, the #1 seal leakoff flow could potentially fall below the limit. Therefore, it is not correct that seal flow would rise above the limit. It is plausible to believe that it would if one believed that a siphon or eductor effect were created as the flow from excess letdown merged with the seal return flow.

D. Incorrect: It is plausible to believe that the maximum flow which excess letdown flow is capable of could out run the pumping capacity of the RCDT pumps because there are radioactive liquid waste pumps which are of insufficient capacity to keep up with excess letdown's maximum flow. Two of these pumps in particular are one of the Tritiated Drain Collector Tank and one of the Floor Drain Collector pumps. Both of these pumps are 20 gpm and serve tanks much larger than the RCDT. Plausibility of this distractor is enhanced because the applicant may believe that the excess letdown flow is diverted directly to the TDCT. However, this is not the case. The TDCT does have a second pump which discharges at 100 gpm.

Question Number: 7

Tier: 1 Group: 1

K/A: 027 Pressurizer Pressure Control System (PZR PCS) Malfunction
G 2.4.18 Knowledge of the specific bases for EOPs.

Importance Rating: 3.3 4.0

10 CFR Part 55: (CFR: 41.10 / 43.1 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to understand the specific basis for the limit on excess letdown flow when placing such in service in accordance with 1-AOI-18, Malfunction of Pressurizer Pressure Control System.

Technical Reference: 1-AOI-18, Malfunction of Pressurizer Pressure Control System
WBN-SDD-N3-77C-4001, Liquid Radwaste Processing System

Proposed references to be provided: None

Learning Objective: 3-OT-STG-062A, CHEMICAL AND VOLUME CONTROL SYSTEM
6. EXPLAIN the CVCS design features and/or interlocks that provide the following:
r. Excess letdown

Cognitive Level:
- Higher ____
- Lower X

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Excess letdown is discharged to the seal return line of CVCS.

Attachment 2
(Page 1 of 2)

Excess Letdown

This caution is non-specific as to whether it is in effect for the duration of excess letdown's in service time or just at the start of placing it in service.

1.0 PLACING EXCESS LETDOWN INSERVICE

CAUTION

As excess letdown is placed in service, the #1 seal leakoff flow could potentially fall below the limit.

A. **PERFORM** the following: [0-M-27B]

- **OPEN** 1-FCV-70-143, EXC LTDN HX SUP CIV-∅A.
- **OPEN** 1-FCV-70-85, EXC LTDN HX OUT CIV-∅A.

B. **CHECK** 1-HS-62-59A, CVCS EXCESS LETDOWN DIVERT FLOW CNTL aligned to one of the following

- NORM
- DIV

NOTE

1-FCV-62-61 and 1-FCV-62-63 should **NOT** be opened if Phase A signal present.

C. **PERFORM** the following: [1-M-5]

1. **IF** 1-HS-62-59A, CVCS EXCESS LETDOWN DIVERT FLOW CNTL in NORM, **THEN**
   - **ENSURE** 1-FCV-62-61, CVCS SEAL WATER RETURN HEADER ISOLATION, OPEN.
   - **ENSURE** 1-FCV-62-63, CVCS SEAL WATER RETURN HEADER ISOLATION, OPEN.
2. **OPEN** 1-FCV-62- 54, EXCESS LTDN ISOL.
3. **OPEN** 1-FCV-62-55, EXCESS LETDOWN ISOLATION.

This caution is definitive: if excess letdown flow is > than the design, then high letdown and CCS temperatures could result.

Attachment 2
(Page 2 of 2)

1.0 PLACING EXCESS LETDOWN INSERVICE (continued)

CAUTION

Excess letdown design flow is 20 gpm, but can go as high as 50 gpm which could cause high letdown and CCS temperatures.

4. **SLOWLY OPEN** 1-HIC-62-56A, EXCESS LETDN FLOW CONTROL to maintain Excess LD HX Outlet temperature below 195°FAND Excess LD HX CCS Outlet Temperature below 137°F.

5. **MONITOR** PZR level returning to normal.

D. **NOTIFY** Radiological Protection Shift Supervisor Excess Letdown is in service.

This supports the plausibility for distractor B.

8.

Given the following conditions:

- Unit 1 was at 100% power.
- An ATWS condition has occurred.
- 1-FR-S.1, Response to Nuclear Power Generation/ATWS, is in progress.
- Step #2, VERIFY turbine TRIPPED is in progress.

Subsequently:

- A PZR PORV **OPENS**.
- **ALL** Control Rods insert to the bottom of the core.
- An AUTOMATIC Safety Injection occurs.

Which ONE of the following describes how the crew will implement the EOPs?

The crew will ________.

A. IMMEDIATELY exit 1-FR-S.1 **AND** Perform 1-E-0, Reactor Trip or Safety Injection

B. REMAIN in 1-FR-S.1 until completed **OR** directed to transition to another procedure

C. EXIT 1-FR-S.1 when the **RED** path clears **AND** perform the steps of 1-E-1, Loss of Reactor or Secondary Coolant

D. SHUT the block valve for the stuck OPEN PORV **THEN** transition to 1-E-0 at 1-FR-S.1 Step 6, MONITOR for SI Signal

<u>CORRECT ANSWER:</u> <u>B</u>

<u>DISTRACTOR ANALYSIS:</u>

A. Incorrect: Plausible since 1-E-0 does have the immediate actions of a Rx trip or Safety Injection, however once an FR is entered it must be completed prior to exit or an exit is directed by a procedure step.

B. Correct: Per the rules of usage in TI-12.04, User's Guide for Abnormal and Emergency Operating Instructions, once 1-FR-S.1 has been entered, the actions of the FR are to be completed prior to exit or until directed to exit by a procedure step of FR-S.1.

C. Incorrect: Plausible if candidate thought that the FR could be exited as soon as the condition which caused them to enter the FR is cleared. This is not in accordance with the TI-12.04.

D. Incorrect: Plausible since at step 6 of 1-FR-S.1, the operators are directed to Perform the immediate action steps of 1-E-0 if SI has actuated. If candidate thought this meant for the crew to exit 1-FR-S.1 instead of performing the actions concurrently.

Question Number: 8

Tier: 1 Group: 1

K/A: 029 Anticipated Transient Without Scram (ATWS)
EG 2.4.31 Knowledge of annunciator alarms, indications, or response procedures.

Importance Rating: 4.2 4.1

10 CFR Part 55: (CFR: 41.10 / 45.3)

10CFR55.43.b: Not applicable

K/A Match: This question matches the K/A by testing the candidate's knowledge of the response procedures and their use for an ATWS condition.

Technical Reference: TI-12.04, User's Guide for Abnormal and Emergency Operating Instructions

Proposed references to be provided: None

Learning Objective: 3-OT-FRS0001, Function Restoration Guidelines 1-FR-S.1 & S.2
2. Apply the rules of usage (TI-12.04) and ANALYZE plant conditions to identify any required procedure transitions in 1-FR-S.1 and FR-S.2.

Cognitive Level:
- Higher ____
- Lower X

Question Source:
- New ____
- Modified Bank ____
- Bank X

Question History: Bank question 029EG2.4.31 7

Comments:

2.4.4 Status Tree Rules of Usage (continued)

C. Status Trees shall be monitored in the following priority:

1. 1-FR-S, Subcriticality,
2. 1-FR-C, Core Cooling,
3. 1-FR-H, Heat Sink,
4. 1-FR-P, PTS,
5. 1-FR-Z, Containment,
6. 1-FR-I, Inventory.

D. If a RED path is diagnosed, then the Function Restoration Instruction will be implemented IMMEDIATELY.

E. If an ORANGE path is diagnosed, then the remaining Status Trees will be checked. If no RED path exits, then the highest priority ORANGE path Function Restoration Instruction will be implemented.

F. Once implemented because of any RED or ORANGE path, that Function Restoration Instruction will be performed to completion or to a point of transition UNLESS a higher priority condition develops.

1. As a Function Restoration Instruction is performed, the status of that tree may change. This change does **NOT** change the priority of an instruction in progress.
2. If a higher priority condition develops, the instruction in effect should be suspended and the higher priority condition addressed.

G. When no RED or ORANGE path exists, a YELLOW path Function Restoration Instruction can be implemented at the operator’s discretion.

9.

Given the following timeline:

00:00:00	1-ES-3.3, Post-SGTR Using Steam Dump is in progress on Unit 1.
00:01:00	The crew is depressurizing the ruptured SG to 400 psig using steam dumps.
00:10:00	Step 15. d. is currently being performed to STOP depressurization of the Ruptured SG.
00:11:00	1-FCV-1-103, CONDENSER A MAIN STEAM DUMP VLV, FAILS to CLOSE from the MCR.
00:12:00	An AUO is dispatched to MANUALLY isolate the failed steam dump.

Which ONE of the following describes the **MAXIMUM** number of Steam Dump Valves that **CAN** be OPENED AND the method the AUO will use to **MANUALLY** isolate the Steam Dumps?

At 00:02:00 a **MAXIMUM** of ____(1)____ steam dump valves can be OPENED.

The dispatched AUO will be able to **MANUALLY** shut a SINGLE valve to isolate ____(2)____.

A. (1) 3
   (2) 1-FCV-1-103 ONLY

B. (1) 12
   (2) 1-FCV-1-103 ONLY

C. (1) 3
   (2) ALL steam dump valves

D. (1) 12
   (2) ALL steam dump valves

<u>CORRECT ANSWER: A</u>

DISTRACTOR ANALYSIS:

A. Correct: The conditions in the stem describe that the operating crew has cooled the RCS down to less than 375°F (which would be step 10 of 1-ES-3.3). WBNP has permissive P-12. P-12 is the LO-LO Tavg Steam Dump block. If RCS Tavg in 2 of 4 loops is less than or equal to 550°F then ALL of the steam dumps would close. This would have been the case when the crew was in 1-E-3, Steam Generator Tube Rupture. The crew would have then bypassed P-12 (still in 1-E-3) to perform their initial cooldown. The ARI for window 68-A (P-12) describes that This interlock can be bypassed for 3 of 12 steam dump valves. Therefore, when the crew was performing the cooldown to less than 375°F, they would have had at most 3 steam dump valves. Additionally, as seen on print 1-47W801-1, 1-FCV-1-103 is isolated via a downstream manual valve 1-ISV-1-627. Also seen on this print is the fact that ONLY 1-FCV-1-103 is isolated by this single manual isolation. Each steam dump is provided with a single downstream isolation valve.

B. Incorrect: It is not correct that 12 steam dump valves could be open given the conditions listed in the steam. It is plausible that the applicant could believe that 12 dump valves could be opened if he failed to comprehend the operation of P-12 or did not correctly interpret the plant conditions which would be had in 1-ES-3.3. Also, it is correct that only 1-FCV-1-103 would be isolated.

C. Incorrect: While it is true that a maximum of 3 steam dump valves could be opened, it is not true that a single valve could isolate all steam dumps. It would be plausible that a single valve would exist as there are other components which are supplied by main steam which can be isolated by a single valve. Examples of these would be the main feed pumps' HP steam supplies and the TDAFWP. Also, the main steam piping to the steam dumps follows a path which provides a convenient location for a single isolation valve.

D. Incorrect: As mentioned previously, it Is not true that 12 steam dumps could be opened or could be isolated by a single valve.

Question Number: 9

Tier: 1 Group: 1

K/A: 038 Steam Generator Tube Rupture (SGTR)
EA1 Ability to operate and monitor the following as they apply to a SGTR:
EA1.43 Manual isolation of steam dump valves

Importance Rating: 3.6* 3.5*

10 CFR Part 55: (CFR 41.7 / 45.5 / 45.6)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to both monitor the steam dump valves in their initial state (prior to isolation) as they would exist during 1-ES-3.3. The applicant must subsequently demonstrate the knowledge required to perform a manual isolation of a failed steam dump valve. The same knowledge applies to the manual isolation of multiple steam dump valves.

Technical Reference: 1-47W801-1; 1-ARI-64-70, Bypass, Intlk, & Permissive
1-ES-3.3, Post-SGTR Cooldown Using Steam Dump

Proposed references to be provided: None

Learning Objective: 3-OT-SYS001B, Steam Dump Control System
4. EXPLAIN the physical connections and/or cause-effect relationships between the Steam Dump Control System and the following systems:
a. Main Steam System
6. EXPLAIN the Steam Dump Control System design features and/or interlocks that provide the following: (IER 11-3 Have a solid understanding of plant design, engineering principles, and sciences.)
c. RCS Cooldown
i. Protection against excessive cooldown

Cognitive Level:
Higher X
Lower ___

Question Source:
New X
Modified Bank ___
Bank ___

Question History: New question for the 2015-301 NRC RO Exam
Comments:

68-A

Source

1-TS-68-2J (Loop 1)
1-TS-68-25J (Loop 2)
1-TS-68-44J (Loop 3)
1-TS-68-67J (Loop 4)

Setpoint

Tavg in 2/4 loops less than or equal to 550 °F

P-12
LO-LO TAVG
STM DUMP BLOCK

(Page 1 of 1)

Probable Cause: A. Plant cooldown

NOTE

The P-12 interlock prevents opening of the steam dump valves if RCS temperature is less than 550 °F. This interlock can be bypassed for 3 of 12 steam dump valves. P-12 light also indicates steam dump valves should be closed.

Corrective Action:

[1] **VERIFY** Tavg less than or equal to 550 °F.

[2] **IF** bypass of LO-LO TAVG interlock is desired, **THEN** **REFER TO** 1-SOI-1.02, STEAM DUMP *SYSTEM.*

References:

W 7246D11 Sht. 40
1-47W611-63-1
1-SOI-1.02

Step	Action/Expected Response	Response Not Obtained
14.	**MONITOR** RCP operating parameters: • Number 1 seal differential pressure greater than 200 psid. • Number 1 seal leakoff flow greater than 0.2 gpm.	**WHEN** normal RCP operation parameters can **NOT** be maintained, **THEN** **STOP** RCP.
15.	**DETERMINE** if RHR should be placed in service:	
	a. **CHECK** the following RHR operation limits: • RCS temperature less than 375°F [360°F ADV]. • RCS pressure less than 400 psig [250 psig ADV], on RVLIS - ICCM PLASMA DISPLAY.	a. **IF** RCS temperature **OR** RCS pressure is greater than or equal to RHR limits, **THEN** ** **GO TO** Step 11.
	b. **PLACE** RHR in shutdown cooling: • **REFER TO** 1-SOI-74.01, Residual Heat Removal System.	
	c. **USE** RHR for RCS cooldown.	This is the step that the crew is at.
	d. **STOP** depressurization of Ruptured S/G by CLOSING the following: • Ruptured S/G MSIV bypass valve • Ruptured S/G PORV • Condenser Steam Dump Valves	

This is the tie in for the steam dumps to the main steam header. Notice the lack of an isolation valve.
These are the steam dumps.
These are the steam dump isolation valves.
SEE NOTE 16
BYPASS & DRAIN
1" WARMING LIN
BYPASS & DRAIN
MAIN STEAM STOP AND CONTROL VALVE LH ASSEMBLY
MAIN STEAM DUMP DRAIN HEADER
CONDENSER ZONE A
CONDENSER ZONE B
COND
INTERCEPT VALVE
INTERCEPT VALVE
FCV 1-67
FCV 1-68
FCV 1-62
FCV 1-61
TV-3
TV-1
GV-3
GV-1
FCV 1-103
FCV 1-104
FCV 1-106
FCV 1-107
FCV 1-108
FCV 1-109
FCV 1-110
FCV 1-112
FCV 1-113
FCV 1-114
FCV 1-124
FCV 1-129
LS 1-207
LS 1-208
LS 1-209
P TEST
1-584
1-585
1-586
1-688
1-689
1-556
1-558
1-60
TW 1-66
PT 1-33D
1-111
1-627
1-628
1-629
1-630
1-631
1-632
1-633
1-634
1-635
1-636
1-637
1-638
1-244A
NOTE 13
37.5" O.D.
36"
28"
12"
20"
10"
8"
6"

10.

Given the following conditions:

- Unit 1 was at 100% power.
- Unit 1 has a faulted SG.

Which ONE of the following describes a reason for ENSURING the MSIVs and MSIV Bypass valves are CLOSED AND if OPERATOR action IS REQUIRED to perform this step?

In accordance with the Westinghouse background document for 1-E-2, Faulted Steam Generator Isolation, a reason for ensuring that the MSIVs are CLOSED is to ____(1)____.

Operators ____(2)____ REQUIRED to manipulate equipment to ensure that the MSIV bypasses are CLOSED.

A. (1) prevent a sustained return to power

(2) ARE

B. (1) prevent a sustained return to power

(2) ARE **NOT**

C. (1) isolate the break and isolate the SGs from each other

(2) ARE

D. (1) isolate the break and isolate the SGs from each other

(2) ARE **NOT**

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: In accordance with the Westinghouse background document for 1-E-2, all MSIVs and bypass valves are checked to be closed in this step [step 1] in an attempt to isolate the break and to isolate the SGs from each other. Therefore, it is correct that the basis for the step listed is to isolate the break. Therefore, it would be plausible to believe that such basis would be included in the WOG document. However, not only is the prevention of the re-criticality not included as a basis for the step, the re-criticality of the core is accounted for by Chapter 15 of the Watts Bar FSAR. Westinghouse studies have shown that for the design basis steam line rupture, which a return to power is predicted (and will be turned by ECCS boron injection). Additionally, Westinghouse has accounted for the fact that the MSIVs will fail to close as the design basis steam line break accounts for both a Uniform cooldown (indicating that ALL SGs are feeding the break – as if the MSIVs were all OPEN) and a non-uniform (MSIVs shut) cooldown. Because of this, Westinghouse: (1) determined that for the design basis steam line break a return to power was predicted whether or not the MSIVs were closed and (2) that ECCS boron injection would provide the required negative reactivity to shut the core down., Also, it is Incorrect to believe that action is required to close the MSIV bypasses.

B. Incorrect: Again it is Incorrect to believe that the basis for closing the MSIVs is to prevent a return to criticality. However it is true that action is not required to ensure that the MSIV bypasses are closed.

C. Incorrect: Again, the Westinghouse background document only recognizes two bases for ensuring that the Main Steam Isolations were closed. Sound reactor principles would indicate to one that shutting such isolations would stop the plant cooldown and thus stop the positive reactivity addition to the core. However, it is not true that the control room staff is required to manipulate anything to ensure that the MSIV bypass valves are closed. As seen in pages 15-18 of 1-SOI-1.01, the MSIV bypass valves are powered and opened when steam line warming is required during plant startup. After the MSIVs are opened, the MSIV bypasses are closed and depowered. Therefore, no manipulations are required by the control room staff to ensure that the MSIV bypasses are closed. It is plausible to believe that control room action would be required because even though these bypass valves are de-energized in accordance with the aforementioned SOI, as seen on print 1-45W600-2, they are designed with the same isolation circuitry as the MSIVs (i.e. that they close on a main steam isolation signal). In effect, one could believe that these valves were normally open because of the associated safeguards which were implicit to the valve design. Because the #2 MSIV failed to automatically close, it is reasonable to believe that the #2 MSIV bypass valve would have failed to close (if it were indeed open).

D. Correct: As mentioned the basis for the isolation step is to isolate the break. Also, no manipulation is required to ensure that the MSIV bypasses were closed.

Question Number: 10

Tier: 1 Group: 1

K/A: 040 Steam Line Rupture
AK3. Knowledge of the reasons for the following responses as they apply to
the Steam Line Rupture:
AK3.01 Operation of steam line isolation valves

Importance Rating: 4.2 4.5

10 CFR Part 55: (CFR 41.5,41.10 / 45.6 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to .understand the reasons for ensuring that the MSIVs and MSIV bypasses are closed during step 1 of 1-E-2, Faulted Steam Generator Isolation.

Technical Reference: Figure 15.4-12a of chapter 15 of the WBNP FSAR
1-45W600-2
1-SOI-1.01, Main Steam System
Westinghouse background document for 1-E-2

Proposed references to be provided: None

Learning Objective: 3-OT-STG-001A, MAIN STEAM
01. DESCRIBE the design criteria, purpose and/or functions of the Main
Steam System, subsystems, and the major system components listed
below:
d. MSIVs & bypass valves

Cognitive Level:
Higher ____
Lower X

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

STEP DESCRIPTION TABLE FOR E-2 Step 1

STEP: Check Main Steamline Isolation And Bypass Valves - CLOSED

PURPOSE: To ensure that the steamline isolation and bypass valves have closed

BASIS:

Since the guideline is entered after symptoms of a faulted SG have been identified, the main steamline isolation signal should have been previously actuated and the MSIVs and bypass valves should have previously received a "CLOSE" signal. On the ERG reference plant, the MSIVs and bypass valves in all main steamlines receive the same isolation signal and, therefore, all SG main steamlines should be isolated. Consequently, all MSIVs and bypass valves are checked to be closed in this step in an attempt to isolate the break and to isolate the SGs from each other. If any valves have not received a "CLOSE" signal or if the valves failed to close, the operator is instructed to manually close the valves.

Some plants may have individual loop low steamline pressure isolation capability. On these plants, the step should be modified to indicate only the MSIVs and bypass valves on those loops for which a main stea isolation signal is present should be checked for closure and manuall if necessary.

Basis for closing the MSIVs:
1. Isolate the break
2. Isolate the SGs from each other

ACTIONS:

- o Determine if main steamline isolation and bypass valves are closed
- o Close appropriate valves

INSTRUMENTATION:

Main steamline isolation and bypass valve indication

CONTROL/EQUIPMENT:

Main steamline isolation and bypass valve controls

Date________ **Initials_____**

5.1 MSIVs and Warming Valves Closed (continued)

CAUTION

During initial warming with RCS T_{avg} >300°F, opening one MSIV bypass at a time with a 2-hr wait between each opening is recommended to allow main steam header to drain and to minimize water hammer

Main Steam piping Heatup Rate should **NOT** exceed 100°F/hr and must **NOT** exceed 200°F/hr.

NOTE

Warm-up time of up to 16 hrs may be needed to warm steam lines properly, dependent on initial temp and conditions.

[12] **(ρ) WHEN** desired to warm the Main Steam Header, **THEN**

PERFORM the following:

NOMENCLATURE	LOCATION	POSITION	UNID	INITIAL
WARMING THROTTLE VALVE S/G LOOP 4	SVR/729	(1)	1-THV-1-8	
WARMING THROTTLE VALVE S/G LOOP 1	SVR/729	(1)	1-THV-1-18	
WARMING THROTTLE VALVE S/G LOOP 2	NVR/729	(1)	1-THV-1-28	
WARMING THROTTLE VALVE S/G LOOP 3	NVR/729	(1)	1-THV-1-48	
SG 1 MSIV BYPASS WARMING LINE	1-M-4	OPEN	1-FCV-1-147	
SG 2 MSIV BYPASS WARMING LINE	1-M-4	OPEN	1-FCV-1-148	
SG 3 MSIV BYPASS WARMING LINE	1-M-4	OPEN	1-FCV-1-149	
SG 4 MSIV BYPASS WARMING LINE	1-M-4	OPEN	1-FCV-1-150	

(1) Valves may be open or throttled, as necessary, to maintain steam line warming rate to less than 100°F/Hr

Date________ **Initials_____**

5.1 MSIVs and Warming Valves Closed (continued)

NOTE

REFER TO Appendix C for MSIVs opening guidelines.

[13] **(ρ) WHEN** steam lines are fully warmed, and Criterion A, B, or C of Appendix C is met**, THEN**

OPEN MSIVs (listed below) ONE at a time**, AND**

ALLOW conditions to stabilize between each opening:

NOMENCLATURE	LOCATION	POSITION	UNID	INITIAL
MSIV SG 1	1-M-4	OPEN	1-FCV-1-4	
MSIV SG 2	1-M-4	OPEN	1-FCV-1-11	
MSIV SG 3	1-M-4	OPEN	1-FCV-1-22	
MSIV SG 4	1-M-4	OPEN	1-FCV-1-29	

[14] **CLOSE** MSIV Bypasses (listed below), **AND**

ENSURE handswitches are placed in CLOSE**:**

NOMENCLATURE	LOCATION	POSITION	UNID	INITIALS	
SG 1 MSIV BYPASS WARMING LINE	1-M-4	CLOSE	1-FCV-1-147		IV
SG 2 MSIV BYPASS WARMING LINE	1-M-4	CLOSE	1-FCV-1-148		IV
SG 3 MSIV BYPASS WARMING LINE	1-M-4	CLOSE	1-FCV-1-149		IV
SG 4 MSIV BYPASS WARMING LINE	1-M-4	CLOSE	1-FCV-1-150		IV

Date________ **Initials**_____

5.1 MSIVs and Warming Valves Closed (continued)

NOTE

LCO tracking of OR 14.10.2 may be discontinued when the handswitches in Step 5.1[15] are returned to OFF.

[15] **PERFORM** the following:

NOMENCLATURE	LOCATION	POSITION	UNID	INITIALS	
MAIN STEAM ISOL VLV LOOP 1 BYP WARMING VLV	A3U/737	OFF	1-HS-1-147B		IV
MAIN STEAM ISOL VLV LOOP 2 BYP WARMING VLV	A4U/757	OFF	1-HS-1-148B		IV
MAIN STEAM ISOL VLV LOOP 3 BYP WARMING VLV	A5U/757	OFF	1-HS-1-149B		IV
MAIN STEAM ISOL VLV LOOP 4 BYP WARMING VLV	A3U/737	OFF	1-HS-1-150B		IV

So, the end result is that the MSIV bypasses are left power disconnected closed.

[16] **IF** Steam Dumps are available to maintain RCS temperature, **THEN**

REFER TO 1-SOI-1.02, Steam Dump System. _________

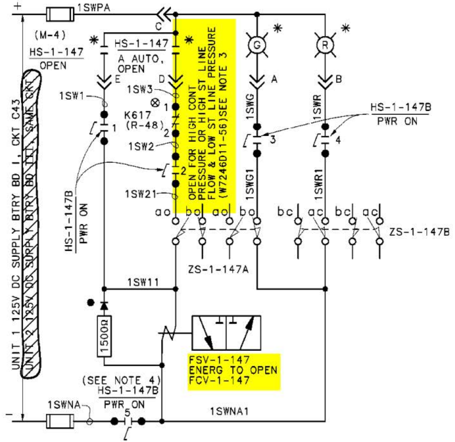
NOTES

1) Temporary gauges will be removed prior to Steam Header pressure exceeding 300 psig.

2) Step 5.1[17] may be **N/A**d if Step 5.1[1] was **N/A**d.

[17] **NOTIFY** Instrument Maintenance (MIG) to remove temporary instruments installed in Step 5.1[1]

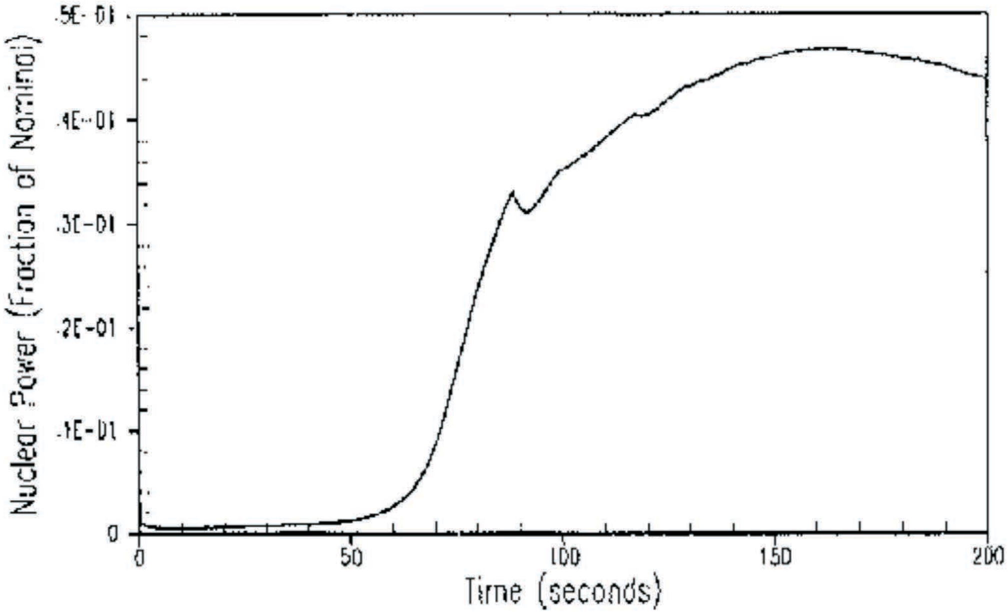
NOMENCLATURE	LOCATION	TEMP GAUGE	UNID	PERFORMED BY
MAIN STEAM LOOP 1 PRESSURE	A3U/713, PNL-1-L-102	REMOVED	1-PT-1-5	Inst Maint.
MAIN STEAM LOOP 2 PRESSURE	SBVRMR/729 PNL 1-L-106	REMOVED	1-PT-1-12	Inst Maint.
MAIN STEAM LOOP 3 PRESSURE	SBVRMR/729 PNL 1-L-106	REMOVED	1-PT-1-23	Inst Maint.
MAIN STEAM LOOP 4 PRESSURE	A3U/713 PNL-1-L-102	REMOVED	1-PT-1-30	Inst Maint.
MAIN STEAM HEADER PRESSURE	T5J/708 1-L-482	REMOVED	1-PT-1-33	Inst Maint.

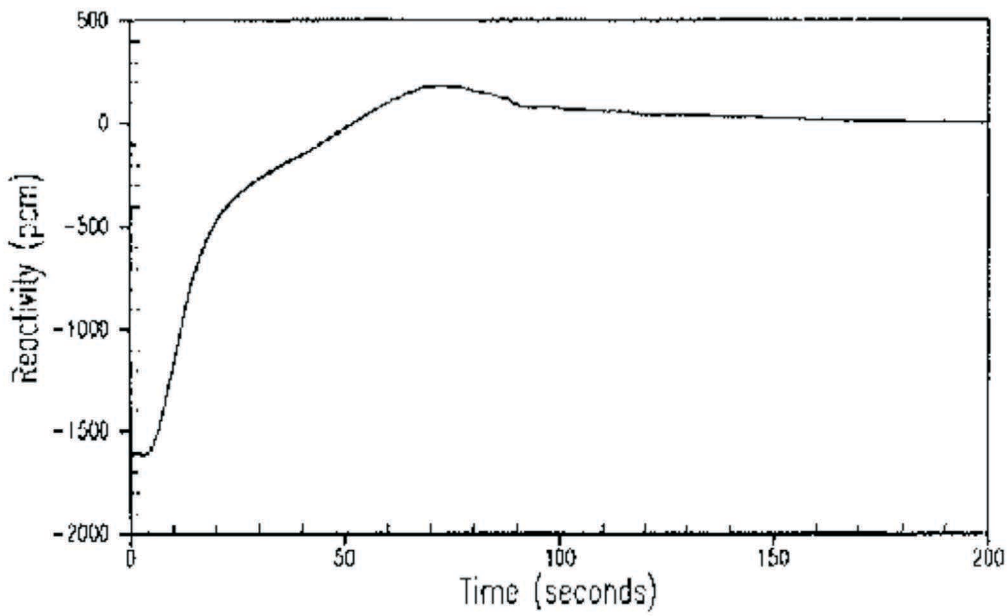


FCV-1-147

STEAM LINE WARMING VALVE (TRAIN A)

TVA NO.	DESCRIPTION	TRAIN	WIRE	FUSE SUPPLY UNIT 1	FUSE SUPPLY UNIT 2	"SSPS" CONTACTS RELAY	"SSPS" CONTACTS CONTACTS	"SSPS" CONTACTS RACK
FCV-1-147	ST L WARNING VALVE LOOP 1	A	1SW	I-C43	III-C43	K617	1,2	R-48
FCV-1-148	ST L WARNING VALVE LOOP 2	B	2SW	II-C48	IV-C48	K617	1,2	R-51
FCV-1-149	ST L WARNING VALVE LOOP 3	A	3SW	I-C44	III-C44	K624	1,2	R-48
FCV-1-150	ST L WARNING VALVE LOOP 4	B	4SW	II-C49	IV-C49	K624	1,2	R-51



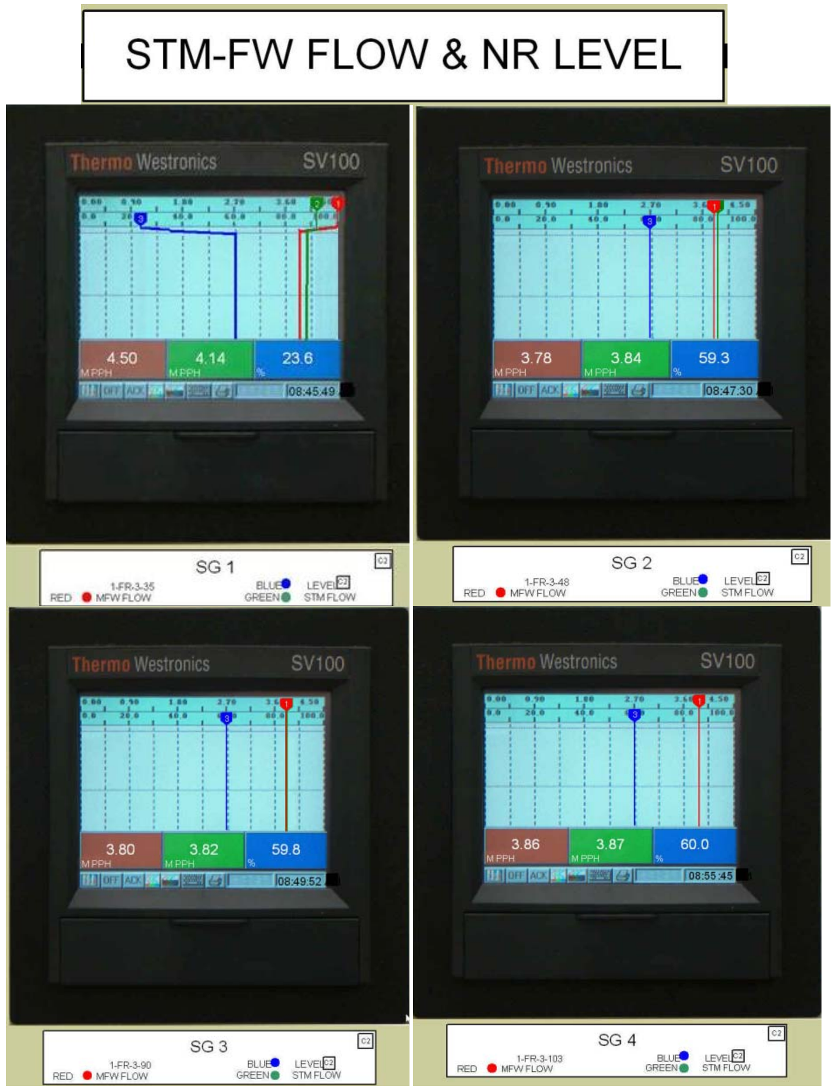


AMENDMENT 6

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
TRANSIENT RESPONSE TO STEAM LINE BREAK WITH SAFETY INJECTION & WITHOUT OFFSITE POWER (CASE B) FIGURE 15.4-12a

11.

Given the following conditions:



Which ONE of the following describes the event which caused the indications shown above?

A. Loss of control air to the MFW Reg Valve

B. A Feed line break in the steam vault room

C. A failure of the Main feed pump speed control

D. A DCS failure which caused the MFW Reg Valve to FAIL OPEN

< SEE THE NEXT PAGE FOR A MAGNIFIED VIEW OF THE RECORDERS >

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: The main feedwater regulating valves fail CLOSED on a loss of instrument air. Therefore, feedwater flow (the red recorder trace) would sharply drop to 0.00 MPPH. Steam flow (the green trace) would remain relatively constant (at 100% power it rises to just at 4.09 MPPH). Level (the blue trace) would drop quickly.

B. Correct: The indications show on the S/G #1 trend recorder resulted from a feed line break which occurred in the main steam vault room (outside of containment). The fault caused feedwater to be robbed from the S/G thus causing S/G level to drop. Feedwater flow increased (as seen) because the MFRV opened in response to a drop in S/G level and thus fed the feed break more. Steam flow rises slightly as the S/G saturation pressure rises slightly due to the reduced heat removal from RCS loop #1 (on account of the reduced feedwater injection).

C. Incorrect: A failure of the MFP speed control would affect all S/Gs equally. One may mistakenly believe that the MFRVs would act quickly enough to compensate for the failure of the MFP speed control; however, if that were the case then then no large effect would be noted. In actuality, the MFP speed change will drive a change in all S/Gs' level which will in turn cause the individual MFRVs to automatically attempt to restore their respective S/G's level.

D. Incorrect: A DCS level control failure which caused the MFRV to fail open would result in S/G level rising. Main Feedwater flow would rise as well.

Question Number: 11

Tier: 1 Group: 1

K/A: 054 Loss of Main Feedwater (MFW)
AA2. Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW):
AA2.08 Steam flow-feed trend recorder

Importance Rating: 2.9 3.3

10 CFR Part 55: (CFR: 43.5 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to interpret the Steam-flow-feed trend record in order to diagnose a specific form of a loss of main feedwater (in the question given a main feedline break to S/G #1 is presented).

Technical Reference:

Proposed references to be provided: None

Learning Objective: 3-OT-AOI3800, Main Steam of Feedwater Line Leak
2. RECOGNIZE Entry Conditions for 1-AOI-38, Main Steam or Feedwater Line Leak. (IER 11-3; Monitoring plant indications and conditions closely)

Cognitive Level:
- Higher X
- Lower ____

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

12.

Which ONE of the following describes a potential implication if SG Pressure is reduced to 50 psig during depressurization of SGs in accordance with 1-ECA-0.0, Loss of All AC Power?

A. A RED Path on PTS may occur

B. Natural Circulation may be impeded

C. Unacceptable Upper Head Voiding may occur

D. The integrity of the RCP seals may be challenged

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: Plausible since the rate of cooldown may be a concern; but final pressure/temperature is being evaluated during the cooldown. Pressurized thermal shock is a concern when RCS remains pressurized, however the RCS is intentionally being de-pressurized.

B. Correct: Nitrogen gas being discharged from the CLAs could accumulate in the RCS hot legs and SG U-tubes and potentially inhibit natural circulation.

C. Incorrect: Plausible since upper head voiding will occur after pressurizer level is lost. However this is an acceptable consequence to minimize RCS inventory loss through RCP seal degradation.

D. Incorrect: Plausible since the cooldown and depressurization of the RCS to cool RCP seals and limit RCS inventory loss, however the RCP seal cooling will only be a concern if the RCS depressurization is stopped or RCS is allowed to heat up, not if the RCS is cooled down further than it should be at this point.

Question Number: 12

Tier: 1 Group: 1

K/A: 055 Station Blackout
EK1.02 Knowledge of the operational implications of the following concepts as they apply to the Station Blackout:
Natural Circulation Cooling

Importance Rating: 4.1 / 4.4

10 CFR Part 55: 41.8, 41.10

10CFR55.43.b: Not applicable

K/A Match: This question matched the K/A by testing the candidates knowledge of the effect of LOWERING RCS pressure below the minimum value during Natural Circulation cool down.

Technical Reference: Westinghouse background document for 1-ECA-0.0, Loss of All AC Power

Proposed references to be provided:

Learning Objective: 3-OT-ECA0000, Emergency Contingency Actions, ECA-0.0, 0.1 & 0.2
3. Explain why intact SGs are depressurized to 300 psig during performance of ECA-0.0.
6. Explain why the operator is directed to maintain RCS press greater than 250 psig during RCS cooldown.

Cognitive Level:

Higher	____
Lower	X

Question Source:

New	____
Modified Bank	____
Bank	X

Question History: Bank question imported from San Onofre Nuclear Generating Station

Comments:

CAUTION: SG pressures should not be decreased to less than (0.07) psig to prevent injection of accumulator nitrogen into the RCS.

PURPOSE: To alert the operator that steam generator pressures must be maintained above the specified limit

BASIS:

Steam generators should be depressurized to maximize delivery (into the RCS) of the water contained in the SI accumulators while minimizing delivery of nitrogen. Maintaining steam generator pressures above a value that prevents introduction of a significant volume of nitrogen into the RCS ensures that accumulator nitrogen will not impede natural circulation.

A steam generator pressure limit is set to preclude significant nitrogen injection into the RCS. To determine the steam generator pressure limit, an ideal gas expansion calculation should be performed based on nominal plant specific values for initial accumulator tank pressure (P_1), initial nitrogen gas volume (V_1), and final nitrogen gas volume (V_2). The final nitrogen gas volume should be equivalent to the total accumulator tank volume. The RCS pressure at empty tank conditions (P_2) is determined from:

$$P_1 V_1^{\gamma} = P_2 V_2^{\gamma}$$

where γ = 1.25 for ideal gas expansion. The steam generator pressure limit is then determined by subtracting the RCS to SG delta P from P_2. The RCS to SG delta P should be calculated as described in the RCP TRIP/RESTART section in the Generic Issues of the Executive Volume. Instrument uncertainties are not included in the determination of the steam generator pressure limit to preclude a bias toward either having more accumulator water injected into the RCS or having less nitrogen injected into the RCS.

ACTIONS:

Determine if SG pressures are greater than (0.07) psig

INSTRUMENTATION:

SG pressure indication for each SG

STEP: Depressurize Intact SGs To (0.08) PSIG

PURPOSE: To depressurize the intact steam generators

BASIS:

Step 16 depressurizes the intact SGs, thereby reducing RCS temperature and pressure to reduce RCP seal leakage and minimize RCS inventory loss. The advantages to performing this action, as well as restrictions that apply during the action, are detailed in Subsection 2.3.

During SG depressurization, SG level must be maintained above the top of the SG U-tubes in at least one SG. Maintaining the U-tubes covered in at least one SG will ensure that sufficient heat transfer capability exists to remove heat from the RCS via either natural circulation or reflux boiling after the RCS saturates. Step 16a requires that SG level be in the narrow range in at least one SG before SG depressurization is initiated in Step 16b. If level is not in the narrow range in at least one SG, RNO 16a instructs the operator to maintain maximum AFW flow until narrow range level is established in one SG. When narrow range level is established, SG depressurization can be started or continued via Step 16b.

Step 16b instructs the operator to reduce SG pressures by depressurizing the intact SGs. Depressurization should be accomplished by opening the PORVs on the intact SGs to establish a maximum steam dump rate, consistent with plant specific constraints. The step is structured assuming that the operator can open and control SG PORVs from the control room. This structure assumes that the PORVs are air-operated and have dc control power and pneumatic power (i.e., either air reservoirs or nitrogen bottles) available. Some plants may not have the capability to open the SG PORVs from the control room. These plants should evaluate their capability to accomplish this step locally via PORV handwheels. Such an evaluation should consider accessibility and communications necessary to accomplish local PORV operation.

Once depressurization is initiated, maintenance of a specified rate is not critical. The depressurization rate should be sufficiently fast to expeditiously reduce SG pressures, but not so fast that SG pressures cannot be controlled. It is important that the depressurization not reduce SG pressures in an uncontrolled manner that undershoots the pressure limit, thus permitting potential introduction of nitrogen from the accumulators into the RCS.

During SG depressurization, AFW flow may have to be increased to maintain the required SG narrow range level. Control of AFW flow will have to be performed from the control room or locally depending on plant specific design. Full AFW flow should be established to any SG in which level drops out of the narrow range.

RCS cold leg temperatures should be monitored during SG depressurization to ensure that the depressurization does not impose a challenge to the Integrity Critical Safety Function. This check is included in Step 16c since guideline ECA-0.0 has priority over the Function Restoration Guidelines and the operator instructed to not implement a Function Restoration Guideline even if a tical Safety Function challenge is detected by the Critical Safety Function tus Trees. Consequently, Step 16c implicitly protects the Integrity tical Safety Function. The SG depressurization should not result in a llenge to the Integrity Critical Safety Function since the resultant RCS d leg temperatures should not approach the temperature limit (i.e., T2 perature) at which a challenge will exist.

The target SG pressure to preclude N2 addition ensures that natural circulation is not impeded.

Once the target SG pressure is reached, the SG PORVs and AFW flow should be controlled to maintain SG pressure at the target value until ac power is restored.

The target SG pressure for Step 16 should ensure that RCS pressure is above the minimum pressure to preclude injection of accumulator nitrogen into the RCS. The target SG pressure should be based on the nominal SG pressure to preclude nitrogen addition, plus margin for controllability (e.g., 100 psi). To determine the steam generator pressure limit, an ideal gas expansion calculation should be performed based on nominal plant specific values for initial accumulator tanks pressure (P_1), initial nitrogen gas volume (V_1), and final nitrogen gas volume (V_2). The final nitrogen gas volume should be equivalent to the total accumulator tank volume.

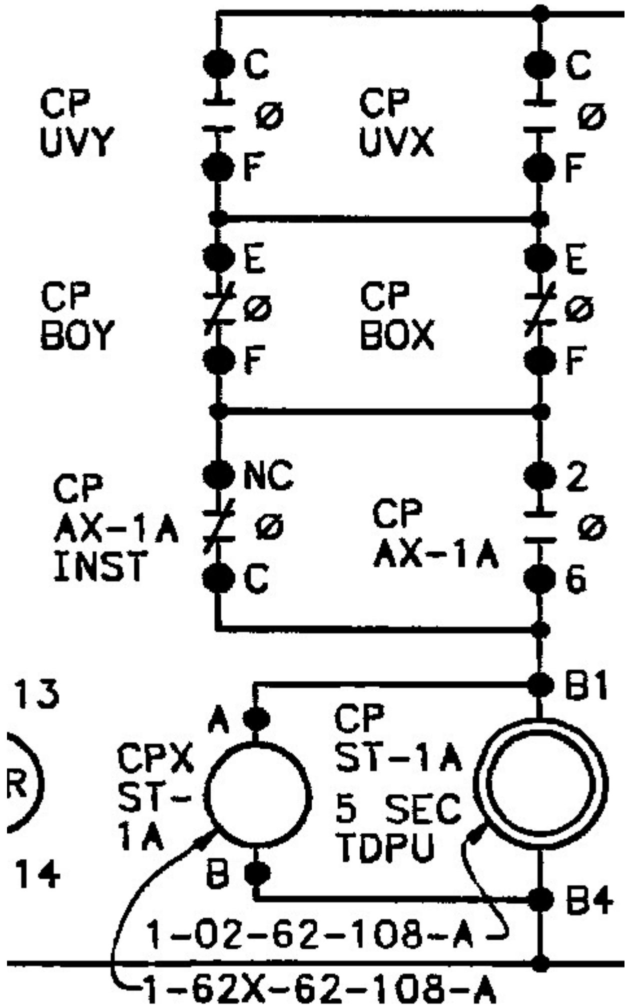
The RCS pressure at empty tank conditions (P_2) is determined from:

$$P_1 V_1^{\gamma} = P_2 V_2^{\gamma}$$

where γ = 1.25 for ideal gas expansion. The steam generator pressure limit is then determined by subtracting the RCS to SG delta p from P_2 and adding the margin to controllability. The RCS to SG delta p should be calculated as described in the RCP TRIP/RESTART section in the Generic Issues of the Executive Volume. Instrument uncertainties are not included in the determination of the steam generator pressure limit to preclude a bias toward either having more accumulator water injected into the RCS or having less nitrogen injected into the RCS.

13.

Given the following portion of the wiring diagram for the 1A-A CCP:



Which ONE of the following completes the statements below?

The BLACKOUT (BOX/BOY) contacts which are depicted in the print excerpt shown above are drawn in the _____(1)_____ state.

The wiring diagram from which the portion shown above was taken is a _____(2)_____ series print.

	(1)	(2)
A.	BLACKOUT	45W760
B.	BLACKOUT	47W611
C.	normal board energized	45W760
D.	normal board energized	47W611

<u>CORRECT ANSWER:</u>   <u>A</u>

<u>DISTRACTOR ANALYSIS:</u>

A. Correct: The wiring diagram excerpt which is presented in the stem of the question was taken from 1-45W760-62-1. Such print contains note 2 which states: “Blackout (BO) relays are shown in the reset state which is the blackout state.” Therefore, during a loss of offsite power (which causes a shutdown board blackout), the charging pump (CP) blackout X (BOX) and blackout Y (BOY) contacts which are depicted will be open. It is plausible to believe that they would be shut as it is a common misconception to believe that the BOX and BOY contacts which are shown on the prints change state during a blackout event. As discussed it is correct that a 45W760 series print is the source for the presented excerpt.

The knowledge of the print series is needed because an index is not readily available for the lookup of the control, logic, low voltage and high voltage prints. While, for example, one could refer to an index to find the single line drawing for the 480VAC Shutdown board 1A1-A, one would have to use the knowledge of the naming convention to find the high voltage print for the 1A-A Charging pump. The naming convention for the logic print is: Unit # - 47W611 – System Number – Sheet. Therefore, the logic scheme for the 1A-A CCP would be depicted on a sheet of the 1-47W611-62 prints. The print excerpt could be found by finding the high voltage print germane to the 1A-A CCP with the naming convention: Unit # - 45W760 – System Number – sheet. Without the possession of the knowledge of the naming convention, one would leaf though thousands of prints before arriving at the correct one.

B. Incorrect: While it is correct that the BOX and BOY contacts would be open during a shutdown board blackout, it is Incorrect that the print excerpt was taken from a 47W611 series print. Such print depicts the electrical logic diagram and does not show contact points. It is a series of electrical print and as such it is plausible to believe that it would be the source for the excerpt.

C. Incorrect: As mentioned it is Incorrect that the BOX and BOY contacts presented would be closed after a blackout of the shutdown board. It is plausible to believe that they would be shut as it is a common misconception to believe that the BOX and BOY contacts which are shown on the prints change state during a blackout event. As discussed it is correct that a

45W760 series print is the source for the presented excerpt.

D. Incorrect: Again it is Incorrect and yet plausible that that the BOX and BOY contacts would be shut. Also, it is Incorrect and yet plausible that the print excerpt was taken from a 47W611 series print.

Question Number: 13

Tier: 1 Group: 1

K/A: 056 Loss of Offsite Power
2.2.41 Ability to obtain and interpret station electrical and mechanical drawings.

Importance Rating: 3.5 3.9

10 CFR Part 55: (CFR: 41.10 / 45.12 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant must be able to recognize the correct series of print which display the contacts for the Blackout relays and thus possess the knowledge required to obtain the station electrical drawings germane to the loss of offsite power. Also, the applicant must be able to correctly identify the status of the blackout relays' contacts (i.e. be able to interpret the drawing).

Technical Reference: 1-45W760-62-1
1-47W611-62-4

Proposed references to be provided: None

Learning Objective: 3-OT-SYS-201B, BLACKOUT AND LOADSHED LOGIC RELAYS
1. Given specific plant conditions, ANALYZE the effect that a loss or malfunction of the following will have on the Blackout and Load Shed Logic Relays.
2) UVX, UVY relays
3) BOX, BOY relays

Cognitive Level:
Higher X
Lower ____

Question Source:
New X
Modified Bank ____
Bank ____

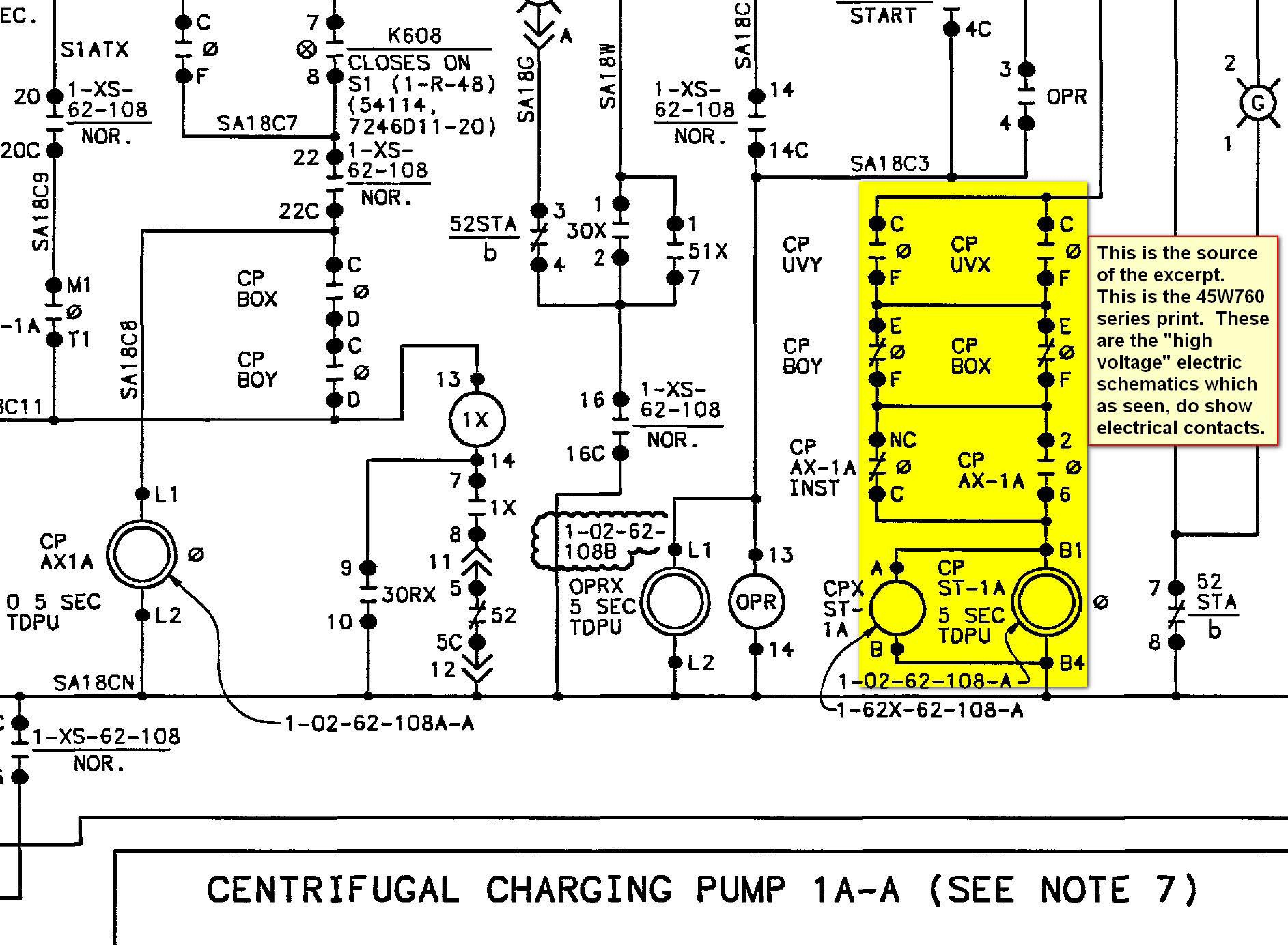
Question History: New question for the 2015-301 NRC RO Exam

Comments:

NOTES:

1. ALL EQUIPMENT IS LOCATED ON THE BOARD FROM WHICH ITS ASSOCIATED LOAD IS POWERED UNLESS OTHERWISE DESIGNATED.
2. BLACKOUT (BO) RELAYS ARE SHOWN IN THE RESET STATE, WHICH IS THE BLACKOUT STATE. UNDERVOLTAGE (UV) RELAYS ARE SHOWN IN THE DE-ENERGIZED STATE, WHICH IS THE UNDERVOLTAGE STATE. FOR BLACKOUT AND UNDERVOLTAGE RELAY OPERATION SEE 45W760-211-16.
3. 52HL CONTACTS ARE SHOWN WITH THE BREAKER IN THE FULLY RAISED AND OPERATING POSITION
4. FUSE NUMBERS SHOWN IN TABLE
APPLICABLE BOARD PREFIX LIS ... ETE
UNIQUE FUSE IDENTIFICATION ... FU-211
EXAMPLE: 1-FU-211-A18/1N

These Notes are taken from the 45W760-62-1 print. The highlighted note is a fundamental knowledge item required to interpret this series of print.



This is the source of the excerpt. This is the 45W760 series print. These are the "high voltage" electric schematics which as seen, do show electrical contacts.

14.

Which ONE of the following describes the procedural guidance contained in the AOIs AND the reason for the action?

For a loss of 120V AC Vital Instrument Power Board _____(1)_____ **ALWAYS** requires that the **Unit 1** reactor be tripped.

This is based on the condition that _____(2)_____ **CANNOT** be controlled.

NOTE: 1-AOI-25.01, Loss of 120V AC Vital Instrument Power Boards 1-I or 2-I
1-AOI-25.04, Loss of 120V AC Vital Instrument Power Boards 1-IV or 2-IV

A. (1) 1-I, 1-AOI-25.01

(2) SG level

B. (1) 1-I, 1-AOI-25.01

(2) PZR level

C. (1) 1-IV, 1-AOI-25.04

(2) SG level

D. (1) 1-IV, 1-AOI-25.04

(2) PZR level

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: DCNs 52853A and 56905 upgraded various controls to the Distributed Control System (DCS). Accordingly, 1-AOI-25.01 was revised and an immediate action to trip the reactor was remove. This may be seen in the 4/7/14 revision to 1-AOI-25.01. DCS takes power from the 120VAC preferred power board and converts that to a logic power which is auctioneered with a logic power created from a 120V AC Vital Instrument board. Therefore, controllers which would have previously failed in automatic and manual (as their plug-mold power source was lost) currently continue to function upon the loss of a vital AC board. This rationale is true of the Main Feed Regulating Valves. Before DCS was installed on the Main Feed System, the MFRV's controllers would fail their associated valves shut on a loss of 120V AC Vital Instrument Boards 1-I or 1-II. Therefore, the reactor required an immediate reactor trip to prevent an automatic reactor trip on the impending loss of S/G level. Currently, the loss of 120V AC Vital Instrument Board 1-I simply causes the 120V AC Preferred Power supply to assume the load imparted by the DCS (it does so seamlessly). Accordingly it is plausible to believe that a reactor trip is required upon the impending loss of S/G level.

B. Correct: As seen in 1-AOI-25.01, section 3.2, "Loss of 120V AC Vital Instrument Power Board 1-I," a reactor trip is tripped on account of the loss of pressurizer level control. Also, it is evident that the reactor trip is not optional as it is a left hand column step. The loss of pressurizer level control results from the loss of control air to the letdown isolation valves (which fail shut). Therefore, pressurizer level would continue to rise until a high pressurizer level automatic trip resulted.

C. Incorrect: 1-AOI-25.04 contains a response not obtained step 3 which states: **TRIP** reactor, and **** GO TO** 1-E-0, Reactor Trip or Safety Injection. Therefore, it is incorrect that a trip is always required as it is only conditionally required and again it is incorrect and yet plausible that a loss of S/G level control precipitates the Rx Trip.

D. Incorrect: While it is true that a reactor trip is required because a loss of Pzr level control exists during the performance of 1-AOI-25.01, it is not true for an entry into 1-AOI-25.04. However, as a conditional Rx Trip is directed in 1-AOI-25.04, it is plausible to believe such.

Question Number: 14

Tier: 1 Group: 1

K/A: 057 Loss of Vital AC Electrical Instrument Bus
AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus:
AK3.01 Actions contained in EOP for loss of vital ac electrical instrument bus

Importance Rating: 4.1 4.4

10 CFR Part 55: (CFR 41.5,41.10 / 45.6 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to identify the reason why the operating crew is directed by 1-AOI-25.01 trip the reactor.

Technical Reference: 1-AOI-25.01, Loss of 120V AC Vital Instrument Power Boards 1-I or 2-I
1-AOI-25.04, Loss of 120V AC Vital Instrument Power Boards 1-IV or 2-IV

Proposed references to be provided: None

Learning Objective: 3-OT-AOI2500, Loss of 125V AC Vital Instrument Power Boards
4. Given a set of plant conditions, DESCRIBE operator actions required in response to AOI-25, Loss of 120V AC Vital Instrument Power Boards.

Cognitive Level:
Higher ____
Lower X

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

3.2 Loss of 120V AC Vital Instrument Power Board 1-I

Step 2 tells one why the Unit 1 reactor is about to be tripped.

1. **DISPATCH** personnel to 120V AC Vital Instrument Power Bd 1-I to restore power using 1-SOI-235.01, 120V AC Vital Power System 1-I.

2. **NOTIFY** Load Dispatcher of impending Reactor Trip due to loss of pressurizer level control.

CAUTION

- **PZR level will continue to rise until Non-Essential air is restored to containment. This procedure takes precedence over 1-AOI-20, Malfunction of Pressurizer Level Control System.**

The instrument air to the letdown orifices has been lost. Therefore, PZR level will continue to rise uncontrollably.

OTES

- Actions to bypass containment air isolation valves, may be stopped, if voltage is restored to Vital Instrument Power Board and failed closed valves.
- Note 1 of TS 3.6.3 is applicable for bypassing containment Isolation valves.

3. **DISPATCH** AUO to restore air to CNTMT using Appendix B.

This is not optional.

4. **(ρ) TRIP** Rx.

5. ****GO TO** 1-E-0, Reactor Trip or Safety Injection WHILE continuing this instruction.

6. **ENSURE** RCS Letdown isolated:
   a. **CLOSE** letdown orifices(s).
   b. **CLOSE** 1-FCV-62-69.
   c. **CLOSE** 1-FCV-62-70.

Revision Log

Revision or Change Number	Effective Date	Affected Page Numbers	Description of Revision/Change
1	01/03/14	2, 4, 13, 15	Clarified that false indication of LO-LO VCT level would result in closure of 1-LCV-62-132, which would result in opening of 1-LCV-62-136. Added prints to support to print list. [PER 796503-001]
02	04/07/14	2-13, 15, 17-24	Revised for implementation of DCNs 52853A & 56905 which upgraded various controls to the Distributed Control System (DCS). Changes include, but are not limited to: • Rx Trip still performed but no longer as Immediate Action. • Removed transfer switches for PZR pressure and level controls. Single instrument failures no longer cause transients. • Deleted section for if on RHR letdown. Minor Editorial: Listed additional Tech Specs. Added System 63 to Appendix A for RWST level Instrumentation.
03	04/08/15	2, 17-19	Made corrections to Appendix A based on operator feedback [PER 937680-001].

When DCS was installed, the Main Feedwater Regulating Valves controllers received their control signal from DCS. They used to receive a signal from the Foxboro analogue controller which were lost (all or in part) on a loss of 120V AC Instrument Power Boards 1-I or 1-II. Therefore, an immediate trip was needed to prevent an automatic trip on the loss of S/G level.

Step	Action/Expected Response	Response Not Obtained

3.1 Loss of 1-IV 120V Vital Instrument Power Bd

Step	Action/Expected Response	Response Not Obtained
1.	**DISPATCH** personnel to 120V AC Vital Instrument Power Bd 1-IV to restore power to board using 1-SOI-235.04, 120V AC Vital Power System 1-IV.	
2.	**MONITOR** PZR pressure NORMAL.	**MAINTAIN** PZR pressure using heaters and sprays.
3.	**MONITOR** Tavg and Tref: • Tavg trending to Tref. • Mismatch less than 3°F.	**MAINTAIN** Tavg and Tref within 3°F by adjusting turbine load or by rod insertion. **IF** Tavg and Tref mismatch can NOT be maintained less than 5°F, **THEN** **TRIP** reactor, and ** **GO TO** 1-E-0, Reactor Trip or Safety Injection.
4.	**PLACE** steam dump controls to OFF: • 1-HS-1-103A, STEAM DUMP FSV “A”. • 1-HS-1-103B, STEAM DUMP FSV “B”.	
5.	**ENSURE** instrument power for AFW Pump 1A-S 1-SW-46-AC-S is in the NORMAL position (Bd 1-III) [A2T/692]	
6.	**REFER TO** Appendix A for equipment availability. [C.2]	
7.	**REFER TO** EPIP-1, Emergency Plan Classification Flowchart.	

This contingency step is taken only after the inability to control Tavg-Tref mismatch.

15.

Given the following conditions:

- Unit 1 is at 100% power.
- CCS Pump 2A-A is running.
- **ALL** 480V SDBDs have control power aligned to their **NORMAL** feeds.

Subsequently:

- 125V DC Vital Battery Board III is de-energized.

Which ONE of the following describes the response of CCS Pump 2A-A AND the handswitch indicating lights for CCS Pump 2A-A?

CCS Pump 2A-A WILL ____(1)____.

The indicating lights on 2-HS-70-59A, CCS PMP 2A-A are ____(2)____.

A. (1) TRIP
   (2) LIT

B. (1) TRIP
   (2) **NOT** LIT

C. (1) CONTINUE RUNNING
   (2) LIT

D. (1) CONTINUE RUNNING
   (2) **NOT** LIT

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: As seen on print 1-45W749-1A, the 2A-A CCS pump is powered from 480V SDBD 2A1-A. The circuit breaker for the 2A-A CCS pump receives control power from the NORMAL control bus for the 480V SDBD 2A1-A. This control bus is powered from the 125V DC Vital Battery Board III. As seen on print 1-45W760-70-1, both the Trip Coil (52TC) and the Closing Coil (52C) are powered from the normal control bus for the 480V SDBD 2A1-A. Therefore, when the bus (i.e. Battery Board III) is lost, the breaker is unable to either trip or close. Therefore, the 2A-A CCS pump will continue running after the loss of control power. It is plausible to believe that the 2A-A CCS pump will be secured following the loss of DC control power because of the blackout scheme for the Shutdown boards. During a loss of the normal 125V DC control power to the shutdown board the UVX (undervoltage relays which cause the individual loads to strip) are de-energized. This is the relay state which would cause the load to strip. Because the emergency control bus has not also been lost, the UVY relay will not also be lost. Because both the UVX and UVY relays must concurrently de-energize to cause a load to be stripped, the CCS pump will not receive a strip signal. The blackout logic applicable to the 2A-A CCS pump is seen on print 1-45W760-211-12. Also observed on print 1-45W760-70-1, all of the indicating lights for the 2A-A CCS pump (including those on 2-HS-70-59A) are powered by the normal control bus for the 480V SDBD 2A1-A. Therefore, when the bus (i.e. Battery Board III) is lost, the lights extinguish. It is plausible to believe that the lights do not extinguish as there are multiple controls in the MCR possessing indicating lights which are powered from a different source than the control power for the device.

B. Incorrect: As described the 2A-A CCS pump will continue to run. Also, it is correct that the indicating lights will be extinguished following the loss of control power.

C. Incorrect: While it is correct that the 2A-A CCS pump will continue to run, it is not correct that the indicating light will be lit for the pump controls. Again it is plausible to believe that such light would be lit as there are multiple controls in the MCR whose indicating lights are powered from a different source then the device's power supply.

D. Correct: As discussed, the pump continues to run and the indicating light is extinguished.

Question Number: 15

Tier: 1 Group: 1

K/A: 058 Loss of DC Power
AA2. Ability to determine and interpret the following as they apply to the Loss of DC Power:
AA2.03 DC loads lost; impact on ability to operate and monitor plant systems

Importance Rating: 3.5 3.9

10 CFR Part 55: (CFR: 43.5 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to understand that the breaker control power is lost following the loss of a Vital Battery Board (i.e. that a DC load is lost). The applicant must then understand the impact on his ability to operate the affected the component and to monitor that components status.

Technical Reference: 1-45W749-1A
1-45W760-70-1
1-45W760-211-12

Proposed references to be provided: None

Learning Objective: 3-OT-AOI2100, Loss of 125V DC Vital Battery Boards
3. IDENTIFY symptoms, including alarms and automatic actuations of AOI-21, Loss of 125V DC Vital Battery Boards.

Cognitive Level:
- Higher ____
- Lower X

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:


This is the normal alignment.
This indicates the source of control power to the breaker.
This is the pump discussed by the question.
2A-A(2-BD-211-A-A)
(1-45W724-3)
2A2-A(2-BD-212-
(1-45W749-2A)
XFMR 2A-A
(2-ØXF-212-A-A)
6900-480V, 3Ø
2000KVA, OA
7% Z
OIL TEMP
ANN
9 10 10
VS
V
TEST
3
C
HGD PNL 5
27S 2A1
27S 2A1X
UV ANN
125V DC NOR FEED FROM BTRY BD III (1-45W703-3)
125V DC ALT FEED FROM BTRY BD I (1-45W703-1)
2-BKR-212-A1/4B-A
2Δ-480-120V
CL
PNL 4A
3-3000-5Y
51E
W
AS
A
86E
TRIP BKR 52E L.O. BKRS 52E & 52N, OC ANN
CLOSE TRIP
100A PNL 5
4B
52E NO
ELECTRICAL
3200A FRAME, 50,000A SY
3Y-480-120V
2Δ-480-120V PNL 5
2C
2B
2D
3D
6
3B
1D
4D
EO
1600A
52
180A
L1
2-BKR-30-104-A
2-BKR-70-59-A
6P
30X
ALL 30X RELAYS ARE IN HGD PNL 5
ANN
(XS)
570A
START AUTO STOP
(C)
A2 112
1P
(PNL 6)
(NOTE 11)
62
2P
PNL 6
TDPU 5-50 SEC
2-BKR-212-A1/4D-A
LOW FLOW FAN 2B FS-30-102
TDPU 35 SEC
ST
CCW A
TDPU .5 SEC
S1
UV
BO
(NOTE 2)
LOW HDR PRESS.
100
350
AUX BLDG GENERAL SUPPLY FAN 2A, 2-MTR-30-104
FUTURE (SEE NOTE 7)
COMPONENT COOL SYSTEM PUMP 2A-A 2-MTR-70-59-A
NOR FEEDER 480V DIESEL AUX BD 2A1-A(1-45W732-1)
30
11
70
1

COMPONENT COOLING SYSTEM PUMP 1A-A (PUMP 2A-A SIMILAR EXCEPT AS NOTED IN TABLE)

NOMENCLATURE	BOARD	WIRE PREFIX	CONTROL SWITCHES	62 RELAY UNID	CCSA RELAY UNID	CCS ST RELAYS	UV RELAYS	BLACKOUT RELAYS	HDR PRESS. BUFFER RELAYS	FUSE ID NO. (NOTE 5)				STATUS MON RELAY
										NOR.TRIP	AUX TRIP	NOR.CLOSE	AUX CLOSE	
COMPONENT COOLING SYSTEM PUMP 1A-A	1A1-A	A13B	1-HS-70-46A,C; 1-XS-70-46	1-02-70-46A-A	1-02-70-46B-A	1-02-70-46-A	CCSAUVX CCSAUVY	CCSA BOX CCSA BOY	CS1A(1-R-73)	A13/11N	A13/11A	A13/12N	A13/12A	1A2112 (A/B)
COMPONENT COOLING SYSTEM PUMP 2A-A	2A1-A	A13B	2-HS-70-59A,C; 2-XS-70-59	2-02-70-59A-A	2-02-70-59B-A	2-02-70-59-A	CCSAUVX CCSAUVY	CCSA BOX CCSA BOY	CS1A(1-R-73)	A13/11N	A13/11A	A13/12N	A13/12A	2A2112 (7/6)

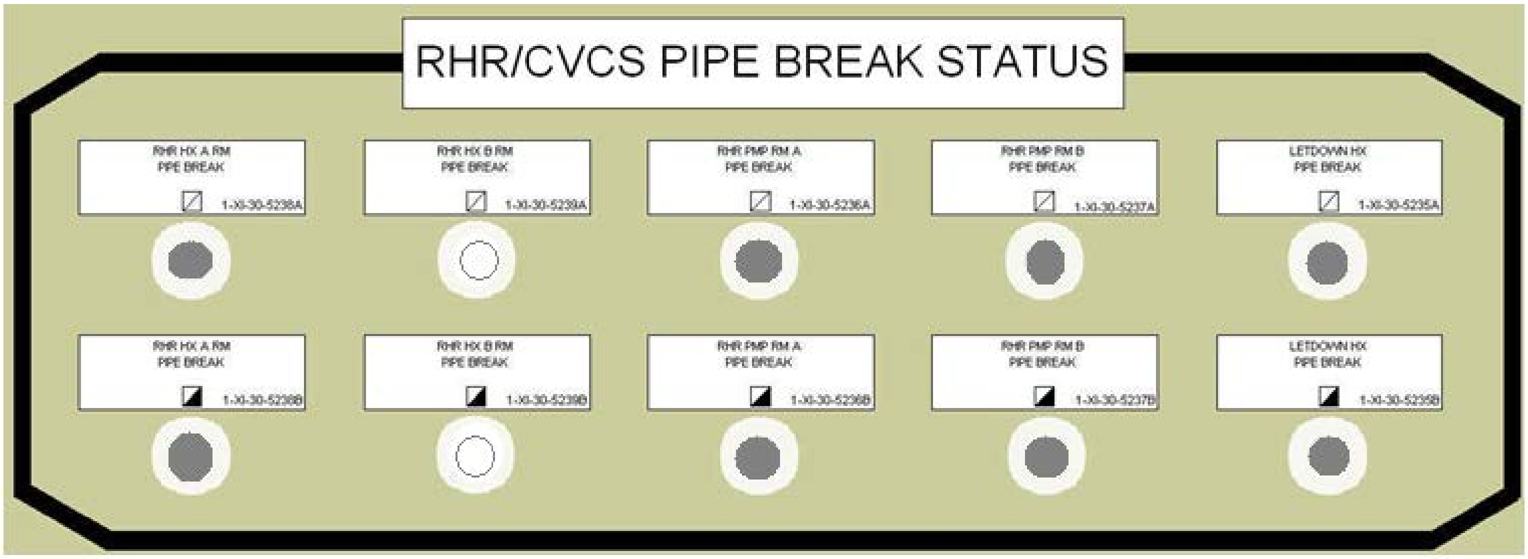
16.

Given the following conditions:

- Unit 1 is at 100% power.

Subsequently:

- Unit 1 experienced a LBLOCA.
- The following **VALID** indications are noted:



- ECA-1.2, LOCA Outside Containment has been completed.
- RCS pressure is **RISING ⇑.**

Which ONE of the following completes the statements listed below?

The INCREASED _____(1)_____ in the applicable room caused the indicated lights to illuminate.

After the completion of ECA-1.2, RHR flow to _____(2)_____ cold legs exists.

A. (1) TEMPERATURE
   (2) 2

B. (1) TEMPERATURE
   (2) 4

C. (1) PRESSURE
   (2) 2

D. (1) PRESSURE
   (2) 4

CORRECT ANSWER: A

DISTRACTOR ANALYSIS:

A. Correct: As seen in SSD- 1-TS-30-5238A, RHR Hx RM B PIPE BREAK light on 1-M-6 is actuated by a temperature switch. The temperature switch detects the rise in room temperature resultant from a high energy line break. Therefore, it is correct that the light will illuminate on temperature.

During a LBLOCA, the RCS pressure will drop to a point at which the RHR pumps will provide low pressure injection flow. If one train of RHR develops a break such that some injection flow is lost to the break; RCS pressure will drop as the equilibrium achieved by the ECCS injection and the break outflow is disturbed (e.g. there will be more fluid going out of the system than there is going into the system).

Given this discussion, the performance of ECA-1.2, will be successful in detecting the fact that after the “B” train RHR ECCS injection is isolated, that the RCS pressure will rise. The procedure will then isolate the “B” RHR train and at its conclusion, two loops of cold leg injection will exist.

B. Incorrect: While it is correct that a high temperature will cause the indicator lights to be lit, it is not correct to believe that injection to four cold legs will exist. It is plausible to believe that because of two reasons. First, one may simply have poor system knowledge and not understand the layout of the cross connect piping between the “A” and “B” trains of RHR. One could believe that a fault of the “B” RHR could be isolated and still allow the “A” train to inject to all four cold legs.

Also, one could fail to realize that ECA-1.2 shuts the cross connect valves 1-FCV-74-33 and 35 (or fail to realize that they are not reopened in step 13).

C. Incorrect: It is incorrect and yet plausible that a rising pressure in a room would indicate a pipe break. The auxiliary building (in which all of these spaces are located) contains multiple pressure detectors which in addition to providing indication and alarm provide inputs to control loops in order that a specific pressure band be maintained.

It is correct that 2 loop injection would remain following the completion of ECA-1.2.

D. Incorrect: Again, it is incorrect and yet plausible that the increased pressure in a space would cause a pipe break light to be lit. Also, it is incorrect and yet plausible that four loop injection would remain following ECA-1.2.

Question Number: 16

Tier: 1 Group: 1

K/A: Westinghouse
E04 LOCA Outside Containment
EK2. Knowledge of the interrelations between the (LOCA Outside Containment) and the following:
EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Importance Rating: 3.8 4.0

10 CFR Part 55: (CFR: 41.7 / 45.7)

10CFR55.43.b: Not applicable

K/A Match: The K/A is matched because the applicant must consider the effect of the LOCA Outside of containment (in this case in the "B" RHR Hx Room) to the RHR system (the RHR Hx Room itself) and then use this interrelation to determine the final status of the ECCS's operation (the overall operation of the facility).

Technical Reference: ECA-1.2, LOCA Outside Containment,

Proposed references to be provided: None

Learning Objective: 3-OT-ECA0101, 1-ECA-1.1 AND 1-ECA-1.2 Loss Of RHR Recirc, LOCA Outside Containment
8. Explain the purpose and basis of each step in ECA-1.1 and 1.2.

Cognitive Level:
- Higher X
- Lower ____

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

SCALING AND SETPOINT DOCUMENT
BISTABLES AND SWITCHES

Instrument No.: 1-TS-30-5238A   Mfrr: SOR   Model: (1)
Vendor ID No: N/A   Contract No: PR W-7698
Type: Switch
Instrument Ranges: Input: 40-225 Deg F   Document: NE SSD 1-T-30-5238A

Switch No. 1 1-TS-30-5238A   Function: (2)

Setpoint	: 134 Deg F	Document:	NE SSD 1-T-30-5238A
Resetpoint	: Fixed	Document:	NE SSD 1-T-30-5238A
Accuracy	: (3)	Document:	NE SSD 1-T-30-5238A
Switch Action	: CL INCR		

Assumptions: None

Head Suppression Data: N/A

Associated Drawings: 1-45W600-57-37, 45N1635-114, 45N1645-3, 45N1695-2, 45N1689-2, 47W600-188, Manual ID WBN-VTM-S382-0010

Remarks:
(1) Model: 201TA-BB125-JJTTX6
(2) Illuminates 1-XI-30-5238A (1-M-6) through relay RHR5 (1-R-74). Annunciates 1-XA-55-6A-113B through relay RHR5 (1-R-74) and relay RHS5 (1-R-80).
(3) Accuracy As Found: +7.9, -19.9 Deg F As Left: +/- 3.0 Deg F
(4) HIGH ALLOWABLE VALUE: 144.0 Deg F
(5) DCN M-16271-A installed instrument & issued NE SSD.
(6) DCN S-36364-A added vent plug requirements.

Function: RHR Heat Exchanger Room A Temperature
Location: EL 713, COL A7-V, PNL 1-L-733A, RHR Heat Exchanger Room A
Reviewed by: D. Echols   Approved by: R. S. Henderson   Date: 01/27/2004

INSTRUMENT CALIBRATION RECORD

Instrument No: 1-TS-30-5238A

HEAD: N/A

WID NO:

TEST			AS FOUND			AS LEFT		
POINT	( )	( )	LO LIMIT	AS FOUND	HI LIMIT	LO LIMIT	AS LEFT	HI LIMIT
1								
2								
3								
4								
5								
6								
7								
8								
9								

BISTABLES/SWITCHES

SET.PT		REQUIRED	AS FOUND			AS LEFT		
SWITCH ID. / ACTION		(Deg F)	LO LIMIT	AS FOUND	HI LIMIT	LO LIMIT	AS LEFT	HI LIMIT
1-TS-30-5238A	SET.PT	134.0	114.1		141.9	131.0		137.0
CL INCR	RESET	Fixed	N/A		N/A	N/A		N/A
	SET.PT							
	RESET							

TEST EQUIPMENT	TVA TAG NO.	CAL DUE DATE

Remarks: (1) IM to functionally check 1-XI-30-5238A on 1-M-6
SAT [ ] UNSAT [ ] If UNSAT, explain.
(2) IM to functionally check 1-XA-55-6A-113B SAT [ ] UNSAT [ ]
If UNSAT, explain.
(3) HIGH ALLOWABLE VALUE OF 144.0 DEG F EXCEEDED. YES [ ] NO [ ]
(4) IM to verify that vent plug is removed from switch housing.
YES [ ]

CREW NO	OOT ( )YES ( )NO	PC ( )YES ( )NO	
PERFORMED BY/DATE	REVIEWED BY/DATE (SIMF)	INSTRUCTION NO.	REV. NO.

Function: RHR Heat Exchanger Room A Temperature

Location: EL 713, COL A7-V, PNL 1-L-733A, RHR Heat Exchanger Room A

Reviewed by: D. Echols Approved by: R. S. Henderson Date: 01/27/2004

Watts Bar Nuclear Plant

Unit 1

Emergency Operating Instruction

ECA-1.2

LOCA Outside Containment

Revision 0005

Quality Related

Level of Use: Continuous Use

The leak postulated for this question is one which exists in the "B" RHR Hx room and is downstream of the Hx outlet check valve (i.e. before the 1-FCV-74-28 throttle valve).

Effective Date: 12-20-2010

Responsible Organization: OPS, Operations

Prepared By: Nicholas Armour

Approved By: Biran McIlnay

Current Revision Description

Minor/editorial revision: Converted to Word 2007 (PCR 4919).

Step	Action/Expected Response	Response Not Obtained

3.0 OPERATOR ACTIONS

These valves were shut at the beginning of the event (i.e. that the Unit was at 100% power.)

1. **ENSURE** RHR suction from RCS CLOSED:
   - 1-FCV-74-1 and 1-FCV-74-9.

     AND
   - 1-FCV-74-2 and 1-FCV-74-8.

This procedure was entered with a VALID pipe break light for the "B" RHR pump room.
Therefore, isolating the RHR "B" train components will isolate the break.

2. **ENSURE** SI pumps hot leg injection 1-FCV-63-156 and 1-FCV-63-157 CLOSED.

3. **ENSURE** RCS letdown isolated:
   - Letdown isolation 1-FCV-62-69 and 1-FCV-62-70 CLOSED.
   - Excess letdown isolation 1-FCV-62-54 and 1-FCV-62-55 CLOSED.

These are the cross tie valves which allow either train of RHR to inject to all four cold legs. If either is shut, then each train of RHR will ONLY be able to inject to two cold legs.

4. **ENSURE** RHR hot leg injection 1-FCV-63-172 CLOSED.

5. **CHECK** RCS press DROPPING or stable.

   RNO: ** **GO TO** Step 14.

The leak is not on the "A" train.

6. **CLOSE** RHR crosstie valve 1-FCV-74-33 or 1-FCV-74-35.

7. **CLOSE** RHR Train A cold leg injection valve 1-FCV-63-93.

   RNO: **IF** 1-FCV-63-93 failed OPEN, **THEN**

   ** **GO TO** Step 10.

Step	Action/Expected Response	Response Not Obtained
8.	**CHECK** LOCA isolated: • RCS press rising.	**OPEN** 1-FCV-63-93. ** **GO TO** Step 10.
9.	**ISOLATE** RHR Train A: a. **STOP** RHR pump A-A, **AND** **PLACE** in PULL TO LOCK. b. **CLOSE** RHR suction valve 1-FCV-74-3. c. ** **GO TO** Step 15.	
10.	**CLOSE** RHR Train B cold leg injection valve 1-FCV-63-94.	**IF** 1-FCV-63-94 failed OPEN, **THEN** ** **GO TO** Step 13.
11.	**CHECK** LOCA isolated: • RCS press rising.	**OPEN** 1-FCV-63-94. ** **GO TO** Step 13.
12.	**ISOLATE** RHR Train B: a. **STOP** RHR pump B-B, **AND** **PLACE** in PULL TO LOCK. b. **CLOSE** RHR suction valve 1-FCV-74-21. c. ** **GO TO** Step 15.	

It won't be since the leak is not on the "A" train.

Once this valve is closed, the postulated leak is isolated from the RCS. As such, pressure will rise.

This is a YES.

Once this is accomplished, the "in's and out's" of the "B" train RHR are isolated.

Step	Action/Expected Response	Response Not Obtained
13.	**ENSURE** RHR crosstie valves 1-FCV-74-33 and 1-FCV-74-35 OPEN.	Notice that this step was skipped (GO TO 15 in a previous step).
14.	**IDENTIFY** break location: • Radiation Protection surveys. • RHR pipe break lights [M-6]. • ECCS pump flows. • Aux bldg flood alarms [M-15; light panel, Aux Bldg 757]. • Radiation area monitor recorders 1-RR-90-1 and 0-RR-90-12A.	**NOTIFY** TSC of failure to identify break location.
15.	**DETERMINE** appropriate Instruction: • **IF** LOCA outside cntmt isolated, **THEN** ** **GO TO** E-1, Loss of Reactor or Secondary Coolant.	**NOTIFY** TSC of failure to isolate break. ** **GO TO** ECA-1.1, Loss of RHR Sump Recirculation.

End of Section

Indicates the isolations provided by ECA-1.2
Area of Leak.
1-FCV 63-172
63-637
SET PRESS. 600 PSI
FE 63-151
63-309A
FT 63-173
63-310A
63-627
SET PRESS. 600 PSI
TO RHR SPRAY CONT ON 1-47W812-1, E-4
CONT ON 47W813-1, H-5, AT PEN. X-24
FM PEN X30 RELIEF CONT ON 1-47W811-1, D-6
OR-63-4
SEE NOTE 20
63-11
VTV-63-16
VTV-63-17
63-506 1" DR
FCV 63-7
63-508
63-520
FLOW RESTRICTOR FE-63-173A
63-518
63-690
3/4" DR
63-691
FE 63-92
63-340A
63-339A
FT 63-92
FLOW RESTRICTOR OR-63-92
RESIDUAL HEAT REMOVAL PUMPS
1B-B
1A-A
RESIDUAL HEAT EXCHANGERS
63-629
63-693
FCV 63-94
FCV 63-93
1-DRV-63-691A
3/4" TC
63-696
3/4" TV
63-628
SET PRESS. 600 PSI
63-626
FLOW RESTRICTOR OR-63-91
FE 63-91
FT 63-91
63-338A
63-337A
LETDOWN HX (RHR) 1-47W809-1, B-8
PI 63-57A
63-664
63-369A
63-370A
63-665
PI 63-57B
63-699
MANWAY
ISV-63-73
K-14
63-729
63-730
63-728
3/4" DR
SEE NOTE 20
FCV

17.

Given the following conditions:

- Unit 1 has experienced a LOCA.
- 1-E-1, Loss of Reactor or Secondary Coolant is in progress.
- CNTMT ISOL PHASE B (125-B) is LIT.
- **NO** CCPs are RUNNING.
- ALL SG WR levels are 38%.

Subsequently:

- The crew transitions to 1-FR-H.1, Loss of Secondary Heat Sink.

Which ONE of the following describes the status of the RCPs AND the FIRST actions the crew will perform in accordance with 1-FR-H.1?

The RCPs were ____(1)____ when the crew entered 1-FR-H.1 AND The FIRST action(s) the crew will perform is to ____(2)____.

A. (1) running

(2) initiate RCS bleed and feed

B. (1) **NOT** running

(2) initiate RCS bleed and feed

C. (1) running

(2) establish MD AFW pump flow

D. (1) **NOT** running

(2) establish MD AFW pump flow

<u>CORRECT ANSWER:</u>   <u>B</u>

<u>DISTRACTOR ANALYSIS:</u>

A. Incorrect: Given the conditions in the stem, the operating crew will reach step 2 of 1-FR-H.1, Loss of Secondary Heat Sink and **ENSURE** at least one charging pump RUNNING. The crew will recognize that the GREEN and WHITE lights LIT on 1-HS-62-108A, CCP A-A (ECCS) indicates that the 1A-A CCP has tripped. The crew would then recognize that because 1-EI-57-66, 6.9 SDB 1B-B VOLTS reads 0 that the 1B-B CCP is unavailable (as it has no power available). Because no CCP is available to RUN, the crew will execute the RNO for step 2 of 1-FR-H.1 which states, ** **GO TO** Cautions prior to Step 18 to initiate RCS bleed and feed. The RCPs would not be in operation at the time of 1-FR-H.1 entry. The RCPs would have been secured in accordance with 1-E-1, Loss of Reactor or Secondary Coolant. Most probably step 1 of 1-E-1 would have been utilized which states, **CHECK** if RCPs should remain in service: a. Phase B DARK [MISSP] **STOP** all RCPs. The crew could also have used the Foldout Page of 1-E-1 which identifies that the RCP trip criterion contains the receipt of a Phase B Isolation. Whichever step was selected for use, the crew would have secured the RCPs prior to entering 1-FR-H.1 upon recognizing that Annunciator 125-B, CNTMT ISOL PHASE B is LIT (which would be one of the attending indications that Phase B would be LIT on the MISSP). It is plausible to believe that RCPs would be in service upon the entry into 1-FR-H.1 because such pumps are normally left in service given that their trip criteria are **NOT** met. Additionally, 1-FR-H.1 directs the securing of the RCPs which could lead one to believe that the pumps would be secured in accordance with such procedure.

B. Correct: The crew would initiate RCS bleed and feed and the RCPs would not be in operating at the time of 1-FR-H.1 entry.

C. Incorrect: As mentioned RCS bleed and feed would be required as at least one CCP was **NOT** running. It is plausible that the applicant believe that he would **ESTABLISH** MD AFW pump flow as at least three SG WR levels are greater than 36%. 1-FR-H.1 contains step 3 which explicates the bleed and feed criteria: Any THREE SG WR levels less than or equal to 26% [36% ADV]. The criteria would require the bracketed adverse value to be used as Phase B had occurred. Also, it is not correct but plausible to believe that the RCPs were in service.

D. Incorrect: It is Incorrect to first attempt to restore MD AFW pump flow but correct to recognize that the RCPs were not in service at the time of entry into 1-FR-H.1.

Question Number: 17

Tier: 1 Group: 1

K/A: Westinghouse
E05 Loss of Secondary Heat Sink
EK1. Knowledge of the operational implications of the following concepts as they apply to the (Loss of Secondary Heat Sink)
EK1.3 Annunciators and conditions indicating signals, and remedial actions associated with the (Loss of Secondary Heat Sink).

Importance Rating: 3.9 4.1

10 CFR Part 55: (CFR: 41.8 / 41.10, 45.3)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to possess the knowledge of the operational implications to both the receipt of annunciators (125-B) and conditions indicating signals (the listed board indications). The applicant must then apply this knowledge to the required actions listed in 1-FR-H.1.

Technical Reference: 1-FR-H.1, Loss of Secondary Heat Sink
1-E-1, Loss of Reactor or Secondary Coolant

Proposed references to be provided: None

Learning Objective: 3-OT-FRH0001 , FUNCTIONAL RESTORATION GUIDES 1-FR-H.1, FR-H.2, H.3, H.4 and 1-FR H.5
3. EXPLAIN the purpose for and basis of each step in 1-FR-H.1, FR-H.2, FR-H.3, FR-H.4 and 1-FR-H.5.
4. EXPLAIN the basis for tripping the RCPs during the performance of FR-H.1. (IER 11-3, Having a solid understanding of plant design, engineering principles and sciences)

Cognitive Level:
- Higher X
- Lower ____

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Foldout Page
(Page 1 of 1)

SI REINITIATION CRITERIA

Manually **START** ECCS pumps as necessary:

- PZR level cannot be maintained greater than 15% [33% ADV], OR
- RCS subcooling less than 65°F [85°F ADV]

RCP TRIP CRITERIA

- Phase B Isolation, **OR**

On phase B, the RCPs are secured. This is the reason that the RCPs were secured upon the entry to 1-FR-H.1.

- One charging pump or one SI pump injecting **AND**

  RCS press reduced uncontrolled to less than 1500 psig.

EVENT DIAGNOSTIC TRANSITIONS

- **IF** any S/G press low or dropping uncontrolled **AND**

  S/G has **NOT** been isolated,
  THEN

  ** **GO TO** 1-E-2, Faulted Steam Generator Isolation.

- **IF** S/G radiation abnormal or S/G level rising uncontrolled,
  THEN

  START SI pumps as necessary, **AND**

  ** **GO TO** 1-E-3, Steam Generator Tube Rupture.

SUMP RECIRC SWITCHOVER CRITERIA

- **IF** RWST level less than 34%,
  THEN

  ** **GO TO** 1-ES-1.3, Transfer to RHR Containment Sump.

AFW OPERATION

- **IF** CST volume less than 5000 gal,
  THEN

  MONITOR AFW pumps to ensure suction transfer.

Step	Action/Expected Response	Response Not Obtained

3.0 OPERATOR ACTIONS

CAUTION

- If total feed flow CAPABILITY of 410 gpm is available, this Instruction should **NOT** be performed.
- If an Intact S/G is available, feed flow should **NOT** be reestablished to any faulted S/G.

Step	Action/Expected Response	Response Not Obtained
1.	**CHECK** if secondary heat sink is required:	
	a. RCS pressure greater than any Intact S/G pressure.	a. **RETURN TO** Instruction in effect.
	b. RCS temperature greater than 375°F [360°F ADV].	b. **PLACE** RHR System in service while continuing in this instruction. • **REFER TO** 1-SOI-74.01, Residual Heat Removal System. **WHEN** adequate RHR shutdown cooling established, **THEN** **RETURN TO** Instruction in effect.
2.	**ENSURE** at least one charging pump RUNNING.	**STOP** all RCPs **AND** ** **GO TO** Cautions prior to Step 18 to initiate RCS bleed and feed.

When 1-FR-H.1 was entered, NO charging pumps were running. Therefore, one will initiate bleed and feed.

Step	Action/Expected Response	Response Not Obtained

CAUTION RCS bleed and feed criteria must be monitored for immediate response if the criteria is exceeded.

If the "immediate" transition to bleed and feed is missed (at step) two, one will continue here.

3. **DETERMINE** if RCS bleed and feed required:

a. **CHECK** RCS bleed and feed required:

- Any THREE S/G WR levels less than or equal to 26% [36% ADV].

OR

- RCS pressure greater than or equal to 2335 psig.

RNO: a. **MONITOR** RCS bleed and feed criteria:

WHEN criteria are met, **THEN**

PERFORM Substep 3b.

** **GO TO** Step 4.

The stem of the question provided that sufficient S/G level was available.

b. **STOP** all RCPs, **AND**

** **GO TO** Cautions prior to Step 18 to initiate RCS bleed and feed.

Step	Action/Expected Response	Response Not Obtained
4.	**ENSURE** S/G blowdown ISOLATED.	Manually **CLOSE** valves.
5.	**MONITOR** CST volume greater than 200,000 gal.	**INITIATE** CST refill USING 1-SOI-59.01, Demineralized Water System. **IF** CST volume drops to less than 5000 gal, **THEN** **MONITOR** AFW pumps to ensure suction transfer.

This step begins the saga of attempting to restore secondary feed.

NOTE If the use of condensate flow is anticipated, then a higher PZR level will better accommodate the level shrink from S/G cooldown and depressurization.

6.	**CONTROL** PZR level between 29% and 63% [47% and 58% ADV].	

Step	Action/Expected Response	Response Not Obtained
7.	**ESTABLISH** MD AFW pump flow:	Try to get AFW back first.
	a. **CHECK** MD AFW pump AVAILABLE.	a. ** **GO TO** Step 8.
	b. **ENSURE** both MD AFW pumps RUNNING.	b. **START** pumps from the shutdown boards.
	c. **ENSURE** MD AFW LCVs OPEN.	c. **OPEN** MD LCVs from the auxiliary control room, **OR** Locally **OPEN** MD LCVs and manual isolation valves USING 1-SOI-3.02, Auxiliary Feedwater System.
	d. **CHECK** MD AFW pump flow greater than 410 gpm.	d. **ENSURE** AFW valve alignment USING 1-SOI-3.02, Auxiliary Feedwater System.
	e. **CHECK** NR level in at least one S/G greater than 29% [39% ADV].	e. **MAINTAIN** total feed flow to S/Gs greater than 410 gpm UNTIL NR level in at least one S/G greater than 29% [39% ADV]. **WHEN** NR level in at least one S/G greater than 29% [39% ADV], **THEN** **RETURN TO** instruction in effect. ** **GO TO** Step 8.
	f. **RETURN TO** Instruction in effect.	

18.

Given the following timeline:

09:00:00	Unit 1 trips.
09:20:00	A Small Break LOCA occurs.
09:50:00	The crew transitioned to 1-ECA-1.1, Loss of RHR Sump Recirculation.
10:30:00	The crew is performing the Response Not Obtained (RNO) for 1-ECA-1.1, Step 19.b, ESTABLISH minimum ECCS flow for decay heat removal.

Which ONE of the following describes the **MINIMUM** ECCS flow rate that satisfies the RNO AND the basis for this requirement in accordance with the Westinghouse Background Document?

In accordance with Figure 1 of 1-ECA-1.1, the **MINIMUM** ECCS flow rate is ____(1)____ gpm AND the basis for establishing this **MINIMUM** ECCS flow rate is to ____(2)____ .

REFERENCE PROVIDED

A. (1) 325
   (2) delay depletion of the RWST

B. (1) 325
   (2) ensure adequate RVLIS level

C. (1) 400
   (2) delay depletion of the RWST

D. (1) 400
   (2) ensure adequate RVLIS level

CORRECT ANSWER: A

DISTRACTOR ANALYSIS:

A. Correct: From 0900 - 1030 (90 Min) Using 1-ECA-1.1, figure 1, the minimum amount of SI flow needed to match decay heat is approximately 325 gpm. The value of 325 gpm is in the acceptable region using the graph from time of trip AND meets the requirement of Minimum Flow to delay RWST depletion. The Basis states the operator is then instructed to establish the minimum SI pump flow needed to match decay heat in order to further decrease SI pump flow and delay RWST depletion.

B. Incorrect: Plausible since the value of 325 gpm would meet the requirements of the step to match the SI flow needed to match decay heat. Also plausible, since maintaining RVLIS is identified in the procedure prior to step 20 and is accomplished by keeping ECCS pumps running (not throttling) for core makeup. Incorrect as to be in step 19.b requires that that RVLIS level was adequate at this point in the procedure

C. Incorrect: Plausible since this flow rate would be in the acceptable range on 1-ECA-1.1 figure 1, however this does not meet the intent of the step which is to reduce flow as low as possible to slow the rate of RWST depletion. The second part is correct.

D. Incorrect: Plausible since this flow rate would be in the acceptable range on 1-ECA-1.1 figure 1, however this does not meet the intent of the step which is to reduce flow as low as possible to slow the rate of RWST depletion. Also plausible, since maintaining RVLIS is identified in the procedure prior to step 20 and is accomplished by keeping ECCS pumps running (not throttling) for core makeup. Incorrect as to be in step 19.b requires that that RVLIS level was adequate at this point in the procedure.

Question Number: 18

Tier: 1 Group: 1

K/A: Westinghouse
E11 Loss of Emergency Coolant Recirculation
EK3. Knowledge of the reasons for the following responses as they apply to the (Loss of Emergency Coolant Recirculation)
EK3.4 RO or SRO function within the control room team as appropriate to the assigned |position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

Importance Rating: 3.6 3.8

10 CFR Part 55: (CFR: 41.5 / 41.10, 45.6, 45.13)

10CFR55.43.b: Not applicable

K/A Match: This question matches the K/A by having the candidate recall the reason for the actions that the crew will take to comply with the requirements of ECA-1.1 while attempting to reduce the depletion rate of the RWST while unable to transfer to CNMT sump recirculation.

Technical Reference: 1-ECA-1.1, Loss of RHR Sump Recirculation
Westinghouse Owners Group background document for 1-ECA-1.1

Proposed references to be provided: Figure 1 of 1-ECA-1.1

Learning Objective: 3-OT-ECA0101, 1-ECA-1.1 AND 1-ECA-1.2 Loss Of RHR Recirc, LOCA Outside Containment
8. Explain the purpose and basis of each step in ECA-1.1 and 1.2.

Cognitive Level:
Higher X
Lower ____

Question Source:
New ____
Modified Bank X
Bank ____

Question History: Modified bank question derived from Sequoyah. Used at Sequoyah for the 05/2013 exam.

QUESTIONS REPORT

for SQN-WBN composite exams

1. W/E11 EK3.4 318

Given the following plant conditions:

- At 0900 the Unit 1 Reactor Trips.
- At 0920 a small break LOCA occurs.
- At 0950 the crew transitioned to ECA-1.1, "Loss of RHR Sump Recirculation", due to the failure of both RHR pumps.
- Crew has established one train of ECCS flow per ECA-1.1.
- SI flow cannot be terminated due to lack of subcooling.
- At 1030 the crew is performing ECA-1.1 Step 20.b, "Monitor if ECCS flow should be terminated:"

Which one of the following:

(1) identifies the **minimum** ECCS flow rate per "Curve 9" of ECA 1.1 that meets the intent of ECA-1.1, Step 20.b RNO,

and

(2) the reason for the minimum ECCS flow rate?

REFERENCE PROVIDED

A.✓ (1) 325 gpm ECCS flow.
(2) To delay RWST depletion.

B. (1) 325 gpm ECCS flow.
(2) To ensure adequate RVLIS level.

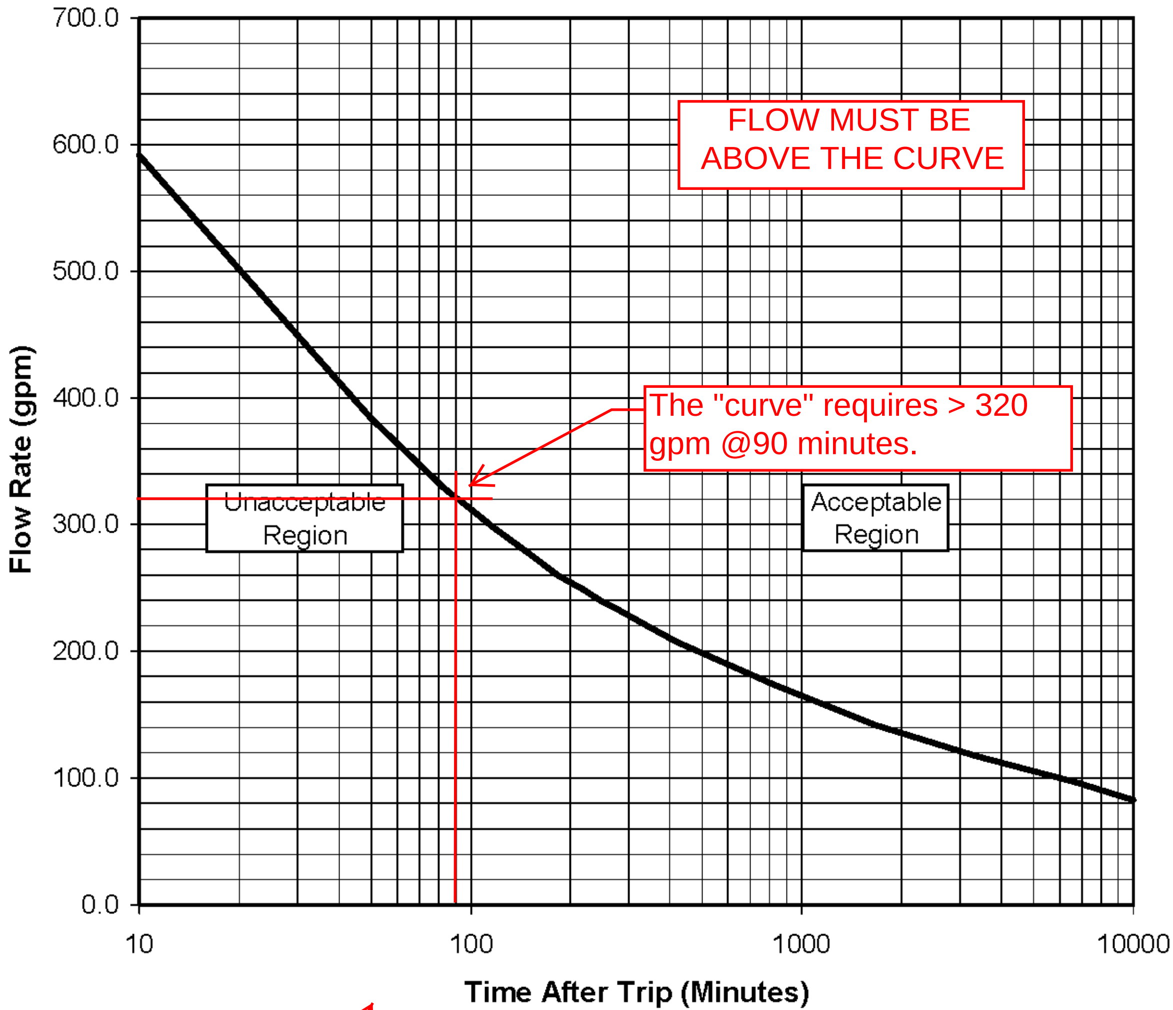
C. (1) 400 gpm ECCS flow.
(2) To delay RWST depletion.

D. (1) 400 gpm ECCS flow.
(2) To ensure adequate RVLIS level.

The plant tripped 90 minutes ago (10:30 - 9:00 = 90 minutes)

Figure 1
(Page 1 of 1)

Minimum SI Flow for Decay Heat vs. Time After Trip



log scale

STEP: Check If SI Can Be Terminated

PURPOSE: To determine if conditions have been established which indicate that one train of SI flow is no longer required

BASIS:

Following the reduction to one train of SI, RCS conditions may be within acceptable limits for SI termination to be allowed. The combination of a minimum subcooling and sufficient liquid level in the vessel to cover the core represents less restrictive SI termination criteria in this guideline because SI flow may prevent a subsequent reduction in RCS pressure and cause considerable depletion of the RWST.

The operator is to use the graph in ECA-1.1 to "delay RWST depletion."

The subcooling criterion will ensure subcooled conditions and the RVLIS indication ensures the existence of an adequate vessel inventory such that core cooling is ensured. Refer to document SI TERMINATION/REINITIATION in the Generic Issues section of the Executive Volume.

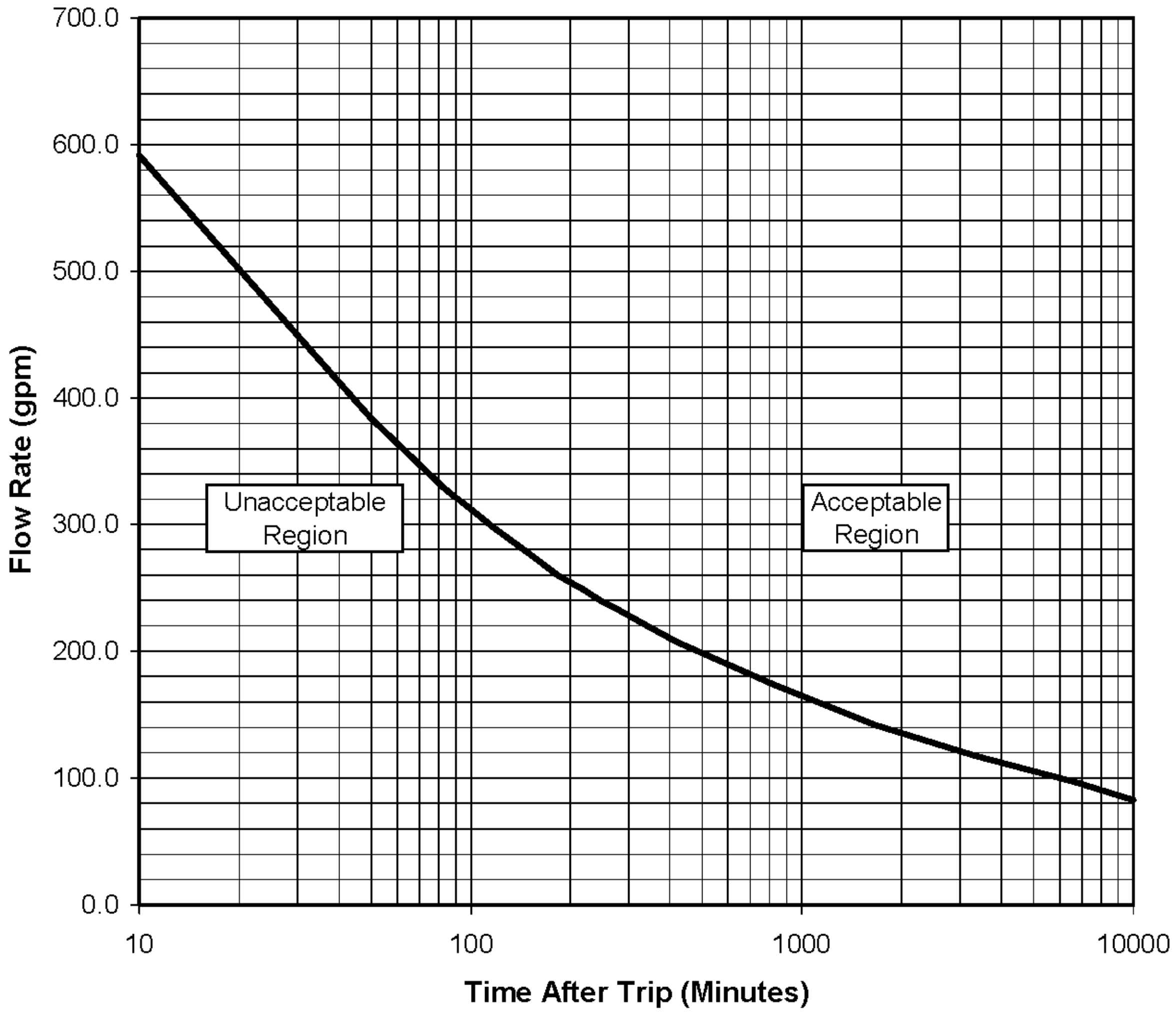
If the termination criteria are not satisfied, then SI is required to ensure core cooling and should not be terminated. If RVLIS indication is adequate but RCS subcooling is not, the operator is then instructed to establish the minimum SI pump flow needed to match decay heat in order to further decrease SI pump flow and delay RWST depletion. This is done by aligning (if necessary) and operating the appropriate SI pumps (charging/SI pumps, high head SI pumps and low-head SI pumps) such that the flow required to match decay heat is established. For most Westinghouse plants, the flow through SI lines cannot be throttled and the exact flow rate required cannot be established. Therefore, in order to establish the minimum SI flow required in this step, the operator should stop appropriate SI pump(s) to establish flow equal to or greater than the minimum SI flow required to match decay heat. The SI flow needed to match decay heat is a function of time and is obtained from Figure 1.

The graph to be used when determining the minimum ECCS injection rate.

Figure 1 is a generic curve with units for flowrate of gpm per MWt. Each utility must develop a plant specific curve for its plant from Figure 1, and this curve would be included in the plant specific emergency operating procedure as Figure ECA11-1. This plant specific curve can be developed by modifying Figure 1 as follows: The Y-axis values for flowrate in gpm/MWt should be multiplied by the plant specific MWt core rating to obtain flowrate values in GPM. The X-axis values for time in minutes are used without modification. A plant specific curve is then plotted as flowrate (gpm) versus

Figure 1
(Page 1 of 1)

Minimum SI Flow for Decay Heat vs. Time After Trip



19.

Given the following conditions:

- Unit 1 is performing a power ascension from 60% to 100%.
- In accordance with the Reactivity Management Plan, the OAC is withdrawing control rods and notices that CBD rod D12 has stopped moving with its group.
- 1-MON-85-5000/1, CERPI MONITOR 1 is displaying Shutdown Banks Rod Positions.
- 1-MON-85-5000/2, CERPI MONITOR 2 is displaying Control Banks Rod Positions.

Subsequently:

- 1-MON-85-5000/2, CERPI MONITOR 2 goes **BLANK**.

Which ONE of the following describes the Unit 1 T/S, LCO 3.1.5, Rod Group Alignment Limits AND the Main Control Room (MCR) **ONLY** Indications for Control Rod Bank Positions?

T/S, LCO 3.1.5 would **NOT** be met when D12 is ____(1)___ steps from the group step counter demand position.

1-MON-85-5000/1, CERPI MONITOR 1 ____(2)____ be used to observe Control Banks Rod Positions.

A. (1) 2
   (2) can

B. (1) 2
   (2) can **NOT**

C. (1) 12
   (2) can

D. (1) 12
   (2) can **NOT**

CORRECT ANSWER: C

DISTRACTOR ANALYSIS:

A. Incorrect: In accordance with Unit 1 Technical Specification LCO 3.1.5, Rod Group Alignment Limits, All shutdown and control rods shall be OPERABLE, with all individual indicated rod positions within 12 steps of the group step counter demand position It is plausible to believe that a difference of 2 steps is the point at which the T/S is **NOT** met as a similar Unit 1 Technical Requirement exists. T/R 3.1.7, Position Indication System, Shutdown declares that The group demand position indicators shall be OPERABLE and capable of determining within +/- 2 steps the demand position for each shutdown or control rod that is not fully inserted. As seen in the CERPI Technical Description, WNA-CT-00053-WAT, each of the Operator Flat Panel Displays possesses 5 operator selectable screens. The two screens which are normally in use are (1) Control Banks A, B, C, and D Rod Positions Display and (2) Shutdown Banks A, B, C, and D Rod Positions Display. Normally, the control room staff will have the shutdown banks' information displayed on the CERPI monitor 1. The control banks' information would be displayed on CERPI monitor 2. In the event of a failure of a monitor the control room staff would select the required information on the remaining operable monitor.

B. Incorrect: It is neither correct that the tolerance for LCO 3.1.5 is within 2 steps nor is it correct that the control room staff can **NOT** select the required rod position information using actions taken from the MCR and the remaining CERPI monitor. It is plausible to believe that the foregoing were true as seen on print 2E10010, all of the information produced by the two CERPI computers (PLC A and B) is passed through a Maintenance and Test Panel (A and B). It is conceivable that one may believe that action taken at the MTP may be required. Additionally, one may simply not recollect that each of the displays can produce the positions of all control rods because of the fact that for the vast majority of the time monitor 1 displays the shutdown banks and monitor 2 displays the control banks.

C. Correct: The LCO 3.1.5 requires that rods be within 12 steps of the group demand and the MCR staff is able to select the required information on the CERPI monitors.

D. Incorrect: While it is correct that LCO 3.1.5 requires that rods be within 12 steps of the group demand, it is not true but plausible that the remaining CERPI monitor can display the required information.

Question Number: 19

Tier: 1 Group: 2

K/A: 005 Inoperable/Stuck Control Rod
AA1. Ability to operate and / or monitor the following as they apply to the Inoperable / Stuck Control Rod:
AA1.05 RPI

Importance Rating: 3.4 3.4

10 CFR Part 55: (CFR 41.7 / 45.5 / 45.6)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to monitor and manipulate the CERPI screens (and a remaining CERPI screen) given that one rod is stuck. The applicant must also be able to detect a stuck rod by identifying the degree of misalignment required by the T/S.

Technical Reference:
TS 3.1.5 Rod Group Alignment Limits
TR 3.1.7 Position Indication System, Shutdown
WNA-GO-00010-WAT CERPI OPERATION AND MAINTENANCE MANUAL
Westinghouse drawing 2E10010

Proposed references to be provided: None

Learning Objective:
3-OT-SYS085A, Control Rod Drive System
2. DESCRIBE the design criteria, purpose and/or functions of the Rod Position
Indicating (RPI) System and the major system components listed below:
a. Computer Enhanced Rod Position Indicator (CERPI)
14. DESCRIBE the following aspects of Technical Specifications and Technical
Requirements for this system:
a. The conditions and required actions with completion time of one hour or
less
b. The Limiting Conditions for Operation, Applicability, and Bases.

Cognitive Level:
Higher ____
Lower X

Question Source:

New _X___

Modified Bank _____

Bank _____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

A misaligned or stuck rod is 12 steps out from its GROUP.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Rod Group Alignment Limits

LCO 3.1.5 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 ≥ 1.6% Δk/k.	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 Restore rod to within alignment limits.	1 hour
	<u>OR</u>	
	B.2.1.1 Verify SDM is ≥ 1.6% Δk/k.	1 hour
	<u>OR</u>	
		(continued)

The GPI (group position indicator - digital step demand meter on 1-M-4) must be able to show within +/- 2 steps, rod control's command to the rods in that group.

TR 3.1 REACTIVITY CONTROL SYSTEMS

TR 3.1.7 Position Indication System, Shutdown

TR 3.1.7 The group demand position indicators shall be OPERABLE and capable of determining within ± 2 steps the demand position for each shutdown or control rod that is not fully inserted.

APPLICABILITY: MODES 3, 4, and 5, when the reactor trip breakers are closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more group demand position indicators inoperable.	A.1 Open reactor trip breakers.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.1.7.1	Determine that each group demand position indicator is OPERABLE by movement of the associated shutdown or control rod 10 steps in any one direction.	Within 4 hours after closing the reactor trip breakers if not completed within previous 31 days. <u>AND</u> 31 days thereafter

5.1.14 System Tunable Parameters Display

5.1.14.1 Following tunable parameters are displayed and can be adjusted by either numeric entry or by increment / decrement buttons as appropriate:

- Rod Bottom Alarm setpoint and hysteresis values
- Rod Bottom Bypass setpoint and hysteresis values
- Rod-to-Rod Deviation Limit value
- Rod-to-Bank Deviation Limit value
- Rod Withdrawal Limit value
- Wrong Direction Alarm setpoint value

5.1.15 Alarm Selection Display

5.1.15.1 A display screen that contains an Enable/Disable toggle button for the following alarms is provided:

- Rod-to-Rod Deviation Alarm
- Rod-to-Bank Deviation Alarm
- Rod Bottom Alarm
- Any Rod On Bottom Alarm
- All Rods On Bottom Alarm
- Rod Withdrawal Limit Alarm
- RDTC Trouble Alarm
- Rod Wrong Direction Alarm

5.2 OPERATOR FLAT PANEL DISPLAY (OFPD)

5.2.1 General

5.2.1.1 There are 2 OFPDs in the Main Control Room.

5.2.1.2 Each OFPD consists of 5 operator selectable screens that are selectable from either the toucscreen or the trackballs, which are connected to each OFPD.

5.2.1.3 Operator functions are limited to selecting screens, monitoring individual rod positions, and evaluating the system status. No operator action is capable of affecting the CERPI system calibration or programming.

5.2.1.4 Date and time is not displayed on OFPD screens.

5.2.1.5 Each OFPD consists of a PC Node Box, a touchscreen flat panel display, and a trackball.

5.2.2 Calibration Adjustments

5.2.2.1 PLC tunable parameters or the PLC calibration constants are not adjustable via the OFPD.

5.2.3 Security Features

This screen shows the control banks.

5.2.3.1 There are no access-protected screens on OFPD.

5.2.4 Control Banks A, B, C, and D Rod Positions Display

This is the screen which is normally displayed on the RIGHT-hand CERPI.

5.2.4.1 This display screen shows the positions of the rods in Control Banks A, B, C, and D in a bar graph format.

5.2.4.2 The ability to identify a rod position value affected by a faulty analog input is provided.

5.2.4.3 An indication for the rod bottom position of each rod is provided.

5.2.4.4 “Any Rod On Bottom”, “All Rods On Bottom”, “Rod-to-Bank Deviation”, “Rod-to-Rod Deviation”, “Rod Wrong Direction”, and “RPI System Trouble” alarm signals are displayed on this screen.

5.2.4.5 RPI System Trouble indicator has touch button capability and once it is touched, screen navigates to System Status Display.

5.2.4.6 Rod Demand from Passive Summer and Rod Speed indicators are provided in bar graph format.

5.2.4.7 Under normal operating conditions, either PLC A or PLC B can be selected to provide the information on this screen. If the main PLC (PLC A) is not available, then the information on this screen will be supplied by PLC B automatically.

And this screen shows the shutdown banks.

5.2.5 Shutdown Banks A, B, C, and D Rod Positions Display

This is the screen which is normally displayed on the LEFT-hand CERPI screen.

5.2.5.1 This display screen shows the positions of the rods in Shutdown Banks A, B, C, and D in a bar graph format.

5.2.5.2 The ability to identify a rod position value affected by a faulty analog input is provided.

5.2.5.3 An indication for the rod bottom position of each rod is provided.

5.2.5.4 "Any Rod On Bottom", "All Rods On Bottom", "Rod-to-Bank Deviation", "Rod-to-Rod Deviation", "Rod Wrong Direction", and "RPI System Trouble" alarm signals are displayed on this screen.

5.2.5.5 RPI System Trouble indicator has touch button capability and once it is touched, screen navigates to System Status Display.

5.2.5.6 Rod Demand from Passive Summer and Rod Speed indicators are provided in bar graph format.

5.2.5.7 Under normal operating conditions, either PLC A or PLC B can be selected to provide the information on this screen. If the main PLC (PLC A) is not available, then the information on this screen will be supplied by PLC B automatically.

This screen, while not typically displayed, can display all of the rods at once.

5.2.6 All Rods Display

5.2.6.1 This display screen shows all system rod position values simultaneously.

5.2.6.2 The ability to identify a rod position value affected by a faulty analog input is provided.

5.2.6.3 "Any Rod On Bottom", "All Rods On Bottom", "Rod-to-Bank Deviation", "Rod-to-Rod Deviation", "Rod Wrong Direction", and "RPI System Trouble" alarm signals are displayed on this screen.

5.2.6.4 RPI System Trouble indicator has touch button capability and once it is touched, screen navigates to System Status Display.

20.

Given the following timeline:

00:00:00	Unit 1 is performing FR-I.2, Low Pressurizer Level. RCS Subcooling indicates 0 °F.
00:01:00	A PORV OPENS. RCS Pressure LOWERS ⇓ 300 psia.

Which ONE of the following describes the response of PZR Level AND the reason for this response?

At time 00:01:05 INDICATED PZR Level will be ____(1)____ ACTUAL PZR Level at time 00:01:00 due to ____(2)____.

	(1)	(2)
A.	HIGHER	the change in specific volume of the PZR water
B.	HIGHER	Containment Temperature and Pressure RISING ⇑
C.	LOWER	the change in specific volume of the PZR water
D.	LOWER	Containment Temperature and Pressure RISING ⇑

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Correct: Assume PZR pressure is 2000 psia and the actual height of water measured in the PZR is 20 feet.

$$P = \frac{\rho g z}{g_c} \qquad P = \frac{gz}{\upsilon g_c} \qquad \rho = \frac{1}{\upsilon}$$

Pressure at 2000 psia

32.2 ft	20 ft	lbm	$lbfs^2$	1 ft^2
s^2		0.02565 ft^3	32.2 ft lbm	144 in^2

P = 5.415

Pressure at 1700 psia

32.2 ft	20 ft	lbm	$lbfs^2$	1 ft^2
s^2		0.02428 ft^3	32.2 ft lbm	144 in^2

P = 5.720

If initial conditions (2000 psia and 20 feet) provided an accurate indication of PZR water level, a pressure of 5.415 psia corresponds to an indicated height of 20 feet.

Then at the new conditions (1700 psia and 20 feet) the higher pressure generated 5.720 psia, since the pressure is higher indicated PZR level would be higher than 20 feet.

To indicate 20 feet of water, the pressure must be equal to 5.415 psia.

Calculate the indicated water level if 5.415 psia was created with the PZR pressure at 2000 psia.

5.415 lbf	0.02565 ft^3	32.2 ft lbm	s^2	144 in^2
in^2	lbm	$lbfs^2$	32.2 ft	1 ft^2

L = 20.01 ft

Calculate the indicated water level if 5.720 psia was created with the PZR pressure at 1700 psia.

5.720 lbf	0.02428 ft^3	32.2 ft lbm	s^2	144 in^2
in^2	lbm	$lbfs^2$	32.2 ft	1 ft2

L = 19.98 ft

B. Incorrect: It is plausible that the applicant may not understand the operation of the PORV and the PRT and may think that a 300 psia decrease in pressure may cause the PRT to rupture and therefore change the external influences on the PZR Level Detector.

C. Incorrect: See explanation above. The 2nd part is correct.

D. Incorrect: See explanation above. It is plausible that the applicant may not understand the operation of the PZR Level Detector OR how the PORV and the PRT will respond and may think that a 300 psia decrease in pressure may cause the PRT to rupture and therefore change the external influences on the PZR Level Detector.

Question Number: 20

Tier: 1 Group: 2

K/A: 028 Pressurizer (PZR) Level Control Malfunction
AK3. Knowledge of the reasons for the following responses as they apply to the Pressurizer Level Control Malfunctions:
AK3.03 False indication of PZR level when PORV or spray valve is open and RCS saturated

Importance Rating: 3.5 4.1

10 CFR Part 55: (CFR 41.5,41.10 / 45.6 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question has established a plant condition in which PZR level has malfunctioned and a condition which the indicated PZR level is false (PZR Level Malfunction and FR-I.2, Low Pressurizer Level). The question then requires that the applicant to understand how the actual and indicated PZR level will differ and the reason for the false indication.

Technical Reference: FR-1.2, Low Pressurizer Level
GFES Sensors and Controllers

Proposed references to be provided: None

Learning Objective:

Cognitive Level:
- Higher X
- Lower ____

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

21.

Given the following conditions:

- Unit 1 Core offload is in progress.
- N31, Channel I SRNI **FAILS**.
- 1-AOI-4, Nuclear Instrumentation Malfunctions is entered.
- Audible count rate **IS** audible in the main control room.
- Audible count rate **IS NOT** audible inside containment.

Which ONE of the following describes the requirements regarding Core Offload and the required operator action in accordance with 1-AOI-4?

Core offload ____(1)____ .

Audio count rate in containment, ____(2)____ be restored by manipulating the switch circled in RED in the following picture.

A. (1) must STOP
   (2) can

B. (1) may CONTINUE
   (2) can

C. (1) must STOP
   (2) can **NOT**

D. (1) may CONTINUE
   (2) can **NOT**

<u>CORRECT ANSWER:</u>   <u>C</u>

<u>DISTRACTOR ANALYSIS:</u>

A. Incorrect: In accordance with 1-AOI-4, the operators must STOP positive reactivity changes or core alterations. A subset of core alterations is fuel offload. Therefore, it is correct that core offload must stop. It is plausible to believe that fuel offload may continue given the failure of only one source range instrument because of two factors. One may believe that because there are two SRNIs that fuel movement may continue as long as one is operational. Secondly, one may believe that fuel offload is permissible because the core is being placed in a more conservative condition (i.e. defueled).

As seen in the excerpted page from lesson plan 3-OT-SYS092A, either SRNI N31 or SRNI N32 may feed the source range level to the scalar/timer. In the question stem, N32 (the remaining OPERABLE SRNI) is selected. The scalar/timer outputs a count rate which is fed to two amplifiers. These amplifier channels are not named N31 and N32 but are named A1 and A2. Amplifier A1 normally only feeds the speaker in the main control room. Amplifier A2 only feeds the speaker inside of containment. In the event of the failure listed in the question, a selector switch (located on the rear of the drawer) may be manipulated to select the main control room amplifier to provide the containment speaker. The stem of the question, therefore, presents not only a failure of SRNI N31, but also a failure of amplifier A2. This condition would be rectified by the second bullet of step 10 of 1-AOI-4 which states: IF audible audio count rate is **NOT** being received in CNTMT, THEN: a. TURN AMPLIFIER SELECT switch to A1 [1-M-13 switch on rear of amplifier]. The switch circled in red on the picture in the stem of the question <u>only controls the channel providing the signal to the scalar/timer</u>; a common misconception is that it selects an amplifier channel (e.g. amplifier channel N31 or N32). One of the most common misconceptions is that this switch affects the amplifier channel for the MCR and that the switch on the rear of the drawer affects the amplifier selection for the containment speaker.

B. Incorrect: Again, it is incorrect and yet plausible that the core offload may continue. Also, it is incorrect and yet plausible that the switch on the front of the drawer would control the signal to the speaker inside of containment.

C. Correct: As discussed, core offload must stop. Only the manipulation of the AMPLIFIER SELECT switch on the back of the drawer will restore the count rate to containment.

D. Incorrect: It is Incorrect and yet plausible to believe that fuel offload may continue. Also, it is incorrect and yet plausible that the switch on the front of the drawer would control the signal to the speaker inside of containment.

Question Number: 21

Tier: 1 Group: 2

K/A: 032 Loss of Source Range Nuclear Instrumentation
2.4 Emergency Procedures / Plan
G2.4.11 Knowledge of abnormal condition procedures.

Importance Rating: 4.0 4.2

10 CFR Part 55: (CFR: 41.10 / 43.5 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to present knowledge of 1-AOI-4 which contains the abnormal condition procedure for the Loss of Source Range Nuclear Instrumentation.

Technical Reference: 1-AOI-4, Nuclear Instrumentation Malfunctions
Westinghouse prints for the SRNI's scalar timer and downstream amplifiers
Selected Page from 3-OT-STG-092A, EXCORE INSTRUMENTATION

Proposed references to be provided: None

Learning Objective: 3-OT-SYS092A, Excore Nuclear Instrumentation
1. DESCRIBE the design criteria, purpose, and/or functions of the Excore
Nuclear Instrumentation System (NIS) and subsystems, and the major
system components listed below: (IER 11-3: Having a solid
understanding of plant design, engineering principles, and sciences)
h. Audio Count Rate Drawer
14. DESCRIBE the following aspects of Technical Specifications and
Technical Requirements for this system:
a. The conditions and required actions with completion time of one hour
or less
b. The Limiting Conditions for Operation, Applicability, and Bases.

Cognitive Level:
- Higher X
- Lower

Question Source:
- New X
- Modified Bank
- Bank

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Step	Action/Expected Response	Response Not Obtained

3.2 Source Range Monitor (SRM) Failure (continued)

9. **IF** in MODE 3, 4, or 5,
THEN:

- **CHECK** audio count rate audible in control room.
  - RNO: **PLACE** audio count rate CHANNEL SELECTOR switch to operable channel [1-M-13].
- **REFER TO** the following Tech Specs:
  - 3.3.1,Reactor Trip System Instrumentation,
  - 3.3.3, PAM Instrumentation,
  - 3.3.4, Remote Shutdown System.

The reason that operators think that the switch on the front of the drawer controls the speaker in the MCR is that this step comes first and has the RNO to manipulate the CHANNEL SELECTOR switch. They fail to realize that this step just states: if the channel that failed is selected as the input to the scalar/timer then you won't have audible count rate in the MCR and must select the operable SRNI.

10. **IF** in MODE 6,
THEN:

- **CHECK** audio count rate audible in control room.
  - RNO: **IF** audible audio count rate is **NOT** being received in the control room, **THEN** **PLACE** audio count rate CHANNEL SELECTOR switch to an operable channel [1-M-13].
- **CHECK** audio count rate audible in Cntmt.
  - RNO: **IF** audible audio count rate is **NOT** being received in Cntmt, **THEN**:
    a. **TURN** AMPLIFIER SELECT switch to A1 [1-M-13 switch on rear of amplifier].
    b. **CHECK** an audible count rate in Cntmt.

This step just recognizes that the containment speaker has a backup amplifier (whereas the MCR speaker does not).

- **REFER TO** Tech Spec 3.9.3, Nuclear Instrumentation.

Table 5 Source Range Audio Count Rate Selector Switch

CONTROL SWITCH	POSITION	OPERATION
 Source Range Audio Count Rate Selector Switch	SR N31	Audio count rate meter monitors source range channel N31
	SR N32	Audio count rate meter monitors source range channel N32

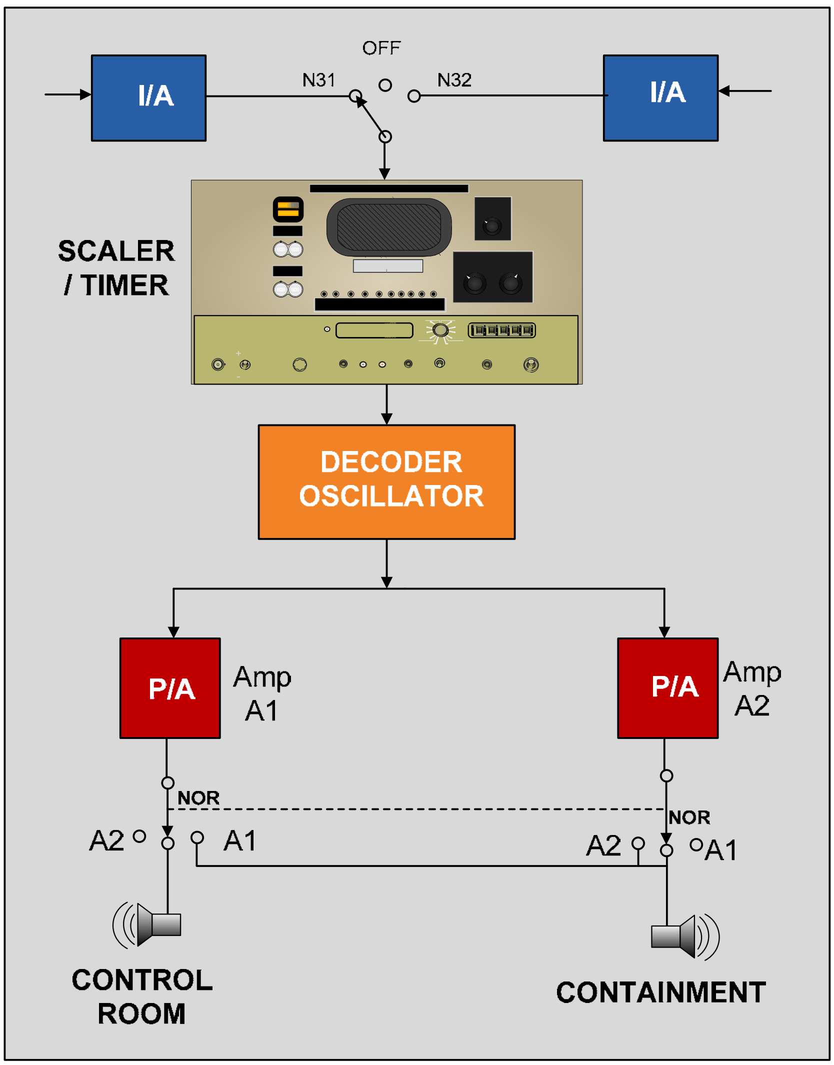


Figure 18 Audio Count Rate

25402-011-V1A-MG00-01508-001-WBT-D-1103

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REVISION STATUS OF SHEETS		
2	1	SHEET NO.
C	D	LATEST REV SYM

REVISIONS				
ZONE	LTR	DESCRIPTION	DATE	APPROVED
	A	DRAWING RELEASED	8-29-68	
	B	RN-K4508 - SHTS 1 & 2 REVISED PER MARKED PRINT. L.F.D.	2-20-69	
	C	RN-K8602 G.O.10330 - SHEET 1, ZONE C4-P6 WAS P514. ZONE A6-REVISED NOTE 2 "NON-REPAIRABLE UNITS" WAS NON-REPAIRABLE UNIT. - SHEET 2-REVISED SWITCHES PER MARKED PRINT.		
D3 C5	D	RN-K 8423 DS501 & DS502 WAS	12/15/69	

BECHTEL POWER CORPORATION	Job Number: 25402
SUPPLIER DOCUMENT REVIEW STATUS	

STATUS CODE:

- 1 ☒ Work may proceed.
- 1C ☐ Work may proceed. Editorial comments need only be incorporated if revised for other purposes.
- 2 ☐ Revise and resubmit. Work may proceed subject to incorporation of changes indicated.
- 3 ☐ Rejected. Revise and resubmit.
- 4 ☐ Review not required. Work may proceed.

Contract No. 65717
WBT-D- 1103

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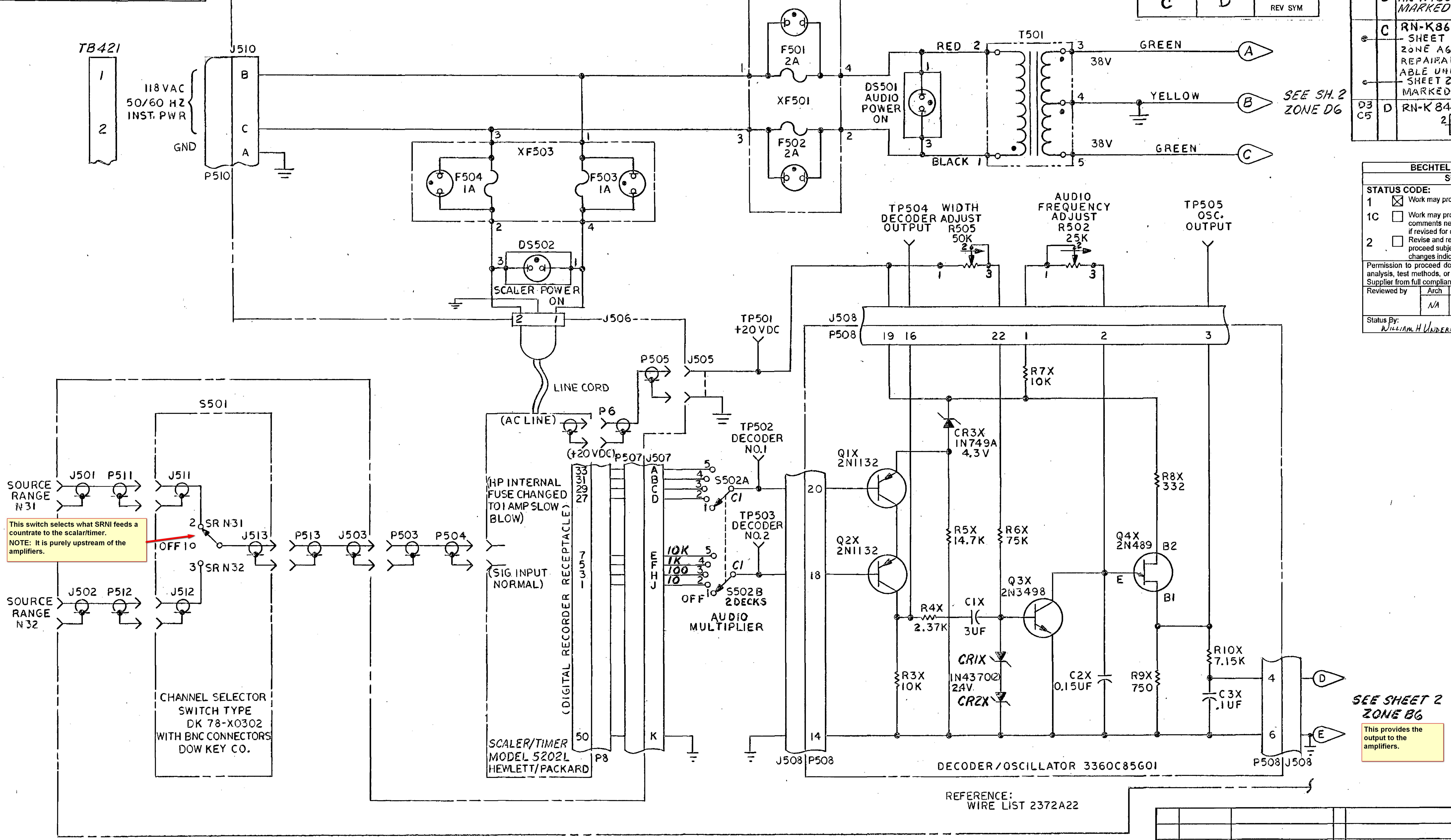
Reviewed by	Arch	Civil	CS	Elect	Mech	MET	PD	Constr	Startup	STE
	NA	NA	WU	NA	NA	NA	NA	NA	NA	NA

Status By: William H Underwood — DATE 12/12/2009



REFERENCE:
WIRE LIST 2372A22

NOTES:

1. UNLESS OTHERWISE SPECIFIED:
   - ALL RESISTORS ARE 1/2 WATT ±1%
   - ALL ZENERS ARE .4W ±10%
   - ALL POTS ARE 10 TURNS, 1.5W, ±5%
   - ARROWS ON POTS INDICATE CW ROTATION
   - ALL CAPACITORS ARE IN MICROFARAD, 200V ±10%
   - ALL RESISTORS ARE IN OHMS, K-INDICATES THOUSANDS, M-MILLIONS
   - ALL FUSES ARE SLO-BLOW
   - DOTS ON TRANSFORMER WINDINGS INDICATE POLARITY
2. A1 & A2 AMPLIFIERS NON REPAIRABLE UNITS, NOTE 1 DOES NOT APPLY

QTY REQD	CODE IDENT NUMBER	PART NO.	*	DESCRIPTION	GOVT OR COML SPEC	Ⓦ SPEC	ITEM NO.
							3
							2
							1

* V=VENDOR ITEM. SEE SOURCE/SPEC CONT DWG — LIST OF MATERIALS

6051D52	G.O. 10310
NEXT ASSY	USED ON
APPLICATION	

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES DO NOT SCALE. TOLERANCES ON DEC.
2 PLACE ±— | 3 PLACE ±— | ANGLES ±—
ALSO SEE SPEC 9QS01
PROCESS SPEC
FINISH SPEC — GOVT — Ⓦ

CONTRACT NUMBER
ORIGINAL DATE OF DWG 29 AUG 68
DRAFTSMAN T.E. VANCE — APPD
CHECKED — APPD
APPD — APPD
DESIGN ACTIVITY APPROVAL
PROCURING ACTIVITY APPROVAL

Westinghouse Electric Corporation
DEFENSE AND SPACE CENTER Ⓦ BALTIMORE, MD., U.S.A.

TITLE
NIS SCHEMATIC DIAGRAM AUDIO COUNT RATE NUCLEAR INSTRUMENTATION SYSTEM

SIZE	CODE IDENT NO.	DWG NO.
D	04804	6052D36

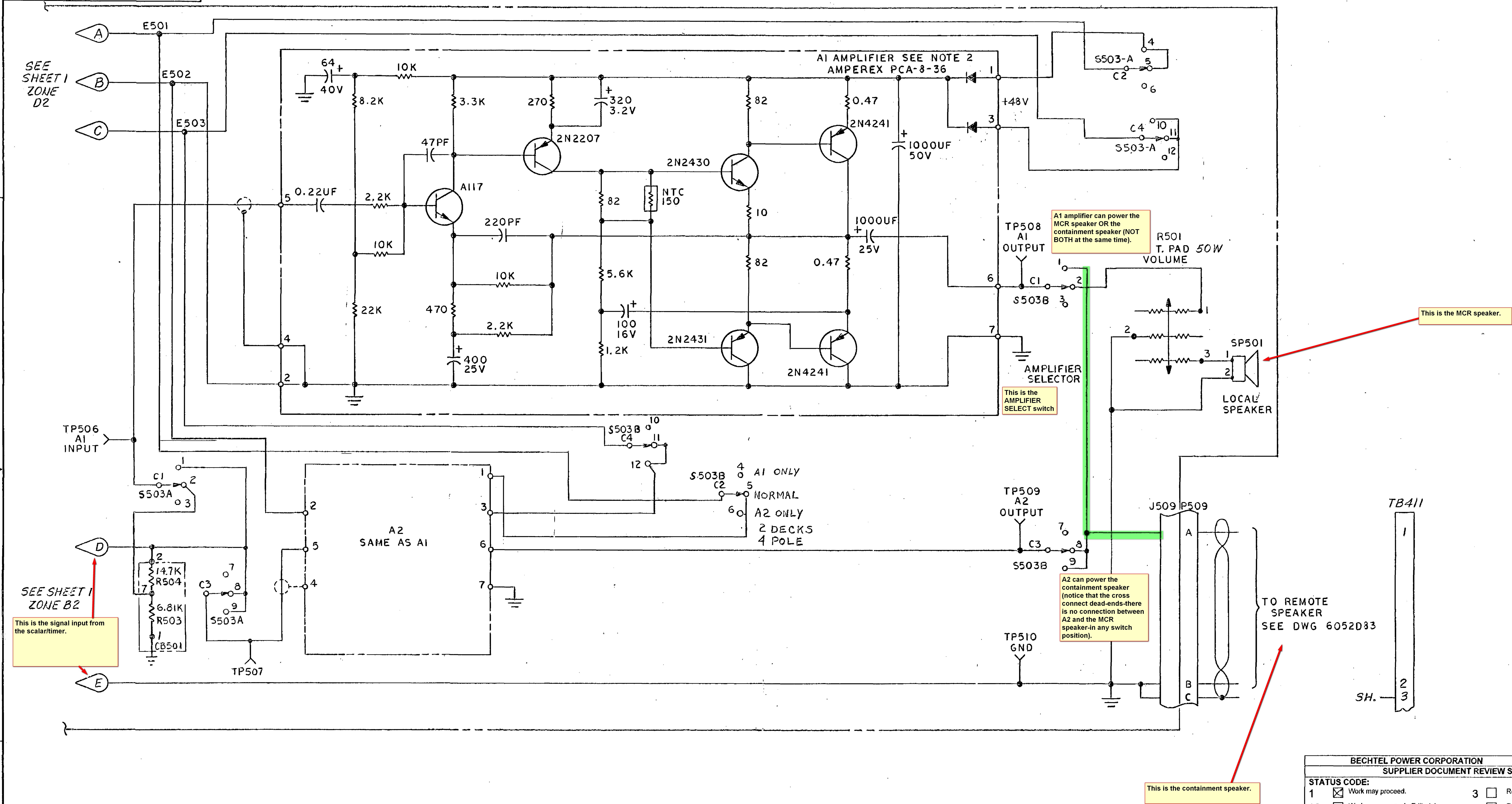
SCALE —/— | WEIGHT | SHEET 1 OF 2

FORM NO. DA 5383-1

AAIA

REVISIONS

ZONE	LTR	DESCRIPTION	DATE	APPROVED
	C	REV. INCORPORATED. SEE SHEET 1		



BECHTEL POWER CORPORATION					Job Number: 25402					

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3 ☐ Rejected. Revise and resubmit.

4 ☐ Review not required. Work may proceed.

Contract No. 65717
WBT-D- 1103

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Reviewed by	Arch	Civil	CS	Elect	Mech	MET	PD	Constr	Startup	STE
	NA	NA	WU	NA	NA	NA	NA	NA	NA	NA

Status By: William H Underwood William H Underwood — DATE 12/12/2009

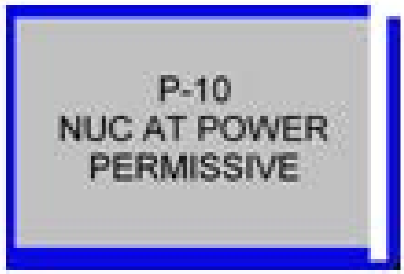
SIZE	CODE IDENT NO.	DWG NO.
D	04804	6052D36
SCALE —/—	WEIGHT	SHEET 2

FORM NO. DA 5363-1

22.

Given the following conditions:

- Unit 1 is in MODE 2.
- The following alarm is **DARK**:



Subsequently:

- 120V AC Vital Instrument Board 1-II de-energizes.

Which ONE of the following describes the Unit 1 response to the failure?

IMMEDIATELY after the failure, the Unit 1 Reactor will ____(1)____.

____(2)____ of the IRNIs remain(s) energized.

A. (1) TRIP
(2) **BOTH**

B. (1) TRIP
(2) **ONLY** ONE

C. (1) REMAIN CRITICAL
(2) **BOTH**

D. (1) REMAIN CRITICAL
(2) **ONLY** ONE

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: The reactor would trip upon the loss of an IRNI. As mentioned below it is plausible to believe that both IRNIs would be energized. It is plausible to believe that the Reactor would trip upon the loss of a 120V AC Vital Instrument Board for a reason other than the loss of an IRNI. Therefore, the combination of the two parts of this distractor remains plausible.

B. Correct: NI-36 would fail causing a reactor trip. It is correct that only IRNI-135 would be energized.

C. Incorrect: The reactor would trip due to IRNI 136 failure and only one intermediate range monitor would be energized. It is plausible to believe that both IRNIs be energized as if a different 120V AC vital board had been lost (e.g. III or IV) both IRNIs would have been energized. As mentioned below, it is plausible to believe that the reactor would have remained critical.

D. Incorrect: 1-AOI-25.02, Loss of 120V AC Vital Instrument Power Boards 1-II or 2-II stipulates that Intermediate Range 1-N-136 will fail upon of loss of power board 1-II. WBNs IRNIs are manufactured by Gamma-Metrics. These NIs utilize control power and instrument power. Both control and instrument power originate from a 120V AC vital power board. Internal to the NI are bistables which trip upon detecting a high flux condition. These bistables are normally energized from instrument power which maintains a contact closed. The secondary of the bistable (the normally closed contact) allows control power to be supplied to the SSPS input relays. The SSPS input relays applicable to NI trips are normally energized. Therefore, upon the loss of the vital instrument board two mechanisms exist to trip the reactor: 1. the NI drawer bistables shift to a tripped state and 2. the SSPS input relays de-energize (achieve their tripped state). As seen in 1-ARI-79-C, Rx Trip First Out, when below permissive P-10, either NC-35F (a bistable internal to IRNI-135) or NC-36F (internal to IRNI-136) will cause an Intermediate Range Flux Trip. It is plausible to believe that the reactor could have remained critical because if the failure had been a different Instrument Power Board then this assumption could have been true (e.g. the loss of 120V AC vital board III or IV). Additionally, it is plausible to believe that the reactor would have remained critical because had the reactor been at a power level above P-10 (10%), the intermediate range trip would have been blocked and the reactor would have remained critical.

Question Number: 22

Tier: 1 Group: 2

K/A: 033 Loss of Intermediate Range Nuclear Instrumentation
AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Intermediate Range Nuclear Instrumentation:
AK1.01 Effects of voltage changes on performance

Importance Rating: 2.7 3.0

10 CFR Part 55: (CFR 41.8 / 41.10 / 45.3)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to recognize the impact which a loss of voltage to a IRNI has upon the operation of the plant when the plant is operating low in the power range (below P-10).

Technical Reference: 1-ARI-76-80, Rx Trip First Out
1-ARI-64-70, Bypass Inlk & Permissive
1-AOI-25.02, Loss of 120V AC Vital Instrument Power Boards 1-II or 2-II
1-AOI-4, Nuclear Instrumentation Malfunctions

Proposed references to be provided: None

Learning Objective: 3-OT-SYS092A, Excore Nuclear Instrumentation
13. Given specific plant conditions, ANALYZE the effect that a loss or
malfunction of the following will have on the NIS: (IER 11-3, Operating
the plant with a conservative bias)
b. Power supplies

Cognitive Level:
Higher X
Lower ____

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

2.1 Alarms (continued)

T. SSPS-B GEN WARNING [115-A]

U. CL ACCUM 1 LEVEL HI/LO [131-A]

V. CL ACCUM 1 PRESS HI/LO [131-B]

W. CL ACCUM 2 LEVEL HI/LO [132-A]

X. CL ACCUM 2 PRESS HI/LO [132-B]

A loss of 120 V AC Vital Instrument Board causes the loss of 1-N-136

2.2 Indications

A. Power Range 1-N-42, Intermediate Range 1-N-136, and Source Range 1-N-132 failure.

B. All channel II trip STATUS LIGHTS will ENERGIZE (Except Cntmt Level, Cntmt Hi-Hi Press And RWST Low Level).

C. Train B CNTMT Isolation Panel 1-XX-55-6F will be DARK.

D. Accumulator 3 & 4 Pressure, fails low

2.3 Automatic Actions

A. Rod withdrawal block due to high flux rod stop.

B. Charging pump suction swaps over to RWST due to de-energized separation relays. 1-LCV-62-135 and 1-LCV-62-136 open and 1-LCV-62-133 closes, which causes borated water to be injected into the RCS.

C. If in service, Train A MCR, EBR, and SDBR Chillers will swap to Train B, and auto start of Train A is disabled.

64-E

P-10
NUC AT POWER
PERMISSIVE

(Page 1 of 1)

Source

Relay K1705 (See Note 1)

Setpoint

NOTE 1
Relay K1705 is actuated when Reactor power on 2/4 PR channels is greater than or equal to10%.

Probable Cause: A. Power escalation above 10%

NOTE 2
P-10 blocks manual reset of both SR channels high voltage supply, blocks SR high flux trip, and supplies input to P-7.

CAUTION
The P-10 permissive will automatically reset if 3/4 PR channels decrease below 10%. The reset of P-10 will unblock the IRM rod stop and high flux trip and the PRM high flux low setpoint reactor trip.

Corrective Action:

[1] **VERIFY** Reactor power greater than or equal to 10%.

[2] **IF** IR high flux trip and rod stop should be blocked, **THEN**
PLACE IR TRIP BLOCK P-10 handswitchs (1-N38A ILLUM and 1-N38B) in BLOCK and
VERIFY Window 65-C illuminates.

[3] **IF** PR high flux low setpoint should be blocked, **THEN**
PLACE PR LO POWER TRIP BLOCK P-10 handswitchs (1-N47A and 1-N47B) in BLOCK and
VERIFY Window 64-D illuminates.

References:

1-45W600-55-9
1-47W611-99-2
W 7246D11 Sht. 40

79-C

Source	**Setpoint**	
NC-35F or NC-36F	IR channel current equal to 25% of full power	**INTERMED RANGE FLUX HI**

The coincidence for the trip is 1 out of 2.

(Page 1 of 1)

NOTE

This trip may be manually blocked above P-10 and is automatically reinstated when power drops below P-10.

Probable Cause:

A. Flux level above setpoint (failure to block when above P-10)
B. Low power reactivity excursion

Corrective Action:

[1] **GO TO** 1-E-0, REACTOR TRIP OR SAFETY INJECTION.

References:

1-47W611-99-2
W 7245D11-26
1-E-0

2.2 Indications (continued)

B. Intermediate Range Monitor (IRM) malfunction:

1. Erratic or loss of indication.
2. INSTRUMENT POWER ON and/or CONTROL POWER ON lights at NIS racks DARK [1-M-13].
3. Both startup rate channels **NOT** indicating same startup rate.
4. SRM does **NOT** energize during shutdown.
5. NON-OPERATE light LIT [1-M-13].

C. Power Range Monitor (PRM) malfunction:

1. Erratic or loss of indication.
2. Delta flux meter failed high, low, or giving erratic indications.
3. CONTROL POWER ON and/or INSTRUMENT POWER ON lights at NIS racks DARK [1-M-13].
4. DCS Operator display for Rod Control and PR NIS indicates abnormal condition for applicable PRM.

2.3 Automatic Actions

A. Source Range Monitor failure:

1. Possible source range monitor high flux Rx trip at 10^5 cps if SRM **NOT** blocked.
2. Possible Cntmt evacuation alarm at 0.5 decade above background.

Loss of ONE IRNI = trip below P-10.

B. Intermediate Range Monitor failure:

1. Intermediate range monitor 20% high flux rod stop (if **NOT** blocked above P-10).
2. Intermediate range monitor 25% high flux Rx trip (if **NOT** blocked above P-10).
3. Possible loss of P-6 block if intermediate range monitor greater than 1.66×10^{-4}% power and intermediate range monitor fails low.

23.

Which ONE of the following describes the control room indications AND the AUTOMATIC actions (if any) **IF** 0-RFV-77-743J, GAS DECAY TANK J RELIEF **FAILS OPEN**?

____(1)____ will provide indication of this failure to the main control room staff AND ____(2)____isolate the failure once it reached its HIGH alarm setpoint in accordance with 1-AOI-31, Abnormal Release of Radioactive Material.

NOTE: *0-RM-90-118A, WGDT RELEASE LINE*
1-RI-90-400, SHIELD BLDG VENT MON

A. (1) 1-RI-90-400
(2) WILL

B. (1) 1-RI-90-400
(2) WILL **NOT**

C. (1) 0-RM-90-118A
(2) WILL

D. (1) 0-RM-90-118A
(2) WILL **NOT**

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: It is correct that 1-RI-90-400 would provide the operators in the main control room with indication of the listed failure. However, it is not correct that an automatic isolation of the failure would occur. As seen on the control print 1-47W610-90-5, the 400 radiation monitor performs NO automatic isolations. It is plausible to believe that it does as there are radiation monitors in the plant which do so.

B. Correct: 1-RI-90-400 provides the indication to the control room staff and would not isolate the listed failure.

C. Incorrect: 1-AOI-31, Abnormal Release of Radioactive Material includes the guidance for an accidental gaseous radwaste release. This AOI contains a listing of symptoms which can be seen to basically list the various radiation monitors within the plant. The operator then utilizes his knowledge of the sample points of the radiation monitors to ascertain whether the radiation problem is associated with the Auxiliary, Reactor, Turbine or Service buildings. The procedure also contains the expected automatic isolations of the radiation monitoring system. It is Incorrect that the radiation monitor for the Waste Gas release header, 0-RM-90-118 would detect a failure of a decay tank relief valve because as seen on print 1-47W830-4, the relief valve header for the tanks ties into the discharge to the plant vent DOWNSTREAM of RE-90-118. Print 1-47W866-1 shows that the discharge to the plant vent is the shield bldg exhaust vent (shield bldg vent stack); this print directs the observer to next look at print 1-47W610-90-5 which shows that 1-XIC-90-400 processes the sample effluent from the shield bldg vent stack and displays the radiation levels on 1-RI-90-400. Therefore, 1-RI-90-400 will alert the main control room staff to the failed waste decay tank relief valve and **NOT** 0-RM-90-118. As seen in ARI-184-A, WGDT REL LINE 0-RM-90-118 RAD HI, 0-FCV-77-119 will automatically close on a high radiation signal isolating the gas decay tank release. However, in this case, 0-RM-90-118 never produces a high radiation signal and as such 0-FCV-77-119 never receives an automatic isolation signal. It is plausible to believe that this distractor is true because 0-RM-90-118 is the radiation monitor which normally inspects the WGDT release path and it would isolate the release path (for a normal tank release) upon a high radiation condition.

D. Incorrect: It is not correct that 0-RM-90-118 would detect the listed failure. Also it is correct that the release would not be isolated. It is plausible to believe that 0-RM-90-118 would not perform an automatic isolation as the majority of the radiation monitors at the plant perform no automatic isolation.

Question Number: 23

Tier: 1 Group: 2

K/A: 060 Accidental Gaseous Radwaste Release
2.4 Emergency Procedures / Plan
2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Importance Rating: 4.5 4.7

10 CFR Part 55: (CFR: 41.10 / 43.2 / 45.6)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to recognize which radiation monitor in the main control room would provide the abnormal indication for a system parameter which would provide for the diagnostic entry level condition for 1-AOI-31.

Technical Reference: 1-AOI-31, Abnormal Release of Radioactive Material
ARI-180-187, Common Radiation Detectors
Picture of 1-RI-90-400
Picture of 0-RM-90-118
Print 1-47W610-90-5
Print 1-47W866-1
Print 1-47W830-4

Proposed references to be provided: None

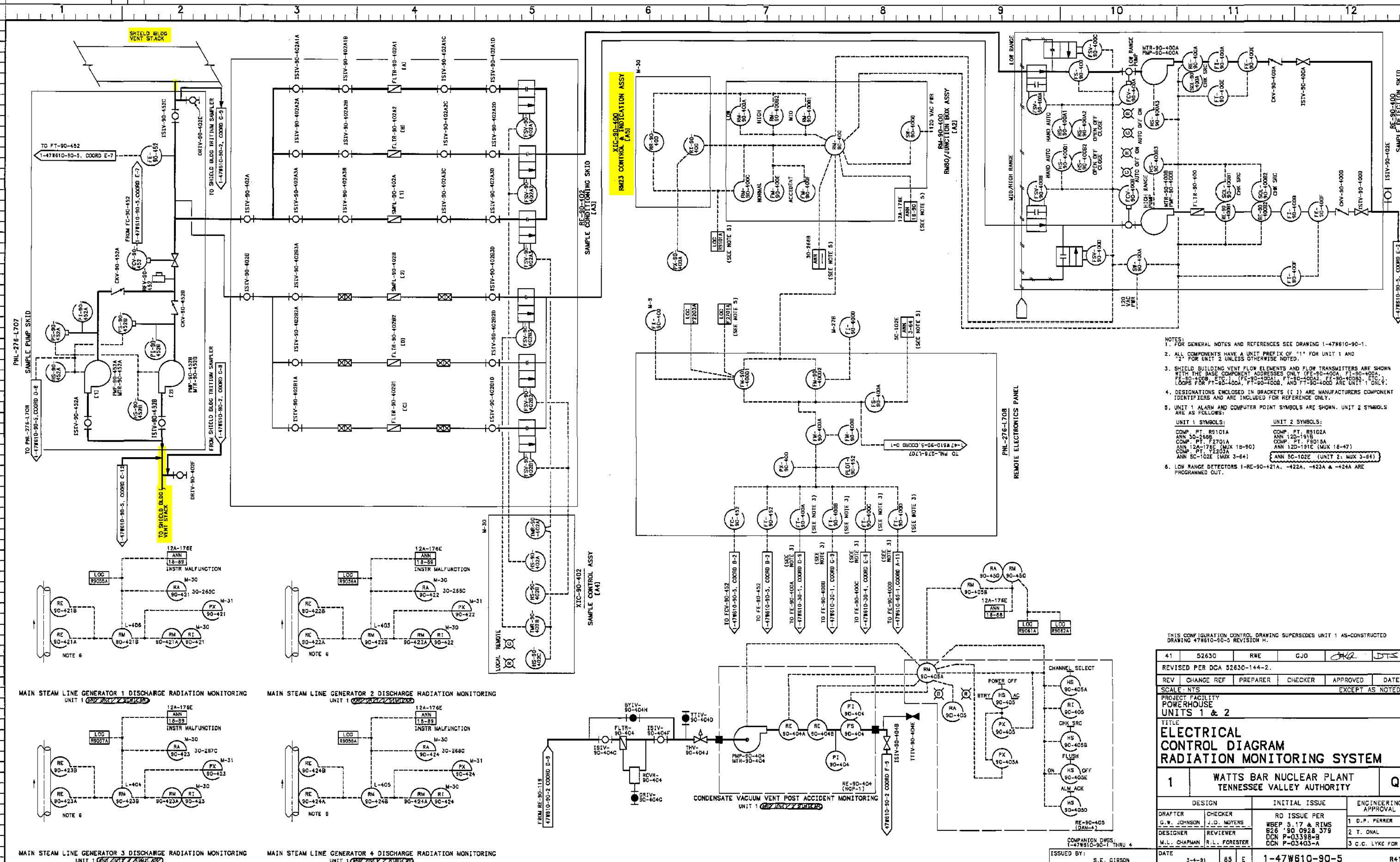
Learning Objective: 3-OT-SYS090A, Radiation Monitoring
3. EXPLAIN the physical connections and/or cause-effect relationships between the Radiation Monitoring System and the following systems:
j. Shield Bldg Vent
k. Aux Bldg Vent

Cognitive Level:
- Higher X
- Lower ____

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:



NOTES:
1. FOR GENERAL NOTES AND REFERENCES SEE DRAWING 1-47W610-90-1.
2. ALL COMPONENTS HAVE A UNIT PREFIX OF "1" FOR UNIT 1 AND "2" FOR UNIT 2 UNLESS OTHERWISE NOTED.
3. SHIELD BUILDING VENT FLOW ELEMENTS AND FLOW TRANSMITTERS ARE SHOWN WITH THE BASE COMPONENT ADDRESSES ONLY (FE-90-400A, FI-90-400A, FE-90-400B, ETC.), (FE-90-400A1, FT-90-400A2, FE-90-400B3, ETC.). LOOPS FOR FT-90-400A, FT-90-400B, AND FT-90-400D ARE UNIT 1 ONLY.
4. DESIGNATIONS ENCLOSED IN BRACKETS ([ ]) ARE MANUFACTURERS COMPONENT IDENTIFIERS AND ARE INCLUDED FOR REFERENCE ONLY.
5. UNIT 1 ALARM AND COMPUTER POINT SYMBOLS ARE SHOWN. UNIT 2 SYMBOLS ARE AS FOLLOWS:

UNIT 1 SYMBOLS:	UNIT 2 SYMBOLS:
COMP. PT. R9101A	COMP. PT. R9102A
ANN 30-268B	ANN 12D-191B
COMP. PT. F2701A	COMP. PT. F9015A
ANN 12A-176E (MUX 18-90)	ANN 12D-191E (MUX 18-47)
COMP. PT. Y2203A	
ANN 5C-102E (MUX 3-64)	ANN 5C-102E (UNIT 2: MUX 3-64)

6. LOW RANGE DETECTORS 1-RE-90-421A, -422A, -423A & -424A ARE PROGRAMMED OUT.

THIS CONFIGURATION CONTROL DRAWING SUPERSEDES UNIT 1 AS-CONSTRUCTED DRAWING 47W610-90-5 REVISION H.

REV	CHANGE REF	PREPARER	CHECKER	APPROVED	DATE
41	52630	RWE	GJO		
REVISED PER DCA 52630-144-2.					

SCALE: NTS EXCEPT AS NOTED

PROJECT FACILITY
POWERHOUSE
UNITS 1 & 2

TITLE
ELECTRICAL CONTROL DIAGRAM RADIATION MONITORING SYSTEM

1 — WATTS BAR NUCLEAR PLANT — TENNESSEE VALLEY AUTHORITY — Q

DESIGN		INITIAL ISSUE	ENGINEERING APPROVAL
DRAFTER: G.W. JOHNSON	CHECKER: J.D. MOYERS	RO ISSUE PER WBEP 5.17 & RIMS B26 '90 0928 379 DCN P-03398-B DCN P-03403-A	1 D.P. PERRER
DESIGNER: M.L. CHAPMAN	REVIEWER: R.L. FORESTER		2 T. ONAL
			3 C.C. LYKE FOR MCB

COMPANION DWGS: 1-47W610-90-1 THRU 4
ISSUED BY: S.E. GIBSON
DATE 3-4-91 | 85 | E | 1-47W610-90-5 | R41

CAD MAINTAINED DRAWING

CONFIGURATION CONTROL DRAWING

1.0 PURPOSE

This instruction provides a response to valid annunciations or indications of an abnormal release of radioactive material.

2.0 SYMPTOMS

2.1 Alarms

NOTE

SHLD BLDG VENT RE-400 RAD HI [268-B] in alarm should be accompanied by higher indicated radiation levels or alarm on additional monitor(s) which indicates the source of the release.

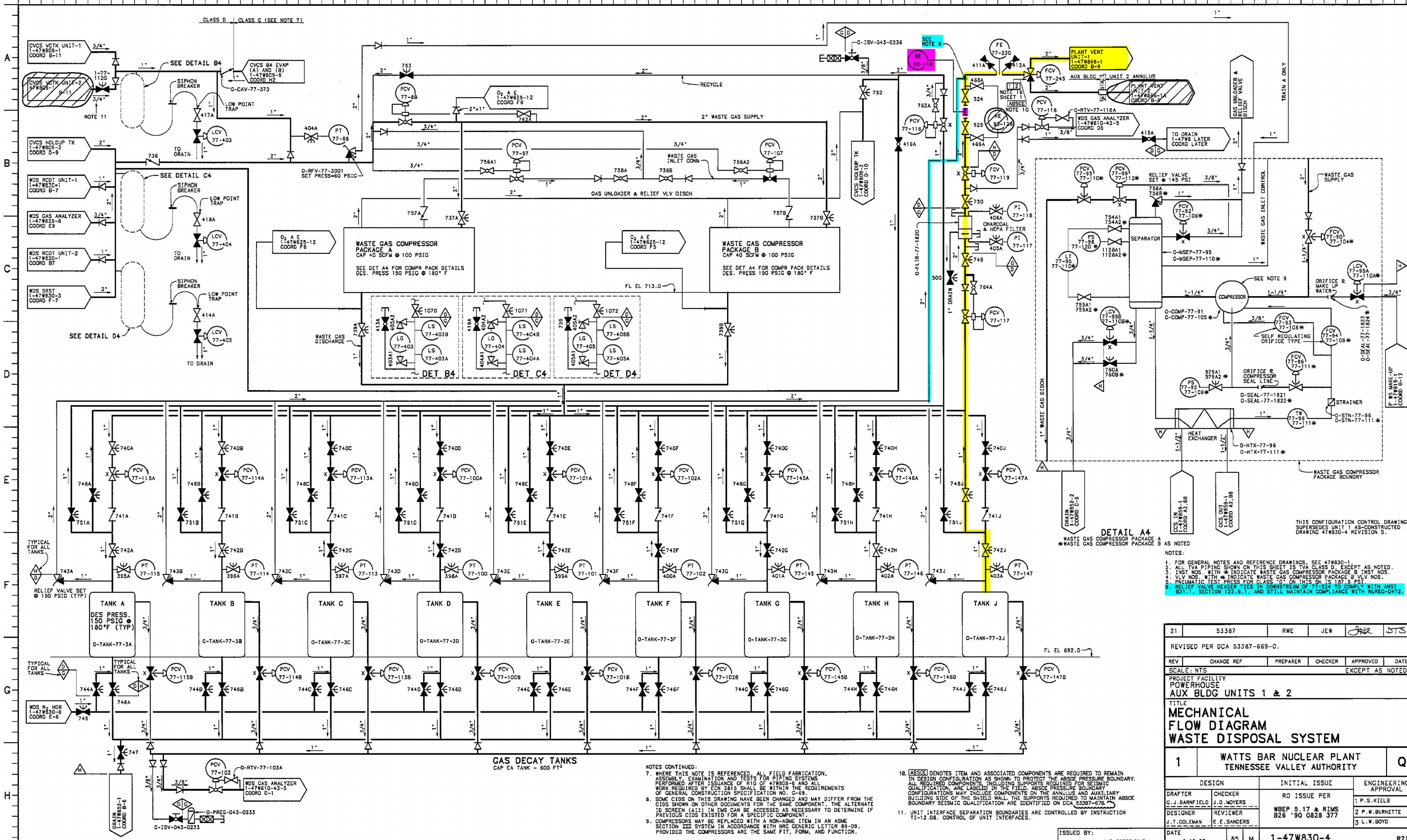
A. Auxiliary Building:

1. 1-RR-90-1 AREA RAH HI [174-B]
2. ERCW DISCH HDR A 0-RM-133/140 LIQ RAD HI [180-B]
3. WDS RELEASE LINE 0-RM-122 LIQ RAD HI [181-A]
4. ERCW DISCH HDR B 0-RM-134/141 LIQ RAD HI [181-B]
5. AB VENT 0-RM-101 RAD HI [183-C]
6. WGDT REL LINE 0-RM-118 RAD HI [184-A]
7. SFP 0-RM-102/103 RAD HI [184-B]
8. 0-RR-90-12 PARTICULATE RAD HI [185-B]

To implement this procedure, the operators must diagnosis which building is affected (and thus section is required) by interpreting the readings of the various radiation monitors.

B. Reactor Building:

1. UPR CNTMT AIR 1-RM-112 RAD HI [173-A]
2. LWR CNTMT AIR 1-RM-106 RAD HI [173-B]
3. CNTMT PURGE EXH 1-RM-130/131 RAD HI [174-A]



THIS CONFIGURATION CONTROL DRAWING SUPERSEDES UNIT 1 AS-CONSTRUCTED DRAWING 47W830-4 REVISION S.

NOTES:

1. FOR GENERAL NOTES AND REFERENCE DRAWINGS, SEE 47W830-1.
2. ALL TVA PIPING SHOWN ON THIS SHEET IS TVA CLASS D, EXCEPT AS NOTED.
3. INST NOS. WITH * INDICATE WASTE GAS COMPRESSOR PACKAGE B INST NOS.
4. VLV NOS. WITH * INDICATE WASTE GAS COMPRESSOR PACKAGE B VLV NOS.
5. PNEUMATIC TEST PRESS FOR CLASS "D" ON THIS SH IS 187.5 PSI.
6. RELIEF VALVE HEADER TIES IN DOWNSTREAM OF 77-524 TO COMPLY WITH ANSI B31.1, SECTION 122.6.1, AND STILL MAINTAIN COMPLIANCE WITH NUREG-0472.

NOTES CONTINUED:

7. WHERE THIS NOTE IS REFERENCED, ALL FIELD FABRICATION, ASSEMBLY, EXAMINATION AND TESTS FOR PIPING SYSTEMS PERFORMED AFTER ISSUANCE OF R10 OF 47W809-6 AND ALL WORK REQUIRED BY ECN 3815 SHALL BE WITHIN THE REQUIREMENTS OF GENERAL CONSTRUCTION SPECIFICATION NO. G-69.
8. SOME CIDS ON THIS DRAWING HAVE BEEN CHANGED AND MAY DIFFER FROM THE CIDS SHOWN ON OTHER DOCUMENTS FOR THE SAME COMPONENT. THE ALTERNATE ID SCREEN (AII) IN EMS CAN BE ACCESSED AS NECESSARY TO DETERMINE IF PREVIOUS CIDS EXISTED FOR A SPECIFIC COMPONENT.
9. COMPRESSORS MAY BE REPLACED WITH A NON-ASME ITEM IN AN ASME SECTION III SYSTEM IN ACCORDANCE WITH NRC GENERIC LETTER 89-09, PROVIDED THE COMPRESSORS ARE THE SAME FIT, FORM, AND FUNCTION.
10. $\boxed{\text{ABSCE}}$ DENOTES ITEM AND ASSOCIATED COMPONENTS ARE REQUIRED TO REMAIN IN DESIGN CONFIGURATION AS SHOWN TO PROTECT THE ABSCE PRESSURE BOUNDARY. ALL REQUIRED COMPONENTS, INCLUDING SUPPORTS REQUIRED FOR SEISMIC QUALIFICATION, ARE LABELED IN THE FIELD. ABSCE PRESSURE BOUNDARY CONFIGURATIONS MAY INCLUDE COMPONENTS ON THE ANNULUS AND AUXILIARY BUILDING SIDE OF THE SHIELD WALL. THE SUPPORTS REQUIRED TO MAINTAIN ABSCE BOUNDARY SEISMIC QUALIFICATION ARE IDENTIFIED ON DCA 53387-670.
11. UNIT INTERFACE SEPARATION BOUNDARIES ARE CONTROLLED BY INSTRUCTION TI-12.08, CONTROL OF UNIT INTERFACES.

ISSUED BY: J.E.GIBBS/WLE

PRINTS REQ'D R

REV	CHANGE REF	PREPARER	CHECKER	APPROVED	DATE
21	53387	RWE	JEW	JKA	DTS
	REVISED PER DCA 53387-669-0.				

SCALE: NTS EXCEPT AS NOTED

PROJECT FACILITY
POWERHOUSE
AUX BLDG UNITS 1 & 2

TITLE
MECHANICAL FLOW DIAGRAM WASTE DISPOSAL SYSTEM

1 | WATTS BAR NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY | Q

DESIGN		INITIAL ISSUE	ENGINEERING APPROVAL
DRAFTER: G.J.BARNFIELD	CHECKER: J.D.MOYERS	RO ISSUE PER WBEP 5.17 & RIMS B26 '90 0828 377	1 P.S.KIELB
DESIGNER: J.T.COLEMAN	REVIEWER: E.E.SANDERS		2 P.W.BURNETTE
			3 L.W.BOYD
DATE: 9-19-90	85 M	1-47W830-4	R21

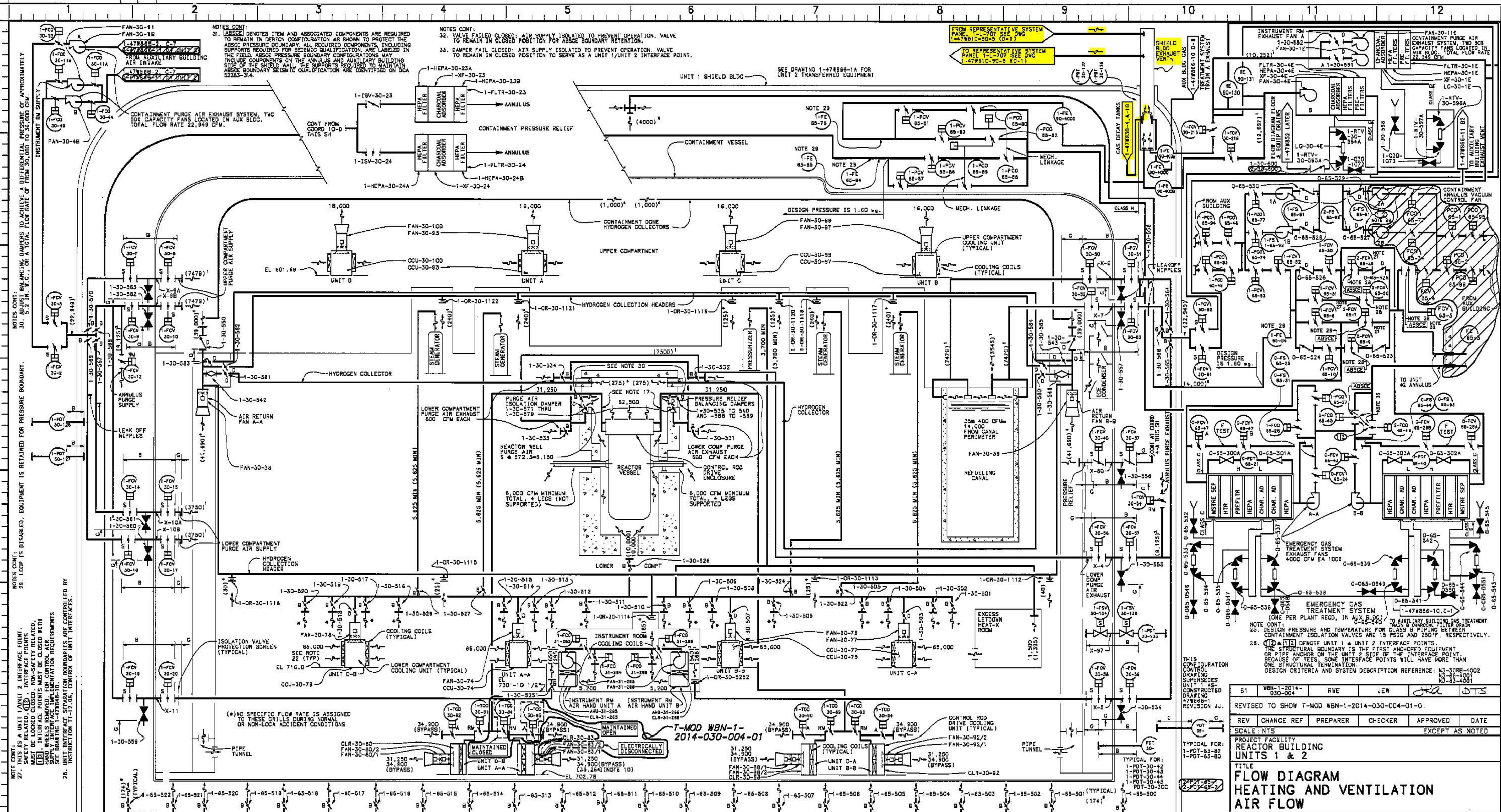
PRINT SIZE F — BR OR PROJ AB CE EE ME NE FE HE SE TE BL BF WB SQ

PRINT SIZE H — BR OR PROJ AB CE EE ME NE FE HE SE TE BL BF WB SQ

CAD MAINTAINED DRAWING

CONFIGURATION CONTROL DRAWING

1-47W866-1 | M | 85



NOTES CONT:
31. ABSCE DENOTES ITEM AND ASSOCIATED COMPONENTS ARE REQUIRED TO REMAIN IN DESIGN CONFIGURATION AS SHOWN TO PROTECT THE ABSCE PRESSURE BOUNDARY. ALL REQUIRED COMPONENTS, INCLUDING SUPPORTS REQUIRED FOR SEISMIC QUALIFICATION, ARE LABELED IN THE FIELD. ABSCE PRESSURE BOUNDARY CONFIGURATIONS MAY INCLUDE COMPONENTS ON THE ANNULUS AND AUXILIARY BUILDING SIDE OF THE SHIELD WALL. THE SUPPORTS REQUIRED TO MAINTAIN ABSCE BOUNDARY SEISMIC QUALIFICATION ARE IDENTIFIED ON DCA 52283-314.

NOTES CONT:
32. VALVE FAILED CLOSED; AIR SUPPLY ISOLATED TO PREVENT OPERATION. VALVE TO REMAIN IN CLOSED POSITION FOR ABSCE BOUNDARY RETENTION.
33. DAMPER FAIL CLOSED; AIR SUPPLY ISOLATED TO PREVENT OPERATION. VALVE TO REMAIN IN CLOSED POSITION TO SERVE AS A UNIT 1/UNIT 2 INTERFACE POINT.

NOTES CONT:
30. ADJUST BALANCING DAMPERS TO ACHIEVE A DIFFERENTIAL PRESSURE OF APPROXIMATELY 5.2 IN. W.G., OR A TOTAL FLOW RATE OF FROM 30,000 TO 36,000 CFM.

NOTES CONT:
29. LOOP IS DISABLED. EQUIPMENT IS RETAINED FOR PRESSURE BOUNDARY.

NOTE CONT:
27. THIS IS A UNIT 1/UNIT 2 INTERFACE POINT. SAFETY RELATED, (1)(2), INTERFACE POINTS MUST BE LOCKED CLOSED. NON-SAFETY RELATED, [1][2], INTERFACE POINTS MUST BE CLOSED WITH HAND WHEELS REMOVED. FOR CONTROL AIR SUPPLY INTERFACE IMPLEMENTATION REQUIREMENTS SEE DRAWING 1-47W848-10.
28. UNIT INTERFACE SEPARATION BOUNDARIES ARE CONTROLLED BY INSTRUCTION TI-12.08, CONTROL OF UNIT INTERFACES.

VALVE MARKER TAG LAST NUMBER USED 1-30-525

NOTE CONT:
25. DESIGN PRESSURE AND TEMPERATURE FOR CLASS B PIPING BETWEEN CONTAINMENT ISOLATION VALVES ARE 15 PSIG AND 250°F, RESPECTIVELY.
26. [1][2] DENOTE UNIT 1 & UNIT 2 INTERFACE POINTS. THE STRUCTURAL BOUNDARY IS THE FIRST ANCHORED EQUIPMENT OR PIPE ANCHOR ON THE UNIT 2 SIDE OF THE INTERFACE POINT. BECAUSE OF TEES, SOME INTERFACE POINTS WILL HAVE MORE THAN ONE STRUCTURAL TERMINATION. DESIGN CRITERIA AND SYSTEM DESCRIPTION REFERENCE: N3-30RB-4002, N3-65-4001, N3-83-4001

NOTES:
1. NUMBERS SHOWN THUS 4000 INDICATE AIR FLOW IN CFM.
2. FOR IDENTIFICATION OF SYMBOLS SEE 30B617-8 THRU -3.
3. FLOWS SHOWN WITHOUT PARENTHESIS ARE FOR NORMAL OPERATION.
4. ( )1 INDICATES MINIMUM FLOW IN PURGE MODE.
5. ( )2 INDICATES ALTERNATE MODE OF DIVERTING SOME PURGE FLOW THRU ANNULUS.
6. ( )3 INDICATES MINIMUM SUPPLY AND MAXIMUM EXHAUST FLOW TO MAINTAIN A POSITIVE PRESSURE WHEN THE INSTRUMENT ROOM ONLY IS BEING PURGED.
7. ( )4 INDICATES MINIMUM FLOW WHEN AIR RETURN FANS ARE OPERATED.
8.
9. ( )5 INDICATES FLOW FROM ANNULUS, OR ENGINEERED SAFEGUARDS WHEN EMERGENCY GAS TREATMENT SYSTEM IS OPERATED.
10. FLOW RATE IN BRACKETS (30,284) WAS MEASURED WITH ALL FOUR LCC'S OPERATING. IT IS OUTSIDE THE G-37 ACCEPTANCE CRITERIA BUT HAS BEEN EVALUATED AND APPROVED AS DOCUMENTED IN CALCULATIONS WBNOSG4130 AND EPMAW122389.
11. FLOWS SHALL BE BALANCED TO THE RATES SHOWN ON THE DIAGRAM PER CONSTRUCTION SPECIFICATION NO. G-37, TESTING AND BALANCING OF HV AND AC SYSTEMS.
12. DESIGN PRESSURE IS 8 IN-WG IN ACCORDANCE WITH SMACNA HIGH VELOCITY DUCT CONSTRUCTION STANDARDS FOR MEDIUM PRESSURE (EXCEPT AS NOTED). DUCTWORK SHALL BE TESTED IN ACCORDANCE WITH TVA GENERAL CONSTRUCTION SPECIFICATION NO. G-37.
13. FOR VALVE MARKER TABULATIONS SEE WATTS BAR NUCLEAR PLANT MISC VALVE REPORT -009 AND -010.
14. ALL DUCTWORK IS CLASS O (ROUND) OR S (RECTANGULAR) EXCEPT WHERE NOTED AS B OR C. ALL PIPING ON THIS DRAWING IS CLASSIFIED AS TVA CLASS B, C, G, OR H AS NOTED.
15. ALL DUCTWORK CONNECTED TO THE SHIELD BUILDING EXHAUST VENT SHALL BE LEAK TESTED TO THE MOST STRINGENT REQUIREMENTS SPECIFIED IN SECTION 4.12.1 OF ANSI-N509.
16. PURGE AIR FILTER HOUSINGS SHALL BE LEAK-TESTED PER ANSI N509-1976. ALLOWABLE LEAKAGE RATES FOR HOUSINGS 1A AND 1B SHALL BE 7.3 CFM AND 6.7 CFM, RESPECTIVELY (REF. NCR 4234R).
17. DUCT WORK IN THIS AREA REMOVABLE TO ALLOW ACCESS TO REACTOR VESSEL HEAD.
18. DESIGN CRITERIA/SYSTEM DESCRIPTION REFERENCE DOCUMENTS (USE THE LATEST REVISION ON ALL WORK UNLESS OTHERWISE SPECIFIED. SEE THE LATEST REVISION OF THE 47B21 SERIES DRAWINGS PIPING SYSTEM CLASSIFICATION.) N3-30RB-4002 — REACTOR BUILDING VENTILATION SYSTEM
19. ALL DUCTWORK SERVING THE AIR RETURN FANS AND LCC COOLERS IS SAFETY CLASS 2b AND SEISMIC CATEGORY I.
20. ALL PURGE DUCTWORK INSIDE THE CONTAINMENT UP TO THE INBOARD CONTAINMENT ISOLATION VALVES IS NON-SAFETY CLASS AND SEISMIC CATEGORY I(L). ALSO, THE SUPPLY DUCT BETWEEN THE DOWNSTREAM ABSCE ISOLATION VALVE AND SHIELD BUILDING ISOLATION VALVES IS NON-SAFETY CLASS AND SEISMIC CATEGORY I(L).
21. ALL PURGE EXHAUST DUCTWORK DOWNSTREAM OF THE OUTBOARD CONTAINMENT ISOLATION VALVES IS SAFETY CLASS 2b AND SEISMIC CATEGORY I.
22. MINIMUM TOTAL FLOWRATE DURING NORMAL OPERATION WITH 3 LOWER COMPARTMENT COOLER UNITS IS 146,000 CFM. MINIMUM TOTAL FLOWRATE DURING NON-LOCA ACCIDENT CONDITIONS IS 112,000 CFM WITH 2 COOLERS OPERATING. DUCTWORK AND GRILLE FLOWRATES FOR NON-LOCA ACCIDENT CONDITIONS ARE SHOWN IN PARENTHESIS ().
23. LEAK TESTING/LEAK CHECKING OF DUCTWORK INSIDE CONTAINMENT AND ANNULUS SHALL BE IN ACCORDANCE WITH G-37, EXCEPT AS NOTED ON DRAWINGS.
24. BASED ON G-37 PARAGRAPHS 4.3.3.2.1, 4.3.3.2.2, 4.3.3.2.3, 4.3.3.2.5, 4.3.3.2.7.1, 4.3.3.2.8, AND 4.3.3.2.9, WHICH PROVIDE EXEMPTIONS FROM LEAK-TESTING OF DUCTWORK, AND EXCEPTION G-37-WBN-1, DUCTWORK FOR LOWER COMPARTMENT COOLING, PURGE AIR, AND CRDM IS EXEMPTED FROM LEAK-TESTING/LEAK-CHECKING REQUIREMENTS, IF TESTED AIRFLOW RATES SATISFY DESIGN VALUES.

DAMPER LEGEND:
A - AUTOMATIC OPERATING FOR NORMAL SERVICE
RM - REMOTE MANUAL OPERATION
S - AUTOMATIC OPERATING FOR ISOLATION
MANUAL (FOR OPEN-CLOSE OPERATION)
BALANCING
BACKDRAFT
FIRE
FAILS OPEN
FAILS CLOSED
FAILS AS IS
LOCKED OPEN
VACUUM RELIEF

REFERENCE DRAWINGS:
1-47W610-30-1 THRU 7
1-47W610-65-1
47W611-30-1 THRU 4
47W611-65-1, 2 & 3
47W544-4
47W865-3
1-47W866-1A, 2, 10 & 11
1-47W852
47W830-4
47W920-30X1 THRU 30X4
47W915 SERIES
47W920 SERIES
47W601-30 SERIES
47W601-65 SERIES
47W913-30 X 1
THRU -43 X 4
47W866-101 STRESS ANALYSIS PROBLEM BOUNDARY

FSAR

ISSUED BY: S.C.GIBSON FOR WSR

THIS CONFIGURATION CONTROL DRAWING SUPERSEDES UNIT 1 AS-CONSTRUCTED DRAWING 47W866-1 REVISION JJ.

REV	CHANGE REF	PREPARER	CHECKER	APPROVED	DATE
61	WBN-1-2014-030-004	RWE	JEW		DTS
REVISED TO SHOW T-MOD WBN-1-2014-030-004-01-0.					

SCALE: NTS — EXCEPT AS NOTED

PROJECT FACILITY: REACTOR BUILDING UNITS 1 & 2

TITLE: FLOW DIAGRAM HEATING AND VENTILATION AIR FLOW

1 — WATTS BAR NUCLEAR PLANT, TENNESSEE VALLEY AUTHORITY — Q

DESIGN		INITIAL ISSUE	ENGINEERING APPROVAL
DRAFTER: DE-HWA L WANG	CHECKER: JDF/EBJ	RO ISSUE PER WBEP 5.17, RIMS B26 '90 0109 378, B26 '90 0117 377 & DCN P-00484-A	1 C.J.WOOLSON
DESIGNER: M.L.CHAPMAN	REVIEWER: H.BENNINGHOFF		2 C.J.WOOLSON
DATE: 6-18-90	85 M	1-47W866-1	3 F.A.KOONTZ

R61

CAD MAINTAINED DRAWING

CONFIGURATION CONTROL DRAWING

184-A

Source

0-RM-90-118A

Setpoint

determined by Chemistry

WGDT REL LINE
0-RM-118
RAD
HI

(Page 1 of 1)

Probable Cause:

A. High activity release from gas decay tanks (GDTs)
B. Loss of power to control relay (WDGS)
C. Loss of ratemeter power

NOTE

0-RM-90-118 has associated ICS computer point R0016A.

Corrective Action:

[1] **ENSURE** 0-FCV-77-119 CLOSED.

[2] **NOTIFY** Chemistry to perform CM-9.09 “Effluent Radiation Monitor Alarm Guidelines” and Radiation Protection to investigate alarm:

[2.1] **IF** 0-FCV-77-119 closure is due to high activity, **THEN INITIATE** new release package, OR **EVALUATE** postponing release to allow additional decay time.

[2.2] **IF** 0-FCV-77-119 closure is due to malfunctioning 0-RM-90-118, **THEN INITIATE** corrective maintenance.

[3] **EVALUATE** gas decay tank lineup.

[4] **REFER TO** AOI-31, Abnormal Release Of Radioactive Material.

[5] **REFER TO** Tech Specs/ODCM.

References:

1-45W600-57-27
1-45W600-77-1
1-47W601-90-34
1-47W610-90-1
AOI-31
CM-9.09
ODCM

24.

Given the following conditions:

- Radiography activities are being conducted **SIMULTANEOUSLY** in **BOTH** the Waste Package Area **AND** the Decontamination Room Area.
- The Radiographer improperly shields the source and causes a **VALID** HIGH radiation condition within the Decontamination Room Area.
- The condition causes an Annunciator to alarm in the MCR.

Which ONE of the following describes the type of radiation monitor that detected the condition AND the alarm window in the MCR?

A(n) ____(1)____ radiation monitor detected the high radiation condition.

The Annunciator window ____(2)____ identify the **SPECIFIC** room in which the high radiation condition occurred.

A. (1) area
   (2) does

B. (1) area
   (2) does **NOT**

C. (1) process
   (2) does

D. (1) process
   (2) does **NOT**

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: As seen in the design criteria, WBN-DCD-40-24, Radiation Monitoring (Unit 1 / Unit 2), the radiation monitoring system includes both process and area radioactivity monitors. Section 3.1.5 of this design criteria documents the Area radiation monitors and lists that 0-RE-90-003 monitors the Waste Packaging Area and that 0-RE-90-004 monitors the Decontamination Room Area. Therefore, it is correct that an area monitor would detect the high radiation condition which is listed in the stem of the question. However, as seen in ARI-174-B, 1-RR-90-1 AREA RAD HI, the annunciator window which came into alarm does not specify which room had the high radiation condition. Step [1] of the ARI lists: CHECK 1-RR-90-1 and associated RMs [0-M-12] to determine which area radiation monitor is alarming. It is plausible to believe that the annunciator window would stipulate which area had the high radiation levels because this specificity in annunciation is seen with the process radiation monitors. These monitors have individual alarm windows. Additionally, it is plausible to believe that this is true because some of the area radiation monitors do have individual alarm windows (e.g. 0-RM-90-135 and 0-RM-90-102/103).

B. Correct: Area radiation monitors would detect the condition and the alarm window would not identify which area had the high radiation condition.

C. Incorrect: It is not true that a process radiation monitor would detect the condition. It is plausible to believe this as a high radiation condition can cause a process radiation monitor to alarm. An excellent example of this is that a refueling water purification filter change can: 1. raise local area radiation levels significantly, 2. cause 0-RM-90-9, Waste Cond Tnk Area to alarm and 3, cause the process monitor 0-RM-90-122, WDS Release Line to alarm. This later alarm occurs because the process monitor is quite close to the filter cubicle. The locations given in the question do not contain any resident process monitors and the alarm of a process monitor is not possible. As described it is not correct but plausible to believe that the annunciator received would indicate the location of the high radiation.

D. Incorrect: Again it is not true but plausible that a process monitor would detect the condition. Also, it is true that the alarm window received would not identify the location of the high radiation condition.

Question Number: 24

Tier: 1 Group: 2

K/A: 061 Area Radiation Monitoring (ARM) System Alarms
AK2. Knowledge of the interrelations between the Area Radiation Monitoring (ARM) System Alarms and the following:
AK2.01 Detectors at each ARM system location

Importance Rating: 2.5 2.6

10 CFR Part 55: (CFR 41.7 / 45.7)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to understand the interrelation of the Annunciation and Alarm system and two detectors with the ARM system.

Technical Reference: WBN-DCD-40-24, Radiation Monitoring
ARI-174-B, 1-RR-90-1 AREA RAD HI

Proposed references to be provided: None

Learning Objective: 3-OT-SYS090A, Radiation Monitoring
3. IDENTIFY the following controls and indications; include main control room, auxiliary control room, and local panels as applicable:
a. Liquid Process Rad Monitors (RMs)
b. Liquid Effluent RM's
c. Gaseous Process RM's
d. Gaseous Effluent RMs
e. Area RMs (ARM)
f. Post Accident Area RMs (PAARM)

Cognitive Level:
Higher ____
Lower X

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

1.0 SCOPE

This document establishes the design requirements of the Radiation Monitoring System (RMS) for the operation of Watts Bar Units 1 and 2 [Source Note 7]. The RMS provides continuous monitoring of radiation levels, provides the results of the monitoring to the operator and in some instances produces signals to initiate automatic control actions. In addition, the RMS includes equipment for off-line tritium, particulate and iodine sample collection with no real-time detection. The RMS consists of process, airborne, area, and effluent radioactivity monitors. The outdoor radiological monitoring instrumentation, laboratory radiation measurement equipment, criticality monitoring equipment, and portable survey instrumentation is outside the scope of this criteria. If a discrepancy exists between this design criteria and any other Nuclear Engineering (NE) design criteria, the appropriate Engineering Manager should be notified by a memorandum. If a discrepancy exists between this design criteria and any other document where the other document is not a NE design criteria, this design criteria shall govern.

2.0 DEFINITIONS

Accuracy - The degree of agreement with the true value of the quantity being measured.

Accessible Area - An area which is designed to be entered without special radiological controls (i.e., is not a high radiation area or airborne radioactivity area), dismantling of installed equipment or the introduction of support equipment such as staging, ladders, etc.

Anticipated Operational Occurrences - Those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to the loss of power to all reactor coolant recirculation pumps, a trip of the turbine generator set, isolation of the main condenser and a loss of all off-site power.

Categories 1, 2, and 3 - References to Categories 1, 2, and 3 are as stated in RG 1.97 (Reference 8.1.11).

Channel - A channel includes all of the instrumentation necessary to provide the monitoring function. A channel includes associated sample lines, skid assemblies, and local readout and alarm devices. A channel also includes equipment in the control room for recording, readout, and alarm and includes the electrical cable that connects monitor components.

Check Source - A test source supplied as an integral part of the monitor channel for use in determining if the channel is functional. This may be a radioactive source, or when a check of the channel exclusive of the detector scintillator is appropriate (i.e., where increased background cannot be tolerated), a pulsed light source or other similar device appropriate for the type of detector.

Class 1E - The safety-related classification of the electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment.

Grab Sample - A volume removed from a process or effluent for laboratory analysis.

Gross beta or gamma radioactivity - The combined radioactivity from the mix of nuclides without distinction as to energy levels of the emissions.

3.1.2 Off-Line Gas Monitors (continued)

The measured noble gas activity level shall be transmitted to the Plant Computer System for display in the TSC.

3.1.3 Continuous Air Monitors (Airborne Particulate Monitors)

The following continuous air (particulate) monitors shall be provided for plant areas where airborne particulate radioactivity may potentially exist and where normal occupancy is required either on a continuous basis or on an infrequent but routine basis:

- * Spent Fuel Pool Monitor
- * Shipping Bay Monitor
- * Holdup Valve Gallery Monitor
- * SI Pump Area Monitor
- * Main Control Room Monitor
- * Waste Packaging Room Monitor
- * Sample Rooms Monitors

* Monitoring of the above listed plant areas are accomplished using portable continuous air monitors supplied and administered by Site Radcon. The requirements of this Design Criteria do not apply to these monitors.

3.1.4 Off-Line Liquid Monitors

The monitors that come under this category are:

Steam Generator Blowdown Liquid Monitor	1,2-RE-90-120 & -121
Waste Disposal System Liquid Monitor	0-RE-90-122
Component Cooling System Monitor	0,1,2-RE-90-123
Condensate Demineralizer Regenerant Waste Discharge Monitor	0-RE-90-225
Essential Raw Cooling Water Effluent Monitors	0-RE-90-133-, 134,-140 & -141

These monitors provide real-time monitoring of the gross gamma radioactivity in liquid samples. Selected off-line liquid monitor channels shall initiate an automatic control function to terminate releases either to the unrestricted area or to the plant environs as described in the following subsections.

A. General Requirements

The monitors shall have pumps or the process connection points shall provide sufficient differential pressure to ensure that an adequate flow rate from the process streams can be predictably delivered to the sample chamber.

3.1.4 Off-Line Liquid Monitors (continued)

3.1.5 Area Radiation Monitors

The following monitors shall continuously monitor ambient radiation levels in the plant buildings to assure that work areas designed for short-term accessibility or normal occupancy have exposure rates which do not exceed the prescribed radiation zone limits, and to provide early warning of abnormal process system operations.

Spent Fuel Pool Area Monitors	1,2-RE-90-001
Personnel Lock Monitors	1,2-RE-90-002
Waste Packaging Area Monitor	0-RE-90-003
Decontamination Room Monitor	0-RE-90-004
Spent Fuel Pool Pumps Area Monitor	0-RE-90-005
Component Cooling Heat Exchanger Area Monitors	1,2-RE-90-006
Sample Rooms Monitors	1,2-RE-90-007
Auxiliary Feedwater Pump Area Monitors	1,2-RE-90-008
Waste Evaporator Condensate Tank Area Monitor	0-RE-90-009
CVCS Board Area Monitors	1,2-RE-90-010
Containment Spray RHR Pump Area Monitor	0-RE-90-011
RB Upper Compartment Refueling Floor Area Monitors	1,2-RE-90-059
RB Upper Compartment Area Monitors	1,2-RE-90-060
RB Lower Compartment Instrument Room Monitors	1,2-RE-90-061
Main Control Room Monitor	0-RE-90-135
Condensate Demineralizer Area Monitor	0-RE-90-230
Condensate Demineralizer Area Monitor	0-RE-90-231
RB Upper Compartment Post Accident Monitors	1,2-RE-90-271 & -272
RB Lower Compartment Post Accident Monitors	1,2-RE-90-273 & -274

A. General Requirements

The monitors shall provide real-time measurement of gross gamma ambient radiation exposure rates in plant areas. These instruments shall be installed in locations which satisfy the guidelines in RG 8.8 (Reference 8.1.6) and ANSI/ANS-HPSSC-6.8.1-1981 (Reference 8.2.3).

The monitors shall have detectors with sufficient range to encompass the minimum and maximum exposure rate expected during normal and anticipated operational occurrences (considering the ambient background radiation conditions at the detectors' location). The monitors should have a minimum of five decades of range as described in ANSI/ANS-HPSSC-6.8.1-1981 (Reference 8.2.3). Certain locations may require a range in excess of five decades. Where multiple detectors are required to cover the anticipated range, the range of the detectors shall overlap. Overlap should be at least one decade. Extended ranges are discussed in the individual monitor sections below.

UNIT 1 RADIATION DETECTORS
0-XA-55-12A

	173	174	175	176	177	178	179	
A	UPR CNTMT AIR 1-RM-112 RAD HI	CNTMT PURGE EXH 1-RM-130/131 RAD HI R				SG BLDN 1-RM-120/121 LIQ RAD HI	CCS HX A 1-RM-123 LIQ RAD HI	A
B	LWR CNTMT AIR 1-RM-106 RAD HI	1-RR-90-1 AREA RAD HI R	VAC PMP EXH 1-RM-119 RAD HI					B
C								C
D	UPR CNTMT AIR 1-RM-112 INSTR MALF	CNTMT PURGE EXH 1-RM-130/131 INSTR MALF R				SG BLDN 1-RM-120/121 INSTR MALF R	CCS HX A OUTLET 1-RM-123 INSTR MALF	D
E	LWR CNTMT AIR 1-RM-106 INSTR MALF	1-RR-90-1 AREA MONITORS INSTR MALF R	VAC PMP EXH 1-RM-119 INSTR MALF	PAS MON PNL M30 / M31 INSTR MALF R				E
F								F
	173	174	175	176	177	178	179	

174-B

Source	**Description**	**Setpoint**
1-RM-90-1	Spent Fuel Pit	determined by Radiation Protection
1-RM-90-2	Personnel Access Outlet	
0-RM-90-3	Waste Package Area	
0-RM-90-4	Decontamination Room Area	
0-RM-90-5	Spent Fuel Pit Pumps	
1-RM-90-6	CCS Hx Area	
1-RM-90-7	Sample Room	
1-RM-90-8	AFWP Area	
0-RM-90-9	Waste Cond Tnk Area	
1-RM-90-10	CVCS Bd Area	
0-RM-90-11	CS and RHR Pump Area	
1-RM-90-59	Upper Cntmt - by equip hatch	
1-RM-90-60	Upper Cntmt - by air lock	
1-RM-90-61	Incore Inst Room	
0-RM-90-230	Cond Demin	
0-RM-90-231	Cond Demin	

1-RR-90-1
AREA
RAD
HI

R

(Page 1 of 1)

Probable Cause:

A. Radiation rise in affected area
B. Proximity of 0-RM-90-3 to Radwaste Mobile Demin package
C. Loss of power to rate meter

Corrective Action:

[1] **CHECK** 1-RR-90-1 and associated RMs [0-M-12] to determine which area radiation monitor is alarming.
[2] **CHECK** Unit 2 area monitors and Aux Bldg vent 0-RM-90-101.
[3] **IF** loss of power indicated (green light out and 174-E LIT), **THEN** **GO TO** of 174-E.
[4] **NOTIFY** Radiation Protection to investigate alarm.
[5] **REFER TO** 1-AOI-31, Abnormal Release Of Radioactive Material.
[6] **IF** during refueling, **THEN** **REFER TO** AOI-29, Dropped Or Damaged Fuel Or Refueling Cavity Seal Failure.

References:

47W610-90-2
45N1651-9, -11
1-AOI-31
AOI-29

25.

Which ONE of the following describes the use of Self-Contained Breathing Apparatuses (SCBAs) in accordance with the Watts Bar Fire Protection Report?

The operating life of the SCBAs used by control room personnel is a **MINIMUM** of ____(1)____.

As this time limit is approached, the wearer of the SCBA will ____(2)____.

A. (1) 30 minutes
   (2) have the SCBA's cylinder replaced

B. (1) 30 minutes
   (2) utilize a manifold system supplied from a storage reservoir

C. (1) 2 hours
   (2) have the SCBA's cylinder replaced

D. (1) 2 hours
   (2) utilize a manifold system supplied from a storage reservoir

<u>CORRECT ANSWER:</u> <u>A</u>

DISTRACTOR ANALYSIS:

A. Incorrect: It is correct that the operating life of the SCBA is 30 minutes.

It is correct that the wearer of the SCBA will have his cylinder be replaced by a helper (another member of the control room staff) as it nears exhaustion.

B. Incorrect: As mentioned it is correct that the operating life of the SCBA is 30 minutes.

Additionally, it is not correct that a manifold system be utilized. It is plausible to believe this because of several reasons. Firstly, the service air system supplies portable breathing air stations (PBAS) via quick disconnect couplings. PBAS are available to supply air to a maximum of 6 personnel (this fact is observed in the system description WBN-SDD-N3-32-4002, Compressed Air System). Secondly, chemistry personnel having breathing air systems built into their Post Accident and Primary sample labs. As seen in system description, N3-43B-4001, Sampling and Water Quality System Post accident Sampling Facility, the breathing air systems is designed to supply $\geq$6cfm/person (based upon 3 people). It would be reasonable to believe that if chemistry personnel were provided with a breathing air system that the occupants of the main control room would be as well.

C. Incorrect: It is not correct that the operating life of the SCBA is 2 hours. As seen on page VIII-34 of Part VIII of the Watts Bar Fire Protection report: The operating life of the self-contained units is a minimum of one-half hour. It is plausible to believe that the operating life is 2 hours because the MSA Air Hawks which are provided to the operations personnel in the MCR do have an air bottle option which is rated at 120 minutes. This is the bottle size which is used at the TVA Nickajack fire training facility (for use with the MSA Fire and Air Hawk SCBAs).

The wearer of the SCBA would have his cylinder replaced as it became exhausted

D. Incorrect: It is not correct that the operating life is 2 hours; it is not correct that a manifold system would be utilized.

Question Number: 25

Tier: 1 Group: 2

K/A: 067: Plant fire on site
AA2. Ability to determine and interpret the following as they apply to the Plant Fire on Site:
AA2.10 Time limit of long-term-breathing air system for control room

Importance Rating: 2.9 3.6

10 CFR Part 55: (CFR: 43.5 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to determine the time limit of the long-term-breathing air system for the control room by understanding the operating life of the SCBA (the breathing air system utilized by the main control room staff during a plant fire).

Technical Reference: Part VIII of the Watts Bar Fire Protection report
N3-43B-4001, Sampling and Water Quality System Post-accident Sampling Facility
WBN-SDD-N3-32-4002, Compressed Air System

Proposed references to be provided: None

Learning Objective: 3-OT-AOI3000, AOI-30.1 &30.2, Plant Fires
8. Given the indications of a plant fire, DETERMINE the following as they apply to 1-AOI-30, Plant Fires
f. Time limit of long-term-breathing air system for control room
g. Time limit for use of respirators

Cognitive Level:
Higher ____
Lower X

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Appendix A Guidelines	Plant Conformance	Alternatives	Remarks
Self-contained breathing apparatus, using full face positive pressure masks, approved by NIOSH (National Institute for Occupational Safety and Health - approval formerly given by the U.S. Bureau of Mines) should be provided for fire brigade, damage control, and control room personnel.	NIOSH-approved self-contained full-face positive pressure breathing apparatus is available for the fire brigade, damage control, and control room personnel. The operating life of the self-contained units is a minimum of one-half hour.		
Control room personnel may be furnished breathing air by a manifold system piped from a storage reservoir, if practical. Service or operating life should be a minimum of one-half hour for the self-contained units.			Not applicable.
At least two extra air bottles should be located onsite for each self-contained breathing unit. In addition, an onsite-6-hour supply of reserve air should be provided and arranged to permit quick and complete replenishment of exhausted supply air bottles as they are returned. If compressors are used as a source of breathing air, only units approved for breathing air should be used. Special care must be taken to locate the compressor in areas free of dust and contaminants.		At least (1) one hour air bottle is available for each self-contained breathing unit. An additional (16) one hour bottles are available onsite to provide a minimum 16-hour supply of reserve air for the self-contained breathing units provided for use by the fire brigade, damage control, and control room personnel.	

3.1.1 General (continued)

The C&SA compressors discharge into two redundant headers provided with manual valves. These headers feed the two CAS receivers which in turn supply air through redundant headers to the CAS dryers. The dryers contain three trains of prefilters, dryers, and afterfilters. Each dryer train is sized to handle CAS requirements for one unit. Manual bypasses are provided around each filter assembly for abnormal operation. The air is then piped to pneumatic equipment throughout the plant.

Air is supplied to the SAS receiver by a header from the CAS receivers. Service air is supplied through a valve which closes if CAS pressure drops below a minimum value of 80 psig to assure that CAS requirements take precedence over the SAS. SAS air is piped from the receiver to service outlets and miscellaneous equipment throughout the plant. Like CAS air, the SAS is delivered oil-free. SAS air is not required to be dried and filtered.

A 4” permanent piping segment with a “Tee”, an isolation valve and connecting flange is provided to the existing service air header for emergency or future compressed air usage. The tie-in is located inside the Turbine Building.

The portion of the SAS downstream of valves 2-33-509 and 2-33-543 will be supplied oil free air from temporarily installed Construction Air Compressors through the tie flange located at 2-33-675. Valves 2-33-509 and 2-33-543 will have their positions controlled as shown on the corresponding flow diagrams to prevent interaction between the Construction Air Compressors and the permanently installed station air compressors.

The SAS shall be the normal source of breathing air. SAS air connects to portable breathing air stations (PBAS) via quick-disconnect couplings. PBAS are available for supplying air to a maximum of 6 personnel. These stations provide breathing air for personnel in areas with potential airborne radioactivity hazards that are not immediately hazardous to health. PBAS provide a separate source of breathing air as required by Reg Guide 8.15 (Ref 7.5.3).

Note: Associated equipment for breathing air (i.e., Post Accident Sampling Facility and Self-Contained Breathing Apparatus) have no physical interface or otherwise with the SAS, therefore it is not discussed here.

The ACAS is supplied by 2 motor-driven, nonlubricated, compressors. Each compressor is sized to supply the total safety-related air requirements in the event of an accident, flood, or loss of the CAS. ACAS flow demand is 35.14 scfm for Unit 1 operation (Ref 7.4.6). The ACAS is separated into 2 independent trains, each containing a compressor, receiver, dryer, and filter. Relief valves are placed on the ACAS compressors and air receivers (Ref 7.4.2). Electric power for the ACAS is provided from normal and emergency sources. The ACAS automatically isolates from the CAS on low CAS pressure.

Attachment 4
(Page 1 of 2)

ABBREVIATIONS AND ACRONYMS

ABGTS	-	Auxiliary Building Gas Treatment System
ac	-	Alternating current
ACAS	-	Auxiliary Control air subsystem
AFW	-	Auxiliary feedwater
ANSI	-	American National Standards Institute
ASME	-	American Society of Mechanical Engineers
CAS	-	Control air subsystem
CIV	-	Containment Isolation Valve
C&SA	-	Control and Station Air
CNTMT	-	Containment
Compr	-	Compressor
Cu	-	Copper
dc	-	Direct current
DP	-	Differential Pressure
EGTS	-	Emergency Gas Treatment System
ERCW	-	Essential Raw Cooling Water
°F	-	Degrees Fahrenheit
FCV	-	Flow Control valve
FMEA	-	Failure Modes and Effect Analysis
FSAR	-	Final Safety Analysis Report
ft^3	-	Cubic Feet
gpm	-	Gallons per minute
hp	-	High pressure
HVAC	-	Heating, Ventilation, and Air-Conditioning
Hz	-	Hertz
ICFM	-	Inlet Cubic Feet per Minute
IR	-	Ingersoll-Rand
lp	-	Low pressure
ph	-	Phase
MCRHZ	-	Main Control Room Habitable Zone
NEMA	-	National Electrical Manufacturer's Association
Ni	-	Nickel
NPT	-	Nominal pipe threads
PA	-	Pressure alarm
PAS	-	Post Accident Sampling
PCV	-	Pressure Control value
PI	-	Pressure indicator
PIC	-	Pressure indicating controller
PBAS	-	Portable Breathing Air Station

Appendix A
(Page 2 of 3)

Preoperational Test Criteria

TABLE 1
COMPONENT REQUIREMENTS

COMPONENT	REQUIREMENTS	DESIGN BASIS	NOTES
LIQUID SAMPLE PANEL L-567	RCS AND CONTAINMENT SUMP SAMPLE FLOW OF ≤ 1900 ML/MIN DURING PURGING & 200 ± 20 ML/MIN DURING SAMPLING	N3-43B-4001 P. 3.1.1 Table 9-2 80 x 62-827371	NOTE 1
CONTAINMENT AIR	CONTAINMENT SAMPLE AIRFLOW	P. 3.2.11	NOTE 1, 5
SAMPLING PANEL L-569	≥ 0.2 ACFM HEAT TRACING MAINTAINS SAMPLES ≥ 280 F	Table 9-2 80 x 62-827371 WB-DC-40-39 p. 5.3	
SAMPLE COOLERS	COOL RCS SAMPLES ≤ 102 F WITH ≤ 95 F CCS COOLING WATER	N3-43B-4001 P. 3.1.1	NOTE 3
ION CHROMATOGRAPH 1&2-XAN-43-284 (U2 ONLY)	FLOW = 15 ± 2 ML/MIN, INCREASING	80 x 62-827371	NOTE 4
DISSOLVED OXYGEN 1&2-XAN-43-321	FLOW = 200 ± 20 ML/MIN, INCREASING	N3-43B-4001 TABLE 9-2	
GAS CHROMATOGRAPH 1&2-XAN-43-338	CARRIER GAS FLOW ≥ 60 ML/MIN PURGE GAS FLOW ≥ 350 ML/MIN	827371	
BREATHING AIR	FLOW ≥ 6 CFM/PERSON (BASED ON 3 PEOPLE)	WB-DC-40-39 P. 4.3	NOTE 2

NOTES:

1. SET POINTS CONTROLLED BY 1-47B601-43 SERIES.
2. SET POINTS CONTROLLED BY 1-47B601-31 SERIES.
3. MUST BE PERFORMED DURING HOT FUNCTIONAL TEST.
4. NOT REQUIRED FOR UNIT 1.
5. MUST VERIFY THAT THE AIR SAMPLES ORIGINATE FROM CONTAINMENT.

26.

Given the following timeline:

00:00:00	Unit 1 is at 100% power.
00:01:00	A Steam Line Break Occurs inside CNTMT.
00:01:15	The 1B-B MDAFWP shears its shaft.
00:01:45	CNTMT Pressure is 3.0 psig and RISING ⇑**.**
00:05:00	ALL MSIV's hand-switches have **RED** lights **LIT.**
00:07:00	1A-A MDAFWP TRIPS due to a motor fault.
00:10:00	1-FR-P.1, Pressurized Thermal Shock is entered.
00:10:30	SG Narrow Range levels are:

SG 1	SG 2	SG 3	SG 4
1%	4%	3%	3%

00:11:00	Step 3 of 1-FR-P.1 is completed

Which ONE of the following describes the status of AFW?

At 00:11:00, the operating crew has aligned AFW to Feed ________.

A. **ALL** SGs at 410 gpm

B. **ALL** SGs at MINIMUM detectable flow

C. **ONLY** the SG supplying the TDAFWP at 410 gpm

D. **ONLY** the SG supplying the TDAFWP at MINIMUM detectable flow

<u>CORRECT ANSWER:</u>   <u>B</u>

<u>DISTRACTOR ANALYSIS:</u>

A. Incorrect: Incorrect: Because the narrow range levels in all S/Gs are less than 39%, the status trees will indicate a red path to 1-FR-H.1, Loss of Secondary Heat Sink if total FW flow is not greater than 410 gpm. Additionally, the loss of secondary heat sink status tree is a higher priority instruction than the pressurized thermal shock tree. Therefore, it is plausible to believe that one would feed S/Gs at a value to preclude generating a red path condition to 1-FR-H.1.

B. Correct: Step 3 of 1-FR-P.1 states: IF all S/Gs Faulted, THEN CONTROL feed flow at minimum detectable flow to each S/G. The WOG Basis document for this step explains that the minimum feed flow of setpoint (S.04) should be maintained to each steam generator. The EOP Setpoints Verification Document, WBN-OSG-4-188 explains that the purpose of S.04 is to allow the SG components to remain in a wet condition thereby minimizing any thermal shock effects if feed flow is increased.

C. Incorrect: The stem of the question indicates that both of the MDAFWPs have failed. Therefore, the only source of AFW is the TDAFWP. A caution prior to step 3 states: If the turbine-driven AFW pump is the only available source of feed flow, steam supply to the turbine-drive AFW pump must be maintained from at least one S/G. Therefore, it is plausible that one may believe that the SG feeding the TDAFWP's steam would be the only SG required to have inventory. Additionally, one may couple this belief with that mentioned previously regarding the prevention of a red path to 1-FR-H.1. Therefore, because of these two items, this distractor is plausible.

D. Incorrect: Again, it is plausible to believe that feeding only the SG supplying the TDAFWP's steam is correct. It is correct to feed at a minimum detectable flow rate.

Question Number: 26

Tier: 1 Group: 2

K/A: E08 Pressurized Thermal Shock
EK1. Knowledge of the operational implications of the following concepts as they apply to the (Pressurized Thermal Shock)
EK1.3 Annunciators and conditions indicating signals, and remedial actions associated with the (Pressurized Thermal Shock).

Importance Rating: 3.5 4.0

10 CFR Part 55: (CFR: 41.8 / 41.10, 45.3)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to rationalize the condition indicating signals (the HI-HI containment pressure which should have closed the MSIVs and the subsequent failure of the MSIVs to close) impact to the performance of 1-FR-P.1. The applicant must subsequently determine the remedial action required to combat a fault of all S/Gs during a Pressurized Thermal Shock event..

Technical Reference: 1-FR-H.1, Loss of Secondary Heat Sink
1-FR-P.1, Pressurized Thermal Shock
WOG basis document for 1-FR-P.1
Setpoint document for the EOPs, WBN-OSG4-188

Proposed references to be provided: None

Learning Objective: 3-OT-FRP0001, Function Restoration Guidelines 1-FR-P.1 & .2
3. Explain why minimum detectable flow is maintained to each S/G of all the S/Gs are faulted.

Cognitive Level:
Higher X
Lower

Question Source:
New X
Modified Bank
Bank

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Step	Action/Expected Response	Response Not Obtained

3.0 OPERATOR ACTIONS

CAUTION

- If total feed flow CAPABILITY of 410 gpm is available, this Instruction should **NOT** be performed.
- If an Intact S/G is available, feed flow should **NOT** be reestablished to any faulted S/G.

Step	Action/Expected Response	Response Not Obtained
1.	**CHECK** if secondary heat sink is required:	
	a. RCS pressure greater than any Intact S/G pressure.	a. **RETURN TO** Instruction in effect.
	b. RCS temperature greater than 375°F [360°F ADV].	b. **PLACE** RHR System in service while continuing in this instruction. • **REFER TO** 1-SOI-74.01, Residual Heat Removal System. **WHEN** adequate RHR shutdown cooling established, **THEN** **RETURN TO** Instruction in effect.
2.	**ENSURE** at least one charging pump RUNNING.	**STOP** all RCPs **AND** ** **GO TO** Cautions prior to Step 18 to initiate RCS bleed and feed.

Step	Action/Expected Response	Response Not Obtained

CAUTION

- If the turbine-driven AFW pump is the only available source of feed flow, steam supply to the turbine-driven AFW Pump must be maintained from at least one S/G.
- If a faulted S/G is necessary for RCS temperature control feed flow to that generator should be maintained.

NOTE

- A Faulted S/G is any S/G that is depressurizing in an uncontrolled manner or is completely depressurized.
- Minimum detectable flow is assured by observing flow indicator response to valve movement.

Step	Action/Expected Response	Response Not Obtained

3.	**CHECK** S/G pressures:
	• All S/G pressures controlled or rising.
	• All S/G pressures greater than 140 psig.
		IF S/G with low or dropping press has **NOT** been isolated, **THEN**
		
		REFER TO 1-E-2, Faulted Steam Generator Isolation, while continuing with this Instruction.
		
		ENSURE TD AFW pump being supplied from intact S/G.
		
		IF all S/Gs Faulted, **THEN**
		
		CONTROL feed flow at minimum detectable flow to each S/G.

Step continued on next page

Step	Action/Expected Response	Response Not Obtained
3.	(continued)	

		IF Faulted S/G necessary for RCS temp control, **THEN**
		
		CONTROL feed flow to minimum detectable flow to that S/G.
		
		IF Faulted S/G **NOT** necessary for RCS temp control, **THEN**
		
		ISOLATE all feedwater to Faulted S/G.

4.	**MONITOR** CST volume greater than 200,000 gal.	**INITIATE** CST refill USING 1-SOI-59.01, Demineralized Water System. **IF** CST volume drops to less than 5000 gal, **THEN** **MONITOR** AFW pumps to ensure suction transfer.
5.	**MINIMIZE** RCS cooldown:	
	a. **CHECK** at least one intact S/G NR level greater than 29% [39% ADV].	a. **CONTROL** total feed flow to maintain greater than 410 gpm UNTIL NR level in at least one Intact S/G greater than 29% [39% ADV].
	b. **CONTROL** feed flow to intact S/G(s) as necessary.	
	c. **CONTROL** S/G pressures as necessary.	

STEP: Check RCS Cold Leg Temperatures - STABLE OR INCREASING

PURPOSE: To determine if RCS cold leg temperature is still decreasing and, if so, to attempt to stop the decrease

BASIS:

Cold leg temperature is the best available indication of vessel downcomer temperature. It is important to terminate, if possible, any cooldown in progress to limit the extent of possible vessel damage due to excessive thermal stresses.

If the RCS cold leg temperatures are decreasing the operator is instructed to eliminate any secondary-side or RHR System instigated RCS cooldown. The items checked in this step are in a preferred order such that the most probable causes of the cooldown are checked first. Therefore, any valves that dump steam are verified to be closed. Next, any cooldown from the RHR System terminated. A cooldown caused by overfeeding the intact SGs is stopped by controlling feed flow consistent with minimum secondary heat sink requirements. The operator checks for any faulted SGs and isolates them. Finally, if a faulted SG is necessary for RCS temperature control or if all SGs are faulted, feed flow to those SGs is controlled at a minimum measurable value to minimize the effects of the RCS cooldown due to the secondary side depressurization.

ACTIONS:

- Determine if RCS cold leg temperatures are stable or increasing
- Determine if SG PORVs are closed
- Determine if condenser steam dump valves are closed
- Determine if RHR System is in service
- Stop any cooldown from RHR System
- Determine if main steamline isolation and bypass valves are closed
- Determine if a faulted SG is necessary for RCS temperature control
- Determine if all SGs are faulted
- Isolate all feedwater to faulted SG(s) unless necessary for RCS temperature control
- Close steam supply valves from faulted SG(s) to turbine-driven AFW pump
- Control feed flow to non-faulted SG(s) to stop RCS cooldown by maintaining total feed flow greater than (S.02) gpm until narrow range level greater than (M.02)% [(M.03)% for adverse containment] in at least one non-faulted SG
- Control feed flow at (S.04) gpm to the faulted SG that is necessary for RCS temperature control
- Control feed flow at (S.04) gpm when all SGs are faulted

INSTRUMENTATION:

- o RCS cold leg temperature indication
- o SG PORVs position indication
- o Condenser steam dump valves position indication
- o SG pressure indication
- o Main steamline isolation and bypass valves position indication
- o Steam supply valves to turbine-driven AFW pump position indication
- o Feed flow indication
- o SG narrow range level indication
- o Plant specific instrumentation to indicate if RHR System is in service

CONTROL/EQUIPMENT:

- o Switches for:
  - SG PORVs
  - Condenser steam dump valves
  - Main steamline isolation and bypass valves
  - Feedline isolation valves
  - Steam supply valves to turbine-driven AFW pump
  - Feed flow control valves (both main and AFW)
- o Plant Specific RHR System controls

KNOWLEDGE:

Thermal stresses caused by rapid RCS temperature transients are not immediately removed if cooldown is stopped or RCS heatup occurs. Therefore, additional steps to minimize RCS pressure and to perform an RCS temperature "soak" are necessary.

PLANT-SPECIFIC INFORMATION:

- o (M.02) SG level just in the narrow range, including allowances for normal channel accuracy and reference leg process errors.
- o (M.03) SG level just in the narrow range, including allowances for normal channel accuracy, post accident transmitter errors, and reference leg process errors, not to exceed 50%.
- o (S.02) The minimum safeguards AFW flow requirement for heat removal, including allowances for normal channel accuracy (typically one MD AFW pump capacity at SG design pressure).

  This flow is equivalent to the minimum AFW flow design requirement that must be delivered to the intact steam generator as assumed in the main feedline break safety analysis.
- o (S.04) Feed flow value in plant specific units corresponding to 25 gpm.

  The 25 gpm value is representative of a minimum measurable feed flow to a steam generator. Plant specific values may depend upon flow instrumentation and the sensitivity of the controls on feed flow.

27.

Given the following timeline:

00:00:00	Unit 1 is at 100% power.
00:00:00	Loss of Offsite power results in a Reactor Trip.
01:00:00	ES-0.3, Natural Circulation Cooldown with Steam Void in Vessel (with RVLIS), is in progress.
	PZR pressure is 785 psig and being LOWERED by auxiliary spray.
	Core Exit TCs indicate 520°F.
	RVLIS Upper Plenum range indicates 81%.

Assuming **NO** additional operator action, which ONE of the following describes the expected RVLIS and PZR level trends as the depressurization continues?

RVLIS Level will ____(1)____ AND PZR Level will ____(2)____.

	(1)	(2)
A.	RISE ⇑	RISE ⇑
B.	RISE ⇑	LOWER ⇓
C.	LOWER ⇓	RISE ⇑
D.	LOWER ⇓	LOWER ⇓

CORRECT ANSWER: C

DISTRACTOR ANALYSIS:

A. Incorrect: Plausible if the candidate determines that the bubble in Rx vessel head will decrease with the use of aux spray, and PZR level would increase with the use of aux. spray.

B. Incorrect: Plausible if the candidate determines that the bubble in Rx vessel head will decrease with the use of aux spray, thus RVLIS level would increase and PZR level would decrease.

C. Correct: With aux spray in service as the PZR pressure is decreased the bubble in Rx vessel head will grow. The expected plant response for this condition is that as the bubble in Rx head increases, RVLIS will decrease and PZR level will increase

D. Incorrect: Plausible since Rx vessel head bubble would grow causing RVLIS to decrease but PZR level would increase not decrease.

Question Number: 27

Tier: 1 Group: 2

K/A: Westinghouse
E10 Natural Circulation with Steam Void in Vessel with/without RVLIS
EA1. Ability to operate and / or monitor the following as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS)
EA1.3 Desired operating results during abnormal and emergency situations.

Importance Rating: 3.4 3.7

10 CFR Part 55: (CFR: 41.7 / 45.5 / 45.6)

10CFR55.43.b: Not applicable

K/A Match: This question matches the K/A by having the candidate determine the expected plant response to LOWERING⇓ RCS pressure during Natural Circulation conditions.

Technical Reference: Westinghouse basis document for ES-0.3, Natural Circulation Cooldown with Steam Void in Vessel (with RVLIS)

Proposed references to be provided: None

Learning Objective: 3-OT-EOP0000, Emergency Instructions Reactor Trip Or Safety Injection, 1-E-0, ES-0.1, 1-ES-0.2, ES-0.3 and ES-0.4
14. Using 1-ES-0.2 DETERMINE the correct response for RCS parameters indicating RCS voiding.

Cognitive Level:
- Higher X
- Lower ___

Question Source:
- New ___
- Modified Bank ___
- Bank X

Question History: Bank question adopted from SQN question ES-0.3-B.2 001.
.

Comments:

STEP: Establish PRZR Level To Accommodate Void Growth

PURPOSE: To ensure that there is adequate space in the PRZR to allow the displacement of fluid from the primary system due to the formation of a void in the vessel

BASIS:

In this guideline as the primary system is cooled and depressurized under natural circulation conditions, a potential for void formation in the upper head region exists. If a void does form, it will displace primary fluid from the vessel into the PRZR as it grows. Therefore, before any further cooldown/depressurization is performed, the PRZR level must be low enough to accommodate this void growth and high enough to cover the PRZR heaters and prevent letdown from isolating. A level between (D.06)% and (D.12)% satisfies these requirements. In addition, PRZR level controls are placed in manual to allow any increase in PRZR level due to void growth.

ACTIONS:

o Determine if PRZR level is between (D.06)% and (D.12)%
o Control charging and letdown as necessary
o Place PRZR level control in manual

INSTRUMENTATION:

o PRZR level indication
o Charging flow indication
o Letdown flow indication
o PRZR level control setting indication
o Charging flow control valve position indication
o Plant specific letdown valve position indication (includes letdown isolation and orifice isolation)

CONTROL/EQUIPMENT:

o Charging flow control valve controls
o Plant specific letdown control valves
o PRZR level control

28.

Given the following timeline:

00:00:00 Unit 1 is at 100% power.

00:00:01 Main Turbine Trips.

00:00:03 A Reactor Trip occurs.

Which ONE of the following describes how DNBR is affected and the RCP design feature that assists in RCS heat removal?

In accordance with the Final Safety Analysis Report (FSAR), at 00:00:02, the departure from nucleate boiling ratio (DNBR) is ____(1)____.

The **MINIMUM** heat removal needed to prevent core damage after a Reactor Trip is **DIRECTLY** ensured by ____(2)____.

A. (1) RISING ⇑
   (2) a flywheel attached to each RCP

B. (1) LOWERING ⇓
   (2) a flywheel attached to each RCP

C. (1) RISING ⇑
   (2) maintaining the RCPs connected to the generator and external network for 30 seconds

D. (1) LOWERING ⇓
   (2) maintaining the RCPs connected to the generator and external network for 30 seconds

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: As seen in table 15.2-1 (sheet 4 of 7) of the Watts Bar Final Safety Analysis Report (FSAR), minimum DNBR occurs 3.7 seconds after a partial loss of cooling occurs. This table considers the case in which one RCP begins coasting down at time 0. It is plausible to believe that DNBR rises at the time given in the stem because DNBR does rise after the reactor is tripped. The case given in the stem would cause an automatic reactor trip as one or more RCPs is secured with nuclear power greater than P-8 (48%). It is correct that a flywheel attached to each RCP acts to assure the minimum heat removal needed to prevent core damage after the reactor trip. This fact is observed in system description WBN-SDD-N3-68-4001, Reactor Coolant System.

B. Correct: DNBR would be LOWERING⇓ at the time specified in the stem of the question. Also, it is correct that the RCP flywheel serves the purpose asked by the question.

C. Incorrect: It is Incorrect and yet plausible that DNBR would be RISING⇑ at the time given in the stem. Additionally, it is Incorrect that core damage is prevented by maintaining the RCPs connected to the generator and external network for 30 seconds. Again, the RCS system description credits the flywheel of each RCP for this prevention. It is plausible to believe this because, as seen in the same system description, In the event of a turbine trip actuated by either the Reactor Trip System or the Turbine Protection System, the generator and RCPs are maintained connected to the external network for 30 seconds to ensure full flow before transfer is made. This protects the turbine/generator from overspeed upon tripping the turbine. Additionally, chapter 15 of the FSAR considers that a Loss of Offsite power occurs after the reactor trip (of the partial loss of flow accident). Therefore, the FSAR regards that all RCPs are lost immediately after the reactor trip.

D. Incorrect: While it is correct that the DNBR would be LOWERING⇓ at the time specified, it is not correct and yet plausible that the 30 second post trip connection of the grid, generator and RCPs would prevent core damage.

Question Number: 28

Tier: 2 Group: 1

K/A: 003 Reactor Coolant Pump System (RCPS)
K5 Knowledge of the operational implications of the following concepts as they apply to the RCPS:
K5.02 Effects of RCP coastdown on RCS parameters

Importance Rating: 2.8 3.2

10 CFR Part 55: (CFR: 41.5 / 45.7)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to understand the effects of the RCP coastdown on the DNBR. Also the applicant must understand what permits the removal of core decay heat during the RCP coastdown phase.

Technical Reference: Watts Bar Final Safety Analysis Report (FSAR), chapter 15
WBN-SDD-N3-68-4001, Reactor Coolant System.

Proposed references to be provided: None

Learning Objective: 3-OT-TS0303, TS 3.3 “INSTRUMENTATION”
1. DESCRIBE the following aspects of Technical Specifications and Technical Requirements for Instrumentation Tech Specs:
a. The Limiting Conditions for Operation, Applicability, and Bases for the LCO.
b. The conditions and required actions with completion time of one hour or less.

Cognitive Level:
- Higher X
- Lower ____

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History: New question for the 2015-301 NRC RO Exam

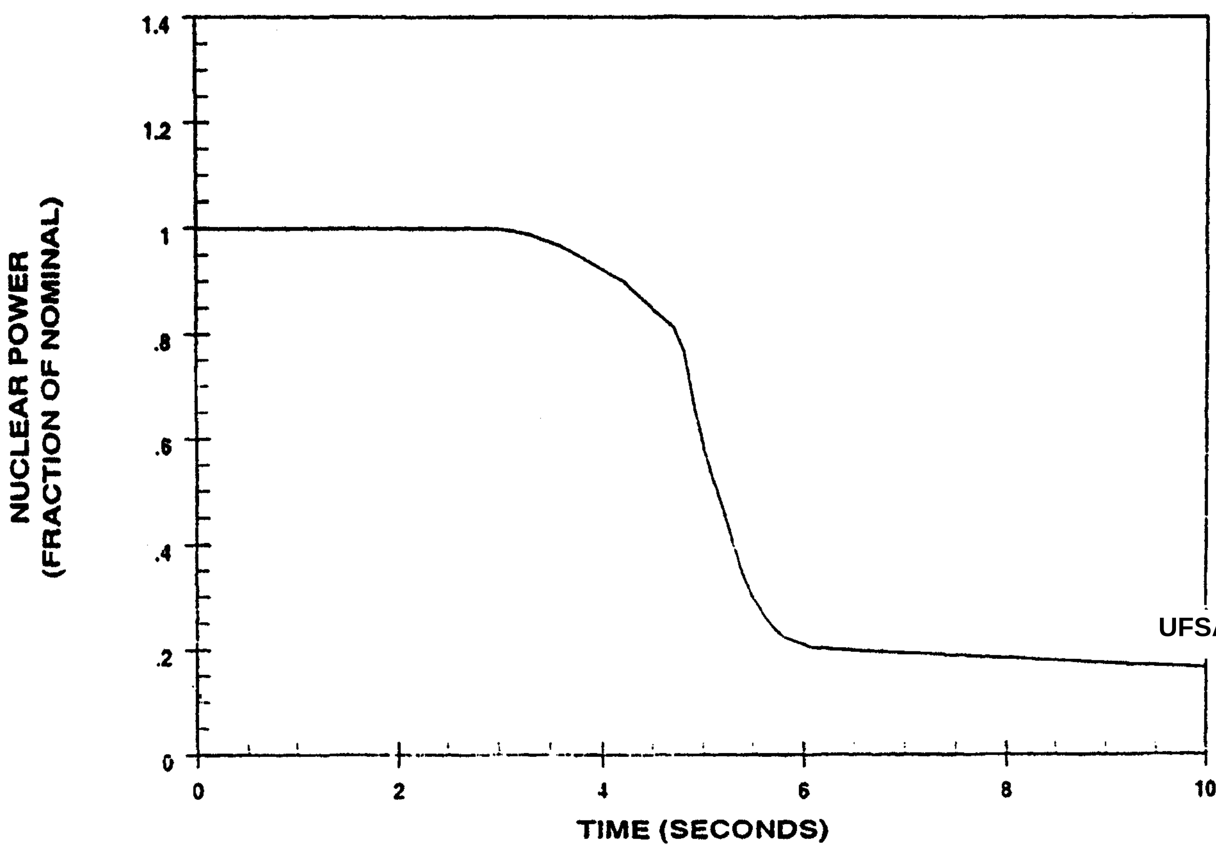
Comments:

TABLE 15.2-1
(Sheet 4 of 7)

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS (Cont'd)

Accident	Event	Time (sec.)
Partial Loss of Forced Reactor Coolant Flow (four loops operating, one pump coasting down)	One pump begins coasting down	0
	Low flow trip setpoint reached	1.32
	Rods begin to drop	2.52
	Minimum DNBR occurs	3.7
Loss of External Electrical Load		
1. With pressurizer control	Loss of electrical load	0
	OT deltaΔT reactor trip point reached	8.5
	Rods begin to drop	10.0
	Minimum DNBR occurs	11.6
2. Without pressurizer control	Loss of electrical load	0
	High pressurizer pressure reactor trip point reached	4.5
	Rods begin to drop	6.5
	Peak Pressurizer pressure occurs	7.9

CAD MAINTAINED DRAWING



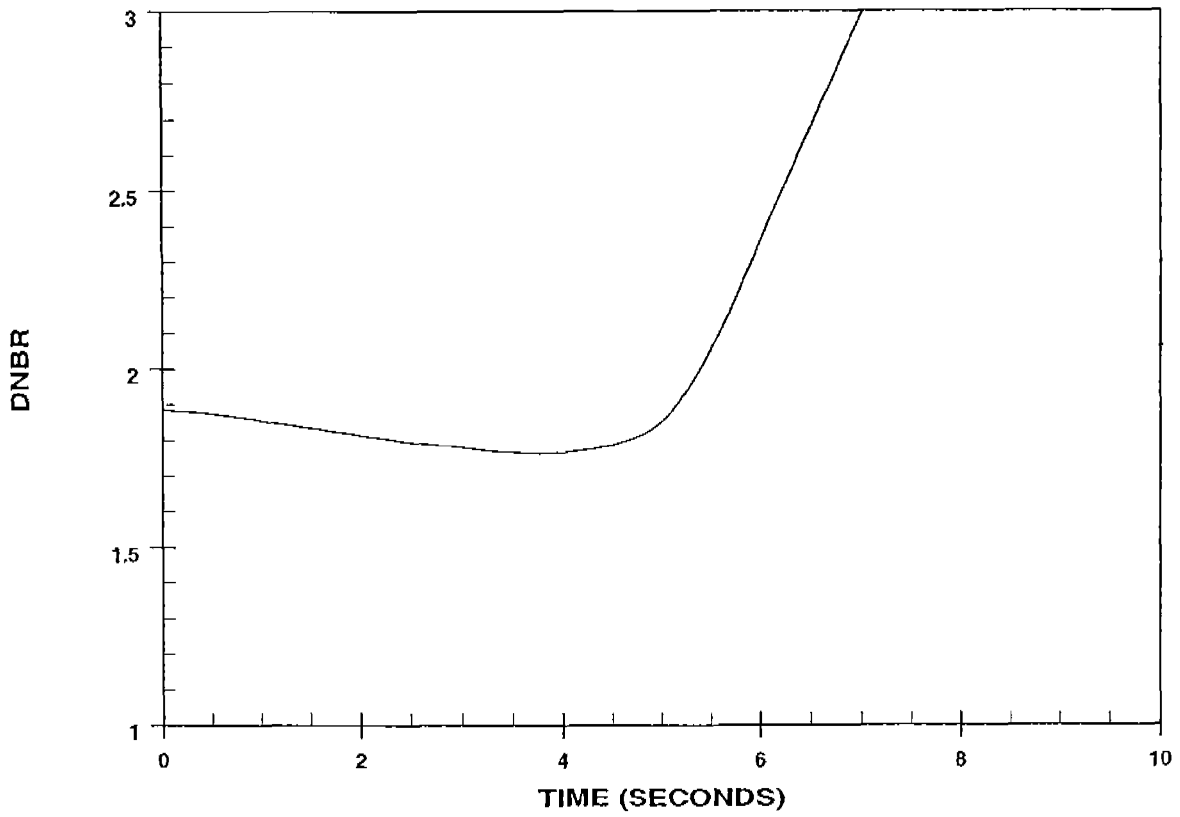
UFSAR AMENDMENT

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

NUCLEAR POWER TRANSIENT
FOUR PUMPS IN OPERATION
ONE PUMP COASTING DOWN

FIGURE 15.2-16

CAD MAINTAINED DRAWING



UFSAR AMENDMENT 2

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

DNBR VERSUS TIME
FOUR PUMPS IN OPERATION
ONE PUMP COASTING DOWN

FIGURE 15.2-17

3.2.3 Reactor Coolant Pumps (continued)

8. Flywheel

   The flywheel stores energy to keep the pump operating for a short but critical period following a power failure. It provides additional inertia to extend pump coast-down. The flywheel is available for inspection by removing the cover.

D. Oil Lift System

An external oil lift system provides lubrication to the upper radial bearings during motor startup. This oil lift system is shown in Figure 2. High-pressure oil is also delivered simultaneously to the upper and lower sets of thrust bearing shoes. The oil lift system will provide from 0.001" to 0.003" of oil film between the runner and the thrust bearing shoes depending on the amount of pump thrust on the bearings. Thus, regardless of thrust condition (either upthrust or downthrust), an oil film between the shoes and runner is present. At the same time, high-pressure oil is being sprayed into the guide bearing chamber for lubricating the upper radial guide bearings. There are low and high oil level alarms in the upper and lower oil reservoirs. The oil lift pumps have spray shields to ensure no lube oil comes in contact with hot reactor coolant in case of component rupture. The oil lift system is required to operate approximately two minutes before RCP startup and approximately one minute after the pump reaches full speed.

E. RCP Power

The RCPs are powered through the 6.9KV RCP boards. Undervoltage to any two of the four pumps will cause a reactor trip above approximately 10% reactor power. Underfrequency to any two of the four pumps will trip all four RCPs and cause a reactor trip above approximately 10% power.

3.2.4 Pressurizer (PZR)

The PZR provides a point in the RCS where liquid and vapor can be maintained in equilibrium under saturated conditions for RCS pressure control. The PZR and its sprays, heaters, PORVs, safety valves, and surge line make up the pressure control equipment. This equipment is designed to accommodate changes in system volume and to limit changes in system pressure due to RCS loop temperature variation during all modes of operation. In particular, the equipment is designed to provide satisfactory operation without reactor trip when the RCS is subjected to the following transients:

a. Loading or unloading at 5% of full power per minute, with automatic reactor control.

b. Instantaneous load transients of ±10% of full power (not exceeding full power), with automatic reactor control.

c. Step-load reduction of 50% of full power with automatic reactor control (10%) and steam dump (40%).

3.4.2 RCS Loop Instrumentation and Control (continued)

The ICCM microprocessor also provides input to a digital saturation meter (one for each train) at the main control board which provides a backup means to assess saturation temperature corresponding to RCS pressure from 0-3000 psig.

The plant process computer is also utilized to continuously monitor pressure and temperature margins to saturation of the RCS.

E. RCS Hot Leg Level Indication (Ref. 7.4.6)

The Westinghouse Ultrasonic Level Measurement System (ULMS) is designed to provide a highly accurate indication of water level in the RCS piping during plant shutdown. Various maintenance and inspection activities performed during plant shutdown require that RCS water level be lowered into the loop piping, often referred to as "mid-loop" operation. Operation above a minimum water level in the hot leg is always necessary to prevent air ingestion by the RHP Pumps, with the potential for a resulting loss of decay heat removal capability. The ULMS was specifically developed to provide the accurate indication of water level necessary during mid-loop operations, due to the narrow operating window and the potential severity of loss of decay heat removal events.

In addition to the ULMS, a differential pressure level transmitter and a liquid level gauge are provided as permanent backup instruments along with a Mansell Level Monitoring System as a temporary backup instrument to monitor the reactor vessel water level during mid-loop operations. A second D/P level transmitter provides wide range indication from the bottom of the hot leg to an elevation above the pressurizer lower level instrument tap.

3.4.3 Steam Generator Instrumentation and Control

See the Main Steam and Main FeedWater System description (Ref. 7.2.1 and 7.2.46).

3.4.4 RCPs Instrumentation and Control

A. RCP Power Supply

The four RCPs are connected to separate buses of the 6.9KV boards which normally receive power from the main generator. These boards receive emergency power from the 6.9KV start buses. The power supply to each RCP consists of a normally closed breaker and a normally open breaker which are electrically interlocked. If the normally closed breaker should sense a loss of power on its bus, it will open and the normally open breaker will close providing power from another bus. During plant startup and shutdown operations the four RCP buses receive power from an offsite source.

Complete loss of normal ac power results in reactor trip on RCP bus undervoltage or underfrequency and turbine trip (Ref. 7.2.44). The flywheel inertia of each RCP sustains RC flow for a period of time (1-1/2 minutes) after the trip sufficient to assure the minimum heat removal needed to prevent core damage after the trip. Natural circulation of the coolant provides sufficient flow for reactor core residual heat removal until power can be restored to the pumps.

3.4.4 RCPs Instrumentation and Control (continued)

In the event of a turbine trip actuated by either the Reactor Trip System or the Turbine Protection System, the generator and RCPs are maintained connected to the external network for 30 seconds to ensure full flow before transfer is made. This protects the turbine/generator from overspeed upon tripping the turbine. In case a generator trip deenergizes the pump buses, the RCP motors will be transferred to offsite power within 6 to 10 cycles. See Table 2 for RCP power supplies.

B. RCP Motor Oil Lift Pump

Each RCP has a normal and an alternate handswitch in the MCR for pump startup. These handswitches each have backup control handswitches and transfer switches. Both sets of controls require prior operation of the oil lift pump for approximately two minutes and are nondivisional.

Each oil lift pump contains its own handswitch and transfer switch. A pressure switch is provided on the high pressure oil lift system. Upon low oil pressure, the switch will actuate a MCR alarm. In addition, the switch is part of an interlock system that will prevent starting of the RCP until the oil lift pump is started manually prior to starting the RCP motor. A local pressure gauge is also provided. The control loop for these instruments is nondivisional.

C. RCP Motor Oil Level Detection

Each RCP contains a control loop for oil reservoir level indication. An annunciator for high or low oil level is located in the MCR. Each control loop contains two lower oil reservoir level indicators and two upper oil reservoir level indicators. All switches of the control loop are nondivisional.

29.

Given the following conditions:

- Unit 1 is at 40%.

Subsequently:

- The #1 RCP trips.

Which ONE of the following describes the effect the RCP trip will have on **INDICATED** secondary plant parameters?

SG #1 Steam Flow WILL **IMMEDIATELY** ____(1)____.

Assuming **NO OPERATOR ACTION** occurs:

FIVE MINUTES AFTER the RCP trip, Feed Flow WILL be provided to the SGs via _____(2)_____ feedwater.

	(1)	(2)
A.	RISE ⇑	AUX
B.	RISE ⇑	MAIN
C.	LOWER ⇓	AUX
D.	LOWER ⇓	MAIN

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: When a RCP trips, the flow in the RCS loop decreases, the temperature of the S/G lowers and due to the saturation conditions existent in the S/G, the pressure in the S/G will lower. As soon as the pressure in the S/G lowers to a value less than that of the other S/Gs, steam flow from the affected S/G will cease.

This portion of the distractor is plausible provided that the applicant does not understand the thermodynamic properties governing the partial loss of flow accident.

It is not correct that the plant would be provided AFW as a feed source. It is plausible to believe this as if the plant tripped, this would be the case. The conditions in the stem indicate that 3 of 4 PR NI channels are less than or equal to 48%. This may be seen in the Response instruction for annunciator 70-C. Also seen in this instruction is that the single loop loss of flow Rx trip is blocked below 48%.

If the applicant did not recognize that power was below the P-8 setpoint, they would believe that the plant would trip on the loss of the RCP and that AFW would be feeding the S/Gs (as MFW would isolate post trip).

B. Incorrect: Again, it is incorrect and yet plausible that steam flow would rise. Again, as the plant does not trip, MFW continues to supply the S/Gs.

C. Incorrect: It is true that steam flow would lower. However, it is not true and yet plausible that AFW supplies the S/Gs.

D. Correct: It is true that steam flow would lower. It is true that MFW continues to feed the S/Gs.

Question Number: 29

Tier: 2 Group: 1

K/A: 003 Reactor Coolant Pump System (RCPS)
K5 Knowledge of the operational implications of the following concepts as they apply to the RCPS:
K5.04 Effects of RCP shutdown on secondary parameters, such as steam pressure, steam flow, and feed flow

Importance Rating: 3.2 3.5

10 CFR Part 55: (CFR: 41.5 / 45.7)

10CFR55.43.b: Not applicable

K/A Match: This question matches the K/A by determining if the candidate knows the effect an RCP trip will have on SG parameters (steam flow and SG level) when the plant is at power.

Technical Reference: FSAR Ch 15, section 15.2.5.1
Westinghouse Transient Accident Analysis
1-AOI-24, RCP Malfunctions During Pump Operation, Revision 0029

Proposed references to be provided: None

Learning Objective: 3-OT-TAA015
1. Discuss loss of coolant flow theory
   a. Derive the time dependent behavior of T_h
   b. Define DNR and DNBR

3-OT-AOI2400
12. Describe basic operator actions to shutdown an RCP.

Cognitive Level:
- Higher X
- Lower ____

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History:

Comments:

70-C

Source	Setpoint
NC41N	3/4 PR Channels less than or equal to 48%
NC42N	
NC43N	
NC44N	

P-8
LO PWR-FLOW
TRIP BLOCKED

(Page 1 of 1)

Probable Cause:
A. Power reduction
B. Unit trip

Corrective Action:
[1] **VERIFY** Reactor power level less than or equal to 48%.

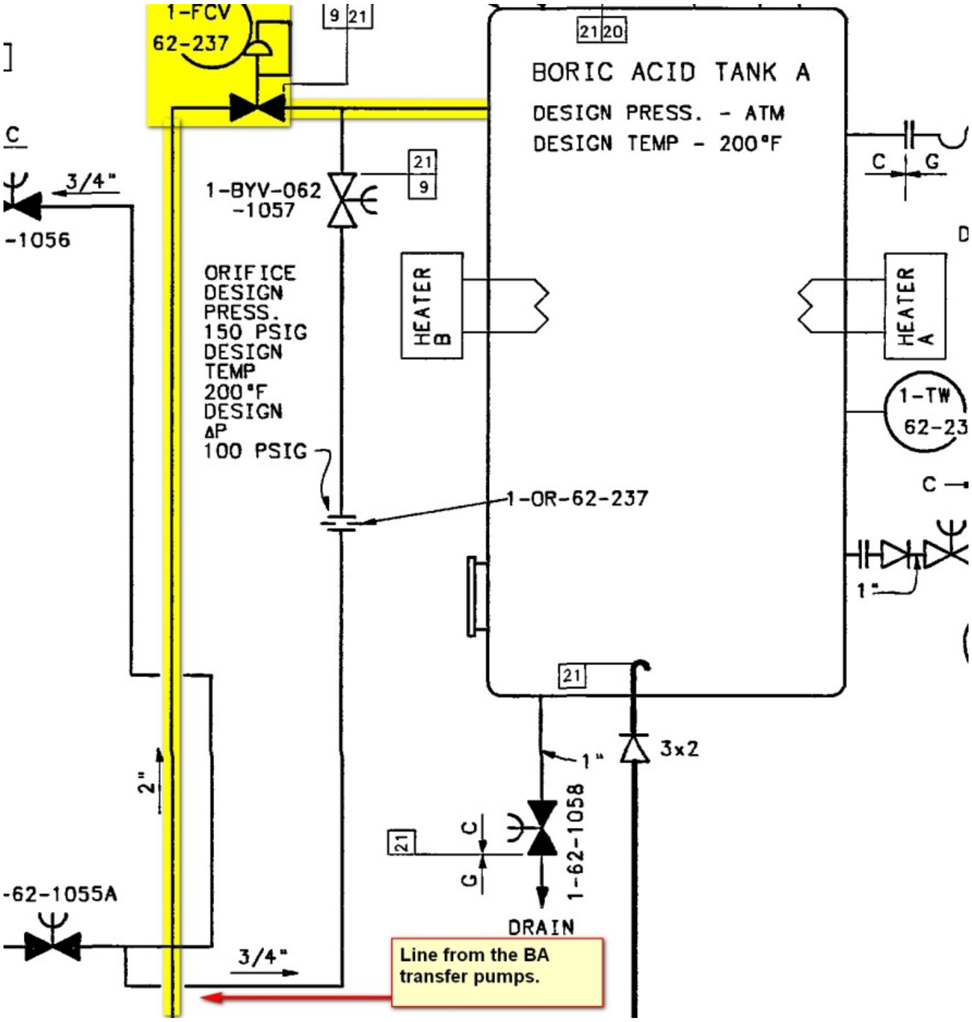
NOTES

1) P-8 illuminated blocks single loop loss of flow Rx trip.
2) P-8 will automatically reset with 2/4 PR channels greater than 48%.

References: W 7246D11 Sht. 2, 5, 8, 11, & 40

30.

Given the following print excerpt:



Which ONE of the following states the purpose of the flowpath highlighted in yellow in the print excerpt shown above?

The highlighted line __________.

A. automatically maintains level in the Boric Acid Tank (BAT) “A” through the automatic action of 1-FCV-62-237, “BAT A RECIRC CONTROL.”

B. is used to manually makeup to the BAT A in accordance with SOI-62.05, “Boric Acid Batching, Transfer, And Storage.”

C. provides the required chemical mixing of the BAT A as 1-FCV-62-237 is normally THROTTLED OPEN to achieve a flow rate specified by chemistry.

D. provides miniflow protection to the Boric Acid Transfer Pumps as 1-FCV-62-237 is normally THROTTLED OPEN to 25%.

<u>CORRECT ANSWER:</u> <u>B</u>

<u>DISTRACTOR ANALYSIS:</u>

A. Incorrect: As seen in SSD-1-LPF-62-237, 1-FCV-62-237 is only controlled by a manual controller. There are no automatic functions associated with this valve. Therefore, it is incorrect that the level in BAT "A" will be controlled by the automatic action of this valve. It is plausible to believe that the BAT "A" has an automatic level control because other tanks in the plant do have such (including those which are part of the CVCS and which do affect reactivity---e.g. the VCT).

B. Correct: As seen in 0- SOI-62.05, "Boric Acid Batching, Transfer, And Storage" 1-FCV-62-237 is manually controlled to raise level in the BAT "A."

C. Incorrect: As the name of 1-FCV-62-237 implies ("BAT A RECIRC CONTROL"), it is used to recirculate the BAT "A." However, as seen in the system description, WBN-SDD-N3-62-4001, "Chemical and Volume Control System," the continuous recirculation of the tank is afforded by the ¾" line which includes the normally open valve 1-BYV-62-1057 and orifice 1-OR-62-237. It is plausible to believe that 1-FCV-62-237 would be left open for chemical mixing because after a makeup to BAT "A," the operators will recirculate the tank using 1-FCV-62-237 to satisfy the Chemistry department's requirements. However, after Chemistry obtains their required samples, 1-FCV-62-237 will be closed.

D. Incorrect: Again, 1-FCV-62-237 is a normally closed valve. It is plausible to believe that it provides miniflow protection to the BATPs because if it were left throttled open, it would do so.

Question Number: 30

Tier: 2 Group: 1

K/A: 004 Chemical and Volume Control System (CVCS)
K5 Knowledge of the operational implications of the following concepts as they apply to the CVCS:
K5.31 Purpose of flow path around boric acid storage tank

Importance Rating: 3.0 3.4

10 CFR Part 55: (CFR: 41.5/45.7)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to demonstrate the knowledge of the purpose for the flow path around the BAT.

Technical Reference: Print 1-47W809-5
WBN-SDD-N3-62-4001, Chemical and Volume Control System
0-SOI-62.05, Boric Acid Batching, Transfer, And Storage
SSD-1-LPF-62-237

Proposed references to be provided: None

Learning Objective: 3-OT-STG-062A, CHEMICAL AND VOLUME CONTROL SYSTEM
6. EXPLAIN the CVCS design features and/or interlocks that provide the following:
i. RCS Boron concentration control and modes of operation of the Reactor Makeup Control System (blender controls)
q. Boron crystallization prevention
s. Chemical injection

Cognitive Level:
Higher ____
Lower X

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

3.2.7 Boric Acid Blender (continued)

The blenders were manufactured by W. They are austenitic stainless steel and consists of a conventional pipe tee. The blenders are provided with water inlet, boric acid inlet, and discharge connections.

The boric acid blenders (W Spin No. CSHDBL) were purchased from W under contract No. 71C60-54114-1.

3.2.8 Orifices

A. Boric Acid Transfer Pump Bypass Orifices

This describes 1-OR-62-237.

There are three pump bypass orifices (one per BAT) located in the auxiliary building. Each orifice is designed to pass the minimum flow required for sufficient recirculation through the piping and tanks with the boric acid transfer pumps. The design operation requirements are listed in Table 11. The orifices normally operate at 100 psig and ambient temperature.

The orifices were supplied by W. The boric acid transfer pump bypass orifices (W Spin No. CSORBA) were purchased from W under contract No. 71C60-54114-1.

B. CCP Miniflow Orifices

There are two CCP miniflow orifices (one per CCP) located in the auxiliary building. These orifices provide a minimum flow for pump protection. The design operating requirements are listed in Table 11. The orifices are shown on Drawing MF-48590 in the CCP manual (Ref. 7.3.9). These orifices (W Spin No. CSORCP) were purchased from W under contract No. 71C62-54114-1.

C. Chemical Mixing Tank Orifices

There is one chemical mixing tank orifice located in the auxiliary building (El. 713). An orifice is provided in the piping upstream of the mixing tank. This orifice limits the flow rate through the tank to 2 gpm to avoid slugging the pump seals with concentrated chemicals. The design operating requirements are listed in Table 11. The orifices are normally operated at ambient temperatures.

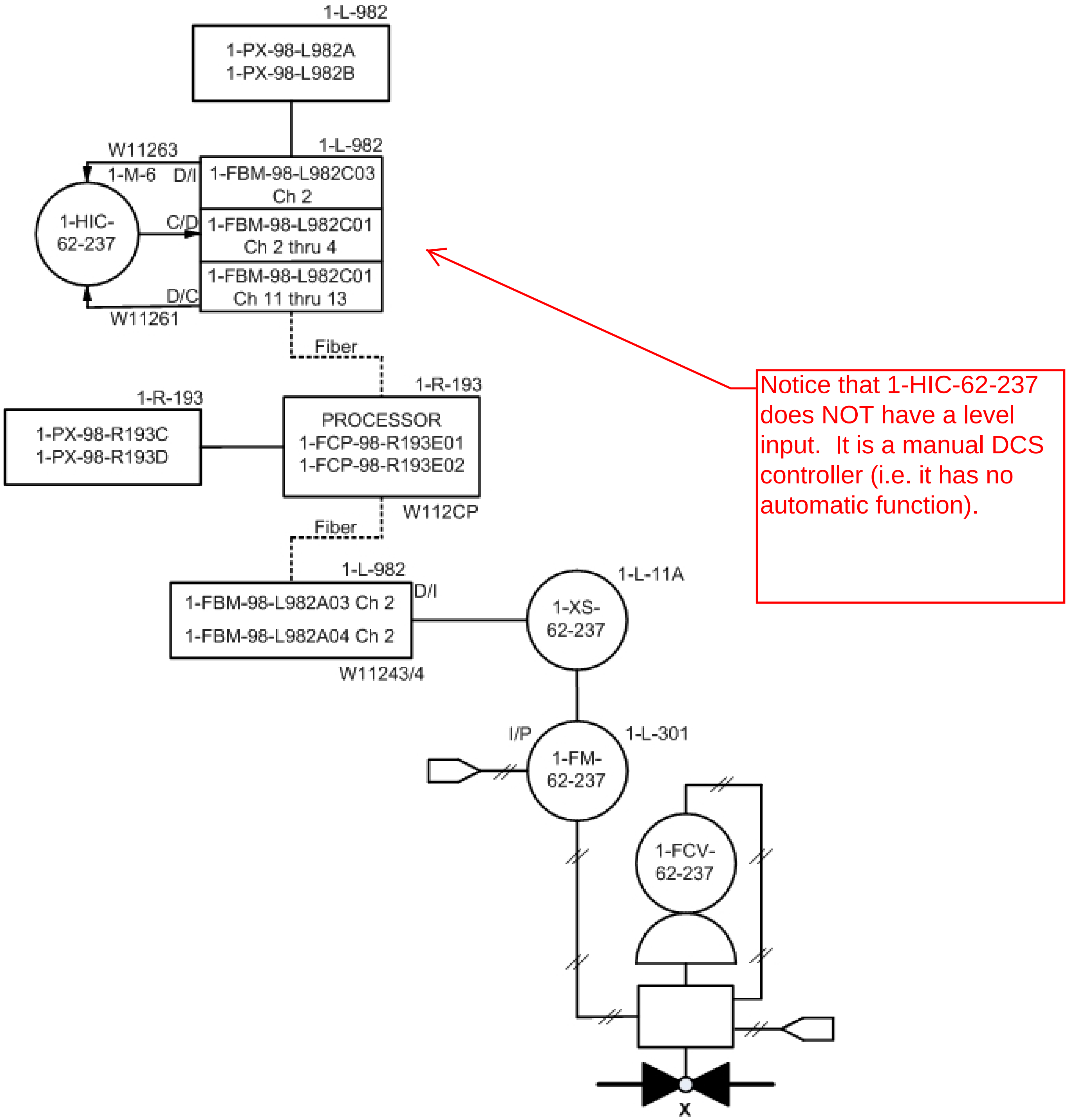
The chemical mixing tank orifices (W Spin No. CSORCM) were purchased from W under contract No. 71C60-54114-1. They are shown on W drawing 271C403 (Ref. 7.3.19).

D. Letdown Orifices

There are four letdown orifices located in the reactor building. Two 75 gpm orifices and one 45 gpm orifice are provided to reduce letdown pressure from RCS conditions and to control the letdown flow. An approximately 3 gpm orifice is provided to limit the rate of thermal change on the Class A welds upstream of the regenerative HX.

SCALING AND SETPOINT DOCUMENT
LOOP DRAWING

Loop Name: Boric Acid Tank A Recirc Flow



Reviewed By: R. J. HUNT Approved By: R.E. McSpadden Date: RF012

Date________ **Initials**

6.3 Transfer BA from BAT C To BAT A

This is the section which a crew uses to makeup to BAT A.

NOTE
This Section is for BAT A in service to U1 via BA Pump 1A.

[1] **IF** BAT A in service to U1, **THEN**

[1.1] **MOMENTARILY PLACE** 1-HS-62-140A, VCT MAKEUP CONTROL [1-M-6], to STOP and RELEASE. ________

[1.2] **CHECK** 1-HS-62-140A, Green light LIT. ________

[2] **RECORD** initial BAT A LEVEL from 1-LI-62-238 [1-M-6]________gals. ________

[3] **RECORD** initial BAT C LEVEL from 1-LI-62-242 [1-M-6]________gals. ________

[4] **REQUEST** SRO to evaluate TR 3.1.1/3.1.2 prior to stopping BA pumps. ________

[5] **PERFORM** the following:

NOMENCLATURE	LOCATION	POSITION	UNID	PERF INITIAL
BA PMP A	1-M-6	STOP	1-HS-62-230A	
BA PMP B	1-M-6	STOP	1-HS-62-232A	

[6] **OPEN** 1-ISV-62-1053B, BA XFER PUMP 1B-B DISCHARGE [A12R/713]. ________

[7] **START** BA Pump 1B, using 1-HS-62-232A, BA PMP B [1-M-6].

During makeup to BAT A, 1-FCV-62-237 is throttled OPEN.

[8] **IF** desired, **THEN**

PLACE BA pump 1B in FAST using 1-HS-62-232D. ________

[9] **THROTTLE** 1-FCV-62-237, BAT A RECIRC CONTROL [1-M-6], as desired to control rate of transfer. ________

Date________ **Initials**

6.3 Transfer BA from BAT C To BAT A (continued)

CAUTION

Damage may occur to BAT Heaters if level drops below 9.5% with heaters ENERGIZED.

[10] **IF** BAT C level decreases to 2270 gals (10%), **THEN**

ENSURE the following:

NOMENCLATURE	LOCATION	POSITION	UNID	PERF INITIAL
BORIC ACID TANK C HEATER A	A13R/713	OFF	0-HS-62-243	
BORIC ACID TANK C HEATER B	A12R/713	OFF	0-HS-62-246	

[11] **IF** BAT is to be drained below heaters, **THEN**

DRAIN and FLUSH completely to prevent precipitation.

1-FCV-62-237 is left closed following a makeup to BAT A. This valve is normally CLOSED.

[12] **WHEN** desired level is obtained, **THEN**

STOP BA Pump 1B, using 1-HS-62-232A, BA PMP B. _________

[13] **CLOSE** 1-FCV-62-237, BAT A RECIRC CONTROL.[C.2] _________

_________ IV

[14] **CLOSE** 1-ISV-62-1053B, BA XFER PUMP 1B-B DISCHARGE.[C.2] _________

_________ IV

[15] **START** BA Pump 1A with 1-HS-62-230A, BA PMP A. _________

[16] **ENSURE** BA pump 1B in SLOW using 1-HS-62-232D. _________

[17] **START** BA Pump 1B with 1-HS-62-232A, BA PMP B (**N/A** if BAT C removed from service). _________

[18] **RECORD** final BAT A LEVEL from 1-LI-62-238_________gals. _________

[19] **RECORD** final BAT C LEVEL from 1-LI-62-242. _________gals. _________

Date________ **Initials**

6.3 Transfer BA from BAT C To BAT A (continued)

[20] **REQUEST** Chemistry Lab sample BAT A. _________

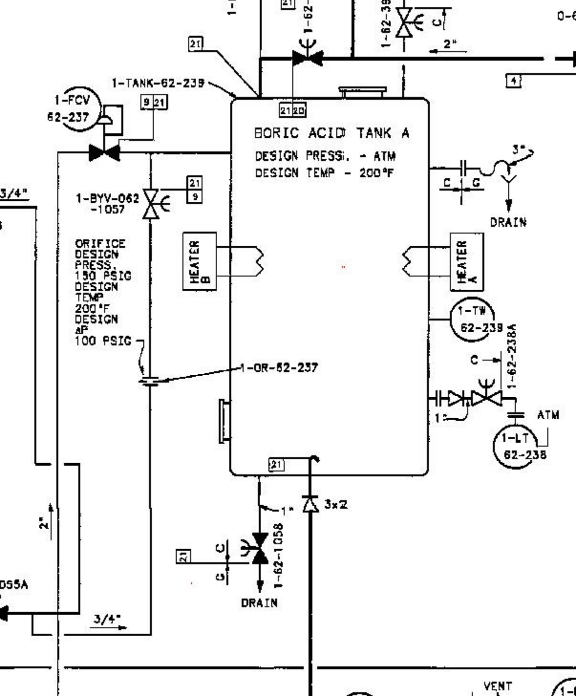
[21] **IF** 1-HS-62-140A, VCT MAKEUP CONTROL, was positioned to STOP in Step 6.3[1], **THEN**

[21.1] **MOMENTARILY PLACE** 1-HS-62-140A, VCT MAKEUP CONTROL [1-M-6], to START and RELEASE. _________

[21.2] **CHECK** 1-HS-62-140A, Red light LIT. _________

[22] **REQUEST** SRO to verify requirements of TR 3.1.5 or 3.1.6 met. _________

End of Section



BORIC ACID TANK A
DESIGN PRESS. - ATM
DESIGN TEMP - 200°F
1-TANK-62-239
1-FCV 62-237
1-BYV-062 -1057
ORIFICE DESIGN PRESS. 150 PSIG DESIGN TEMP 200°F DESIGN ΔP 100 PSIG
1-OR-62-237
HEATER B
HEATER A
1-TW 62-239
1-62-238A
1-LT 62-238
ATM
DRAIN
1-62-1058
DRAIN
3x2
VENT
3/4"
2"
3"
1"
21
9
20
4

31.

Which ONE of the following describes the operation of the motor operated mini-flow valves; in regards to the RHR system flow rates, in accordance with 1-SOI-74.01, Residual Heat Removal System?

The motor operated mini-flow valve will be **OPENED** if RHR pump flow falls below ____(1)____ gpm AND will be **CLOSED** when the flow INCREASES above ____(2)____ gpm.

A. (1) 250
(2) 1100

B. (1) 250
(2) 1400

C. (1) 750
(2) 1100

D. (1) 750
(2) 1400

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: As seen in Appendix A, RHR Pump Operating Flow Limits of 1-SOI-74.01, the minimum flow operating of the RHR pumps is 750-1400 gpm. These setpoints were modified upon recommendation from Westinghouse from their original values of 250-1100 gpm. The modification occurred after two RHR pumps were seized and analysis from Westinghouse concluded that low mini-flow setpoints were subjecting the RHR pumps to a temperature swing in excess of 200ºF. When the analysis for the Fire Safe Shutdown was performed, the setpoints for the RHR pump mini-flow was 250-1100 gpm. Therefore, as seen on page III-16 of PART III – SAFE SHUTDOWN CAPABILITIES of the Fire Protection Report, the setpoints are listed as 250-1100 gpm. Therefore, it is plausible to believe that the mini-flow setpoints are 250-1100 gpm.

B. Incorrect: Again it is Incorrect and yet plausible that the low setpoint for the RHR pump mini-flow valve be 250 gpm. It is correct that the high setpoint for such valve is 1400 gpm.

C. Incorrect: It is correct that the low setpoint for the RHR pump mini-flow valve be 750 gpm. It is Incorrect and yet plausible that the high setpoint be 1100 gpm.

D. Correct: It is correct that the RHR pump mini-flow setpoints are 750-1400 gpm.

Question Number: 31

Tier: 2 Group: 1

K/A: 005 Residual Heat Removal System (RHRS)
A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including:
A1.02 RHR flow rate

Importance Rating: 3.3 3.4

10 CFR Part 55: (CFR: 41.5 / 45.5)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to demonstrate the knowledge of the setpoints for RHR mini-flow operation. The applicant must possess this knowledge to be able to monitor for changes in RHR flow rate to prevent exceeding design limits (specifically, RHR pump overheating).

Technical Reference: 1-SOI-74.01, Residual Heat Removal System
PART III – SAFE SHUTDOWN CAPABILITIES of the Fire Protection Report

Proposed references to be provided: None

Learning Objective: 3-OT-STG-074, THE RESIDUAL HEAT REMOVAL SYSTEM
6. EXPLAIN the RHR System design features and/or interlocks of the following:
c. Function of RHR pump miniflow recirculation

Cognitive Level:
Higher ____
Lower X

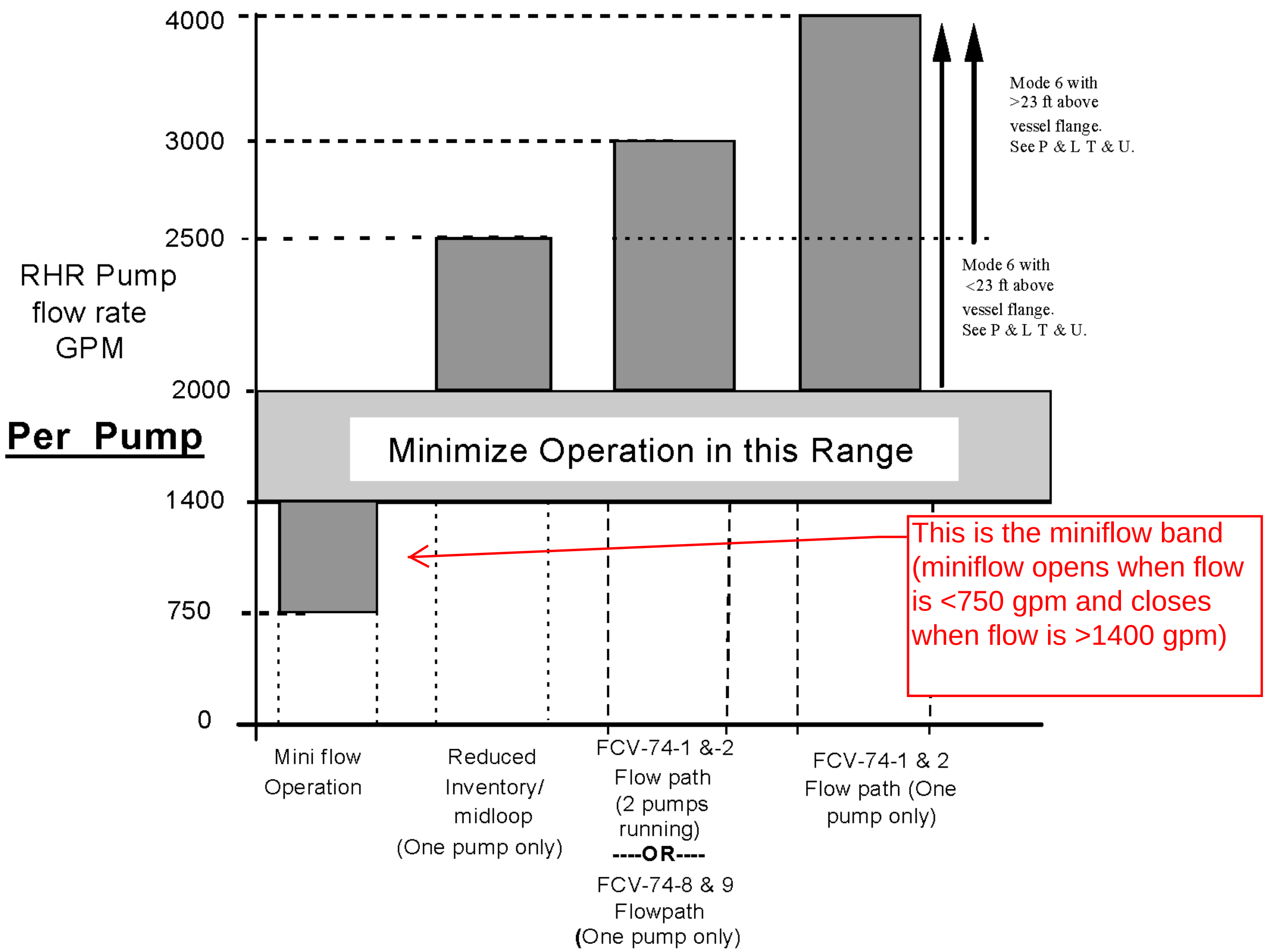
Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Appendix A
(Page 1 of 1)

RHR Pump Operating Flow Limits



A minimum-flow return line from the downstream side of each RHR heat exchanger to the corresponding pump's suction line is provided to assure that the RHR pumps do not overheat under low flow conditions. A motor-operated valve located in each minimum flow line will be opened if RHR pump flow falls below 250 gpm and will be closed when the flow increases above 1100 gpm.

The cooldown rate of the reactor coolant is controlled by regulating the flow through the tube side of the RHR heat exchangers. A bypass line, which serves both residual heat exchangers, is used to regulate the temperature of the return flow to the RCS as well as to maintain a constant flow through the RHR system.

The RHR system can be placed in operation when the pressure and temperature of the RCS are about 400 psig and 350^{0}F, respectively. If one of the pumps and/or one of the heat exchangers is not operable, safe operation of the plant is not affected; however, the time for cooldown is extended.

These were the old setpoints.

4.5.1 Residual Heat Removal Pumps

Two identical pumps are installed in the RHR system. Each pump is sized to deliver sufficient reactor coolant flow through the residual heat exchangers to meet the plant cooldown requirements. A seal heat exchanger for each pump is supported by operation of CCS.

4.5.2 RHR Safety Valves

The RHR system safety valves provide RCS cold overpressure protection whenever the RHR system is in operation. The valves are located inside containment, one each on the RHR system suction and discharge path, and discharge to the pressurizer relief tank. The valves are set at 450 psig and 600 psig, respectively.

4.6 Safety Injection System Accumulators - Key 36

During normal plant operating conditions, the safety injection system accumulators are pressurized by nitrogen gas in order to inject borated water in the RCS when RCS pressure falls below 600 psi unintentionally. During a controlled depressurization, the accumulators are isolated to prevent injection of safety injection system accumulator borated water. However, if the isolation valves remain open, the borated water will be injected and nitrogen pressure will decrease due to the increased empty volume of the accumulators. Injection of nitrogen into the RCS occurs when RCS pressure is less than 150 psi.

The manual isolation of the accumulators is assumed as a post-fire activity. The isolation valve at each accumulator is closed only when the RCS is intentionally depressurized below 1000 psig. If the cables associated with these valves were damaged by fire, isolation is performed locally, governed by appropriate plant procedures (post-fire). In the event the valves are inaccessible, RCS pressure will be maintained greater than 150 psi to preclude nitrogen injection into the RCS via the accumulators.

32.

Given the following conditions:

- Unit 1 is at 100% power.

Which ONE of the following describes the physical location of 1-BKR-74-2A, LOOP 4 HOT LEG TO RHR SUCTION (1-FCV-74-2) AND the NORMAL position of this breaker?

1-BKR-74-2A is located on a____(1)____ board AND is required to be ____(2)____.

A. (1) Rx MOV
(2) ON

B. (1) Rx MOV
(2) OFF & Locked

C. (1) C&A Vent
(2) ON

D. (1) C&A Vent
(2) OFF & Locked

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: It is correct that the breaker listed is located on a Rx MOV board.

It is also not correct that 1-BKR-74-2A be left ON or shut. As seen in 0-PI-OPS-17.1, 18 month Locked Breaker Verification, this breaker is positioned OFF (for Appendix R concerns). It is plausible to believe that this breaker would be left on because many MOVs have their supplies left ON. Also, circuit interlocks exist which prevent this valve from opening given that RCS parameters are not within limits. Finally, administrative restrictions exist which prohibit manipulating this breaker's valve.

B. Correct: The breaker listed is located on a Rx MOV board. Also, it is locked OFF.

C. Incorrect: As seen on both 1-SOI-74.01 ATT 1P, Residual Heat Removal System Power Checklist 1-SOI-74.01 ATT 1P, 1-BKR-74-2A resides on the 480V Reactor MOV Board 1B1-B. It is plausible to believe that this breaker were to reside on a C&A vent board because there are safety related MOVs which do have there supply breakers located on a C&A vent board.

Again, it is incorrect and yet plausible that this breaker would be left ON.

D. Incorrect: Again it is Incorrect and yet plausible that the breaker listed reside on a C&A vent board. Also, it is correct that the breaker listed be locked OFF.

Question Number: 32

Tier: 2 Group: 1

K/A: 005 Residual Heat Removal System (RHRS)
K2 Knowledge of bus power supplies to the following:
K2.03 RCS pressure boundary motor-operated valves

Importance Rating: 2.7 2.8

10 CFR Part 55: (CFR: 41.7)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to correctly identify on which board the supply breaker for a RHR pressure boundary valve is located (1-BKR-74-2A). The applicant must further identify the locking requirement applicable to this bus power supply breaker.

Technical Reference: 1-SOI-74.01 ATT 1P, Residual Heat Removal System Power Checklist
0-PI-OPS-17.1, 18 month Locked Breaker Verification

Proposed references to be provided: None

Learning Objective: 3-OT-STG-074, THE RESIDUAL HEAT REMOVAL SYSTEM
5. LIST the bus power supplies to the following RHR System components:
b. RHR MOVs 1-FCV-74-1, 2, 8, 9, 3, 21

Cognitive Level:

Higher	
Lower	X

Question Source:

New	X
Modified Bank	
Bank	

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Performed On _________

NOMENCLATURE	LOCATION	POSITION	UNID	PERF INITIAL	VERIFIER INITIAL
480V Reactor MOV Board 1A1-A					
LOOP 4 HOT LEG TO RHR SUCTION (1-FCV-74-1)	C/10D	ON	1-BKR-74-1B		CV
LOOP 4 HOT LEG TO RHR SUCTION (1-FCV-74-1)	C/5B	0-PI-OPS-17.1	1-BKR-74-1A		
480V Reactor MOV Board 1A2-A					
1-FCV-74-2 BYPASS RHR SUCTION (1-FCV-74-8)	C/6D	0-PI-OPS-17.1	1-BKR-74-8		
480V Reactor MOV Board 1B1-B					
LOOP 4 HOT LEG TO RHR SUCTION (1-FCV-74-2)	C/5B	0-PI-OPS-17.1	1-BKR-74-2A		
LOOP 4 HOT LEG TO RHR SUCTION (1-FCV-74-2)	C/10D	ON	1-BKR-74-2B		CV
1-FCV-74-1 BYPASS RHR SUCTION (1-FCV-74-9)	C/5C	0-PI-OPS-17.1	1-BKR-74-9		

Appendix A
(Page 2 of 5)

Appendix A Locked Breaker List					Date________	
Nomenclature	**Location Board / Compartment**		**Position**	**UNID**	**Perf Initial**	**Verif Initial (CV)**
ERCW STRAINER 1A-A ISOL (1-FCV-67-22)	Rx MOV Bd 1A2-A	7B	OFF♦ & Locked	1-BKR-067-22		
AB ERCW SUP HDR 1A ISOL (1-FCV-67-81)	Rx MOV Bd 1A2-A	8A	OFF♦ (2) & Locked	1-BKR-067-81		
AB AIR CLR ERCW SUP HDR 1A ISOL (1-FCV-67-127)	Rx MOV Bd 1A2-A	10B	OFF♦ (2) & Locked	1-BKR-067-127		
CCS HX C HDR 1A SUPPLY (1-FCV-67-147)	Rx MOV Bd 1A2-A	11B	OFF (2) & Locked	1-BKR-067-147B		
CCS HX C OUT ERCW HDR A FLOW CNTL (0-FCV-67-151)	Rx MOV Bd 1A2-A	12A	OFF♦ & Locked	0-BKR-067-151		
CCS HX C HDR 1B SUPPLY (1-FCV-67-223)	Rx MOV Bd 1A2-A	13A	OFF♦ (2) & Locked	1-BKR-067-223B		
1A ESF EQUIP CCS SUP HDR (1-FCV-70-2)	Rx MOV Bd 1A2-A	14A	OFF♦ (2) & Locked	1-BKR-070-2		
MISC EQUIP CCS SUP HDR (1-FCV-70-4)	Rx MOV Bd 1A2-A	14B	OFF♦ (2) & Locked	1-BKR-070-4		
CCS HX A OUTLET (1-FCV-70-8)	Rx MOV Bd 1A2-A	14D	OFF♦ (2) & Locked	1-BKR-070-8		
CCS HX A/C OUTLET XTIE (1-FCV-70-10)	Rx MOV Bd 1A2-A	14E	OFF♦ (2) & Locked	1-BKR-070-10		
CCS HX A/C INLET XTIE (1-FCV-70-23)	Rx MOV Bd 1A2-A	15B	OFF♦ (2) & Locked	1-BKR-070-23		
CCS HX A INLET (1-FCV-70-25)	Rx MOV Bd 1A2-A	15D	OFF♦ (2) & Locked	1-BKR-070-25		
CCS HX A HDR 1B ERCW SUP (1-FCV-67-458)	Rx MOV Bd 1A2-A	15E	OFF♦ (2) & Locked	1-BKR-067-458		
CCP MIN FLOW VALVE (1-FCV-62-99) (SHUNT TRIP BKR)	Rx MOV Bd 1B1-B	2E2	OFF & Locked	1-BKR-062-99A		
SIP COLD LEG INJECTION (1-FCV-63-22)	Rx MOV Bd 1B1-B	2F2	OFF & Locked	1-BKR-63-22A		
SIS CL ACCUM 2 OUT ISOL (1-FCV-63-98)	Rx MOV Bd 1B1-B	3F2	OFF & Locked	1-BKR-63-98A		
LOOP 4 HOT LEG TO RHR SUCTION (1-FCV-74-2)	Rx MOV Bd 1B1-B	5B	OFF ♦ (1) & Locked	1-BKR-074-2A		
1-FCV-74-1 BYPASS RHR SUCTION (1-FCV-74-9)	Rx MOV Bd 1B1-B	5C	OFF ♦ (1) & Locked	1-BKR-074-9		

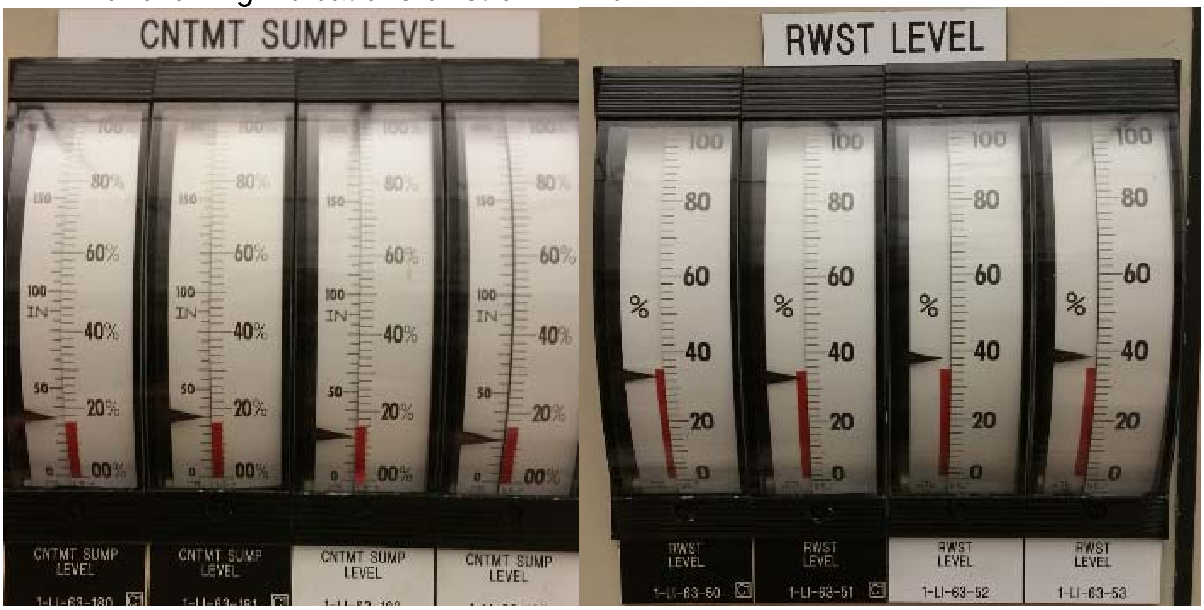
(1) Use padlock in place of seals
(2) ACB may be ON to position valve, but MUST be OFF & Locked during normal operation (Appendix R).
(♦) Positions marked with an ♦ are identified/required by NE to be in stated position.

Remarks: ______________________________

33.

Given the following conditions:

- Unit 1 has suffered a large break LOCA.
- The following indications exist on 1-M-6:



Based on these conditions, it is expected that transfer to the containment sump WILL ______________.

A. occur, due to all required conditions having been met

B. **NOT** occur due to RWST level being too high

C. **NOT** occur due to containment sump level being too low

D. **NOT** occur due to an SI signal still being supplied to the RHR sump valves

CORRECT ANSWER: A

DISTRACTOR ANALYSIS:

A. Correct: RWST level, containment sump level and the presence of the SI signal satisfy all conditions for the transfer to the containment sump signal to occur. The required logic of 2 of 4 channels is satisfied.

B. Incorrect: RWST level is less than the required 34.63% (on 2 of 4 channels) to satisfy the logic for transfer to the containment sump.

C. Incorrect: Containment sump level is greater than the required 16.1% (on 2 of 4 channels) to satisfy the logic for transfer to the containment sump.

D. Incorrect: Plausible, since an SI signal is in the logic for the automatic transfer to the containment sump and the SI reset and the transfer to the containment sump reset due to SI are separate.

Question Number: 33

Tier: 2 Group: 1

K/A: 006 Emergency Core Cooling System (ECCS)
A3 Ability to monitor automatic operation of the ECCS, including:
A3.08 Automatic transfer of ECCS flow paths

Importance Rating: 4.2 4.3

10 CFR Part 55: (CFR: 41.7 / 45.5)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to recognize the point at which the RHR pump suction transfer will occur.

Technical Reference: NE SSD 1-L-63-50
NE SSD 1-L-63-180
1-47W611-63-2
1-47W611-63-5

Proposed references to be provided: None

Learning Objective: 3-OT-STG-074, THE RESIDUAL HEAT REMOVAL SYSTEM
4. EXPLAIN, the physical connections and/or cause-effect relationships between
the RHR System and the following systems:
b. SIS
d. RWST

Cognitive Level:
Higher X
Lower ____

Question Source:
New ____
Modified Bank X
Bank ____

Question History: Modified from question 006 A4.05 32 used on the 5/2009 WBN NRC EXAM

Comments:

QUESTIONS REPORT
for SQN-WBN composite exams

1. 006 A4.05 032

Given the following plant conditions:

- Safety Injection (SI) is actuated.
- RWST level is 32%.
- Containment sump level is 18%.

Based on these conditions, it is expected that sump swapover will ______________.

A. NOT occur since containment sump level is too low.

B. NOT occur because an SI signal is still present to the RHR sump valves.

C. NOT occur since RWST level is too high.

D.✔ occur, since all required conditions have been met.

LOOP ACCURACY

Comparator Functions	Safety Analysis Limit	Setpoint	Tech Spec Allowable Value
Auto Recirc Low Interlock & Alarm 1-LS-63-50A-D (LB-913A)	181.0" / 134.6" (UAL/LAL) above tank floor 40.57% / 28.58% span (1)	158.0" above tank floor 34.63% span (9)	155.6" above tank floor 34.0% span (8)
Lo-Lo Alarm 1-XA-55-6C-126D (15-77)	71.8" / 33.0" (UAL/LAL) above tank floor 12.35% / 2.33% span (1)	52.4" above tank floor 7.34% span (9)	N/A

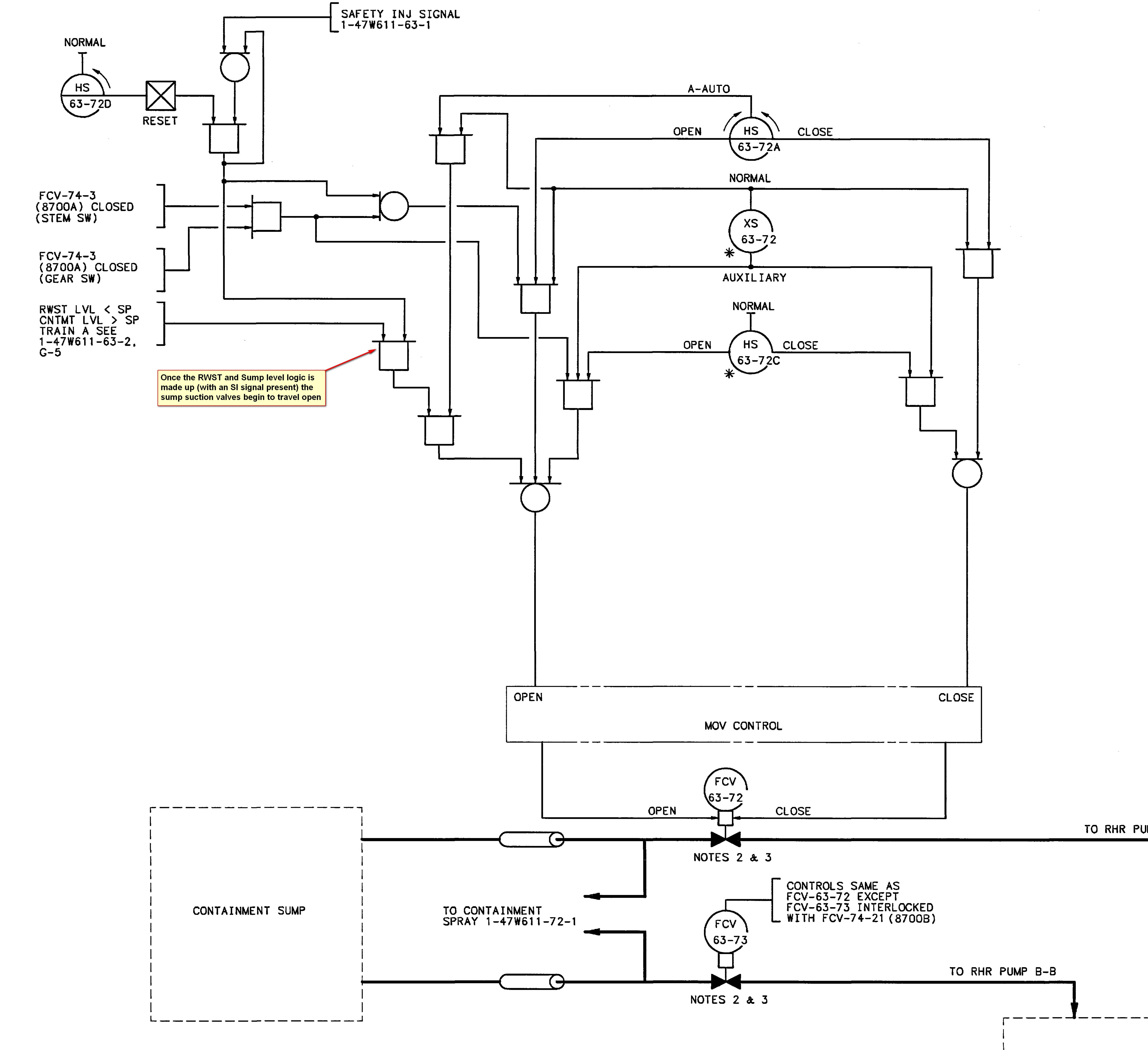
Loop accuracy and settings for 1-LS-63-50

Comparator Functions	Safety Analysis Limit	Setpoint	Tech Spec Allowable Value
Auto Recirc High Interlock 1-LS-63-180-D (LB-920A)	0.5% 7" above the floor (1)	16.1% INCR 38.2" above the floor (7)	15.6% 37.2" above the floor (8)
PAMS Indication Fcts	**Loop Allowable Value**	**Acceptable As-Found**	**Acceptable As-Left**
Containment Sump Level	±7.55% (12)	±6.29%	±1.52%

Loop accuracy and settings for 1-LS-63-180

Note that there is a setpoint of 34.63% RWST level and 16.1% sump level. There is also a T/S Allowable value of 34.0% RWST level and 15.6% sump level. The WOG setpoint document uses 34.63% and 16.1% (i.e. it takes the setpoint of the RWST and the T/S Allowable value for use).


RWST LEVEL LOOPS
LT 63-50 I
LT 63-51 II
LT 63-52 III
LT 63-53 IV
LS 63-50A
LS 63-51A
LS 63-52A
LS 63-53A
A 6C-126C
2/4
TRAIN A
LEVEL < SETPOINT
2/4
TRAIN B
Note that coincidence is 2 of 4.
2/4
TRAIN A
LEVEL > SETPOINT
2/4
TRAIN B
TRAIN A TRIP SIGNAL TO FCV-63-72 (8811A) 1-47W611-63-5, B-2
TRAIN B SIGNAL TO FCV-63-73 (8811B) 1-47W611-63-5, D-4
LS 63-180
LS 63-181
LS 63-182
LS 63-183
A 6C-127E
CNTMT FLOOR LEVEL LOOPS
LT 63-180 I
LT 63-181 II
LT 63-182 III
LT 63-183 IV


SAFETY INJ SIGNAL 1-47W611-63-1
NORMAL
HS 63-72D
RESET
A-AUTO
OPEN
HS 63-72A
CLOSE
NORMAL
XS 63-72
AUXILIARY
FCV-74-3 (8700A) CLOSED (STEM SW)
FCV-74-3 (8700A) CLOSED (GEAR SW)
RWST LVL < SP CNTMT LVL > SP TRAIN A SEE 1-47W611-63-2, G-5
NORMAL
OPEN
HS 63-72C
CLOSE
Once the RWST and Sump level logic is made up (with an SI signal present) the sump suction valves begin to travel open
OPEN
CLOSE
MOV CONTROL
FCV 63-72
OPEN
CLOSE
TO RHR PU
NOTES 2 & 3
CONTAINMENT SUMP
TO CONTAINMENT SPRAY 1-47W611-72-1
CONTROLS SAME AS FCV-63-72 EXCEPT FCV-63-73 INTERLOCKED WITH FCV-74-21 (8700B)
FCV 63-73
TO RHR PUMP B-B
NOTES 2 & 3

34.

Which ONE of the following describes the operation of the Primary Relief Tank (PRT)?

The PRT is cooled by dispersing primary water into the PRT ____(1)____ the quench water level.

As primary water is added to the PRT, 1-FCV-68-310, PRT Drain to the RCDT ____(2)____ **AUTOMATICALLY** cycle to maintain PRT level.

A. (1) ABOVE
   (2) WILL

B. (1) ABOVE
   (2) WILL **NOT**

C. (1) BELOW
   (2) WILL

D. (1) BELOW
   (2) WILL **NOT**

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: Plausible because the Primary Water is dispersed above the water level but the RCDT tank pump B will not require a manual start (it automatically starts when 1-FCV-68-310 gets fully open.) The

B. Correct: Primary Water for cooling is dispersed through a spray header above the water level and if draining of the PRT is required to lower the level, 1-FCV-68-310, PRT Drain to RCDT is opened by the OAC. When the valves gets fully open the RCDT B pump will automatically start.

C. Incorrect: Plausible because there is a dispersion header below the water level (sparging header) to disperse steam entering the PRT and the RCDT tank pump B automatically starting when 1-FCV-68-310 gets fully open is correct.

D. Incorrect: Plausible because there is a dispersion header below the water level (sparging header) to disperse steam entering the PRT and the RCDT tank pump B will not require a manual start (it automatically starts when 1-FCV-68-310 gets fully open.)

Question Number: 34

Tier: 2 Group: 1

K/A: 007 Pressurizer Relief Tank/Quench Tank System (PRTS)
K4 Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following:
K4.01 Quench tank cooling

Importance Rating: 2.6 2.9

10 CFR Part 55: (CFR: 41.7)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to relate the knowledge of the PRT design features for accomplishing one of its functions (maintain the tank cooled during a steam discharge).
.

Technical Reference: System Description N3-68-4001, Reactor Coolant System
1-47W813-1
1-47W611-68-1

Proposed references to be provided: None

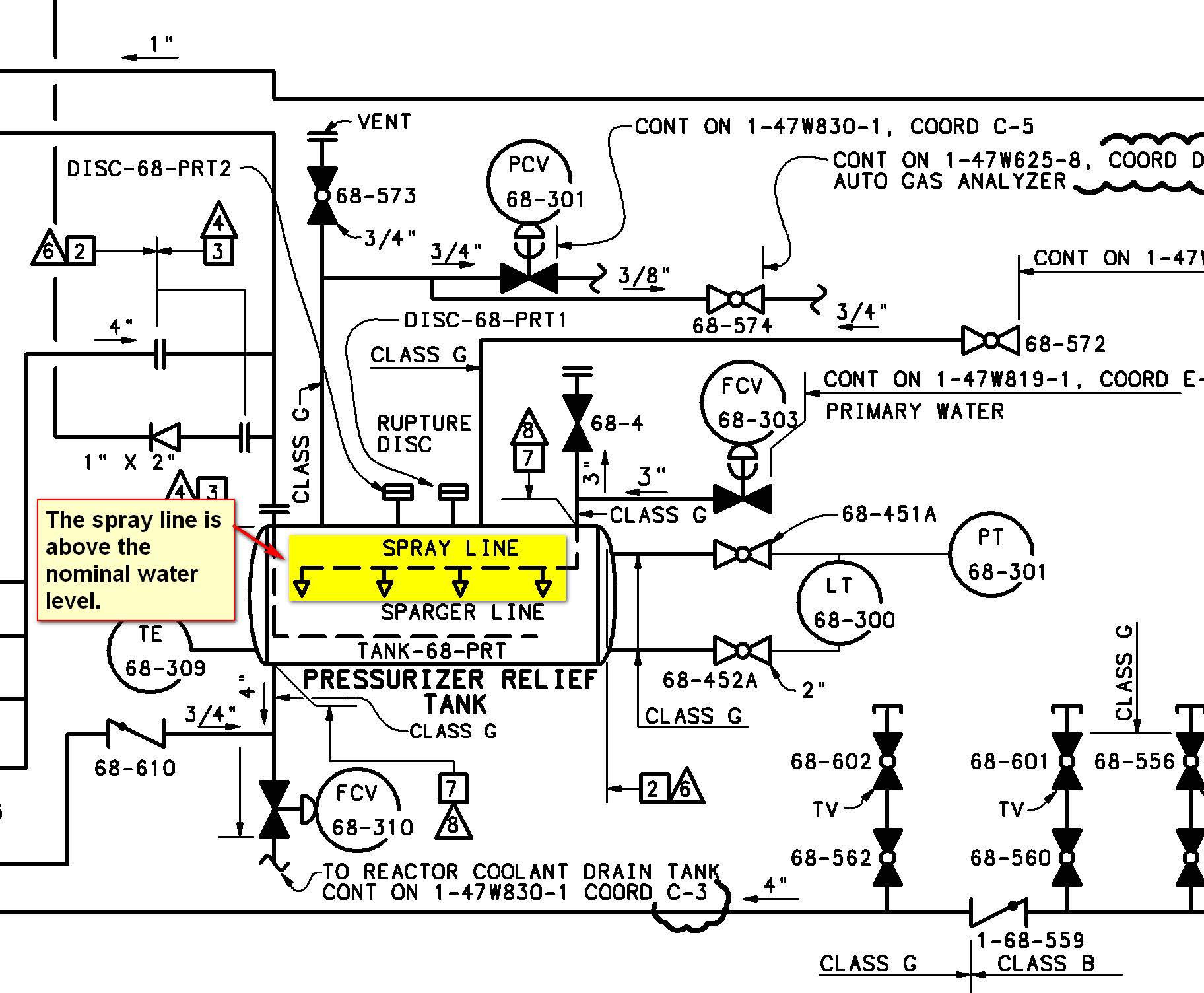
Learning Objective: 3-OT-STG-068C, PRESSURIZER AND PRESSURIZER RELIEF TANK
10. EXPLAIN, the physical connections and/or cause-effect relationships between the Pressurizer Relief Tank (PRT) and the following:
f. Primary water
11. EXPLAIN the Pressurizer Relief Tank (PRT) design features and/or interlocks that provide the following:
a. Tank cooling
c. Makeup water

Cognitive Level:
Higher ____
Lower X

Question Source:
New ____
Modified Bank ____
Bank X

Question History: WBN Bank question

Comments:



The spray line is above the nominal water level.
SPRAY LINE
SPARGER LINE
TANK-68-PRT
PRESSURIZER RELIEF TANK
VENT
68-573
DISC-68-PRT2
DISC-68-PRT1
RUPTURE DISC
PCV 68-301
CONT ON 1-47W830-1, COORD C-5
CONT ON 1-47W625-8, COORD D
AUTO GAS ANALYZER
68-574
68-572
FCV 68-303
CONT ON 1-47W819-1, COORD E-
PRIMARY WATER
68-4
CLASS G
68-451A
PT 68-301
LT 68-300
68-452A
TE 68-309
68-610
FCV 68-310
TO REACTOR COOLANT DRAIN TANK
CONT ON 1-47W830-1 COORD C-3
68-602
68-562
68-601
68-560
68-556
TV
1-68-559
CLASS B
1" X 2"

3.2.5 Pressurizer Relief Tank (PRT) (continued)

Steam is discharged into the PRT through a sparger pipe under the water level. This condenses and cools the steam by mixing it with water that is near ambient temperature. The tank is equipped with an internal spray and a drain which are used to cool the tank following a discharge. The tank is protected against a discharge exceeding the design value by two rupture discs which discharge into the reactor containment. The tank is carbon steel with a corrosion-resistant coating on the wetted surfaces. A flanged nozzle is provided on the tank for the PZR discharge line connection. This nozzle and the discharge piping and sparger within the vessel are austenitic stainless steel.

The tank design is based on the requirement to condense and cool a discharge of PZR steam equal to 110% of the volume above the full-power PZR water level set point. The tank is not designed to accept a continuous discharge from the PZR. The volume of water in the tank is capable of absorbing the heat from the assumed discharge. If the temperature in the tank rises above 128°F during plant operation, the tank is cooled by spraying in cool water from the primary water system and draining out the warm mixture to the waste disposal system.

The spray rate is designed to cool the tank from 200°F to 120°F in approximately one hour following the design discharge of PZR steam. The volume of nitrogen gas in the tank is selected to limit the maximum pressure following a design discharge to 50 psig.

The PRT rupture discs have a relief capacity equal or greater than the combined capacity of the PZR safety valves. The tank design pressure (and the maximum rupture disc blowout point) is twice the calculated pressure resulting from the maximum safety valve discharge. This margin is to prevent deformation of the disc. The rupture disc holders (15 psig minimum external pressure) are designed for full vacuum to prevent tank collapse if the contents cool following a discharge without nitrogen being added. Failure of the PRT does not affect the integrity of the RCPB nor does it affect the capability to shut down the plant safely.

PRT spray for cooling of its contents is manually controlled by opening FCV-68-303 from the MCR. Discharge from the PRT to the WDS vent header is manually controlled by the opening of PCV-68-301 from the MCR. Draining the PRT to the RCDT is manually controlled by opening FCV-68-310 from the MCR.

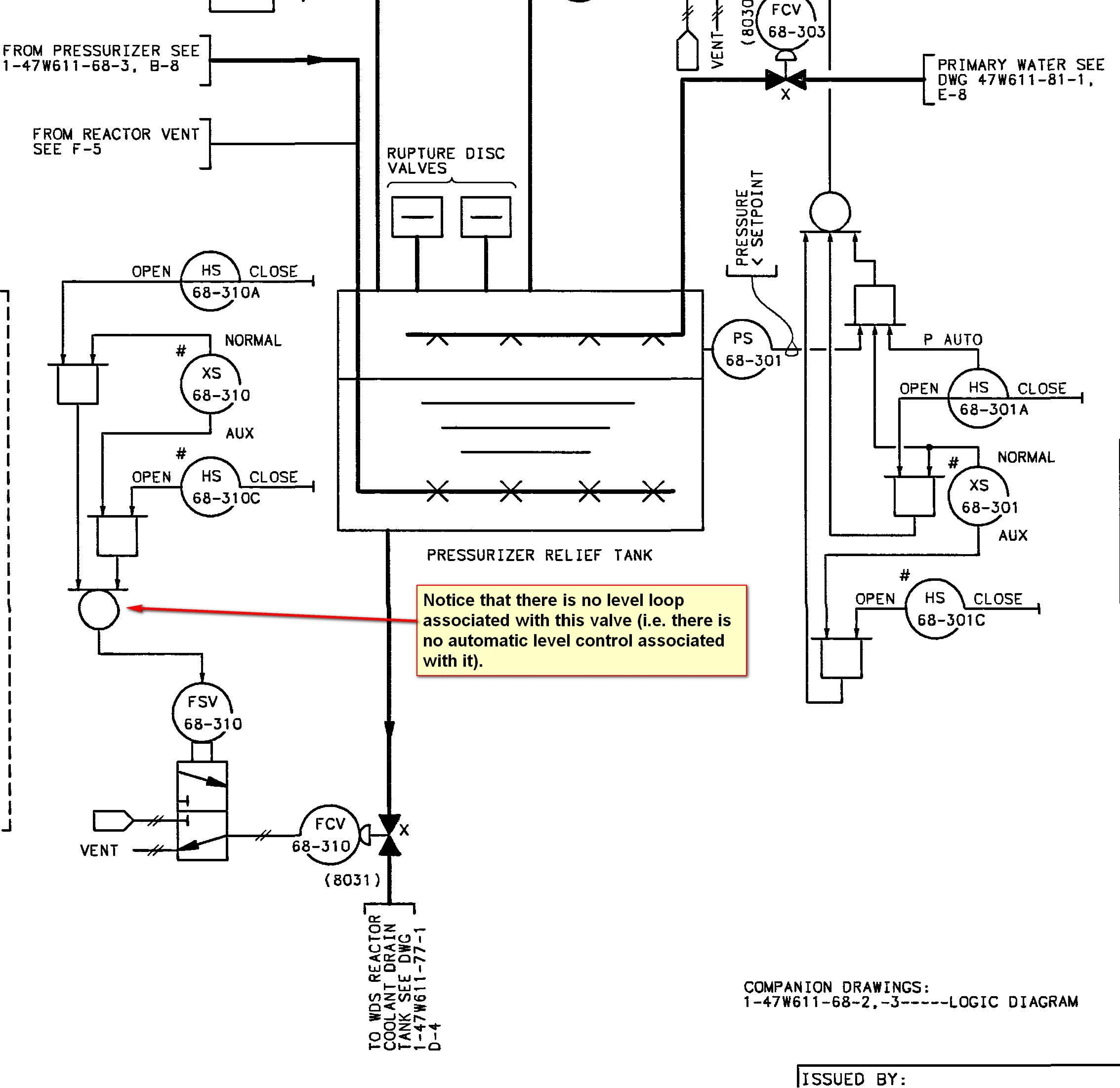
3.2.6 RCS Valves

All RCS valves are constructed primarily of stainless steel. Where stainless steel is not utilized, such as for hard surfacing and packing, other compatible materials are in contact with the coolant.

The (MOVs) uniqueness requirements are defined in Ref. 7.1.8.

The RCS valves in Table 19 are classified as active per the following rules:

a. Only WBN plant specific Design Basis Events (DBEs) that can be logically coupled with an SSE were considered. These DBEs were studied to identify the active pumps and valves required to mitigate the DBE and place the plant in a safe shutdown condition.

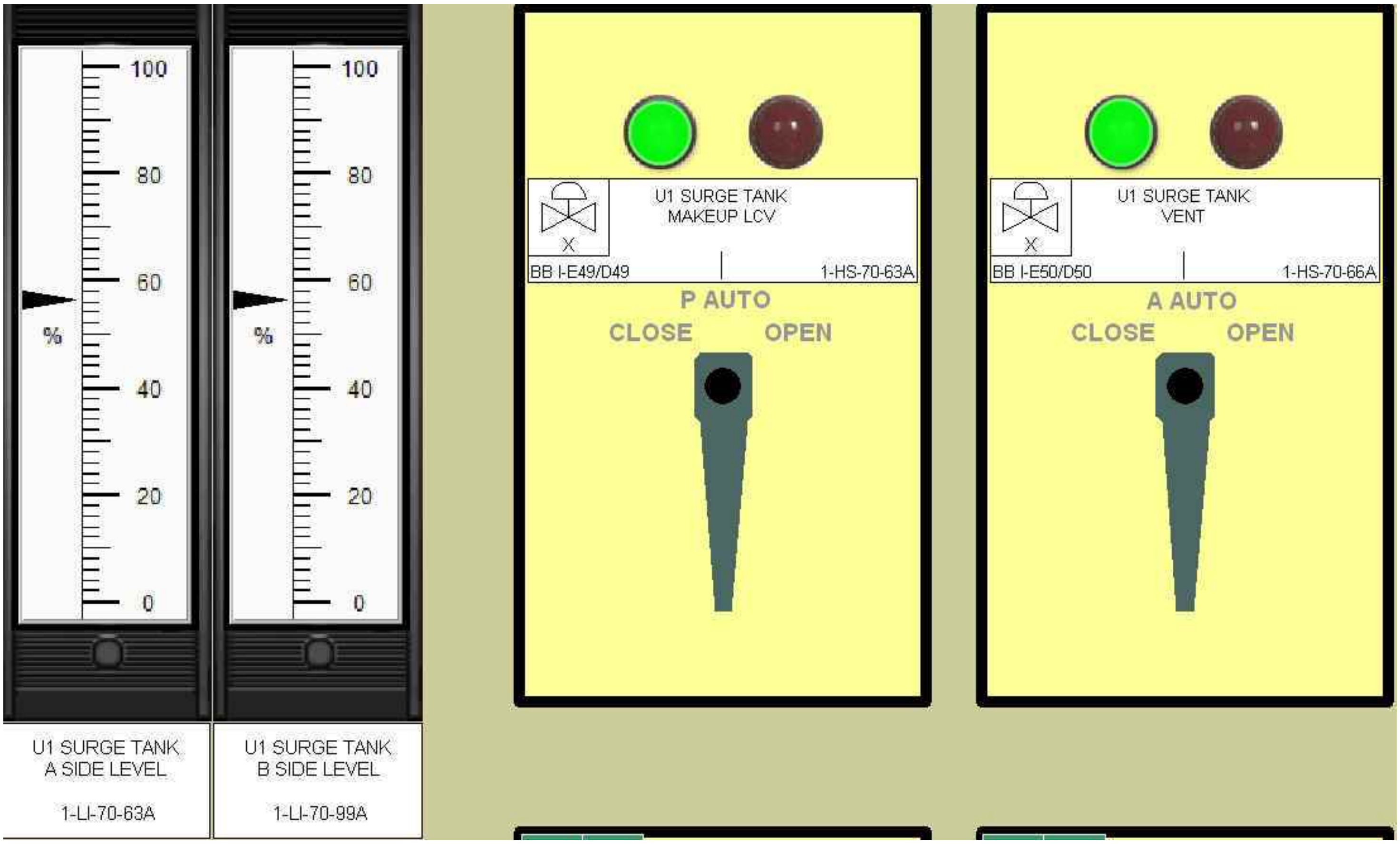


Notice that there is no level loop associated with this valve (i.e. there is no automatic level control associated with it).

35.

Given the following conditions:

- Unit 1 is at 100% power.
- The following is observed:



Which ONE of the following completes the statements listed below?

The status of the Unit 1 “A” train CCS surge tank indicates that __________.

NOTE: 1-AOI-15, Loss of Component Cooling Water (CCS)
0-AOI-10, Loss of Control Air

A. conditions are NORMAL and that NO further actions are required

B. Control air has been LOST to 1-LCV-70-63 and 1-FCV-70-66; 0-AOI-10 WILL be entered.

C. CCS in-leakage is probable and section 3.4, CCS In-Leakage of 1-AOI-15 WILL be entered

D. because the valve alignment for the CCS surge tank is correct, a leak exists on the Unit 1 “A” train CCS and section 3.3, CCS Out Leakage of 1-AOI-15 WILL be entered

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: The condition of the CCS surge tank is abnormal. Firstly, the level of the CCS surge tank is seen to be 57%. This level is abnormally low; this may be seen in the diagnostics for section 3.3, “CCS Out Leakage” of 1-AOI-15 which state: “Surge Tank level less than 60% and dropping uncontrolled.” Additionally, annunciator 249-A, “U1 SURGE TANK LEVEL HI/LO,” would be expected as its LO setpoint is 57%. Other abnormalities include the fact that both the vent and makeup valves are closed. The vent valve is normally open (it will close on a high radiation signal). The makeup valve is designed to open when level in the tank is less than 60%.

B. Correct: As seen on print 1-47W859-1, both 1-LCV-70-63 and 1-FCV-70-66 are air operated valves which fail closed upon either a loss of air or signal. One may observe in ARI 249-A the probable cause which states: “Surge Tank vent closed causing positive or negative Surge Tank press; giving erroneous level indication.” The aggregate of the indications seen in the stem show that control air has been lost to both of the mentioned valves. The vent valve caused surge tank level to indicate abnormally low and even though the level was below the makeup setpoint (of < 60%) the failed closed makeup valve was unable to provide any flow.

C. Incorrect: The level is less than that normally expected. However, if one believed that normal level was less than that shown (e.g. at mid-scale or 50%) then the indications would present a potential in-leakage to the CCS system. The diagnostics for section 3.4, “CCS In-Leakage” of 1-AOI-15 contain a check for CCS surge tank level high or rising uncontrolled OR a CCS rad monitor alarm present. Because the vent valve closes on a rad monitor alarm it is plausible to believe that such is present. Accordingly, if one misunderstood the normal surge tank level, then one would believe that in-leakage (e.g. from the thermal barriers of the RCPs) was occurring.

D. Incorrect: Again, two valves are observed to be out of their expected alignment. However, if the valves were in the expected alignment, the indicated level would indicate a CCS inventory loss and the crew would implement section 3.3, “CCS Out Leakage” of 1-AOI-15.

Question Number: 35

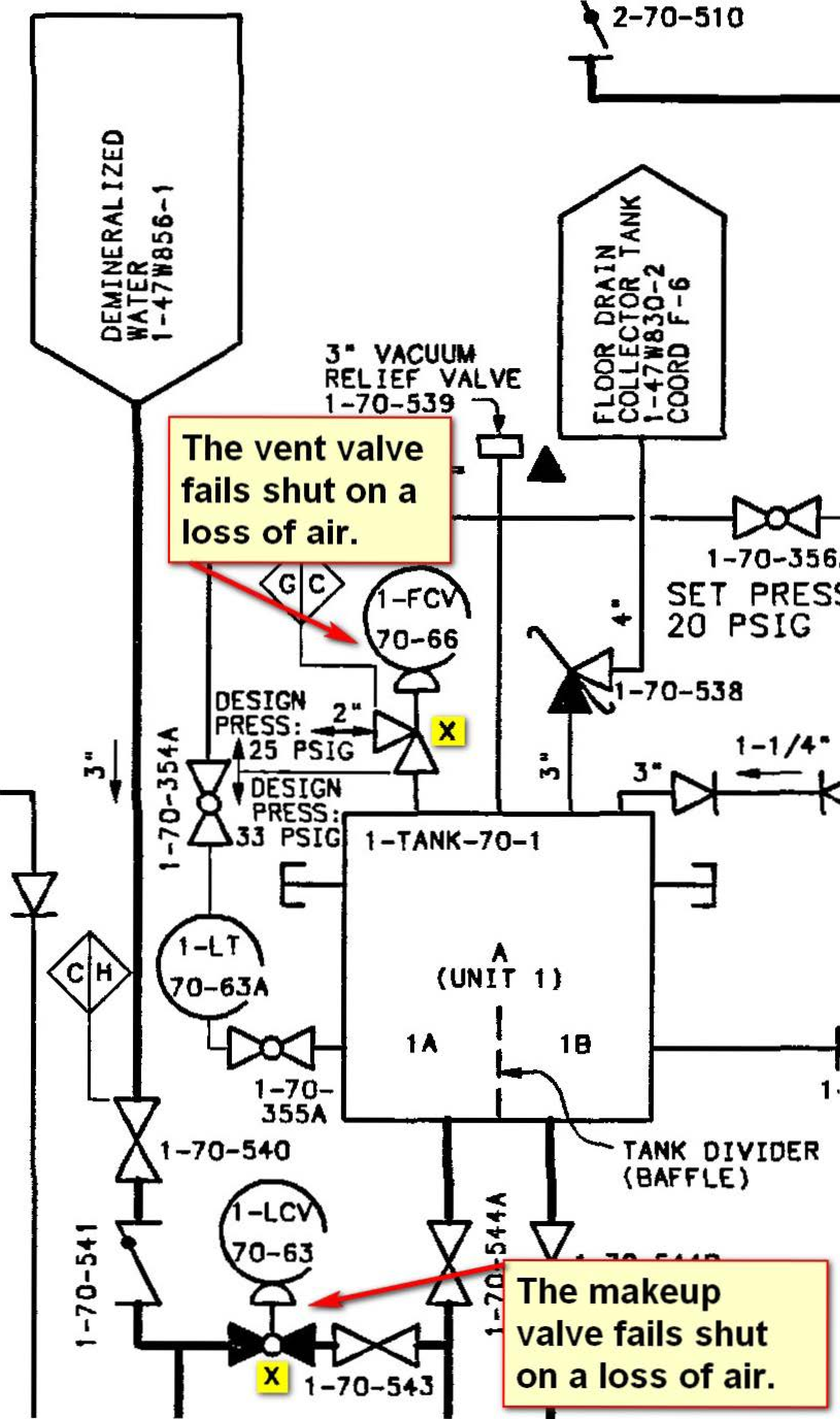
Tier: 2 Group: 1

K/A: 008 Component Cooling Water System (CCWS)
A2 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:
A2.05 Effect of loss of instrument and control air on the position of the CCW valves that are air operated

Importance Rating: 3.3 3.5
10 CFR Part 55: (CFR: 41.5 / 43.5 / 45.3 / 45.13)
10CFR55.43.b: Not applicable
K/A Match: K/A is matched because the applicant is required to predict the impact to the makeup of the CCS surge tank by understanding the failure position of the makeup level control valve (given that it loses control air) and then be able to utilize ARI-249-A to recover from the failure.
Technical Reference: ARI-249-A, CCS
1-47W859-1
1-AOI-15, Loss of Component Cooling Water (CCS)

Proposed references to be provided: None
Learning Objective: 3-OT-STG-070A, COMPONENT COOLING WATER SYSTEM
5. EXPLAIN, the physical connections and/or cause-effect relationships
between the CCS and the following systems:
e. Radiation Monitoring System
f. Sources of Make-up Water
Cognitive Level:
Higher X
Lower
Question Source:
New X
Modified Bank
Bank
Question History: New question for the 2015-301 NRC RO Exam

Comments:


2-70-510
DEMINERALIZED
WATER
1-47W856-1
FLOOR DRAIN
COLLECTOR TANK
1-47W830-2
COORD F-6
3" VACUUM
RELIEF VALVE
1-70-539
The vent valve fails shut on a loss of air.
G C
1-FCV
70-66
20 PSIG
4"
1-70-538
DESIGN
PRESS:
25 PSIG
2"
X
1-1/4"
3"
3"
3"
1-70-354A
DESIGN
PRESS:
33 PSIG
1-TANK-70-1
1-LT
70-63A
C H
A
(UNIT 1)
1A
1B
1-70-
355A
TANK DIVIDER
(BAFFLE)
1-70-540
1-70-541
1-LCV
70-63
1-70-544A
X
1-70-543
The makeup valve fails shut on a loss of air.

3.0 OPERATOR ACTIONS

3.1 Diagnostics

IF	GO TO Subsection	PAGE
LOSS OF CCS FLOW: • Pump trip • Multiple low flow alarm • Low header pressure	3.2	6
Surge Tank level less than 60% and dropping uncontrolled **OR** Indications of out leakage.	3.3	11
Surge Tank level greater than 72% or rising uncontrolled, **OR** CCS Rad Monitor alarm.	[illegible]	[illegible]
Loss of ERCW to CCS HX A	[illegible]	[illegible]3
Loss of ERCW to CCS HX C	3.6	37
Loss of CCS while RHR Shutdown Cooling is in service	1-AOI-14	

The AOI utilizes 60% as a decision point.

a high level or a rad monitor alarm (which causes the surge tank vent to shut)

249-A

U1 SURGE TANK LEVEL HI/LO

(Page 1 of 2)

Source	**Setpoint**
1-LS-70-99A/B	85%
1-LS-70-99B/A	57%

The low alarm would be expected for the conditions shown in the stem of the question.

Probable Cause:

A. Leak into or out of CCS
B. Makeup system malfunction
C. Gas bubble in system
D. Valve misalignment
E. Surge Tank vent closed causing positive or negative Surge Tank press; giving erroneous level indication (if level in BOTH sides of tank is less than 60%, a false level may be indicated due to interior baffle design)

The surge tank vent is normally open.

It is abnormal that the makeup valve is seen to be shut in the stem of the question.

NOTE

Auto makeup should initiate at 60% lowering level.

Corrective Action:

[1] **MONITOR** Surge Tank level.

[2] **IF** level is low, **THEN**:

[2.1] **ENSURE** 1-LCV-70-63, U1 SURGE TANK MAKEUP LCV, OPEN.

[2.2] **IF** Surge Tank level lowers below 52% or 1-FCV-70-183, SAMPLE HX CCS OUTLET is CLOSED, **THEN**
NOTIFY Chemistry to suspend sampling.

[2.3] **IF** Surge Tank loss imminent, **THEN**
GO TO AOI-15, LOSS OF COMPONENT COOLING WATER (CCS). [C.1]

[2.4] **IF** level is **NOT** maintained due to loss of makeup, **THEN**
CONSIDER installing ERCW spool piece for emergency source.

[2.5] **IF** level is **NOT** maintained due to loss of air to makeup valve, **THEN**
CONSIDER installing bottled gas on 1-LCV-70-63.

Continued on Next Page

36.

Which ONE of the following describes the required actions for a Loss of Component Cooling Water (CCS) with Unit 1 operating at 100% power; in accordance with 1-AOI-15, Loss of Component Cooling Water?

The operating crew will initiate a Unit 1 reactor trip if the ____(1)____ CCS train is lost.

RCPs can be operated for a **MAXIMUM** of ____(2)____ minutes after loss of CCS flow.

	(1)	(2)
A.	1A	10
B.	1A	12
C.	1B	10
D.	1B	12

<u>CORRECT ANSWER:</u> <u>A</u>

<u>DISTRACTOR ANALYSIS:</u>

A. Correct: In accordance with 1-AOI-15, the reactor is tripped upon the loss of the 1A CCS train (with the Unit at power). This is seen in several instances within the contents of the AOI. For example, step 3. States: CHECK 1A Train flows NORMAL … e. TRIP Reactor. f. STOP RCPs. Additionally, it is correct that the RCPs can be operated for up to 10 minutes after loss of CCS flow. This fact is seen in the CAUTION in section 3.2 of 1-AOI-15 (prior to step 3.).

B. Incorrect: While it is correct that the reactor is tripped upon the loss of the 1A CCS train, it is not correct that the RCPs can be operated for up to 12 minutes after loss of CCS flow. It is plausible to believe this because the same caution in 1-AOI-15 which describes the impact to the RCPs contains the following: CCP may survive for only 10 to 12 minutes after loss of CCS to lube oil cooler.

C. Incorrect: It is Incorrect that the loss of 1B CCS train would cause the crew to initiate a reactor trip. It is plausible to believe this because if 1B CCS train provided cooling to the RCPs then its loss would require tripping the reactor (with the Unit at power). Additionally, it is correct that the RCPs can be operated for up to 10 minutes after loss of CCS flow.

D. Incorrect: Again it is Incorrect but plausible that the loss of 1B CCS train would cause the crew to initiate a reactor trip. It is also, as mentioned, Incorrect and yet plausible that the RCPs could be operated for up to 12 minutes after loss of CCS flow.

Question Number: 36

Tier: 2 Group: 1

K/A: 008 Component Cooling Water System (CCWS)
2.4 Emergency Procedures / Plan
2.4.11 Knowledge of abnormal condition procedures.

Importance Rating: 4.0 4.2

10 CFR Part 55: (CFR: 41.10 / 43.5 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to utilize the guidance provided in 1-AOI-15 to ascertain which CCS train's loss will precipitate a reactor trip. Furthermore, the applicant must recall the guidance provided in the same AOI to the operational impact to the RCPs given that a loss of their CCS occurs.

Technical Reference: 1-AOI-15, Loss of Component Cooling Water (CCS)

Proposed references to be provided: None

Learning Objective: 3-OT-STG-070A, COMPONENT COOLING WATER SYSTEM
5. EXPLAIN, the physical connections and/or cause-effect relationships
between the CCS and the following systems:
a. RCP

Cognitive Level:
Higher ____
Lower X

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Step	Action/Expected Response	Response Not Obtained

3.2 Loss of CCS Flow

Step	Action/Expected Response	Response Not Obtained
1.	**CHECK** at least one U-1 Train A header supply pump RUNNING AND pumping forward: • 1A-A • 1B-B	**START** available U-1 Train A CCS pump.
2.	**IF** loss of CCS is due to a loss of AC power train, **THEN** **REFER** to AOI-43 series and continue with this procedure.	

This is the reason that 12 is plausible

RCPs can be operated for UP to 10 minutes.

IONS

- **CCP may survive for only 10 to 12 minutes after loss of CCS to lube oil cooler.**
- **RCPs can be operated for up to 10 minutes after loss of CCS flow.**

Step	Action/Expected Response	Response Not Obtained
3.	**CHECK** 1A Train flows NORMAL • CCS Low Flow alarms on 0-M-27 CLEAR.	**IF** 1A Train CCS flow lost, **THEN** **PERFORM** the following: a. **ENSURE** CCP 1B-B Running **THEN** **STOP** and **LOCKOUT** CCP 1A-A. b. **INITIATE** alignment of ERCW to CCP 1A-A lube oil heat exchanger USING Attachment 1 (may use placard posted locally in CCP room 1A-A).

"A" train is lost.

Step continued on next page

3.2 Loss of CCS Flow (continued)

Step 3 continued.

c. **ISOLATE** letdown and charging.

1) **CLOSE** Orifices valves
2) **CLOSE** 1-FCV-62-69 & 70
3) **CLOSE** 1-FCV-62-90 & 91
4) **ENSURE** excess letdown 1-FCV-62-54 & 55 isolated.

d. **STOP** and **LOCKOUT** the following pumps:

- TBBPs 1-A & 1-B,
- CCS pumps 1A-A & 1B-B (If aligned to A Trn)
- CS pump 1A-A,
- SI pump 1A-A,
- RHR pump 1A-A,
- CCP 1A-A (IF ERCW not aligned and greater than 10 min has elapsed)

If "A" train CCS is lost, then the Rx must be tripped.

e. **TRIP** Reactor.

f. **STOP** RCPs.

g. ****GO TO** 1-E-0, Reactor Trip or Safety Injection,
WHILE CONTINUING this Instruction.

37.

Which ONE of the following describes the power supply to the 1C and 1D PZR heaters?

A. 480V Unit Boards

B. 6.9kV Unit Boards

C. 480V Shutdown Boards

D. 6.9kV Shutdown Boards

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: It is not correct that the 1C and 1D heater groups are powered from the 480V Unit Boards. Two items lend plausibility to the belief that the Unit Boards would power the 1C and 1D heater groups. First, as seen in the labeling for these heater groups, there is NO train designation (i.e. a –A or –B). This would lead one to believe that the component is **NOT** safety related and as such **NOT** powered from shutdown power. Additionally, as seen in T/S LCO 3.4.9, Pressurizer: The pressurizer shall be OPERABLE with: b. Two groups of pressurizer heaters OPERABLE with the capacity of each group > 150kW. Because the Unit's T/S do not require that four groups of heaters be operable, it is plausible to believe that only two would receive shutdown power. Also, it is plausible to believe that because the pressurizer heaters are actually 480VAC heaters that they would be powered from a 480VAC board.

B. Incorrect: It is not correct that the 1C and 1D heater groups are powered from the 6.9kV Unit Boards. It is plausible to believe this for a reason tantamount to the "A" distractor with the exception that an applicant may understand the voltage of the supply to the heaters is 6.9kV.

C. Incorrect: While it is correct that shutdown power supplies the 1C and 1D heaters, it is not correct that the supply originates from a 480V shutdown board. It is plausible to believe that because the pressurizer heaters are actually 480VAC heaters that they would be powered from a 480VAC board.

D. Correct: It is correct that the 1C backup group and 1D control group of pressurizer heaters are powered from the 6.9kV Shutdown Boards.

Question Number: 37

Tier: 2 Group: 1

K/A: 010 Pressurizer Pressure Control System (PZR PCS)
K2 Knowledge of bus power supplies to the following:
K2.01 PZR heaters

Importance Rating: 3.0 3.4

10 CFR Part 55: (CFR: 41.7)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to possess the knowledge that ALL of the groups of pressurizer heaters are powered from the 6.9kV shutdown boards.

Technical Reference: 1-45W724-1
1-45W724-2
Unit 1 Technical Specifications

Proposed references to be provided: None

Learning Objective: 3-OT-STG-068C, PRESSURIZER AND PRESSURIZER RELIEF TANK
5. LIST the power supplies to the following Pressurizer Level Control System components:
a. PZR Heaters

Cognitive Level:
Higher ____
Lower X

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 92%; and

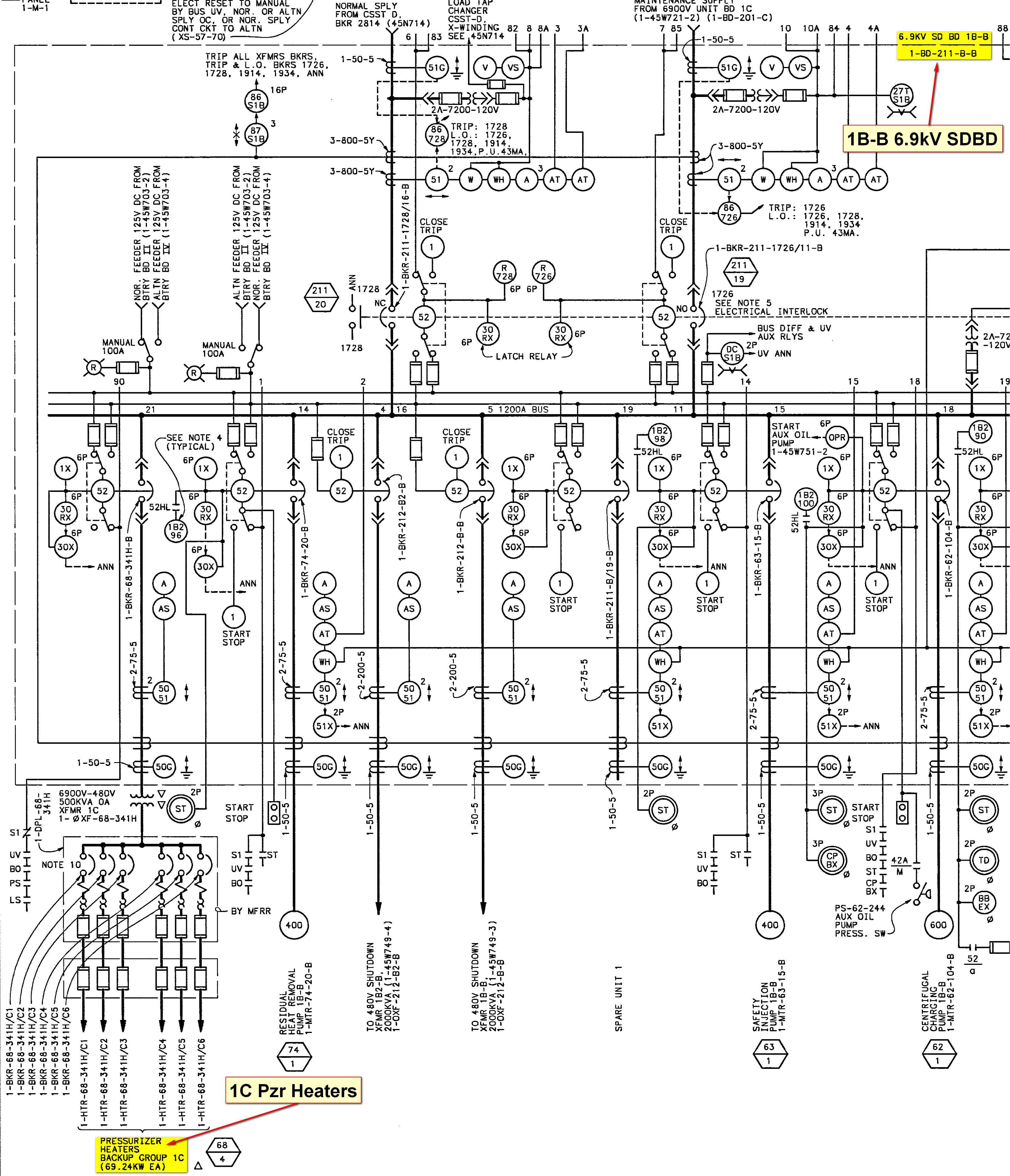
b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open.	6 hours
	AND	
	A.2 Be in MODE 4.	12 hours
B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 4.	12 hours

6.9KV SD BD 1A-A
1-BD-211-A-A
1A-A 6.9 kV SDBD
FROM CSST D, BKR 1812
(45N714)
(DETAIL 1, 1-45W727)
PREVENTS AUTO TRANSFER TO DEAD/ DEGRADED SOURCE
2Λ-7200-120V
TRIP: 1932
L.O.: 1718, 1716, 1912, 1932, P.U. 43MA
PERMIT AUTO CLOSING OF BKR 1912 & COMP INPUT
ALTN SUPPLY 125V DC FROM BTRY BD I (1-45W703-1)
NOR. SUPPLY 125V DC FROM BTRY BD III (1-45W703-3)
MANUAL 100A
TDDO 62 1932/1 ANN
LATCH RELAY
CLOSE TRIP
1-BKR-211-1932/1-A
1-BKR-211-1912/6-A
ELECTRICAL INTERLOCK
UV ANN
UV AUX RLYS
BUS DIFF & UV AUX RLYS
1200A BUS
SEE NOTE 5 (TYPICAL)
START AUX OIL PUMP 1-45W751-1
1-BKR-68-341F-A
1-BKR-72-27-A
1-BKR-212-A2-A
1-BKR-212-A-A
1-BKR-74-10-A
1-BKR-63-10-A
START STOP
SPRAY ACT
6900V-480V 500KVA OA XFMR 1D 1-OXF-68-341F
1-SCR-68-341F BY MFRR
1-DPL-68-341F
NOTE 10
BY MFRR
PS-62-247 AUX OIL PUMPS PRESS. SW
CONTAINMENT SPRAY PUMP 1A-A 1-MTR-72-27-A
TO 480V SHUTDOWN XFMR 1A2-A, 2000KVA (1-45W749-2) 1-OXF-212-A2-A
TO 480V SHUTDOWN XFMR 1A-A, 2000KVA (1-45W749-1) 1-OXF-212-A-A
RESIDUAL HEAT REMOVAL PUMP 1A-A 1-MTR-74-10-A
SAFETY INJECTION PUMP 1A-A 1-MTR-63-10-A
1-BKR-68-341F/D1
1-BKR-68-341F/D2
1-BKR-68-341F/D3
1-BKR-68-341F/D4
1-BKR-68-341F/D5
1-BKR-68-341F/D6
1-HTR-68-341F/D1
1-HTR-68-341F/D2
1-HTR-68-341F/D3
1-HTR-68-341F/D4
1-HTR-68-341F/D5
1-HTR-68-341F/D6
1D Pzr Heaters
PRESSURIZER HEATERS CONTROL GROUP 1D (69.24KW EA)



1-M-1
ELECT RESET TO MANUAL BY BUS UV, NOR. OR ALTN SPLY OC, OR NOR. SPLY CONT CKT TO ALTN (XS-57-70)
NORMAL SPLY FROM CSST D, BKR 2814 (45N714)
LOAD TAP CHANGER CSST-D, X-WINDING SEE 45N714
MAINTENANCE SUPPLY FROM 6900V UNIT BD 1C (1-45W721-2) (1-BD-201-C)
6.9KV SD BD 1B-B
1-BD-211-B-B
1B-B 6.9kV SDBD
TRIP ALL XFMRS BKRS, TRIP & L.O. BKRS 1726, 1728, 1914, 1934, ANN
1-50-5
2A-7200-120V
TRIP: 1728
L.O.: 1726, 1728, 1914, 1934, P.U. 43MA.
3-800-5Y
TRIP: 1726
L.O.: 1726, 1728, 1914, 1934
P.U. 43MA.
NOR. FEEDER 125V DC FROM BTRY BD II (1-45W703-2)
ALTN FEEDER 125V DC FROM BTRY BD IV (1-45W703-4)
ALTN FEEDER 125V DC FROM BTRY BD II (1-45W703-2)
NOR. FEEDER 125V DC FROM BTRY BD IV (1-45W703-4)
CLOSE TRIP
1-BKR-211-1728/16-B
1-BKR-211-1726/11-B
1726 SEE NOTE 5 ELECTRICAL INTERLOCK
BUS DIFF & UV AUX RLYS
UV ANN
LATCH RELAY
MANUAL 100A
5 1200A BUS
SEE NOTE 4 (TYPICAL)
START AUX OIL PUMP 1-45W751-2
START STOP
1-BKR-68-341H-B
1-BKR-74-20-B
1-BKR-212-B2-B
1-BKR-212-B-B
1-BKR-211-B/19-B
1-BKR-63-15-B
1-BKR-62-104-B
2-75-5
2-200-5
6900V-480V 500KVA OA XFMR 1C 1-ØXF-68-341H
1-DPL-68-341H
NOTE 10
BY MFRR
PS-62-244 AUX OIL PUMP PRESS. SW
1-BKR-68-341H/C1
1-BKR-68-341H/C2
1-BKR-68-341H/C3
1-BKR-68-341H/C4
1-BKR-68-341H/C5
1-BKR-68-341H/C6
1-HTR-68-341H/C1
1-HTR-68-341H/C2
1-HTR-68-341H/C3
1-HTR-68-341H/C4
1-HTR-68-341H/C5
1-HTR-68-341H/C6
1C Pzr Heaters
PRESSURIZER HEATERS BACKUP GROUP 1C (69.24KW EA)
RESIDUAL HEAT REMOVAL PUMP 1B-B 1-MTR-74-20-B
TO 480V SHUTDOWN XFMR 1B2-B, 2000KVA (1-45W749-4) 1-OXF-212-B2-B
TO 480V SHUTDOWN XFMR 1B-B, 2000KVA (1-45W749-3) 1-OXF-212-B-B
SPARE UNIT 1
SAFETY INJECTION PUMP 1B-B 1-MTR-63-15-B
CENTRIFUGAL CHARGING PUMP 1B-B 1-MTR-62-104-B

38.

Which ONE of the following describes the EAGLE 21 system response to a Channel I input failure and the effect on the system's ability to calculate a trip setpoint?

If the channel I input for ____(1)____ **FAILS**, EAGLE-21 will be unable to calculate a **VALID** setpoint for the channel I ____(2)____ trip.

	(1)	(2)
A.	the PRNI	OPΔT
B.	Loop 1 Flow	OPΔT
C.	PZR Pressure	OTΔT
D.	Turbine Impulse Pressure	OTΔT

CORRECT ANSWER: C

DISTRACTOR ANALYSIS:

A. Incorrect: The OPΔT trip uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with reactor coolant average temperature and the rate of change of reactor coolant average temperature. These facts may be found in multiple locations; provided as supporting material is the T/S basis discussion for T/S LCO 3.3.1. It is plausible to believe that the loss of a channel I PRNI is detrimental to the ability of the OPΔT calculator because as seen in Note 2: Overpower ΔT of table 3.3.1-1 (of the Unit's T/S), the setpoint calculation contains the term $-f_2(\Delta I)$. Delta Flux (ΔI) is produced from the appropriate Power Range Nuclear Instrument (PRNI). As applied at WBN, delta flux (and thus, PRNI) is not included in the OPΔT calculation because as seen in the Note 2: $f_2(\Delta I)=0$ for all $f_2\Delta I$. Incidentally, it is a common misconception that OPΔT utilizes the PRNIs as the name overpower implies that it uses power as an input.

B. Incorrect: As seen in the basis for the Unit's T/S (again these facts may be found in multiple locations), the OTΔT setpoint is automatically calculated from reactor coolant average temperature, pressurizer pressure and axial power distribution f(ΔI). The inputs to OTΔT may also be seen in print 1-47W610-68-8. Also given by this reference (and multiple others) is that The Over temperature ΔT trip Function is provided to ensure that the design limit DNBR is met. Therefore, it is plausible to believe that a RCS parameter which does impact the DNBR (namely RCS flow) would input into the OTΔT setpoint. However, the OTΔT setpoint assumes normal full loop flow. This is one of the reasons why the ΔTs present in the RCS must be verified at the beginning of a operation cycle.

C. Correct: As described pressurizer pressure is an input which affects the setpoint for the OTΔT trip.

D. Incorrect: As mentioned OPΔT uses only temperatures as inputs (even though its name is over power). It is plausible that it would utilize an indication of power as an input (i.e. Turbine Impulse Pressure). In the case of channel I, 1-R-2 (Unit 1's rack 2) of the EAGLE-21 protective system, calculates the setpoints for both OPΔT and OTΔT. This fact is seen on print 1-47W610-99-1. 1-R-2 is included in the collection of channel I racks collectively known as protection set I. One of the channel I (or protection set I) racks, 1-R-4, receives turbine impulse pressure and uses such for a protective system function. This fact is seen on print 1-47W610-99-2. Therefore, it is additionally plausible that a channel I rack would use turbine impulse pressure for the production of the OPΔT setpoint.

Question Number: 38

Tier: 2 Group: 1

K/A: 012 Reactor Protection System
A1 Ability to predict and/or monitor Changes in parameters (to prevent exceeding design limits) associated with operating the RPS controls including:
A1.01 Trip setpoint adjustment

Importance Rating: 2.9* 3.4*

10 CFR Part 55: (CFR: 41.5 / 45.5)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to have knowledge of the parameters associated with the RPS and then be able to monitor the correct parameter predicting what its failure will have upon a RPS setpoint. The failures described which if unresolved would place the plant outside of design limits.

Technical Reference: Unit 1 Technical Specifications
Unit 1 Technical Specifications Basis
1-47W610-68-8
1-47W610-99-1
1-47W610-99-2

Proposed references to be provided: None

Learning Objective: 3-OT-TS0303, TS 3.3 “INSTRUMENTATION”
1. DESCRIBE the following aspects of Technical Specifications and Technical Requirements for Instrumentation Tech Specs:
a. The Limiting Conditions for Operation, Applicability, and Bases for the LCO.
b. The conditions and required actions with completion time of one hour or less.

Cognitive Level:
Higher ____
Lower X

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Table 3.3.1-1 (page 7 of 9)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.2% of ΔT span.

$$\Delta T \left\{ \frac{1 + \tau_4 s}{1 + \tau_5 s} \right\} \leq \Delta T_0 \left\{ K_1 - K_2 \frac{(1 + \tau_1 s)}{(1 + \tau_2 s)} \left[T - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_O is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the indicated T_{avg} at RTP, $\leq$ 588.2°F.

P is the measured pressurizer pressure, psig
P' is the nominal RCS operating pressure, $\geq$ 2235 psig

$K_1 \leq 1.16$	$K_2 \geq 0.0183$/°F	$K_3 = 0.000900$/psig
$\tau_1 \geq 33$ sec	$\tau_2 \leq 4$ sec	
$\tau_4 \geq 3$ sec	$\tau_5 \leq 3$ sec	

$f_1(\Delta I)$ =	$-2.62\{22 + (q_t - q_b)\}$	when $q_t - q_b < -22\%$ RTP
	0	when -22% RTP $\leq q_t - q_b \leq 10\%$ RTP
	$1.96\{(q_t - q_b) - 10\}$	when $q_t - q_b > 10\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

Table 3.3.1-1 (page 8 of 9)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.0% of ΔT span.

$$\Delta T\left(\frac{1+\tau_4 s}{1+\tau_5 s}\right) \le \Delta T_0\left\{K_4 - K_5\left(\frac{\tau_3 s}{1+\tau_3 s}\right)T - K_6\left[T - T''\right] - f_2(\Delta I)\right\}$$

This is always 0.

Where: ΔT is measured RCS ΔT, °F.
ΔT_O is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T'' is the indicated T_{avg} at RTP, $\le$ 588.2°F.

$K_4 \le 1.10$

$K_5 \ge 0.02$/°F for increasing T_{avg}
0/°F for decreasing T_{avg}

$K_6 \ge 0.00162$/°F when $T > T''$
0/°F when $T \le T''$

$\tau_3 \ge 5$ sec

$\tau_4 \ge 3$ sec

$\tau_5 \le 3$ sec

$f_2(\Delta I) = 0$ for all ΔI.

A/D
DFP-99-0213
CALC POWER RANGE CHECK
A/D
DFP-99-0213
CALC POWER RANGE CHECK
N-411A QL
N-411B QU
SUM
TRIP DELAY
N-411 DELTA I
CALCULATE DELTA I CAL
N-411 DELTA I CAL
CALC FLUX IMBALANCE f1
CALC FLUX IMBALANCE f2
UXIMB f1
T-412 TAVG
T3S
1+T3S
A
N-411 FLUXIMB f2
T-411 TAVGR/L
CALCULATE OVERPOWER SP
E
T-411 OPSP
D/A
DAC-99-0215
D/A
DAC-99-0215
D/A
DFP-99-0212
CALC PRESS. RANGE CHECK
P-455Q PZR PRESS
(RTD CURRENT)
T-412 TAVG
B
TS-68-2K DEENG DECR
TS-68-2J DEENG DECR
1+T1S
1+T2S
T-411 TAVGL/L
CALC OVER-TEMP SP
N-411 FLUXIMB f1
LO TAVG TB-412G
LO-LO TAVG TB-412D
T-411 OTSP
D/D
DDC-99-0214
D/D
DDC-99-0214
D/A
DAC-99-0215
T-411B NR TCOLD1
D/D
The three inputs to OT delta T are Temperature, Pressure and Delta I.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Source Range Neutron Flux (continued)

In MODE 3, 4, or 5 with the reactor shut down, the Source Range Neutron Flux trip Function must also be OPERABLE. If the CRD System is capable of rod withdrawal, the Source Range Neutron Flux trip must be OPERABLE to provide core protection against a rod withdrawal accident. If the CRD System is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. However, their monitoring Function must be OPERABLE to monitor core neutron levels and provide visual indication and audible alarm of reactivity changes that may occur as a result of events like a boron dilution. The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.3, "Nuclear Instrumentation."

6. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include pressurizer pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase. The Overtemperature ΔT trip Function uses each loop's ΔT as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;

- pressurizer pressure — the Trip Setpoint is varied to correct for changes in system pressure; and

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

- axial power distribution - the f(ΔI) Overtemperature ΔT Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation is included for delays associated with fluid transport from the core to the loop temperature detectors (RTDs), and thermowell and RTD response time delays.

ΔT_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured for each loop. T' represents the 100% RTP Tavg value as measured by the plant for each loop. ΔT_0 and T' normalize each loop's ΔT setpoint to the actual operating conditions existing at the time of measurement, thus forcing the setpoint to reflect the equivalent full power conditions as assumed in the accident analyses. Differences in RCS loop ΔT and Tavg can be due to several factors, e.g., measured RCS loop flow greater than minimum measured flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT and Tavg values. Loop specific values of ΔT_0 and T' must be determined at the beginning of each fuel cycle at full power, steady-state conditions (i.e., power distribution not affected by xenon transient conditions) and will be checked quarterly and updated, if required. Tolerances for ΔT_0 and T' have been included in the determination of the Overtemperature ΔT setpoint.

The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature ΔT is indicated in two

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

loops. The pressure and temperature signals are used for other control functions. The actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux—High Setpoint trip.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature — the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and
- rate of change of reactor coolant average temperature — including dynamic compensation for delays associated with fluid transport from the core to the loop temperature detectors (RTDs),and thermowell and RTD response time delays.

ΔT_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured for each loop. T” represents the 100% RTP Tavg value as measured by the plant for each loop. ΔT_0 and T” normalize each loop’s ΔT setpoint to the actual operating conditions existing at the time of measurement, thus forcing the setpoint to reflect the equivalent full power conditions as assumed in the accident analyses. Differences in RCS loop ΔT and Tavg can be due to several factors, e.g., measured RCS loop flow greater than minimum measured flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT and Tavg values. Loop specific values of ΔT_0 and T” must be determined at the beginning of each fuel cycle at full power, steady-state conditions (i.e., power distribution not affected by xenon transient conditions) and will be checked quarterly and updated, if required. Tolerances for ΔT_0 and T” have been included in the determination of the Overtemperature ΔT setpoint.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

The Overpower ΔT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower ΔT is indicated in two loops. The temperature signals are used for other control functions. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE. Note that the Overpower ΔT trip Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel.

(continued)

FROM
TE-68-2B1
TE-68-2B2
TE-68-2B3
(NOTE 1)

FROM
PT-68-63
PT-68-340
(NOTE 1)

FROM
TE-68-14A
TE-68-14B
(NOTE 1)

FROM
TE-68-1
TE-68-18
TE-68-24
TE-68-41
(NOTE 1)

FROM
NIS NEUTRON FLUX
UPPER ION CHAMBER
LOWER ION CHAMBER
(NOTE 1)

RACK 1-R-2 (I)

MMI CART 1,2,3 OR 4
COORD B-1

Turbine Impulse Pressure is used by this Protection Set I rack.
FROM PT-1-20A FT-1-21A FT-3-90A
FROM PT-1-73 LT-63-180
FROM PT-1-27A FT-1-28A FT-3-103A
(NOTE 1)
RACK 1-R-4 (I)
MMI CART 1,2,3 OR 4
1-47W610-99-1, COORD B-1
XHP-99-0401
I/E
EAI-99-0401
TP
EAI-99-0403
EAI-99-0402
XHP-99-0402
BLOW-99-0400A
BLOW-99-0400B
TESTER SUBSYSTEM (NOTE 4)
A/D
DFP-99-0413
PX 99-0400B
(NOTE 3)
XHT-99-0402
LCP-99-0411
PM-1-20A PM-1-27A PM-1-73 FM-3-90A
FM-3-103A LM-63-180 (NOTE 8)
PX 99-0400A
(NOTE 2)
DLH-99-0409
DLH-99-0408
TSP-99-0407
XHI-99-0401
FRONT TEST PANEL
(NOTE 7)
TO FIBER OPTIC CONVERTER 1-R-2
SEC-261-0004
(NOTE 14)
D/A
DAC-99-0415
D/D
DDC-99-0414
HDAC-99-0403
HDAC-99-0404
DDC-99-0405
TS 99-0400
99 7
TO I/O BOARDS (NOTE 5)
E/I
EAO-99-0414
EAO-99-0413
EPT-99-0411
EPT-99-0409
EPT-99-0410
DOOR SWITCH
ECO-99-0412
TO MCR INDICATORS PLANT COMPUTERS PROCESS CONTROL (NOTE 10)
TO SSPS (NOTE 9)
5-18
W
6A-115F
ANN 3-71 NOTE 12
4A-66D
ANN 19-46 BYPASS
6A-111F
ANN 19-45 CHANNEL FAILURE
6A-110F
ANN 19-44 TROUBLE
5-19
NOTE 11

39.

Which ONE of the following describes the power supply to a Reactor Trip Breaker UV Trip Coil?

A Reactor Trip Breaker UV Trip Coil power is supplied by a ________ Board.

A. 48V-DC Plant Battery

B. 125V-Vital DC Battery

C. 120V-AC Preferred Power

D. 120V-AC Vital Instrument Power

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: Plausible since 48 VDC power is used for the UV relays (This 48V power supply provides power for the RPS UV relays which cause a Reactor Trip (52 UV relay in breaker trip logic)) however this supply provides power for plant phone system.

B. Incorrect: Plausible since 125 VDC power is the control power to the Reactor Trip breakers control power.

C. Incorrect: Plausible since 120 VAC is used in RPS logic, however it is 120 VAC vital Instrument power that used for RPS. 120 VAC-Preferred power provides power primarily to secondary 120V control functions

D. Correct: 120V AC is converted to 48V DC in the logic cabinets. This 48V power supply provides power for the RPS UV relays which cause a Reactor Trip (52 UV relay in breaker trip logic)

Question Number: 39

Tier: 2 Group: 1

K/A: 012 Reactor Protection System
K2 Knowledge of bus power supplies to the following:
K2.01 RPS channels, components, and interconnections

Importance Rating: 3.3 3.7

10 CFR Part 55: (CFR: 41.7)

10CFR55.43.b: Not applicable

K/A Match: This question matches the K/A by having testing the knowledge of power supply to portions of the RPS and which of those power supplies are used to cause RTA &/or RTB to trip.

Technical Reference: 1-45W600-99-1

Proposed references to be provided: None

Learning Objective: 3-OT-SYS085A, Control Rod Drive System
13. Given specific plant conditions, ANALYZE the effect that a loss or malfunction
of the following will have on the Rod Control or Rod Position Indicating
Systems:
e. Loss of control power reactor trip breakers

Cognitive Level:
Higher ____
Lower X

Question Source:
New ____
Modified Bank ____
Bank X

Question History: Bank question used on the SQN 2/2010 exam.

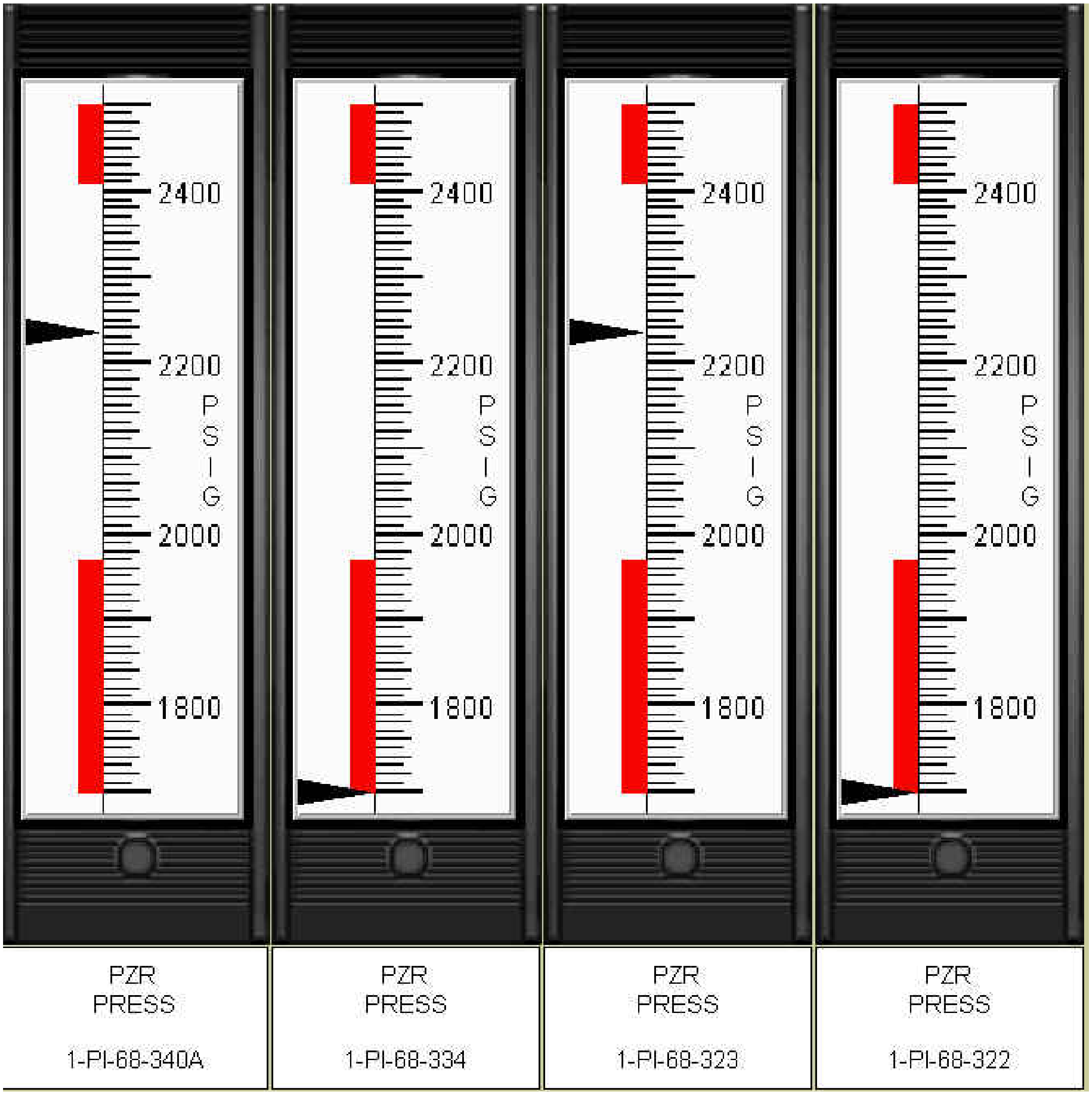
Comments:

125V DC SUPPLY BTRY BD I, BKR 319
ANN "125V DC VITAL BTRY BD I ABNORMAL"
52H/RTA
TTRP1
50605A
1BP1
TURBINE TRIP BUS A (1-45W600-47-2)
SEE CKT BELOW
RELAY X1A (BKR RTA)
FU-99-L116/RTAP
FU-99-L116/RTAN
FU-99-L116/BYAP
FU-99-L116/BYAN
1BP
1BN
1BN1
1BT
[1-RT1] CLOSE
[1-RT1] TRIP (M-4)
[1-RT2] TRIP (M-6)
[1-S1A1] HS-63-133B ACTUATE (M-4)
[1-S1A2] HS-63-133A ACTUATE (M-6)
S1 BLOCK AUTO SHUNT TRIP
STA
52M
52 SR
52Y
LIM SW
52 SH
52a RTA
52b RTA
X5A
X1A
COMPUTER (47A615-0)
Y0006D+
Y0006D-
TURBINE FIRST OUT ANN (1-45W600-47-1) SEE CKT AT RIGHT
1M398
SDFX
RELAY X5A (BKR RTA)
TURBINE TRIP BUS A (1-45W600-47-2)
SEE CKT BELOW
SPARE
RELAY X1A (BKR RTA)
TP3
TP4
BREAKER AUXILIARY CONTACTS
The UV coil receives 48V from the undervoltage driver board of the "A" train SSPS. 120V AC Vital Instrument Boards 1-I and 1-III provide power to the redundant supplies of the "A" train SSPS. These supply power to the undervoltage driver board.
TRAIN A REAC PROT SYS UNDERVOLTAGE TRIP (R-47) (W 1082H70, SH 13)
TEST AUTO SHUNT TRIP
52SH CKT (BKR RTA)
52 UV
50801A
50802A
TP1
TP2
1KΩ
REACTOR TRIP BREAKER RTA (TRAIN A)
125V DC SUPPLY BTRY BD II, BKR 319
ANN "125V DC VITAL BTRY BD II ABNORMAL"
52H/RTB
TTRP2
50605B
1CP1
TURBINE TRIP BUS B (1-45W600-47-2)
RELAY X1B (BKR RTB)
FU-99-L116/RTBP
FU-99-L116/RTBN
FU-99-L116/BYBP
FU-99-L116/BYBN
1CP
1CN
1CN1
1CT
STB
X5B
X1B
Y0007D+
Y0007D-
1M399
SDGX
RELAY X5B (BKR RTB)
TURBINE TRIP BUS B (1-45W600-47-2)
RELAY X1B (BKR RTB)
"B" train is functionally the same as the "A" train.
TRAIN B REAC PROT SYS UNDERVOLTAGE TRIP (R-50) (W 1082H70, SH 13)
52SH CKT (BKR RTB)
50801B
50802B
REACTOR TRIP BREAKER RTB (TRAIN B)

40.

Given the following conditions:

- Unit 1 is at 100%.
- The following is observed:



Which ONE of the following describes the response of SSPS?

SSPS will ________.

A. initiate a Reactor Trip **ONLY**

B. initiate a Safety Injection **ONLY**

C. **NOT** initiate **ANY** actuation signals

D. initiate **BOTH** a Reactor Trip **AND** a Safety Injection

CORRECT ANSWER: A

DISTRACTOR ANALYSIS:

A. Correct: Reactor trip occurs due to 2/4 channels failing low (322 had failed low as given in the stem of the question and now a subsequent channel fails low). No safety injection will occur as it uses channels I,II and III. The SI logic will see 1 of 3 channels meeting the initiation criteria.

B. Incorrect: Reactor trip occurs due to 2/4 channels (322 had failed low as given in the stem of the question). Plausible since, if the original failure had been a different channel (II or III) it would have caused an SI. If the swaps the logic between the safety injection and the reactor trip, that individual may believe that a safety injection could occur but a rx trip would not.

C. Incorrect: As mentioned, the reactor trip logic would be met (2/4 channels). If the applicant believes that both the trip and SI logics were 2 of 3 (using channels I, II and III) then they would arrive at this distractor as the correct.

D. Incorrect: This distractor is incorrect and yet plausible given the assumption that both the trip and safety injection logics were 2 of 4.

Question Number: 40

Tier: 2 Group: 1

K/A: 012 Reactor Protection System
K6 Knowledge of the effect of a loss or malfunction of the following will have on the RPS:
K6.01 Bistables and bistable test equipment

Importance Rating: 2.8 3.3

10 CFR Part 55: (CFR: 41.7 / 45/7)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to understand the impact which is had on the protective system of two different Eagle bistables.

Technical Reference: 1-47W611-99-2
1-47W611-63-1

Proposed references to be provided: None

Learning Objective: 3-OT-TS0303, TS 3.3 “INSTRUMENTATION”
1. DESCRIBE the following aspects of Technical Specifications and Technical Requirements for Instrumentation Tech Specs:
a. The Limiting Conditions for Operation, Applicability, and Bases for the LCO.
b. The conditions and required actions with completion time of one hour or less.

Cognitive Level:
- Higher X
- Lower ____

Question Source:
- New ____
- Modified Bank X
- Bank ____

Question History: Modified from Bank question 013 K5.02 39 which was used on the SQN 09/2010 NRC exam.

Comments:

1. 013 K5.02 039

Given the following plant conditions:

- Unit 1 is at 100% power.
- Pressurizer pressure Channel I, 1-PT-68-340, has been removed from service for surveillance testing with its associated bistables tripped.
- Pressurizer pressure Channel IV, 1-PT-68-322, fails high.

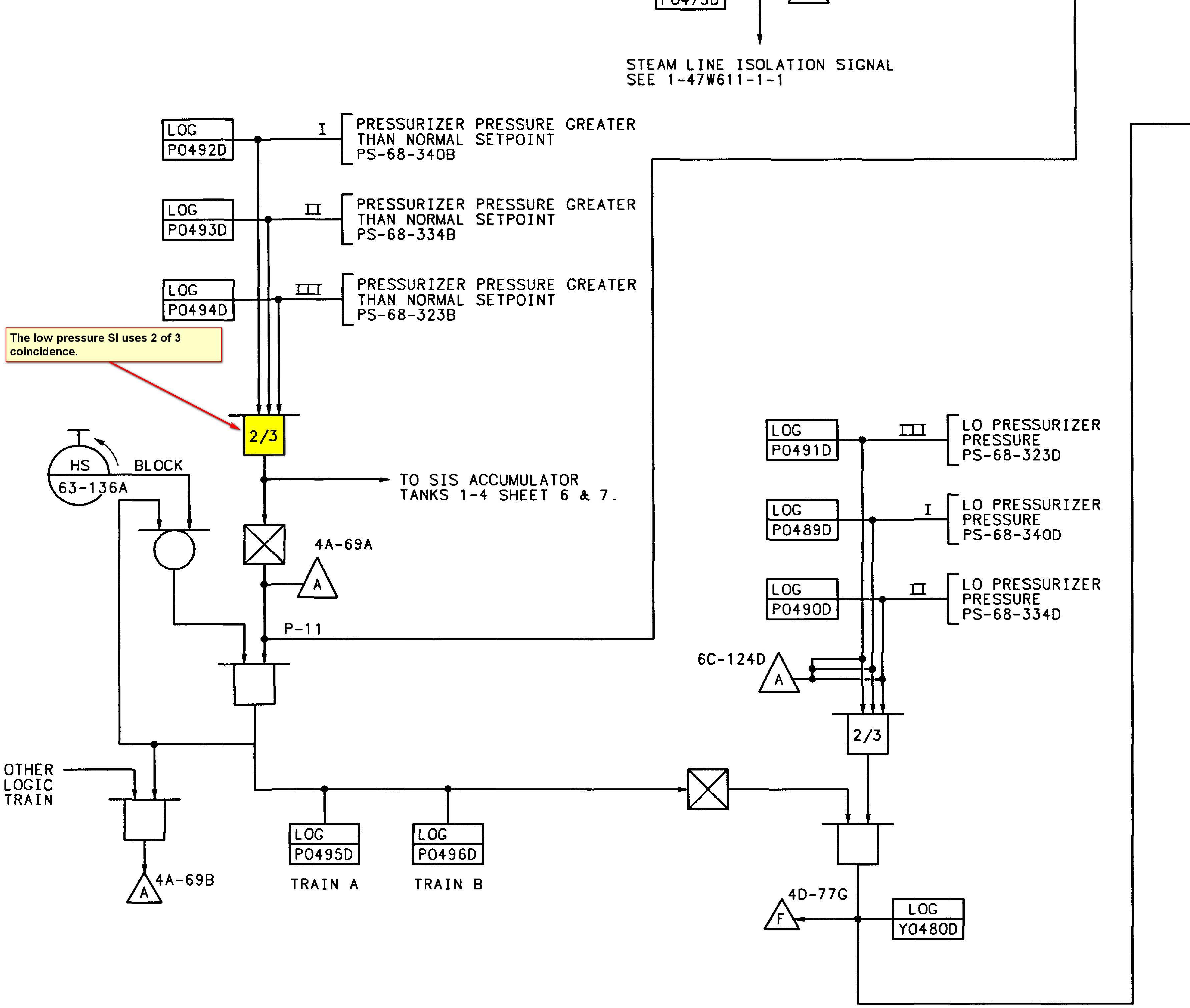
Which ONE of the following describes the immediate result of these conditions?

A. Both Reactor trip and Safety Injection occur.
PZR PORV, 1-PCV-68-340 opens; PZR PORV, 1-PCV-68-334 opens.

B. A Reactor trip but **NO** Safety Injection occurs.
PZR PORV, 1-PCV-68-340 opens; PZR PORV, 1-PCV-68-334 remains closed.

C. Both Reactor trip and Safety Injection occur.
PZR PORV, 1-PCV-68-340 remains closed; PZR PORV, 1-PCV-68-334 opens.

D✓ A Reactor trip but **NO** Safety Injection occurs.
PZR PORV, 1-PCV-68-340 remains closed; PZR PORV, 1-PCV-68-334 remains closed.



PRESSURIZER PRESSURE

PRESSURIZER HIGH PRESSURE
I
II
III
IV
PS 68-340A
PS 68-334A
PS 68-323A
PS 68-322A
LOG P0480D
LOG P0481D
LOG P0482D
LOG P0497D
6C-124B
A
2/4
4D-77C
F
LOG P0483D
PRESSURIZER HIGH PRESSURE REACTOR TRIP 1-47W611-99-1
The High and Low Pzr Pressure trips use 2 of 4 coincidence.
PRESSURIZER LOW PRESSURE
(FIXED SET POINT LEAD/LAG COMPENSATION)
PS 68-340E
PS 68-334E
PS 68-323E
PS 68-322E
LOG P0484D
LOG P0485D
LOG P0486D
LOG P0487D
6C-124C
A
2/4
4D-77D
F
LOG P0488D
PRESSURIZER LOW PRESSURE REACTOR TRIP 1-47W611-99-1

1. 013 K5.02 039

Given the following plant conditions:

- Unit 1 is at 100% power.
- Pressurizer pressure Channel I, 1-PT-68-340, has been removed from service for surveillance testing with its associated bistables tripped.
- Pressurizer pressure Channel IV, 1-PT-68-322, fails high.

Which ONE of the following describes the immediate result of these conditions?

A. Both Reactor trip and Safety Injection occur.
PZR PORV, 1-PCV-68-340 opens; PZR PORV, 1-PCV-68-334 opens.

B. A Reactor trip but **NO** Safety Injection occurs.
PZR PORV, 1-PCV-68-340 opens; PZR PORV, 1-PCV-68-334 remains closed.

C. Both Reactor trip and Safety Injection occur.
PZR PORV, 1-PCV-68-340 remains closed; PZR PORV, 1-PCV-68-334 opens.

D.✓ A Reactor trip but **NO** Safety Injection occurs.
PZR PORV, 1-PCV-68-340 remains closed; PZR PORV, 1-PCV-68-334 remains closed.

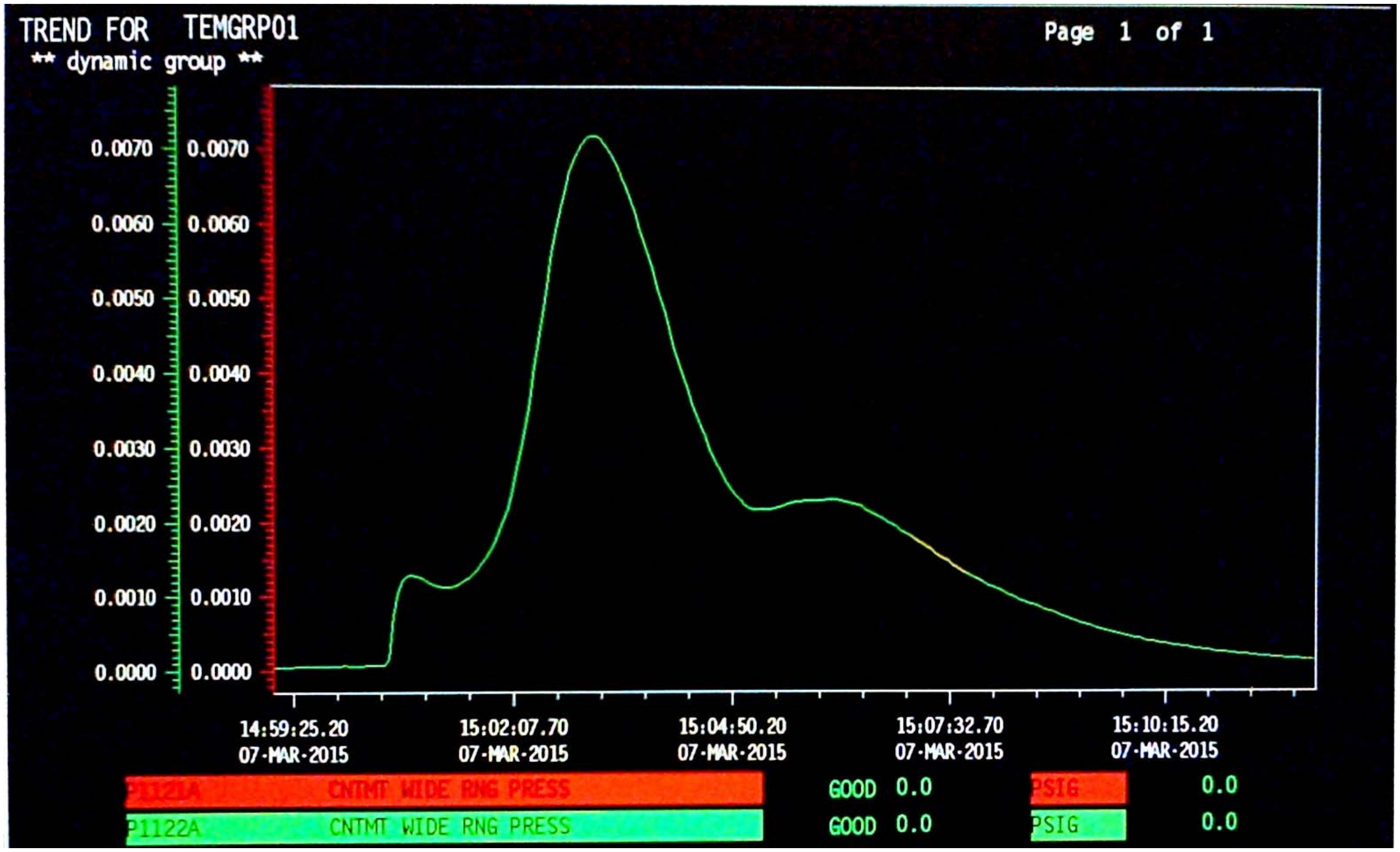
This is the original question.

41.

Given the following conditions:

- Unit 1 is at 100% power.
- ALL Containment Cooling is **FULLY** Operational in accordance with the SOIs.

Which ONE of the following describes an event which causes the following containment pressure transient?



A. Starting an additional lower containment cooler

B. A RCS leak inside of containment

C. Securing one of the running lower containment coolers

D. Instrument Air leak inside containment which does **NOT** result in the isolation of instrument air to containment

CORRECT ANSWER: C

DISTRACTOR ANALYSIS:

A. Incorrect: If an additional lower containment cooler is started, the available cooling capacity available to lower containment is increased. The cooler will immediately act to lower the temperature of lower containment and because of the effect which can be approximated by the ideal gas law, will lower pressure slightly. The originally running LCCs will sense the lowered temperature and will reduce their cooling. Therefore, the temperature will return to that originally seen and as such the pressure will return to that originally seen.

The LCCs may be simplified as a recirculation fan which draws air across a cooling coil. If one believe that the LCCs drew air from somewhere other than the containment atmosphere, then they could reasons that the small pressure spike was due to this initial increased admission (until the other LCCs reduced their air intake flow).

Also, one may understand that the LCCs discharge air at a velocity which will literally blow a person over. Because of this, it is plausible to believe that starting this very power blower will initially cause a pressure spike.

B. Incorrect: A RCS leak inside of containment will cause containment pressure to rise. If the leak is small enough, containment pressure will rise and establish a higher equilibrium value. The equilibrium value is one in which the energy input into the containment atmosphere is in balance with the heat removal capability of the containment coolers. This distractor is plausible if the applicant believes that the containment coolers (through automatic action of their temperature control valves) will cause containment pressure to turn and return to near 0 psig.

C. Correct: When the Lower Containment Cooler A-A is secured, overall cooling to containment is reduced. The temperature and pressure of the containment will rise. The rise in temperature will cause the other running LCCs' Temperature control valves to open further. The remaining running LCCs will restore containment temperature and pressure to their values before the securing of the LCC.

D. Incorrect: An instrument air break inside of containment will cause containment pressure to rise. It would be plausible to believe that this distractor is correct if air had isolated (on a low air header pressure). If the break were large enough to cause the applicable reactor compartment air header to isolate, then the observed containment pressure trend would be similar to that shown in the time plot. Pressure would rise and subsequently be restored through the actions of both the containment pressure reliefs and the annulus vacuum subsystem.

Question Number: 41

Tier: 2 Group: 1

K/A: 022 Containment Cooling System (CCS)
A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including:
A1.02 Containment pressure

Importance Rating: 3.6 3.8

10 CFR Part 55: (CFR: 41.5 / 45.5)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to interpret a time plot of containment pressure (monitor this parameter) and determine the initiating cause of the time plot. Thus, the applicant demonstrates the capability to understand the impact on containment pressure when securing (or operating) a Lower Containment Cooler.

Technical Reference:

Proposed references to be provided: None

Learning Objective: 3-OT-STG-030C, REACTOR BUILDING VENTILATION SYSTEM
7. EXPLAIN the Reactor Building Ventilation System design features and/or interlocks that provide the following:
h. Temperature Indicating Controller settings for control of TCVs

Cognitive Level:
- Higher X
- Lower ____

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

42.

Regarding the following hand-switches:

Located on 1-M-5

Located on 1-M-6

Which ONE of the following describes the logic for a **MANUAL** actuation of a phase B Isolation?

A. The operation of **ANY ONE** of the 4 hand-switches will actuate **BOTH** trains

B. The **SINGLE** operation of either 1-HS-30-64A or 1-HS-30-68A will actuate **ONLY** the A train

C. The **SEQUENTIAL** operation of one pair of hand-switches (either those on 1-M-5 or those on 1-M-6) will actuate **BOTH** trains

D. The **SIMULTANEOUS** operation of one pair of hand-switches (either those on 1-M-5 or those on 1-M-6) will actuate **BOTH** trains

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: It is not correct that the operation of one of the four hand-switches will produce a phase B signal (which in turn yields the containment spray actuation). The plausibility for this distractor is seen if the applicant believes that the manual actuation of phase B is similar to the manual actuation of phase A. To manually actuate phase A, the operator would take either of the two available hand-switches to ACTUATE. This fact may be observed on print 1-47W611-88-1. Also seen on this print is the operation of the manual phase B signal. Phase B requires that two hand-switches be simultaneously taken to Actuate. The fact that the switches be simultaneously positioned is indicated by the arrow appearing above and to the left of the switch symbol. If one switch were taken and then released and then the other taken and then released, the AND circuit downstream of the switches would at no time receive a signal of other than one 0 and one 1. Additionally, the pair of switches operated must be in the same pair. There are two pairs of switches. One pair is on 1-M-5 and one is on 1-M-6. One could not, as seen in print 1-47W611-88-1 simultaneously operate one of the switches on 1-M-5 and one of the switches on 1-M-6 to produce a phase B.

B. Incorrect: Again it is Incorrect that the operation of only one switch would actuate a single train of phase B. The plausibility of this distractor is that this is exactly how two of the other isolation signals function. Specifically, the Control Room Isolation signal has two pairs of hand-switches. One pair is on panel 1-M-6 and the other pair is on panel 2-M-6. Either pair of hand-switches contains an A train and a B train switch. For example the Control Room Isolation switches on 1-M-6 includes 1-HS-31-177A, MCR ISOL TR-A and the switches on 2-M-6 includes 2-HS-31-177A, MCR ISOL TR-A. If either of the two hand-switches (assuming U2 were in operation) were taken to ACTUATE, then a CRI would occur. The other isolation which functions in the manner described above is the Auxiliary Building Isolation. It too has a pair of switches on 1-M-6 and a pair on 2-M-6. By design, the operation of an A train switch on either Unit would cause the appropriate train of ABI to occur. Another method by which this distractor is plausible is the fact that isolation signal resets operate exactly in the manner described above.

C. Incorrect: As previously discussed, the sequential operation of a pair of phase B switches will not cause an actuation. The plausibility of this distractor originates not only from the fact the design of phase B could certainly have stipulated that a sequential operation of two hand-switches be utilized for the signals actuation. This design would certainly have befit the human factors design basis that only single handed operations be allowed. Also, the distractor is plausible as the sequential operation of the Safety Injection reset pushbuttons is required to produce the desired annunciator (and system) response. Take, for example, step 12 of 1-E-1, Loss of Reactor or Secondary Coolant which states: **RESET** SI **AND CHECK** the following: SI ACTUATED permissive DARK AUTO SI BLOCKED permissive LIT. To achieve this step (which appears throughout the Emergency procedure set), the operator will push the A train SI reset button, check that the Annunciator 70-B, AUTO SI BLOCKED is LIT, push the B train SI reset button, and finally check that Annunciator 70-A, SI ACTUATED is DARK. This sequential operation of safeguards switches is practiced time and time again.

D. Correct: As discussed the simultaneous operation of either pair of hand-switches will cause the phase B actuation.

Question Number: 42

Tier: 2 Group: 1

K/A: 026 Containment Spray System (CSS)
A4 Ability to manually operate and/or monitor in the control room:
A4.01 CSS controls

Importance Rating: 4.5 4.3

10 CFR Part 55: (CFR: 41.7 / 45.5 to 45.8)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to demonstrate the capability to operate the phase B controls in the main control room. The phase B controls are those which initiate a containment spray signal.

Technical Reference: 1-47W611-88-1

Proposed references to be provided: None

Learning Objective: 3-OT-TS0303, TS 3.3 “INSTRUMENTATION”
1. DESCRIBE the following aspects of Technical Specifications and Technical Requirements for Instrumentation Tech Specs:
a. The Limiting Conditions for Operation, Applicability, and Bases for the LCO.
b. The conditions and required actions with completion time of one hour or less.

Cognitive Level:
- Higher X
- Lower ____

Question Source:
- New ____
- Modified Bank X
- Bank ____

Question History: Modified bank question 026 A4.01 43 which was used on Sequoyah’s 09/2010 NRC exam.

Comments:

QUESTIONS REPORT

for SQN-WBN composite exams

1. 026 A4.01 043

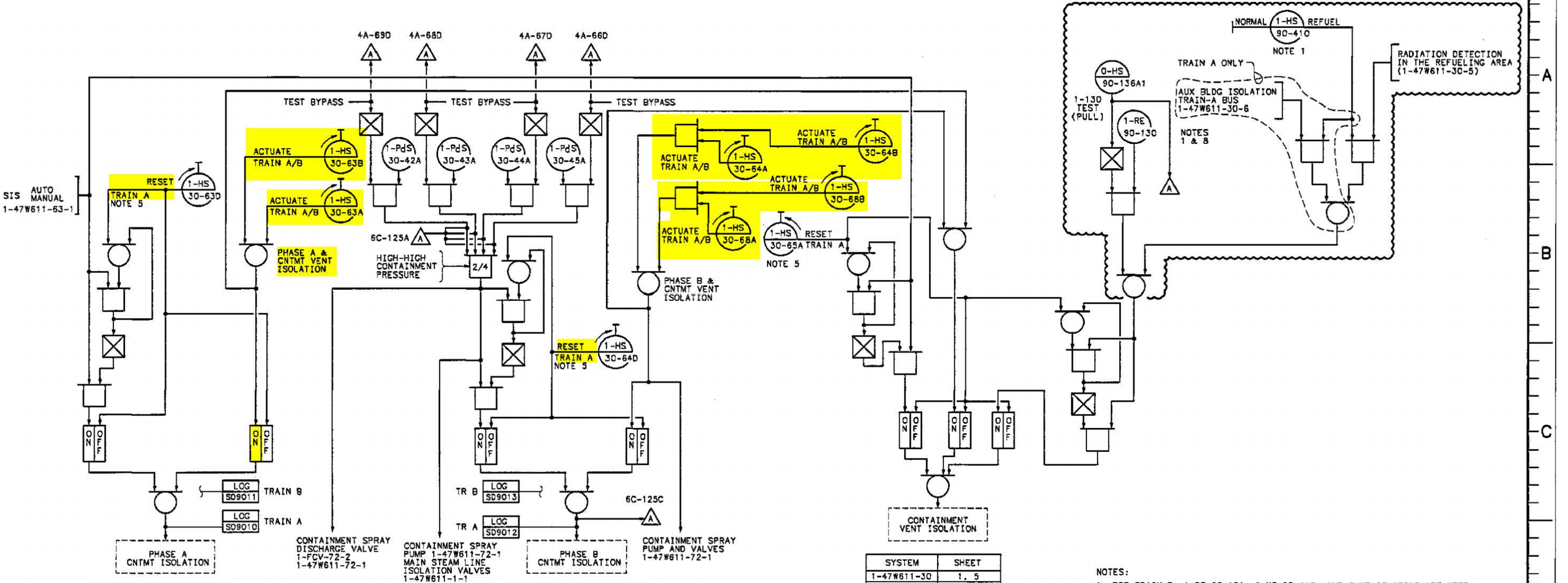
Which ONE of the following is correct regarding the logic for MANUAL actuation of a phase B Isolation?

A. Operation of any one of 4 handswitches will actuate both trains.

B. Operation of either 1-HS-30-64A (M-6) or 1-HS-30-68A (M-5) will actuate train A ONLY.

C. Operation of the paired handswitches (M-5 or M-6) operated sequentially will actuate both trains.

D.✓ Operation of the paired handswitches (M-5 or M-6) operated simultaneously will actuate both trains.



SYSTEM	SHEET
1-47W611-30	1, 6
1-47W611-61	2
1-47W611-62	1
1-47W611-63	3, 5, 6, 8
1-47W611-65	1, 2, 3
1-47W611-68	1
1-47W611-70	3
1-47W611-77	1, 5
1-47W611-81	1
1-47W611-1	3
1-47W611-31	7

SYSTEM	SHEET
1-47W611-30	3, 4
1-47W611-70	3
1-47W611-32	2
1-47W611-67	3

SYSTEM	SHEET
1-47W611-30	1, 5
1-47W611-88	1

NOTES:

1. FOR TRAIN B, 1-RE-90-131, 1-HS-90-415, AND 0-HS-90-136A2 ARE USED. 0-HS-90-136A1 & 136A2 ARE MULTI SELECT SW'S USED TO TEST ONE CHANNEL AT A TIME.

3. ALL ISOLATION SIGNALS AND CIRCUITRY THAT ARE PART OF ENGINEERED SAFEGUARDS ARE REDUNDANT.

4. NOTE DELETED.

5. EACH TRAIN A RESET HAS AN ASSOCIATED TRAIN B RESET. SEE TVA DRAWING 1-47W610-30-1 FOR TRAIN B "RESET" HANDSWITCH NUMBERS.

6. VALVES 1-FCV-43-22 AND 1-FCV-43-23 MAY BE NORMALLY OPEN WITH 1-FSV-43-23 ENERGIZED TO ALLOW CONTINUOUS RCS FLOW DUE TO THERMAL CYCLING CONCERNS (REF. EDC-51134).

QUESTIONS REPORT

for SQN-WBN composite exams

1. 026 A4.01 043

Which ONE of the following is correct regarding the logic for MANUAL actuation of a phase B Isolation?

A. Operation of any one of 4 handswitches will actuate both trains.

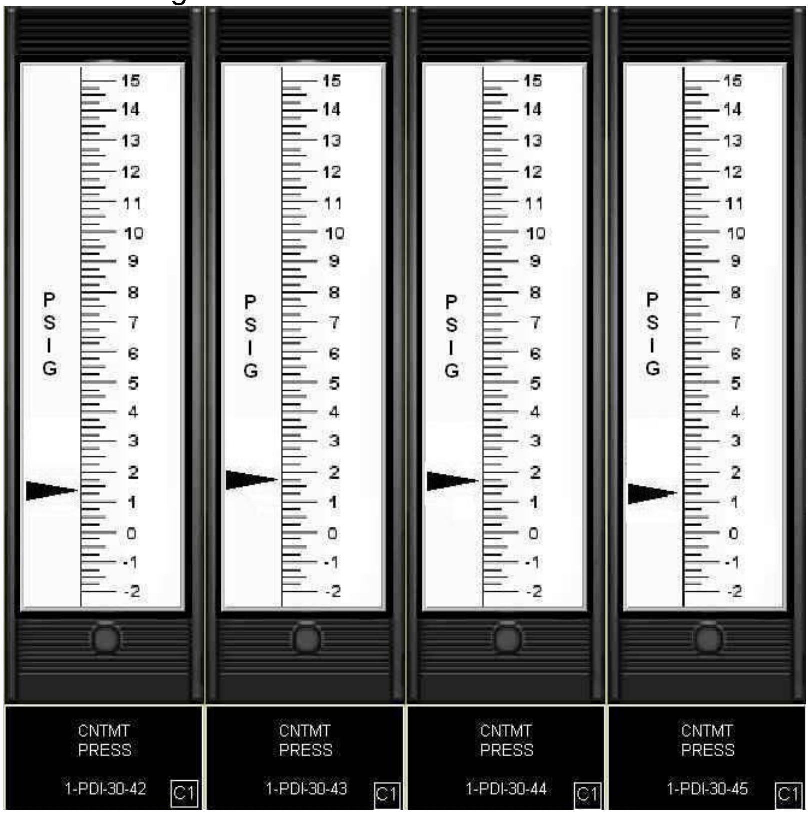
B. Operation of either 1-HS-30-64A (M-6) or 1-HS-30-68A (M-5) will actuate train A ONLY.

C. Operation of the paired handswitches (M-5 or M-6) operated sequentially will actuate both trains.

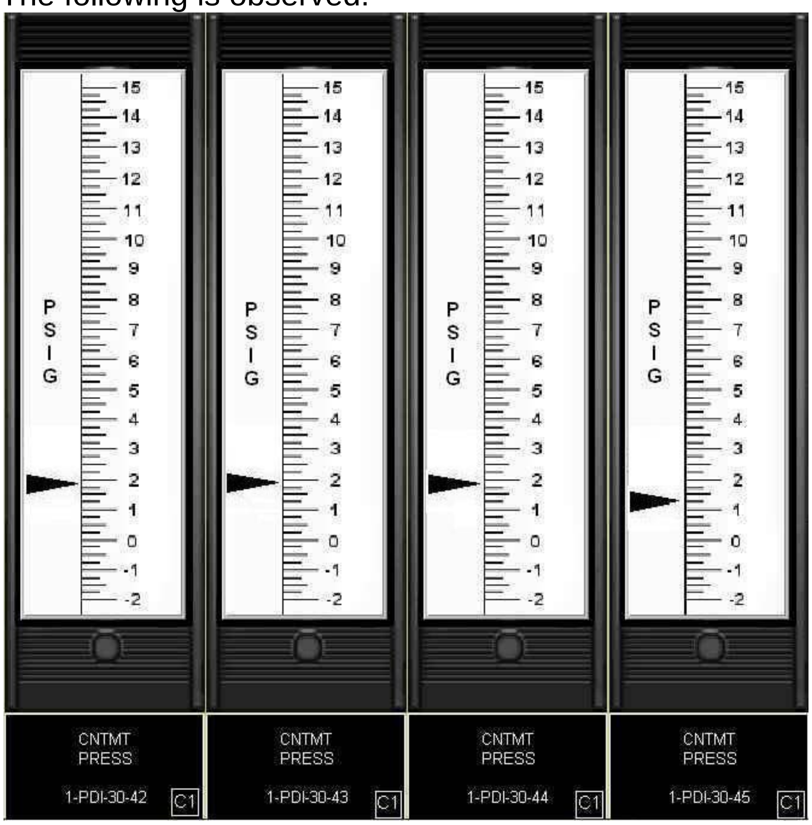
D.✓ Operation of the paired handswitches (M-5 or M-6) operated simultaneously will actuate both trains.

43. Given the following timeline:

00:00:00 Unit 1 is at 100% power.
00:01:00 A LOCA occurs.
00:02:00 The following is observed:

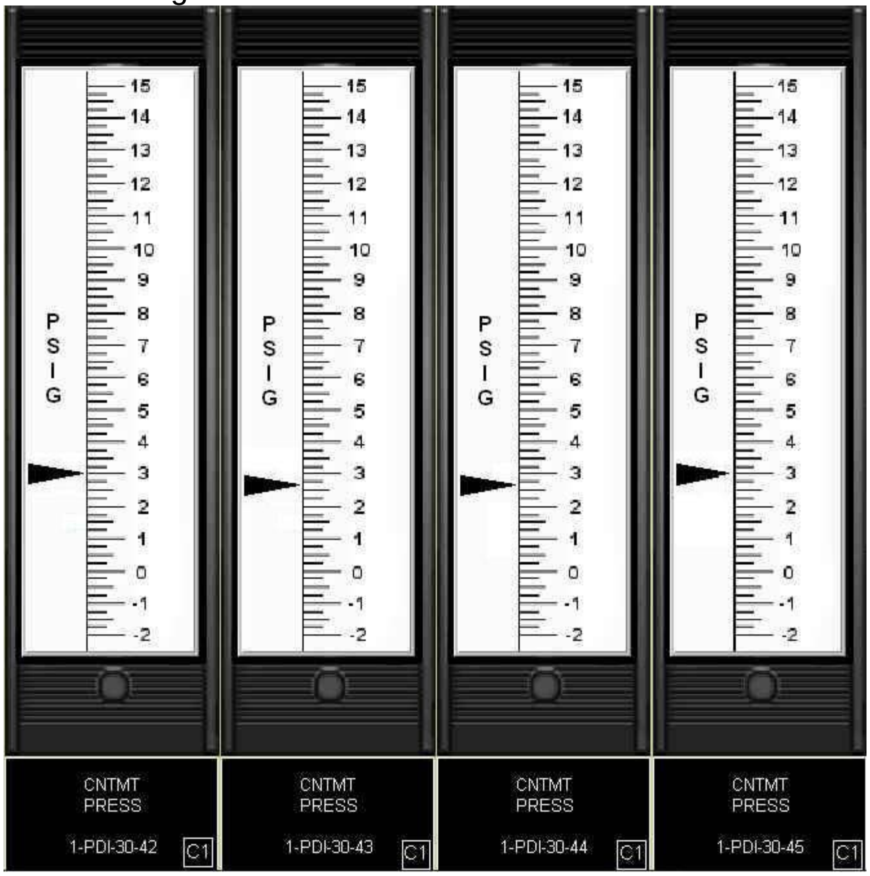


00:03:00 The following is observed:

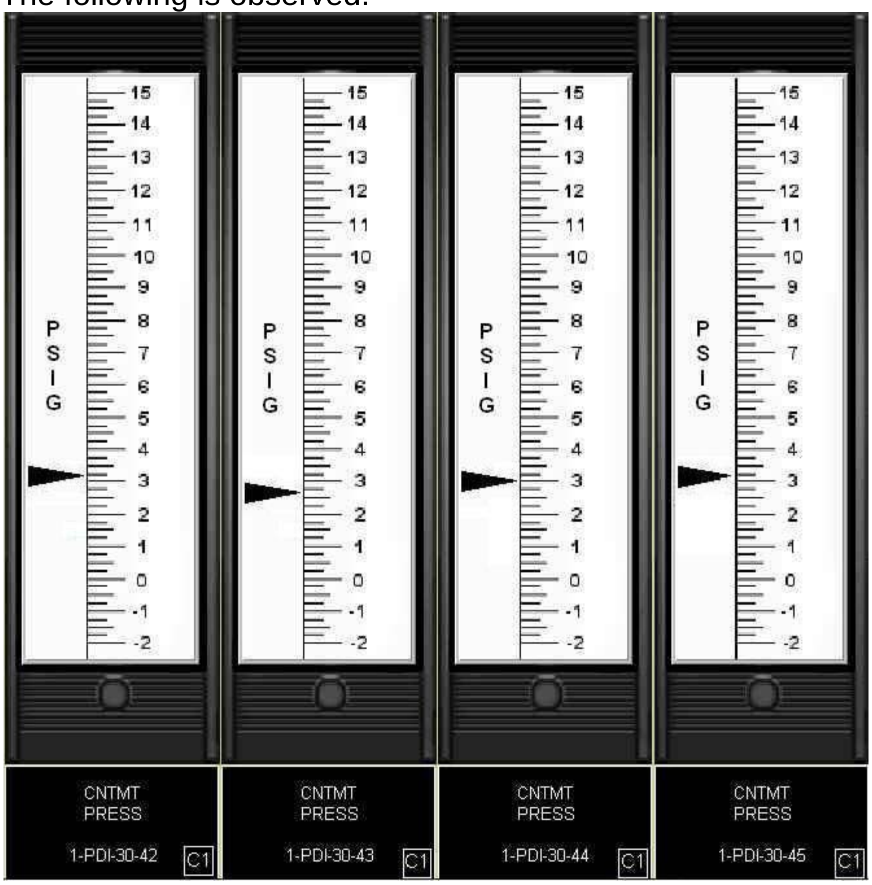


< QUESTION CONTINUED ON THE NEXT PAGE >

00:04:00 The following is observed:



00:05:00 The following is observed:



< QUESTION CONTINUED ON THE NEXT PAGE >

From the times and indications listed above, which ONE of the following completes the statement listed below?

The MSIVs would **FIRST** be expected to AUTOMATICALLY CLOSE at __________.

A. 00:02:00

B. 00:03:00

C. 00:04:00

D. 00:05:00

CORRECT ANSWER: C

A. Incorrect: This distractor presents two of the four channels of containment pressure in excess of the High containment pressure setpoint of 1.5 psig. While the coincidence is correct, the setpoint of 1.5 is incorrect. As seen in T/S table 3.3.2-1 (this value may be found in many other references), the containment pressure-High High setpoint applicable to the automatic steam line isolation is 2.8 psig. The Containment High setpoint of 1.5 would be applicable to the automatic safety injection.

B. Incorrect: This distractor presents three of the four channels of containment pressure in excess of the High containment pressure setpoint of 1.5 psig. Again, both the coincidence required and the setpoint are incorrect.

C. Correct: This distractor presents two of the four channels of containment pressure in excess of the High-High containment pressure setpoint of 2.8 psig. As seen on print 1-47W611-88-1, the coincidence for the automatic steam line isolation is 2 of 4.

D. Incorrect: This distractor presents three of the four channels of containment pressure in excess of the High-High containment pressure setpoint of 2.8 psig. While the MSIVs would be expected to be closed at this time, they would have first closed at 0400.

Question Number: 43

Tier: 2 Group: 1

K/A: 039 Main and Reheat Steam System
A3 Ability to monitor automatic operation of the MRSS, including:
A3.02 Isolation of the MRSS

Importance Rating: 3.1 3.5

10 CFR Part 55: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: The K/A is matched because the applicant must be able to monitor for the automatic isolation of the Main Steam System during a casualty which causes containment pressure to rise.

Technical Reference: 1-47W611-88-1
T/S Table 3.3.2-1, Engineered Safety Feature Actuation System Instrumentation

Proposed references to be provided: None

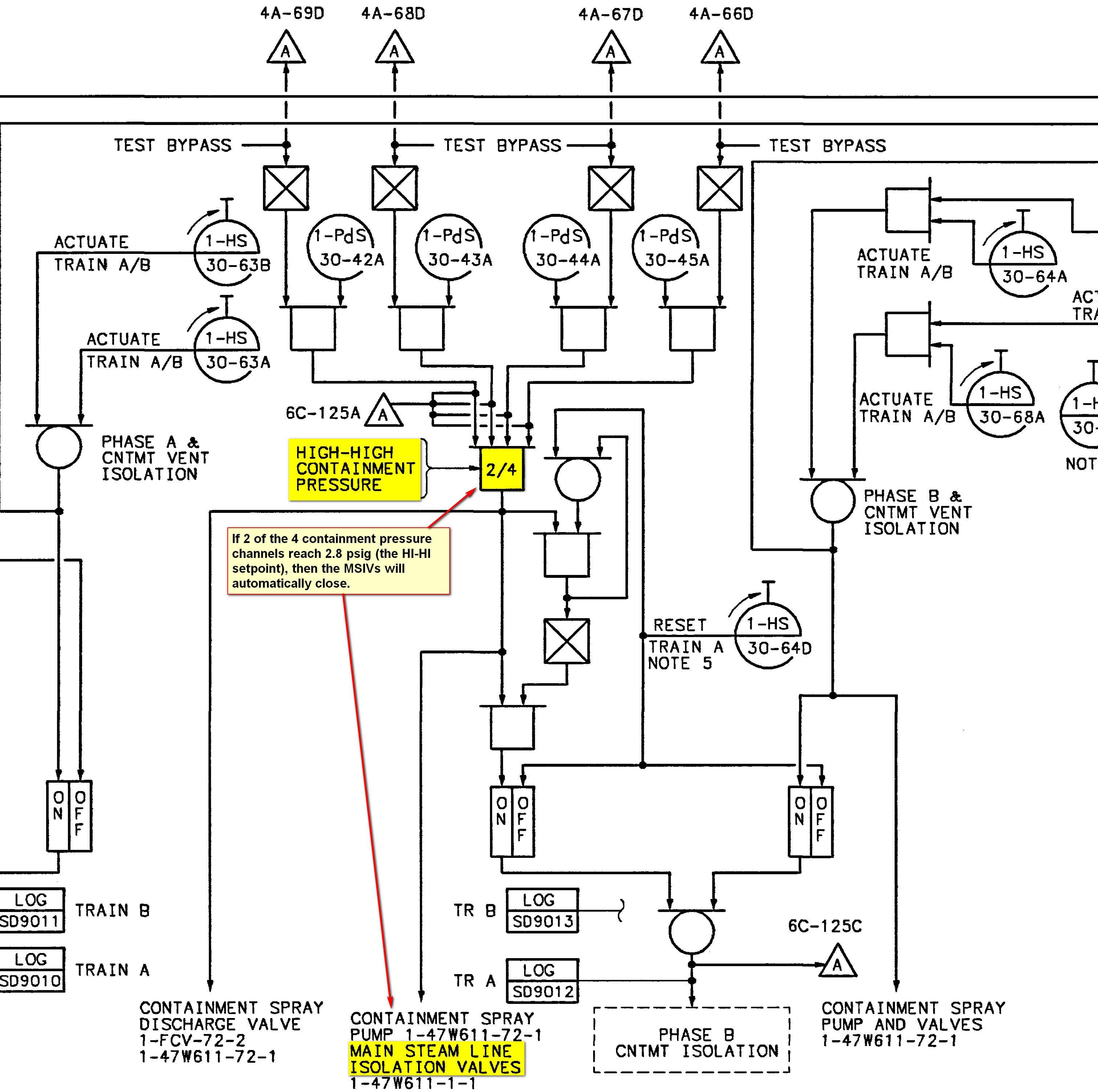
Learning Objective: 3-OT-TS0303, TS 3.3 “INSTRUMENTATION”
1. DESCRIBE the following aspects of Technical Specifications and Technical Requirements for Instrumentation Tech Specs:
a. The Limiting Conditions for Operation, Applicability, and Bases for the LCO.
b. The conditions and required actions with completion time of one hour or less

Cognitive Level:
Higher X
Lower

Question Source:
New X
Modified Bank
Bank

Question History: New question for the 2015-301 NRC RO Exam

Comments:



SYSTEM	SHEET
1-47W611-30	3, 4
1-47W611-70	3
1-47W611-32	2
1-47W611-67	3

Table 3.3.2-1 (page 3 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
4. Steam Line Isolation (continued)						
c. Containment Pressure-High High	1, $2^{(c)}$, $3^{(c)}$	4	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≤ 2.9 psig	2.8 psig
d. Steam Line Pressure						
(1) Low	1, $2^{(c)}$, $3^{(a)(c)}$	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR .3.2.10	≥ $666.6^{(b)}$ psig	$675^{(b)}$ psig
(2) Negative Rate-High	$3^{(d)(c)}$	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≤ $108.5^{(e)}$ psi	$100^{(e)}$ psi
5. Turbine Trip and Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1, $2^{(f)}$, $3^{(f)}$	2 trains	H	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. SG Water Level-High High(P-14)	1, $2^{(f)}$, $3^{(f)}$	3 per SG	I	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10 $^{(h)}$	≤ 83.1%	82.4%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. North MSV Vault Room Water Level - High	1, $2^{(f),(g)}$	3/vault Room	O	SR 3.3.2.6 SR 3.3.2.9	≤ 5.31 inches	4 inches
e. South MSV Vault Room Water Level - High	1, $2^{(f),(g)}$	3/vault Room	O	SR 3.3.2.6 SR 3.3.2.9	≤ 4.56 inches	4 inches

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.
(b) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.
(c) Except when all MSIVs are closed and de-activated.
(d) Function automatically blocked above P-11 (Pressurizer Interlock) setpoint and is enabled below P-11 when safety injection on Steam Line Pressure Low is manually blocked.
(e) Time constants utilized in the rate/lag controller are t_3 and $t_4 \geq 50$ seconds.
(f) Except when all MFIVs, MFRVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.
(g) MODE 2 if Turbine Driven Main Feed Pumps are operating.
(h) For the time period between February 23, 2000, and prior to turbine restart (following the next time the turbine is removed from service), the response time test requirement of SR 3.3.2.10 is not applicable for 1-FSV-47-027.

44.

Given the following conditions:

- Unit 1 is at 100% power.

Subsequently:

- EHC POWER FAILURE (72-C) is FLASHING.
- The light above 1-HS-47-24, TURBINE TRIP is GREEN.
- Unit 1 is in MODE 3.
- The position of SG 2 MFW isol vlv is **UNKNOWN**.

Which ONE of the following describes T/S LCO 3.3.3, Post Accident Monitoring (PAM) Instrumentation AND when the valves go CLOSED in reference to the picture above?

In accordance with T/S LCO 3.3.3, actions ____(1)____ required to be taken.

The valves received a closure signal when RCS Tavg dropped to less than ____(2)____ °F.

	(1)	(2)
A.	are	550
B.	are	564
C.	are **NOT**	550
D.	are **NOT**	564

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: As seen in Table 6.2.4-1 of the stations Final Safety Analysis Report, containment penetration is in part isolated by motor operated valve 3-47. Therefore, this valve is a containment isolation valve (contained within system 3, Main Feedwater). In accordance with T/S LCO 3.3.3, Post Accident Monitoring (PAM) Instrumentation, whenever One or more Functions with one channel [is] inoperable, then action A.1 must be completed. This action states to Restore required channel to Operable status and has a completion time of 30 days. The function which has the inoperable component in this case is function 11-Containment Isolation Valve Position- which is found in Table 3.3.3-1 of the Unit 1 technical specifications. Therefore is is correct that the loss of the valve position indication in the question is a failure to meet T/S LCO 3.3.3. The stem of the question presents a normal and uncomplicated reactor trip. The reactor trip resulted from a Turbine Trip which was caused by an EHC power failure (this sequence of events may be substantiated using 1-ARI-71-75, Turbine Trip First Out.). A normal and uncomplicated reactor trip will always result in a Feedwater Isolation because the automatic steam dump system will always stabilize RCS Tavg at a value less than 564°F. A reactor trip coupled with Tavg less than 564°F will produce a Feedwater isolation. The value of 564°F may be found (in addition to many other places) in Nuclear Engineering Scaling and Setpoint Document SSD-1-T-68-2. Therefore, it is Incorrect to select 550°F for this setpoint. It is plausible to do so as 550°F is the Lo Lo Tavg (P-12) setpoint. Lo Lo Tavg, as seen in 1-ARI-64-70, Bypass Intlk, & Permissive, is the steam dump block and even though its name implies a more severe action (the Lo Lo versus Lo), it simply causes the steam dumps to close versus actuating a Feed Water Isolation.

B. Correct: Again, it is correct that T/S LCO actions are required and that the setpoint for Feedwater isolation is 564°F.

C. Incorrect: It is not correct that the conditions listed do not incur T/S LCO required actions. It is plausible to believe so for several reasons. Firstly, if the Unit had stabilized in Mode 4 (when the failure of indication occurred), then Table 3.3.3-1 would demonstrate that no actions were required as the Modes of applicability are seen to be 1,2 and 3. Secondly, it is plausible to believe that the valve with the failed position indication is not a containment isolation valve and thus not included in the T/S LCO. A good example of this would be the Main Feed Regulating valve which is in series with 1-ISV-3-47. The MFRV also closes on a FWI but is not a containment isolation valve. Also, as discussed it is Incorrect and yet plausible that the FWI would occur at 550°F.

D. Incorrect: While it is correct that the FWI occurs at 564°F, it is not correct and yet plausible that T/S LCO actions are not required.

Question Number: 44

Tier: 2 Group: 1

K/A: 059 Main Feedwater (MFW) System
G2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Importance Rating: 3.9 4.6

10 CFR Part 55: (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to recognize that a failure of the position indication for a main feed water valve represents entry-level conditions for the unit's Technical Specifications..

Technical Reference: 1-ARI-71-75, Turbine Trip First Out
1-ARI-64-70, Bypass, Intlk, & Permissive
SSD-1-T-68-2
FSAR Table 6.2.4-1, Containment Penetrations and Barriers
T/S LCO 3.3.3, Post Accident Monitoring (PAM) Instrumentation

Proposed references to be provided: None

Learning Objective: 3-OT-TS0303, TS 3.3 "INSTRUMENTATION"
1. DESCRIBE the following aspects of Technical Specifications and Technical Requirements for Instrumentation Tech Specs:
a. The Limiting Conditions for Operation, Applicability, and Bases for the LCO.
b. The conditions and required actions with completion time of one hour or less.

Cognitive Level:
Higher X
Lower ____

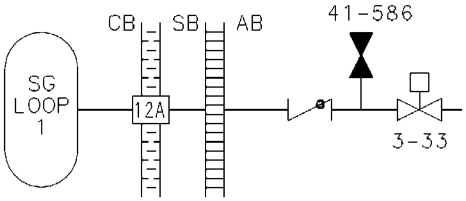
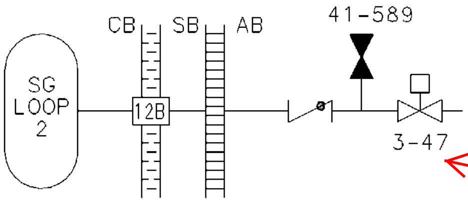
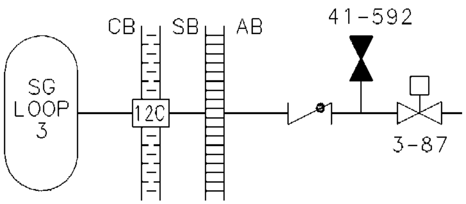
Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

UFSAR - Amendment 1

PENETRATION DATA							VALVE DATA										VALVE STATUS									
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 731 AZ 7°40'	47W803-1 47W862-2	57	W	H	BC E	FEEDWATER (3) & (41)	AB AB	33 586	A -	GA GA	MO M	AT LM	LM -	FW -	- -	O C	C V	C C	AI AI	C C	Y N	N N	A A	N	MK70	22,24
 EL 731 AZ 172°20'	47W803-1 47W862-2	57	W	H	BC E	FEEDWATER (3) & (41)	AB AB	47 589	B -	GA GA	MO M	AT LM	LM -	FW -	- -	O C	C V	C C	AI AI	C C	Y N	N N	A A	N	MK69	22,24
 EL 731 AZ 187°40'	47W803-1 47W862-2	57	W	H	BC E	FEEDWATER (3) & (41)	AB AB	87 592	A -	GA GA	MO M	AT LM	LM -	FW -	- -	O C	C V	C C	AI AI	C C	Y N	N N	A A	N	MK68	22,24

This is the valve whose position indication is affected.

3.3 INSTRUMENTATION

3.3.3 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3 The PAM instrumentation for each Function in Table 3.3.3-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.3-1.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE-------------- Not applicable to Functions 3, 4, 14, and 16. ------------------------------- One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.9.8.	Immediately

(continued)

Table 3.3.3-1 (page 1 of 2)
Post Accident Monitoring Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS/TRAINS	CONDITION REFERENCED FROM REQUIRED ACTION D.1
1.	Intermediate Range Neutron Flux(g)	1[(a)], 2[(b)], 3	2	E
2.	Source Range Neutron Flux	2[(c)] , 3	2	E
3.	Reactor Coolant System (RCS) Hot Leg Temperature (T-Hot)	1,2,3	1 per loop	E
4.	RCS Cold Leg Temperature (T-Cold)	1,2,3	1 per loop	E
5.	RCS Pressure (Wide Range)	1,2,3	3	E
6.	Reactor Vessel Water Level [(f) (g)]	1,2,3	2	F
7.	Containment Sump Water Level (Wide Range)	1,2,3	2	E
8.	Containment Lower Comp. Atm. Temperature	1,2,3	2	E
9.	Containment Pressure (Wide Range) (g)	1,2,3	2	E
10.	Containment Pressure (Narrow Range)	1,2,3	4	E
11.	Containment Isolation Valve Position (g)	1,2,3	2 per penetration flow path [(d)(i)]	E
12.	Containment Radiation (High Range)	1,2,3	2 upper containment 2 lower containment	F
13.	RCS Pressurizer Level	1,2,3	3	E
14.	Steam Generator (SG) Water Level (Wide Range)[(g)]	1,2,3	1/SG	E
15.	Steam Generator Water Level (Narrow Range)	1,2,3	3/SG	E
16.	AFW Valve Status [(j)]	1,2,3	1 per valve	E
17.	Core Exit Temperature-Quadrant 1[(f)]	1,2,3	2 [(e)]	E

(continued)

72-C

Source

DC Bus Trip

EHC POWER FAILURE

(Page 1 of 1)

Probable Cause:

A. Loss of ±15V dc power supply (Set A or Set B)
B. Loss of 48V dc power supply (Set A or Set B)

Corrective Action:

[1] **IF** greater than or equal to 50% power (P-9), **THEN**
VERIFY Turbine Trip/Rx Trip, **AND**
GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION.

[2] **IF** less than 50% power (P-9), **THEN**
VERIFY Turbine Trip, **AND**
GO TO 1-AOI-17, TURBINE TRIP.

References:

1-45W600-47-2
1-45W600-55-12
WBN-VTM-W120-2872
1-AOI-17
1-E-0

SCALING and SETPOINT DOCUMENT
ACCURACY & CALIBRATION SECTION

LOOP ID:	DELTA T/T AVG LOOP 1 PS I
PROCESS RANGE:	530.0 to 650.0°F - Narrow Range THot 510.0 to 630.0°F - Narrow Range TCold 0 to 150% power - Delta T 530.0 to 630.0°F - T Avg 1700 to 2500 psig - Pressurizer Pressure 0 to 60 PU - NIS Upper/Lower Flux
SAFETY RELATED:	Yes. Up to and including the isolators (EAO boards or SSPS). The peripheral devices are categorized as follows: All MCR indicators - quality-related; Plant Computer log points and MCR annunciators, non-safety related; XS-68-2B and TR-68-2A - quality related; Compliance: See TI-49 (Reference 56).
LOOP FUNCTION:	Perform measurement and processing for Delta and Average Temperature. Provides Comparator Outputs to Low TAvg (Feedwater Isolation), Low-Low TAvg (Permissive P-12), Overtemperature and Overpower Delta-T Reactor Trip, Turbine Runback and Block Rod Withdrawal. Also provides Analog Outputs to the LOG (Plant Computer), Temperature Recorder and to the Main Control Board Indicators and to the ICCM. (8)

R6

LOOP ACCURACY

Comparator Functions	**Safety Analysis Limit**	**Setpoint**	**Tech Spec Allowable Value**
Lo TAvg (FW Isolation)	(1)	564.0°F (1) (13)	(1) (14)
Lo Lo TAvg (P-12)	(1)	550.0°F (1) (13)	(1) (14)
Overtemp Delta T Trip	FSAR Figure 15.1-1 (2)	0 PU (2) (13)	(2) (14)
OverTemp Turb Rbk & Blk Rod Withdrawal	(1)	OT Delta T trip setpoint - 1 PU (1) (13)	(1) (14)
Overpower Delta T Trip	FSAR Figure 15.1-1 (3)	0 PU (3) (13)	(3) (14)
Overpower Turb Rbk & Blk Rod Withdrawal	(1)	OP Delta T trip setpoint - 1 PU (1) (13)	(1) (14)
Loop Indication Functions	**Technical Specification/** TRM/ODCM Sections Applicable **LCO Value**	**Main Control Board Indication Uncertainty**	
RCS Average Temperature	≤ 593.2°F (LCO 3.4.1) (21)	N/A (21)	
RCS Average Temperature	≥ 551°F (LCO 3.4.2)	± 3.6°F (22)	

68-A

P-12
LO-LO TAVG
STM DUMP BLOCK

(Page 1 of 1)

Source	**Setpoint**
1-TS-68-2J (Loop 1)	Tavg in 2/4 loops less than or equal to 550 °F
1-TS-68-25J (Loop 2)	
1-TS-68-44J (Loop 3)	
1-TS-68-67J (Loop 4)	

This causes the steam dumps to initially close and be restricted from opening.

Probable Cause: A. Plant cooldown

NOTE

The P-12 interlock prevents opening of the steam dump valves if RCS temperature is less than 550 °F. This interlock can be bypassed for 3 of 12 steam dump valves. P-12 light also indicates steam dump valves should be closed.

Corrective Action:

[1] **VERIFY** Tavg less than or equal to 550 °F.

[2] **IF** bypass of LO-LO TAVG interlock is desired, **THEN** **REFER TO** 1-SOI-1.02, STEAM DUMP *SYSTEM.*

References:

W 7246D11 Sht. 40
1-47W611-63-1
1-SOI-1.02

45.

Given the following conditions:

- Unit 1 is at 3% power.
- The MDAFWPs are in service supplying the SGs.
- 1-POS-3-164, positioner to 1-LCV-3-164, air output FAILS to 0 psig.
- 1-POS-3-164A, positioner to 1-LCV-3-164A, air output FAILS to 0 psig.

Which ONE of the following describes the response of the NORMAL AND BYPASS Level Control Valves?

The **NORMAL** level control valve 1-LCV-3-164 will ____(1)____.

The **BYPASS** level control valve 1-LCV-3-164A will ____(2)____.

NOTE: *1-LCV-3-164, MD AFW PUMP 1A-A SG 1 LEVEL CONTROL*
1-LCV-3-164A, SG 1 AUX FEEDWATER 1-FCV-3-164 BYPASS

	(1)	(2)
A.	OPEN	OPEN
B.	OPEN	CLOSE
C.	CLOSE	OPEN
D.	CLOSE	CLOSE

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: As seen in SSD-1-LPL-3-164 (the normal level control valve), 1-LCV-3-164 fails open on a loss of control air and 1-LCV-3-164A (the bypass level control valve) fails closed on a loss of control air. The purpose of the valve positioners in this case is to receive a pneumatic signal from 1-LM-3-164A and translate that into the appropriate pneumatic output signal to control the valve actuator. Therefore, when the positioners fail and output 0 psig, the valves in effect suffer a loss of control air and travel to their failed position. It is plausible to believe that 1-LCV-3-164A fails open as its sister valve does (as well as many other air operated valves).

B. Correct: As mentioned the failure positions of both valves is correct.

C. Incorrect: It is neither correct; that 1-LCV-3-164 will close, nor that 1-LCV-3-164A will open. It is plausible to believe this for the aforementioned rationale. An item of note is that it is completely plausible that an AFW level control valve would fail closed as this is the case with the TDAFWP's LCVs. On a loss of control air, the TDAFWP's LCVs fail closed.

D. Incorrect: While it is correct that 1-LCV-3-164A will close, it is not correct and yet plausible that 1-LCV-3-164 will close.

Question Number: 45

Tier: 2 Group: 1

K/A: 061 Auxiliary / Emergency Feedwater (AFW) System
K6 Knowledge of the effect of a loss or malfunction of the following will have on the AFW components:
K6.01 Controllers and positioners
(CFR: 41.7 / 45.7)

Importance Rating: 2.5 2.8

10 CFR Part 55:

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to understand the effect on the AFW level control valves 1-LCV-3-164 and 1-LCV-3-164A that failures of the associated positioners have

Technical Reference: SSD-1-LM-3-164A

Proposed references to be provided: None

Learning Objective: 3-OT-SYS003B, Auxiliary Feedwater System
12. Given specific plant conditions, ANALYZE the effect that a loss or malfunction of the following will have on the Auxiliary Feedwater System:
a. Instrumentation failures
b. Control Air

Cognitive Level:
Higher ____
Lower X

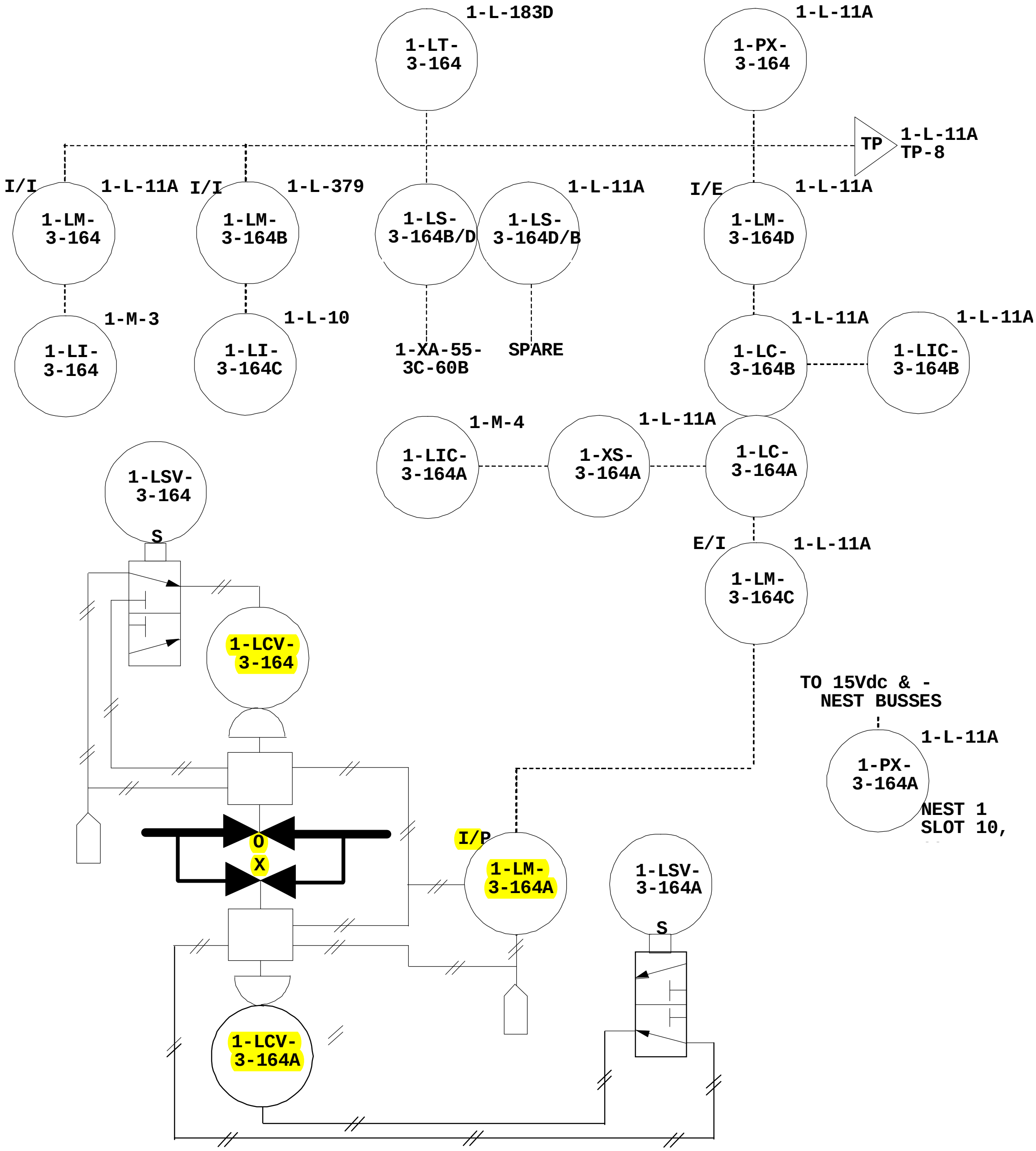
Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

SCALING AND SETPOINT DOCUMENT
LOOP DRAWING

LOOP NAME: STEAM GENERATOR 1 MOTOR DRIVEN AFW LEVEL CONTROL



Reviewed by: R. S. Henderson Approved by: K. D. Garrison Date: 10/13/2009

SCALING AND SETPOINT DOCUMENT
PROCESS CONTROL VALVES AND CONTROL DEVICES

Instrument No.: 1-LCV-3-164 Mfrr: Masoneilan Model:37-20721
Vendor ID No: N/A Contract No: 83577

Function: CV
Valve Action: Air to Close

Positioner Mfrr: Masoneilan Model No: 7432-713 Cam Type: U/A

Calibrated Ranges: Input:	9-15 PSIG (4)	Document:	47B601-3
Output:	0-100% OPEN	Document:	MFRR
Stroke:	N/A	Document:	N/A
Accuracy:	+/- 6%	Document:	MIG

Associated Interlock: 1-LSV-3-164; ENG to close and DEENG to modulate 1-LCV-3-164.

Associated Drawings: WBN-VTM-M120-0010

Remarks: (1) Pressure Regulator: 1-PREG-3-164 Setting: 65.0 PSIG
As-Found 63 to 65.5 PSIG / As-Left 64.5 to 65.5 PSIG
(2) DCN F-29644-A revised action of valve positioner from direct to reverse. This was done by rolling the output wires of 1-LM-3-164C. The input wires at the valve positioner were not changed, leaving black to (+) terminal and white to (-) terminal.
(3) DCN W-32189-B changed positioner (1-POS-3-164) from current in/pressure out to pressure in/pressure out.
(4) Input is to positioner (1-POS-3-164) which is reverse acting. Valve fails open with loss of supply air.
(5) DCN W-37869-A changed regulator setting.
(6) DCN 51815-A changed 1-PREG-3-164 setpoint.

SCALING AND SETPOINT DOCUMENT
PROCESS CONTROL VALVES AND CONTROL DEVICES

Instrument No.: 1-LCV-3-164A   Mfrr: Leslie   Model:Mark P
Vendor ID No: None   Contract No: 87379

Function: CV
Valve Action: Air to Open
Positioner Mfrr: Bailey   Model No: AV1121000   Cam Type: Linear

Calibrated Ranges:	Input: 3-9 PSIG (5)	Document:	47B601-3
	Output: 0-100% OPEN	Document:	MFRR
	Stroke: N/A	Document:	N/A
	Accuracy: +/- 6%	Document:	MIG

Associated Interlock: 1-LSV-3-164A; ENG to close and DEENG to modulate 1-LCV-3-164A

Associated Drawings: WBN-VTM-B045-0470, Manual ID WBN-L170-0010

Remarks: (1) Pressure Regulator: 1-PREG-3-164A1 Setting: 59 +0/-4 PSIG
(2) DCN S-21327-A revised 1-PREG-3-164A1 setting.
(3) DCN W-28198-A revised 1-PREG-3-164A1 setting.
(4) DCN W-32189-B changed positioner (1-POS-3-164A) range and changed action of 1-LS-3-164A.
(5) Input is to positioner (1-POS-3-164A).
Valve fails closed with loss of supply air.
(6) EDC 51696-A revised 1-PREG-3-164A1 setpoint.
(7) DCN 52127-A replaced positioner for 1-LCV-3-164A.

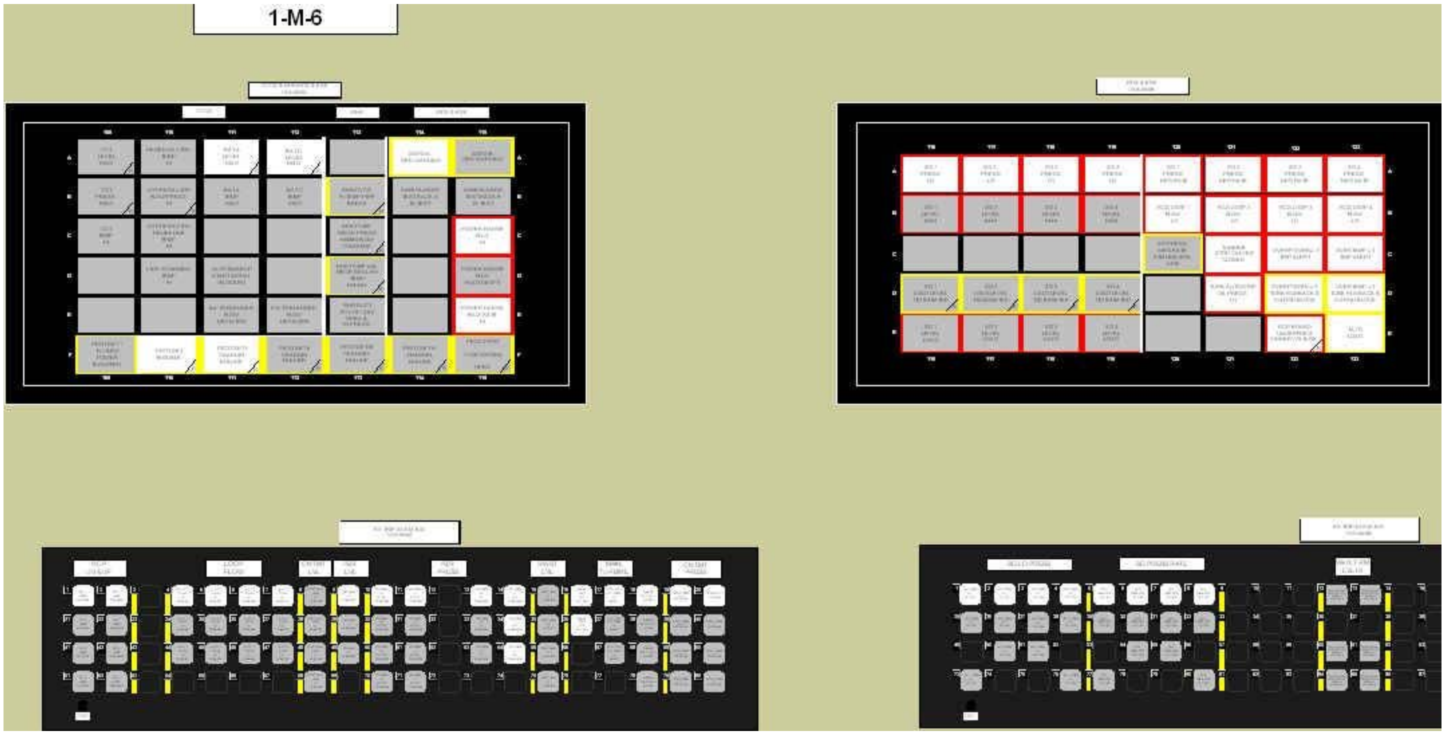
Function: Steam Generator 1 Motor Driven AFW Level Control
Location: EL 737, COL A3-S

Reviewed by: R. S. Henderson   Approved by: K. D. Garrison   Date: 10/13/2009

46.

Given the following conditions:

- An electrical board has failed
- The following indications are observed:



Which ONE of the following describes the electrical board that failed?

120 VAC Vital Instrument Board _______ has FAILED.

A. 1-I

B. 1-II

C. 1-III

D. 1-IV

CORRECT ANSWER: A

DISTRACTOR ANALYSIS:

A. Correct: The indications depicted occur upon the loss of 120 VAC Vital Instrument Board 1-I.

B. Incorrect: The annunciator pattern which would be received is similar. Channel II bistables (i.e. a different row of bistables) would be lit versus Channel I bistables. Also, both of the CERPI monitors would remain powered.

C. Incorrect: Again, the annunciator pattern would be different. Channel III bistables would be lit versus Channel I bistables. Both of the CERPI monitors would remain powered.

D. Incorrect: Again, the annunciator pattern would be different. Channel IV bistables would be lit versus Channel I bistables. The right hand CERPI monitors would be deenergized and thus dark.

Question Number: 46

Tier: 2 Group: 1

K/A: 062 A.C. Electrical Distribution
2.4.46 Ability to verify that the alarms are consistent with the plant conditions.

Importance Rating: 4.1 4.4

10 CFR Part 55: (CFR: 41.10 / 43.5 / 45.3 / 45.12)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to verify that the alarms and indications are consistent with the loss of 120 VAC Vital Instrument Board 1-I.

Technical Reference: 1-AOI-25.01, "Loss of 120V AC Vital Instrument Power Boards 1-I or 2-I"

Proposed references to be provided: None

Learning Objective:

Cognitive Level:
- Higher X
- Lower

Question Source:
- New X
- Modified Bank
- Bank

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Watts Bar Nuclear Plant

Unit 1

Abnormal Operating Instruction

1-AOI-25.01

Loss of 120V AC Vital Instrument Power Boards 1-I or 2-I

Revision 0003

Quality Related

Level of Use: Continuous Use

Effective Date: 04-08-2015

Responsible Organization: OPS, Operations

Prepared By: Scott Warrington

Approved By: William Sprinkle

1.0 PURPOSE

This instruction provides actions to respond to a loss of 120V AC Vital Instrument Power Boards 1-I and 2-I. This instruction will stabilize the unit and provide a list of equipment which may NOT be available due to a loss of the board. [C.1, C.3]

2.0 SYMPTOMS

The alarm pattern seen for 120VAC Vital Instrument Board 1-I is seen in the stem of the question.

2.1 Alarms

A. 125 DC VITAL CHGR/BATT 1 ABNORMAL [17-A]

B. 125 DC VITAL BATT BD 1 ABNORMAL CKTS ISOLATED [17-B]

C. 120V AC VITAL PWR BD 1-I UV/CKT TRIP [17-D].

D. 120V AC VITAL INVERTER 1-I / 0-I ABNORMAL [17-C].

E. NORTH/SOUTH VALVE VAULT RM LEVEL HI/MFW ISOLATION [57-C]

F. SG FEEDWATER FLOW HI [58-B]

G. SG 1 LEVEL HI [60-B]

H. SG 1 STM-FW FLOW MISMATCH [60-C]

I. SG 2 LEVEL HI [61-B]

J. SG 2 STM-FW FLOW MISMATCH [61-C]

K. SG 3 STM-FW FLOW MISMATCH [62-C]

L. SG 4 LEVEL HI [63-B]

M. SG 4 STM-FW FLOW MISMATCH [63-C]

N. C-5 LO TURB IMPULSE PRESS ROD BLOCK [66-A]

O. RCS COPS ARM PERMISSIVE [67-B]

P. DCS TROUBLE [82-F]

Q. POWER RANGE OVERPOWER ROD WD STOP [83-A]

R. POWER RANGE UPR DETECTOR FLUX DEVN [83-B]

S. POWER RANGE LWR DETECTOR FLUX DEVN [83-C]

2.1 Alarms (continued)

T. POWER RANGE CHANNEL DEVIATION [83-E]

U. RVLIS SYS MALFUNCTION [85-F]

V. CERPI TROUBLE [86-C]

W. RWST LEVEL LO-LO [126-D]

X. CL ACCUM 1 LEVEL HI/LO [131-A]

Y. CL ACCUM 1 PRESS HI/LO [131-B]

Z. CL ACCUM 2 LEVEL HI/LO [132-A]

AA. CL ACCUM 2 PRESS HI/LO [132-B]

BB. AC VITAL PWR BD 2-I UV/CKT TRIP [135-F].

2.2 Indications

A. Power range 1-N-41, Intermediate range 1-N-135, and Source Range 1-N-131 failure.

B. Rx Trip SI Status panel 1-XX-55-5 will be DARK.

C. Train A CNTMT Isolation Panel 1-XX-55-6E will be DARK.

D. Loss of Channel I S/G feedwater flow and steam flow inputs to DCS.

E. Loss of letdown due to Non-Essential air isolation to CNTMT.

F. 1-LT-68-339, PZR Level indication fails low.

G. 1-PT-68-340, PZR pressure indication fails low.

And because the input relay is deenergized (for SSPS), a trip bistable will be generated for the loss of F and G (and many other trip or ESFAS functions).

2.3 Automatic Actions

A. Rod withdrawal block due to high flux and C-5 interlock.

B. Charging suction swap to RWST due to de-energized separatio 1-LCV-62-132, VCT OUTLET VALVE, to close and 1-LCV-62-1 and 1-LCV-62-136, RWST CVCS SUPPLY HDR ISOLATION, to open injecting borated water to the RCS.

C. If in service, Train B MCR, EBR, and SDBR Chillers will swap to Train A, and auto start of Train B is disabled.

Appendix A
(Page 5 of 6)

120V AC Vital Instrument Power Board 1-I Loads

SYSTEM 74 - **RHR:**

- 1-FCV-74-1 will require placing 1-XS-74-1 to AUX position [Rx MOV Bd 1A1-A, Compt 5B] to open valve when RCS pressure is less than 380 psig.
- 1-FCV-74-16 & -32 will require placing 1-XS-74-16 & -32 to AUX position [1-L-11-A, Aux Control Room] to control RHR cool down temperature.

SYSTEM 82

- Emergency start for 1A-A and 2A-A D/Gs on 1-M-1 does not work. D/Gs will Emergency start from 0-M-26.

SYSTEM 85 - CERPI

- 1-MON-85-5000/1, CERPI Monitor A.

SYSTEM 88 - Containment Isolation:

- Train A CRI and CVI will occur after restoration of the board.
- If the refueling logic switch, 1-HS-90-410-A [1-R-73], is in the REFUEL position, then an "A" Train ABI (partial ABI associated with high rad in refuel area) will also occur when the CVI is initiated.

SYSTEM 90 - Radiation Monitors:

- 1-RM-90-106, Lower Containment Air Monitor control power **NOT** operable.
- 0-RM-90-125, MCR Train A Air Intake Monitor control power **NOT** operable.
- 1-RM-90-130, Containment Train A Purge Exhaust Monitor control power **NOT** operable.

SYSTEM 92 - Excore Nuclear Instrumentation:

- All Channel I NIS Control and Instrument Power.

SYSTEM 98-Distributed Control System:

- Primary power supply to DCS cabinets 1-L-981, 983, and 1-R-193. (ref 1-SOI-98.01 for specific loads)

47.

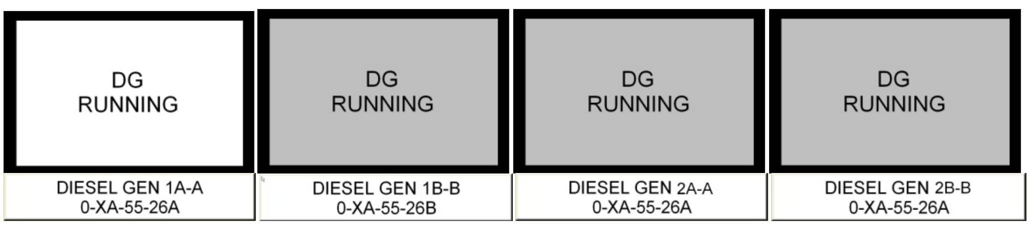
Given the following timeline:

00:00:00 Unit 1 is at 100% power.
00:00:01 The 1A-A SDBD experiences a BLACKOUT.

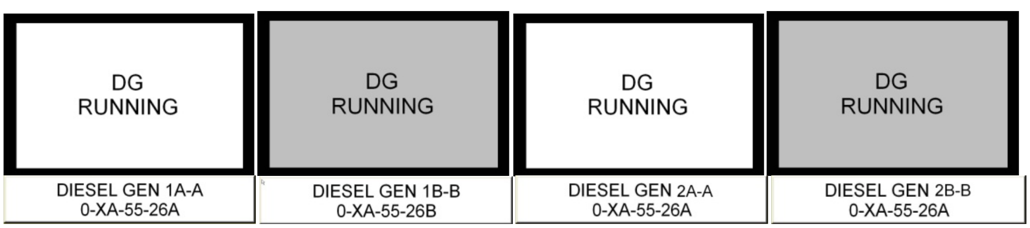
At 00:00:15, which ONE of the following combinations of alarms will be seen in the MCR?

A.

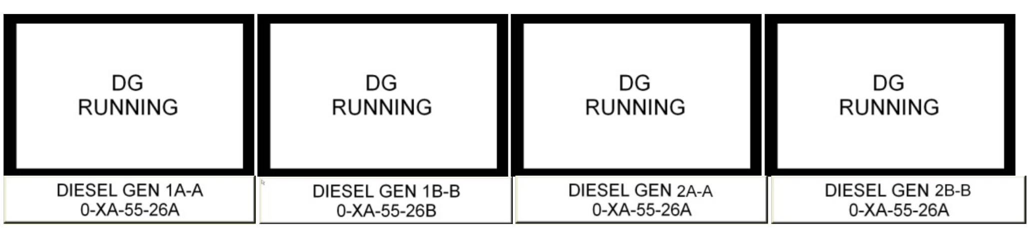
B.



C.



D.



CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: It is not correct that a BLACKOUT on one 6.9kV SDBD will cause none of the EDGs to start. An undervoltage condition on one 6.9kV Shutdown Board will cause all of the EDGs to start. In the case presented in the question, the 27LV relays which are monitoring the 1A-A 6.9kV SDBD (loss of voltage relays) will actuate at <87% of nominal shutdown board voltage. Therefore, as the board voltage decays to 0, the 27LV relays will actuate to pick up two timer relays LV-1 and LV-2. Only 1 of these two timers is required to cause (after 0.75 seconds) the LV relay to energize. When the LV relay energizes, the Normal (from CSST C), the alternate (from CSST D) and the maintenance feeder breaker (from a Unit board-this breaker is normally racked down) will trip open. At this point, the shutdown board is truly blacked out (0 volts). One half second after voltage drops below 70% of nominal (remember that the board is blacked out at this time), the 27D1A relays will act to energize the 27D1AX and 27D1AY relays. These two relays will open contacts in the ES1AY relay circuit which de-energizes ES1AY and ES1AY1. Note that ES stands for emergency start. With the diesels aligned in standby, ES1AY will act to start the diesel generator for the specific shutdown board and ES1AY1 will operate to start the other three diesels. Print 1-45W760-211-1 shows an excellent summary of the high level relay actions affecting the shutdown board. It demonstrates the 27LV, 27S and 27D relays' operation. Print 1-45W760-82-6 demonstrates in detail the emergency start of both the affected shutdown board's EDG and the common emergency start of the three other EDGs. In accordance with multiple documents in the combined license basis, the EDGs are designed to start and reenergize the SDBDs within 10 seconds.

It is plausible to believe this because 14 seconds has elapsed between the blackout of the SDBD and the time at which the indications would be seen. If one believed that greater than 14 seconds was required for an EDG to start and repower the SDBD, then one would arrive at this answer.

B. Incorrect: Again, it is incorrect that only the 1A-A EDG would be running. It is plausible to believe this because one may misunderstand the operation of the common emergency start.

C. Incorrect: It is incorrect that only the "A" train EDGs would be running. It is plausible to believe this simply because of the concept of affiliating the trains (e.g. believing that the “A” train EDGs would start on the blackout of an “A” train SDBD).

D. Correct: Correct, as discussed, a blackout of one 6.9kV SDBD will initiate the Common Emergency start of all of the EDGs.

Question Number: 47

Tier: 2 Group: 1

K/A: 062 A.C. Electrical Distribution
K1 Knowledge of the physical connections and/or cause effect relationships between the ac distribution system and the following systems:
K1.02 ED/G

Importance Rating: 4.1 4.4

10 CFR Part 55: (CFR: 41.10 / 43.5 / 45.3 / 45.12)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to understand the cause effect relationship between the blackout of the 1A-A 6.9kV SDBD and its associated ED/G.

Technical Reference: 1-45W760-211-1
1-45W760-82-6

Proposed references to be provided: None

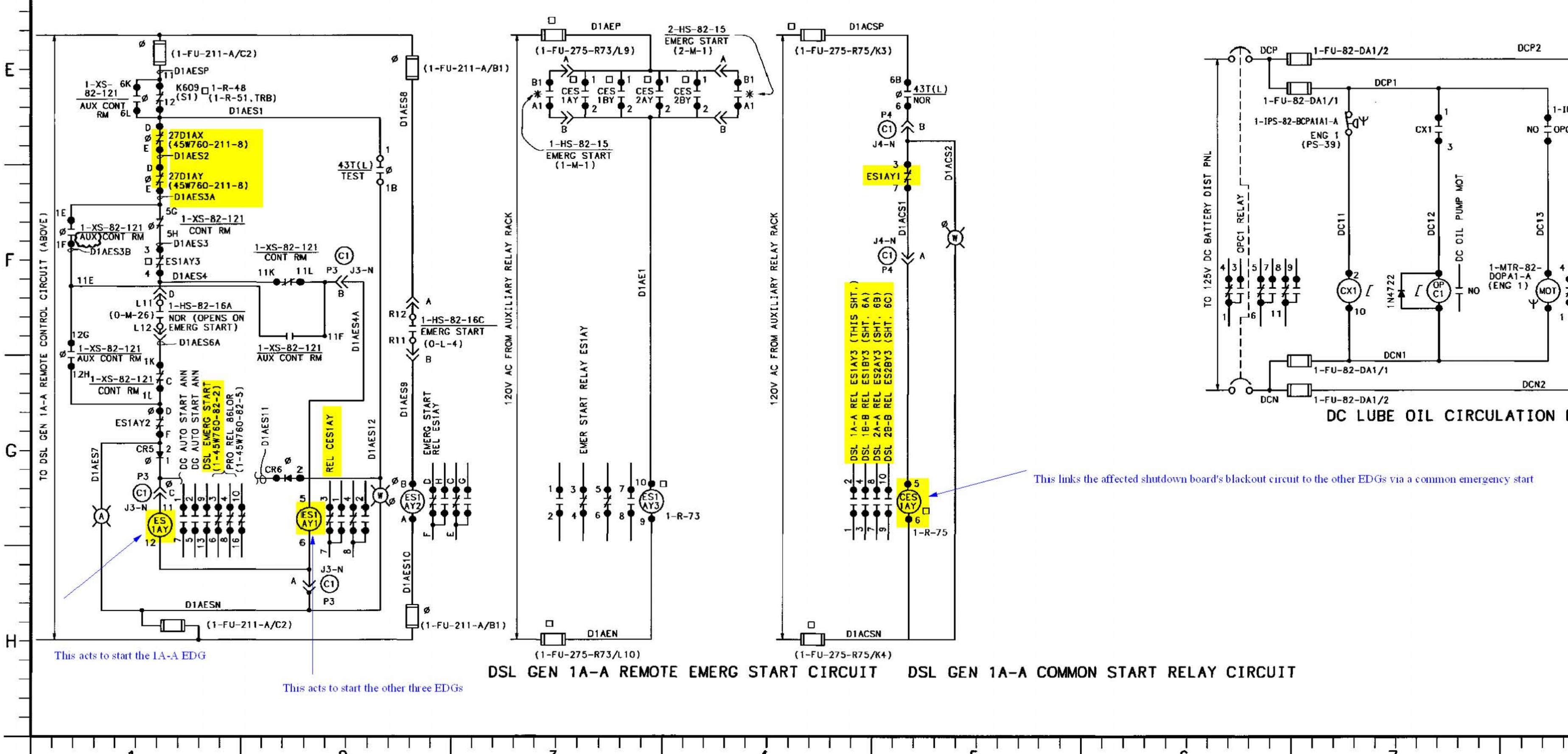
Learning Objective: 3-OT-SYS-201B, BLACKOUT AND LOADSHED LOGIC RELAYS
9. Given specific plant conditions, ANALYZE the effect that a loss of shutdown boards or malfunction of the Blackout and Load Shed Logic Relays will have on the following:
a. Diesel Generators

Cognitive Level:
Higher X
Lower

Question Source:
New X
Modified Bank
Bank

Question History: New question for the 2015-301 NRC RO Exam

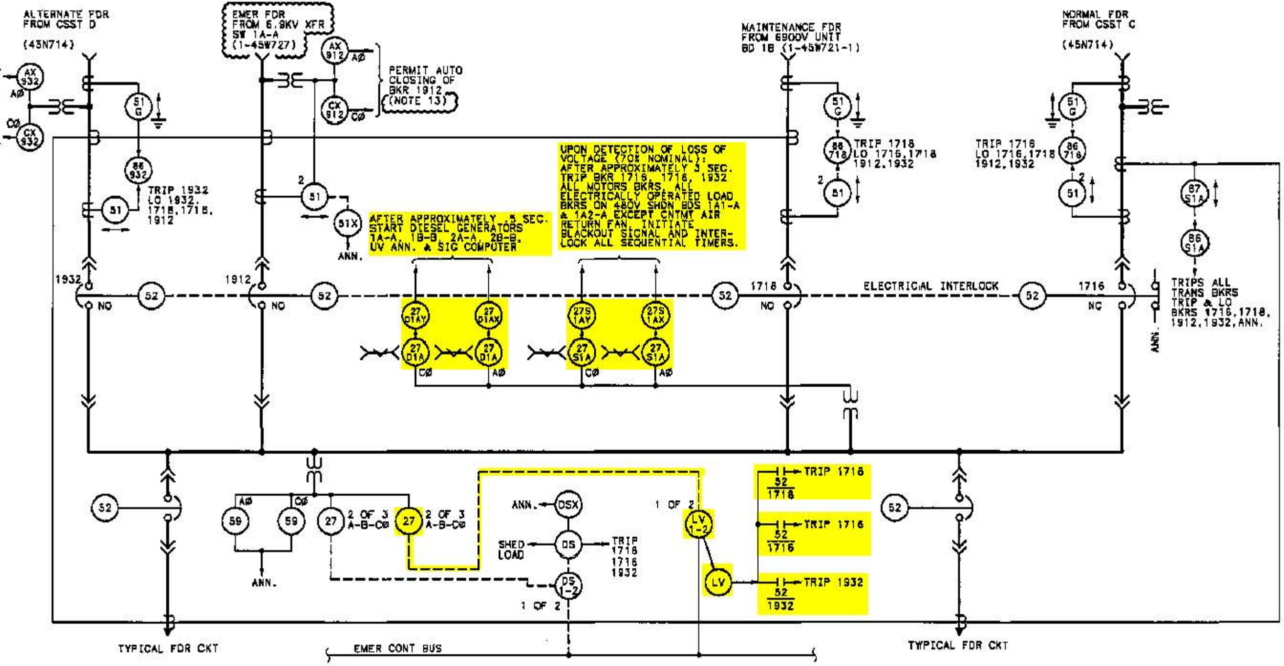
Comments:



DSL GEN 1A-A REMOTE EMERG START CIRCUIT

DSL GEN 1A-A COMMON START RELAY CIRCUIT

DC LUBE OIL CIRCULATION



*REQUIRED FOR 220 & 250V-DC OPERATION ONLY

KEY DIAGRAM
6900V SHUTDOWN BD 1A-A
(ELECTRICAL INTERLOCK SIMILAR FOR BOARDS 1B-B, 2A-A, 2B-B)

SHTDN BD	MAINT FDR BKR	NOR. FDR BKR	EMER FDR BKR	ALT FDR BKR

NOTES:

1. ALL EQUIPMENT IS LOCATED ON THE BOARD FROM WHICH ITS ASSOCIATED LOAD IS POWERED UNLESS OTHERWISE DESIGNATED.
2. BLACKOUT (BOX & BOY) RELAYS ARE SHOWN IN THE RESET STATE, WHICH IS THE BLACKOUT STATE. UNDERVOLTAGE (UVX AND UVY) RELAYS ARE SHOWN IN THE DE-ENERGIZED STATE, WHICH IS THE UNDERVOLTAGE STATE. FOR BLACKOUT AND UNDERVOLTAGE RELAY OPERATION SEE 1-45W760-211-16.
3. FUSE NUMBERS SHOWN IN TABLES SHOULD BE COMBINED WITH THE APPLICABLE BOARD PREFIX LISTED BELOW TO FORMULATE COMPLETE UNIQUE FUSE IDENTIFICATION NUMBER:
   6.9KV SHTDN BDS: 1-FU-211-
   EXAMPLE:
   1-FU-211-A15/1N
   THIS NUMBER WILL CHANGE TO 2 FOR UNIT 2.
4. AUTOMATIC OPEN-CIRCUIT FAST TRANSFER FROM NORMAL TO ALTERNATE COMMON STATION SERVICE TRANSFORMER INITIATED BY ANY TRANSFORMER OR LINE FAILURE RELAY SUCH AS FAULT PRESSURE, TRANSFORMER OVERCURRENT, GROUND CURRENT, LINE PROTECTION, OR DIFFERENTIAL RELAYS (VIA 86TT DWG 1-75W502). LOSS OF VOLTAGE & DEGRADED OFFSITE (NORMAL OR ALTERNATE) SOURCE OF ACTUATION OF CSST TRANSFORMER C OR D RESULT IN A TRANSFER TO THE STANDBY DIESEL GENERATOR THROUGH LOAD SHEDDING AND SEQUENCING. FOR DETAILED DESCRIPTION OF THE TRANSFER SCHEME. SEE DESIGN CRITERIA WB-DC-30-28.
5. THE FUSE INFORMATION FOR CLASS 1E FUSES OR FUSES USED TO PROTECT ELECTRICAL PENETRATIONS SHOWN IN THIS DRAWING SERIES, OR AS GIVEN FOR THESE CIRCUITS IN VENDOR CONTRACT DRAWINGS REFERENCED IN THIS DRAWING SERIES, IS FOR INFORMATION ONLY. REFER TO THE MASTER FUSE REPORT DRAWING SERIES 45B6000, FOR CONFIGURATION CONTROLLED FUSE INFORMATION. FOR THOSE FUSES NOT LISTED IN THE MASTER FUSE REPORT, THESE DRAWINGS WILL REMAIN THE CONTROLLING DOCUMENT.
6.
7. THIS CIRCUIT PERMITS ONLY MANUAL TRANSFER BETWEEN NORMAL, ALTERNATE AND MAINTENANCE BREAKERS, AND FROM THE DIESEL GENERATORS TO THE NORMAL, ALTERNATE AND MAINTENANCE POWER SOURCES.
8. DELETED
9. UNDERVOLTAGE RELAYS (27DAT, DBT, DCT) INITIATE A TWO-OUT-OF-THREE COINCIDENCE LOGIC TO INITIATE AN ALARM IN THE CONTROL ROOM AND LOAD SHED THE 6.9KV SHUTDOWN BOARD AND TRANSFER TO THE DIESEL GENERATOR AFTER A MAXIMUM DELAY OF 11.5 SECONDS
10. THE TERMINAL NUMBERS SHOWN FOR MCC-INTERNAL COMPONENTS ARE THE MCC-TERMINAL BLOCK/INTERNAL WIRE NUMBERS RATHER THAN THE ACTUAL DEVICE TERMINAL NUMBERS.
11. MAINTENANCE FEEDER BREAKERS 1718, 1818, 1726 & 1826 WILL BE ADMINISTRATIVELY LOCKED "NORMALLY OPEN" DURING NORMAL OPERATION AND WILL BE USED ONLY DURING COLD SHUTDOWN FOR MAINTENANCE PURPOSE , TRANSFER FROM THIS SOURCE TO ANOTHER SOURCE WILL BE MANUAL.
12. FOR GENERAL NOTES, INCLUDING TECHNICAL INFORMATION ABOUT CIRCUIT DESIGN, SEE DRAWING 1-45W760-0-1.
13. USE OF THE 6900V 3MW FLEX DIESEL GENERATORS BYPASS THIS PERMISSIVE. SEE NOTES 16 & 17, DRAWING 1-15E500-2.

48.

Given the following conditions:

- Spare charger 6-S has been placed in service to Vital Battery Board I, in accordance with 0-SOI-236.01, Section 8.1.

Subsequently:

- The operating crew is directed to place Vital Battery V in service to Vital Battery Board II.

Which ONE of the following describes the Battery V voltage and which charger is in service?

PRIOR to Vital Battery V being placed in service, Vital Battery V voltage will be ____(1)____ than that of Vital Battery II.

In accordance 0-SOI-236.05, **AFTER** Vital Battery V is placed in service on Battery Bd II, 125V VITAL BATTERY CHARGER ____(2)____ will be in service to Vital Battery Board II.

NOTE: *0-SOI-236.01, 125V DC Vital Battery Board I*
0-SOI-236.01, Section 8.1, 125V Battery Bd I Transfer to Spare 125V DC Vital Battery Charger 6-S
0-SOI-236.05, 125V DC Vital Battery Board V

	(1)	(2)
A.	higher than	II
B.	higher than	V
C.	the same as	II
D.	the same as	V

CORRECT ANSWER: A

DISTRACTOR ANALYSIS:

A. Correct: As seen in 0-SOI-236.05, 125V DC Vital Battery Board V, Vital Battery V has two more cells than the other Vital Batteries and as such operates at a higher voltage. Therefore, it is correct that prior to Vital Battery V being placed in service, Vital Battery V voltage will be higher than that of Vital Battery II. As seen in 0-SOI-236.05, 125V DC Vital Battery Board V, an assumption is made (prior to placing Vital Battery V in service to a Vital Battery Board) that either the normal or spare charger is in service. If no charger is in service, the user of the procedure is directed to use the 0-SOI-236 series to place either the normal or spare charger in service. The reason for this is that (as noted in the procedure): "At no time is Vital Battery Charger V to be used to supply battery system loads." Vital Battery Charger V is not a safety related charger and is not allowed to be connected to an operable DC source (i.e. Battery Board).

B. Incorrect: While it is correct that the normal operating voltage of VB V is higher than that of its counterparts, it is not correct that Vital Battery Charger V is allowed to be aligned to Vital Battery Board II. As seen in the simplified drawing in 0-SI-0-3, Weekly Log, it is fully physically possible to align the Vital Battery Charger V to another Vital Battery Board. However, also seen in the same procedure is the note which says that such alignment is not allowed.

C. Incorrect: While it is correct that Vital Battery Charger II must be used, it is not correct that VB V has the same voltage as the other 4 vital batteries.

D. Incorrect: It is incorrect that both the VB V has the same nominal voltage as the other four batteries and that Vital Battery Charger V may be aligned to VBB II.

Question Number: 48

Tier: 2 Group: 1

K/A: 063 D.C. Electrical Distribution
K1 Knowledge of the physical connections and/or cause effect relationships between the DC electrical system and the following systems:
K1.03 Battery charger and battery

Importance Rating: 2.9 3.5

10 CFR Part 55: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

10CFR55.43.b: Not applicable

K/A Match: The K/A is matched because the applicant must understand cause effect relationships of the Vital Battery V and Vital Battery Board II in that one must understand that the operating voltage of VB V is higher than that of the normally in service VB II and that the charger normally affiliated with VB V is not allowed to be used with VBB II.

Technical Reference: 0-SOI-236.01, 125V DC Vital Battery Board I
0-SOI-236.05, 125V DC Vital Battery Board V
1-45W703-1

Proposed references to be provided: None

Learning Objective: 3-OT-SYS057P, Plant DC Systems
4. EXPLAIN the physical connections and/or cause-effect relationships between the DC Systems and the following systems:
c. Battery charger and battery
9. Given plant conditions, IDENTIFY the applicable DC Systems Precautions and Limitations related to the following: (IER 11-3, Operating the plant with a conservative bias)
a. SOI-236.01 125V DC Vital Battery Board I
b. SOI-239.01 250V Battery Board I
c. SOI-100.01 Communications System

Cognitive Level:
Higher X
Lower ____

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

3.0 PRECAUTIONS AND LIMITATIONS

A. All breakers on a board should be OFF or OPEN and all protective grounds must be removed before energizing board.

B. A circuit that blows a fuse after being replaced once should have the circuit checked before replacing the fuse a second time, unless the SRO and plant conditions dictate otherwise.

C. To prevent explosive mixtures of H_2 and O_2 from accumulating in battery rooms, a battery room exhaust fan should be in service at all times.

D. A permanent or portable eyewash station shall be available in the battery rooms.

E. Acid spills should be neutralized with a solution of baking soda and water and all traces of the spill should be wiped up.

F. There shall be no smoking, open flames, or arcs in the battery rooms.

G. Battery board ground alarm setpoint (90v) is set at 60% of scale. Positive or negative grounds should be maintained as low as possible.

H. Do **NOT** allow continued operation of battery chargers if output voltage exceeds 140 volts DC while supplying plant loads or the output current exceeds 250 amps. Continued operation in excess of these values can cause equipment failures; therefore, prompt action should be taken to remove the chargers from service.

Because of the two extra cells, VB V possesses a higher voltage. After VB V is placed into service to a VBB, the float voltage originating from the vital battery charger must be increased.

If 480V power is removed from the charger for more than 5 minutes, the charger will automatically start up in the equalize mode of operation when power is restored. The Float pushbutton will have to be pressed and released in order to return to the float operating mode.

When placing the charger in service, transitioning between float and equalize modes, or when making sudden large magnitude load changes, there may be alarm actuations due to momentary voltage fluctuations on the charger. If these alarms occur, the operator will need to PRESS and RELEASE the PRESS TO RESET ALARMS pushbutton and ensure local alarms clear before proceeding to subsequent steps.

K. Vital Battery V has two more cells than the other Vital Batteries and operates at a higher voltage. Only Vital Battery Charger V can be used to perform an equalizing charge on Vital Battery V due to the other chargers having a high voltage cutout function that is lower than this higher voltage.

L. At no time is Vital Battery Charger V to be used to supply battery system loads.

Vital Battery Charger V is not safety related.

Date________ Initials

8.0 INFREQUENT OPERATIONS

8.1 Placing Vital Battery V in service on Battery Bd I, II, III or IV Using Normal or Spare Charger

NOTES

1) This section assumes either the normal or spare charger is in service. If no charger is in service, one must be placed in service per the applicable 0-SOI-236 series instruction.

2) At no time is Vital Battery Charger V to be used to supply battery system loads.

[1] **ENSURE** 125V dc Vit Batt Bd V in service per Section 5.0. _________

[2] **ENSURE** one exhaust fan is operating. _________

[3] **IDENTIFY** below which 125V DC Vital Batt Bd to be placed in service with 125V Vital Batt V. (**N/A** others)

Board to be Aligned to Vital Battery V	INITIALS
125V dc Vital Battery Board I	
125V dc Vital Battery Board II	
125V dc Vital Battery Board III	
125V dc Vital Battery Board IV	

NOTE

If selected board aligned to spare charger then next step may be N/Ad.

[4] **ENSURE** 125V Vital Batt Bd to be placed in service with 125V Vital Batt V is in service. (**N/A** others)

VITAL BATTERY BOARD	PROCEDURE	INITIALS
125V dc Vital Battery Board I	0-SOI-236.01	
125V dc Vital Battery Board II	0-SOI-236.02	
125V dc Vital Battery Board III	0-SOI-236.03	
125V dc Vital Battery Board IV	0-SOI-236.04	

[5] **PLACE** 0-BKR-236-5/CB2, DC OUTPUT, [Vit Batt Chgr V] in OFF. _________

____
CV

Date________ **Initials**

8.1 Placing Vital Battery V in service on Battery Bd I, II, III or IV Using Normal or Spare Charger (continued)

[6] **PLACE** 0-BKR-236-5/CB1, AC INPUT, [Vit Batt Chgr V] in OFF. _________

____
CV

[7] **PLACE** 0-BKR-236-5/102, SUPPLY FROM 125V VIT BATT CHGR V [5th Vital Batt Bd], OFF. _________

____
CV

[8] **NOTIFY** UO prior to switching on Vital Battery Boards. _________

[9] **IF** normal charger is to remain in service with Vital Battery V, **THEN**

GO TO Step 8.1[12]. _________

NOTE

If spare charger in service then next steps 8.1[10] and 8.1[11] may be **N/A**d.

[10] **PLACE** Spare Charger in service using appropriate procedure. (**N/A** others)

Vital Battery Board	Procedure	Initials
125V dc Vital Battery Board I	0-SOI-236.01	
125V dc Vital Battery Board II	0-SOI-236.02	
125V dc Vital Battery Board III	0-SOI-236.03	
125V dc Vital Battery Board IV	0-SOI-236.04	

Date________ **Initials**

8.1 Placing Vital Battery V in service on Battery Bd I, II, III or IV Using Normal or Spare Charger (continued)

NOTE

Vital Batt V is 62 cells with 137 to 140 volts of float voltage

[11] **PERFORM** the following:

[11.1] **ENSURE** 6-S, 7-S, 8-S, or 9-S charger (as appropriate) set to FLOAT. _________

[11.2] **RECORD** float voltage. _____________VDC. _________

[11.3] **IF** recorded voltage is **NOT** 137 to 140 VDC, **THEN**

NOTIFY Electrical Maintenance to adjust voltage to within 137 to 140 VDC. _________

[12] **PLACE** feeder breaker to 125V DC DIST PNL 5A or 5B in ON position to set up feed to desired Vital Batt Bd (**N/A** other) [125V Vit Batt Bd V, pnl 2]:

Desired Battery Bd Feed	Breaker to CLOSE or place in ON	PERF INITIALS	VERIFIER INITIALS
Feed to Battery Bds I and III	0-BKR-236-5/201		CV
Feed to Battery Bds II and IV	0-BKR-236-5/202		CV

Date________ **Initials**

8.1 Placing Vital Battery V in service on Battery Bd I, II, III or IV Using Normal or Spare Charger (continued)

CAUTION

Breakers in 125V DC DIST PNL 5A or 5B (el 757, outside of Battery Bd rooms II and III) are in OFF position when placed in the UP position.

NOTE

Closing 0-BKR-236-5/301 or 0-BKR-236-5/302 will cause an alarm on annunciator point 21-B. Closing 0-BKR-236-5/401 or 0-BKR-236-5/402 will cause an alarm on annunciator point 21-C.

[13] **PLACE** one of the following breakers in 125V DC DIST PNL 5A or 5B [el 757, cabinets outside of Batt Bd rooms II and III] in ON to align feed to desired Vital Batt Bd. (**N/A** three others).

Desired Battery Bd Feed	Breaker to CLOSE or place in ON	PERF INITIALS	VERIFIER INITIALS
Feed to Battery Bds I	0-BKR-236-5/301		CV
Feed to Battery Bds II	0-BKR-236-5/401		CV
Feed to Battery Bds III	0-BKR-236-5/302		CV
Feed to Battery Bds IV	0-BKR-236-5/402		CV

Date________ **Initials**

8.1 Placing Vital Battery V in service on Battery Bd I, II, III or IV Using Normal or Spare Charger (continued)

CAUTION

Board will be isolated from battery after performance of next step. Do **NOT** transfer shutdown boards or start large pumps until battery is on board.

NOTES

1) Opening breakers in the following step may cause an alarm on annunciator points 17B, 18B, 19B, or 20B.

2) In Modes 1, 2, 3, and 4, TS 3.8.4 should be applicable. In Modes 5 and 6, TS 3.8.5 may be applicable.

[14] **PERFORM** the following:

[14.1] **ENSURE** in-service charger (as appropriate) set to FLOAT. _________

[14.2] **RECORD** float voltage. ______________VDC. _________

[14.3] **IF** recorded voltage is **NOT** 137 to 140 VDC, **THEN**

NOTIFY Electrical Maintenance to adjust voltage to within 137 to 140 VDC. _________

[15] **OPEN** one of the following to disconnect normal vital batt from the vital batt bd [Vital Batt Bd - Pnl 1]: (**N/A** three others)

Assoc. Alarm	BATTERY BOARD	UNID	PERF INITIALS	VERIFIER INITIALS
17B	125v dc Vit Batt Bd I	0-BKR-236-1/109		CV
18B	125v dc Vit Batt Bd II	0-BKR-236-2/109		CV
19B	125v dc Vit Batt Bd III	0-BKR-236-3/109		CV
20B	125v dc Vit Batt Bd IV	0-BKR-236-4/109		CV

Date________ **Initials**

8.1 Placing Vital Battery V in service on Battery Bd I, II, III or IV Using Normal or Spare Charger (continued)

[16] **PLACE** one of the following switches to appropriate position: (**N/A** others) [Pnl 0s in each 125V Vit Batt Bd Rm]

- 0-XS-236-1/0, MCR AMMETER SELECTOR SWITCH, to BTRY V position. ________ ________ CV

- 0-XS-236-2/0, MCR AMMETER SELECTOR SWITCH, to BTRY V position. ________ ________ CV

- 0-XS-236-3/0, MCR AMMETER SELECTOR SWITCH, to BTRY V position. ________ ________ CV

- 0-XS-236-4/0, MCR AMMETER SELECTOR SWITCH, to BTRY V position. ________ ________ CV

[17] **ENSURE** selected breaker placed in ON position. (**N/A** others) [Pnl 0 in associated vital batt bd rms]

NOMENCLATURE	LOCATION	UNID	PERF INITIALS	VERIFIER INITIALS
BACKUP SUPPLY FROM 125V VITAL BATTERY BD V	0-DPL-236-1 vital batt rm 1	0-BKR-236-1/0		CV
BACKUP SUPPLY FROM 125V VITAL BATTERY BD V	0-DPL-236-2 vital batt rm 2	0-BKR-236-2/0		CV
BACKUP SUPPLY FROM 125V VITAL BATTERY BD V	0-DPL-236-3 vital batt rm 3	0-BKR-236-3/0		CV
BACKUP SUPPLY FROM 125V VITAL BATTERY BD V	0-DPL-236-4 vital batt rm 4	0-BKR-236-4/0		CV

Date________ **Initials**

8.1 Placing Vital Battery V in service on Battery Bd I, II, III or IV Using Normal or Spare Charger (continued)

NOTE

In Modes 1, 2, 3 and 4, TS 3.8.4 may be exited. In Modes 5 and 6, TS 3.8.5 may be exited.

[18] **CHECK** current flow toward Vital Battery V (negative or zero) to ensure normal or spare Vital Charger is carrying load and Vital Battery V tied to board (selected Battery Board ammeter). ________

NOTES

1) The Battery Board is now connected to Vital Batt V. Transfers of Shutdown Boards or starting of large pumps may be resumed as necessary.

2) The normal Vital Battery is now disconnected without a charger. If a spare battery charger is in service to the 5th Vital Battery, then the normal charger should be placed on the normal vital battery unless a condition exists which would prohibit the battery from remaining on a charger. Sections 8.5 through 8.8 should be used for placing the normal battery charger in service to its disconnected vital battery.

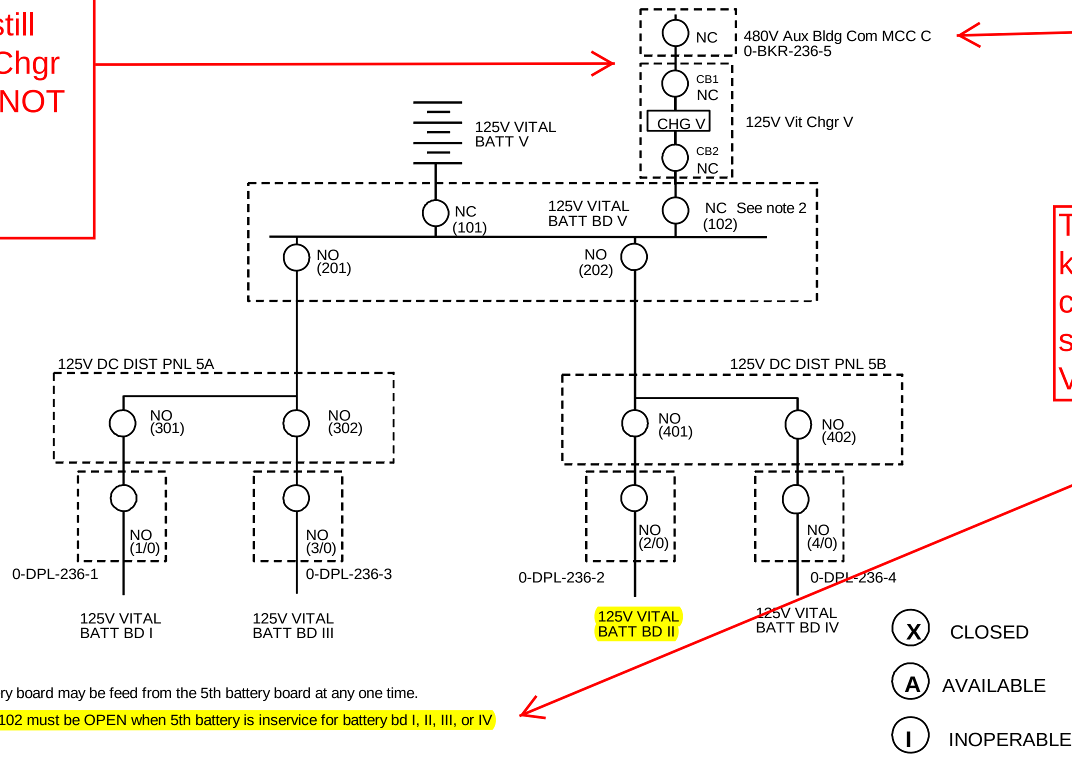
End of Section

Appendix A
(Page 10 of 24)

It is fully possible to align Vital Batt BD V to another Vital Batt Bd and still have the Vit Chgr V in service (NOT allowed but physically possible).

Notice that the power supply to the Vit Chgr V is not safety related

This prohibits keeping vital charger V in service to another VBB.



NOTE 1: Only one battery board may be feed from the 5th battery board at any one time.

NOTE 2: 0-BKR-236-5/102 must be OPEN when 5th battery is inservice for battery bd I, II, III, or IV

INITIALS OF DATA COLLECTOR: ____________________ DATE ____________________

REMARKS: ____

49.

Given the following conditions:

- Unit 1 is at 100%.

HANDSWITCH ON 0-M-26

Which ONE of the following describes the Emergency Diesel Generators response to depressing the handswitch shown above and the Essential Raw Cooing Water (ERCW) alignment?

Depressing the switch will start ____(1)____ EDG(s).

ERCW ____(2)____.

A. (1) **ALL**
   (2) AUTOMATICALLY aligns to the starting DG

B. (1) **ALL**
   (2) is NORMALLY flowing to the standby DGs

C. (1) **ONLY** ONE
   (2) AUTOMATICALLY aligns to the starting DG

D. (1) **ONLY** ONE
   (2) is NORMALLY flowing to the standby DGs

CORRECT ANSWER: C

DISTRACTOR ANALYSIS:

A. Incorrect: While it is correct that ERCW would be automatically aligned to a DG, it is not correct but plausible that the handswitch depicted would start all of the DGs.

B. Incorrect: It is not correct that all of the DGs will start when 1-HS-82-16A is depressed. It is plausible to believe that because there are MCR hand-switches (1-HS-82-15 and 2-HS-82-15) which will cause all four of the DGs to emergency start. This is seen on print 1-45W760-82-6. It is also plausible to believe this because on this same print, 1-HS-82-16A appears in-line with the other causes of a common emergency start of the DGs; however, it appears downstream of the branch to ES1AY1 and as such will not cause the common start. Also, it is Incorrect and yet plausible that ERCW would be normally flowing to the standby DGs.

C. Correct: It is correct that ERCW automatically aligns to the DGs and that only one DG will start when 1-HS-82-16A is depressed.

D. Incorrect: As seen on print 1-45W760-82-6, it is correct that depressing 1-HS-82-16A will only emergency start one diesel generator (specifically, the 1A-A). Because 1-HS-82-16A opens the current flow path to relay ES1AY after the branch to relay ES1AY1, the common emergency start of the diesel generators is not actuated. It is not correct that ERCW is normally flowing to the standby DGs. As seen in 0-SOI-82.01, Diesel Generator (DG) 1A-A precaution and limitation HH: ERCW automatically aligns to a DG at 40 rpm. It is plausible to believe that ERCW would be normally flowing to the standby DGs as this was the case before Design Change 56341 was implemented in November of 2013. This design change considered the overall two unit flow balancing requirements for ERCW and required that the ERCW supplies to the DGs be isolated when the DGs were in standby. Currently, ERCW to the DGs is automatically aligned as the DG comes to speed and is manually secured as part of the standby alignment.

Question Number: 49

Tier: 2 Group: 1

K/A: 064 Emergency Diesel Generators (ED/G)
A4 Ability to manually operate and/or monitor in the control room:
A4.06 Manual start, loading, and stopping of the ED/G

Importance Rating: 3.9 3.9

10 CFR Part 55: (CFR: 41.7 / 45.5 to 45.8)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to understand which hand-switches in the MCR will manually start all of the DGs and which will manually start only one DG. The applicant must also be able to monitor the proper operation of the DG cooling water (ERCW) during a manual diesel start.

Technical Reference: SOI-82.01, Diesel Generator (DG) 1A-A revision 0090 (HISTORIC)
0-SOI-82.01, Diesel Generator (DG) 1A-A
1-45W760-82-6

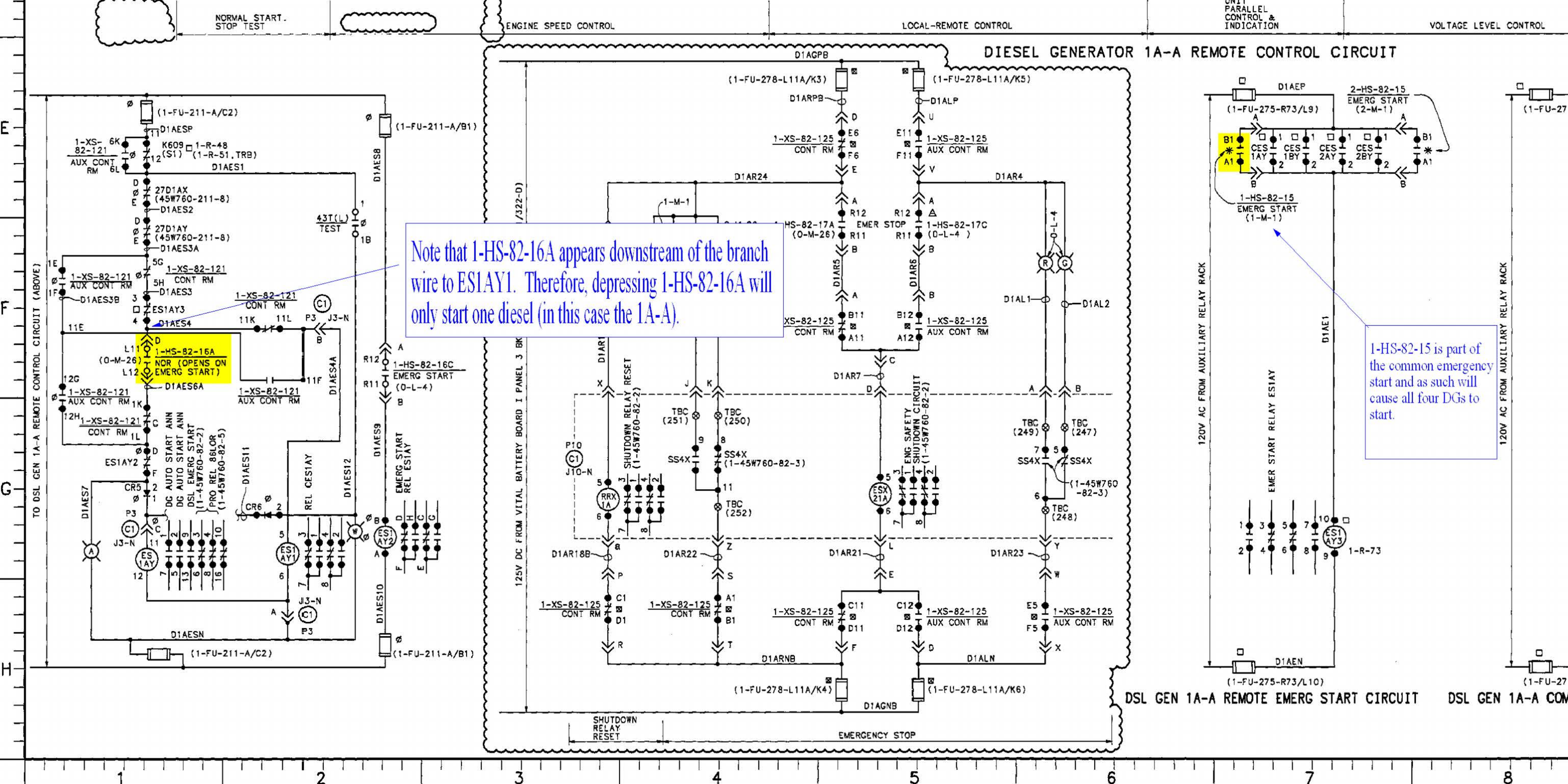
Proposed references to be provided: None

Learning Objective: 3-OT-SYS082A, DIESEL GENERATOR OVERVIEW
1. EXPLAIN the DG design features and/or operational interlocks that provide the following: (IER 11-3: Having a Solid Understanding of Plant Design and System Interrelationships)
a. Normal & Emergency starts (automatic and manual)
b. DG control during emergency start

Cognitive Level:
Higher X
Lower ____

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam
Comments:


NORMAL START. STOP TEST
ENGINE SPEED CONTROL
LOCAL-REMOTE CONTROL
UNIT PARALLEL CONTROL & INDICATION
VOLTAGE LEVEL CONTROL
DIESEL GENERATOR 1A-A REMOTE CONTROL CIRCUIT
Note that 1-HS-82-16A appears downstream of the branch wire to ES1AY1. Therefore, depressing 1-HS-82-16A will only start one diesel (in this case the 1A-A).
1-HS-82-16A NOR (OPENS ON EMERG START)
1-HS-82-15 is part of the common emergency start and as such will cause all four DGs to start.
1-HS-82-15 EMERG START (1-M-1)
2-HS-82-15 EMERG START (2-M-1)
1-HS-82-16C EMERG START (0-L-4)
TO DSL GEN 1A-A REMOTE CONTROL CIRCUIT (ABOVE)
125V DC FROM VITAL BATTERY BOARD I PANEL 3 BKR (322-D)
120V AC FROM AUXILIARY RELAY RACK
EMER START RELAY ES1AY
SHUTDOWN RELAY RESET
EMERGENCY STOP
DSL GEN 1A-A REMOTE EMERG START CIRCUIT
DSL GEN 1A-A COM

3.0 PRECAUTIONS AND LIMITATIONS (continued)

FF. The DG may surge from 400-450 RPM after an Idle Start, which is to be expected and does **NOT** have any adverse consequences.

GG. The fuel oil transfer pumps will automatically start when the gauge indicates between mark 6-9 (out of the 16 marks for tank gauge).

HH. ERCW automatically aligns to a DG at 40 rpm. If alignment of ERCW **CANNOT** immediately be obtained, the DG should be Emergency Stopped.

II. Alarm window 225-E TR A/B ERCW TO C&SS COMPR FLOW HI may alarm on some DG starts due to high differential between the two ERCW headers.

Revision Log

Revision or Change Number	Effective Date	Affected Page Numbers	Description of Revision/Change
81	10/04/12	2, 68	Incorporated changes associated with DCN #58383 by adding a new DG Emergency Stop/Reset Transfer switch 1-XS-82-125, which is located on 1-PNL-276-L011A in ACR.
82	02/05/13	2, 13, 42, 43, 95	Added steps to use strobotac for failure of single Air Intake Damper group and commitment statement including WBPER458545.[PER598595-001]
83	03/25/13	2, 12	Added P&L GG to clarify when the DG transfer pump will automatically start. [PER 620257-001]
84	06/07/13	2, 88-90	PER 699018: Add sections to shutdown and restore DG room exhaust fans as part of a tagout.
85	08/22/13	2, 91, 92	Added Section 8.15 for manual exhaust fan start in low temperature conditions. Requested by Work Control for PMT performance following clearance.
86	11/18/13	2, 12, 29, 42	Added P&L and step to alert operator that 1-FCV-67-66 will automatically open at 40 rpm and emergency stop required if this does not occur. [DCN56341]
87	11/27/13	2, 29	Added step to close 1-FCV-67-66 for standby alignment.
88	02/27/14	2, 13, 27-35	Minor/editorial revision to ad notes regarding the effects on ERCW header pressure from the new DG ERCW alignment. [PER 817463-001] [PER 826190-001] Moved IV steps to end of 5.2 Verification of Standby Alignment to help prevent human error traps.
89	04/06/14	2, 93	Revised note to clarify normal position of Exhaust Fan 1 and 2 iaw 1-SI-0-2 series. [PER 827633-001]
90		ALL	Cancelled to 0-SOI-82.01, Rev 0.

50.

Given the following timeline:

00:00:00 Unit 1 is at 100% power.
00:01:00 An error by the Protective Relay test group causes the 1A-A 6.9kV SDBD to experience a Blackout.
00:10:00 In accordance with 0-SOI-82.01, section 8.3.1, an operator is preparing to parallel the 1A-A EDG to offsite power.
1-XI-82-1, TRAIN 1A-A SYNCHROSCOPE is rotating **CLOCKWISE** as shown below.
The synchroscope has **NOT** completed a full revolution.

Synchroscope at 00:10:00 Synchroscope at 00:10:07

Which ONE of the following describes the direction and speed of the synchroscope rotation?

The direction of synchroscope rotation is ____(1)____ AND the speed of synchroscope rotation is ____(2)____.

NOTE: *0-SOI-82.01, Diesel Generator (DG) 1A-A,Section 8.3.1, Removing DG from Service-DG Tied on to SD Bd*

REFERENCE PROVIDED

A. (1) CORRECT
(2) CORRECT

B. (1) CORRECT
(2) TOO FAST

C. (1) INCORRECT
(2) CORRECT

D. (1) INCORRECT
(2) TOO FAST

CORRECT ANSWER: A

DISTRACTOR ANALYSIS:

A. Correct: As seen in step [15] of section 8.3.1 of 0-SOI-82.01, the operator is to **ADJUST** 1-HS-82-13, SPEED CONTROL, [0-M-26] to obtain desired clockwise rotation (15 or more seconds) on 1-XI-82-1, TRAIN 1A-A SYNCHROSCOPE. Therefore, the direction of synchroscope rotation is correct. The two pictures of the synchroscope depict two states of the instrument over a 7 second span. Because the needle has travelled about 120°, the total time for one rotation of the meter is 360°/120° x 7 seconds = 21 seconds. Even if one assumes that the needle has travelled 150°, the time for one rotation becomes 16.8 seconds. Because the rotation is longer than 15 seconds, it is acceptable. The actual angle of displacement of the needle is just at 135°.

B. Incorrect: Again the direction of synchroscope rotation is correct. However, the rotation of the meter is not too fast. It is plausible to believe that it is if one calculates the speed of rotation based upon a counterclockwise rotation of the synchroscope. If one did so one would determine that the speed for rotation is 10.5 seconds per rotation (based upon an interpretation that the synchroscope needle rotated 240° in 7 seconds) or 12.0 seconds (based upon an interpretation that the synchroscope needle rotated 210° in 7 seconds).

C. Incorrect: Again, it is incorrect that the direction of rotation is incorrect. However, as demonstrated the speed of rotation is correct.

D. Incorrect: It is both incorrect that the direction of rotation is incorrect and that the speed of rotation is too fast.

Question Number: 50

Tier: 2 Group: 1

K/A: 064 Emergency Diesel Generator (ED/G) System
A4 Ability to manually operate and/or monitor in the control room:
A4.09 Establishing power from the ring bus (to relieve ED/G)

Importance Rating: 3.2 3.3

10 CFR Part 55: (CFR: 41.7 / 45.5 to 45.8)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to monitor the proper conditions needed to bring the normal offsite power supply to the 6.9kV shutdown board back into service (i.e. relieving the ED/G) following a blackout condition. .

Technical Reference: 0-SOI-82.01, Diesel Generator (DG) 1A-A

Proposed references to be provided: Redacted page 61 of 0-SOI-82.01

Learning Objective: 3-OT-SYS082A, DIESEL GENERATOR OVERVIEW
8. Given plant conditions, IDENTIFY the applicable Diesel Generator Precautions and Limitations contained in SOI-82 series.

Cognitive Level:
Higher X
Lower ____

Question Source:
New ____
Modified Bank X
Bank ____

Question History: Modified bank Question (original Q was SYS082I.04). The original Q started with offsite power on the SDBD. This Q starts with the DG as the only power source on the SDBD.

Comments:

Date________ INITIALS

8.3.1 Removing DG from Service-DG Tied on to SD Bd (continued)

CAUTION

When adjusting speed and voltage, care must be taken to prevent overshooting desired values. Voltage control response is approximately five times faster than speed control response.

Slow in the fast direction is the desired indication on the synchroscope. Therefore, a clockwise rotation of greater than 15 seconds is desired.

[12] **MATCH** generator Incoming Frequency (1-XI-82-2) with Running Frequency (1-XI-82-3) using 1-HS-82-13, SPEED CONTROL [0-M-26]. _______

[13] **MATCH** running voltage (1-EI-82-5) to incoming voltage (1-EI-82-4) with 1-HS-82-12, VOLTAGE REGULATOR [0-M-26]. _______

[14] **PLACE** 1-HS-82-18, DG MODE SELECTOR, in PARALLEL [0-M-26]. _______

[15] **ENSURE** DG Frequency and Voltage are MATCHED with 6.9 kV SD Bd, **AND**

ADJUST 1-HS-82-13, SPEED CONTROL, [0-M-26] to obtain desired clockwise rotation (15 or more seconds) on 1-XI-82-1,TRAIN 1A-A SYNCHROSCOPE. _______

____
CV

[16] **IF** a white breaker disagreement light is lit for any of the following handswitches, **THEN**

CLEAR the disagreement by taking the handswitch to TRIP. (**N/A** handswitches with no disagreement)

NOMENCLATURE	LOCATION	POSITION	UNID	PERF INITIAL
1716-NORMAL FROM CSST C	0-M-26	TRIP	1-HS-57-41B	
1932 ALTERNATE FROM CSST D	0-M-26	TRIP	1-HS-57-97B	
1718-MAINTENANCE FROM 6.9 UNIT BD 1B	0-M-26	TRIP	1-HS-57-44B	

1. Given the following timeline:

   0000 - Unit 1 is at 100% power.

   0001 - In accordance with section 8.3.2 of SOI-82.01, an operator is preparing to parallel the 1A-A DG to the 1A-A 6.9kV SDBD. He notes that the 1A-A DG's synchroscope is rotating clockwise as shown below. He notes that the synchroscope has **NOT** completed a full revolution.

Synchroscope at 0001:00

Synchroscope at 0001:05

This is the original question.

Which ONE of the following completes the statements below:

The direction of synchroscope rotation is ___(1)___.

The speed of synchroscope rotation is ___(2)___.

Note: SOI-82.01, "Diesel Generator (DG) 1A-A"
Section 8.3.2, "Removing DG from Service-DG NOT Tied on to SD Bd"

REFERENCE PROVIDED

	(1)	(2)
a.	correct	acceptable
b.✔	correct	unacceptable
c.	incorrect	acceptable
d.	incorrect	unacceptable

Date________ **INITIALS**

8.3.1 Removing DG from Service-DG Tied on to SD Bd (continued)

> **CAUTION**
>
> When adjusting speed and voltage, care must be taken to prevent overshooting desired values. Voltage control response is approximately five times faster than speed control response.

[12] **MATCH** generator Incoming Frequency (1-XI-82-2) with Running Frequency (1-XI-82-3) using 1-HS-82-13, SPEED CONTROL [0-M-26]. ________

[13] **MATCH** running voltage (1-EI-82-5) to incoming voltage (1-EI-82-4) with 1-HS-82-12, VOLTAGE REGULATOR [0-M-26]. ________

[14] **PLACE** 1-HS-82-18, DG MODE SELECTOR, in PARALLEL [0-M-26]. ________

[15] **ENSURE** DG Frequency and Voltage are MATCHED with 6.9 kV SD Bd, **AND**

ADJUST 1-HS-82-13, SPEED CONTROL, [0-M-26] to obtain [redacted] (15 or more seconds) on 1-XI-82-1,TRAIN 1A-A SYNCHROSCOPE. ________

____
CV

[16] **IF** a white breaker disagreement light is lit for any of the following handswitches, **THEN**

CLEAR the disagreement by taking the handswitch to TRIP. (**N/A** handswitches with no disagreement)

NOMENCLATURE	LOCATION	POSITION	UNID	PERF INITIAL
1716-NORMAL FROM CSST C	0-M-26	TRIP	1-HS-57-41B	
1932 ALTERNATE FROM CSST D	0-M-26	TRIP	1-HS-57-97B	
1718-MAINTENANCE FROM 6.9 UNIT BD 1B	0-M-26	TRIP	1-HS-57-44B	

51.

Given the following conditions:

- CASK DECON COLL TANK(CDCT) LEVEL HI (0-L-2D-9) is LIT.

The following windows alarm and are acknowledged:

Which ONE of the following completes the statements listed below?

____(1)____ would cause the annunciators shown above.

In accordance with 0-SOI-77.01, Liquid Waste Disposal, the high level in the CDCT ____(2)____

A. (1) High activity in the liquid rad waste piping
   (2) is unable to be REDUCED until 0-RM-90-122 is restored

B. (1) A loss of power to the radiation monitor
   (2) is unable to be REDUCED until 0-RM-90-122 is restored

C. (1) High activity in the liquid rad waste piping
   (2) can be REDUCED after fuses are removed and a temporary jumper is installed

D. (1) A loss of power to the radiation monitor
   (2) can be REDUCED after fuses are removed and a temporary jumper is installed

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: As seen in ARI-180-187, Common Radiation Detectors, both the loss of power to the rate meter and the loss of signal from the detector will cause Annunciator 181-C, WDS RELEASE LINE 0-RM-122 INSTR MALF to alarm. Also contained in this alarm response instruction is that the causes for Annunciator 181-A, WDS RELEASE LINE 0-RM-122 LIQ RAD HI are as follows: A. High activity in release line B. Loss of power to control relay (WDRS) C. Loss of power to rate meter. D. Background radiation rise at monitor To further develop the causes of Annunciator 181-A, one may refer to the vendor manual for the readout modules manufactured by Sorrento Electronics, WBN-VTD-G292-1150. The radiation monitor instrument loop contains (in order of signal progression) a detector, preamplifier, and readout module. The preamplifier delivers a sequence of negative pulses to the readout module. These negative pulses are filtered by the readout module and then converted into a dc current which is proportional to the counting rate of input pulses. Finally, the dc current is fed to a meter amplifier circuit which provides a current to voltage transfer function. This voltage is fed into the summing junction of an operational amplifier which controls the trip circuit of the readout module. In the foregoing, one may observe that if the signal from the detector is lost, the sequence of negative pulses will not exist. Because the pulse rate is 0, the dc current which is generated will be 0. As a result, the voltage fed to the summing junction of the operational amplifier will be 0 and will never reach the high radiation trip setpoint. Therefore, a loss of signal from the detector cannot cause a High Radiation alarm. It is plausible to believe that a loss of signal would place the readout module's Bistable Trip Circuits into their tripped state because of the existence of the Operate (Instrument Malfunction) circuit. If the detector signal should fail, the Operate relay of the readout module, K101 will de-energize causing its contacts to change state. The applicant may therefore believe that the trip bistables are affected in the same mode as the operate relay by a loss of signal. It is also Incorrect that the high level in the CDCT would be unable to be corrected until the radiation monitor is restored. Section 8.16, Release of the Cask Decontamination Collector Tank to Cooling Tower Blowdown of 0-SOI-77.01 contains step [3] which takes action **IF** 0-RE-90-122 is INOPERABLE. These actions include: **REMOVE** fuses N-17 and N-18 in Panel 1-R-76. **PLACE** temporary jumper (clip-type) at TB-248 across terminals 8 and 10 in Panel 1-R-72. It is plausible to believe that the high level cannot be corrected until the radiation monitor is restored because there are procedures such as SOI-77.02, Waste Gas Disposal System which do not contain steps for the conditions listed in the question. During a gas tank release, 0-RM-90-118 monitors the effluent. If 0-RM-90-118 were to lose power, the release would be terminated (tantamount to the response of the liquid radioactive waste system). The steps in SOI-77.02 in part state: **NOTIFY** Instrument Maintenance to adjust setpoint of 0-RE-90-118 to avoid inadvertent isolation of release. This step would not allow the gas decay tank release to continue.

B. Incorrect: It is correct that a loss of power to the radiation monitor would produce the annunciators depicted. It is not correct and yet plausible that a procedural compliant liquid release would be impossible until the radiation monitor was restored.

C. Incorrect: As mentioned it is Incorrect and yet plausible that a loss of signal would produce a High Radiation alarm. It is correct that steps within 0-SOI-77.01 could be used to compensate for the loss of the radiation monitor.

D. Correct: As discussed, a that a loss of power to the radiation monitor would produce the annunciators depicted. It is correct that steps within 0-SOI-77.01 could be used to compensate for the loss of the radiation monitor.

Question Number: 51

Tier: 2 Group: 1

K/A: 073 Process Radiation Monitoring (PRM) System
A2 Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to Correct: control, or mitigate the consequences of those malfunctions or operations:
A2.01 Erratic or failed power supply

Importance Rating: 2.5 2.9

10 CFR Part 55: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to first predict that a loss of power to the 0-RM-90-122 monitor has on that monitor's output to the RONAN system. Subsequently, the applicant must be able to use 0-SOI-77.01 to mitigate the consequence imposed by the loss of power and correct the high level existent in a liquid radioactive waste tank.

Technical Reference: SOI-77.02, Waste Gas Disposal System
ARI-0-L-2D, Liquid Waste Panel
0-SOI-77.01, Liquid Waste Disposal
ARI-180-187, Common Radiation Detectors
WBN-VTD-G292-1150, Vendor Manual for the General Atomics Radiation Analyzer Readout Module

Proposed references to be provided: None

Learning Objective: 3-OT-SYS090A, Radiation Monitoring
3. Given plant conditions, DETERMINE if any of the following Radiation Monitoring System alarms would be present and the actions required by the ARI: (IER 11-3, Monitoring plant indications and controls closely)
b. [0-XA-55-12B, ALL] All annunciators on COMMON RADIATION DETECTORS panel from ARI-180-187

Cognitive Level:
- Higher X
- Lower ____

Question Source:
- New X
- Modified Bank ____
- Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

181-A

Source
0-RM-90-122A

Setpoint
determined by Chemistry

WDS RELEASE LINE
0-RM-122
LIQ RAD
HI

(Page 1 of 1)

Probable Cause:

A. High activity in release line
B. Loss of power to control relay (WDRS)
C. Loss of power to ratemeter
D. Background radiation rise at monitor

NOTE
0-RM-90-122 has associated ICS computer point R1022A.

Corrective Action:

[1] **ENSURE** 0-RCV-77-43, WASTE CONDENSATE DISCHARGE RADIATION ISOLATION VALVE closed.
[2] **CHECK** 0-RE-90-9 (ARM)
[3] **NOTIFY** Chemistry to perform CM-9.09 "Effluent Radiation Monitor Alarm Response Guidelines".
[4] **NOTIFY** Radiation Protection to investigate alarm.
[5] **IF** release was in progress, **THEN**
EVALUATE discharge lineup.
[6] **REFER TO** AOI-31, Abnormal Release Of Radioactive Material.
[7] **REFER TO** EPIP-1.

References:

1-45W600-57-27
1-45W600-77-1
1-47W610-90-2
AOI-31
EPIP-1
CM-9.09

181-C

Source	**Setpoint**
0-RM-90-122A	Instrument Failure

WDS RELEASE LINE
0-RM-122
INSTR MALF

(Page 1 of 1)

Probable Cause:

A. Loss of signal from detector
B. Failure of readout module's internal power supply
C. Loss of power to ratemeter
D. Loss of power to control relay
E. Function switch in ALARM ADJ

Corrective Action:

[1] **ENSURE** 0-RCV-77-43, WASTE CONDENSATE DISCHARGE RADIATION ISOLATION VALVE, CLOSED.
[2] **CHECK** 0-RM-90-122A and 0-RR-90-122 [0-M-12] OR ICS computer points R1022A.
[3] **NOTIFY** Chemistry to perform CM-9.09 "Effluent Radiation Monitor Alarm Guidelines".
[4] **NOTIFY** Instrument Maintenance to investigate alarm.
[5] **REFER TO** ODCM Section 1/2.1.1

References:

AOI-31
1-45W600-57-18
1-45W600-77-1
CM-9.09
ODCM

Date _________

8.16 Release of the Cask Decontamination Collector Tank to Cooling Tower Blowdown

NOTES

1) The Unit SRO should confirm there are no planned interruptions in Hydro Plant operation which would result in isolation of normal CTBD diffuser flow, in order to minimize transportation of contaminants to the yard holding pond.

2) MI-0.018, NSSS FILTER CHANGEOUT, may be referenced if 0-RM-90-122 alarms on hi rad levels with filters installed on CDCT system.

[1] **ENSURE** Section 8.15 complete. _________

[2] **IF** 0-RE-90-122 is OOS, **THEN** (**N/A** part of this Step **NOT** performed):

PERFORM 0-RE-90-122 INOPERABLE steps, **OR** _________

REQUEST Instrument Maintenance (IM) to ENSURE 0-RE-90-122 properly aligned for service. [C.8] _________

[3] **IF** 0-RE-90-122 is INOPERABLE, **THEN**

PERFORM the following:

[3.1] **REMOVE** fuses N-17 and N-18 in Panel 1-R-76. _________ _________ CV

[3.2] **PLACE** temporary jumper (clip-type) at TB-248 across terminals 8 and 10 in Panel 1-R-72. _________ _________ CV

[3.3] **IDENTIFY** jumper with an Information Tag noting procedure and step number. _________

Date________ INITIALS

8.9 Gas Decay Tank Release Followed by Nitrogen Header Purge (continued)

[13] **OBTAIN** Decay Tank Release Permit (0-ODI-90-5) from Chemistry, **AND**

COMPLETE in conjunction with this Instruction. _________

NOTE

IV for Step [14] may be performed by Chemistry.

[14] **ENSURE** 0-ODI-90-5 Release Permit approved by SM/SRO. _________

_________ IV

[15] **OBTAIN** SRO approval and verification that release is authorized, and instructions are correct for release of the applicable Gas Decay Tank.

_________ SRO

[16] **IF** 0-RE-90-118 is INOPERABLE, **THEN**

ENSURE Chemistry Countroom is performing compensatory measures,

AND

This step will NOT compensate for a loss of power to the 0-RM-90-118

NOTIFY Instrument Maintenance to adjust setpoint of 0-RE-90-118 to avoid inadvertent isolation of release, **AND**

PERFORM Independent Verification for Steps [19], [20], and [21]. [C.4] _________

CAUTION

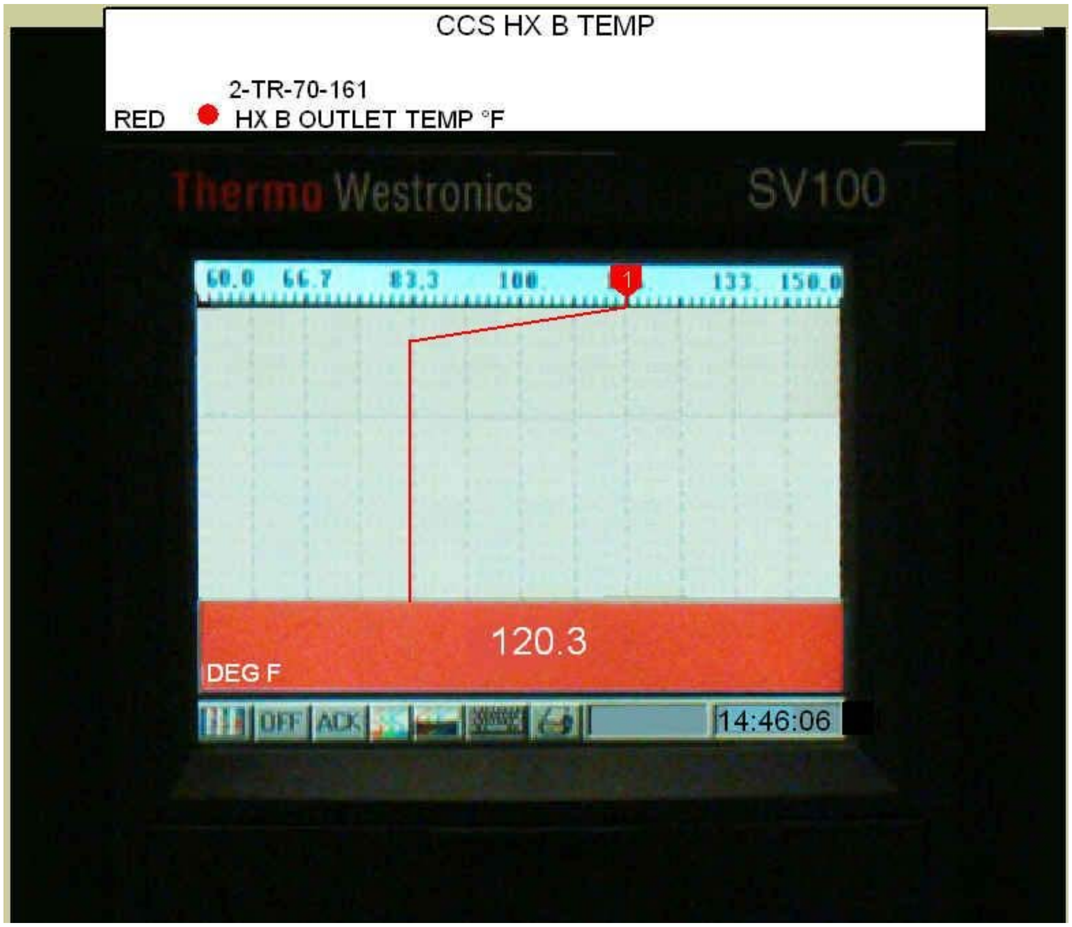
ABGTS should **NOT** be operated for testing purposes or for releases of waste decay tank at the same time both containment purge exhaust fans are in operation. Operation of one purge exhaust fan with the ABGTS is acceptable.

[17] **START** ABGTS Fan A-A per SOI-30.06. _________

52.

Given the following conditions:

- Unit 1 is at 100%.
- The following indication is observed on 0-M-27B.



Which ONE of the following describes the cause of the above indications?

A Loss of the ________ ERCW header.

A. 1A

B. 1B

C. 2A

D. 2B

CORRECT ANSWER: C

DISTRACTOR ANALYSIS:

A. Incorrect: The closed cooling water system is called Component Cooling Water (CCS) at WBN. The service water system at WBN is called Essential Cooling Water System (ERCW). CCS has three heat exchangers which are all cooled by ERCW. The A trained CCS Hxs are A and B (even though the B nomenclature would imply affiliation with the B train). The B trained CCS HXs is therefore C. The A and B CCS Hx are normally cooled by ERCW header 2A and alternately cooled by ERCW 1A. The C CCS Hx is normally cooled by ERCW header 2B and alternately cooled by ERCW header 1B. The isolations between the same trained ERCW headers are locked shut in accordance with the fire protection report. Use of the alternate headers will therefore incur Operating Requirements (ORs). Use of the alternate headers will also incur required actions in accordance with the technical specifications as two ERCW headers will be cross connected and thus rendering both inoperable. These alternate headers are used during outages and the abnormal operating procedures. All of the aforementioned may be found on print 1-47W845 sheets 2 and 5 (for two notes). It is plausible to believe that the 1A ERCW header supplies the B CCS Hx because even though it normally is not aligned, it physically can supply the Hx

B. Incorrect: The 1B ERCW header does not supply the B CCS Hx. It is plausible to believe such because as discussed the very name of the B CCS Hx leads one to believe that it is affiliated with the B train and thus cooled by a B trained ERCW header.

C. Correct: The 2A ERCW header does supply the B CCS Hx.

D. Incorrect: The 2B ERCW header does not supply the B CCS Hx. It is plausible to believe such because as discussed the very name of the B CCS Hx leads one to believe that it is affiliated with the B train and thus cooled by a B trained ERCW header. Furthermore, it is plausible to believe this if one remembers that Unit 2’s ERCW supplies the CCS Hxs.

Question Number: 52

Tier: 2 Group: 1

K/A: 076 Service Water System (SWS)
K3 Knowledge of the effect that a loss or malfunction of the SWS will have on the following:
K3.01 Closed cooling water

Importance Rating: 3.4* 3.6

10 CFR Part 55: (CFR: 41.7 / 45.6)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to associate a trend observed on a CCS Hx's temperature recorder with the appropriate malfunction of the ERCW system.

Technical Reference: 1-47W845-2
1-47W845-5

Proposed references to be provided: None

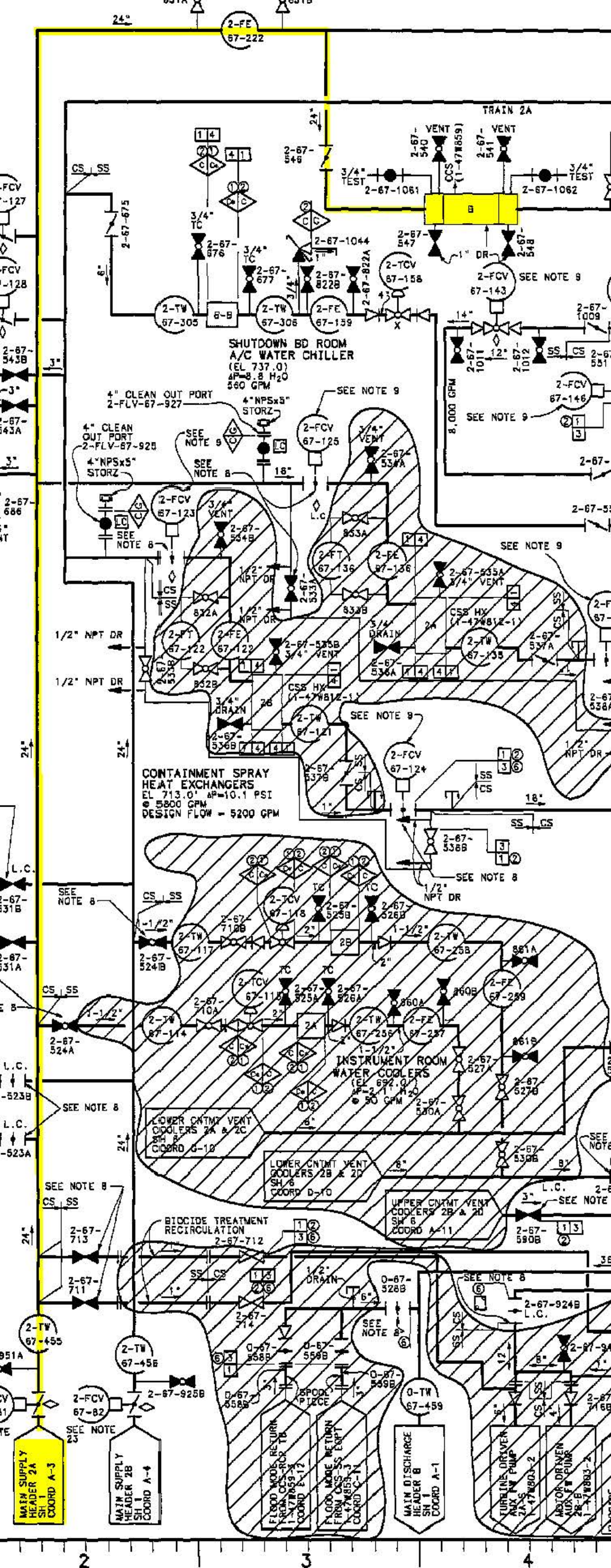
Learning Objective: 3-OT-STG-070A, COMPONENT COOLING WATER SYSTEM
5. EXPLAIN, the physical connections and/or cause-effect relationships
between the CCS and the following systems:
c. ERCW

Cognitive Level:
Higher X
Lower ____

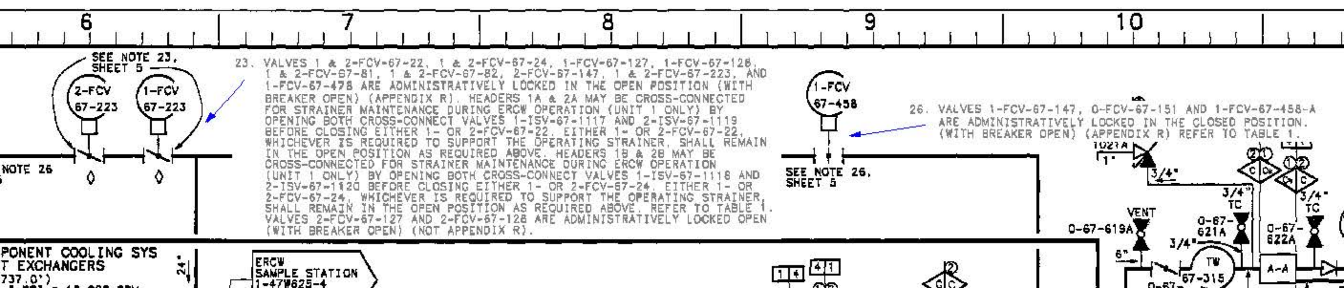
Question Source:
New X
Modified Bank ____
Bank ____

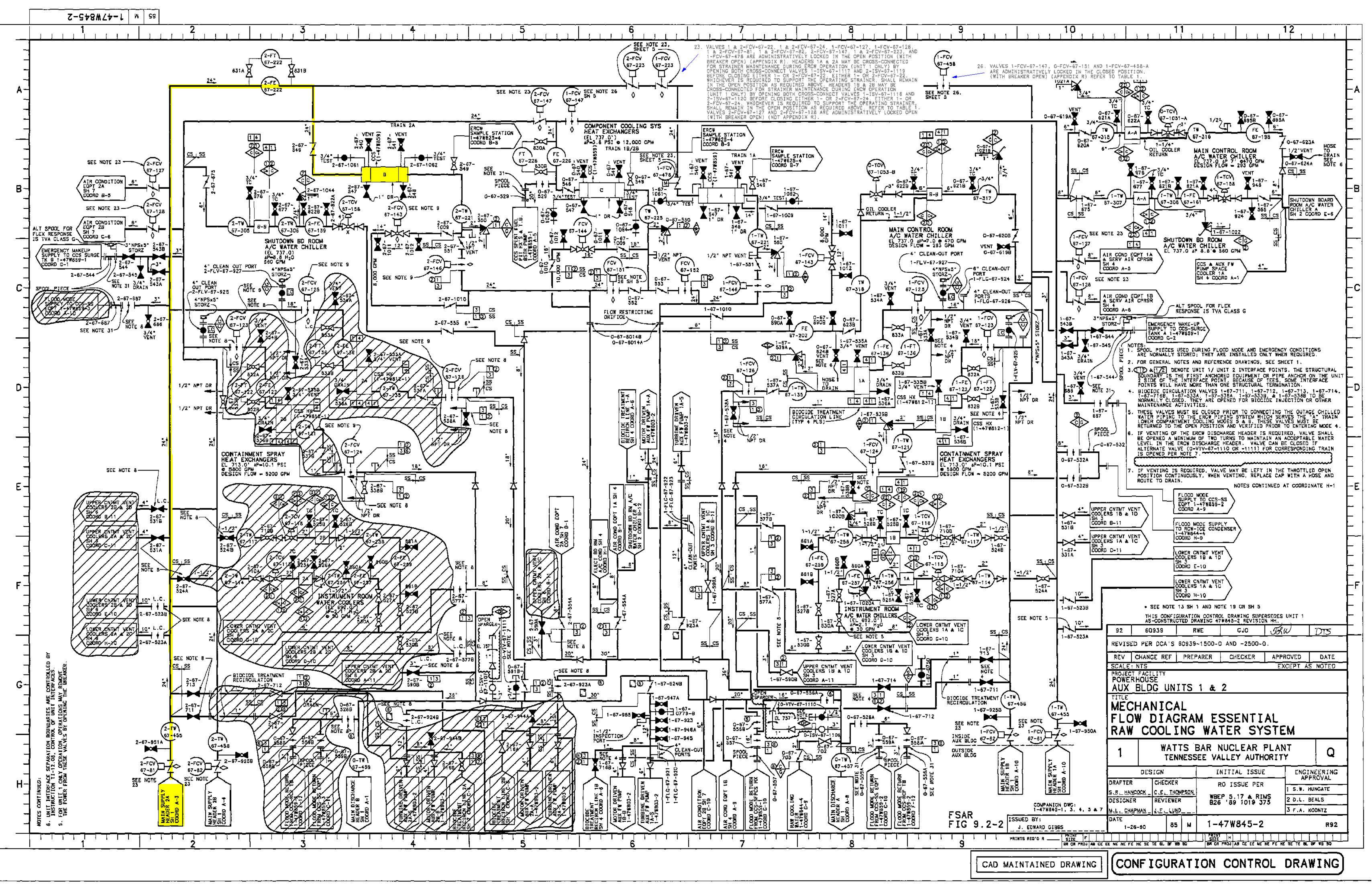
Question History: New question for the 2015-301 NRC RO Exam

Comments:



TRAIN 2A
SHUTDOWN BD ROOM
A/C WATER CHILLER
(EL 737.0)
ΔP=8.8 H2O
560 GPM
4" CLEAN OUT PORT
2-FLV-67-927
4" CLEAN
OUT PORT
2-FLV-67-928
4"NPSx5"
STORZ
SEE NOTE 9
SEE NOTE 8
CONTAINMENT SPRAY
HEAT EXCHANGERS
EL 713.0' ΔP=10.1 PSI
@ 5800 GPM
DESIGN FLOW = 5200 GPM
CSS HX
(1-47W812-1)
INSTRUMENT ROOM
WATER COOLERS
LOWER CNTMT VENT
COOLERS 2A & 2C
SH 6
COORD G-10
LOWER CNTMT VENT
COOLERS 2B & 2D
SH 6
COORD D-10
UPPER CNTMT VENT
COOLERS 2B & 2D
SH 6
COORD A-11
BIOCIDE TREATMENT
RECIRCULATION
2-67-712
SPOOL
PIECE
MAIN SUPPLY
HEADER 2A
SH 1
COORD A-3
MAIN SUPPLY
HEADER 2B
SH 1
COORD A-4
FLOOD MODE RETURN
FROM CCS-RCP TB
1-47W859-3
COORD E-12
FLOOD MODE RETURN
FROM CCS-SS EQPT
1-47W859-3
COORD C-11
MAIN DISCHARGE
HEADER B
SH 1
COORD A-1
TURBINE DRIVEN
AUX FW PUMP
2A-S
1-47W803-2
MOTOR DRIVEN
AUX FW PUMP
2B-B
1-47W803-2
8,000 GPM
2-FE 67-222
2-TW 67-305
2-TW 67-306
2-FE 67-159
2-TCV 67-158
2-FCV 67-143
2-FCV 67-146
2-FCV 67-125
2-FCV 67-123
2-FT 67-136
2-FE 67-136
2-FT 67-122
2-FE 67-122
2-TW 67-135
2-TW 67-121
2-FCV 67-124
2-TW 67-117
2-TCV 67-118
2-TW 67-258
2-FE 67-259
2-TCV 67-115
2-TW 67-114
2-TW 67-256
2-FE 67-257
2-TW 67-455
2-TW 67-456
2-FCV 67-82
0-TW 67-459
2
3
4


6
7
8
9
10
SEE NOTE 23,
SHEET 5
2-FCV
67-223
1-FCV
67-223
NOTE 26
23. VALVES 1 & 2-FCV-67-22, 1 & 2-FCV-67-24, 1-FCV-67-127, 1-FCV-67-128, 1 & 2-FCV-67-81, 1 & 2-FCV-67-82, 2-FCV-67-147, 1 & 2-FCV-67-223, AND 1-FCV-67-478 ARE ADMINISTRATIVELY LOCKED IN THE OPEN POSITION (WITH BREAKER OPEN) (APPENDIX R). HEADERS 1A & 2A MAY BE CROSS-CONNECTED FOR STRAINER MAINTENANCE DURING ERCW OPERATION (UNIT 1 ONLY) BY OPENING BOTH CROSS-CONNECT VALVES 1-ISV-67-1117 AND 2-ISV-67-1119 BEFORE CLOSING EITHER 1- OR 2-FCV-67-22. EITHER 1- OR 2-FCV-67-22, WHICHEVER IS REQUIRED TO SUPPORT THE OPERATING STRAINER, SHALL REMAIN IN THE OPEN POSITION AS REQUIRED ABOVE. HEADERS 1B & 2B MAY BE CROSS-CONNECTED FOR STRAINER MAINTENANCE DURING ERCW OPERATION (UNIT 1 ONLY) BY OPENING BOTH CROSS-CONNECT VALVES 1-ISV-67-1118 AND 2-ISV-67-1120 BEFORE CLOSING EITHER 1- OR 2-FCV-67-24. EITHER 1- OR 2-FCV-67-24, WHICHEVER IS REQUIRED TO SUPPORT THE OPERATING STRAINER, SHALL REMAIN IN THE OPEN POSITION AS REQUIRED ABOVE. REFER TO TABLE 1. VALVES 2-FCV-67-127 AND 2-FCV-67-128 ARE ADMINISTRATIVELY LOCKED OPEN (WITH BREAKER OPEN) (NOT APPENDIX R).
1-FCV
67-458
SEE NOTE 26,
SHEET 5
26. VALVES 1-FCV-67-147, 0-FCV-67-151 AND 1-FCV-67-458-A ARE ADMINISTRATIVELY LOCKED IN THE CLOSED POSITION. (WITH BREAKER OPEN) (APPENDIX R) REFER TO TABLE 1.
1027A
1"
3/4"
3/4"
TC
3/4"
TC
VENT
0-67-619A
0-67-
621A
0-67-
622A
3/4"
6"
TW
67-315
A-A
PONENT COOLING SYS
T EXCHANGERS
737.0')
24"
ERCW
SAMPLE STATION
1-47W625-4



1-47W845-2
COMPONENT COOLING SYS HEAT EXCHANGERS
SHUTDOWN BD ROOM A/C WATER CHILLER
MAIN CONTROL ROOM A/C WATER CHILLER
CONTAINMENT SPRAY HEAT EXCHANGERS
INSTRUMENT ROOM WATER COOLERS
INSTRUMENT ROOM A/C WATER CHILLERS
23. VALVES 1 & 2-FCV-67-22, 1 & 2-FCV-67-24, 1-FCV-67-127, 1-FCV-67-128, 1 & 2-FCV-67-81, 1 & 2-FCV-67-82, 2-FCV-67-147, 1 & 2-FCV-67-223, AND 1-FCV-67-478 ARE ADMINISTRATIVELY LOCKED IN THE OPEN POSITION (WITH BREAKER OPEN) (APPENDIX R). HEADERS 1A & 2A MAY BE CROSS-CONNECTED FOR STRAINER MAINTENANCE DURING ERCW OPERATION (UNIT 1 ONLY) BY OPENING BOTH CROSS-CONNECT VALVES 1-ISV-67-1117 AND 2-ISV-67-1119 BEFORE CLOSING EITHER 1- OR 2-FCV-67-22. EITHER 1- OR 2-FCV-67-22, WHICHEVER IS REQUIRED TO SUPPORT THE OPERATING STRAINER, SHALL REMAIN IN THE OPEN POSITION AS REQUIRED ABOVE. HEADERS 1B & 2B MAY BE CROSS-CONNECTED FOR STRAINER MAINTENANCE DURING ERCW OPERATION (UNIT 1 ONLY) BY OPENING BOTH CROSS-CONNECT VALVES 1-ISV-67-1118 AND 2-ISV-67-1120 BEFORE CLOSING EITHER 1- OR 2-FCV-67-24. EITHER 1- OR 2-FCV-67-24, WHICHEVER IS REQUIRED TO SUPPORT THE OPERATING STRAINER, SHALL REMAIN IN THE OPEN POSITION AS REQUIRED ABOVE. REFER TO TABLE 1. VALVES 2-FCV-67-127 AND 2-FCV-67-128 ARE ADMINISTRATIVELY LOCKED OPEN (WITH BREAKER OPEN) (NOT APPENDIX R).
26. VALVES 1-FCV-67-147, 0-FCV-67-151 AND 1-FCV-67-458-A ARE ADMINISTRATIVELY LOCKED IN THE CLOSED POSITION. (WITH BREAKER OPEN) (APPENDIX R) REFER TO TABLE 1.
NOTES:
1. SPOOL PIECES USED DURING FLOOD MODE AND EMERGENCY CONDITIONS ARE NORMALLY STORED; THEY ARE INSTALLED ONLY WHEN REQUIRED.
2. FOR GENERAL NOTES AND REFERENCE DRAWINGS, SEE SHEET 1.
3. 1 2 & 1/2 DENOTE UNIT 1/ UNIT 2 INTERFACE POINTS. THE STRUCTURAL BOUNDARY IS THE FIRST ANCHORED EQUIPMENT OR PIPE ANCHOR ON THE UNIT 2 SIDE OF THE INTERFACE POINT. BECAUSE OF TEES, SOME INTERFACE POINTS WILL HAVE MORE THAN ONE STRUCTURAL TERMINATION.
4. BIOCIDE CIRCULATION VALVES 1-67-711, 1-67-712, 1-67-713, 1-67-714, 1-67-716B, 1-67-533A, 1-67-538A, 1-67-533B, & 1-67-538B TO BE NORMALLY CLOSED. THEY ARE OPENED FOR BIOCIDE INJECTION OR OTHER MAINTENANCE ACTIVITIES.
5. THESE VALVES MUST BE CLOSED PRIOR TO CONNECTING THE OUTAGE CHILLED WATER PIPING TO THE ERCW PIPING SYSTEM WHICH SERVES THE "A" TRAIN LOWER COMPARTMENT COOLING MODES 5 & 6. THESE VALVES MUST BE RETURNED TO THE OPEN POSITION AND VERIFIED PRIOR TO ENTERING MODE 4.
6. IF VENTING OF THE ERCW DISCHARGE HEADER IS REQUIRED, VALVE SHALL BE OPENED A MINIMUM OF TWO TURNS TO MAINTAIN AN ACCEPTABLE WATER LEVEL IN THE ERCW DISCHARGE HEADER. VALVE CAN BE CLOSED IF ALTERNATE VALVE (0-VTV-67-1110 OR -1111) FOR CORRESPONDING TRAIN IS OPENED PER NOTE 7.
7. IF VENTING IS REQUIRED, VALVE MAY BE LEFT IN THE THROTTLED OPEN POSITION CONTINUOUSLY. WHEN VENTING, REPLACE CAP WITH A HOSE AND ROUTE TO DRAIN.
NOTES CONTINUED AT COORDINATE H-1
NOTES CONTINUED:
8. UNIT INTERFACE SEPARATION BOUNDARIES ARE CONTROLLED BY INSTRUCTION TI-12.08, CONTROL OF UNIT INTERFACES.
9. FOR UNIT 1 ONLY OPERATION, OPERATIONS MAY REMOVE THE POWER FROM THESE VALVES BY OPENING THE BREAKER.
* SEE NOTE 13 SH 1 AND NOTE 19 ON SH 5
THIS CONFIGURATION CONTROL DRAWING SUPERSEDES UNIT 1 AS-CONSTRUCTED DRAWING 47W845-2 REVISION HH.
92 | 60939 | RWE | GJO | TRW | DTS
REVISED PER DCA'S 60939-1500-0 AND -2500-0.
REV | CHANGE REF | PREPARER | CHECKER | APPROVED | DATE
SCALE: NTS EXCEPT AS NOTED
PROJECT FACILITY
POWERHOUSE
AUX BLDG UNITS 1 & 2
TITLE
MECHANICAL
FLOW DIAGRAM ESSENTIAL
RAW COOLING WATER SYSTEM
1
WATTS BAR NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY
Q
DESIGN
DRAFTER S.R. HANCOCK
CHECKER C.E. THOMPSON
DESIGNER M.L. CHAPMAN
REVIEWER J.F. LUND
INITIAL ISSUE
RO ISSUE PER
WBEP 5.17 & RIMS B26 '89 1019 375
ENGINEERING APPROVAL
1 S.W. HUNGATE
2 D.L. BEALS
3 F.A. KOONTZ
DATE 1-26-90
85 M
1-47W845-2
R92
COMPANION DWG: 1-47W845-1, 3, 4, 5 & 7
ISSUED BY: J. EDWARD GIBBS
FSAR FIG 9.2-2
CAD MAINTAINED DRAWING
CONFIGURATION CONTROL DRAWING

53.

Given the following timeline:

00:00:00	Unit 2 is at 100% power.
00:01:00	The station's Non-essential control air is lost and Non-essential air header pressure is 0 psig.

Which ONE of the following describes the response of the Unit 2 MSIVs to this condition?

At 00:01:01, the Unit 2 MSIVs will ________.

A. have CLOSED

B. remain OPEN as the MSIVs are provided with essential control air

C. remain OPEN until the MSIVs air accumulator's inventory is depleted

D. remain OPEN as a N_2 backup is provided to the MSIVs' instrument air supply

CORRECT ANSWER: C

DISTRACTOR ANALYSIS:

A. Incorrect: In accordance with WBN system description NPG-SDD-WBN2-1-4002, An air accumulator is provided at each MSIV air supply line to prevent MSIV closure on loss of control air. [page 55 of NPG-SDD-WBN2-1-4002] This distractor is plausible from two perspectives. Firstly, if the applicant simply fails to recall that the MSIVs possess an air accumulator then the applicant would deduce that a instrument air pressure of 0 psig would result in the MSIVs travelling to their failed position of closed. Secondly, it is a common misconception to believe that the MSIVs have actuators which are similar to other common Air Operated Valves. Typical AOVs have an air admission port and a bleed orifice. A continuous supply of air is required to maintain the AOV in the desired position (otherwise the bleed orifice will depressurize the valve operator). The MSIV has a pneumatic cylinder for an operator. Air is admitted via a solenoid valved line and air may be vented off via a solenoid valved line. If the air supply is lost and the vent line is not opened, the valve will not change state as there is no exit path for the contained instrument air.

B. Incorrect: As seen on print 2-47W848-9, the MSIVs (shown on the print as FCV-1-11 and FCV-1-22 – the other two MSIVs are depicted on 2-47W848-6) are afforded Non-essential control air. It is plausible to believe that these valves be afforded essential control air as are many other air operated valves in the plant. Amplifying the plausibility of this is that the MSIVs fulfill a T/S LCO; that being, T/S LCO 3.7.2, MSIVs. The applicant may assume that due to the fact that the valves are safety related that their air supply must be safety related.

C. Correct: Given the information presented in the analysis for distractor A, one may understand that the MSIVs will remain open until their air accumulators inventory deplete due to operating cylinder leakage. (which as seen in 0-MI-1.002, Main Steam Isolation Valve Maintenance is miniscule).

D. Incorrect: As seen on prints 2-47W848-9 and 2-47W848-6 there is no backup N2 supply to the instrument air for the MSIVs. It is plausible to believe that one does exist because several valves impacting the steam generators do receive a backup N2 supply. Examples of such are SG PORVs and AFW LCVs and PCVs. Additional strengthening to the plausibility for the existence of a backup N2 supply is that U2 was selected for the Unit affected. More air operated valves are receiving backup N2 supplies on U2 than on U1. Most of these backups are being provided due to either Appendix R analysis or Fukishima response.

Question Number: 53

Tier: 2 Group: 1

K/A: 078 K1.05
078 Instrument Air System (IAS)
K1 Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems:
K1.05 MSIV air

Importance Rating: 3.4* 3.5*

10 CFR Part 55: CFR: 41.2 to 41.9 / 45.7 to 45.8

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because it requires the applicant to both understand the physical connection of the instrument air system to the MSIVs and to comprehend the cause-effect relationship that a loss of the instrument air system would have on the MSIVs (and the MSIVs' air accumulators).

Technical Reference: 2-47W848-6
2-47W848-9
NPG-SDD-WBN2-1-4002
0-MI-1.002
T/S LCO 3.7.2

Proposed references to be provided: None

Learning Objective: 3-OT-STG-001A, MAIN STEAM
04. EXPLAIN the physical connections and/or cause-effect relationships
between the Main Steam System and the following systems:
k. Instrument Air

Cognitive Level:
Higher ____
Lower X

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam
Comments:

3.3.2 MSIV Controls (continued)

The redundant exhaust valves are energize-to-close and they are arranged in parallel paths so that one train of the exhaust valves needs to be deenergized to exhaust air from the operator. The exhaust valves in a train are arranged in series to preclude inadvertent closing of the MSIV. The test valves are energize-to-close and they are normally opened (deenergized) for exhaust air from the operator to go through a low resistance path for MSIV quick closing. By closing a test exhaust valve and an exhaust valve in the same train, exhaust from MSIV operator will be routed through a high resistance flow path for slow MSIV closing. All solenoid valves, except the test solenoid valves, are energized to open the MSIV.

Redundant sets of solenoid valves are independently operated. Power supplies for Unit 1 train A and train B solenoid valves come from 125V DC Vital Battery Boards I and II respectively (Table 2.2). Similarly, power supplies for Unit 2 train A and train B solenoid valves come from the 125V DC Vital Battery Boards III and IV, respectively (Table 2.2). The supply solenoid valve, one exhaust valve and the test solenoid valve in the same train are powered from a fused circuit of the applicable battery board. The second exhaust solenoid valve is powered from another fused circuit of the same board. Separate fused power supply circuits for the two exhaust valves are provided to prevent MSIV closure in the event of a power supply circuit fault.

Control air (System 32) is used to supply all MSIV operations. An air accumulator is provided at each MSIV air supply line to prevent MSIV closure on loss of control air. The control air is a quality air which has been filtered and dried to assure that the solenoid valves will not be degraded by poor air quality and failed to respond on deenergization.

The MSIV is provided with position switches to indicate its fully closed, 90 and 100% open positions; and for MSIV test closing control circuit. The MSIV positions are indicated in the MCR.

Redundant solenoid valves, limit switches, wiring and terminals for field interface cables are physically separated in protective enclosures and conduits to assure that effects of harsh environment resulting from high energy line break will not prevent the solenoid valves from performing their safety function. The solenoid valve safety function is to deenergize and remain in this state to keep MSIV closed. These solenoid valves are subjected to Environmental Qualification Program and their qualification is documented in Ref. 7.2.53.

The main steam isolation signal is generated by the Solid State Protection System (SSPS) which is a part of System 99 - Reactor Protection System (Ref. 7.2.44). The main steam isolation signal is generated from either of the following conditions:

a) High-high containment pressure,

b) Low steamline pressure in any steamline (lead-lag compensated, P-11 permissive),

c) High steamline negative pressure rate in any steamline (rate-lag compensated, P-11 permissive).

Refer to Figure 3.3-1 for the main steam isolation signal development logic.

53178-045-1
HOLD
53178-046-0
HOLD
CLASS H
CLASS G
TURBINE BUILDING
CONTROL BUILDING
FROM TURB BLDG EL 729'
1-47W848-3 COORD H-1
TO EL 757'
1-47W848-7 COORD A-2
This is the non-essential air header. Class G is non-safety related piping.
These two check valves prevent the accumulators (air tanks) from depressurizing during a loss of the non-essential air header.
These are the MSIVs
(V402)
2-FCV-1-4
(V403)
2-FCV-1-29
AIR TANK
2-32-592
2-32-593
2-32-594
(V236)
2-FCV-3-185
(239)
2-FCV-3-188
(V724)
2-FCV-3-245
(V404)
2-FCV-1-150
(V442)
2-TCV-44-41
(V595)
2-TIC-44-41
(V21)
2-FCO-30-109
(V439)
2-FCO-30-21
(V454)
2-FCO-30-104
(V453)
2-FCO-30-105
(V458)
2-FCO-30-108
(V1047)
2-L-218B
(V807)
2-FCO-30-294
(V1010)
2-TCV-31-378
(V806)
2-FCO-30-295
(V444)
ABSCE
NOTE 15
UNIT 2

53917-059-0
HOLD
53276-306-0
HOLD
53917-060-0
HOLD
53276-306-0
HOLD
53917-060-0
HOLD
53178-040-0
HOLD
NON ESSENTIAL REACTOR BLDG AND NORTH VALVE ROOM
(V798) FCV-30-51
3548
(V775) FCV-43-58
3556
(V774) FCV-43-55
3557
(V776) FCV-43-61
3558
(V777) FCV-43-64
TCV-67-92
3562
(V773) FCV-43-35
3573
(V788) FCV-30-12
3574
(V786) FCV-30-9
EL 703'
(V638) FCV-63-95
3660
(V430) PNL L-173
ABSCE
NOTE 3
(V1000) FCV-31-326
3666
(V1004) FCV-31-330
3667
3663
(V791) FCV-30-16
1438
1440
1442
1445
1447
1449 SPARE
1450
1451
1456
1459
1439
1441
1443
1444
3575
(V795) FCV-30-37
1446
3576
(V784) FCV-30-7
1448
2-32-610
AIR TANK
MSIV
(V682) FCV-1-22
2-32-609
1462
1453
1452
(V679) FCV-1-148
3577
3578
(V722) FCV-3-239
(V237) FCV-3-186
3579
(V238) FCV-3-187
3580
1464
1465
NORTH VLV ROOM
SHIELD WALL
1454
1455
1463
2-32-611
(V676) FCV-1-11
MSIV
2-32-612
AIR TANK
1457
1458
3668
(V805) FCV-30-59
1460
1330
(V258A) TC-30-312
3669
(V1048) FCO-30-312
3670
(V258C) TC-30-314
3671
3673
(V687) FCV-1-149
3674
(V723) FCV-3-242

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Four MSIVs shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when all MSIVs are closed and de-activated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
C. -------------NOTE---------------- Separate Condition entry is allowed for each MSIV. ------------------------------------- One or more MSIVs inoperable in MODE 2 or 3.	C.1 Close MSIV. <u>AND</u> C.2 Verify MSIV is closed and de-activated.	8 hours Once per 7 days

(continued)

Date ________

6.6 Air Cylinder Refurbishment and Bench Pressure Testing (continued)

NOTES

1) Air cylinder should now be leak tested for leakage past piston and all static seals (2 and 17 included), in addition to rod seal (11).

2) 125 psi supply pressure required. Air or nitrogen may be used. For air, Control Air System air may be used, or Service Air may be used if a dryer and 50 micron filter are used. To conserve nitrogen, cylinder may be filled with air, then final pressure obtained with nitrogen, if desired.

[36] **ENSURE** cylinder cap port (above piston) is open to atmosphere, **THEN:**

A. **CONNECT** test rig to cylinder head port (below piston).

B. **PRESSURIZE** to 125 psig.

C. **USING** Snoop leak detector, **TEST** tube end seal and piston rod seal.

D. **VERIFY** no leakage.

____
Initials

[37] **ENSURE** rod end still pressurized to 125 psig, **THEN**

CLOSE test rig shutoff valve to isolate air in cylinder, **AND**

PERFORM following:

A. **MONITOR** test gauge for 1 minute, **AND**

OBSERVE pressure drop and **RECORD**.

MSIV air operating cylinder leakage is miniscule.

B. **IF** pressure drop is 0 psi, piston seal leakage is acceptable, **THEN**

GO TO Step 6.6[3].

54.

Given the following conditions:

- Unit 1 is at 100% power.
- A leak develops on the Control Air system.
- Control air pressure is 72 psig and LOWERING⇓.

Which ONE of the following identifies the system response that occurred to maintain Train A Essential Air Header pressure?

A. 1-FCV-32-80, AUX AIR TO RX BLDG TR A, SHUTS

B. Auxiliary air compressor A-A STARTED and LOADED

C. 1-FCV-32-110, NON-ESS AUX AIR TO RX BLDG, SHUTS

D. 0-PCV-33-4, Service Air Receiver isolation valve, OPENED

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: Plausible since this valve does go closed to isolate essential air from control air, however this valve doesn't go closed until system pressure decreases to < 70 psig.

B. Correct: At 83 psig decreasing the A-A Aux air compressor starts in an attempt to maintain A essential air header pressure above 70 psig to prevent auto closure of the containment supply valves.

C. Incorrect: Plausible if the candidate thought that by closing the nonessential air to CNMT that this would maintain essential air header pressure; however, this valve does not go closed until pressure is < 70 psig.

D. Incorrect: Plausible, if candidate thought that Service air could be used to supply essential air, however PCV-33-4 goes closed on decreasing pressure (<80 psig) not open.

Question Number: 54

Tier: 2 Group: 1

K/A: 078 Instrument Air System (IAS)
K4 Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following:
K4.02 Cross-over to other air systems

Importance Rating: 3.2 3.5

10 CFR Part 55: (CFR: 41.7)

10CFR55.43.b: Not applicable

K/A Match: Question matches K/A by having the candidate determine when the standby Aux air compressors automatically start and supply essential air.

Technical Reference: 0-AOI-10

Proposed references to be provided: None

Learning Objective: 3-OT-SYS032A, Control Air System
6. EXPLAIN the Control and Service Air System design features and/or interlocks that provide the following: (IER 11-3, Having a solid understanding of plant design, engineering principles and sciences)
b. Cross-over to other air systems
d. Automatic isolation of sections of the air system
f. Automatic control of station air pressure

Cognitive Level:

Higher ___
Lower X

Question Source:

New ___
Modified Bank ___
Bank X

Question History: Bank question 078 K4.02 number 54.

Comments:

2.3 Automatic Actions

A. Possible Rx trip due to MFW reg valves or heater drain valves failing closed.

B. MSIVs may fail closed.

C. Operating compressors load in sequence and compressors selected for auto will start on dropping press.

D. Aux Air compressors start at 83 psig dropping.

This is the reason that the "A" answer is correct.

E. Service air isolates from control air at 80 psig dropping.

This is 0-PCV-33-4 shutting.

F. Aux air isolates from control air at 79.5 psig dropping.

G. Essential and non-essential air to the Rx bldg isolates at 70 psig dropping.

H. ERCW to the C&SS air compressors isolate on high flow in conjunction with low press on the compressor ERCW supply headers.

These are 1-FCV-32-80, 102 and 110 shutting.

55.

Given the following timeline:

00:00:00	The polar crane is secured after placing the reactor head on the head stand.
00:01:00	SOURCE RANGE HI FLUX AT SHUTDOWN(81-B) alarms and is acknowledged.
00:01:15	Rising counts are observed on both 1-NI-131 and 1-NI-132.

Which ONE of the following describes if a horn can be heard and the required actions in accordance with 1-ARI-81-87, NIS & Rod Controls?

A horn ____(1)____ be heard in upper containment.

The crew will **FIRST** ____(2)____.

NOTE: *1-AOI-34, Immediate Boration*

A. (1) WILL
(2) ENSURE all personnel are evacuated from CNTMT

B. (1) WILL **NOT**
(2) ENSURE all personnel are evacuated from CNTMT

C. (1) WILL
(2) INITIATE an immediate boration in accordance with 1-AOI-34

D. (1) WILL **NOT**
(2) INITIATE an immediate boration in accordance with 1-AOI-34

CORRECT ANSWER: A

DISTRACTOR ANALYSIS:

A. Correct: As seen in the Westinghouse Technical Manual for the Nuclear Instrumentation system, WBN-VTD-W120-2826, The evacuation horn assembly is also mounted near the reactor. This horn works in conjunction with the Source Range high flux at shutdown bistable relay driver assembly to provide an evacuation alarm for on-site personnel. Referring to the annunciator response for window 81-B. SOURCE RANGE HI FLUX AT SHUTDOWN, one will find that such alarm is generated by bistables NC31CX and NC32CX. On print 1-45W600-57-13, one may find that these two bistable relays are normally energized during the time that their associated source range nuclear instrument detects that the high flux at shutdown is zero (or does not exist). When such a condition occurs, the bistable relay will de-energize causing a contact observed on print 1-45W600-57-17 to shut. This contact causes 120V AC power to be supplied to the containment evacuation horn. This alarm horn is located in upper containment. Considering the steps contained in the annunciator response for window 81-B, one will see that the operating crew will **ENSURE** all personnel are evacuated from CNTMT. This is step [2.1] in such instruction. The next step directs the crew to 1-AOI-34, Immediate Boration. In accordance with TI-12.04, User's Guide for Abnormal and Emergency Operating Instructions, numbered procedure steps are to be performed in the order enumerated within the procedure.

B. Incorrect: It is not correct that a horn would not sound in upper containment. It is plausible to believe that this would be the case because one may believe that an alarm horn does not exist because of the facts that the plant paging system allow for the announcement of emergencies and that during core alterations a fuel handling supervisor is stationed (such individual is a SRO and is in constant communication with the main control room). He would be able to direct the containment evacuation upon the first indication of RISING⇑ counts. Also, as seen in 1-GO-7, Refueling Operations, step [11] directs that, Demonstrate that Plant P/A system AUDIBLE in CNTMT OR A means for containment evacuation has been established. This coupled with the fact that 1-GO-7 does not test the alarm horn inside of upper containment would again lead one to believe that the P/A were the primary mode by which a containment evacuation were conducted. Therefore, it would be reasonable to believe that a separate alarm horn would not be required. Again, it is correct that the first action which the control room staff took is to ensure that personnel were evacuated from containment.

C. Incorrect: While it is correct that the alarm horn will sound within upper containment, it is not correct that the operating crew will initiate a boration prior to ensuring that all personnel are evacuated from CNTMT. It is plausible to believe that this is the case because of the fact that in accordance with TI-12.04, the term ensure is defined as To observe that an expected characteristic or condition exists and, if necessary, to take actions to make the condition occur. Therefore, in accordance with the ARI, the operating crew would only begin a boration after containment was evacuated. In accordance with TI-12.04, initiate is To commence or begin. Generally used to cause the start or beginning of an effort which can **NOT** be completed in a short period of time. Therefore it would be reasonable to believe that the operating crew would *initiate* a containment evacuation and then *initiate* a boration prior to *ensuring* that containment were indeed evacuated.

D. Incorrect: As discussed it is Incorrect and yet plausible that an alarm horn would not sound and that the operating crew would first initiate a boration of the RCS.

Question Number: 55

Tier: 2 Group: 1

K/A: 103 Containment System
A2 Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to Correct: control, or mitigate the consequences of those malfunctions or operations
A2 04 Containment evacuation (including recognition of the alarm)

Importance Rating: 3.5* 3.6*

10 CFR Part 55: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to associate the SOURCE RANGE HI FLUX AT SHUTDOWN annunciator with an alarm horn within upper containment. This association demonstrates the knowledge that a high flux at shutdown will impact the containment system by causing an alarm physically within the containment. The applicant must then use knowledge of 1-ARI-81-87 to identify the action which they would first take.

Technical Reference: WBN-VTD-W120-2826
TI-12.04, User's Guide for Abnormal and Emergency Operating Instructions
1-GO-7, Refueling Operations
1-ARI-81-87, NIS & Rod Controls
1-45W600-92-1
1-45W600-57-13
1-45W600-57-17

Proposed references to be provided: None

Learning Objective: 1-AOI-29, Dropped or Damaged Fuel or Refueling Cavity Seal Failure
3. IDENTIFY Alarms, symptoms, automatic actuations, and other indications of AOI-29, Dropped or Damaged Fuel or Refueling Cavity Seal Failure.
4. Given a set of plant conditions, DESCRIBE operator actions required in response to the following per AOI-29, Dropped or Damaged Fuel or Refueling Cavity Seal Failure:

Cognitive Level:

Higher ______
Lower __X__

Question Source:

New __X__
Modified Bank ______
Bank ______

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Table 1-8. Front Panel Control and Indicator Functions (Continued)

Name	Reference Designation	Function
DETECTOR CURRENT COMPARATOR SECTION (Continued)		
UPPER SECTION switch	S605	A five position, make-before-break switch which when placed in the N41, N42, N43, or N44 position enables the removal of a faulty upper section power range channel. The circuit automatically adjusts the gain of the averaging amplifier alarm so that the set point stays the same. When the switch is in the NORMAL position, all four upper power range channels are averaged and compared.
LOWER SECTION switch	S606	A five-position, make-before-break switch which when placed in the N41, N42, N43 or N44 position enables the removal of a faulty lower section power range channel. The circuit automatically adjusts the gain of the averaging amplifier alarm so that the set point stays the same. When the switch is in the NORMAL position all four lower power range channels are averaged and compared.

1.6.3 REMOTE EQUIPMENT.

A preamplifier, a loud speaker, a volume control, and an evacuation horn assembly are supplied as part of the system for installation at a remote location. The description of the preamplifier appears in paragraph 1.2.3.11 under the Source Range channel discussion. The volume control and speaker are mounted near the reactor and operate in conjunction with the remote speaker output of the Audio Count Rate Channel drawer assembly. The volume control is mounted in a housing and has an impedance of 8 ohms and a power rating of 50 watts. The volume control provides a means of adjusting the sound level of the remote speaker over a range of approximately 0 to 100 decibels.

The evacuation horn assembly is also mounted near the reactor. This horn works in conjunction with the Source Range high flux at shutdown bistable relay driver assembly to provide an evacuation alarm for on-site personnel. The horn is screw mounted and has two horns separated by 180 degrees. The horn requires 118 volts at 50/60 Hertz.

81-B

Source	**Setpoint**
Bistables: 1N31AX 1N32AX NC31CX NC32CX	0.5 decade increase above background count rate at shutdown (variable)

SOURCE RANGE
HI FLUX
AT SHUTDOWN

(Page 1 of 1)

Probable Cause:

A. Positive reactivity insertion from:
1. RCS cooldown
2. RCS dilution
3. Rod withdrawal
4. Xenon decay

Corrective Action:

[1] **ENSURE** CNTMT evacuation alarm initiates.

[2] **IF** count rate continues to rise, **THEN**

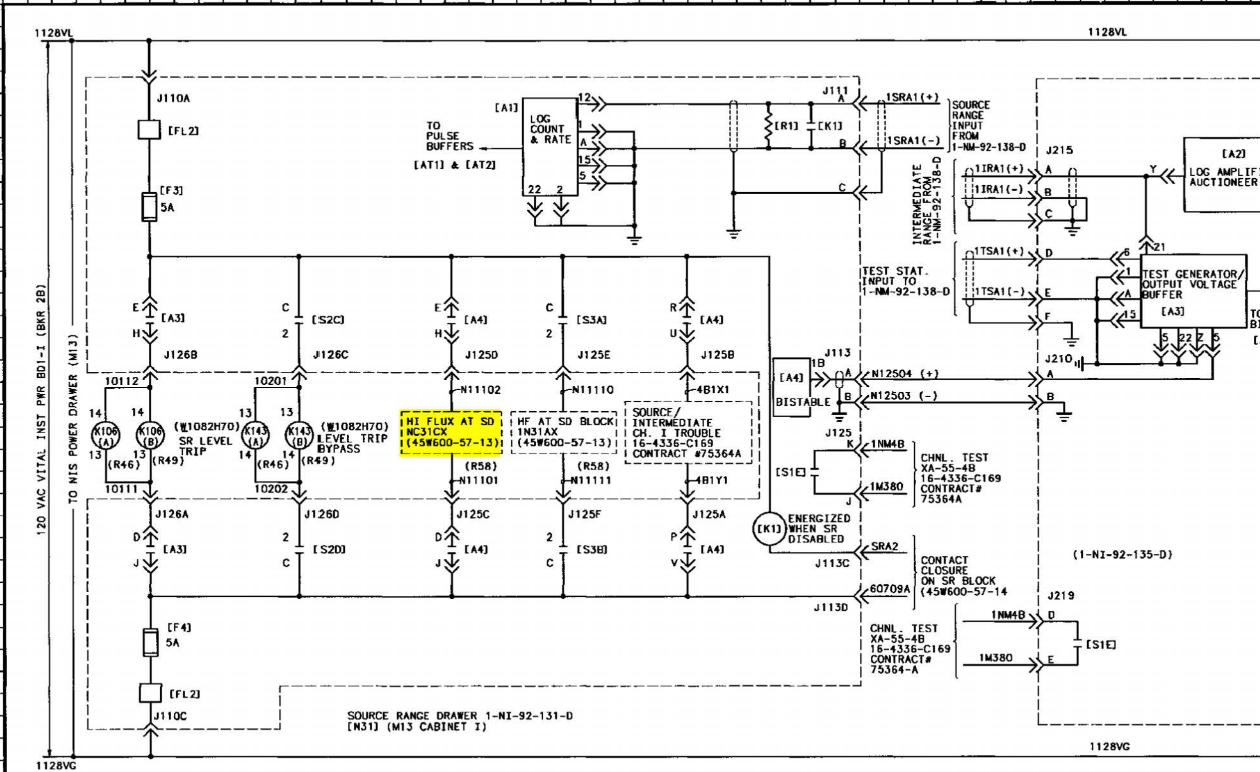
[2.1] **ENSURE** all personnel are evacuated from CNTMT.

[2.2] **GO TO** 1-AOI-34, *IMMEDIATE BORATION*.

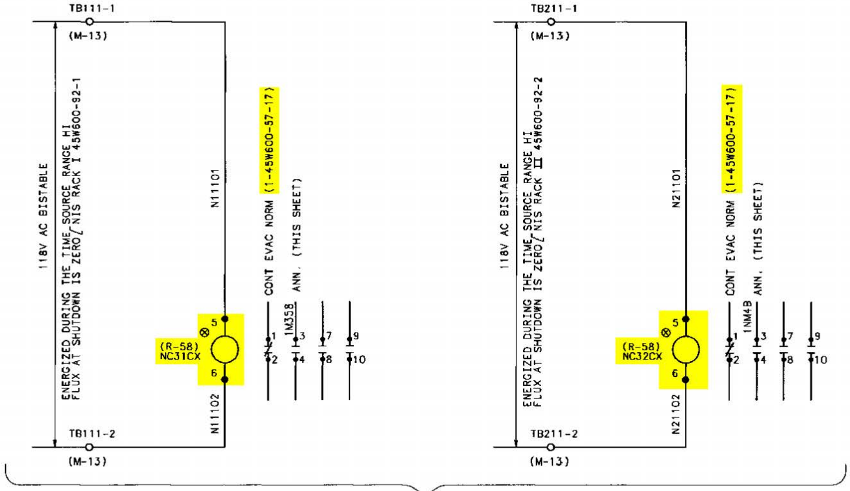
[3] **GO TO** 1-AOI-4, NUCLEAR *INSTRUMENTATION MALFUNCTION*.

References:

1-45W600-92-1
1-45W600-92-2
1-45W600-57-13
1-AOI-4
1-AOI-34

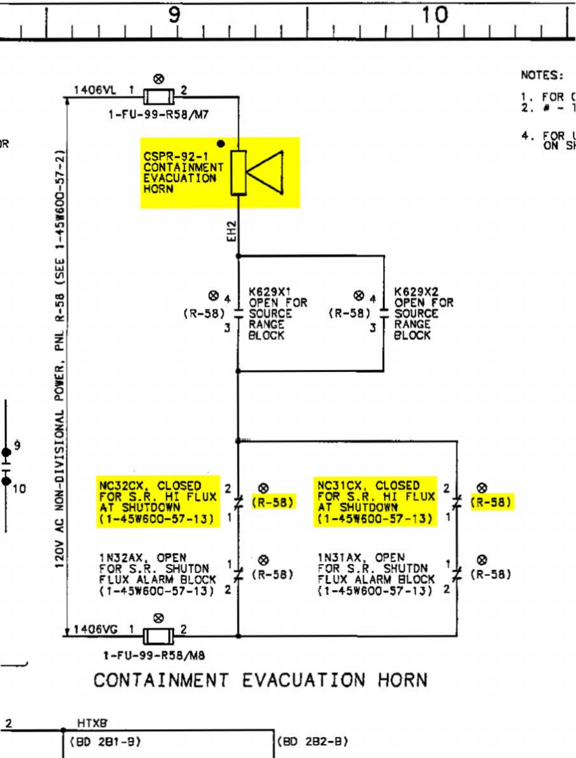

1128VL
1128VG
120 VAC VITAL INST PWR BD1-I (BKR 2B)
TO NIS POWER DRAWER (M13)
J110A
[FL2]
[F3]
5A
[A1]
LOG COUNT & RATE
TO PULSE BUFFERS
[AT1] & [AT2]
J111
1SRA1(+)
1SRA1(-)
[R1]
[K1]
SOURCE RANGE INPUT FROM 1-NM-92-138-D
J215
[A2]
LOG AMPLIFIER AUCTIONEER
INTERMEDIATE RANGE FROM 1-NM-92-138-D
1IRA1(+)
1IRA1(-)
1TSA1(+)
1TSA1(-)
TEST STAT. INPUT TO 1-NM-92-138-D
TEST GENERATOR/ OUTPUT VOLTAGE BUFFER
[A3]
J210
J126B
J126C
J125D
J125E
J125B
[S2C]
[S3A]
[A4]
10112
10111
10201
10202
(W1082H70) SR LEVEL TRIP
(W1082H70) LEVEL TRIP BYPASS
N11102
N11101
N11110
N11111
4B1X1
4B1Y1
HI FLUX AT SD NC31CX (45W600-57-13)
HF AT SD BLOCK 1N31AX (45W600-57-13)
SOURCE/ INTERMEDIATE CH. I TROUBLE 16-4336-C169 CONTRACT #75364A
(R46)
(R49)
(R58)
J113
[A4] BISTABLE
N12504 (+)
N12503 (-)
J125
1NM4B
1M380
[S1E]
CHNL. TEST XA-55-4B 16-4336-C169 CONTRACT# 75364A
J126A
J126D
J125C
J125F
J125A
[S2D]
[S3B]
[K1] ENERGIZED WHEN SR DISABLED
SRA2
J113C
60709A
J113D
CONTACT CLOSURE ON SR BLOCK (45W600-57-14
(1-NI-92-135-D)
J219
CHNL. TEST XA-55-4B 16-4336-C169 CONTRACT# 75364-A
[F4]
5A
[FL2]
J110C
SOURCE RANGE DRAWER 1-NI-92-131-D
[N31] (M13 CABINET I)

LAYS



SOURCE RANGE HI FLUX AT SHUTDOWN SEPARATION RELAYS

6 7 8 9



CONTAINMENT EVACUATION HORN

2.2.5 Sequence of Step Performance (continued)

B. A required task need **NOT** be fully completed before proceeding to the next action.

1. Efficient implementation of the EOI network is important; therefore completion of the task is **NOT** required before continuing with the next step.

2. It is sufficient to begin a task and have assurance it is progressing satisfactorily before continuing with the next step; e.g., when opening a valve, the operator verifies that the red light has come on.

3. Completion of steps treated in this manner should be confirmed after sufficient time has elapsed for the action to complete; e.g., the operator verifies that the valve is full open based upon the red light on and the green light off.

C. Substeps are identified as sequential or non-sequential by the substep designator.

1. Sequential substeps are identified by letters or numbers with a parenthesis. Sequential substeps must be performed in the order listed.

2. Non-sequential substeps are identified by bullets. Bulleted substeps may be performed in any order and they also have an implied “AND” logic (i.e., all listed steps must be performed) UNLESS an “OR” conjunction is specifically used, or an implied “OR” logic is stated in the high level action (e.g., “PERFORM one or more of the following” or “IF any of the following are met”, etc.).

3. In general, the rules for completing a given task in steps apply to substeps.

EXAMPLE 1 (Sequential Substeps)

12**.** **ENSURE** ERCW operation:
  a. At least four ERCW pumps RUNNING, one on each shutdown board preferred.
  b. D/G ERCW supply valves OPEN

EXAMPLE 2 (Non-Sequential Substeps)

3. **CHECK** S/G press:
  • All S/G press controlled or rising
  • All S/G press greater than 150 psig

Date ___________ **Initials**

5.1 Reactor Disassembly (continued)

NOTE

If the Transfer Canal has been filled and draining is **NOT** required, the following two steps may be marked **N/A**.

[9] **CHECK** SFP gate seal to the Transfer Canal does **NOT** leak excessively (leakage greater than SFP makeup flow capability) when Transfer Canal is drained for Transfer Tube Blind Flange removal. ________

[10] **IF** water is in the Transfer Canal and draining is required, **THEN**

DEWATER Fuel Transfer Canal per SOI-78.01. ________

[11] **INITIATE** Setup of Communications between the following stations: ________

- Main Control Room (MCR) ☐
- SFP-Side Fuel Transfer Control Console ☐
- Rx-Side Fuel Transfer Control Console, if manned. ☐
- Refueling Machine ☐
- Demonstrate that Plant P/A system AUDIBLE in Cntmt. ☐
  OR or
  A means for containment evacuation has been established. ☐

[12] **ENSURE** BOTH cavity drain plugs INSTALLED per MI-68.001. ________

Appendix A
(Page 3 of 8)

EOI Glossary

DETERMINE - To evaluate the status of plant parameters or a system in order to establish whether or NOT an action should be performed. Action steps containing this verb usually end with a substep directing performance of the desired action.

DISPATCH - To send to a particular destination.

DO NOT - Used to emphasize that something should NOT be done or implemented at that specific time.

DUMP - The operator action of opening steam dump valves in order to cooldown the RCS.

EMERGENCY (OR IMMEDIATE) BORATE - To initiate actions to inject a high concentration of boric acid solution into the RCS through the normally recognized emergency boration flow path.

EMERGENCY START - Operator action to depress the “Emergency Start” push button for a diesel generator or otherwise initiate an emergency start.

EMERGENCY STOP - Operator action to depress the “Emergency Stop” push button for a diesel generator.

ENERGIZE - To take actions necessary to supply electrical power to equipment or circuitry.

ENSURE - To observe that an expected characteristic or condition exists and, if necessary, to take actions to make the condition occur. Typically, the expectation comes from some previous automatic or operator action.

EQUALIZE - The process of making two or more variables or parameters the same.

ESTABLISH - To bring about. To take necessary actions to cause a specified set of conditions to exist.

EVALUATE - To determine the significance or worth of something usually by careful appraisal or study. Generally, to formulate a decision regarding a course of action by careful study of applicable conditions. For example, “Evaluate the need for continued containment spray system operation”.

FAULTED - Used to describe a secondary system component, e.g., steam generator, with a feedwater line or a steam line break. This term is initial letter capped when used in this manner.

GO TO - Transition point during procedure performance. The transition may be to another step or another instruction.

Appendix A
(Page 4 of 8)

EOI Glossary

IDENTIFY - To determine which of several similar components is applicable to a specified condition as outlined by the instruction.

IMMEDIATE (OR EMERGENCY) BORATE - To initiate actions to inject a high concentration of boric acid solution into the RCS through the normally recognized emergency boration flow path.

IMMINENT - About to occur at any moment

IMPLEMENT - To carry out. To initiate and maintain a prescribed course of action(s).

INITIATE - To commence or begin. Generally used to cause the start or beginning of an effort which can **NOT** be completed in a short period of time.

INSPECT - To examine closely in a critical manner, generally used to require a search for a potential problem or error.

INTACT - Describes a steam generator which is neither ruptured nor faulted, the S/G pressure boundary has no breaks. This term is initial letter capped when used in this manner.

INVESTIGATE - To observe or study by close examination in a systematic manner. Generally used to cause a search for problems or information.

ISOLATE - To separate one component from another. Generally used to require the securing of flow to and from a component.

LEAVE - Maintain.

LOCAL OR LOCALLY - An action performed by an operator outside the Main Control Room. A member of the Control Room Team normally requests personnel other than a team member to perform these actions. Feedback from personnel performing local actions should be required by the Control Room Team.

LOWER - To cause a parameter to become smaller in magnitude.

MAINTAIN - To continuously control a given plant parameter within a specified range or other specified limit as required by the instruction.

MANUAL OR MANUALLY - An action performed by the operator in the control room. These actions are normally in lieu of actions which are normally performed automatically.

56.

Given the following timeline:

00:00:00 1-PDI-30-42, CNTMT PRESS is 0 psig and STABLE ⇔.

00:01:00 1-PI-77-2, RCDT PRESSURE is RISING ⇑.

1-LI-77-1, RCDT LEVEL is 5% and RISING ⇑.

00:20:00 1-LI-77-1 is 45% and STABLE ⇔.

0-PIS-77-88, WASTE GAS VENT HEADER PRESSURE is 2.2 psig and STABLE ⇔.

Which ONE of the following describes a cause of the above conditions AND the expected Radioactive Collecting Drain Tank (RCDT) pressure?

The observed trend was caused by flushing the ____(1)____.

Given **ONLY** the events listed in the timeline, at 00:25:00, 1-PI-77-2 will indicate ____(2)____ psig.

A. (1) RCP #3 seal leakoff lines
(2) 0

B. (1) RCP #3 seal leakoff lines
(2) 2.2

C. (1) instrument lines at panel 1-L-171
(2) 0

D. (1) instrument lines at panel 1-L-171
(2) 2.2

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: It is not correct that flushing the instrument lines at panel 1-L-171 would cause the events listed in the timeline. As seen on print 1-47W851-1, the drain headers from panels 1-L-170 and 1-L-171 (and any other instrument panel inside of containment for that matter) are admitted to the Reactor Building Floor and Equipment Drain Sump's Pocket Sump. It is plausible to believe that these would enter the RCDT because both the Pocket Sump and the RCDT are liquid radioactive waste tanks within containment and both are designed to receive Tritiated water.

The final pressure of the RCDT would not be 0 psig. It would be 0 psig if it were vented to the containment atmosphere (such as is the RBFES). The RCDT is vented to the waste gas header and as such would remain at its pressure (nominally 2.2 psig).

B. Incorrect: As mentioned, it is not correct and yet plausible that the instrument line flush would cause the RCDT parameters to increase.

It is correct that the pressure of the RCDT would stabilize at 2.2 psig. The RCDT is piped to the waste gas vent header. The vent header leads to the suction of the waste gas compressors which act to maintain their suction and thus the waste gas header within a pressure band of 2.0 to 3.5 psig.
Therefore, sending water inventory to the RCDT will initially cause RCDT pressure to rise and then cause the waste gas vent header to equalize RCDT pressure to its that of its own.

C. Incorrect: Prints 1-47W830-1 and 1-47W809-1 demonstrate that the RCP #3 seal leakoff lines are piped to the 4" outlet piping of the RCDT. Therefore, it is correct that any flushing of #3 seal leakoff lines would admit water to the RCDT.
Again, it is incorrect and yet plausible that the stable final pressure of the RCDT would be 0 psig.

D. Correct: As discussed, it is correct that flushing seal leakoff lines would affect the RCDT and that the final pressure of the RCDT would be that of the waste gas header.

Question Number: 56

Tier: 2 Group: 2

K/A: 002 Reactor Coolant System
K5 Knowledge of the operational implications of the following concepts as they apply to the RCS:
K5.05 Reason for drain tank pressure rise during water inventory operations

Importance Rating: 2.9 3.3

10 CFR Part 55: (CFR: 41.5 / 45.7)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is to understand the implication had on the reactor coolant drain tanks

Technical Reference: WBN-SDD-N3-77A-4001, Gaseous Waste Disposal System
1-47W830-1
1-47W809-1
1-47W851-1

Proposed references to be provided: None

Learning Objective: 3-OT-STG-077A, LIQUID RADWASTE PROCESSING SYSTEM
4. EXPLAIN the physical connections and/or cause-effect relationships
between the Liquid Radwaste System and the following systems:
e. Waste gas vent header
f. PRT
g. RCDT
h. Reactor Coolant System (RCS)

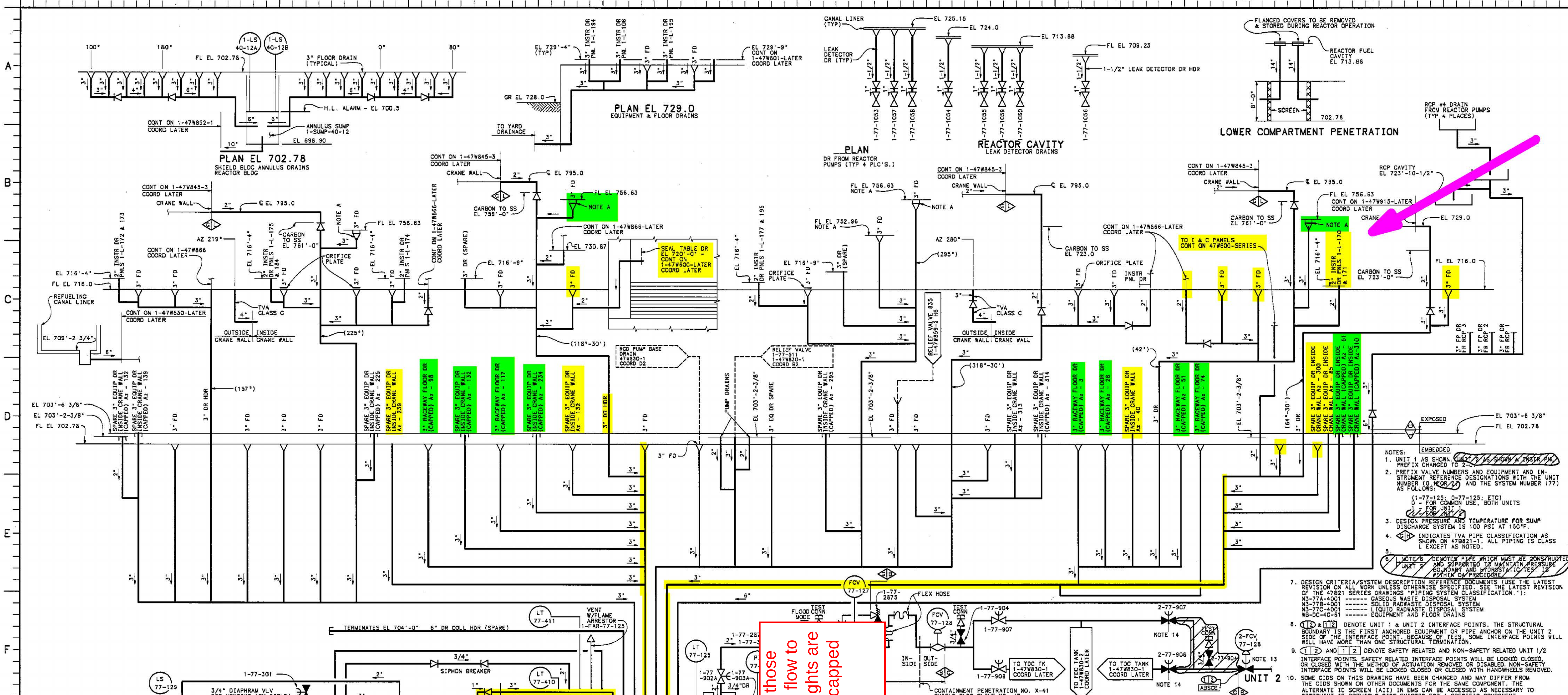
Cognitive Level:

Higher X
Lower ____

Question Source:

New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam



NOTES:

1. UNIT 1 AS SHOWN. ~~UNIT 2 AS SHOWN & INSTR PNL PREFIX CHANGED TO 2-L.~~
2. PREFIX VALVE NUMBERS AND EQUIPMENT AND INSTRUMENT REFERENCE DESIGNATIONS WITH THE UNIT NUMBER (0, 1 ~~OR 2~~) AND THE SYSTEM NUMBER (77) AS FOLLOWS:
   (1-77-125; 0-77-125; ETC)
   0 - FOR COMMON USE, BOTH UNITS
   1 - FOR UNIT 1
   ~~2 - FOR UNIT 2~~
3. DESIGN PRESSURE AND TEMPERATURE FOR SUMP DISCHARGE SYSTEM IS 100 PSI AT 150°F.
4. G|H INDICATES TVA PIPE CLASSIFICATION AS SHOWN ON 47B821-1. ALL PIPING IS CLASS L EXCEPT AS NOTED.
5.
6. ~~NOTE 6 UNIT 2 DENOTES PIPE WHICH MUST BE CONSTRUCTED AND SUPPORTED TO MAINTAIN PRESSURE BOUNDARY AND HYDROSTATIC TEST IS WITHIN QA PROCEDURE~~
7. DESIGN CRITERIA/SYSTEM DESCRIPTION REFERENCE DOCUMENTS (USE THE LATEST REVISION ON ALL WORK UNLESS OTHERWISE SPECIFIED. SEE THE LATEST REVISION OF THE 47B21 SERIES DRAWINGS "PIPING SYSTEM CLASSIFICATION."):
   N3-77A-4001 ------ GASEOUS WASTE DISPOSAL SYSTEM
   N3-77B-4001 ------ SOLID RADWASTE DISPOSAL SYSTEM
   N3-77C-4001 ------ LIQUID RADWASTE DISPOSAL SYSTEM
   WB-DC-40-61 ------ EQUIPMENT AND FLOOR DRAINS
8. (1|2) & [1|2] DENOTE UNIT 1 & UNIT 2 INTERFACE POINTS. THE STRUCTURAL BOUNDARY IS THE FIRST ANCHORED EQUIPMENT OR PIPE ANCHOR ON THE UNIT 2 SIDE OF THE INTERFACE POINT. BECAUSE OF TEES, SOME INTERFACE POINTS WILL WILL HAVE MORE THAN ONE STRUCTURAL TERMINATION.
9. (1|2) AND [1|2] DENOTE SAFETY RELATED AND NON-SAFETY RELATED UNIT 1/2 INTERFACE POINTS. SAFETY RELATED INTERFACE POINTS WILL BE LOCKED CLOSED, OR CLOSED WITH THE METHOD OF ACTUATION REMOVED OR DISABLED. NON-SAFETY INTERFACE POINTS WILL BE LOCKED CLOSED OR CLOSED WITH HANDWHEELS REMOVED.
10. SOME CIDS ON THIS DRAWING HAVE BEEN CHANGED AND MAY DIFFER FROM THE CIDS SHOWN ON OTHER DOCUMENTS FOR THE SAME COMPONENT. THE ALTERNATE ID SCREEN (AII) IN EMS CAN BE ACCESSED AS NECESSARY TO DETERMINE IF PREVIOUS CIDS EXISTED FOR A SPECIFIC COMPONENT.
11. SOCK FILTERS MAY BE ADDED TO FLOOR DRAINS IN THE REACTOR BUILDING AS NECESSARY TO PREVENT DEBRIS AND SILT FROM ENTERING THE SYSTEM. THE SOCK FILTERS SHALL BE MONITORED AND REPLACED AS NECESSARY TO PREVENT RADIOACTIVE HOT SPOT, CRUD TRAPS, DAMAGE AND BLOCKAGE. THE FILTERS IN THE REACTOR BUILDING SHALL ONLY BE USED DURING AN OUTAGE AND BE REMOVED PRIOR TO RETURNING THE PLANT BACK TO SERVICE.
12. ABSCE DENOTES ITEM AND ASSOCIATED COMPONENTS ARE REQUIRED TO REMAIN IN DESIGN CONFIGURATION AS SHOWN TO PROTECT THE ABSCE PRESSURE BOUNDARY. ALL REQUIRED COMPONENTS, INCLUDING SUPPORTS REQUIRED FOR SEISMIC QUALIFICATION, ARE LABELED IN THE FIELD. ABSCE PRESSURE BOUNDARY CONFIGURATIONS MAY INCLUDE COMPONENTS ON THE ANNULUS AND AUXILIARY BUILDING SIDE OF THE SHIELD WALL. THE SUPPORTS REQUIRED TO MAINTAIN ABSCE BOUNDARY SEISMIC QUALIFICATION ARE IDENTIFIED ON DCA 53387-666.
13. VALVE FAILED CLOSED; AIR SUPPLY ISOLATED TO PREVENT OPERATION. VALVE TO REMAIN IN CLOSED POSITION FOR ABSCE BOUNDARY RETENTION.
14. UNIT INTERFACE SEPARATION BOUNDARIES ARE CONTROLLED BY INSTRUCTION TI-12.08, CONTROL OF UNIT INTERFACES.

NOTE A:
DRAINS ARE PERMANENTLY PLUGGED.

NOTE B:
HYDROSTATIC TEST PRESURE FOR CLASS C PIPING IS 150 PSI. HYDROSTATIC TEST PRESSURE FOR CLASS B PIPING IS 125 PSI PRESSURE FOR CLASS B PIPING IS 125 PSI AND NOTED ON DRAWING. CONST SHALL DETERMINE IF THE COMPONENTS IN THE SYSTEM ARE LIMITING COMPONENTS TO THE HYDROSTATIC TEST. ASME CODE CASES INVOLVING HYDROSTATIC TESTING MAY BE USED IF EN DES APPROVES THEIR APPLICATION. HYDROSTATIC TEST PRESSURE DATA IS HISTORICAL INFORMATION AND NO LONGER MAINTAINED AS DESIGN OUTPUT.

NOTE C:
PNEUMATIC TEST MAY PERFORMED IN ACCORDANCE WITH ASME SECTION III AND ANSI B31.1 IN LIEU OF HYDROSTATIC TEST.

TVA REFERENCE DRAWINGS	
NO.	TITLE
47W476	CONTAINMENT DRAINS & EMBEDDED PIPING - REACTOR BLDG
47W482	SUMP PUMP PIPING - REACTOR BLDG
47W830	FLOW DIAGRAM - WASTE DISPOSAL SYSTEM
47W852	FLOW DIAGRAM - STATION DRAINAGE AUX BLDG
47W801	FLOW DIAGRAM - MAIN STEAM & REHEAT STEAM
17W300	YARD PIPING
47W915	HEATING & VENTILATING
47W851-101	STRESS ANALYSIS PROBLEM BOUNDARY FLOOR EQUIPMENT DRAINS

THIS CONFIGURATION CONTROL DRAWING SUPERSEDES UNIT 1 AS-CONSTRUCTED DRAWING 47W851-1 REVISION S.

ISSUED BY: J.EDWARD GIBB/WLE

REV	CHANGE REF	PREPARER	CHECKER	APPROVED	DATE
30	53387	RWE	JEW	JKa	DTS
REVISED PER DCA 53387-665-0.					

SCALE: NTS EXCEPT AS NOTED

PROJECT FACILITY
POWERHOUSE
UNITS 1 & 2

TITLE
MECHANICAL FLOW DIAGRAM FLOOR & EQUIPMENT DRAINS

1 WATTS BAR NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY Q

DESIGN: DRAFTER G.W.JOHNSON; CHECKER D.L.COOPER; DESIGNER J.T.COLEMAN; REVIEWER E.E.SANDER
INITIAL ISSUE: RO ISSUE PER WBEP 5.17 & RIMS B26 '90 0829 378
ENGINEERING APPROVAL: 1 P.S.KIRKHAM; 2 P.W.BURNETT; 3 L.W.BOYD
DATE 9-19-90 85 M 1-47W851-1 R30

CAD MAINTAINED DRAWING

CONFIGURATION CONTROL DRAWING

3.2.1 Waste Gas Compressors (continued)

M. Pipe Joints

All GWDS piping joints are welded except where flanged connections are necessary for maintenance (e.g., there are 16 flanged connections per compressor package).

3.2.2 Decay Tank Component Description

The nine GWDS gas decay tanks are vertical, cylindrical carbon steel tanks with a volume of 600 ft^3 each. The dimensions of the tanks are approximately 8 feet 8 inches outside diameter by approximately 10 feet vertical height. They are fitted with 3 nozzles, a 1-inch inlet, a 1-inch outlet, and a 3/4-inch sample line. Each tank has 2 manholes, both 16 inches in diameter: One is located on the top of the tank and the other on the side. Tanks have a design pressure of 150 lb/in^2g (Table 9-5), a design temperature of 180°F, a normal operating pressure of 0-100 lb/in^2g, and a normal operating temperature of 40-140°F. They have a service life of 40 years and are TVA Class D.

3.2.3 Waste Gas Filter Component Description

A cleanup system combining a charcoal absorber and HEPA filters removes particulates and gaseous iodine compounds from the gas discharged. The system is designed to handle a maximum flow of 100 cfm at a maximum pressure and temperature of 5.3 lb/in^2g and 180°F. The maximum pressure drop across a clean element is 6 inches H_2O at a flow rate of 100 SCFM. Collection efficiency of the particulate filter exceeds 99 percent. The iodine cartridge has a minimum collection efficiency for methyl iodine of 70 percent and 95 percent collection efficiency for elemental iodine.

3.3 Instrumentation and Controls

Instrument setpoints and functions are provided in reference 7.54.

3.3.1 Vent Header Pressure System (PCV 77-89)

This system indicates and controls the vent header pressure. Pressure indication is provided on the Waste Processing System (WPS) panel. When the backup waste gas compressor is in automatic mode, the controller will start the compressor on high pressure and actuate a high pressure alarm. The backup compressor is stopped automatically when the vent header pressure drops back to normal. In manual mode, both compressors may be turned on or off regardless of the vent header pressure. The vent header high pressure alarm is actuated regardless of the mode of operation of the compressor. A low pressure alarm is also provided with indication and alarms on the WPS panel.

3.3.2 Compressor Control Instrumentation

The WG compressor controls are a part of the compressor packages. These consist of pressure, flow, and level controls. Refer to the simplified flow diagram (Figure 8.4).

Under normal conditions one compressor package is in automatic with the second acting as backup. On high inlet pressure (2.0 lb/in^2g) the starter switch of the compressor in automatic is energized. Auxiliary contacts on this switch provide operating power for solenoid valves and pressure switches in the waste gas compressor package.

The WGC will start if its suction pressure is 2.0 psig.

9.2 TABLE 9-2 - Gaseous Waste Systems Components

Gaseous Waste Systems Components

Waste Gas Decay Tanks

Number	9
Volume, Each, ft^3	600
Design Pressure, lb/in^2g	150
Design Temperature, °F	180
Normal Operating Pressure, lb/in^2g	0-110
Normal Operating Temperature, °F	50-140
Material of Construction	Carbon Steel Vertical Cylinder
Design Type Code/Class	ASME III/3
ANS Safety Class	D

Waste Gas Compressors

Number	2
Type	Water Sealed Centrifugal
Design Flow Rate, N_2 (at 140°F, 2 lb/in^2g)cfm	40
Design Pressure, lb/in^2g	150
Design Temperature, °F	180
Normal Operating Pressure, lb/in^2g	
Suction	2.0-3.5
Discharge	0-100
Normal Operating Temperature, °F	70-130
Design Code/Class	[illegible]
ANS Safety Class	[illegible]

The waste gas compressor starts at 2.0psig suction pressure. It then runs and a common inlet valve (to both of the WGCs) modulates to maintain suction pressure around a setpoint of 3.0 psig. Therefore, the action of both of these components maintains the Waste Gas Header between 2.0 and 3.5 psig.

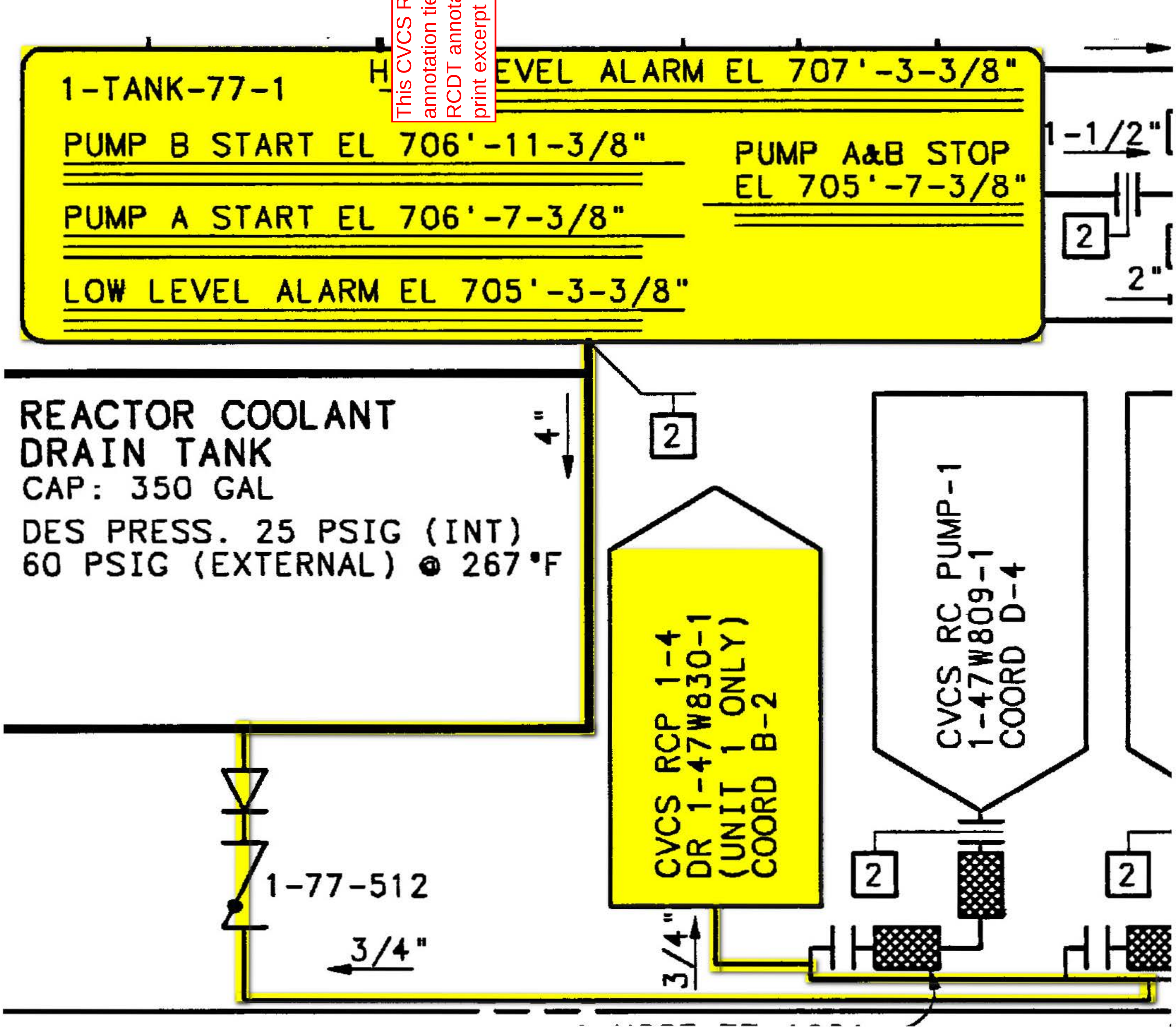
Waste Gas Filter

Number	[illegible]
Maximum Flow (at 5.3 lb/in^2g, 180°F) SC[illegible]	[illegible]
Material	SS
Maximum Clean Pressure Drop (at 100 scfm) in H_2O	6
Type	Charcoal/HEPA Combination
Design Code	ASME VIII
ANS Safety Class	G

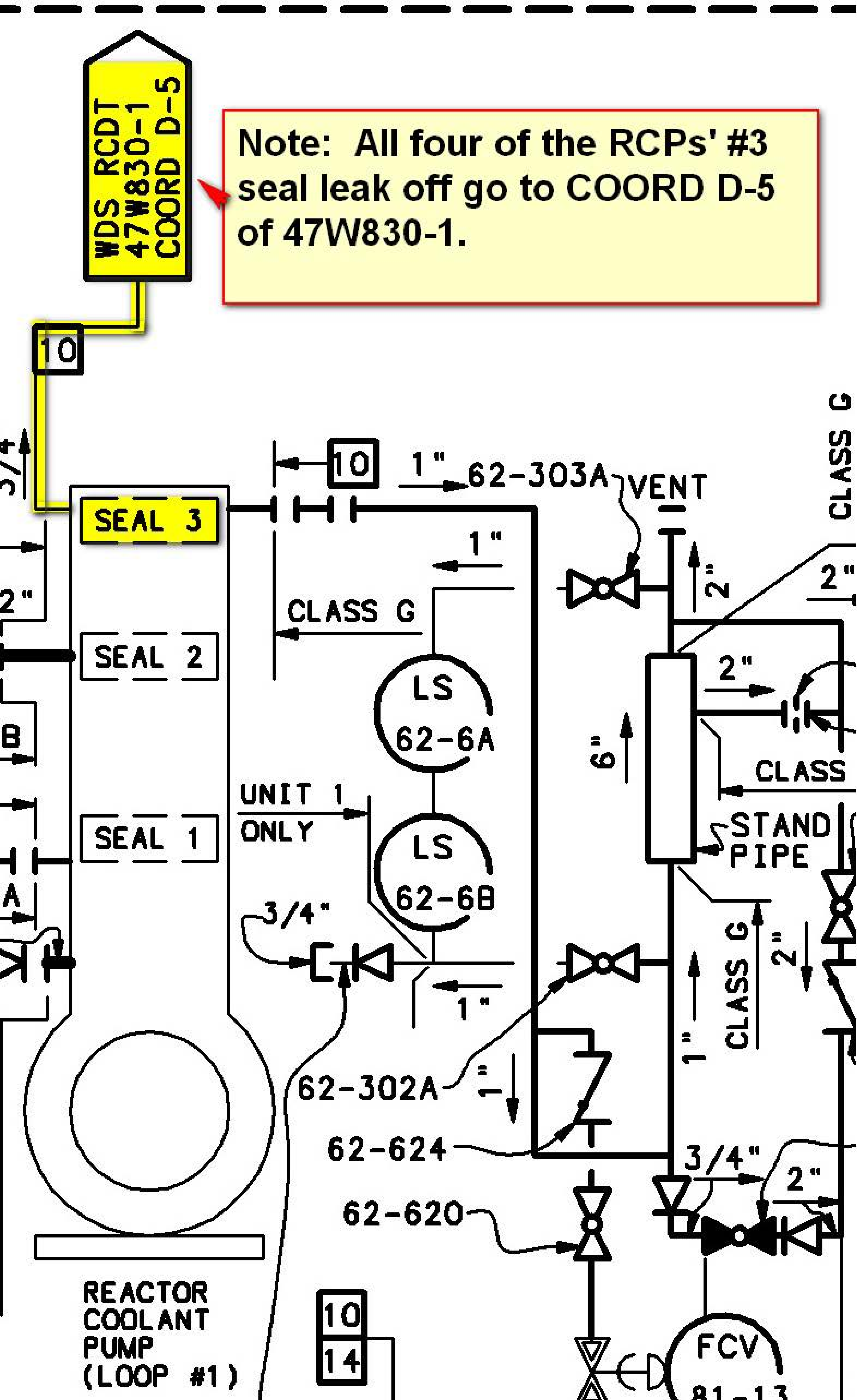
Sequential Automatic Gas Analyzer

Number	1 (shared)
Oxygen	Electrochemical of the Polarographic type 0-20% O_2
Hydrogen	by Thermal Conductivity 0-100% H_2
Automatic stepping switch*	8 steps
Recorded Readout	8 points
Temperature	120°F

*See Table 9-4.



This CVCS
annotation
RCDT
print excerpt
1-TANK-77-1
EVEL ALARM EL 707'-3-3/8"
PUMP B START EL 706'-11-3/8"
PUMP A START EL 706'-7-3/8"
LOW LEVEL ALARM EL 705'-3-3/8"
PUMP A&B STOP
EL 705'-7-3/8"
1-1/2"
2
2"
REACTOR COOLANT
DRAIN TANK
CAP: 350 GAL
DES PRESS. 25 PSIG (INT)
60 PSIG (EXTERNAL) @ 267°F
4"
2
CVCS RCP 1-4
DR 1-47W830-1
(UNIT 1 ONLY)
COORD B-2
CVCS RC PUMP-1
1-47W809-1
COORD D-4
2
2
1-77-512
3/4"
3/4"



Note: All four of the RCPs' #3 seal leak off go to COORD D-5 of 47W830-1.

57.

Which ONE of the following describes the bus power supply to the 1A-A Centrifugal Charging Pump?

A. 1A-A 480V SDBD

B. 1A 480V UNIT BD

C. 1A-A 6.9kV SDBD

D. 1A 6.9kV UNIT BD

CORRECT ANSWER: C

DISTRACTOR ANALYSIS:

A. Incorrect: The power supply to the 1A-A CCP is the 6.9kV SDBD 1A-A and not the 480V SDBD 1A-A. It is plausible to believe that it would derive power from a 480V board because there are plants whose large motors are 480V powered.

B. Incorrect: The CCPs were originally installed as the "High Head Safety Injection Pumps." WBNP's original design involved the continuous use of a positive displacement charging pump which would serve the non-safety related purpose of chemical and volume control. This charging pump did not possess a safety related qualification. As use of the positive displacement pump was phased out, the CCPs assumed its non-safety related role of chemical and volume control. The relation of this non-safety related role to a power supply generates the plausibility of selecting the Unit Boards as an answer.

C. Correct: As seen on print 1-45W724-1, the power supply to the 1A-A CCP is the 1A-A 6.9kV SDBD.

D. Incorrect: Again, while it is correct that a 6.9kV board powers the 1A-A CCP, it is not correct that a unit board is such supply.

Question Number: 57

Tier: 2 Group: 2

K/A: 011 Pressurizer Level Control System
K2 Knowledge of bus power supplies to the following:
K2.01 Charging pumps

Importance Rating: 3.1 3.2

10 CFR Part 55: (CFR: 41.7)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to identify the bus power supply for the 1A-A Charging Pump.

Technical Reference: 1-45W724-1

Proposed references to be provided: None

Learning Objective: 3-OT-STG-062A, CHEMICAL AND VOLUME CONTROL SYSTEM
5. LIST the bus power supplies to the following CVCS components:
a. Centrifugal Charging pumps (CCPs)

Cognitive Level:

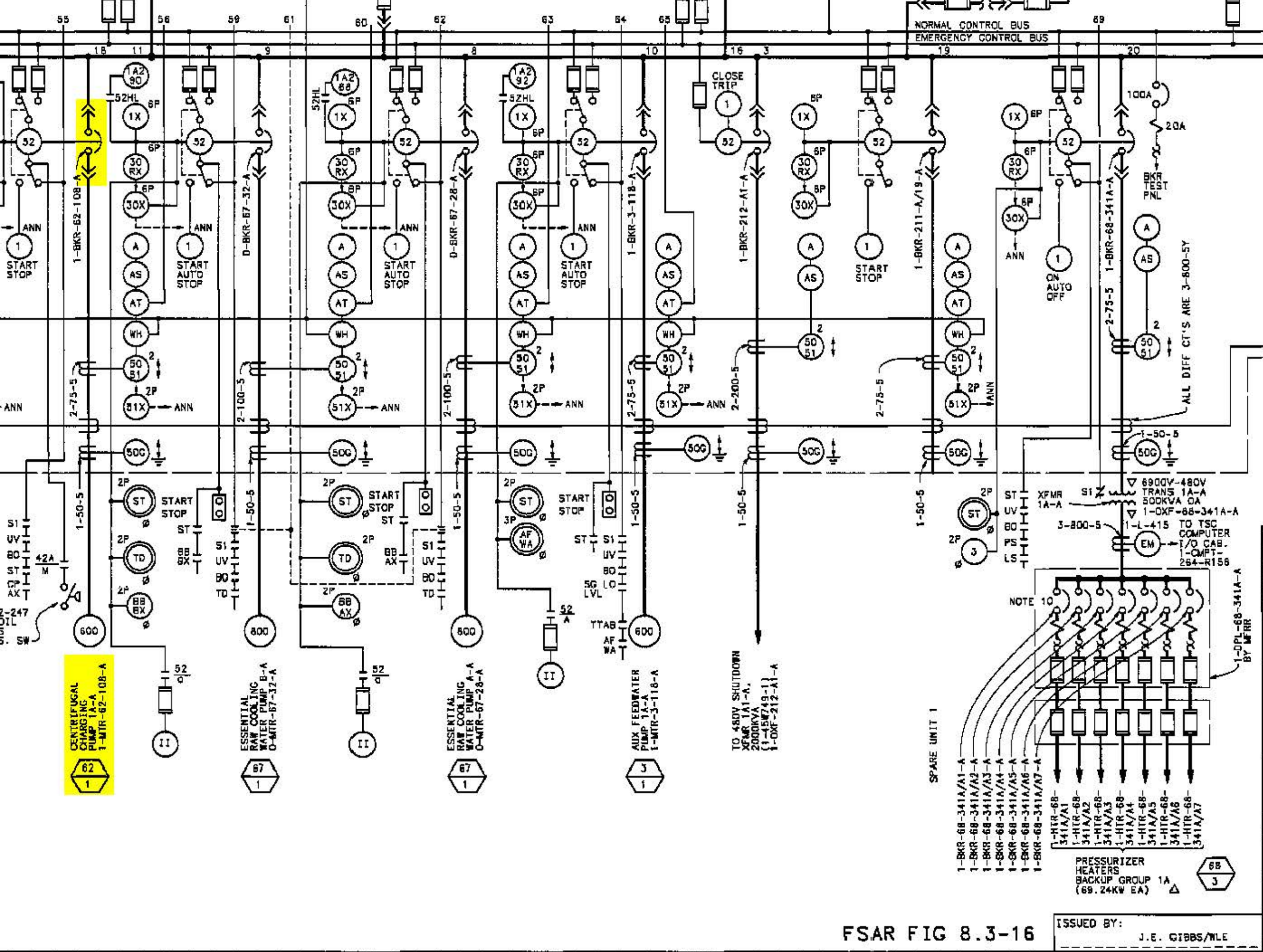
Higher	
Lower	X

Question Source:

New	
Modified Bank	
Bank	X

Question History: WBN Bank question used for ILT periodic exams.

Comments:



EXTERNAL CONNECTION DIAGRAMS ----- 45W1724 SERIES
INTERNAL CONNECTIONS DIAGRAMS:
SEE TVA CONTRACT (74C2-84376)

SYMBOLS:
ø ----- EQUIPMENT LOCATED ON 6.9KV SD BD LOGIC RELAY PANEL 1A-A
* ----- EQUIPMENT LOCATED ON A STATUS INDICATING BOARD IN THE AUXILIARY CONTROL ROOM.
Δ ----- EQUIPMENT NOT SAFETY RELATED
S --- SYSTEM NUMBER OF EQUIPMENT
D --- SHEET NUMBER OF SCHEMATIC IN SERIES 1-45W760

THIS CONFIGURATION CONTROL DRAWING SUPERSEDES UNIT 1 AS-CONSTRUCTED DRAWING 45W724-1 REVISION FF.

REV	CHANGE REF	PREPARER	CHECKER	APPROVED	DATE
28	60853	RWE	GJO	JKQ	DTS
REVISED PER DCA 60853-77-1.					

SCALE: NTS EXCEPT AS NOTED

PROJECT FACILITY
AUXILIARY BUILDING
UNITS 1 & 2

TITLE
WIRING DIAGRAMS
6900V SHUTDOWN BOARD 1A-A
SINGLE LINE

1 WATTS BAR NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY Q

DESIGN		INITIAL ISSUE	ENGINEERING APPROVAL
DRAFTER D.H. WANG	CHECKER D.L. COOPER	RO ISSUE PER	1 C.H. SUDDUTH
DESIGNER M.L. CHAPMAN	REVIEWER S.M. GALLAGHER	WBEP 5.17 & RIMS B26 '90 0829 377	2 S.J. HUNTER
			3 C.C. LYKE/MCB

ISSUED BY: J.E. GIBBS/MLE

DATE 9-21-90 | 85 | E | 1-45W724-1 | R28

FSAR FIG 8.3-16

PRINTS REQ'D R ____ PRINT SIZE F
PRINT SIZE H

58

Which ONE of the following describes how CERPI calculates Rod Insertion Limits?

The CERPI system calculates the Rod Insertion Limit using ________ as an input.

A. Calorimetric Power

B. Nuclear Instrument Power

C. Turbine Impulse Pressure

D. Delta Temperature Percent Power

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: It is Incorrect that the plant calorimetric provides an input into the rod insertion limit. It is plausible to believe this for the exact reason seen in the analysis for distractor B.

B. Incorrect: As discussed, it is Incorrect that NIS provides an input into the rod insertion limit. It is plausible to believe this because as seen in WNA-CT-00053-WAT: The Rod Insertion Limit (RIL) algorithm calculates the minimum allowable control bank positions as a function of plant power. Therefore, because this distractor provides a measure of plant power the distractor is plausible.

C. Incorrect: It is Incorrect that the turbine impulse pressure provides an input into the rod insertion limit. It is plausible to believe this for the exact reason seen in the analysis for distractor B.

D. Correct: As seen in the CERPI Technical Description, WNA-CT-00053-WAT: The Rod Insertion Limit (RIL) algorithm calculates the minimum allowable control bank positions as a function of plant power to ensure sufficient shutdown reactivity margin. The calculated RIL is compared to the control bank demand positions by two bistables; LO and LO-LO. The MTP [maintenance test panel] receives Delta Temperature Percent Power (%ΔT) from the ICS in every second and it updates PLCs via AF100 bus. %ΔT shall be processed by the following algorithm: RIL LO-LO Setpoint = $K_1 \times \%\Delta T + K_2$.

Question Number: 58

Tier: 2 Group: 2

K/A: 014 Rod Position Indication System
K1 Knowledge of the physical connections and/or cause effect relationships between the RPIS and the following systems:
K2.02 NIS

Importance Rating: 3.0 3.3

10 CFR Part 55: (CFR: 41.3 to 41.9 / 45.7 to 45.8)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the K/A asks for the power input into the RPI system. The question requires that the applicant recall the power input which is used to produce the Rod Insertion Limit.

Technical Reference: CERPI Technical Description, WNA-CT-00053-WAT

Proposed references to be provided: None

Learning Objective: 3-OT-SYS085A, Control Rod Drive System
2. DESCRIBE the design criteria, purpose and/or functions of the Rod Position
Indicating (RPI) System and the major system components listed below:
e. Rod Insertion Limit (RIL, CERPI Software)
5. EXPLAIN the physical connections and/or cause-effect relationships between
the Rod Control system and the following systems:
a. Rod Position Indicators (RPIs)
b. Excore Nuclear Instrumentation (NIS)

Cognitive Level:

Higher ___
Lower X

Question Source:

New X
Modified Bank ___
Bank ___

Question History: New question for the 2015-301 NRC RO Exam

Comments:

4.2.6.3.5 If any one of the 8 bistable outputs (one for each bank) is active, the output of the Rod to Bank Deviation Alarm relay will be de-energized to alarm.

4.2.6.3.6 If the quality of any of the Bank Demand Position transmitted by the ICS has Bad quality, then the algorithm uses the value of the last Good quality value of that Bank Demand.

4.2.6.3.7 Rod to Bank Deviation Alarm can be disabled by using the access restricted (by utilizing Function Enable Keylock Switch) Alarm Selection display on the MTP.

4.2.7 Rod Insertion Limit Alarms

4.2.7.1 The Rod Insertion Limit (RIL) algorithm calculates the minimum allowable control bank positions as a function of plant power to ensure sufficient shutdown reactivity margin. The calculated RIL is compared to the control bank demand position by two bistables; LO and LO-LO.

4.2.7.2 The MTP receives Delta Temperature Percent Power (%ΔT) from the ICS in every second and it updates PLCs via AF100 bus.

4.2.7.3 %ΔT shall be processed by the following algorithm:

RIL LO-LO Setpoint = K_1 x %ΔT + K_2

where K1 and K2 are as follows per Reference 9:

Control Bank / Coefficient	A	B	C	D
K_1 (steps/%ΔT)	0	2.30	2.29	2.29
K_2	211	214	86	-42.45

Table 4.2.7.2-1 – RIL LO-LO Setpoint Algorithm Constants

4.2.7.4 The RIL LO-LO Setpoint is limited to a maximum value of 211 steps per Reference 10.

4.2.7.5 RIL LO Setpoint is 10 steps greater than the RIL LO-LO Setpoint and it is limited to a maximum value of 221 steps and minimum value of 0 step.

4.2.7.6 Control bank demand position is an input to RIL LO and RIL LO-LO bistables which trips when the demand position goes below RIL LO Setpoint and RIL LO-LO Setpoint values, respectively.

4.2.7.7 The hysteresis value for both RIL LO and RIL LO-LO bistables is 5 steps.

4.2.7.8 K_1 and K_2 coefficients, RIL maximum and minimum limit values, bistable setpoint and hysteresis values are adjustable via the PLC programming tool.

4.2.7.9 If the quality of any of the Control Bank Demand Position transmitted by the ICS has Bad quality, then the algorithm uses the value of the last Good quality value of that Control Bank Demand.

4.2.7.10 If the quality the Delta Temperature Percent Power transmitted by the ICS has Bad quality, then the algorithm uses the value of the last Good quality Delta Temperature Percent Power.

4.2.7.11 RIL LO Alarm and RIL LOLO Alarm outputs have re-flash capability. Thus, while an alarm condition exists, if a new alarm condition occurs then RIL LO Alarm and RIL LOLO Alarm outputs are energized for 2 Seconds and de-energized (to alarm) again to indicate the new alarm condition.

4.2.8 Rod Speed Indicator

4.2.8.1 Rod Speed Indicator algorithm calculates the rod speed during a rod motion based on the analog rod speed indication signal. Rod speed shall be between 8 Steps/minute to 72 Steps/minute during a rod motion and 0 Steps/minute otherwise.

4.2.8.2 Analog Rod Speed signal is processed by the following algorithm:

Rod Speed during a rod movement = - (Analog_Speed_Signal + 1.21) / 0.115

4.2.9 Rod Demand from Passive Summer Indicator

4.2.9.1 Rod demand from passive summer analog signal is received, scaled, and displayed in order to help the Operator to know the magnitude of rod movement to be expected when the rod control system is switched from Manual Rod Control to Automatic Rod Control.

59.

Which ONE of the following describes the In-core Monitoring System?

The T/C Map Display of 1-XI-68-100, RVLIS-ICCM PLASMA DISPLAY will show _________ for a **BAD** thermocouple input?

A. 0000

B. 9999

C. BBBB

D. XXXXB

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: As seen in the equipment description for the Inadequate Core Cooling Monitor-86, WBN-VTD-W120-3020, 0000 is displayed for the MAX value for a quadrant with no good thermocouples. It is plausible to believe that this would be displayed as a thermocouple does indeed fail low. One could believe that this failure would be displayed on the ICCM.

B. Incorrect: As seen in the equipment description for the Inadequate Core Cooling Monitor-86, WBN-VTD-W120-3020, 9999 is displayed for the MIN value for a quadrant with no good thermocouples. It is plausible to believe that this would be displayed if one believe that a thermocouple failed high and that such would be displayed on the ICCM.

C. Incorrect: It is not correct that "BBBB" will be displayed in the place of a thermocouple reading. Rather, a reverse video "B" will. It is plausible to believe this because the Distributed Control System (installed on both the primary and secondary plants) displays "BAD" for a bad input.

D. Correct: As seen in the equipment description for the Inadequate Core Cooling Monitor-86, WBN-VTD-W120-3020, "if a calculated analog input point quality code of BAD…is assigned to any of these values, then a reverse video B…will be displayed next to the new value of X's, for a BAD quality code." As seen in the vendor manual for the ICCM, the T/C Map Display page displays temperatures to four digits. Therefore, the correct mode of showing a BAD thermocouple input would be to display XXXXB.

Question Number: 59

Tier: 2 Group: 2

K/A: 017 In-Core Temperature Monitor System
K6 Knowledge of the effect of a loss or malfunction of the following ITM system components:
K6.01 Sensors and detectors

Importance Rating: 2.7 3.0

10 CFR Part 55: (CFR: 41.7 / 45.7)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to understand the effect which a failed Thermocouple has on the ICCM display.

Technical Reference: Equipment description for the Inadequate Core Cooling Monitor-86, WBN-VTD-W120-3020

Proposed references to be provided: None

Learning Objective: 3-OT-STG-068F, RVLIS AND ICCM
10.
Given specific plant conditions, ANALYZE the effect that a loss or malfunction of the following will have on RVLIS:
b. RTD failure

Cognitive Level:

Higher ____
Lower X

Question Source:

New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

WBN-VTD-W120-3020

```
VESSEL LEVEL DIAGNOSTIC PAGE            TIME 12:46:39          MESSAGE 100
LT-68-369      -5.334  30  00   UPPER RANGE     73.025 34  00
LT-68-368       0.000  60  80   LOWER RANGE     70.000 64  00
LT-68-367      18.424  30  00   DYNAMIC HEAD    85.229 34  00
TE-68-373      79.426  30  00
TE-68-376      80.393  30  00   RVLIS LEVEL     73.025 34  00
TE-68-377      80.435  30  00
TE-68-378      80.250  30  00
TE-68-379      79.401  30  00   PEN 1:   RVLIS LEVEL        73.025
1PY-406A     2504.473  30  00   PEN 2:   RVLIS LEVEL        73.025
                                PEN 3:   RVLIS LEVEL        73.025
THOT          600.485  34  00
1TY-413T      600.485  30  00                       RCP-1       00  00
1TY-423T      423.224  30  00                       RCP-2       00  00
1TY-411C       93.072  30  00                       RCP-3       00  00
                                                    RCP-4       00  00
                                MALFUNCTION 00      XIS-68-387  00  00
DIAGNOSTIC INFORMATION                              XIS-68-388  00  00
                                                    XIS-68-389  00  00
```

WBN REV 0
WBN PAGE 195

Figure 5-10 Typical RCS Level Diagnostics Display Page

5.10.4.1 T/C Map Display Pages

The core map display pages (figures 5-11 and 5-12) give a train-oriented view of the core exit thermocouple layouts (CETC). These graphic examples are presented only to familiarize the operator with the mechanics of the ICCM-86 System, and do not necessarily represent exact replicas of the display pages that actually appear on the screen. The rounded whole number value of each CETC is shown in its assigned core location.

If a CETC value becomes greater than or equal to some critical setpoint, then it will be displayed in reverse video.

If a calculated analog input point quality code of BAD or SUSPECT (without a manually entered value) is assigned to any of these values, then a reverse video "B" or "S," respectively, will be displayed next to the new value of X's, for a "BAD" quality code, or D's, for a "SUSPECT" quality code.

If a quality code of SUSPECT (with a manually entered value) is assigned to any of these values, then a reverse video "S" will be displayed next to the current numeric value.

A calculated analog input point quality code of POOR assigned to any of these values will cause a reverse video "P" to be displayed next to the current numeric value.

5.10.4.2 Incore T/C Quadrant Summary Display Pages

WBN REV 0
WBN PAGE 197

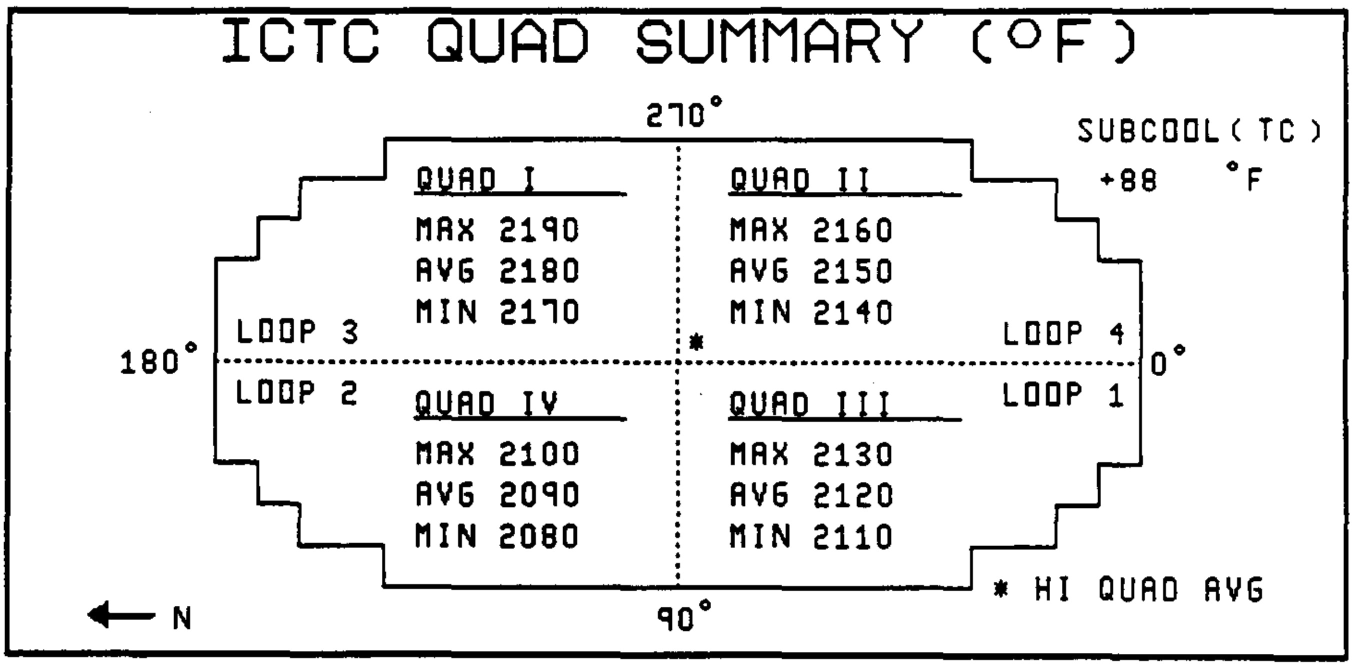


Figure 5-13 Typical ICTC Quadrant Summary Display Page, Unit 1

The incore T/C quadrant summary pages are shown in figures 5-13 and 5-14. These graphic examples are presented only to familiarize the operator with the mechanics of the ICCM-86 System, and do not necessarily represent exact replicas of the display pages that actually appear on the screen. In a train-oriented view of the reactor vessel, the T/C quadrant summary page displays the rounded whole number values for each quadrant's minimum and maximum thermocouples. The average T/C temperature will also be displayed for each of the four quadrants. An asterisk will

WBN - VTD - W120 - 3020

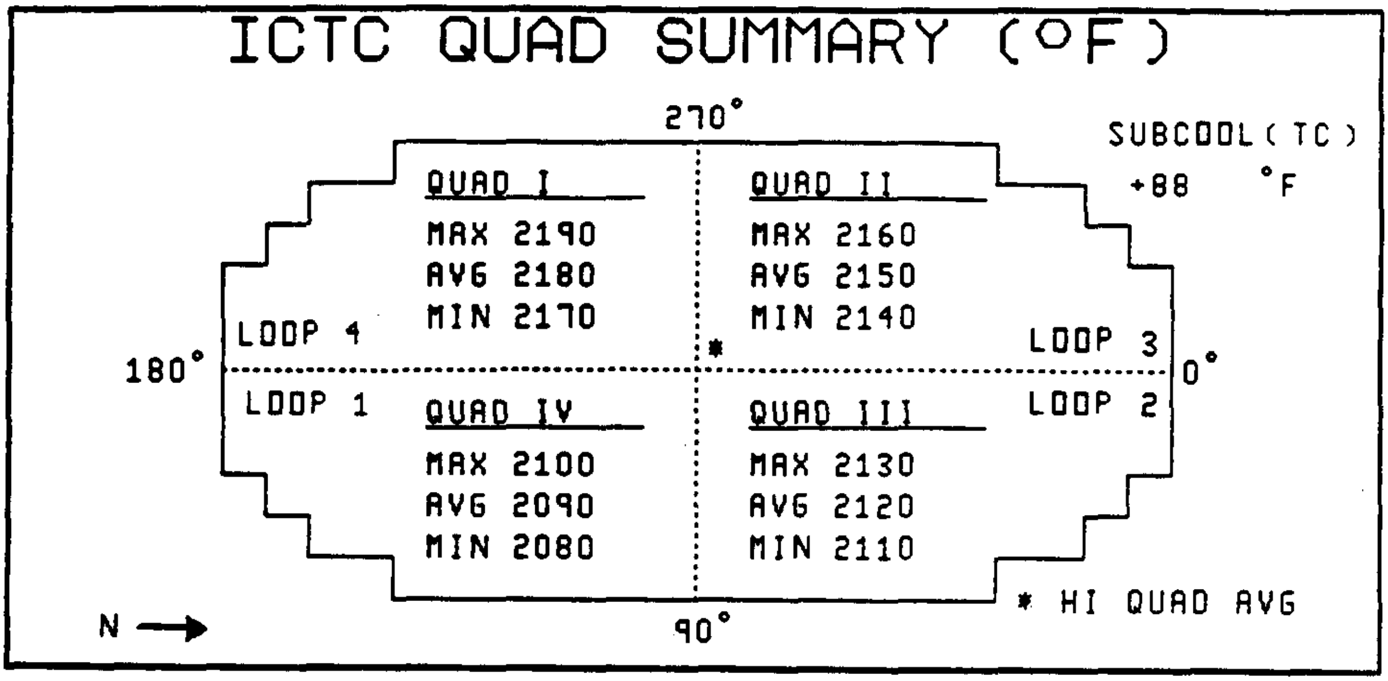


Figure 5-14 Typical ICTC Quadrant Summary Display Page, Unit 2

appear in the quadrant with the high auctioneered average temperature and the note, * HI QUAD AVG, will appear in the lower right-hand corner of the page. The corresponding loop numbers will appear in each quadrant's displayed section.

In addition to the T/C values being displayed, the rounded whole number value for SUBCOOL (T/C or TH) will also appear in the upper right-hand corner of the page. If the subcooling value becomes less than an entered setpoint, its respective value will be displayed in reverse video.

If a calculated analog input point quality code of BAD is assigned to any average quadrant value, then a reverse video "B" will be displayed next to the new value of X's for a "BAD" quality code.

If a quality code of POOR is assigned to any average quadrant value, then a reverse video "P" will be displayed next to the current numeric value.

If there are no "GOOD" thermocouples in any one quadrant, then the MAX and MIN values for that quadrant will be 0000 and 9999 respectively.

Notice the reversal: low number to MAX and high number to MIN

WBN - VTD - W120 - 3020

2300°F) from the field wiring. Each signal is then converted to a -5 to 5-volts input voltage range. An open thermocouple will cause the output signal to go offscale low. The thermocouple input board is described in section 15 of this volume (volume I).

1.7 REMOTE DISPLAY MODULE

The train A and train B modular remote displays are units containing an alphanumeric graphic plasma display, pushbuttons which enable page selection, printed circuit boards, and other electronics that process data and format the information for display.

Detailed descriptions of the display modules and graphic displays can be found in section 5 of this volume (volume I).

1.8 MAINTENANCE TERMINAL

The Maintenance Terminal is a portable personal computer which comes packaged in a conveniently sized suitcase along with an AC adaptor, interconnecting cable for connection to the ICCM cabinet, and an XTALK communications package on a 3-1/2 inch diskette. Verifying or modifying scaling values and other constants is accomplished via the Maintenance Terminal and the transaction results are stored on disk. The Maintenance Terminal is disconnected during normal System operation.

WBN REV 0
WBN PAGE 52

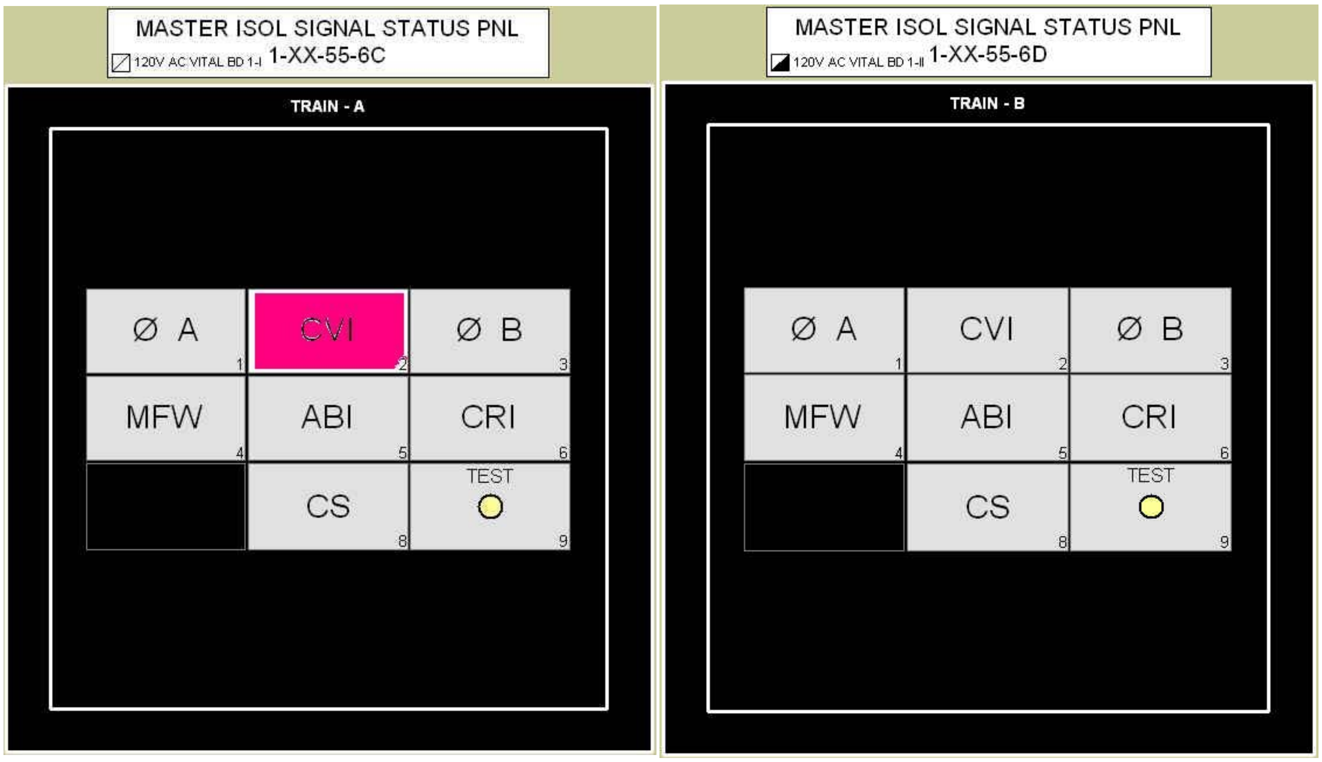
60.

Given the following conditions:

- Unit 1 is at 100%.
- The B train CNTMT Purge is in service to Lower CNTMT.

Subsequently:

- The following indications are noted:



Which ONE of the following describes the status of the CNTMT Purge Fans and the Purge Damper position indications?

The running supply and exhaust CNTMT Purge Fans will ____(1)____.

Purge Damper position indication ____(2)____ available on panel **1-M-6**.

A. (1) STOP
   (2) IS

B. (1) STOP
   (2) IS **NOT**

C. (1) continue running
   (2) IS

D. (1) continue running
   (2) IS **NOT**

<u>CORRECT ANSWER:</u> <u>A</u>

<u>DISTRACTOR ANALYSIS:</u>

A. Correct: Print 1-47W611-30-1 demonstrates the logic for the operation of the containment purge supply and exhaust fans. One may observe on this print that the supply and exhaust operate as a pair (i.e. the same start or stop signal will operate both the supply and exhaust fan of a given train). This print shows that either a Containment Vent Isolation Train A or a CVI Train B will cause both of the trains of containment purge fans to receive a stop signal. Therefore, it is correct that the A train CVI will cause the B train containment purge supply and exhaust fans to stop. One may note that the dampers immediately adjacent to the supply and exhaust fans receive the same signal that the fan does. This logic does not hold true for the remainder of the containment purge dampers.

It is correct that containment purge damper position indication is available on 1-M-6. The containment isolation status panels on 1-M-6 (1-XX-55-6E and 6F) show those dampers and solenoid valves which are isolated on a CVI.

B. Incorrect: It is true that both trains of containment purge are stopped with the actuation of either train of CVI. It is not true that position indication only available on 1-M-9. 1-M-9 is the MCR panel which possesses the controls for containment purge and thus the indications. However, again, 1-M-6 does possess the status panels which do reflect the damper positions for containment purge.

C. Incorrect: It is not correct that the B train purge fans will continue to run. It is plausible to believe this because if the logic attributed to the majority of the containment purge dampers were attributed to the fans then this would be the case. The containment purge dampers which are located either in the annulus or inside of containment are actuated only by the train of CVI to which they are affiliated. This is seen on print 1-47W611-30-1. It is correct that containment purge damper position indication is available on 1-M-6.

D. Incorrect: It is not correct but plausible that the B train purge fans will continue to run. It is not true but plausible that position indication only available on 1-M-9.

Question Number: 60

Tier: 2 Group: 2

K/A: 029 Containment Purge System (CPS)
A3 Ability to monitor automatic operation of the Containment Purge System including:
A3.01 CPS isolation

Importance Rating: 3.8 4.0

10 CFR Part 55: (CFR: 41.7 / 45.5)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to demonstrate the ability to anticipate the response of the containment purge system to a single train of the containment vent isolation signal. Also, the applicant is required to demonstrate the ability to monitor the CPS isolation by understanding on which control board to observe such isolation.

Technical Reference: 1-47W611-30-1
.jpg picture of 1-M-6

Proposed references to be provided: None

Learning Objective: 3-OT-TS0303, TS 3.3 “INSTRUMENTATION”
1. DESCRIBE the following aspects of Technical Specifications and Technical Requirements for Instrumentation Tech Specs:
a. The Limiting Conditions for Operation, Applicability, and Bases for the LCO.
b. The conditions and required actions with completion time of one hour or less.

Cognitive Level:

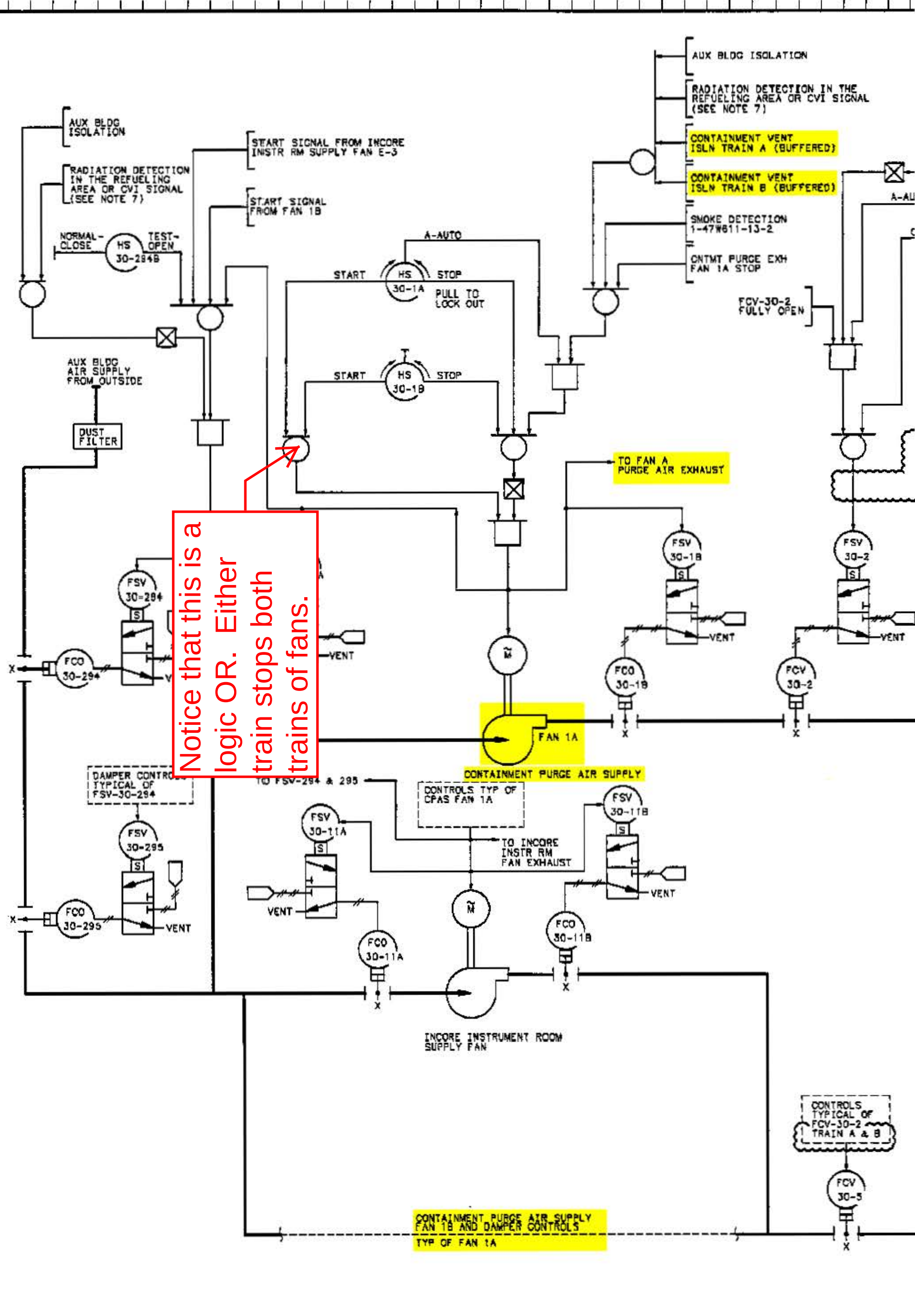
Higher X
Lower ____

Question Source:

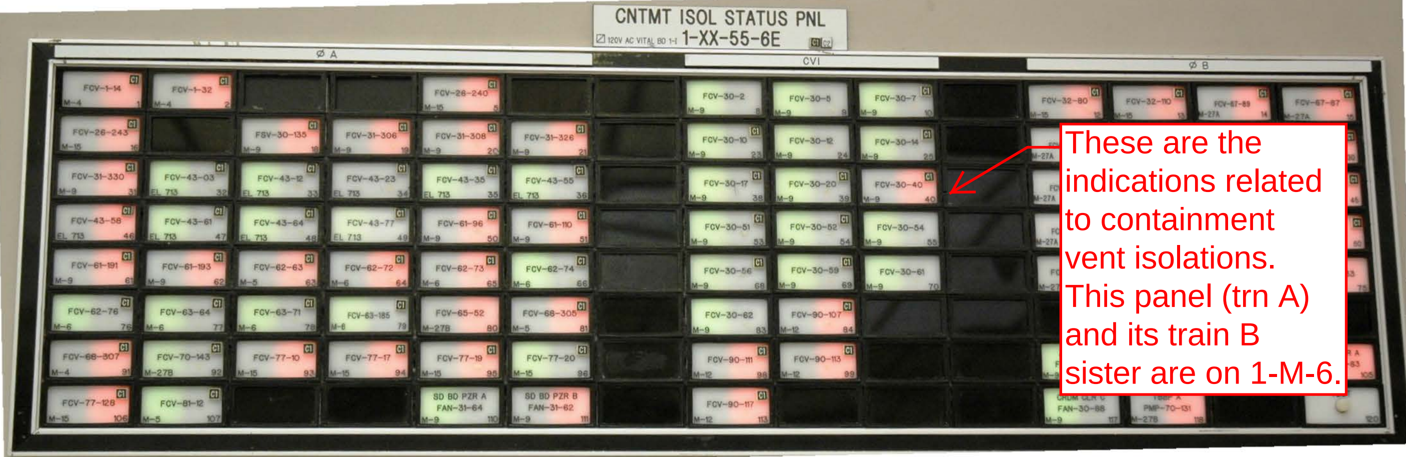
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:



AUX BLDG ISOLATION
RADIATION DETECTION IN THE REFUELING AREA OR CVI SIGNAL (SEE NOTE 7)
CONTAINMENT VENT ISLN TRAIN A (BUFFERED)
CONTAINMENT VENT ISLN TRAIN B (BUFFERED)
SMOKE DETECTION 1-47W611-13-2
CNTMT PURGE EXH FAN 1A STOP
FCV-30-2 FULLY OPEN
A-AU
AUX BLDG ISOLATION
RADIATION DETECTION IN THE REFUELING AREA OR CVI SIGNAL (SEE NOTE 7)
START SIGNAL FROM INCORE INSTR RM SUPPLY FAN E-3
START SIGNAL FROM FAN 1B
NORMAL-CLOSE
HS 30-294B
TEST-OPEN
A-AUTO
START
HS 30-1A
STOP
PULL TO LOCK OUT
START
HS 30-1B
STOP
AUX BLDG AIR SUPPLY FROM OUTSIDE
DUST FILTER
TO FAN A PURGE AIR EXHAUST
FSV 30-1B
FSV 30-2
FSV 30-294
VENT
FCO 30-294
FCO 30-1B
FCV 30-2
FAN 1A
CONTAINMENT PURGE AIR SUPPLY
Notice that this is a logic OR. Either train stops both trains of fans.
DAMPER CONTROL TYPICAL OF FSV-30-294
TO FSV-294 & 295
CONTROLS TYP OF CPAS FAN 1A
TO INCORE INSTR RM FAN EXHAUST
FSV 30-295
FSV 30-11A
FSV 30-11B
FCO 30-295
FCO 30-11A
FCO 30-11B
VENT
INCORE INSTRUMENT ROOM SUPPLY FAN
CONTROLS TYPICAL OF FCV-30-2 TRAIN A & B
FCV 30-3
CONTAINMENT PURGE AIR SUPPLY FAN 1B AND DAMPER CONTROLS
TYP OF FAN 1A



CNTMT ISOL STATUS PNL
120V AC VITAL BD 1-I 1-XX-55-6E
Ø A
CVI
Ø B
FCV-1-14 M-4 1
FCV-1-32 M-4 2
FCV-26-240 M-15 5
FCV-30-2 M-9 8
FCV-30-5 M-9 9
FCV-30-7 M-9 10
FCV-32-80 M-15 12
FCV-32-110 M-15 13
FCV-67-89 M-27A 14
FCV-67-87 M-27A 15
FCV-26-243 M-15 16
FSV-30-135 M-9 18
FCV-31-306 M-9 19
FCV-31-308 M-9 20
FCV-31-326 M-9 21
FCV-30-10 M-9 23
FCV-30-12 M-9 24
FCV-30-14 M-9 25
FCV-31-330 M-9 31
FCV-43-03 EL 713 32
FCV-43-12 EL 713 33
FCV-43-23 EL 713 34
FCV-43-35 EL 713 35
FCV-43-55 EL 713 36
FCV-30-17 M-9 38
FCV-30-20 M-9 39
FCV-30-40 M-9 40
FCV-43-58 EL 713 46
FCV-43-61 EL 713 47
FCV-43-64 EL 713 48
FCV-43-77 EL 713 49
FCV-61-96 M-9 50
FCV-61-110 M-9 51
FCV-30-51 M-9 53
FCV-30-52 M-9 54
FCV-30-54 M-9 55
FCV-61-191 M-9 61
FCV-61-193 M-9 62
FCV-62-63 M-5 63
FCV-62-72 M-6 64
FCV-62-73 M-6 65
FCV-62-74 M-6 66
FCV-30-56 M-9 68
FCV-30-59 M-9 69
FCV-30-61 M-9 70
FCV-62-76 M-6 76
FCV-63-64 M-6 77
FCV-63-71 M-6 78
FCV-63-185 M-6 79
FCV-65-52 M-27B 80
FCV-68-305 M-5 81
FCV-30-62 M-9 83
FCV-90-107 M-12 84
FCV-68-307 M-4 91
FCV-70-143 M-27B 92
FCV-77-10 M-15 93
FCV-77-17 M-15 94
FCV-77-19 M-15 95
FCV-77-20 M-15 96
FCV-90-111 M-12 98
FCV-90-113 M-12 99
FCV-77-128 M-15 106
FCV-81-12 M-5 107
SD BD PZR A FAN-31-64 M-9 110
SD BD PZR B FAN-31-62 M-9 111
FCV-90-117 M-12 113
FAN-30-88 M-9 117
PMP-70-131 M-27B 118
120
These are the indications related to containment vent isolations. This panel (trn A) and its train B sister are on 1-M-6.

61.

Given the following conditions:

- 0-GATE-79-5, Spent Fuel Pit and Fuel Transfer Canal Gate is INSTALLED.

- SFP level is 749’ 1 ½”.

- The Unit 1 Reactor cavity is EMPTY.

- The flange on the Unit 1 Containment side of the fuel transfer tube is REMOVED.

- Due to a failure of 0-LS-78-3, SFP LEVEL HI/LO (128-A), is DISABLED.

Which ONE of the following describes the response of the SFP level AND the means by which SFP level will be monitored as a response to the failure of annunciator window 128-A, in accordance with 0-SOI-78.01, Spent Fuel Pool Cooling and Cleaning System?

SFP level will ____(1)____ **IF** 1-ISV-78-600, Fuel Transfer Tube Isol is OPENED.

SFP level will be monitored ____(2)____.

	(1)	(2)
A.	LOWER ⇓	at the Spent Fuel Pit itself
B.	LOWER ⇓	using a meter on 1-M-6
C.	REMAIN the same ⇔	at the Spent Fuel Pit itself
D.	REMAIN the same ⇔	using a meter on 1-M-6

<u>CORRECT ANSWER:</u> <u>C</u>

DISTRACTOR ANALYSIS:

A. Incorrect: As seen in the system description WBN-SDD-N3-78-4001, Spent Fuel Pool Cooling and Cleaning System, the manual gate valve1-ISV-78-600 is located on the auxiliary building side of the transfer tube. One may also see this on mechanical print 47W455-1 as well as the intertie between the transfer canal and the equipment pit of the reactor cavity; also of note on this print is the fact that the containment side of the fuel transfer tube has a flange which is installed when the plant is not refueling. Prints 47W454-2 and 47W454-3 show a top down view of the transfer canal and spent fuel pit. Print 47W454-3 also shows the level placard located on the south west wall of the spent fuel pit and the gate which can be installed to separate the transfer canal from the spent fuel pit. Print 47W454-6 shows the details of the level placard. From these resources one may see that the opening of 1-ISV-78-600 will allow water to flow from either the auxiliary building or the containment given that either possesses a water inventory greater than the other. In the question posed, the auxiliary building (SFP) possesses a greater inventory than the containment side but because the SFP is isolated from the flow path by 0-GATE-79-5, Spent Fuel Pit and Fuel Transfer Canal Gate the level of the SFP will remain constant. It is plausible to believe that level in the SFP would lower if one did not understand the intertie between the various refueling volumes. For example, if one believed that 1-ISV-78-600 were attached directly to the SFP or to the fuel cask loading area then level would lower. Also, if the gate between the SFP and the transfer canal were not installed, then the level in the SFP would lower.

0-SOI-78.01 contains precaution and limitation K. which states: If MCR SFP alarm is **NOT** available, local monitoring of level with direct communication to MCR should be established. MCR panel 1-M-6, while it contains the annunciator 128-A does not contain any meter or indicator which presents the actual level of the SFP. It is plausible to believe that it does because there are level indicators which are used during outage and which would indicate the SFP as it is cross connected with the cavity. Therefore, when the annunciator is lost, the appropriate action is to direct an operator to monitor the SFP level at the SFP itself.

B. Incorrect: As discussed it is Incorrect and yet plausible that the level in the SFP would not lower. Also, it is Incorrect and yet plausible that an operator would monitor the SFP level at 1-M-6..

C. Correct: It is correct that level in the SFP would remain the same. It is also correct that level would be monitored from the SFP itself.

D. Incorrect: While it is correct that level in the SFP would remain the same, it is not correct and yet plausible that the level of the SFP would be monitored at 1-M-6.

Question Number: 61

Tier: 2 Group: 2

K/A: 033 Spent Fuel Pool Cooling System (SFPCS)
A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel Pool Cooling System operating the controls including:
A1.01 Spent fuel pool water level

Importance Rating: 2.7 3.3

10 CFR Part 55: (CFR: 41.5 / 45.5)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to first predict the change in SFP level which would result from the opening of 1-ISV-78-600. Additionally the applicant must be able to select the correct location to monitor the level of the SFP given a loss of annunciation related to the spent fuel cooling system.

Technical Reference: WBN-SDD-N3-78-4001, Spent Fuel Pool Cooling and Cleaning System
47W455-1; 47W454-2; 47W454-3; 47W454-6
0-SOI-78.01, Spent Fuel Pool Cooling And Cleaning System

Proposed references to be provided: None

Learning Objective: 3-OT-SYS079A, FUEL HANDLING
1. STATE the location of the following Fuel Handling System components:
a. New Fuel Pit
b. Spent Fuel Pit
c. Cask Loading area
d. Transfer Canal
e. Upenders (including equipment, controls and power supply)
f. Refueling Machine (Including equipment, controls and power supplies)
g. RCCA Change Fixture
h. Fuel Transfer Tube gate valve

Cognitive Level:
Higher X
Lower ___

Question Source:
New X
Modified Bank ___
Bank ___

Question History: New question for the 2015-301 NRC RO Exam

Comments:

5.2 System Leakage

Leakage detection is provided for the SFP by leakage channels located on the back side of each welded joint of the floor and walls of SFP steel liner. Leakage into these channels will drain to the perimeter leakage channels located at the bottom of the SFP. The leakage will then flow into the SFP drain pipe to a normally open manual gate valve. Visual detection of the leakage from the SFP may be witnessed as the leakage exits the manual valve and drips into a funnel. The leakage is then routed to the tritiated drain collector tank (TDCT) of the waste disposal system. In the event of excessive leakage, the manual gate valve may be closed to prevent further leakage. Similar type design of leakage channels and visual display of leakage are also provided for the fuel transfer canal and the cask setdown pool. Nonqualified instrumentation are provided in the SFP and the TDCT with MCR and local, low and high level alarms, respectively.

Leakage from the SFPCCS pumps is routed to the station drainage system. Abnormal equipment leakage is detected by visual methods.

5.3 Loss of Coolant Accident (LOCA)

In the event of a LOCA, the CCS flow to the shell side of the SFP heat exchangers may be diverted to ensure an adequate flow of cooling water for safeguard loads after an accident.

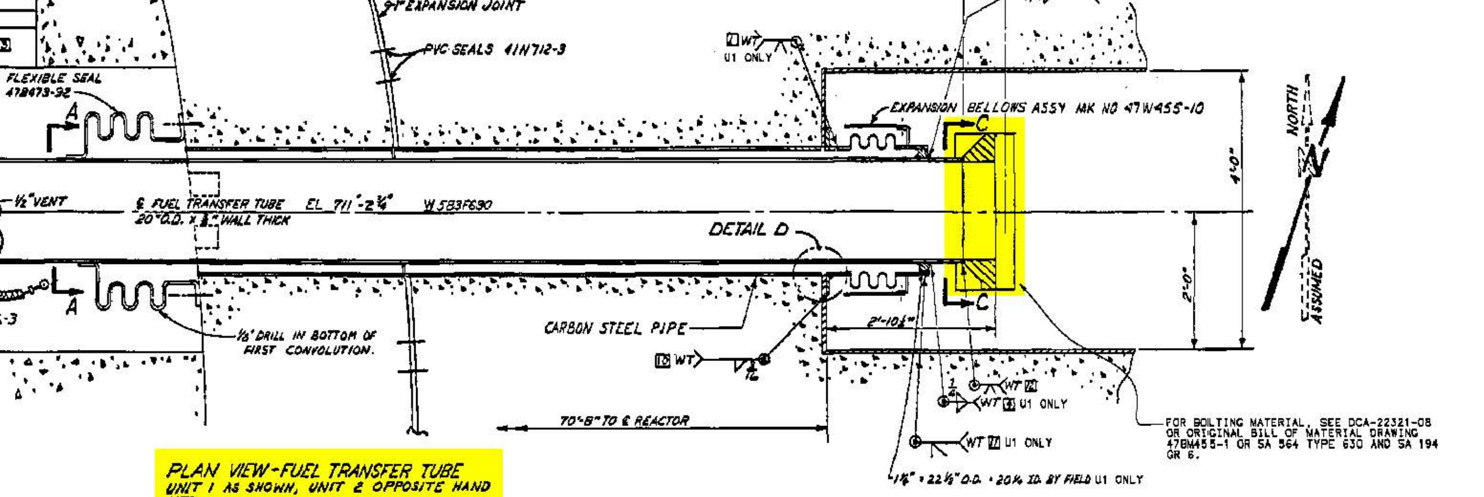
5.4 Failure of the Refueling Cavity Water Seal

SFP water level would decrease in the event of a leak at the Refueling Cavity Water Seal because the transfer tube would be open connecting the refueling canal, and the transfer canal with the SFP. The decreasing SFP water level would be detected by instrumentation located in the SFP that alarms in the main control room. The SFP would be isolated from the refueling cavity by closing the manual gate valve (78-ISV-600) on the auxiliary building side of the transfer tube. If the water level in the refueling cavity decreases to the elevation of the reactor vessel flange before the SFP can be isolated, there will be two feet of water above the stored fuel in the SFP. If during this time a spent fuel assembly were in transit between the reactor core and the SFP and was in the upright position, it could become uncovered. The consequences of uncovering the spent fuel assembly would be possible high radiation level, fuel cladding failures, and release of radioactivity into the containment and/or auxiliary building. For this scenario, the events would be bounded by the design basis analysis for a fuel handling accident as described in Section 15.4.5 of the WBN FSAR. SFP water temperature would approach and remain at the boiling temperature until the SFPCCS service is restored.

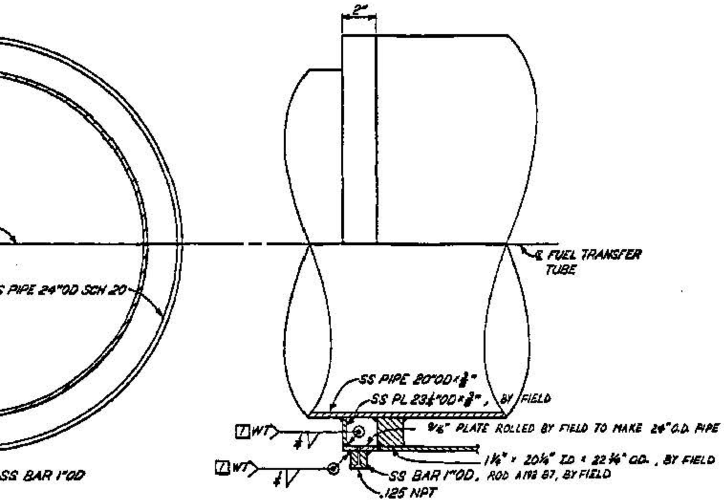
6.0 MAINTENANCE AND TESTING COMMITMENTS

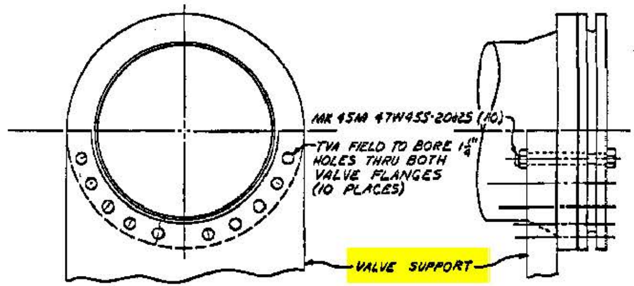
Preoperational testing requirements for specific items not contained in other design output documents are provided in Appendix A.

Each fuel pool gate shall be visually inspected for defects in the structure, seals or fittings prior to each use. The relief valves shall be checked to prevent over inflation of the seals. Leakage of the seal shall be checked by pressure decay test after the gate is installed. Seals may also be checked in the stored position.



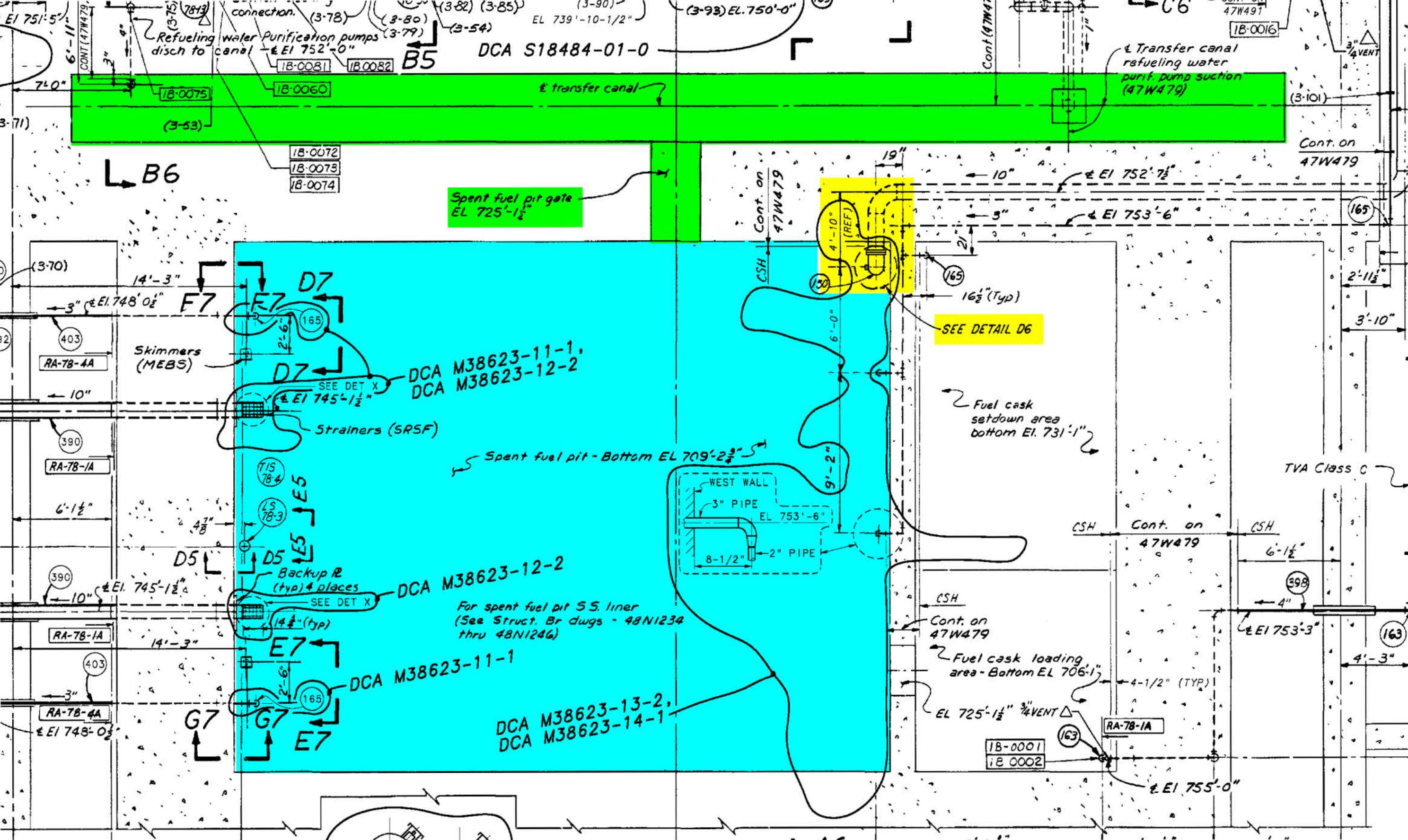
PLAN VIEW-FUEL TRANSFER TUBE
UNIT 1 AS SHOWN, UNIT 2 OPPOSITE HAND
NTS





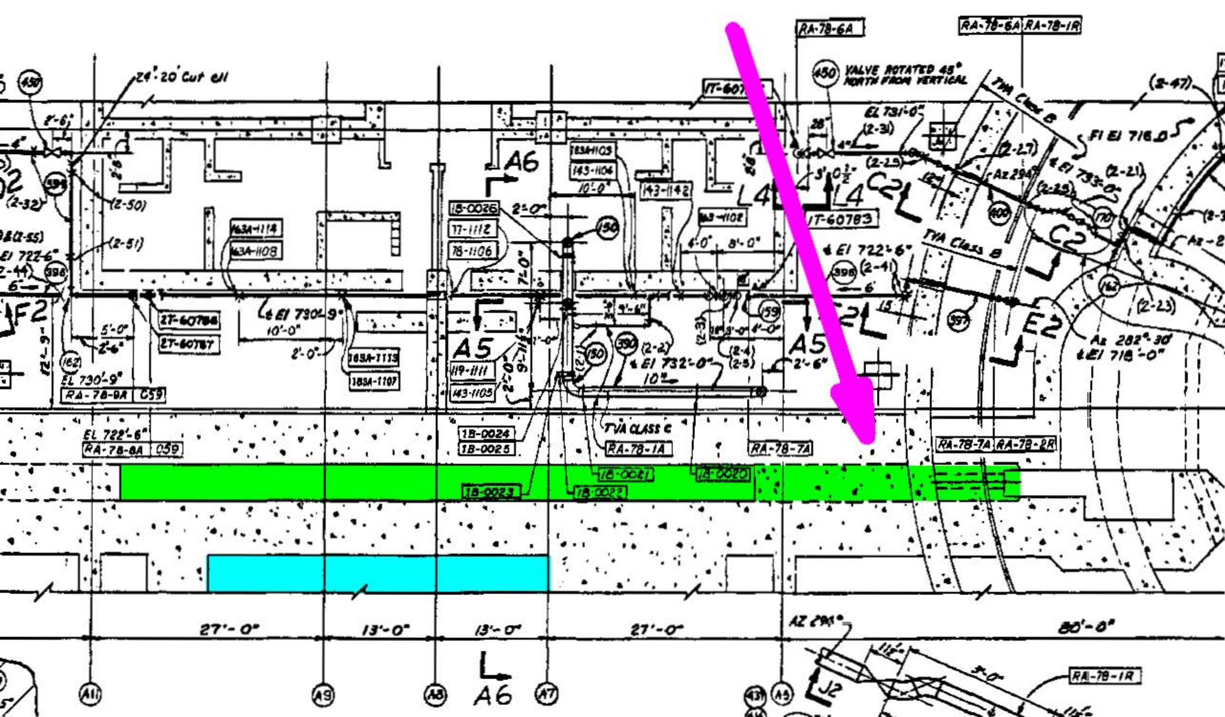
SECTION C-C
NTS

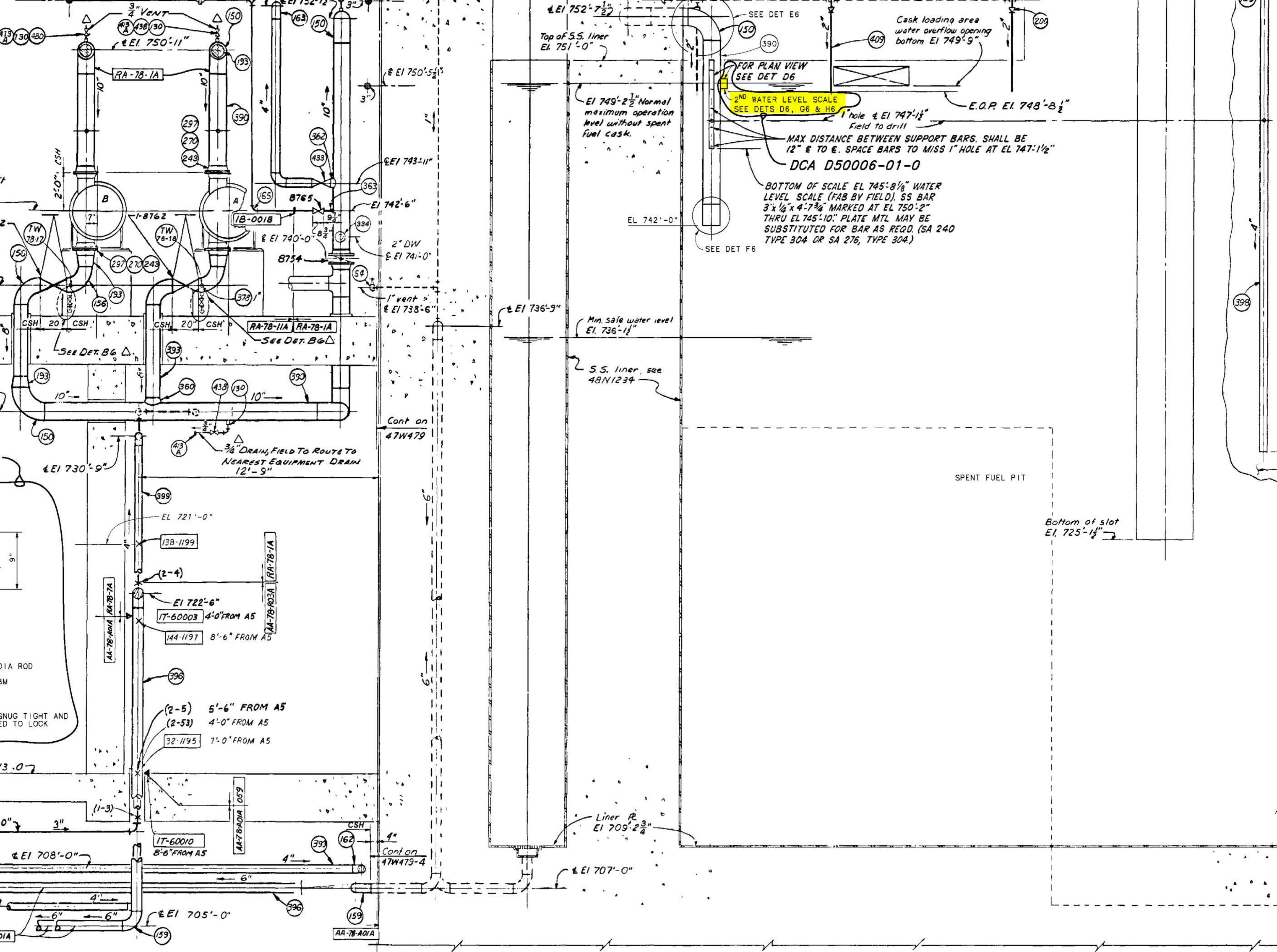
1/2" SCH 80S
BY FIELD



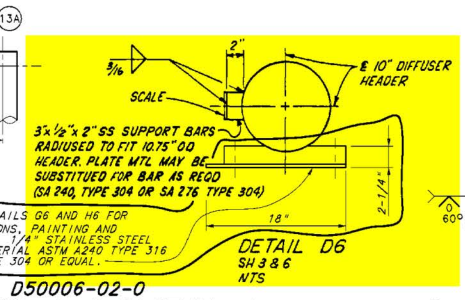
DCA S18484-01-0
Refueling water Purification pumps disch to canal
℄ El 752'-0"
IB-0081
IB 0082
IB-0075
IB-0060
℄ transfer canal
(3-53)
(3-101)
℄ Transfer canal refueling water purif. pump suction (47W479)
IB-0016
47W491
IB-0072
IB-0073
IB-0074
B5
B6
C6
Spent fuel pit gate EL 725'-1½"
Cont. on 47W479
19"
10"
℄ El 752'-7½"
3"
℄ El 753'-6"
4'-10" (REF)
CSH
165
130
16½" (Typ)
SEE DETAIL D6
(3-70)
14'-3"
F7
D7
3" ℄ El. 748'-0½"
403
RA-78-4A
Skimmers (MEBS)
10"
SEE DET X
℄ El 745'-1½"
DCA M38623-11-1,
DCA M38623-12-2
Strainers (SRSF)
390
RA-78-1A
Spent fuel pit - Bottom EL 709'-2¾"
6'-0"
9'-2"
Fuel cask setdown area bottom El. 731'-1"
TVA Class C
6'-1½"
4⅞"
TIS 78-4
LS 78-3
E5
D5
WEST WALL
3" PIPE
EL 753'-6"
8-1/2"
2" PIPE
CSH
Cont. on 47W479
6'-1½"
Backup ℞ (typ) 4 places
℄ El. 745'-1½"
DCA M38623-12-2
For spent fuel pit S.S. liner (See Struct. Br dwgs - 48N1234 thru 48N1246)
14¾" (typ)
398
4"
℄ El 753'-3"
163
E7
DCA M38623-11-1
Fuel cask loading area - Bottom EL 706'-1"
4-1/2" (TYP)
4'-3"
G7
DCA M38623-13-2,
DCA M38623-14-1
EL 725'-1½" ¾" VENT
RA-78-1A
IB-0001
IB 0002
℄ El 755'-0"
2'-11½"
3'-10"
Cont. on 47W479
7'-0"
6'-11"
3"
4"

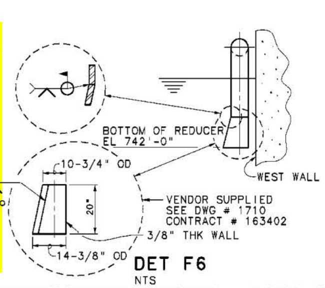
Location of 1-ISV-78-600





SECTION A6 - A6
(Sheets 1, 2 & 3)





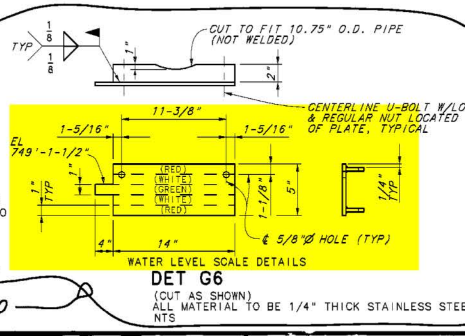
NOTES FOR DET G6

PAINT COATING MATERIAL
RED: AMERLOCK 400NT, SAFETY RED 1979, OR EQUAL
WHITE: AMERLOCK 400NT, WHITE, OR EQUAL
GREEN: AMERLOCK 400NT, RT-2305 GREEN, OR EQUAL

THE STRIPES USED ARE GREEN TO REPRESENT THE NORMAL OPERATING RANGE FROM 749'-1/2" TO 749'-1-1/2", WHITE (YELLOW WAS NOT READILY AVAILABLE) TO REPRESENT THE RANGE FROM 749'-1-1/2" TO 749'-2-1/2" AT THE HIGH LEVEL ALARM INDICATOR AND THE RANGE FROM 749'-1/2" TO 748'-11-1/2" AT THE LOW LEVEL ALARM INDICATOR; AND UPPER AND LOWER RED STRIPES TO INDICATE LEVELS ABOVE THE LEVEL ALARM RANGES. (FOR ADDITIONAL HELP IN IDENTIFYING THE WATER LEVEL THE GREEN LINE INDICATING THE NORMAL LEVEL IS EXTENDED TO A 1 INCH WIDE POINTER ON THIS SECOND LEVEL INDICATOR).

FOR THIS APPLICATION THE AMERLOCK 400NT PAINT IS ACCEPTABLE FOR USE ON THE STAINLESS STEEL MATERIAL AND IN THE FUEL PIT WATER.
DO NOT PAINT WITHIN 1.5 INCH DIAMETER OF HOLE CENTERLINES TO CLEAR NUTS.

DCA D50006-03-0



3.0 PRECAUTIONS AND LIMITATIONS

A. SFP spills may cause contamination spread and unnecessary radiation exposure. Take immediate corrective actions including informing Rad Protection.

B. Loose items may fall into SFP, and damage Fuel assembly, racks, etc. Loose items must be secured to personnel or another piece of stationary equipment.

C. The SFP water temperature is **NOT** to be permitted to exceed 150°F during normal operation and anticipated refueling activities, pursuant to the FSAR.

D. The SFP Filter requires replacing when ΔP of 20 psi or greater, OR Rad Protection discretion (radiation level).

E. Exceeding 100 gpm or 140°F through SFP Demin can damage resins. During refueling outages, flows up to 180 gpm through SFP Demin using the RWPS pumps are acceptable. Ultrasonic flow-meters may be used to determine flows during refueling cleanup

F. If Skimmer Loop is in service, air may enter skimmer head, causing pump cavitation. SFP level should be checked periodically to ensure proper relation between skimmer head and vortex created by pump suction.

G. SFP Skimmer Filter is replaced at 20 psi ΔP.

H. When taking suction from Refueling Cavity with RWPS Pump(s), monitor suction press after starting pump(s) due to high loop in suction pipe which could cause loss of pump suction.

I. RWPS Filter damage may occur if 200 gpm is exceeded.

J. During transfer activity, RWST and SFP levels must be monitored to ensure correct valve alignments and Tech Spec requirements.

K. If MCR SFP level alarm is **NOT** available, local monitoring of level with direct communication to MCR should be established.[C.1]

L. Pump Starting Guidelines are in GOI-7.[C.1]

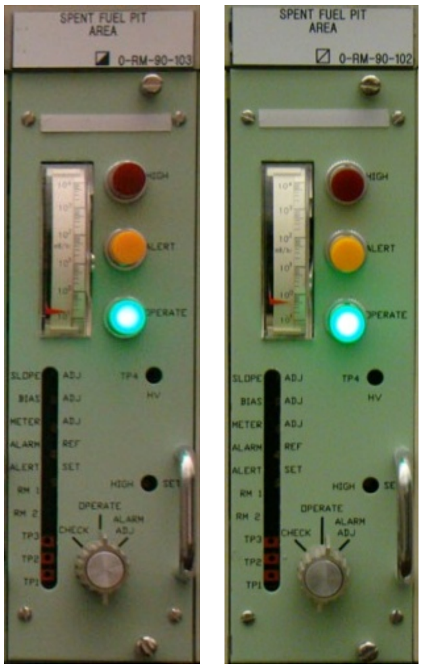
M. Spent Fuel Pit minimum temperature assumed for criticality analysis is 39°F(4°C), based on Sys 79 System Description-Nuclear Fuel Report PFE-R07, Revision 0, dated October 1996, “Criticality Analysis Summary Report, Watts Bar Nuclear Plant (WBN)” (L38 961015 802).

N. Instrument maintenance department should be notified to ensure required instruments will be placed in service as necessary to support system operation.

62.

Given the following timeline:

00:00:00	Unit 1 is at 100% power. Spent fuel movement is in progress in the SFP. The FH AREA EXH FAN A is **RUNNING.**
00:15:00	A fuel assembly is dropped while trying to extract it from its storage rack. Visual inspection shows that it is slightly bent.
00:20:00	0-RM-90-102 and 0-RM-90-103 display a **STABLE** ⟺ reading as shown.



00:21:00	The control room staff enters 1-AOI-29, Dropped or Damaged Fuel or Refueling Cavity Seal Failure section 3.4, Dropped or Damaged Irradiated Fuel Assembly in Spent Fuel Pit Area.

Which ONE of the following describes the status of FH AREA EXH FAN A AND the FIRST action the crew will implement in accordance with 1-AOI-29?

At 00:20:01, the FH AREA EXH FAN A is ____(1)____.

The **FIRST** action which will be taken is to ____(2)____ .

A. (1) running
   (2) **CHECK** ABI actuated

B. (1) secured
   (2) **CHECK** ABI actuated

C. (1) running
   (2) **EVACUATE** the affected area

D. (1) secured
   (2) **EVACUATE** the affected area

CORRECT ANSWER: C

DISTRACTOR ANALYSIS:

A. Incorrect: It is correct that the “A” FHE fan will continue running. The rate meters shown in the stem of the question do not show a high radiation condition.

Also, step 1 of 1-AOI-29 is “**EVACUATE** the affected area.” Step 7 of the AOI initiates ABI. Therefore, it is incorrect to believe that the first action would be to evacuate. It is plausible to believe that the initiation of the signal which by design is credited for protecting the health and safety of the public (the ABI which limits the release to the public) would be the immediate action.

B. Incorrect: The conditions shown in the stem of the question do not indicate that a high radiation signal has been reached. If such had been the case, the Rate Meters shown would have the red alarm light LIT. As seen in prints: 1-45W600-3, 1-45W760-30-14 and 1-45W600-57-30, a high radiation signal from either of the spent fuel pit radiation monitors will cause both of the FHE fans to be automatically secured. It is plausible to believe that the FHE fan would be secured as damage to the fuel assembly was noted.

Again, it is incorrect and yet plausible that the initiation of the ABI would be the first action taken.

C. Correct: It is correct that the “A” FHE fan would remain running. It is also correct that the first action taken in 1-AOI-29 is to evacuate the affected area.

D. Incorrect: Again, it is incorrect and yet plausible that the “A” FHE fan would be secured. It is correct that the first action taken in 1-AOI-29 is to evacuate the affected area.

Question Number: 62

Tier: 2 Group: 2

K/A: 034 Fuel Handling Equipment System (FHES)
A2 Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System ; and (b) based on those predictions, use procedures to Correct: control, or mitigate the consequences of those malfunctions or operations:
A2.01 Dropped fuel element

Importance Rating: 3.6 4.4

10 CFR Part 55: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to predict the impact that a dropped fuel element has upon the FH AREA EXH FAN A (a component of the fuel handling equipment system). Subsequently, the applicant must be able to prioritize the actions contained in the abnormal response instruction for the dropped fuel assembly in the spent fuel pit area; thus, demonstrating their ability to use the procedure to mitigate the consequences of the accident.

Technical Reference: 1-45W600-30-4
1-AOI-29, Dropped or Damaged Fuel or Refueling Cavity Seal Failure

Proposed references to be provided: None

Learning Objective: 1-AOI-29, Dropped or Damaged Fuel or Refueling Cavity Seal Failure
3. IDENTIFY Alarms, symptoms, automatic actuations, and other indications of AOI-29, Dropped or Damaged Fuel or Refueling Cavity Seal Failure.
4. Given a set of plant conditions, DESCRIBE operator actions required in response to the following per AOI-29, Dropped or Damaged Fuel or Refueling Cavity Seal Failure:

Cognitive Level:

Higher X

Lower ____

Question Source:

New X

Modified Bank ____

Bank ____

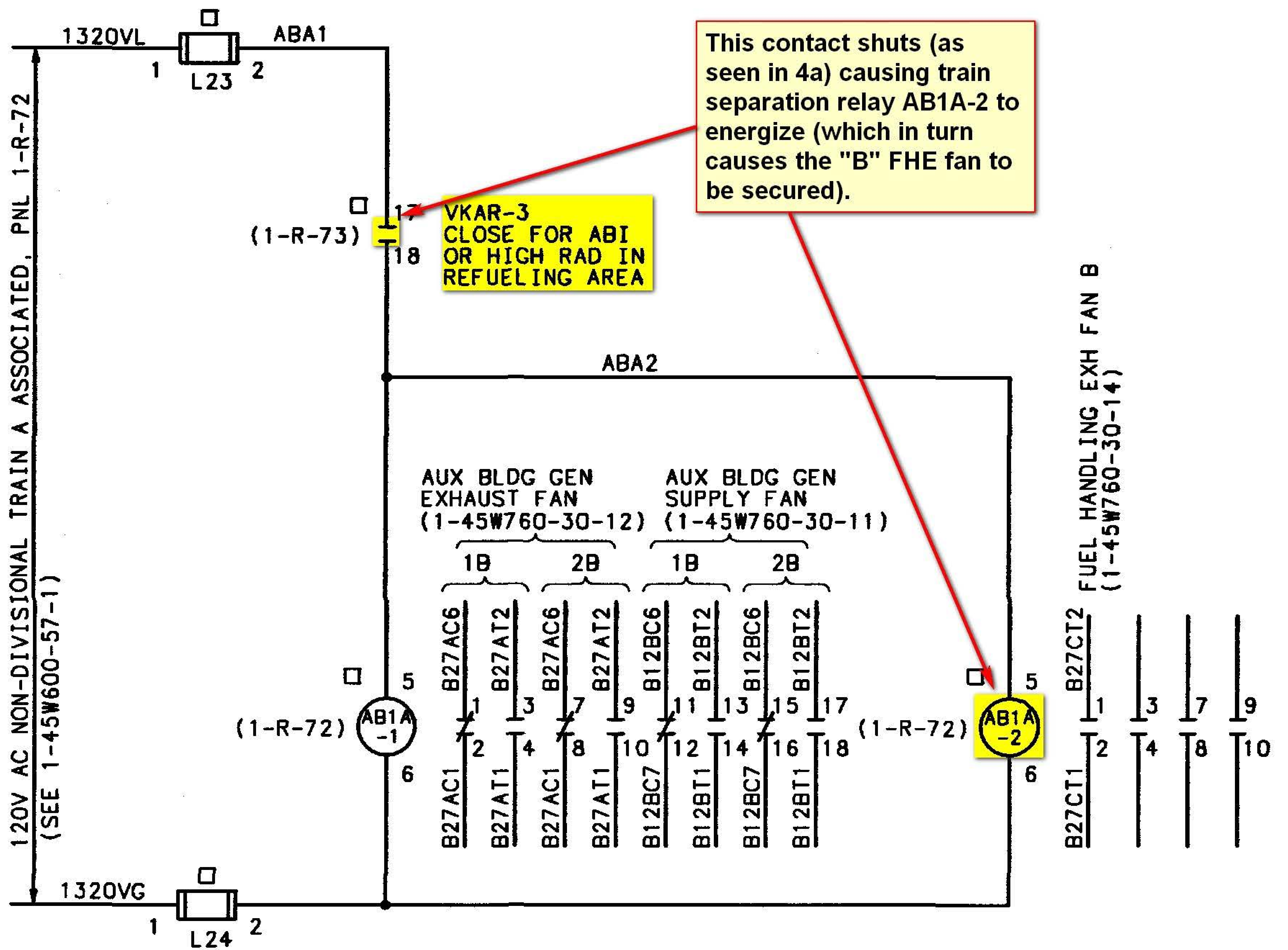
Question History: New question for the 2015-301 NRC RO Exam

Comments:

VKAL
KO-R102
(0-M-12) CLOSES FOR HIGH RAD IN REFUELING AREA
Step 1: This current path is made up during an actual high rad event
Step 3: This contact closes.
VKAR-6
VKA1
VKA3
VKA6
VKA7
SEE NOTE 4
SEE NOTE 8
CLOSES FOR UNIT 1 CONTAINMENT ISOLATION PHASE A
CLOSES FOR UNIT 2 CONTAINMENT ISOLATION PHASE A
K-606 (2-R-48)
(54114-1 7246D11-19)
1-TS-30-103 CLOSED FOR HIGH TEMP AUX BLDG GEN SUPPLY DUCT (UNIT 1)
2-TS-30-104 CLOSED FOR HIGH TEMP AUX BLDG GEN SUPPLY DUCT (UNIT 2)
VKAR -2
VKAR -3
VKAR -4
VKAR -5
VKAR -7
VKAR -6
(1-45W600-57-30)
(1-45W760-30-14)
Step 4a: This relay energizes causing this contact to shut.
Step 4b: This relay energizes causing this contact to shut.
Step 2: VKAR-6 is energized.
AUX BLDG GEN EXH FANS
AUX BLDG GEN SUPPLY FANS
SEP RELAYS
TRAIN A AUXILIARY BUILDING ISOLATION & HIGH RADIATION IN REFUELING AREA LOGIC BUS
VKBL
KO-R103
(0-M-12) CLOSES FOR HIGH RAD IN

FUEL HANDLING EXHAUST FAN A (0-MTR-30-136) (SEE NOTE 3)
(480V SHUTDOWN BOARD 2A2-A ONLY)

This excerpt is from 1-45W760-30-14.



AUXILIARY BUILDING GENERAL EXH AND SUPPLY FANS 1B & 2B & FUEL HANDLING EXH FAN B SEPARATION RELAYS

(COMMON TO PLANT)

Step	Action/Expected Response	Response Not Obtained

3.4 Dropped or Damaged Irradiated Fuel Assembly in Spent Fuel Pit Area

CAUTION

- Maintenance must be notified to IMMEDIATELY ensure at least one door is closed on both upper and lower personnel air locks, and close any other open penetrations required per TI-68.002.
- Entry to the Refuel Floor and other affected areas shall be coordinated with Radiation Protection.

Step	Action/Expected Response	Response Not Obtained
1.	**EVACUATE** the affected area: • Fuel Handling SRO **NOTIFY** personnel in the general area of radiation concern. • **ANNOUNCE** for all personnel to evacuate the affected area.	
2.	**NOTIFY** Shift Manager and Radiation Protection of radiation release.	
3.	**EVALUATE** conditions and necessary protective measure prior to reentry into affected area.	
4.	**ENSURE** closure initiated for personnel air locks and any other open penetrations in accordance with TI-68.002.	
5.	**CHECK** 0-RM-90-101A, B, C Aux Bldg Vent monitor NORMAL [0-M-12].	**NOTIFY** Radiation Protection to begin surveys and monitoring of Refuel Floor.
6.	**CHECK** SFP area monitors 0-RM-90-102 and -103 NORMAL [0-M-12].	**STOP** and **LOCKOUT** Fuel Handling Area Exh Fans.

Step	Action/Expected Response	Response Not Obtained

3.4 Dropped or Damaged Irradiated Fuel Assembly in Spent Fuel Pit Area (continued)

Step	Action/Expected Response	Response Not Obtained
7.	**CHECK** ABI actuated: • ABI Train A window LIT [MISSP] • ABI Train B window LIT [MISSP]	**INITIATE** ABI by placing 1-HS-30-101A and 1-HS-30-101B to ACTUATE [1-M-6].
8.	**CHECK** the following fans OFF [1-M-9]: • Aux Bldg Supply Fans • Aux Bldg Exhaust Fans • Fuel Handling Area Exh Fans	Manually **STOP** fans as necessary. The ABI is initiated regardless of the actual radiation levels. It is the accident signal which will mitigate any radioactive release to the public.
9.	**CHECK** Train A and Train B ABGTS running [0-M-25].	**START** ABGTS Train A and Train B.
10.	**CHECK** the following ABGTS dampers OPEN: • 1-FCO-30-146B, Train A Suct • 1-FCO-30-146A, Train A Disch • 2-FCO-30-157B, Train B Suct • 2-FCO-30-157A, Train B Disch	Manually **OPEN** dampers.
11.	**IF** Containment or Annulus is OPEN to the Aux Bldg, **THEN** **CHECK** Containment Vent Isolation has occurred: • CVI Train A window LIT [MISSP] • CVI Train B window LIT [MISSP] • 1-XX-55-6E CVI Train A green • 1-XX-55-6F CVI Train B green	**IF** CVI has **NOT** occurred, **THEN**: a. **STOP** Purge Air Supply & Exh Fans. b. **CLOSE** Purge Air Cntmt Isol Dampers. c. **CLOSE** Cntmt Press Relief Valves [1-M-9]. d. **MONITOR** Cntmt Pressure.

63.

Given the following conditions:

- Unit 1 SG #1 is faulted.
- Unit 1 has tripped and Safety Injection has actuated.
- When the MSIVs hand-switches were placed to **CLOSED**, the #1 MSIV failed to **CLOSE**.
- The control room is **MINIMALLY** staffed.

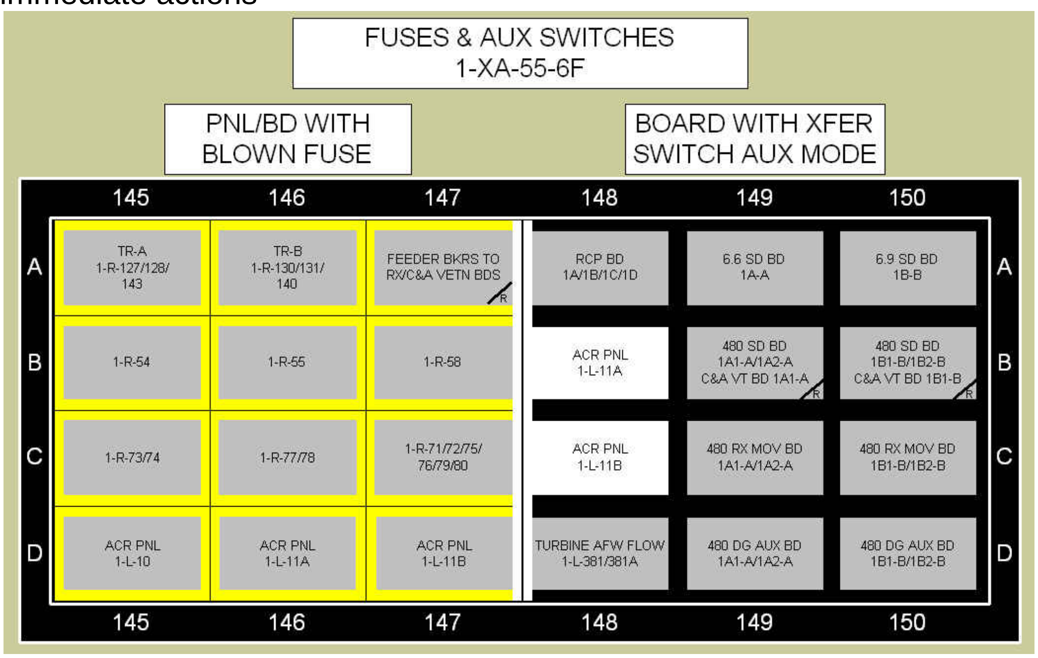
Which ONE of the following describes when AFW flow may be isolated AND the indications on 1-M-4?

In accordance with 0-TI-12.04, User's Guide For Abnormal And Emergency Operating Instructions, AFW flow to SG #1 may be isolated after the ____(1)____ of 1-E-0 are complete.

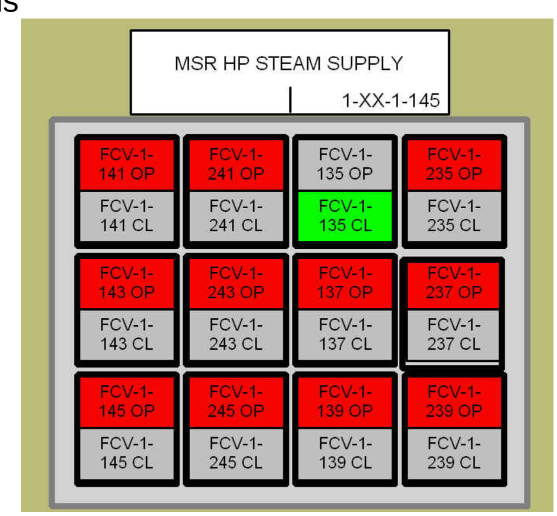
In accordance with 1-E-2, the **FIRST** MCR indication that compensatory measures for the failure of the #1 MSIV to close **HAVE** been taken is _____(2)_____.

NOTE: 1-E-0, Reactor Trip or Safety Injection
1-E-2, Faulted Steam Generator Isolation

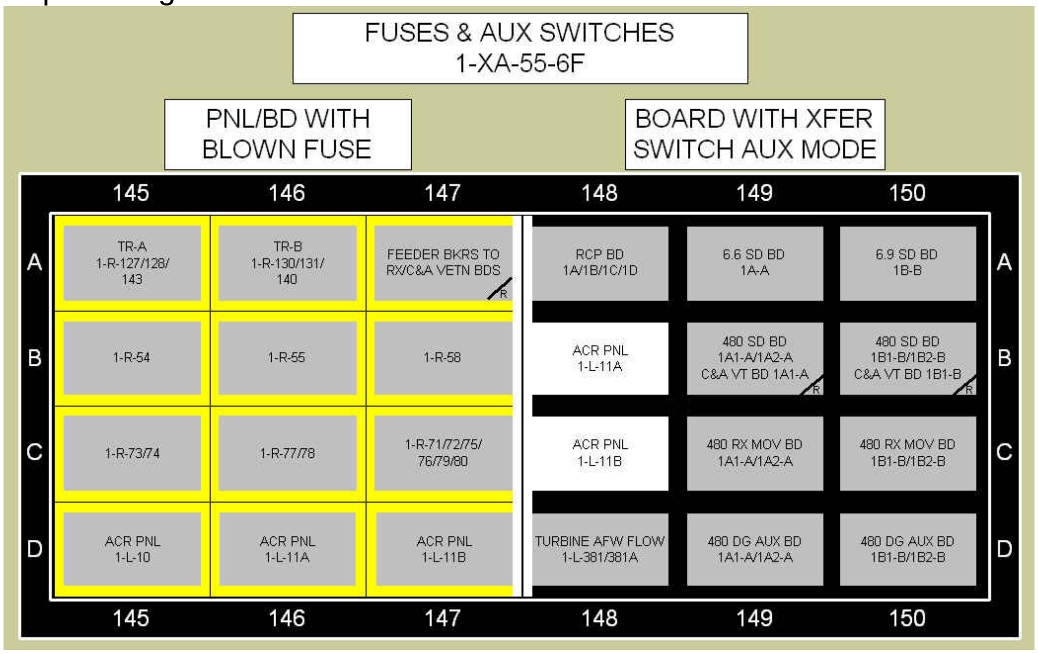
A. (1) immediate actions
(2)



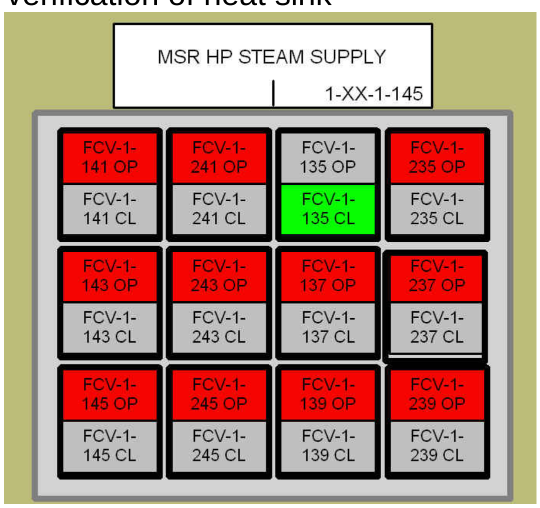
B. (1) immediate actions
(2)



C. (1) steps through the verification of heat sink
(2)



D. (1) steps through the verification of heat sink
(2)



CORRECT ANSWER: C

DISTRACTOR ANALYSIS:

A. Incorrect: 0-TI-12.04 states: “Early isolation of auxiliary feedwater (AFW) should be performed after positive identification of a ruptured, faulted, or ruptured and faulted S/G providing the following guidelines are met: a. Prior to isolation, the steps of 1-E-0 are complete through verification of heat sink.” 0-TI-12.04 also states: “Diagnostic or repair actions will be delayed until the immediate actions are complete to allow for evaluation of plant response.” Therefore, it is incorrect that AFW may be isolated to the faulted S/G immediately following the first four steps (the immediate steps) of 1-E-0. It is plausible to believe this as the second excerpt of 0-TI-12.04 shows a requirement which would independently support the isolation of AFW immediately following the completion of the immediate actions.

As seen in 1-E-2, the first action taken to compensate for a failed open MSIV (by a minimally staffed crew) is to manipulate AUX transfer switches. The resulting indication for such component operation is that of the “ACR PNL 1-L-11A” and “ACR PNL 1-L-11B” lights lit on 1-XX-55-6F, FUSES & AUX SWITCHES indicating panel.

B. Incorrect: Again, it is incorrect and yet plausible that the isolation of AFW to a faulted S/G would occur following the completion of the immediate action steps of 1-E-0.

It is incorrect that a green position would be seen on 1-FCV-1-135 before the ACR PNL lights were observed. The manipulation of 1-HS-1-135A is directed by Attachment 2 of 1-E-2. If the control room is minimally staffed, Attachment 2 is always performed at the conclusion of Attachment 1 (if no resolution is had to the failure of the MSIV to close). It is plausible to believe that the green position is seen first because if an extra operator were available, then the Attachment 2 could be performed in parallel with Attachment 1. A board operator could certainly manipulate a HS on 1-M-2 prior to an AUO reaching the ACR and manipulating two transfer switches.

C. Correct: As discussed, it is correct that AFW may be isolated to a faulted S/G after the verification of heat sink. It is also correct that the ACR PNL lights would be observed prior to the green indication on 1-FCV-1-135.

D. Incorrect: While it is correct that AFW may be isolated to a faulted S/G after the verification of heat sink, it is not correct and yet plausible that the green indication on 1-FCV-1-135 would be noted prior to the ACR PNL lights.

Question Number: 63

Tier: 2 Group: 2

K/A: 035 Steam Generator System
A4 Ability to manually operate and/or monitor in the control room:
A4.06 SG isolation on steam leak or tube rupture/leak

Importance Rating: 4.5 4.6

10 CFR Part 55: (CFR: 41.7 / 45.5 to 45.8)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the question requires ability to appropriately operate components to isolate AFW to a faulted S/G and then to monitor the correct indications for the isolation of the steam flowpath.

Technical Reference: 1-E-2, Faulted Steam Generator Isolation
1-E-0, Reactor Trip or Safety Injection
0-TI-12.04, User's Guide For Abnormal And Emergency Operating Instructions, Revision

Proposed references to be provided: None

Learning Objective: 3-OT-EOP0200, 1-E-2, Faulted Steam Generator Isolation
4. LIST from memory all the components/valves that must be isolated in order to ISOLATE a faulted S/G. (No valve numbers required).

Cognitive Level:
Higher X
Lower

Question Source:
New X
Modified Bank
Bank

Question History: New Question for the 2015-301 NRC RO exam.

Comments:

2.7 Prudent Operator Actions (continued)

9. Adjusting a S/G PORV to 90% if a known SG tube leak/rupture exists.

10. Early isolation of auxiliary feedwater (AFW) should be performed after positive identification of a ruptured, faulted, or ruptured and faulted S/G providing the following guidelines are met:

    a. Prior to isolation, the steps of 1-E-0 are complete through verification of heat sink.

    b. For a ruptured S/G, ensure that narrow range (NR) level on the affected S/G is >29% and heat sink minimum requirements are met for the unaffected S/Gs.

    c. For a faulted or ruptured and faulted S/G, ensure heat sink minimum requirements are met for the unaffected S/Gs.

11. When an ATWS event is in progress and after the crew has attempted to open the reactor trip breakers from the second reactor trip hand switch and the trip is unsuccessful, the response should be as follows:

    a. The RO starts inserting control rods.

    b. The BOP/CRO immediately trips the turbine, then starts AFW.

12. If **NO** SI condition exists, heat sink may be verified and AFW throttled as necessary to stabilize RCS temperature prior to transition out of 1-E-0.

B. In deciding if taking prudent action is appropriate, the following elements should be considered as a whole:

1. Plant safety status should be maintained or enhanced. Prudent mitigation or preemptive action should **NOT** degrade plant status or put the plant in a less safe state or challenge it more than the initiating event. It should **NOT** cause a RED or ORANGE path critical safety function condition. For example, closing the MSIVs to isolate the steam dumps at 100% power would challenge the plant more than the original problem by lifting Pressurizer and S/G safeties and causing an overheat and load rejection event. However, in Mode 3, closing the MSIVs to stop an uncontrolled cooldown and positive reactivity addition would have minimal impact on plant status, while enhancing reactor safety.

2. Prudent operator actions should be consistent with procedural guidance for similar situations. For example, if a steamline break occurs in Mode 3 and S/G pressure is approaching 675 psig, a manual MSIV Isolation would anticipate the automatic response and is consistent with procedural guidance and plant design response.

2.2.3 Foldout Page

A. This page presents actions or transitions which are applicable at any time in the given instruction. Upon transition from an instruction, the current instruction's foldout page becomes applicable and use of the previous instruction's foldout page is discontinued.

1. In the control room, the foldout page information is presented on the back of each page of instructions for which there is a foldout page.

2. The information on the foldout page should be continuously monitored to determine when operator action is necessary.

B. Transitions to other instructions allow immediate response to new symptoms as they appear.

2.2.4 Immediate Action Steps

Steps that have been designated as "Immediate Actions" are contained in three emergency instructions and selected AOIs. These steps are intended to be performed, if necessary, without the written instruction being available.

A. Operators are required to be able to complete the intent of immediate operator action steps.

B. During immediate operator action steps the operators will ensure automatic actions have occurred or initiate signals as appropriate. Diagnostic or repair actions will be delayed until the immediate actions are complete to allow for evaluation of plant response.

C. The immediate actions of 1-E-0 will be addressed as follows:

1. Steps 1 through 4 will be performed in order by the OAC and completion, along with any discrepancies, will be communicated to the Procedure Reader.

2. The BOP will acknowledge alarms as necessary to reduce the noise level, and perform backup verification of steps 1 through 4.

3. Procedure Reader will read the immediate action high level step. The OAC will confirm the high level step by verbalizing the low level steps to the procedure reader.

4. When reentering 1-E-0 from another EOI, the first four high level steps must be reconfirmed, it is **NOT** necessary to reperform each low level action.

Step	Action/Expected Response	Response Not Obtained
5.	**PERFORM** Appendixes A and B , 1-E-0, pages 16-26	
6.	**ANNOUNCE** reactor trip and safety injection over PA system.	
7.	**ENSURE** secondary heat sink available with either: • Total AFW flow greater than 410 gpm, **OR** • At least one S/G NR level greater than 29% [39% ADV].	** **GO TO** 1-FR-H.1, Loss of Secondary Heat Sink.

After this step, the crew may isolated AFW to a faulted steam generator (provided that heat sink is met).

Step	Action/Expected Response	Response Not Obtained

3.0 OPERATOR ACTIONS

CAUTION If a faulted S/G is **NOT** needed for RCS cooldown, it should remain isolated during subsequent recovery actions.

1. **ENSURE** all MSIVs and MSIV bypasses CLOSED.

   The stem of the question gave that this was attempted and that the #1 MSIV failed to close.

   RNO: Manually **CLOSE** valves.

   IF valves can **NOT** be closed, **THEN**

   Locally **REMOVE** power to valves:

   - **DISPATCH** NAUO to perform Attachment 1 (1-E-2).

NOTE If it is known that a steam leak exists in the Turbine building, the following step should **NOT** be performed until the affected steam header is depressurized.

2. **PLACE** steam dump controls OFF:
   - 1-HS-1-103A, STEAM DUMP FSV “A”.
   - 1-HS-1-103B, STEAM DUMP FSV “B”.

3. **DETERMINE** if Cntmt spray should be stopped:

   a. **MONITOR** Cntmt pressure less than 2.0 psig.

   RNO: a. **WHEN** Cntmt pressure less than 2.0psig, **THEN**

   PERFORM substeps 3b thru e.

   b. **CHECK** at least one Cntmt spray pump running.

   RNO: b. ****GO TO** Step 4

   c. **RESET** Cntmt spray signal.

Step continued on next page.

Attachment 1
(Page 1 of 5)

Isolation of MSIVs and MSIV Bypass Valves

This is the reason for the statement about the control room is minimally staffed. During the casualty, the BOP would still be performing the equipment verification of 1-E-0 and thus be unavailable.

1.0 INSTRUCTIONS

NOTE
If additional MCR Operators are available, US may direct performance of Attachment 2, MSIV and Bypass Valves Fail Open, to be performed in parallel with Attachment 1.

1.1 Isolation of MSIVs and MSIV Bypass Valves

The #1 MSIV failed to close.

A. **IF** any MSIV will **NOT** close, **THEN**

GO TO Section 1.2 (Attachment 1).

B. **IF** any MSIV bypass valve will **NOT** close, **THEN**

GO TO Section 1.3 (Attachment 1).

Attachment 1
(Page 2 of 5)

Isolation of MSIVs and MSIV Bypass Valves

1.2 MSIV Isolation

A. **PLACE** affected MSIV transfer control switch in AUX position:
[Auxiliary Control Room, Panels 1-L-11A and 1-L-11B]

S/G (Circle affected S/G)	EQUIPMENT	AUX TRANSFER SWITCH	AUX POSITION √
1	MSIV Loop 1, Train A, 1-FCV-1-4	1-XS-1-4A	☐
	MSIV Loop 1, Train B, 1-FCV-1-4	1-XS-1-4B	☐
2	MSIV Loop 2, Train A, 1-FCV-1-11	1-XS-1-11A	☐
	MSIV Loop 2, Train B, 1-FCV-1-11	1-XS-1-11B	☐
3	MSIV Loop 3, Train A, 1-FCV-1-22	1-XS-1-22A	☐
	MSIV Loop 3, Train B, 1-FCV-1-22	1-XS-1-22B	☐
4	MSIV Loop 4, Train A, 1-FCV-1-29	1-XS-1-29A	☐
	MSIV Loop 4, Train B, 1-FCV-1-29	1-XS-1-29B	☐

When these transfer switches are taken to AUX, the indication on 1-XA-55-6F (as seen in the distractors) occurs in the MCR.

CONSULT UO to verify affected MSIV closed.

Notice that the caveat of the perform in parallel if additional operators are available did not apply. Therefore, this is performed after Attachment 1.

Attachment 2
(Page 1 of 3)

MSIV and Bypass Valves Fail Open

1.0 MSIV AND BYPASS VALVES FAIL OPEN

A. **IF** affected MSIV or MSIV Bypass still open after completion of Attachment 1 or performance directed by US, **THEN**:

1. **CLOSE** the following HP steam and bypass isolation valves.

This is the first of the valves to be operated. Its position indication (in the MCR) is seen in the B and D distractors.

MSR A2 HP STM ISOL	1-M-2	1-HS-1-135A	☐
MSR B2 HP STM ISOL	1-M-2	1-HS-1-137A	☐
MSR C2 HP STM ISOL	1-M-2	1-HS-1-139A	☐
MSR A1 HP STM ISOL	1-M-2	1-HS-1-141A	☐
MSR B1 HP STM ISOL	1-M-2	1-HS-1-143A	☐
MSR C1 HP STM ISOL	1-M-2	1-HS-1-145A	☐
MSR A2 HP STM BYPASS ISOL	1-M-2	1-HS-1-235A	☐
MSR B2 HP STM BYPASS ISOL	1-M-2	1-HS-1-237A	☐
MSR C2 HP STM BYPASS ISOL	1-M-2	1-HS-1-239A	☐
MSR A1 HP STM BYPASS ISOL	1-M-2	1-HS-1-241A	☐
MSR B1 HP STM BYPASS ISOL	1-M-2	1-HS-1-243A	☐
MSR C1 HP STM BYPASS ISOL	1-M-2	1-HS-1-245A	☐

2. **CLOSE** the following HP steam warming valves.

MSR A1 WARMING LINE	1-M-2	1-HS-1-142	☐
MSR A2 WARMING LINE	1-M-2	1-HS-1-136	☐
MSR B1 WARMING LINE	1-M-2	1-HS-1-144	☐
MSR B2 WARMING LINE	1-M-2	1-HS-1-138	☐
MSR C1 WARMING LINE	1-M-2	1-HS-1-146	☐
MSR C2 WARMING LINE	1-M-2	1-HS-1-140	☐

3. **ENSURE** BOTH steam Seal Supply valves in CLOSED position:

   a. 1-HS-47-180A, HP SEAL STEAM SUPPLY ISOL [1-M-2] ☐

   b. 1-HS-47-181A, HP SEAL STEAM SUPPLY BYPASS [1-M-2] ☐

64.

Which ONE of the following describes if condenser vacuum can be established AND the effect a loss of all condenser vacuum pumps WILL have on saturation temperature?

In accordance with 1-SOI-2&3.01, Condensate and Feedwater System, condenser vacuum ____(1)____ be established following a refueling outage if no condenser vacuum pumps are available.

With Unit 1 at 100% power, a loss of all condenser vacuum pumps WILL cause the saturation temperature of the condenser to ____(2)____.

	(1)	(2)
A.	CAN	RISE ⇑
B.	CAN	LOWER ⇓
C.	CAN **NOT**	RISE ⇑
D.	CAN **NOT**	LOWER ⇓

CORRECT ANSWER: C

DISTRACTOR ANALYSIS:

A. Incorrect: It is not correct that initial condenser vacuum could be drawn without the use of the condenser vacuum pumps. It is correct that saturation temperature would rise. Admitting a initial amount of steam via the Steam Dumps would initially draw a vacuum due to the condensing action of the steam, but the air and non condensable gasses would build up and vacuum would degrade subsequently.

B. Incorrect: Again, it is not correct and yet plausible that initial condenser vacuum could be drawn without the vacuum pumps. This may be observed in section 5.2 of 1-SOI-2&3.01, Condensate and Feedwater System. It is impracticable to build a large condenser which is air tight. As such, some method must exist to remove air and other non-condensable gases which migrate into the condenser. Plants either utilize air ejectors or vacuum pumps for this purpose. Additionally, some method must exist to initially cause a vacuum to exist in the condenser. Again, this necessitates the use of either an air ejector or a vacuum pump. It is plausible to believe that the vacuum pumps are not required to establish condenser vacuum following a refueling outage because one may either believe that air ejectors are present and thus used for this purpose or one may not understand that the condensing action of steam (which maintains the vacuum at power) is the method by which initial vacuum is formed.

It is not correct that saturation temperature lowers. If the inevitably present non-condensable gases are not removed from the condenser, then the gas concentration present will increase. As this occurs, the heat transfer surface area of the condenser will decrease. This occurs because the gases tend to blanket the tubes of the condenser. A reduction of the surface area will cause the heat transfer to become reduced and thus the vacuum of the condenser to lower. Because of the relationship of temperature and pressure within a saturated system, the saturation temperature at which the steam will condense increases. It is plausible to believe that the saturation temperature lowers because one may believe that saturation temperature varies as does the saturation pressure (i.e. that as one lowers the other does as well).

C. Correct: It is correct that initial condenser vacuum could not be achieved without the CVPs. It is correct that saturation temperature would rise.

D. Incorrect: It is correct that initial condenser vacuum could not be achieved without the CVPs. It is not correct and yet plausible that saturation temperature would lower.

Question Number: 64

Tier: 2 Group: 2

K/A: 055 Condenser Air Removal System (CARS)
K3 Knowledge of the effect that a loss or malfunction of the CARS will have on the following:
K3.01 Main condenser

Importance Rating: 2.5 2.7

10 CFR Part 55: (CFR: 41.7 / 45.6)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to understand the effect that a loss of the condenser vacuum pumps (the CARS at WBN) would have on both the initial production of main condenser vacuum and on the saturation temperature of the main condenser when the Unit is at power.

Technical Reference: 1-SOI-2&3.01, Condensate and Feedwater System
Generic article regarding the non-condensable gases in a condenser.

Proposed references to be provided: None

Learning Objective: 3-OT-AOI1100, Loss of Condenser Vacuum
3. IDENTIFY Alarms, symptoms, automatic actuations, and other indications of AOI-11, Loss of Condenser Vacuum

Cognitive Level:

Higher ____
Lower X

Question Source:

New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Date________ **INITIALS**

5.2 Establishing Condenser Vacuum

NOTE

Since the Condenser Vacuum Exhaust (CVE) effluent must be monitored for a SG tube leak, and the CVE monitors must remain isolated until vacuum is achieved to avoid water damage, the following comp measures must be taken per guidance in System Description N3-2-4002, Section 4.10:

- SG blowdown rad monitors should be placed in service.
- Sampling provisions of the ODCM will be used to assess radioactive effluents out of the CVE.

[1] **ENSURE** Section 5.1, Short Cycle Operation, COMPLETE. _________

[2] **ENSURE** Atmospheric Condensate Drain Tank is available to receive water per 1-SOI-5&6.01, **THEN**

PLACE Injection Water in service to the Condensate Booster Pumps (if ready to start) and Main Feedwater Pumps per applicable steps of 1-SOI-54.01, if **NOT** already in service. _________

[3] **ENSURE** 1-RE-90-119, COND VAC PMP AIR EXH RAD MON, is **NOT** in service. _________

[4] **PLACE** Turbine on Turning Gear per 1-SOI-47.01. _________

[5] **ESTABLISH** Steam Seals per 1-SOI-47.03. _________

[6] **ESTABLISH** Gland Seal Water to Boot Seal and Vacuum Breaker per 1-SOI-37.01. _________

[7] **ENSURE** water level in each Condenser Vacuum Pump (CVP) reservoir is approx. 50% [T3H/685]:

A. 1-LG-2-168, COND VAC PMP A SEAL WATER LEVEL. _________

B. 1-LG-2-173, COND VAC PMP B SEAL WATER LEVEL. _________

C. 1-LG-2-178, COND VAC PMP C SEAL WATER LEVEL. _________

[8] **PLACE** 1-HS-30-883, CONDENSER VACUUM PUMP AREA COOLER [T3F/685], in AUTO. _________

Date________ **INITIALS**

5.2 Establishing Condenser Vacuum (continued)

[9] **OPEN** 1-FCV-2-255, with 1-HS-2-255, EXH BYPASS COND VAC PMPS [1-M-3]. _________

[10] **OPEN** the following:

A. 1-ISV-24-589, COND VACUUM PUMP CLR 1C RCW INLET ISOL [T3F/685]. _________

B. 1-ISV-24-593, CVP SEAL WTR COOLER 1A RCW INLET ISOL [T3G/685]. _________

C. 1-ISV-24-591, CVP SEAL WTR COOLER 1B RCW INLET ISOL [T3G/685]. _________

NOTE

Venting seal water pump is necessary if the pump and/or seal water tank has been drained.

[11] **IF** necessary, **THEN**

VENT CVPs' seal water pump(s) by performing the following: (**N/A** substeps if venting **NOT** performed)

[11.1] **REMOVE** cap from the following valve(s):

- 1-VTV-2-1045, COND VACUUM WATER PMP A VENT. _________
- 1-VTV-2-1046, COND VACUUM WATER PMP B VENT. _________
- 1-VTV-2-1047, COND VACUUM WATER PMP C VENT. _________

[11.2] **ENSURE** water level in seal water tank is above level of vent valve. _________

Date________ INITIALS

5.2 Establishing Condenser Vacuum (continued)

[11.3] **OPEN** to vent trapped air from pump, **THEN**

CLOSE the following valve(s):

- 1-VTV-2-1045. _________
- 1-VTV-2-1046. _________
- 1-VTV-2-1047. _________

[11.4] **REINSTALL** cap on the following valve(s):

- 1-VTV-2-1045. _________
- 1-VTV-2-1046. _________
- 1-VTV-2-1047. _________

[12] **PERFORM** the following:

NOMENCLATURE	LOCATION	POSITION	UNID	PERF INITIALS
COND VACUUM PMP A CONTROLS	T2F/680.5	RESET	1-HS-2-171B	
COND VACUUM PMP B CONTROLS	T2F/680.5	RESET	1-HS-2-176B	
COND VACUUM PMP C CONTROLS	T2F/680.5	RESET	1-HS-2-181B	

[13] **START** CVP(s) as necessary [1-M-3]:

A. 1-HS-2-171A, COND VACUUM PMP A. _________

B. 1-HS-2-176A, COND VACUUM PMP B. _________

C. 1-HS-2-181A, COND VACUUM PMP C. _________

[14] **ENSURE** 1-FCV-6-330, VACUUM BREAKER COND A, CLOSED. _________

Date________ **INITIALS**

5.2 Establishing Condenser Vacuum (continued)

[15] **WHEN** vacuum is established (C-9 permissive has been acknowledged), **THEN**

PLACE 1-RE-90-119, COND VAC PMP AIR EXH RAD MON, in service per 0-SOI-90.02. ________

[16] **ENSURE** Chemistry notified when condenser vacuum exhaust system and/or radiation monitors returned to service. ________

[17] **IF** additional CVPs are needed to maintain backpressure 0 to 5 in. Hg on 1-P/TR-2-2 [1-M-3], **THEN**

START CVPs as necessary:

A. 1-HS-2-171A, COND VACUUM PMP A. ________

B. 1-HS-2-176A, COND VACUUM PMP B. ________

C. 1-HS-2-181A, COND VACUUM PMP C. ________

Date________ **INITIALS**

5.2 Establishing Condenser Vacuum (continued)

NOTE
1) Ammonia is injected into the Condensate System for pH control.
2) Ammonia injection rate is higher during unit startup; therefore maximizing Seal Water Tank turnover is necessary. Throttling open drain valves in the following step helps reduce ammonia concentration in the general area by raising the turnover rate of the Seal Water Tank volume, which allows ammonia to exit via drain while still entrained in the condensate.

[18] **ESTABLISH** continuous drain for the in-service pump(s) to reduce ammonia concentration by throttling the drain valve to a position which maximizes drain flow within the capabilities of the drain system and the auto makeup:

A. 1-DRV-37-572, GLAND SEAL WATER DRAIN - PUMP 1A. _________

B. 1-DRV-37-571, GLAND SEAL WATER DRAIN - PUMP 1B. _________

C. 1-DRV-37-565, GLAND SEAL WATER DRAIN - PUMP 1C. _________

CAUTION
CVP Discharge Reliefs open if 1-FCV-2-255 is closed before condenser pressure reaches 5 in. Hg.

[19] **WHEN** 1-P/TR-2-2, COND PRESS & TEMPERATURE, is 0 to 5 in. Hg, **THEN**

CLOSE 1-FCV-2-255 with 1-HS-2-255, EXH BYPASS COND VAC PMPS [1-M-3]. _________

Date________ **INITIALS**

5.2 Establishing Condenser Vacuum (continued)

CAUTION
Max Hotwell and CST temp is 140°F.

[20] **IF** Sparging Steam is required, **THEN**

THROTTLE 1-PFV-12-513, CONDENSER SPARGING SUPPLY [T8F/708], to maintain less than 140° on 1-P/TR-2-2, COND PRESS & TEMPERATURE. _________

[21] **IF** Startup filters are required, **THEN**

PLACE Startup filters in service per 1-SOI-2.02. _________

[22] **IF** Condensate Polishers are required, **THEN**

PLACE Condensate Polishers in service per 1-SOI-14.01. _________

[23] **VERIFY** Condensate is within chemical limits. _________

[24] **IF** MFPT Condenser Drain Pumps are required, **THEN**

PLACE in service per 1-SOI-5 & 6.01. _________

[25] **WHEN** main condenser vacuum is established and stable, **THEN**

STOP one of the following Condenser Vacuum Pumps and place in P-AUTO: (Pumps **NOT** selected may be **N/A'd**)

A. 1-HS-2-171A, COND VACUUM PMP A. _________

B. 1-HS-2-176A, COND VACUUM PMP B. _________

C. 1-HS-2-181A, COND VACUUM PMP C. _________

Date________ **INITIALS**

5.2 Establishing Condenser Vacuum (continued)

[26] **ISOLATE** continuous drain for any pumps STOPPED:

A. 1-DRV-37-572, GLAND SEAL WATER DRAIN - PUMP 1A. _________

B. 1-DRV-37-571, GLAND SEAL WATER DRAIN - PUMP 1B. _________

C. 1-DRV-37-565, GLAND SEAL WATER DRAIN - PUMP 1C. _________

End of Section

Main Categories

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Large Steam System Condensers

Heat Exchanger Knowledge | Heat Exchanger Companies Suppliers

The steam condenser, shown in Figure 9 below, is a major component of the steam cycle in power generation facilities. It is a closed space into which the steam exits the turbine and is forced to give up its latent heat of vaporization. It is a necessary component of the steam cycle for two reasons. One, it converts the used steam back into water for return to the steam generator or boiler as feedwater. This lowers the operational cost of the plant by allowing the clean and treated condensate to be reused, and it is far easier to pump a liquid than steam. Two, it increases the cycle's efficiency by allowing the cycle to operate with the largest possible delta- T and delta-P between the source (boiler) and the heat sink (condenser).

Because condensation is taking place, the term latent heat of condensation is used instead of latent heat of vaporization. The steam's latent heat of condensation is passed to the water flowing through the tubes of the condenser.

After the steam condenses, the saturated liquid continues to transfer heat to the cooling water as it falls to the bottom of the condenser, or *hotwell*. This is called subcooling, and a certain amount is desirable. A few degrees subcooling prevents condensate pump cavitation. The difference between the saturation temperature for the existing condenser vacuum and the temperature of the condensate is termed *condensate depression*. This is expressed as a number of degrees condensate depression or degrees subcooled. Excessive condensate depression decreases the operating efficiency of the plant because the subcooled condensate must be reheated in the boiler, which in turn requires more heat from the reactor, fossil fuel, or other heat source.

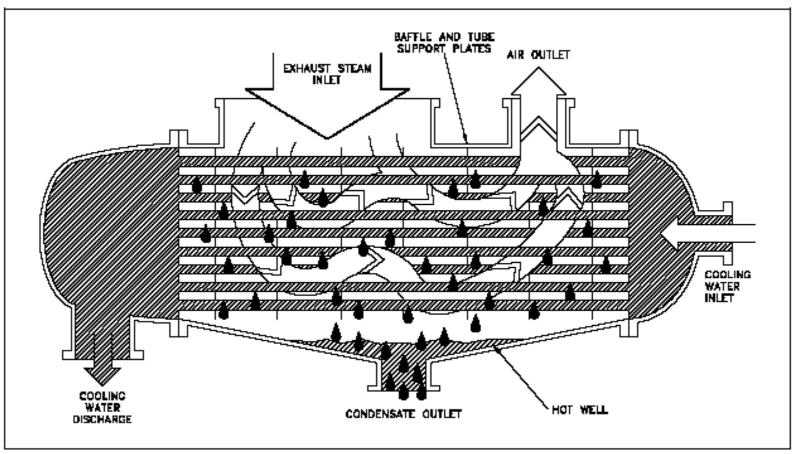


Figure 9 Single-Pass Condenser

Discussion of non-condensible gases.

There are different condenser designs, but the most common, at least in the large power generation facilities, is the straight-through, single-pass condenser illustrated Figure 9 above. This condenser design provides cooling water flow through straight tubes from the inlet water box on one end, to the outlet water box on the other end. The cooling water flows once through the condenser and is termed a single pass. The separation between the water box areas and the steam condensing area is accomplished by a tube sheet to which the cooling water tubes are attached. The cooling water tubes are supported within the condenser by the tube support sheets. Condensers normally have a series of baffles that redirect the steam to minimize direct impingement on the cooling water tubes. The bottom area of the condenser is the hotwell. This is where the condensate collects and the condensate pump takes its suction. If noncondensable gasses are allowed to build up in the condenser, vacuum will decrease and the saturation temperature at which the steam will condense increases.

Non-condensable gasses also blanket the tubes of the condenser, thus reducing the heat transfer surface area of the condenser. This surface area can also be reduced if the condensate level is allowed to rise over the lower tubes of the condenser. A reduction in the heat transfer surface has the same effect as a reduction in cooling water flow. If the condenser is operating near its design capacity, a reduction in the effective surface area results in difficulty maintaining condenser vacuum.

The temperature and flow rate of the cooling water through the condenser controls the temperature of the condensate. This in turn controls the saturation pressure (vacuum) of the condenser. To prevent the condensate level from rising to the lower tubes of the condenser, a hotwell level control system may be employed. Varying the flow of the condensate pumps is one method used to accomplish hotwell level control. A level sensing network controls the condensate pump speed or pump discharge flow control valve position. Another method employs an overflow system that spills water from the hotwell when a high level is reached.

Condenser vacuum should be maintained as close to 29 inches Hg as practical. This allows maximum expansion of the steam, and therefore, the maximum work. If the condenser were perfectly air-tight (no air or noncondensable gasses present in the exhaust steam), it would be necessary only to condense the steam and remove the condensate to create and maintain a vacuum. The sudden reduction in steam volume, as it condenses, would maintain the vacuum. Pumping the water from the condenser as fast as it is formed would maintain the vacuum. It is, however, impossible to prevent the entrance of air and other noncondensable gasses into the condenser. In addition, some method must exist to initially cause a vacuum to exist in the condenser. This necessitates the use of an air ejector or vacuum pump to establish and help maintain condenser vacuum.

Air ejectors are essentially jet pumps or eductors, as illustrated in Figure 10 below. In operation, the jet pump has two types of fluids. They are the high pressure fluid that flows through the nozzle, and the fluid being pumped which flows around the nozzle into the throat of the diffuser. The high velocity fluid enters the diffuser where its molecules strike other molecules. These molecules are in turn carried along with the high velocity fluid out of the diffuser creating a low pressure area around the mouth of the nozzle. This process is called entrainment. The lowpressure area will draw more fluid from around the nozzle into the throat of the diffuser. As the fluid moves down the diffuser, the increasing area converts the velocity back to pressure. Use of steam at a pressure between 200 psi and 300 psi as the high pressure fluid enables a singlestage air ejector to draw a vacuum of about 26 inches Hg.

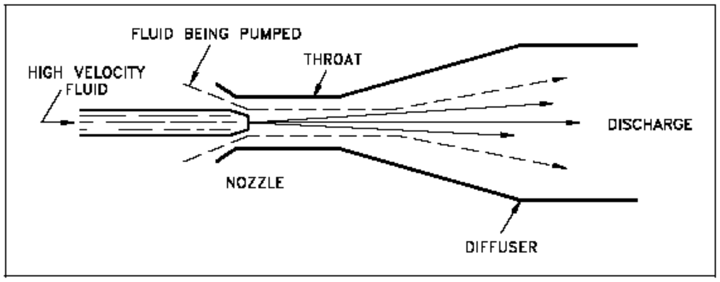


Figure 10 Jet Pump

Normally, air ejectors consist of two suction stages. The first stage suction is located on top of the condenser, while the second stage suction comes from the diffuser of the first stage. The exhaust steam from the second stage must be condensed. This is normally accomplished by an air ejector condenser that is cooled by condensate. The air ejector condenser also preheats the condensate returning to the boiler. Two-stage air ejectors are capable of drawing vacuums to 29 inches Hg.

A vacuum pump may be any type of motor-driven air compressor. Its suction is attached to the condenser, and it discharges to the atmosphere. A common type uses rotating vanes in an elliptical housing. Single-stage, rotary-vane units are used for vacuums to 28 inches Hg. Two stage units can draw vacuums to 29.7 inches Hg. The vacuum pump has an advantage over the air ejector in that it requires no source of steam for its operation. They are normally used as the initial source of vacuum for condenser start-up.

65.

Given the following conditions:

- Unit 1 is in MODE 3.
- A Loss of Control Air has occurred.
- Control Air header pressure is 60 psig and LOWERING ⇓.
- Power is available to the A Station Air compressor.
- Unit 2 Construction Air is available.

Which ONE of the following describes the response of the AUO?

In accordance with 0-AOI-10, section 3.3, an AUO will be dispatched to ____(1)____.

After the AUO completes this task and control air header pressure is restored, 0-PCV-33-4, Service Air Isol Valve ____(2)____ AUTOMATICALLY RE-OPEN.

NOTE: *0-AOI-10, Loss of Control Air*
Section 3.3, Loss of Control Air in Mode 1,2,3 or 4

A. (1) START the ‘A’ Station Air compressor
(2) WILL

B. (1) START the ‘A’ Station Air compressor
(2) WILL **NOT**

C. (1) ALIGN Unit 2 Construction Air Compressor to Unit 1 Control Air header
(2) WILL

D. (1) ALIGN Unit 2 Construction Air Compressor to Unit 1 Control Air header
(2) WILL **NOT**

<u>CORRECT ANSWER:</u> <u>B</u>

DISTRACTOR ANALYSIS:

A. Incorrect: Section 3.3 of 0-AOI-10, Loss of Control Air addresses a loss of control air in mode 1, 2, 3 or 4. Step 3. of this section directs: **START** C&SS Compressors locally as required USING Attachment 1, Local Restart of the C&SS Compressors. Investigation of Attachment 1 will reveal that it provides direction to start the A and/or B station air compressor and alerts the operator that If 480V Aux Bldg Common Bd is energized, Compressors C will also be available and will start if there is a demand. It does not provide any guidance for starting the D station air compressor. Print 1-45W600-32-1 shows the circuitry for 0-PCV-33-4. Contained on this print is PS-33-4 which opens on a decreasing pressure (nominally set to 80 psig). When this pressure switch opens, the solenoid valve which admits control air to 0-PCV-33-4 de-energizes and thus causes 0-PCV-33-4 to shut. This action isolates control air from service air. PS-33-4 requires a manual reset; this is seen on print 1-45W600-32-1. It is plausible to believe that it does not as numerous valves in the plant will close and open automatically.

B. Correct: It is correct that the A station air compressor would be started. It is also correct that 0-PCV-33-4 would require a manual reset to cause it to reopen. This manual reset would be conducted by an AUO (the reset is local to the valve’s control panel).

C. Incorrect: It is not correct that Unit 2 construction air would be aligned to restore the Unit 1 control air header pressure. It is plausible to believe this because 0-SOI-32.01, Control Air System contains section 8.4, Manual Alignment of Unit 2 Construction Air Compressor to Unit 1 Control / Service Air Header. This section directs the AUO to open 1 valve to restore control air header pressure. Before the replacement of the reciprocating air compressors (the old A,B and C) which occurred this cycle, any time that the D air compressor was tagged out for maintenance, the on-shift crews would direct AUOs to walk down (i.e. locate the 2-ISV-33-543 valve) such that they could restore control air pressure quickly. Therefore, because physically, the quickest was to restore air header pressure is to cross connect Unit 2 construction air to Unit 1 it is plausible that this would be directed by the AOI-10. However, 0-AOI-10 does not make allowance for using this method. Again, it is plausible but Incorrect to believe that 0-PCV-33-4 would automatically reopen.

D. Incorrect: It is Incorrect and yet plausible that Unit 2 construction air would be aligned to restore the Unit 1 control air header pressure. It is correct 0-PCV-33-4 would not automatically reopen.

Question Number: 65

Tier: 2 Group: 2

K/A: 079 Station Air System
G 2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Importance Rating: 3.8 4.0

10 CFR Part 55: (CFR: 41.10 / 43.5 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to possess the knowledge that an AUO would start the A station air compressor and that as a result of his action the service air isolation valve would not automatically reopen..

Technical Reference: 0-AOI-10, Loss of Control Air
0-SOI-32.01, Control Air System
1-45W600-32-1

Proposed references to be provided: None

Learning Objective: 3-OT-AOI1000, Loss of Control Air
5. DESCRIBE the reasons and applicable conditions for the notes,
cautions, and major actions of AOI-10, Loss of Control Air. (IER 11-3, Operating the plant with a conservative bias)

Cognitive Level:
Higher ____
Lower X

Question Source:
New X
Modified Bank ____
Bank ____

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Step	Action/Expected Response	Response Not Obtained

3.3 Loss of Control Air in Mode 1, 2, 3 or 4

Step	Action/Expected Response	Response Not Obtained
1.	**DISPATCH** personnel to check C&SS Compressors.	
2.	**CHECK** 480V power available to at least 2 C&SS compressors.	**REFER TO** 1-AOI-35, Loss of Offsite Power.
3.	**START** C&SS Compressors locally as required USING Attachment 1, Local Restart of the C&SS Compressors.	
4.	Locally **CHECK** air dryers and filters operating properly with 2 stations in service: • Regeneration time is 3 minutes for A & B Dryers only. • Filter diff press less than 5 psid. • One tower per station is drying.	**OPEN** the following to bypass any filter which is malfunctioning: a. 0-BYV-32-233, Station A Prefilter Bypass. b. 0-BYV-32-228, Station A Afterfilter Bypass. c. 0-BYV-32-222, Station B Prefilter Bypass. d. 0-BYV-32-357, Station B Afterfilter Bypass. e. 0-BYV-32-199, Station C Prefilter Bypass. f. 0-BYV-32-207, Station C Afterfilter Bypass.
5.	**ENSURE** Emergency Purge Shutoff valve not closed as indicated on Dryer Panel for C Dyer only.	**ENSURE** remaining dryers are in service and remove dryer from service.
6.	**DISPATCH** personnel to identify and isolate air leak.	
7.	**REFER TO** Appendix A for list of potentially inoperable valves and equipment.	

Attachment 1
(Page 1 of 2)

Local Restart of C&SS Air Compressors

Step	Action/Expected Response	Response Not Obtained

NOTE 1 Continuous contact between UO and NAUO should be maintained to aid quick recovery of air compressors.

NOTE 2 Attachment 1 is posted on panel 0-L-240 near the station air compressors. The posting must be revised if this Attachment is revised. Changes in revision level or page number will **NOT** require re-posting, as long as the content does **NOT** change

Step	Action/Expected Response	Response Not Obtained
1.	Locally **CHECK** 0-PCV-33-4, SERVICE AIR SUPPLY ISOLATION, CLOSED [T7M/708].	**CLOSE** 0-PCV-33-4 using 0-HS-33-4.

NOTE A local audible alarm will sound when 0-HS-32-25E or 0-HS-32-26B is pressed in the following steps.

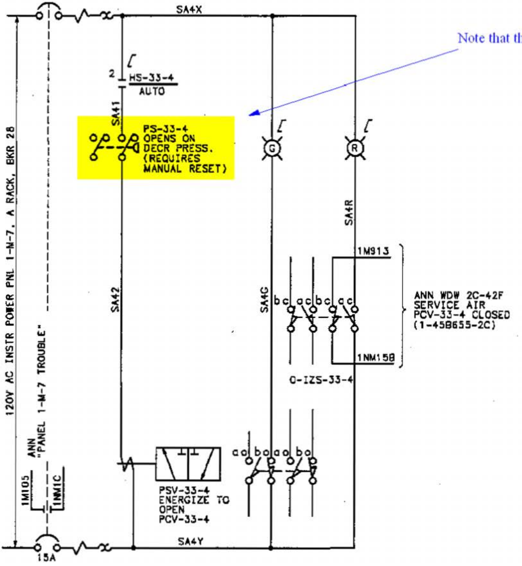
Step	Action/Expected Response	Response Not Obtained
2.	**PRESS** 0-HS-32-25E to restore power to Compressor A.	
3.	**PRESS** 0-HS-32-26B to restore power to Compressor B.	
4.	**ENSURE** 0-HS-32-25D is in the CLOSE position.	**PLACE** 0-HS-32-25D in CLOSE.

Attachment 1
(Page 2 of 2)

Local Restart of C&SS Air Compressors

Step	Action/Expected Response	Response Not Obtained
5.	**ENSURE** 0-HS-32-26A is in the CLOSE position.	**PLACE** 0-HS-32-26A in CLOSE.
6.	**ENSURE** A and/or B Station Air Compressor is running and supplying air pressure.	**IF** neither Air Compressor is running, **ENSURE** Header pressure is <setpoint on 0-CNTL-32-25 and/or -26.
7.	**MONITOR** Compressor operation: • Oil press >34 psig. • Compressors auto-loading. **NOTE** If 480V Aux Bldg Common Bd is energized, Compressors C will also be available and will start if there is a demand.	
8.	**CHECK** 480V Aux Bldg Common Bd ENERGIZED.	** **GO TO** Step 10.
9.	**ENSURE** 0-HS-32-27A is in the CLOSE position.	**PLACE** 0-HS-32-27A in CLOSE.
10.	**RETURN** TO Instruction in effect.	

Note that the pressure switch requires a manual reset



0-PCV-33-4
SERVICE AIR ISOLATION VLV

CONTACTS	POSITION (FRONT VIEW)

Date________ **INITIALS**

8.4 Manual Alignment of Unit 2 Construction Air Compressor to Unit 1 Control/ Service Air Header

CAUTION

Performance of this section should only be performed with concurrence of US/SM with a degraded or in anticipation of a degrading control air header pressure.

NOTE

2-ISV-33-543 will require an extension ladder to access it. 2-ISV-33-509 will require a scaffold to be erected.

[1] **UNLOCK** and **OPEN** one of the following:

- 2-ISV-33-543 SERVICE AIR TURBINE BLDG SUPPLY HDR ISOL.[T9D/708] _________

OR

- 2-ISV-33-509 SERVICE AIR SUPPLY HDR ISOL. [T9K/708] _________

[2] **IF** 0-PCV-33-4 SERVICE AIR SUPPLY ISOLATION is NOT OPEN, **THEN**

THROTTLE 0-ISV-33-502 SERVICE AIR 0-PCV-33-4 BYPASS OPEN. [T7M/708] _________

A. **ENSURE** 0-HS-33-4, SERVICE AIR SUPPLY ISOLATION [T7M/708], in AUTO. _________

B. **WHEN** Service Air pressure equals Control Air pressure, **THEN**

RESET 0-PS-33-4, SERVICE AIR SUPPLY ISOLATION PRESSURE [T7M/708]. _________

Date________ **INITIALS**

8.4 Manual Alignment of Unit 2 Construction Air Compressor to Unit 1 Control/ Service Air Header (continued)

NOTE

0-PCV-33-4, Service Air Supply Isolation from Control Air, will take from 5 to 10 minutes to travel from full closed to full open position.

C. **ENSURE** ANNUNCIATOR WINDOW 42-F [1-M-2], SERVICE AIR PCV-33-4 CLOSED, **NOT** LIT. ________

D. **CLOSE** 0-ISV-33-502, SERVICE AIR 0-PCV-33-4 BYPASS [T7M/708]. ________

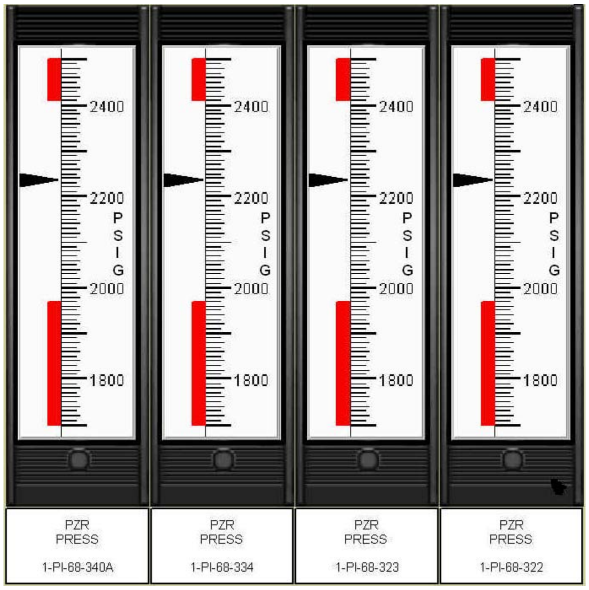
[3] **CONTACT** Unit 2 Operations and notify them that Unit 1 is receiving air from Unit 2 construction air compressors. ________

End of Section

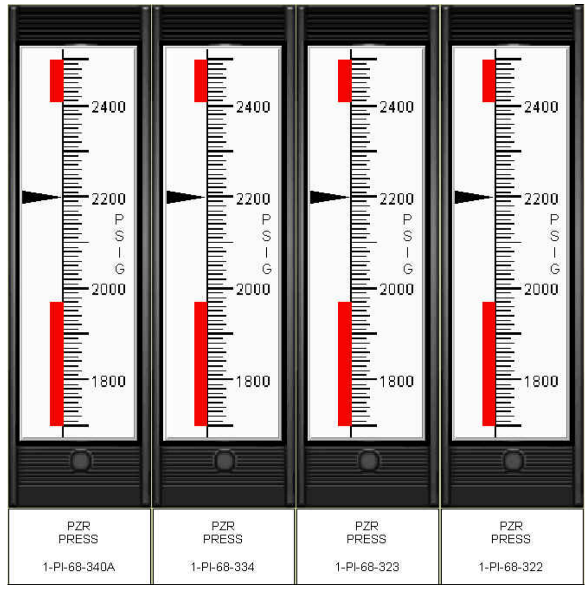
66.

Given the following conditions:

00:00:00 Unit 1 is at 100%.



Observed at 00:00:00



Observed at 00:00:05

In accordance with OPDP-1, Conduct of Operations, which ONE of the following completes the **MINIMUM** standard for the OAC's reporting the information depicted above to the operating crew?

Announce: "Crew update."

____(1)____ to ensure **ALL** crew members are listening.

Announce: "Pressurizer pressure is 2200 ____(2)____."

Announce: "End of update"

A. (1) Wait
(2) psig

B. (1) Wait
(2) psig and LOWERING ⇓

C. (1) DO **NOT** wait
(2) psig

D. (1) DO **NOT** wait
(2) psig and LOWERING ⇓

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: OPDP-1 defines the conduct of operations at WBNP. It contains section 5.3 which addresses various forms of briefings used by the control room staff. Among these is the crew update. As seen in OPDP-1, If a crew member has information that should be know by the majority of crew members he/she passes this information in a concise manner by: Obtaining the crew's attention by stating crew update / crew announcement Ensuring crew members are listening through the use of raised hands or crew members stating ready. Concisely announcing the information and Stating end of update / end of announcement. Therefore, one may observe that the sender of the information (in this question the sender is the OAC) is required to wait to ensure that all of the members of the crew are ready before passing the information.

OPDP-1 also contains section 7.2 which discusses the following: Communication of indicator reading should be provided in the format of PARAMETER – VALUE – UNITS – TREND (with rate when appropriate). Therefore, in accordance with OPDP-1, the minimum standard is not met by simply stating that pressure is 2200 psig. It is plausible to believe that this would be the MINIMUM standard because of the fact that one is also required to Concisely announce the information to be passed. Also, it is very well shown in the written simulator comments from both Licensed operator requalification training and Initial License training that the provision of trend is not uniformly made. A classic argument for the concept that trend would not be required in this case would be that stating that pressure is 2100 psig is sufficient because this is a standard trigger value used to initiate a manual reactor trip and it does not matter what the trend is if while at 100% power, pressurizer pressure is noted to be 2100 psig. It would be patently obvious to any licensed operator that pressures were LOWERING⇓. It is plausible to believe that one would give the crew update and immediately announce to the Unit Supervisor: US, Tripping the Unit 1 reactor. However, in accordance with OPDP-1, this is not the case.

B. Correct: As discussed it is correct that one would wait to obtain the recognition of all of the other members of the crew. It is correct that the trend must be provided in accordance with OPDP-1.

C. Incorrect: As seen it is not correct that the information of the crew update can be passed before the remainder of the crew acknowledges such. This is continually a contentious issue both in LOR and ILT and in this case one may observe that waiting for all of the other crew members to announce ready or raise their hand may not allow the OAC time to insert a manual reactor trip (and safety injection) before an automatic action occurs. However, it is the requirement of OPDP-1 that this be done. It is very plausible that one not wait and simply pass the information because as previously discussed, the OAC would make the crew update and summarily notify the Unit Supervisor that he was tripping the Unit 1 reactor.

D. Incorrect: Again it is plausible but Incorrect that the OAC not wait before passing the information of the crew update. It is correct though that he provide value and trend.

Question Number: 66

Tier: 3 Group: ______

K/A: 2.1 Conduct of Operations
2.1.18 Ability to make accurate, clear, and concise logs, records, status boards, and reports.

Importance Rating: 3.6 3.8

10 CFR Part 55: (CFR: 41.10 / 45.12 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to make the most concise and accurate report (crew update) procedurally allowed by OPDP-1, Conduct of Operations.

Technical Reference: OPDP-1, Conduct of Operations

Proposed references to be provided: None

Learning Objective: 3-OT-SPP1000, OPDP-1, CONDUCT OF OPERATIONS and NPG-SPP-10.0
14. State the types of briefings and associated standards as described in OPDP-1.(IER 11-3 - Precisely Controlling Plant Evolutions)

Cognitive Level:
- Higher X
- Lower ______

Question Source:
- New X
- Modified Bank ______
- Bank ______

Question History: New question for the 2015-301 NRC RO Exam

Comments:

3.8.3 Briefings (continued)

3. Shift Turnover Brief

Shift briefs are conducted at the beginning and middle of each shift. Shift Management directs the briefs. Personnel participation is required; with the exception of control room monitoring that is expected to continue. (See Attachment 3 for Beginning of Shift Brief and Mid-Shift Brief).

- If shift activities do not facilitate 100% crew attendance, a crew member knowledgeable of the watch station status will provide turnover information on that watch station.

4. Event Based Brief

Event Based Briefs are conducted as determined by the Unit Supervisor utilizing a format similar to that contained in Attachment 4. Event based briefs ensure that key information and strategies are communicated in a concise manner and that crew members are given an opportunity to share critical information. Attachment 4 is not required to be filled out. It is a general guideline to follow and does not need to be in hand during the brief.

5. Crew Updates

During many conditions the communication of important information to the crew may not need a scripted brief. In these cases the "Crew Update" can be used. If a crew member has information that should be known by the majority of crew members he/she passes this information in a concise manner by:

Notice that one must ensure that crew members are listening (even if a rapidly developing transient is in progress in which immediate action is needed).

- Obtaining the crew's attention by stating "crew update" / "crew announcement"
- Ensuring crew members are listening through the use of raised hands or crew members stating "ready."
- Concisely announcing the information and
- Stating "end of update" / "end of announcement".

3.9 Understanding Plant Design And Interaction

3.9.1 Training and Qualification

A. Operations training and qualification programs help ensure the knowledge and skills needed by non-licensed operators Assistant Unit Operators (AUOs), licensed reactor operators (UOs), licensed Senior Reactor Operators (SROs), Shift Technical Advisors (STAs), and Shift Managers (SMs) to effectively perform routine operations activities in the plant and properly respond to and combat unusual incidents and to anticipate and prevent such events.

B. Operations managers and supervisors are directly involved in the training and qualification of Operations personnel. This involvement includes close coordination with the Training Department to:

3.10.2 Error Prevention Tools (continued)

Verif." Column.

7. Slash through the circled IV from the step just performed.

D. Communication

1. Three-Way Communication is used for plant equipment manipulations, procedure performance and plant /equipment parameter information that will result in decision making, direction being given or actions being taken. See NPG-SPP-22.202, Human Performance Tools.

- Ensure all communications are clear, concise, and free of ambiguity.
- For non-face-to-face verbal communication or to gain the individual's attention the sender identifies themselves by stating their name or title.
- Use equipment noun names when practical and the phonetic alphabet when communicating component, train, channel, or procedure step designators.
- Avoid words during verbal communication that could be mistaken for each other, like "increase' and "decrease."
- Communication of indicator readings should be provided in the format of PARAMETER – VALUE – UNITS – TREND (with rate when appropriate).
- Use the appropriate unit designator, system designator, or noun name and appropriate phonetic alphabet component or train designator when communicating equipment nomenclature.

  FOR EXAMPLE

  Letdown Isolation 1-FCV-62-69 should be verbalized as "Letdown Isolation one FCV sixty two sixty nine" or "Letdown Isolation one FCV sixty two tack sixty nine."

a. Sender Responsibilities (Direction)

- Address intended receiver by name or title
- Obtain the attention of the intended receiver and speak clearly
- Send the intended message in just enough words to minimize the chance of receiver misunderstanding
- Require confirmation from the intended receiver of the information by ensuring a repeat back

b. Receiver Responsibilities (Repeat Back/Answer)

- Repeat direction/request back to sender (paraphrasing is allowed).
- If responding to a request for information, respond with "yes" or "no"
- Maintain a questioning attitude, if you question or do not understand a person's direction then reconfirm with sender that instructions were correct as given.

3.10.2 Error Prevention Tools (continued)

- Take no action until message is clearly understood and confirmed.

c. Sender Responsibilities (Confirmation)

- If the receiver's repeat-back is satisfactory, reply in the affirmative.
- If unsatisfactory, reply in the negative and repeat the original communication.

EXAMPLE: THREE -WAY COMMUNICATIONS	
Sender (Request for Information)	
SRO:	"*Name or title*, verify Reactor Coolant pressure is 1000 psig and stable"
Receiver (Answer)	
UO:	" No. Pressure is 900 psig and slowly going down."
Sender (Acknowledge)	
SRO:	"Understand"

EXAMPLE: THREE -WAY COMMUNICATIONS	
Sender (Direction)	
SRO:	"*Name or title*, place RCIC SYSTEM FLOW TRANSFER SWITCH 3-XS-71-36 Bravo in the EMERGENCY position"
Receiver (Repeat Back)	
UO:	" Place 3-XS-71-36 Bravo in the EMERGENCY position"
Sender (Confirmation)	
SRO:	"That's correct."

2. Phone and Radio Communications

a. Phone Communications

- Phone communications utilize 3 Way communications as described in this procedure.
- Names or titles are used to ensure sender and receiver are understood.
- Formality and professional language are the standard for phone communications to ensure miscommunications do not occur.

b. Radio Communications

- Radio communications utilize 3-Way communications as described in this procedure.
- If reception is poor or radio usage is not allowed other methods of

67.

Given the following conditions:

00:00:00 Unit 1 is at 100% power.

00:01:00 Unit 1 has a Turbine runback to 80%.

At the times specified, **ALL** Excore channels indicate the following:

	Rated Thermal Power (%)	AFD (%)
00:01:05	95	(-) 3.0%
00:01:10	90	(-) 6.0%
00:01:15	85	(-) 15.5%
00:01:20	80	(-) 20.0%

Which ONE of the following identifies the **FIRST** time that T/S LCO 3.2.3, AFD will be **NOT MET**.

REFERENCE PROVIDED

A. 00:01:05

B. 00:01:10

C. 00:01:15

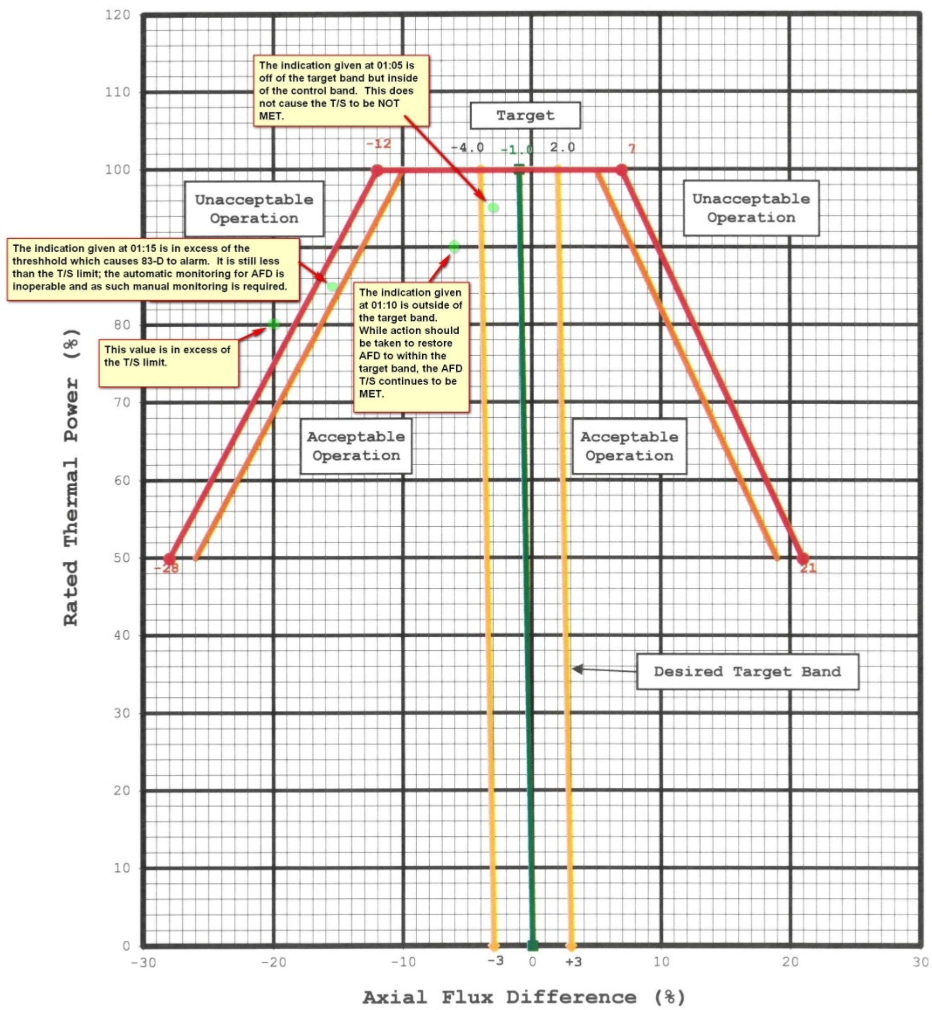
D. 00:01:20

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

WBN 1	NUCLEAR OPERATING BOOK (NOB)	NOB SHEET A-1 Revision 161 Page 1 of 1

Target Band vs. Power Level
WBN Unit 1 Cycle 13



CURRENT AFD AND TARGET BAND CONSTANTS
K2024X = -28%, K2025X = -12%, K2026X = +7%, K2027X = +21%,
K2028 = -1.0%, K2029 = -3.0, K2030 = 3.0

Question Number: 67

Tier: 3 Group: ______

K/A: G 2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.

Importance Rating: 3.9 4.2

10 CFR Part 55: (CFR: 41.10 / 43.5 / 45.12)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to use the AFD graph to determine whether or not the T/S is violated.

Technical Reference: Unit 1 Nuclear Operating Book (NOB)

Proposed references to be provided: Redacted page A-1 of the Nuclear Operating Book (NOB)

Learning Objective: 3-OT-TS0302 TS 3.2 “Power Distribution Tech Specs”
2. Given a copy of the Technical Specifications and the Technical Requirements Manual, DETERMINE the following for Power Distribution Technical Specification: (IER 11-3, Controlling Plant Evolutions Precisely, Monitor Plant Indications Closely)
a. Applicable Limiting Condition for Operation for a given plant condition.

Cognitive Level:

Higher X
Lower ______

Question Source:

New X
Modified Bank ______
Bank ______

Question History: New question for the 2015-301 NRC RO Exam

Comments:

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the COLR.

---NOTE---
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

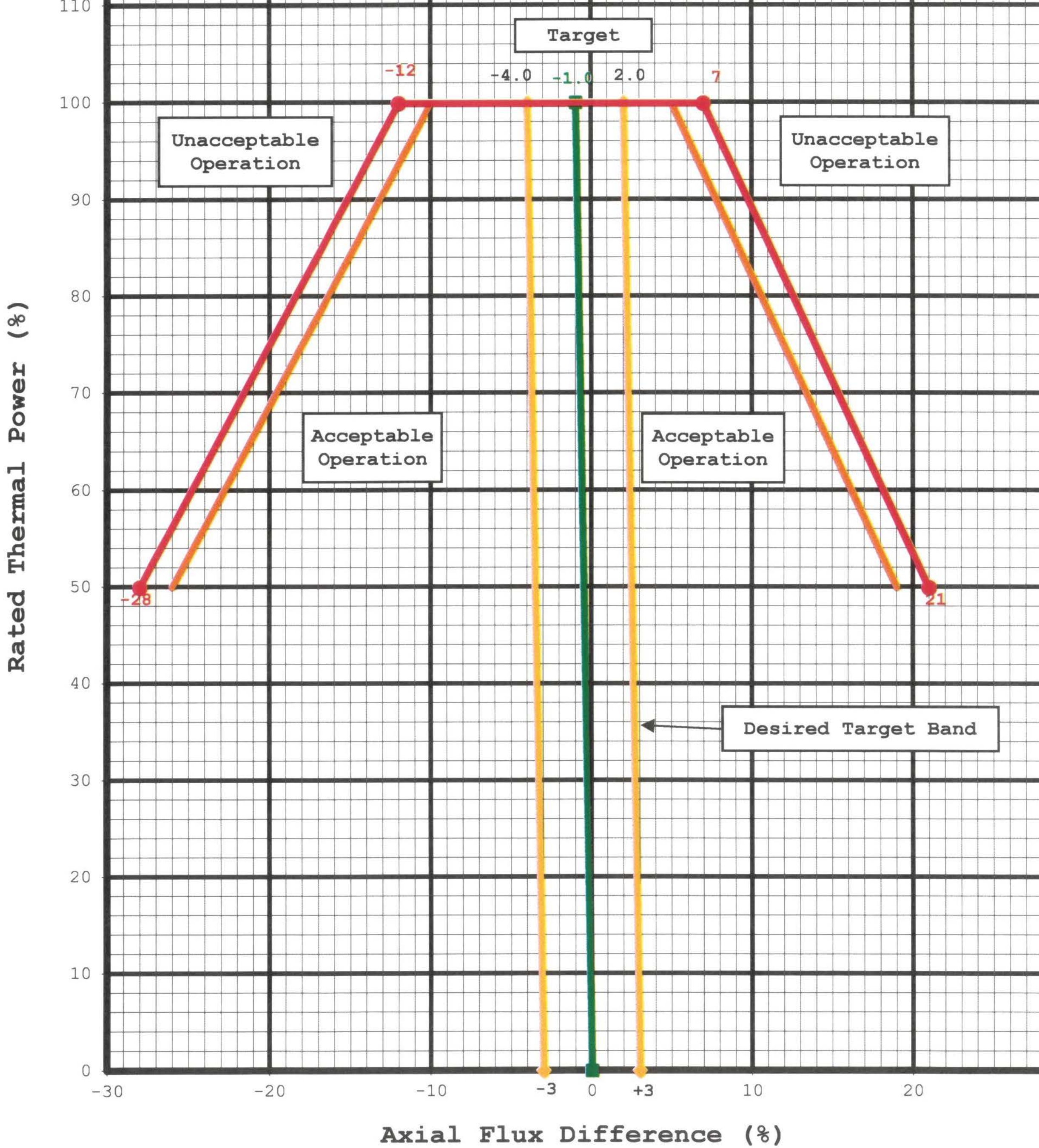
APPLICABILITY: MODE 1 with THERMAL POWER $\geq$ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

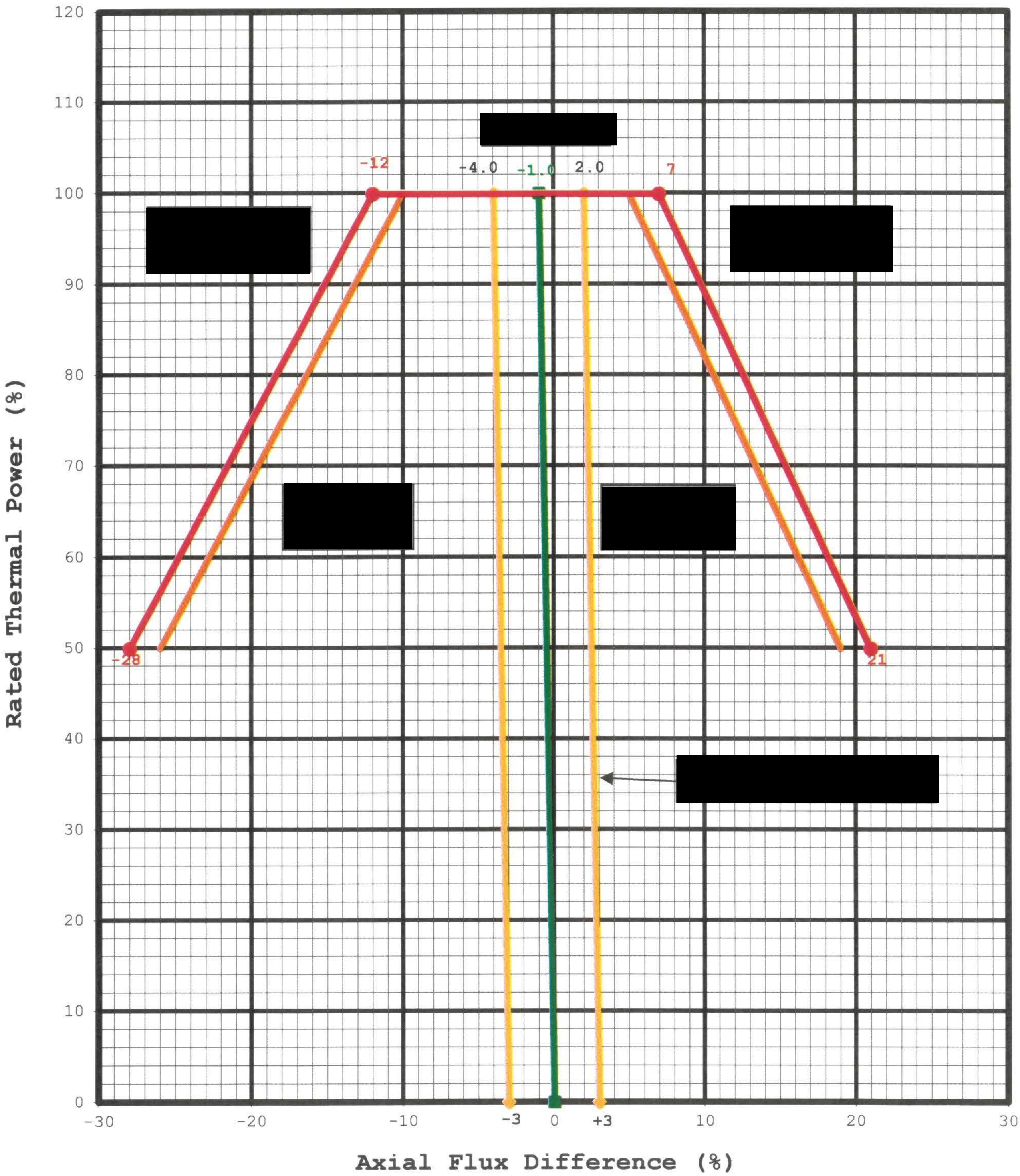
SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days <u>AND</u> Once within 1 hour and every 1 hour thereafter with the AFD monitor alarm inoperable



Target
-12
-4.0
-1.0
2.0
7
Unacceptable Operation
Unacceptable Operation
Acceptable Operation
Acceptable Operation
-28
21
Desired Target Band
Rated Thermal Power (%)
110
100
90
80
70
60
50
40
30
20
10
0
-30
-20
-10
-3
0
+3
10
20
Axial Flux Difference (%)

Target Band vs. Power Level

WBN Unit 1 Cycle 13



CURRENT AFD AND TARGET BAND CONSTANTS
K2024X = -28%, K2025X = -12%, K2026X = +7%, K2027X = +21%,
K2028 = -1.0%, K2029 = -3.0, K2030 = 3.0

68.

Which ONE of the following describes the MCR Log Taking in accordance with 1-SI-0-2B-01, 0700-0900, Shift and Daily Surveillance Log Mode One?

The Reactor Operator will obtain data available in the MCR from _____(1)_____.

This data will be compared to criteria which are _____(2)_____ that contained in the Unit 1 Technical Specifications.

A. (1) **ONLY** the MCR board indications
   (2) **ALWAYS** identical to

B. (1) **ONLY** the MCR board indications
   (2) **SOMETIMES** different than

C. (1) the MCR board indications and the ICS
   (2) **ALWAYS** identical to

D. (1) the MCR board indications and the ICS
   (2) **SOMETIMES** different than

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: It is not correct that data supporting the performance of 1-SI-0-2B-01 be solely obtained from the MCR board indications. In fact, this surveillance will specify that some points are preferentially obtained from the Integrated Computer System (ICS). Take, for example, CNTMT Lower Compartment Average Air Temperature. This is reference 63 in 1-SI-0-2B-01 and lists the calculated point U9020 as the source for data. U9020 is a data point which is produced by the ICS. It would be plausible to believe that one would only use the pedigreed indicators on the MCR panels for the completion of this surveillance as such is being used to satisfy the Unit's technical specifications. As the ICS is not safety related, it would be plausible to believe that it would not be used to survey the license requirements of the Unit. However, as seen in TI-49, Compliance Instruments, the ICS does provide sundry inputs into the various surveillance instructions of the plant. Because the ICS is used for such purposes, it is subject to the calibration requirements of NPG-SPP-06.7, Instrumentation Setpoint, Scaling And Calibration Program and as long as such requirements are met, may be used to satisfy the surveillance procedures. It is also not correct that the Technical Specification Operability Limits are always equal to the Technical Specification Surveillance Limits. While this may be the case at many other nuclear sites, it is not the case at WBNP. Two design output drawings 47W605-242 and 47W605-243 list those parameters who possess Surveillance Limits which differ from their Operability Limits. Take the Lower Containment Average Air Temperature for consideration again. T/S LCO 3.6.5, Containment Air Temperature states that: Containment average air temperature shall be: b. $\geq$ 100°F and $\leq$ 120°F for the containment lower compartment. Reference 63 of 1-SI-0-2B-01 lists that the T S Limit of the same parameter is $\geq$ 102.5°F and $\leq$ 117.5°F. Therefore, an average temperature of 119.5 would render the lower containment inoperable even though its temperature was inside of the band contained within the T/S LCO. The two drawings of the 47W605 series captured the commitment which WBNP made to incorporate instrument errors into the surveillance instructions. WBNP did not solely use the instrument errors already built into the standardized improved Technical Specifications. It is plausible to believe that the surveillance satisfying the Technical Specifications would use the requirements of the Technical Specifications as this would be common sense as well as the practice at many other nuclear plants.

B. Incorrect: Again, it is not correct and yet plausible that data supporting the performance of 1-SI-0-2B-01 be solely obtained from the MCR board indications. Also, it is correct that the acceptance criteria of the SI are not always the same as that contained in the Unit's technical specifications.

C. Incorrect: It is correct that the ICS will be used in concert with the MCR board indications to obtain data for 1-SI-0-2B-01. Also, it is not correct and yet plausible that the acceptance criteria of the SI is the same as that contained in the Unit's technical specifications.

D. Correct: It is correct that the ICS will be used in concert with the MCR board indications to obtain data for 1-SI-0-2B-01. Also, it is correct that the acceptance criteria of the SI are not always the same as that contained in the Unit's technical specifications.

Question Number: 68

Tier: 3 Group: ______

K/A: G 2.2.12 Knowledge of surveillance procedures.

Importance Rating: 3.7 4.1

10 CFR Part 55: (CFR: 41.10 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to have the fundamental (generic) knowledge of which sources of data are allowed to be used for surveillance procedures. 1-SI-0-2B-01 was selected for specificity and clarity but the concept is applicable to the surveillance procedures at WBN. The applicant is then required to understand the source of the acceptance criteria..

Technical Reference: 1-SI-0-2B-01
TI-49, Compliance Instruments
T/S LCO 3.6.5, Containment Air Temperature
47W605-242

Proposed references to be provided: None

Learning Objective: 3-OT-TS0000 Technical Specification Overview
12. EXPLAIN briefly the information that can be found in the Technical Specification and Technical Requirements "Surveillance Requirements" section.

Cognitive Level:

Higher	
Lower	X

Question Source:

New	X
Modified Bank	
Bank	

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Data Sheet 1
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Mode 1 Surveillance Log: 0700 - 1900

Date _________

Ref	Location	Description	Instrument #	T S Limit	Data	Acc NOT Met
60	ICS	SGBD flow to CTBD	F0619A OR 1-FIT-15-42	Operable and Channel Check	gpm	If inoperable, SRO is to be notified and Section 6.1L should be referenced. A flow reading from either the computer point or the local indicator will satisfy the channel check requirement.
	1-PNL-276 -L657 T8J/712				gpm	If either the computer point or the local indicator is inoperable, a SR is to be initiated and SR number recorded in remarks.
61	T8J/685 OR ICS	Station Sump Discharge Radiation Mon	0-RI-90-212A ☐ OR point R1010A ☐	Operable and Channel Check	cpm	If monitor is inoperable, notify SRO, and Chemistry Countroom per Section 6.1K.
62	S4Sn/741 0-L-399 OR ICS	SBVS Duct Monitor	Stack Volume 0-FI-90-320/1B ☐ OR point F2702A ☐	Operable and Channel Check	scfm(4)	If inoperable, SRO is to be notified and Section 6.0. L should be referenced. Perform Data Sheet 62 every 4 hours until declared Operable.
63	ICS	CNTMT Upper Compartment Average Air Temperature	U9019	≥87.1°F to ≤107.9°F(3)	°F(1)	If ICS is INOPERABLE, Data Sheet 63 must be performed.
		CNTMT Lower Compartment Average Air Temperature	U9020	≥102.5°F to ≤117.5°F(2)	°F(1)	

(1) Minimum number of operable channels is verified by computer point routines. Points will indicate unreliable if less than the required number of inputs are available. A print value review of points T1000A through T1033A can be performed to determine which points are invalid.

(2) Maximum Lower Compartment average air temperature can be raised to 119°F, and Minimum Lower Compartment average air temperature can be lowered to 101°F, if ALL of the temperature inputs and U9020 are reliable (NONE of the computer points have a quality code of "BAD"). The points can be displayed on computer Primary Mimic CONTAINMENT TEMPERATURES.

(3) Maximum Upper Compartment average air temperature can be raised to 108.5°F, and Minimum Upper Compartment average air temperature can be lowered to 86.5°F, if ALL of the temperature inputs and U9019 are reliable (NONE of the computer points have a quality code of "BAD"). The points can be displayed on computer Primary Mimic CONTAINMENT TEMPERATURES.

(4) Inoperable if display is ≤3,000 SCFM

	Initials indicate acc met OR action in "Acc not Met" column initiated.	Performer's Initials	SRO's Initials

Remarks: ____

____

Appendix B
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Compliance ICS Computer Point List

Date ________

This list consists of all ICS computer points used for compliance. The following column headings are used:

COMPUTER POINT - The ICS computer point used as a Compliance device by the listed Instruction.

LOOP/INSTRUMENT - The loop number associated with the ICS computer point.

INSTRUCTION - The number(s) of the Instruction where the ICS computer point is used.

NOTES

1) These Offsite Dose Instructions (ODIs) record data taken from their associated System Operating Instruction (SOI). The applicable computer points are listed in the SOI.

2) Devices which can **NOT** be calibrated. Some examples are: thermocouples which input directly to the plant computer; computer calculated points where the individual inputs to the calculation are calibrated under their specific loops; or computer points normalized by the computer such as Incore flux detectors.

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Compliance ICS Computer Point List

ICS COMPUTER POINT	LOOP /INSTRUMENT	INSTRUCTION	NOTES
F0125A	1-LPF-62-40	1-SI-68-33	0
F0127A	1-LPF-62-27	1-SI-68-33	0
F0129A	1-LPF-62-14	1-SI-68-33	0
F0131A	1-LPF-62-1	1-SI-68-33	0
F0400A	1-LPF-68-6A	1-SI-0-2	0
F0401A	1-LPF-68-6B	1-SI-0-2	0
F0402A	1-LPF-68-6D	1-SI-0-2	0
F0403A	1-LPF-3-35A	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-41	0
F0404A	1-LPF-3-35B	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-41	0
F0407A	1-LPF-3-235	1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
F0408A	1-LPF-1-152	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-1-905	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
F0420A	1-LPF-68-29A	1-SI-0-2	0
F0421A	1-LPF-68-29B	1-SI-0-2	0
F0422A	1-LPF-68-29D	1-SI-0-2	0
F0423A	1-LPF-3-48A	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
F0424A	1-LPF-3-48B	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
F0427A	1-LPF-3-238	1-SI-0-24	0
		1-SI-92-1	0

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Compliance ICS Computer Point List

ICS COMPUTER POINT	LOOP /INSTRUMENT	INSTRUCTION	NOTES
F0427A (cont)	1-LPF-3-238	TI-6	0
		TI-41	0
F0428A	1-LPF-1-156	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-1-905	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
F0440A	1-LPF-68-48A	1-SI-0-2	0
F0441A	1-LPF-68-48B	1-SI-0-2	0
F0442A	1-LPF-68-48D	1-SI-0-2	0
F0443A	1-LPF-3-90A	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
F0444A	1-LPF-3-90B	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
F0447A	1-LPF-3-241	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
F0448A	1-LPF-1-160	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-1-905	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
F0460A	1-LPF-68-71A	1-SI-0-2	0
F0461A	1-LPF-68-71B	1-SI-0-2	0
F0462A	1-LPF-68-71D	1-SI-0-2	0
F0463A	1-LPF-3-103A	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0

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Compliance ICS Computer Point List

ICS COMPUTER POINT	LOOP /INSTRUMENT	INSTRUCTION	NOTES
F0463A (cont)	1-LPF-3-103A	TI-6	0
		TI-41	0
F0464A	1-LPF-3-103B	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-68-31	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
F0467A	1-LPF-3-244	1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
F0468A	1-LPF-1-164	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-1-905	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
F0619A	1-FIT-015-42	1-ODI-90-2	0
		1-SI-0-2	0
F2702A	0-LPF-90-320	0-ODI-90-8	0
F2704A	0-LPF-90-300	0-ODI-90-22	0
F9015A	2-LPF-90-400	1-ODI-90-26	0
L0112A	1-LPL-62-130	1-SI-68-32	0
L0480A	1-LPL–68–339	1-SI-68-32	0
L0481A	1-LPL–68–335	1-SI-68-32	0
L0482A	1-LPL–68–320	1-SI-68-32	0
L0485A	1-LPL–68–300	1-SI-68-32	0
L1040A	1-LPL-63-50	1-SI-63-911	0
		1-SI-63-912	0
L1041A	1-LPL-63-51	1-SI-63-911	0
		1-SI-63-912	0
LEFMSTAT		1-SI-92-1	2
N0001A		1-SI-0-15	2
		1-SI-0-20	2
		1-SI-0-22	2
		1-SI-92-2	2
		1-SI-92-3	2
		1-SI-92-4	2
		TI-41	2
N0002A		1-SI-0-15	2
		1-SI-0-20	2
		1-SI-0-22	2
		1-SI-92-2	2
		1-SI-92-3	2
		1-SI-92-4	2
		TI-41	2

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Compliance ICS Computer Point List

N0003A		1-SI-0-15	2
		1-SI-0-20	2
		1-SI-0-22	2
		1-SI-92-2	2
		1-SI-92-3	2
		1-SI-92-4	2
		TI-41	2
N0004A		1-SI-0-15	2
		1-SI-0-20	2
		1-SI-0-22	2
		1-SI-92-2	2
		1-SI-92-3	2
		1-SI-92-4	2
		TI-41	2
N0005A		1-SI-0-15	2
		1-SI-0-20	2
		1-SI-0-22	2
		1-SI-92-2	2
		1-SI-92-3	2
		1-SI-92-4	2
		TI-41	2
N0006A		1-SI-0-15	2
		1-SI-0-20	2
		1-SI-0-22	2
		1-SI-92-2	2
		1-SI-92-3	2
		1-SI-92-4	2
		TI-41	2
N0041A	1-INM-92-1/301	1-SI-0-11	0
		1-SI-0-21	0
		1-SI-0-22	0
		1-SI-92-2	0
		1-SI-92-3	0
		1-SI-92-4	0
		TI-41	0
N0042A	1-INM-92-1/302	1-SI-0-11	0
		1-SI-0-21	0
		1-SI-0-22	0
		1-SI-92-2	0
		1-SI-92-3	0
		1-SI-92-4	0
		TI-41	0
N0043A	1-INM-92-2/301	1-SI-0-11	0
		1-SI-0-21	0
		1-SI-0-22	0
		1-SI-92-2	0
		1-SI-92-3	0
		1-SI-92-4	0

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Compliance ICS Computer Point List

ICS COMPUTER POINT	LOOP /INSTRUMENT	INSTRUCTION	NOTES
N0043A (cont)	1-INM-92-2/301	TI-41	0
N0044A	1-INM-92-2/302	1-SI-0-11	0
		1-SI-0-22	0
		1-SI-92-2	0
		1-SI-92-3	0
		1-SI-92-4	0
		TI-41	0
N0045A	1-INM-92-3/301	1-SI-0-11	0
		1-SI-0-21	0
		1-SI-0-22	0
		1-SI-92-2	0
		1-SI-92-3	0
		1-SI-92-4	0
		TI-41	0
N0046A	1-INM-92-3/302	1-SI-0-11	0
		1-SI-0-21	0
		1-SI-0-22	0
		1-SI-92-2	0
		1-SI-92-3	0
		1-SI-92-4	0
		TI-41	0
N0047A	1-INM-92-4/301	1-SI-0-11	0
		1-SI-0-21	0
		1-SI-0-22	0
		1-SI-92-2	0
		1-SI-92-3	0
		1-SI-92-4	0
		TI-41	0
N0048A	1-INM-92-4/302	1-SI-0-11	0
		1-SI-0-21	0
		1-SI-0-22	0
		1-SI-92-2	0
		1-SI-92-3	0
		1-SI-92-4	0
		TI-41	0
N0049A	1-INM-92-1/303	1-SI-0-10	0
		1-SI-0-11	0
		1-SI-0-20	0
		1-SI-0-24	0
		1-SI-0-28	0
		1-SI-92-1	0
		1-SI-92-2	0
		1-SI-92-3	0
		1-SI-92-4	0
		TI-41	0
N0050A	1-INM-92-2/303	1-SI-0-10	0
		1-SI-0-11	0

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ICS COMPUTER POINT	LOOP /INSTRUMENT	INSTRUCTION	NOTES
N0050A (cont)	1-INM-92-2/303	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-0-28	0
		1-SI-92-1	0
		1-SI-92-2	0
		1-SI-92-3	0
		1-SI-92-4	0
		TI-41	0
N0051A	1-INM-92-3/303	1-SI-0-10	0
		1-SI-0-11	0
		1-SI-0-20	0
		1-SI-0-24	0
		1-SI-0-28	0
		1-SI-92-1	0
		1-SI-92-2	0
		1-SI-92-3	0
		1-SI-92-4	0
		TI-41	0
N0052A	1-INM-92-4/303	1-SI-0-10	0
		1-SI-0-11	0
		1-SI-0-20	0
		1-SI-0-24	0
		1-SI-0-28	0
		1-SI-92-1	0
		1-SI-92-2	0
		1-SI-92-3	0
		1-SI-92-4	0
		TI-41	0
P0142A	1-LPP-62-92	1-SI-68-33	0
P0400A	1-LPP-1-2A	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-68-31	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0401A	1-LPP-1-2B	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-68-31	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0402A	1-LPP-1-5	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0

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ICS COMPUTER POINT	LOOP /INSTRUMENT	INSTRUCTION	NOTES
P0403A	1-LPP-3-37	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0420A	1-LPP-1-9A	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0421A	1-LPP-1-9B	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0422A	1-LPP-1-12	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0423A	1-LPP-3-50	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0440A	1-LPP-1-20A	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0441A	1-LPP-1-20B	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0442A	1-LPP-1-23	1-SI-0-20	0
		1-SI-0-24	0

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ICS COMPUTER POINT	LOOP /INSTRUMENT	INSTRUCTION	NOTES
P0442A (cont)	1-LPP-1-23	1-SI-92-1	0
		TI-6	0
		TI-41	0
P0443A	1-LPP-3-92	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0460A	1-LPP-1-27A	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0461A	1-LPP-1-27B	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0462A	1-LPP-1-30	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0463A	1-LPP-3-105	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
P0480A	1-LPP-68-340	1-SI-0-23	0
		1-SI-68-32	0
P0481A	1-LPP-68-334	1-SI-68-33	0
P0482A	1-LPP-68-323	1-SI-68-33	0
P0483A	1-LPP-68-322	1-SI-68-33	0
P0499A	1-LPP-68-66A	1-SI-68-44	0
P0701A	1-LPP-68-63	1-SI-68-44	0
P4005A	1-LPP-30-126	1-SI-0-2	0
P4006A	1-LPP-30-127	1-SI-0-2	0

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ICS COMPUTER POINT	LOOP /INSTRUMENT	INSTRUCTION	NOTES
P0702A	1-LPP-68-64	1-SI-68-44	0
Q0340A	1-ET-57-16A	1-SI-0-23	2
P1000A	1-LPD-30-42	1-SI-0-2	0
P1001A	1-LPD-30-43	1-SI-0-2	0
P1002A	1-LPD-30-44	1-SI-0-2	0
P1003A	1-LPD-30-45	1-SI-0-2	0
R0001A	1-LPR-90-119	1-ODI-90-25	0
R0011A	0-LPR-90-132B	0-ODI-90-8	0
R0016A	0-LPR-90-118	0-ODI-90-5	1
R0020A	0-LPR-90-101B	0-ODI-90-22	0
R1010A	0-LPR-90-212	0-ODI-90-3	0
R1011A	0-LPR-90-225	0-ODI-90-23	1
R1020A	1-LPR-90-120	1-ODI-90-2	1
R1021A	1-LPR-90-121	1-ODI-90-2	1
R1022A	1-LPR-90-122	0-ODI-90-1	1
R1027A	1-LPR-90-130	1-ODI-90-15	0
		1-ODI-90-18	0
R1028A	1-LPR-90-131	1-ODI-90-15	0
		1-ODI-90-18	0
R9101A	1-LPR-90-400	1-ODI-90-15	0
		1-ODI-90-18	0
		1-ODI-90-26	0
R9102A	2-LPR-90-400	1-ODI-90-26	0
T0400A	1-LPT-68-2	1-SI-0-10	0
		1-SI-0-11	0
		1-SI-0-23	0
		1-SI-0-27	0
		1-SI-68-32	0
T0403A	1-LPT-68-2	1-SI-0-23	0
T0406A	1-LPT-68-18	1-SI-0-10	0
T0418A	1-LPT-3-36	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
		TI-6.001	0
T0419A	1-LPT-68-1	1-SI-68-44	0
T0420A	1-LPT-68-25	1-SI-0-10	0
		1-SI-0-11	0
		1-SI-0-23	0
		1-SI-0-27	0
		1-SI-68-32	0

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ICS COMPUTER POINT	LOOP /INSTRUMENT	INSTRUCTION	NOTES
T0423A	1-LPT-68-25	1-SI-0-23	0
T0426A	1-LPT-68-41	1-SI-0-10	0
T0438A	1-LPT-3-49	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
		TI-6.001	0
T0439A	1-LPT-68-24	1-SI-68-44	0
T0440A	1-LPT-68-44	1-SI-0-10	0
		1-SI-0-11	0
		1-SI-0-23	0
		1-SI-0-27	0
		1-SI-68-32	0
T0443A	1-LPT-68-44	1-SI-0-23	0
T0446A	1-LPT-68-60	1-SI-0-10	0
T0458A	1-LPT-3-91	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
		TI-6.001	0
T0459A	1-LPT-68-43	1-SI-68-44	0
T0460A	1-LPT-68-67	1-SI-0-10	0
		1-SI-0-11	0
		1-SI-0-23	0
		1-SI-0-27	0
		1-SI-68-32	0
T0463A	1-LPT-68-67	1-SI-0-23	0
T0466A	1-LPT-68-83	1-SI-0-10	0
T0478A	1-LPT-3-104	1-SI-0-20	0
		1-SI-0-24	0
		1-SI-92-1	0
		TI-6	0
		TI-41	0
		TI-6.001	0
T0479A	1-LPT-68-65	1-SI-68-44	0
T0480A	1-LPT-68-319	1-SI-68-44	0
T0481A	1-LPT-68-324	1-SI-68-44	0
T0497A	1-LPT-68-2	1-SI-92-1	0
T0630A	1-LPT-74-14	1-SI-68-44	0
T0631A	1-LPT-74-25	1-SI-68-44	0

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ICS COMPUTER POINT	LOOP /INSTRUMENT	INSTRUCTION	NOTES
T1000A	1-TE-30-210A	1-SI-0-2	2
T1001A	1-TE-30-210B	1-SI-0-2	2
T1002A	1-TE-30-210C	1-SI-0-2	2
T1003A	1-TE-30-210D	1-SI-0-2	2
T1004A	1-TE-30-210E	1-SI-0-2	2
T1005A	1-TE-30-210F	1-SI-0-2	2
T1014A	1-TE-30-210O	1-SI-0-2	2
T1015A	1-TE-30-210P	1-SI-0-2	2
T1016A	1-TE-30-210Q	1-SI-0-2	2
T1017A	1-TE-30-210R	1-SI-0-2	2
T1018A	1-TE-30-210S	1-SI-0-2	2
T1019A	1-TE-30-210T	1-SI-0-2	2
T1020A	1-TE-30-210U	1-SI-0-2	2
T1021A	1-TE-30-210V	1-SI-0-2	2
T1022A	1-TE-30-210W	1-SI-0-2	2
T1023A	1-TE-30-210X	1-SI-0-2	2
T1024A	1-TE-30-210Y	1-SI-0-2	2
T1025A	1-TE-30-210Z	1-SI-0-2	2
T1026A	1-TE-30-210AA	1-SI-0-2	2
T1027A	1-TE-30-210AB	1-SI-0-2	2
T1028A	1-TE-30-210AC	1-SI-0-2	2
T1029A	1-TE-30-210AD	1-SI-0-2	2
T1030A	1-TE-30-210AE	1-SI-0-2	2
T1031A	1-TE-30-210AF	1-SI-0-2	2
T1032A	1-TE-30-210AG	1-SI-0-2	2
T1033A	1-TE-30-210AH	1-SI-0-2	2
T0418A	1-LPT-3-36	TI-6.001	0
T0438A	1-LPT-3-49	TI-6.001	0
T0458A	1-LPT-3-91	TI-6.001	0
T0478A	1-LPT-3-104	TI-6.001	0
T2612A	1-LPT-67-455	1-SI-0-2	0
T2613A	1-LPT-67-456	1-SI-0-2	0
T2614A	2-LPT-67-455	1-SI-0-2	0
T2615A	2-LPT-67-456	1-SI-0-2	0
T2816A	0-TE-65-56	0-SI-65-6-A	2
T2817A	0-TE-65-57	0-SI-65-6-A	2
T2819A	0-TE-65-62	0-SI-65-6-B	2
T2836A	0-TE-65-61	0-SI-65-6-B	2

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Compliance ICS Computer Point List

ICS COMPUTER POINT	LOOP /INSTRUMENT	INSTRUCTION	NOTES
U0049	Stp Ct CBAG1,2	1-SI-0-15	2
U0050	Stp Ct CBBG1,2	1-SI-0-15	2
U0051	Stp Ct CBCG1,2	1-SI-0-15	2
U0052	Stp Ct CBDG1,2	1-SI-0-11	2
		1-SI-0-12	2
		1-SI-0-15	2
		1-SI-0-20	2
		1-SI-0-23	2
		1-SI-0-24	2
		TI-41	2
U0053	Stp Ct SBAG1,2	1-SI-0-15	2
U0054	Stp Ct SBBG1,2	1-SI-0-15	2
U0055	Stp Ct SBCG1	1-SI-0-15	2
U0056	Stp Ct SBDG1	1-SI-0-15	2
U0484		1-SI-0-12	2
		1-SI-0-24	2
U0900	1-LPF-2-256	1-ODI-90-25	2
U0902	0-LPF-90-320	0-ODI-90-8	2
U0903	1-LPF-90-400	1-ODI-90-26	2
U0904	0-LPF-90-300	0-ODI-90-22	2
U0905		1-SI-0-10	2
		1-SI-0-11	2
U0906		1-SI-0-10	2
		1-SI-0-11	2
U0907		1-SI-0-10	2
		1-SI-0-11	2
U0908		1-SI-0-10	2
		1-SI-0-11	2
		1-SI-0-23	2
U0909		1-SI-0-10	2
		1-SI-0-11	2
		1-SI-0-23	2
U0910		1-SI-0-10	2
		1-SI-0-11	2
U1118		1-SI-0-10	2
		1-SI-0-11	2
		1-SI-0-12	2
		1-SI-0-20	2
		1-SI-0-24	2
		1-SI-92-1	2
		1-SI-92-2	2
		1-SI-92-3	2
		TI-41	2
U1125		1-SI-0-2	2
U1126		1-SI-0-2	2

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ICS COMPUTER POINT	LOOP /INSTRUMENT	INSTRUCTION	NOTES
U1127		1-SI-0-10	2
		1-SI-0-11	2
		1-SI-0-12	2
		1-SI-0-20	2
		1-SI-0-24	2
		1-SI-92-1	2
		1-SI-92-2	2
		1-SI-92-3	2
		TI-41	2
U1141	N43 DELTA-FLUX	1-SI-0-21	2
		1-SI-92-3	2
U1142	N42 DELTA-FLUX	1-SI-0-21	2
		1-SI-92-3	2
U1143	N44 DELTA-FLUX	1-SI-0-21	2
		1-SI-92-3	2
U1144	N41 DELTA-FLUX	1-SI-0-21	2
		1-SI-92-3	2
U1145		1-SI-0-21	2
U1146		1-SI-0-21	2
U1151	1-INM-92-3/302	1-SI-0-21	2
U1152	1-INM-92-2/302	1-SI-0-21	2
U1153	1-INM-92-4/302	1-SI-0-21	2
U1154	1-INM-92-1/302	1-SI-0-21	2
U1159	1-INM-92-3/301	1-SI-0-21	2
U1160	1-INM-92-2/301	1-SI-0-21	2
U1161	1-INM-92-4/301	1-SI-0-21	2
U1162	1-INM-92-1/301	1-SI-0-21	2
U1169		1-SI-0-10	2
U1254		1-SI-92-1	2
U2118		1-SI-0-10	2
		1-SI-0-11	2
		1-SI-0-12	2
		1-SI-0-20	2
		1-SI-0-24	2
		1-SI-92-1	2
		1-SI-92-2	2
		1-SI-92-3	2
		TI-41	2
U2125		1-SI-0-2	2
U2126		1-SI-0-2	2
U9019		1-SI-0-2	2
U9020		1-SI-0-2	2
UF1015	2-LPF-90-400	1-ODI-90-26	2
UL1005		1-SI-68-32	2
UN2000		1-SI-68-32	2

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ICS COMPUTER POINT	LOOP /INSTRUMENT	INSTRUCTION	NOTES
UT5000		1-SI-68-32	2
Y0703A	1-LPF-3-147A	1-SI-3-912	0
Y0704A	1-LPF-3-155B	1-SI-3-912	0
Y0708A	1-LPF-3-163B	1-SI-3-912	0
Y0709A	1-LPF-3-170A	1-SI-3-912	0
Y2203A	1-LPF-90-400	1-ODI-90-15	0
		1-ODI-90-18	0
		1-SI-0-2	0
Y8000A	0-LPF-27-98	0-PI-ENV-3.1	0
Y8001A	0-TT-27-99	0-PI-ENV-3.1	0

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be:

a. ≥ 85°F and ≤ 110°F for the containment upper compartment, and

b. ≥ 100°F and ≤ 120°F for the containment lower compartment.

--NOTE--
The minimum containment average air temperatures in MODES 2, 3, and 4 may be reduced to 60°F.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limits.	A.1 Restore containment average air temperature to within limits.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

CONTAINMENT TEMPERATURE

INSTRUMENT NUMBER	PARAMETER DESCRIPTION	MODE	TS OPERABILITY LIMIT	TS SURVEILLANCE LIMIT	ALARM SP (SEE NOTE 4)	COMPUTER PT	SURVEILLANCE REQUIREMENT
1-TE-30-210A *	CONTAINMENT DOME	1	≥85°F & ≤110°F	≥87.1°F & ≤107.9°F	N/A	T1000A	SR 3.6.5.1
"	"	2,3 & 4	≥60°F & ≤110°F	≥62.1°F & ≤107.9°F	N/A	"	"
1-TE-30-210B *	PRZR ENCLOSURE CEILING	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1001A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210C *	SG ENCLOSURE CEILING	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1002A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210D *	SG ENCLOSURE CEILING	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1003A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210E *	REACTOR SHIELD WALL	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1004A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210F *	REACTOR SHIELD WALL	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1005A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210O *	OPP. REFUELING GATE	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1014A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210P *	PRESS. SUPPLY PLATFORM	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1015A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210Q *	ICE CONDENSER WALL	1	≥85°F & ≤110°F	≥87.1°F & ≤107.9°F	N/A	T1016A	SR 3.6.5.1
"	"	2,3 & 4	≥60°F & ≤110°F	≥62.1°F & ≤107.9°F	N/A	"	"
1-TE-30-210R *	ICE CONDENSER WALL OPP SIDE	1	≥85°F & ≤110°F	≥87.1°F & ≤107.9°F	N/A	T1017A	SR 3.6.5.1
"	"	2,3 & 4	≥60°F & ≤110°F	≥62.1°F & ≤107.9°F	N/A	"	"
1-TE-30-210S *	PRZR ENCLOSURE WALL	1	≥85°F & ≤110°F	≥87.1°F & ≤107.9°F	N/A	T1018A	SR 3.6.5.1
"	"	2,3 & 4	≥60°F & ≤110°F	≥62.1°F & ≤107.9°F	N/A	"	"
1-TE-30-210T *	ICE CONDENSER WALL OPP PRZR	1	≥85°F & ≤110°F	≥87.1°F & ≤107.9°F	N/A	T1019A	SR 3.6.5.1
"	"	2,3 & 4	≥60°F & ≤110°F	≥62.1°F & ≤107.9°F	N/A	"	"
1-TE-30-210U *	SG ENCLOSURE WALL	1	≥85°F & ≤110°F	≥87.1°F & ≤107.9°F	N/A	T1020A	SR 3.6.5.1
"	"	2,3 & 4	≥60°F & ≤110°F	≥62.1°F & ≤107.9°F	N/A	"	"
1-TE-30-210V *	SG ENCLOSURE WALL OPP SIDE	1	≥85°F & ≤110°F	≥87.1°F & ≤107.9°F	N/A	T1021A	SR 3.6.5.1
"	"	2,3 & 4	≥60°F & ≤110°F	≥62.1°F & ≤107.9°F	N/A	"	"
1-TE-30-210W *	ICE CONDENSER PLATE, RC PUMP 2	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1022A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210X *	ICE CONDENSER PLATE, RC PUMP 4	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1023A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210Y *	ICE CONDENSER PLATE, SG	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1024A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210Z *	ICE CONDENSER PLATE, SG	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1025A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210AA*	CONTAINMENT SUMP	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1026A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210AB*	FAN COMPARTMENT WALL	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1027A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210AC*	FAN COMPARTMENT WALL	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1028A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210AD*	INSTRUMENT ROOM WALL	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1029A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210AE*	ACCUMULATOR ROOM WALL	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1030A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210AF*	ACCUMULATOR ROOM WALL	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1031A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210AG*	ACCUMULATOR ROOM WALL	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1032A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"
1-TE-30-210AH*	ACCUMULATOR ROOM WALL	1	≥100°F & ≤120°F	≥102.5°F & ≤117.5°F	N/A	T1033A	SR 3.6.5.2
"	"	2,3 & 4	≥60°F & ≤120°F	≥62.5°F & ≤117.5°F	N/A	"	"

* SEE NOTE 2, 5

RCP SEAL FLOW

INSTRUMENT NUMBER	PARAMETER DESCRIPTION	TECH SPEC OPERABILITY LIMIT	TECH SPEC SURVEILLANCE LIMIT	ALARM SP (SEE NOTE 4)	COMPUTER PT	SURVEILLANCE REQUIREMENT
1-FI-62-1A	RCP 1 SEAL WATER FLOW	≤10 GPM	≤9.4 GPM	N/A	F0131A	SR 3.5.5.1
1-FI-62-14A	RCP 2 SEAL WATER FLOW	≤10 GPM	≤9.4 GPM	N/A	F0129A	SR 3.5.5.1
1-FI-62-27A	RCP 3 SEAL WATER FLOW	≤10 GPM	≤9.4 GPM	N/A	F0127A	SR 3.5.5.1
1-FI-62-40A	RCP 4 SEAL WATER FLOW	≤10 GPM	≤9.4 GPM	N/A	F0125A	SR 3.5.5.1

NOTES:

1. THIS DRAWING WILL CONTAIN ONLY THOSE INSTRUMENTS WHICH ARE USED TO EVALUATE TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS FOR TECHNICAL SPECIFICATION OPERABILITY LIMITS. IT WILL NOT CONTAIN INSTRUMENTS THAT ARE USED TO EVALUATE ODCM OR TECHNICAL MANUAL REQUIREMENTS.
2. THE COMPLIANCE READOUT DEVICE FOR THESE INSTRUMENT NUMBERS IS THE INTEGRATED COMPUTER SYSTEM (ICS). REFER TO THE "COMPUTER POINT" COLUMN FOR THE APPLICABLE COMPUTER POINT.
3. WBN CALCULATION TI-49, "ANALYSIS OF COMPLIANCE INSTRUMENTS LIST TO DETERMINE REQUIREMENTS FOR COMPLIANCE CALCULATIONS" IS UTILIZED AS A REFERENCE. THIS CALCULATION DETERMINES THOSE INSTRUMENTS WHOSE ACCURACY IS NOT CRITICAL IN DETERMINING COMPLIANCE WITH TECHNICAL SPECIFICATION OPERABILITY LIMITS.
4. THE ALARM SETPOINTS THAT ARE LISTED ON THIS DRAWING ARE NOT TO BE USED AS INDICATION FOR ENTERING OR EXITING LIMITED CONDITIONS OF OPERATIONS. THEY ARE TO BE ONLY USED AS INFORMATION.
5. INDIVIDUAL POINTS ARE NOT REQUIRED TO BE MAINTAINED WITHIN THE TEMPERATURE RANGE, ONLY THE ASSOCIATED WEIGHTED AVERAGE TEMPERATURE MUST BE MAINTAINED WITHIN LIMITS FOR TECH SPEC COMPLIANCE. TEMPERATURE LIMITATIONS EXIST WITH OTHER DESIGN BASIS CONSIDERATIONS.

REV	CHANGE REF	PREPARER	CHECKER	APPROVED	DATE
4	51723	JEW	GJB	[signature]	9-12-06
REVISED PER DCA 51723-34-0.					

SCALE: NTS EXCEPT AS NOTED

PROJECT FACILITY
CONTROL BUILDING
UNIT 1

TITLE
ELECTRICAL TECH SPEC COMPLIANCE TABLES

WATTS BAR NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY

INITIAL ISSUE	ENGINEERING APPROVAL
RO ISSUE PER: EDC E-50185-A DCA E50185-01-0	1 MARK A FEGLEY
	2 E.S. JONES
	3 D.L. OSBORNE

CCD | DATE 8-16-99 | 85 | E | 1-47W605-242 | R4

69.

Which ONE of the following describes the Modes of Operation in accordance with the Unit 1 Technical Specifications?

Unit 1 will enter MODE 5 from MODE 4 when ____(1)____ is ≤ ____(2)____.

	(1)	(2)
A.	Tavg	200 °F
B.	Tavg	350 °F
C.	Tcold	200 °F
D.	Tcold	350 °F

CORRECT ANSWER: A

DISTRACTOR ANALYSIS:

A. Correct: As seen in Table 1.1-1 of the Unit 1 Technical Specifications, Average Reactor Coolant Temperature (Tavg) is used as a factor into the Unit's mode of operation. Also seen in this table is that Mode 5 is achieved when Tavg $\leq$200°F.

B. Incorrect: It is correct that Tavg is used as a factor into the Unit's mode of operation. It is not correct that Mode 5 is achieved when Tavg $\leq$350°F. When Tavg is <350, the Unit enters Mode 4.

C. Incorrect: It is not correct that Tcold be used as a factor into the Unit's mode of operation. It is plausible to believe that it is because Tcold is used to satisfy the monitoring of several other Technical Specifications. For example, Tcold is used to determine the heatup and cooldown rates of the RCS and thus, the monitoring of T/S LCO 3.4.3, RCS Pressure and Temperature (P/T) Limits. Tcold is also an input into the cold over-pressure mitigation system and thus used to satisfy T/S LCO 3.4.12.

It is correct that Mode 5 is achieved when Tavg $\leq$200°F.

D. Incorrect: Again it is Incorrect and yet plausible that Tcold be used as a factor into the Unit's mode of operation. It is not correct that Mode 5 is achieved when Tavg $\leq$350°F. When Tavg is <350, the Unit enters Mode 4.

Question Number: 69

Tier: 3 Group: ______

K/A: G 2.2.35 Ability to determine Technical Specification Mode of Operation.

Importance Rating: 3.6 4.5

10 CFR Part 55: (CFR: 41.7 / 41.10 / 43.2 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to correctly identify the appropriate point at which Mode 5 is entered. The applicant must also identify the correct temperature (Tavg vs. Tcold) to use when determining a mode change. In doing so, the applicant is exercising the knowledge required to determine the Mode of Operation..

Technical Reference: Unit 1 Technical Specifications

Proposed references to be provided: None

Learning Objective: 3-OT-TS0000 Technical Specification Overview
9. State the number and respective title of the different operational modes according to Technical Specifications.

Cognitive Level:

Higher ______
Lower X

Question Source:

New X
Modified Bank ______
Bank ______

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Tavg NOT Tcold

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER (a)	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown (b)	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown (b)	< 0.99	NA	≤ 200
6	Refueling (c)	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

70.

Given the following conditions:

- Unit 1 is at 7% power

In accordance with TI-12.07A, Containment Access Modes 1-4, which ONE of the following tasks does **NOT** pose a radiological hazard requiring authorization for Special Access into containment?

A. Physically checking a lower ice condenser door shut

B. Locally inspecting the RCP #2 oil reservoir sight glasses

C. Isolating a leak on the letdown line by closing 1-ISV-68-580 (IC/705 AZ 221)

D. Venting the RHR suction header using 1-TV-74-504 and 1-TV-74-543 (716-AZ 301 #4 Accum Rm)

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: The lower ice condenser doors are inside of the polar crane wall. This distractor is actually the product of a common ILT misconception which is realized when an operator receives an Ice Condenser Door Open annunciator. With no other indications except this annunciator, the student will inevitably dispatch a simulated operator to inspect the door (as the existence of such presents a short term technical specification LCO required action). Additionally, it is possible to inspect some ice condenser lower doors by simply stepping a foot or two out of the Upper polar crane wall door in Accumulator room #4.

B. Incorrect: The inspection of the RCP oil sight glasses requires an operator to enter inside of the polar crane wall. Concerning RCP #2, one would cross the seal table area and then open the seal table area to Loop #2 polar crane wall door. The operator could then wall a few feet past that door before looking up and to his left to inspect the sight glasses.

C. Incorrect: 1-ISV-68-580 is a manual valve located just inside the polar crane wall door on the Loop 2/3 side of lower containment. This isolation is the first letdown isolation off of the RCS loop #3. Because this valve is located inside of the polar crane wall the operation of such when in Modes 1 or 2 will require special authorization in accordance with TI-12.07A. As seen in TI-12.07A, Special Access authorization is also required for entries inside the Polar Crane Wall or Regen Hx Room during Modes 1 or 2. It is plausible to believe that this task would require general access authorization because it is a task which would be of short duration and would require an operator to step only about two feet through the Loop 2/3 door. Also, one may believe that 1-ISV-68-580 were located in an area outside of the polar crane wall. Letdown isolations are provided which are outside of the polar crane wall.

D. Correct: The venting of the RHR common suction header is performed almost at the center of the accumulator room #4. This room is between the steel containment vessel and the polar crane wall. 1-TV-74-504 and 543 are ½ kerotest valves and do not provide an exceptionally fast venting of the large common suction line piping. Additionally, the location of these valves subjects the operator to a fairly high radiation field. Therefore, it is outside of the polar crane wall and even though that the venting would be relatively slow and present the operator to a fairly substantial dose, not subject to any other authorization but general access.

Question Number: 70

Tier: 3 Group: ______

K/A: 2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Importance Rating: 3.4 3.8

10 CFR Part 55: (CFR: 41.12 / 43.4 / 45.10)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to understand the radiological risks of several abnormal tasks and then to be able to recognize which permissions are required for those tasks.

Technical Reference: TI-12.07A, Containment Access Modes 1-4

Proposed references to be provided: None

Learning Objective: 3-OT-RAD0003 , RADIATION PROTECTION STANDARDS AND GUIDELINES
8. Identify the responsibilities of the following concerning the ALARA program: a. Radiation Protection Manager
b. TVA NPG Organization
c. Employee

Cognitive Level:
- Higher X
- Lower ______

Question Source:
- New X
- Modified Bank ______
- Bank ______

Question History: New question for the 2015-301 NRC RO Exam

Comments:

These are the lower ice doors which are located inside of the polar crane wall. This one is located just inside of the door in the polar crane wall which connects the upper level of the #4 accumulator room to the ice deck.


This is the oil sightglass on the #2 RCP. It is located off of a balcony inside of the polar crane wall.

This valve is located just inside of the Polar Crane wall door for the Loop 2/3 side of containment. It is INSIDE of the polar crane wall.
XLSG-3
1-ISV-68-580
RCS LOOP 3
LETDOWN ISOL
1
SYSTEM 62
BOUNDARY START

Performed On ________

NOMENCLATURE	LOCATION	POSITION	UNID	INITIALS	INITIALS

Inside Polar Crane Wall (Cont.)

NOMENCLATURE	LOCATION	POSITION	UNID	INITIALS	INITIALS
PRESSURIZER VENT	IC/788 AZ 105	[1]CLOSED	1-VTV-68-577		IV
RCS LOOP 3 HOT LEG SAMPLE	IC/720 AZ 194	OPEN	1-SMV-68-578		IV
RCS LOOP 3 EXCESS LETDOWN ISOL	IC/716 AZ 235	OPEN	1-ISV-68-579		IV
RCS LOOP 3 LETDOWN ISOL	IC/705 AZ 221	OPEN	1-ISV-68-580		IV
RCS LOOP 3 DRAIN	IC/704 AZ 221	CLOSED	1-DRV-68-581		IV
RCS LOOP 3 DRAIN	IC/704 AZ 221	CLOSED	1-DRV-68-582		IV
CVCS LETDOWN RELIEF DISCH DRAIN	IC/712 AZ 60	CLOSED	1-DRV-68-584		IV
RCS LOOP 2 PZR SPRAY LINE DRAIN	IC/732 AZ 110	[1]CLOSED	1-DRV-68-593		IV
PRESSURIZER SPRAY LINE VENT	IC/783 AZ 105	[1]CLOSED	1-VTV-68-594		IV
REACTOR VESSEL HEAD VENT LINE VENT	IC/755 AZ 30	[1, 2] CLOSED	1-VTV-68-604		IV
CVCS LETDOWN RELIEF DISCH DRAIN	IC/710 AZ 60	[1]CLOSED	1-DRV-68-605		IV

[1] **VERIFY** cap/blind flange installed

[2] May be N/A'd if Reactor Vessel Head removed.

Annulus

NOMENCLATURE	LOCATION	POSITION	UNID	INITIALS	INITIALS
PRESSURIZER RELIEF TANK GAS ANALYZER SUPPLY	OC/716 AZ 315	OPERABLE OPEN	1-FCV-68-307		IV
CONTROL AIR ISOLATION VALVE TO 1-FCV-68-307	OC/716 AZ 315	OPEN	1-ISV-32-3499		IV

BIT Room

NOMENCLATURE	LOCATION	POSITION	UNID	INITIALS	INITIALS
PRESSURIZER RELIEF TANK NITROGEN SUP PRESS CNTL	A4X/713 BIT RM	OPERABLE	1-PCV-68-304		IV
CONTROL AIR ISOLATION VALVE TO 1-PCV-68-304	A4X/713 BIT RM	OPEN	1-ISV-32-3199		IV
PRESSURIZER RELIEF TANK NITROGEN SUP FLOW CNTL	A4X/713 BIT RM	OPEN	1-FCV-68-305		IV
CONTROL AIR ISOLATION VALVE TO 1-FCV-68-305	A4X/713 BIT RM	OPEN	1-ISV-32-3198		IV
1-FCV-68-304 CNTL ISOL	A4X/713 BIT RM	OPEN	1-RTV-68-307A		IV
PRESSURIZER RELIEF TANK N_2 SUP HDR TEST VENT	A4X/713 BIT RM	CLOSED	1-TV-68-847		IV

2.2 Developmental References (continued)

I. RCI-128, ALARA Program Implementation

J. TI-134, Control of Portable Two Way Radios

K. TI-229, Temporary Shielding Program

L. WBN FSAR Questions 22.26, 212.116, and 212.129

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General Precautions and Limitations

A. All access portals (Airlocks el 757/716, equipment hatches el 757, and Annulus el 713) to the Containment building SHALL be controlled to prevent unauthorized entry while in Modes 1 through 4. Radiation Protection (RP) shall maintain positive access control of Containment and Annulus in accordance with RP procedures.

B. All entries into Containment or Annulus while in Modes 1 through 4 are authorized by completion of applicable sections of Appendix A, *Containment/Annulus Entry Authorization*.

C. One Appendix A is required for each area entered (Upper Containment, Lower Containment, Annulus). Appendix A's are typically issued for a 24 hour period only. However the SM may approve an extension beyond 24 hours. Authorization requirements for entries are described as follows:

1. *General Access* is authorized by the completion of Appendix A, Section 1.0, which requires the incore detectors to be TAGGED and in either of the two approved storage locations, as described in TI-41, (Normal STORAGE in Crane Wall or approximately ten feet below bottom of core limit in any core thimble). This ensures that a detector is in a location which would not expose entry personnel. In addition, this section also permits entry inside the Polar Crane Wall when below Mode 2.

2. *Special Access* is authorized by completion of Appendix A, Section 2.0, for any entry into Containment or Annulus when the incore detectors are NOT TAGGED or NOT in their approved storage location. Special Access authorization is also required for entries inside the Polar Crane Wall or Regen Hx Room during Modes 1 or 2.

3. All entries requiring Special Access Authorization shall require approvals in accordance with RCI-128, *ALARA Program Implementation* to ensure that appropriate controls are established to prevent personnel overexposure (i.e., Special RWP and APR).

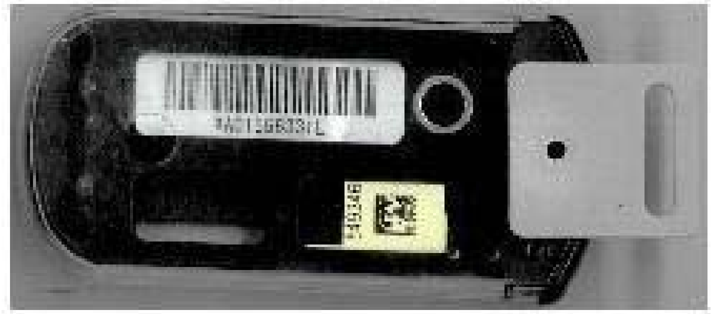

These valves are located in the #4 Accumulator room. This room is outside of the polar crane wall.
1-RFV-74-505
RHR PUMP SUCTION HDR
RELIEF
1-TV-74-543
RHR SUCTION HEADER
TEST VENT
1-TV-74-504
RHR SUCTION HEADER
TEST VENT

Performed On ________

NOMENCLATURE	LOCATION	POSITION	UNID	PERF INITIAL	VERIFIER INITIAL
Containment					
RHR SUCTION HDR VLV LOW POINT DRAIN	702-AZ315 Inside Polar Crane Wall	CLOSED	1-DRV-74-542		IV
RHR SUCTION HDR VLV LOW POINT DRAIN	702-AZ315 Inside Polar Crane Wall	CLOSED	1-DRV-74-503		IV
RHR SUCTION HDR DRAIN	702-AZ345 Inside Polar Crane Wall	CLOSED	1-DRV-74-500		IV
RHR SUCTION HDR DRAIN	702-AZ345 Inside Polar Crane Wall	CLOSED	1-DRV-74-541		IV
LOOP 4 HOT LEG TO RHR SUCTION	706-AZ345 Inside Polar Crane Wall	0-PI-OPS-17.0	1-FCV-74-1		
1-FCV-74-1 BYPASS RHR SUCTION	706-AZ345 Inside Polar Crane Wall	0-PI-OPS-17.0	1-FCV-74-9		
RHR SUCTION HEADER TEST VENT	716-AZ301 #4 Accum Rm	0-PI-OPS-17.0	1-TV-74-504		
RHR SUCTION HEADER TEST VENT	716-AZ301 #4 Accum Rm	0-PI-OPS-17.0	1-TV-74-543		
LOOP 4 HOT LEG TO RHR SUCTION	716-AZ301 #4 Accum Rm	0-PI-OPS-17.0	1-FCV-74-2		
1-FCV-74-2 BYPASS RHR SUCTION	716-AZ304 #4 Accum Rm	0-PI-OPS-17.0	1-FCV-74-8		

71.

FRONT



BACK

Which ONE of the following describes Dosimeter use at Watt Bar?

In order to check out an electronic dosimeter, the operator must scan the barcode on the ____(1)____ of the OSL.

The OSL is returned to Radiation Protection every ____(2)____ months so that it may be read.

A. (1) BACK
   (2) THREE

B. (1) BACK
   (2) SIX

C. (1) FRONT
   (2) THREE

D. (1) FRONT
   (2) SIX

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: WBNP uses a Landauer InLight Optically Stimulated Luminescence Dosimeter. This dosimeter has two bar codes. The bar code on the front only contains either a four or five digit number. This is the unique number assigned to the employee which Rad Con uses for tracking purposes. Therefore, when a certain employee obtains an electronic dosimeter, he scans this bar code such that the computerized dosimetry issuing station recognizes his specific account. The bar code on the back of the OSL is affixed to the actual dosimetry element (i.e. the actual element which records the radiation received). In this way, each employee will have their unique number assigned to one or more protective cases which house the dosimetry elements. The dosimetry elements are tied electronically to the specific case by relating the dosimetry serial number to the employees unique number. As seen the serial numbers are not equivalent in value. It is plausible to believe that the bar code on the back would be used because both bar codes are of the same font (7 of 9 bar code) and general appearance. To this extent, a standard bar code reader is able to process both equally well.

Previous to the use of the OSL, WBNP issued TLDs which were read on a quarterly basis. With the advent of the OSL, WBNP has shifted to a six month read frequency.

B. Incorrect: Again it is incorrect that the bar code on the back would be used for dosimetry issue. It is correct that the OSL is returned to Rad Con each six months.

C. Incorrect: While it is correct that the bar code on the front of the OSL is used for dosimetry issue, it is not correct that the OSL be returned to Rad Con every 3 months.

D. Correct: It is correct that both the bar code on the front of the OSL be used and that the OSL be returned every six months for processing.

Question Number: 71

Tier: 3 Group: ______

K/A: 2.3.15 Knowledge of radiation monitoring systems, such as fixed radiation monitors and | alarms, portable survey instruments, personnel monitoring equipment, etc.

Importance Rating: 2.9 3.1

10 CFR Part 55: (CFR: 41.12 / 43.4 / 45.9)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to possess the practical knowledge of a piece of personnel monitoring equipment (Optically Stimulated Luminescence Dosimeter)

Technical Reference: RCDP-15, Dosimetry Operations

Proposed references to be provided: None

Learning Objective: 3-OT-RAD0004 , Personnel Dosimetry
1. Describe the operation of Optically Stimulated Luminescent Dosimeters (IER 11-3, Having a solid understanding of plant design, engineering principles, and sciences.)

Cognitive Level:

Higher ______
Lower X

Question Source:

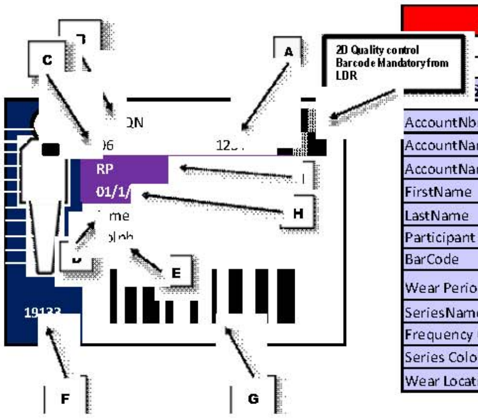
New X
Modified Bank ______
Bank ______

Question History: New question for the 2015-301 NRC RO Exam

Comments:

3.2.2 Ordering (continued)

1. The label example is shown below



ACCOUNTNBR	ACCOUNTNAME1	ACCOUNTNAME2	FIRSTNAME	LASTNAME
1234	TVA-SQN	0006	James	Rolph

Suggested Entry	Field	Length	Data	Data Type
AccountNbr	A	6	1234	alphanumeric
AccountName1	B	17	TVA-SQN	alphanumeric
AccountName2	C	10	0006	alphanumeric
FirstName	D	17	James	alphanumeric
LastName	E	17	Rolph	alphanumeric
Participant ID Nbr	F	6	19133	alphanumeric
BarCode	G	12	19133	alphanumeric
Wear Period *Entry must be date*	H	11	01/1/13-06/30/13	date
SeriesNameOrParticipantTitle	I	17	RP	alphanumeric
Frequency Color (H bar)	.....	2	BL (Blue)	.....
Series Color (V bar)	.....	2	PP (Purple)	.....
Wear Location	.....	.....	Whole Body	.....

> **NOTE**
>
> **The fields shown in the example and described below may be modified to meet a specific licensee or department needs. For example, the Emergency Planning Tennessee Emergency Management Agency dosimeters will have a different label scheme than the personnel dosimeters for occupationally exposed workers.**

A= Customer account # specific to each site

B = TVA site

C= TVA

D = worker first name

E = Worker last name

F = Plant ID

G = Bar Code for plant ID

This can be related to the helpful drawing shown above to see that the bar code on the front relates to the unique number assigned to the employee by rad con.

H= Wear period

I = Slot number

2. Background color is White for worker, area and environmental badges and yellow for controls.

3. Frequency Color (H bar) the color around the worker icon:

   Blue for Semi-annual (January – June)

72.

Which ONE of the following completes the statement below?

In accordance with 10 CFR 20.1201, Occupational dose limits for adults, individual worker's total effective dose equivalent (TEDE) shall be maintained less than __________ REM.

A. 0.5

B. 1.0

C. 3.0

D. 5.0

CORRECT ANSWER: B

DISTRACTOR ANALYSIS:

A. Incorrect: NPG-SPP-05.1, Radiological Controls details the Nuclear Power Groups administrative dose level program. This program lists five levels of administrative dose limits. The first of these is a dose equivalent of up to 0.5 TEDE. This limit does not require a signed authorization. This fact demonstrates the plausibility for this distractor. This distractor is not correct however because the next level of the administrative dose level program also does not require a signed authorization and is up to 1.0 TEDE.

B. Incorrect: Correct: the administrative dose level program allows an individual to receive up to 1.0 TEDE before obtaining signed authorization.

C. Incorrect: It is not correct that an individual is allowed to receive up to 3.0 TEDE before obtaining signed authorization. It is plausible to believe this because this is the Lens Dose Equivalent (LDE) listed in the second step of the administrative dose level program. The complete second step of the program is Up to 1.0 TEDE (or 3.0 LDE or 10 SDE) all sources.

D. Incorrect: It is not correct that an individual is allowed to receive up to 5.0 TEDE. It is plausible to believe this because, as seen in 10 CFR 20.1201(a)(1)(i), The total effective dose equivalent being equal to 5 REM. Therefore, by law, the occupational dose limit for an adult is 5.0 TEDE. However, by administrative process, the dose limit is less than this.

Question Number: 72

Tier: 3 Group: ______

K/A: G 2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.

Importance Rating: 3.2 3.7

10 CFR Part 55: (CFR: 41.12 / 43.4 / 45.10)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to possess the knowledge of the radiation exposure limits for normal conditions.

Technical Reference: 10 CFR 20.1201, Occupational dose limits for adults
NPG-SPP-05.1, Radiological Controls

Proposed references to be provided: None

Learning Objective: 3-OT-RAD0003 , RADIATION PROTECTION STANDARDS AND GUIDELINES
4. List TVA radiation dose goals.

Cognitive Level:

Higher ______
Lower X

Question Source:

New X
Modified Bank ______
Bank ______

Question History: New question for the 2015-301 NRC RO Exam

Comments:

3.2.3 Exposure Control (continued)

TABLE 1

ADMINISTRATIVE DOSE LEVEL PROGRAM

Dose Equivalent (Rem)	**Requirement**	**Authorization to exceed (signatures)**
Up to **0.5 TEDE** (or 1.5 LDE or 5.0 SDE) at TVA	Statement of current year dose and previous years dose signed by individual	Not applicable
Up to **1.0 TEDE** (or 3.0 LDE or 10 SDE) all sources	NRC FORM-4 or equivalent to document current year and previous years dose equivalent	Not applicable
To exceed **1.0 TEDE** (or 3.0 LDE or 10 SDE) all sources	Same as above	RPM/RSO
To exceed **5.0 TEDE[3] all sources**	Form-4 information must be verified and a Planned Special Exposure initiated	RPM/RSO, Plant Manager[1], and Site VP[2] or SED as appropriate.
To exceed **1N[4] all sources**	Form-4 must be verified	RPM/RSO, Plant Manager[1], and Site VP[2] or SED as appropriate.

[1] At non-nuclear plant sites, this will be the RSO’s immediate supervisor.

[2] At non-nuclear plant sites, this will be the applicable TVA VP.

[3] Authorizations for a planned special exposure will only be considered in an exceptional situation when alternatives that might avoid the dose estimated to result from the planned special exposure are unavailable or impractical.

[4] Total effective dose equivalent should not exceed 1N rem, where N equals the individual’s age in years at last birthday, without the authorization signatures delineated.

4. ADLs are based on dosimeter(s) used in determining the reported dose. Results which exceed an administrative level, based on other dosimeter data, do not violate the ADL.

NOTE

Exceeding an ADL as a result of an incorrect signed documentation of current year or previous years dose does not constitute a violation of the intent of the ADL.

5. When visitor or contract personnel have more restrictive dose limits than TVA - the more restrictive limits will be used. It is the responsibility of the contractor to provide written notification to RP (or RSO) of any company administrative limit.

6. To ensure that ADLs are not exceeded, an administrative control system shall be maintained.

7. Individuals under the age of 18 shall not enter radiologically controlled areas.

Subpart C--Occupational Dose Limits

Source: 56 FR 23396, May 21, 1991, unless otherwise noted.

§ 20.1201 Occupational dose limits for adults.

(a) The licensee shall control the occupational dose to individual adults, except for planned special exposures under § 20.1206, to the following dose limits.

(1) An annual limit, which is the more limiting of--

(i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or

(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).

(2) The annual limits to the lens of the eye, to the skin of the whole body, and to the skin of the extremities, which are:

(i) A lens dose equivalent of 15 rems (0.15 Sv), and

(ii) A shallow-dose equivalent of 50 rem (0.5 Sv) to the skin of the whole body or to the skin of any extremity.

(b) Doses received in excess of the annual limits, including doses received during accidents, emergencies, and planned special exposures, must be subtracted from the limits for planned special exposures that the individual may receive during the current year (see § 20.1206(e)(1)) and during the individual's lifetime (see § 20.1206(e)(2)).

(c) When the external exposure is determined by measurement with an external personal monitoring device, the deep-dose equivalent must be used in place of the effective dose equivalent, unless the effective dose equivalent is determined by a dosimetry method approved by the NRC. The assigned deep-dose equivalent must be for the part of the body receiving the highest exposure. The assigned shallow-dose equivalent must be the dose averaged over the contiguous 10 square centimeters of skin receiving the highest exposure. The deep-dose equivalent, lens-dose equivalent, and shallow-dose equivalent may be assessed from surveys or other radiation measurements for the purpose of demonstrating compliance with the occupational dose limits, if the individual monitoring device was not in the region of highest potential exposure, or the results of individual monitoring are unavailable.

(d) Derived air concentration (DAC) and annual limit on intake (ALI) values are presented in table 1 of appendix B to part 20 and may be used to determine the individual's dose (see § 20.2106) and to demonstrate compliance with the occupational dose limits.

(e) In addition to the annual dose limits, the licensee shall limit the soluble uranium intake by an individual to 10 milligrams in a week in consideration of chemical toxicity (see footnote 3 of appendix B to part 20).

(f) The licensee shall reduce the dose that an individual may be allowed to receive in the current year by the amount of occupational dose received while employed by any other person (see § 20.2104(e)).

[56 FR 23396, May 21, 1991, as amended at 60 FR 20185, Apr. 25, 1995; 63 FR 39482, July 23, 1998; 67 FR 16304, Apr. 5, 2002; 72 FR 68059, Dec. 4, 2007]

73.

Which ONE of the following completes the statements listed below?

There _____(1)_____ which may be **DIRECTLY** entered.

The Immediate Actions listed in certain EOPs ______(2)_____ without the written instruction being available.

A. (1) is **ONLY ONE** EOP
(2) **ONLY** require that the operator perform the left hand column step

B. (1) is **ONLY ONE** EOP
(2) require that the operator perform both the left hand column step **AND** the right hand column-“response not obtained”-step

C. (1) are **TWO** EOPs
(2) **ONLY** require that the operator perform the left hand column step

D. (1) are **TWO** EOPs
(2) require that the operator perform both the left hand column step **AND** the right hand column-“response not obtained”-step

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: Again it is incorrect and yet plausible that there is only one EOP which may be directly entered. Also, it is incorrect that an immediate action only requires the operator to execute the left hand step from memory. It is plausible to believe such as some of the immediate actions have response not obtained steps which are quire lengthy and difficult to remember (such as the RNO for step 3 of 1-E-0).

B. Incorrect: It is incorrect that there is only one EOP which may be directly entered. As seen in 0-TI-12.04, implementation of the EOI set is limited to the two cases of: implementing 1-ECA-0.0 and implementing 1-E-0. It is plausible to believe that the entry into 1-ECA-0.0 would be made much in the manner that the entry into 1-FR-S.1, ATWS is (i.e. while the implementation of 1-FR-S.1 appears direct, the implicit EOI network entry is made via the response not obtained of the immediate action step 1 of 1-E-0). Along such lines, one could reason that the entry into 1-ECA-0.0 while appearing direct was actually made via the implicit transition from the immediate action step 3 of 1-E-0.

As seen in 0-TI-12.04, the immediate action steps are “intended to be performed, if necessary, without the written instruction being available.” There is no relaxation providing the need to execute only the left hand step and not the right hand “response not obtained” step.

C. Incorrect: It is correct that there are two EOPs which are directly entered. Also, it is incorrect and yet plausible that only the left hand column of an immediate action step be performed (if needed) without the written instruction..

D. Correct: It is correct that there are two EOPs which are directly entered. Also, it is correct that both columns of an immediate action are required to be performed (if needed) without the written instruction.

Question Number: 73

Tier: 3 Group: ______

K/A: G 2.4.1 Knowledge of EOP entry conditions and immediate action steps.

Importance Rating: 4.6 4.8

10 CFR Part 55: (CFR: 41.10 / 43.5 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to recognize the entry conditions for 1-ECA-0.0 and 1-FR-S.1 while simultaneously recalling the immediate actions of 1-ECA-0.0. .

Technical Reference: 1-ECA-0.0, Loss of Shutdown Power
1-E-0, Reactor Trip or Safety Injection
TI-12.04, User's Guide for Abnormal and Emergency Operating Instructions
WOG background document for 1-ECA-0.0

Proposed references to be provided: None

Learning Objective: 3-OT-TI1204, USER's GUIDE FOR AOIs and EOIs
1. Describe the conditions for entry into the EOP network
11. Identify the three (3) Emergency Instructions which contain "Immediate Actions."

Cognitive Level:
- Higher ______
- Lower X

Question Source:
- New X
- Modified Bank ______
- Bank ______

Question History: New question for the 2015-301 NRC RO Exam

Comments:

1.0 INTRODUCTION

1.1 Purpose

This Instruction provides guidance for the use of the Emergency Operating Instructions (EOIs) and Abnormal Operating Instructions (AOIs) and ensures standardized implementation of the EOIs.

1.2 Applicability

A. These guidelines will be followed by the Main Control Room (MCR) crew during emergency conditions as well as training exercises.

B. This Instruction provides the rules of usage for the set of Emergency Operating Instructions including the Status Trees. The EOI set includes Emergency Instructions (E and ES), Emergency Contingency Actions (ECA), and Function Restoration Instructions (FR).

C. The rules explained in Section 2.2.7 (Operator Performance of Action Steps) also apply to Abnormal Operating Instructions (AOIs) that are written in the same two column format. Section 2.3 (Types of Action Steps in Two Column Instructions) is descriptive of EOIs and AOIs.

2.0 DETAILS

The EOIs direct operator actions to the most urgent safety or operational conditions. This Instruction provides the rules of priority and usage which, when properly applied, will direct the operator through the EOI network. The rules of usage define the intended use of two column format instructions. The rules of priority define the hierarchy of the individual instructions and must be followed to ensure proper implementation.

2.1 Implementation of Emergency Operating Instructions

Implementation of the EOI set is limited to two specific conditions:

Only two EOPs are direct entry: 1-E-0 and 1-ECA-0.0.

A. Any time a complete loss of both trains of Shutdown Power occurs the MCR Team enters the EOI network by implementing 1-ECA-0.0, Loss of Shutdown Power. This includes any time during the performance of any other instruction.

B. Any time a reactor trip or safety injection occurs, or is required, the MCR Team enters the EOI network by implementing 1-E-0, Reactor Trip or Safety Injection.

2.2.3 Foldout Page

A. This page presents actions or transitions which are applicable at any time in the given instruction. Upon transition from an instruction, the current instruction's foldout page becomes applicable and use of the previous instruction's foldout page is discontinued.

1. In the control room, the foldout page information is presented on the back of each page of instructions for which there is a foldout page.

2. The information on the foldout page should be continuously monitored to determine when operator action is necessary.

B. Transitions to other instructions allow immediate response to new symptoms as they appear.

2.2.4 Immediate Action Steps

Steps that have been designated as “Immediate Actions” are contained in three emergency instructions and selected AOIs. These steps are intended to be performed, if necessary, without the written instruction being available.

A. Operators are required to be able to complete the intent of immediate operator action steps.

B. During immediate operator action steps the operators will ensure automatic actions have occurred or initiate signals as appropriate. Diagnostic or repair actions will be delayed until the immediate actions are complete to allow for evaluation of plant response.

C. The immediate actions of 1-E-0 will be addressed as follows:

1. Steps 1 through 4 will be performed in order by the OAC and completion, along with any discrepancies, will be communicated to the Procedure Reader.

2. The BOP will acknowledge alarms as necessary to reduce the noise level, and perform backup verification of steps 1 through 4.

3. Procedure Reader will read the immediate action high level step. The OAC will confirm the high level step by verbalizing the low level steps to the procedure reader.

4. When reentering 1-E-0 from another EOI, the first four high level steps must be reconfirmed, it is **NOT** necessary to reperform each low level action.

2.2.4 Immediate Action Steps (continued)

D. The immediate actions for 1-FR-S.1, Nuclear Power Generation/ATWS, and 1-ECA-0.0, Loss of Shutdown Power, are consistent with the initial actions of 1-E-0, Reactor Trip or Safety Injection, and will be performed in the same manner.

E. The immediate actions contained in Abnormal Operating Instructions will be performed in order and completion, along with discrepancies, will be communicated to the Procedure Reader.

   1. 1-AOI-2, Malfunction of Reactor Control System

   2. 1-AOI-4, Nuclear Instrumentation Malfunctions

   3. AOI-18, Malfunction of Pressurizer Pressure Control System

2.2.5 Sequence of Step Performance

The EOI steps have been written in a specified sequence. This sequence is designed to ensure timely operator action for the most limiting evolution.

A. The operator is expected to perform all high level action steps in the specified sequence.

   1. High level steps and substeps for which action may be delayed until plant conditions allow performance are identified by the step wording.

   2. The Procedure Reader is responsible for keeping track of steps for which performance must be delayed.

   3. When one operator is involved in completing a step, the Procedure Reader may direct the other operator to perform a subsequent step provided the subsequent step does **NOT** depend upon completion of the previous step.

Step	Action/Expected Response	Response Not Obtained

3. **CHECK** 6.9 kV shutdown boards:

   a. At least one board energized from:

      - CSST (offsite),

      OR

      - D/G (blackout) with ERCW flow to running DG(s) [0-M-27A]:
        - 1-HS-67-66A **OR** 1-HS-67-68A
        - 2-HS-67-66A **OR** 2-HS-67-68A
        - 1-HS-67-67A **OR** 1-HS-67-65A
        - 2-HS-67-67A **OR** 2-HS-67-65A

This step has grown so "out of hand" that some operators have begun to claim that the right hand column cannot be expected to be performed without an instruction.

Response Not Obtained:

RESTORE power to at least one train of shutdown boards:

1) **EMERGENCY START** All D/G's [1-M-1].

2) **IF** D/G did **NOT** start from 1-M-1, **THEN**

   EMERGENCY START All D/G's [0-M-26]

3) **IF** both trains of shutdown boards remain de-energized, **THEN**

   PLACE 6.9kV SD Bd transfer switch in MAN [1-M-1], **AND**

   CLOSE supply breaker from energized source.

4) **ENSURE** ERCW to running DG(s) [0-M-27A]:
   - 1-HS-67-66A **OR** 1-HS-67-68A
   - 2-HS-67-66A **OR** 2-HS-67-68A
   - 1-HS-67-67A **OR** 1-HS-67-65A
   - 2-HS-67-67A **OR** 2-HS-67-65A

5) **IF** ERCW flow **CANNOT** be aligned, **THEN**

   EMERGENCY STOP affected DG(s)

IF power can **NOT** be restored to at least one train of shutdown boards, **THEN**

** **GO TO** 1-ECA-0.0, Loss of Shutdown Power.

74.

Given the following conditions:

- Unit 1 is at 100% power.
- A contract worker uses a plant phone and the x3911 line to report that a temporary pump located outside of the Unit 2 TDAFWP room has caught fire.
- Heavy smoke is present at the scene of the fire.

Which ONE of the following describes the appropriate response in accordance with 1-AOI-30.1, Plant Fires?

The RO actuating the fire alarm will ____(1)____.

The control room staff will ____(2)____.

***NOT**E:* *0-SOI-13.01, Fire Detection System*

A. (1) INITIATE Fire Alarm and then MANUALLY RESET Fire Alarm

(2) initiate an ABI

B. (1) INITIATE Fire Alarm and then MANUALLY RESET Fire Alarm

(2) secure ventilation in accordance with 0-SOI-13.01

C. (1) **ONLY** INITIATE Fire Alarm, the alarm will AUTOMATICALLY RESET

(2) initiate an ABI

D. (1) **ONLY** INITIATE Fire Alarm, the alarm will AUTOMATICALLY RESET

(2) secure ventilation in accordance with 0-SOI-13.01

<u>CORRECT ANSWER:</u> <u>B</u>

<u>DISTRACTOR ANALYSIS:</u>

A. Incorrect: It is correct that in accordance with 1-AOI-30.1, that the operator must **INITIATE** Fire Alarm and then **RESET** Fire Alarm. This is seen in steps B. and C. of Appendix B of the AOI. The reason for this is that the fire alarm is actuated and secured by one red pushbutton which sits on the reactor operator's desk in the MCR. If one does not push the button a second time, the fire alarm will not automatically secure itself. It is Incorrect that the main control room will initiate an ABI (in order to secure the auxiliary building ventilation). Step 5 of 1-AOI-30.1 directs: **REFER TO** 0-SOI-13.01 Fire Detection System, for ventilation systems required and USE Appendix A of this Instruction, as necessary, for further guidance. Appendix A addresses fire dampers which fail to automatically close. 0-SOI-13.01, Fire Detection System contains Appendix B, Operator Actions. Appendix B of the SOI directs which ventilation fans are to be secured for a fire in a given location. For general areas in the auxiliary building, one may see that either the Unit 1, Unit 2 or both Units General Supply and Exhaust Fans are secured. It is plausible to believe that would actuate an ABI because doing so would secure the Auxiliary building ventilation in a predictable manner. The WARNING which appears above the Operator Actions for the Auxiliary Building in Appendix B was included because of a corrective action which resulted from the use of the SOI. The operating crew using the Appendix realized that the alterations to the Aux Bldg ventilation being directed were not (at the time) existent in a procedure and would cause a gross imbalance in Aux Bldg pressure-to the extent that damage to dampers and doors could occur. The crew asked the fire protection engineer (who had input into the SOI) about how the securing of ventilation could be done and the crew asked about the feasibility of using an ABI. The decision was made to include the warning discussed.

B. Correct: Again, it is correct that one must actuate and subsequently reset the fire alarm. Also, it is correct that the crew would secure ventilation in accordance with 0-SOI-13.01.

C. Incorrect: It is not correct that a manual reset of the fire alarm is not required. It is plausible to believe this because the accountability alarm does have an automatic reset. The accountability alarm is located in the vicinity of the Fire Detection panel, 0-M-29 and will cease several minutes after being actuated. It is also Incorrect and yet plausible that an ABI would be used to secure Aux Bldg ventilation.

D. Incorrect: It is not correct and yet plausible that a manual reset of the fire alarm is not required. It is correct that the crew would secure ventilation in accordance with 0-SOI-13.01.

Question Number: 74

Tier: 3 Group: ______

K/A: 2.4 Emergency Procedures / Plan
2.4.27 Knowledge of fire in the plant procedures.

Importance Rating: 3.4 3.9

10 CFR Part 55: (CFR: 41.10 / 43.5 / 45.13)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to possess the knowledge of how the fire in the plant procedure 1-AOI-30.1 addresses the actuation of the fire alarm as well as how 1-AOI-30.1 secures ventilation for a space affected by smoke.

Technical Reference: 0-SOI-13.01, Fire Detection System
1-AOI-30.1, Plant Fires

Proposed references to be provided: None

Learning Objective: 3-OT-AOI3000, AOI-30.1 & 30.2, “PLANT FIRES”
4. Given a set of plant conditions, DESCRIBE operator actions required in response to the following per 1-AOI-30, Plant Fires.

Cognitive Level:
- Higher ______
- Lower X

Question Source:
- New X
- Modified Bank ______
- Bank ______

Question History: New question for the 2015-301 NRC RO Exam

Comments:

Appendix B
(Page 1 of 1)

Operator Actions Guideline

A. **OBTAIN** the following: ☐

NAME of person reporting fire (if verbal): ___________________________

LOCATION of fire: ____

TYPE of fire: ____

SEVERITY of fire: ____

DATE & TIME: ____

B. **INITIATE** Fire Alarm. ☐

C. **RESET** Fire Alarm. ☐

Must push the fire alarm button to sound the alarm and then push the button again to secure the alarm.

D. **NOTIFY** Fire Operations Unit by radio. ☐

E. **ANNOUNCE** fire location over PA. ☐

F. **REPEAT** fire announcement. ☐

G. **INITIATE** fire alarm. ☐

H. **RESET** fire alarm when necessary. **TIME** of reset: __________________ ☐

I. **ENSURE** fire pumps running as necessary:

HPFP: 1A-A ☐ 2-A-A ☐ 1B-B ☐ 2B-B ☐ DGFP: ☐

J. **ANNOUNCE** location of Incident Command Post over PA and radio. ☐

K. **IF** a confirmed fire is located in **ANY** of the following:

1. Control Building ☐
2. Auxiliary Building ☐
3. Reactor Building ☐
4. Annulus ☐

THEN NOTIFY ALL AUOs to report with gear and Standby for AOI-30.2. ☐

Step	Action/Expected Response	Response Not Obtained

3.0 OPERATOR ACTIONS (continued)

5. **REFER TO** SOI-13.01, Fire Detection System, for ventilation systems required and USE Appendix A of this Instruction, as necessary, for further guidance.

Step 5 directs the use of the SOI; this SOI recommends the appropriate ventilation lineups.

75.

Given the following conditions:

- Unit 1 is at 100% power.
- Supplemental Condenser Circulating Water System (SCCW) has been placed in service in accordance with SOI-27.03, section 5.2.

Subsequently:

- COOLING TOWER BASIN LEVEL HI (164-D) is LIT.

Which ONE of the following describes the appropriate alarm response?

The operator responding to this annunciator will validate the alarm by comparing the alarm setpoint to the actual reading obtained ____(1)____.

In accordance with ARI-159-165, Sumps & CCW, the operator responding to this annunciator will **ADJUST** 0-FCV-112, SUPPLEMENTAL CCW SYSTEM SUPPLY FLOW CONTROL by ____(2)____.

NOTE: SOI-27.03, Supplemental Condenser Circulating Water System; Section 5.2, Placing SCCW In Service

A. (1) on the ICS
(2) operating the valve from 1-M-15

B. (1) on the ICS
(2) dispatching an AUO to operate the valve from a local control

C. (1) locally at the cooling tower basin
(2) operating the valve from 1-M-15

D. (1) locally at the cooling tower basin
(2) dispatching an AUO to operate the valve from a local control

CORRECT ANSWER: D

DISTRACTOR ANALYSIS:

A. Incorrect: It is not correct that the operator would refer to ICS to obtain cooling tower basin level. Watts Bar is a closed loop condenser circulating water system. The CCW pumps take suction from the cooling tower basin, provide cooling water to the condenser and then the return water to the condenser is discharged back to the cooling tower. To compensate for evaporative losses as well as cooling tower blowdown, three forms of CCW makeup are provided. These are the raw cooling water bypass strainers, essential raw cooling water discharge and the supplemental condenser circulating water system (SCCW). The SCCW system was a modification to the original plant design and takes water originally designed to provide the Watts Bar Coal Plant (taken from over twenty feet below the surface of Watts Bar Lake) and discharges such into the Unit 2 cooling tower basin. This may be seen on print 1-47W831-1. The Unit 2 cooling tower basin is connected to the Unit 1 cooling tower and the cold injection water is fed into the suction of the Unit 1 CCW pumps. The hot water in the Unit 1 basin is discharged to the original discharge point of the Watts Bar Coal plant. Coupled with the normal operation of CCW, this feed and bleed provides a very effective cooling. If the supply valve and the discharge valve for SCCW are not balanced properly, then the Unit 1 basin can either drain or overfill. Because balancing a 90” butterfly valve and a 78” butterfly valve is not an easy task, this evolution can taken well over a shift (and require the next shift to respond to cooling tower level alarms). Both the SOI for placing SCCW in service as well as the ARI for addressing the cooling tower high level alarm direct that the SCCW inlet be throttled (0-FCV-27-112). As this was a design modification, controls for these valves are not on the water service panel of the MCR (1-M-15). Additionally, cooling tower basin level is not a parameter monitored on the ICS. This may be seen on 1-47W610-27-2. It is plausible to believe that the ICS would monitor cooling tower basin level because disastrous results come from either a high or low level. A high level will cause an overfill which is an environmental reportable event. A low level will cause the CCW pumps to lose suction and trip; causing a turbine and reactor trip. Parameters such as holding pond level are monitored on ICS and as such it would be reasonable to monitor the basin level. One of the original makeup methods for the cooling tower basin level is RCW makeup via the bypass strainer. This makeup passes through the control valve 1-FCV-24-191. The handswitch for this valve is on 1-M-15 (seen on print 1-47W611-24-1) and as such it is plausible to believe that controls for SCCW would be provided on 1-M-15 especially given the consequences that a loss of basin level control could have.

B. Incorrect: Again, it is not correct and yet plausible that ICS would be used to obtain cooling tower basin level. Also, it is correct that 0-FCV-27-112 would be operated locally.

C. Incorrect: It is correct that basin level need be obtained locally. Also, it is Incorrect and yet plausible that SCCW would be adjusted locally.

D. Correct: It is correct that basin level need be obtained locally. Also, it is correct that SCCW would be adjusted locally.

Question Number: 75

Tier: 3 Group: ______

K/A: 2.4 Emergency Procedures / Plan
2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Importance Rating: 4.2 4.0

10 CFR Part 55: (CFR: 41.10 / 43.5 / 45.3)

10CFR55.43.b: Not applicable

K/A Match: K/A is matched because the applicant is required to demonstrate the ability to validate the alarm setpoint in an ARI by correctly identifying the source of the data. Next the applicant must demonstrate the ability to operate the controls identified in the ARI.

Technical Reference: 1-47W831-1; 1-47W611-24-1; 1-47W610-27-2
ARI-159-165, Sumps and CCW
SOI-27.03, Supplemental Condenser Circulating Water System

Proposed references to be provided: None

Learning Objective: 3-OT-SYS027A, CONDENSER CIRCULATING WATER
4. EXPLAIN the physical connections and/or cause-effect relationships between the Condenser Circulating Water System and the following systems:
a. Supplemental CCW System (SCCW)

Cognitive Level:

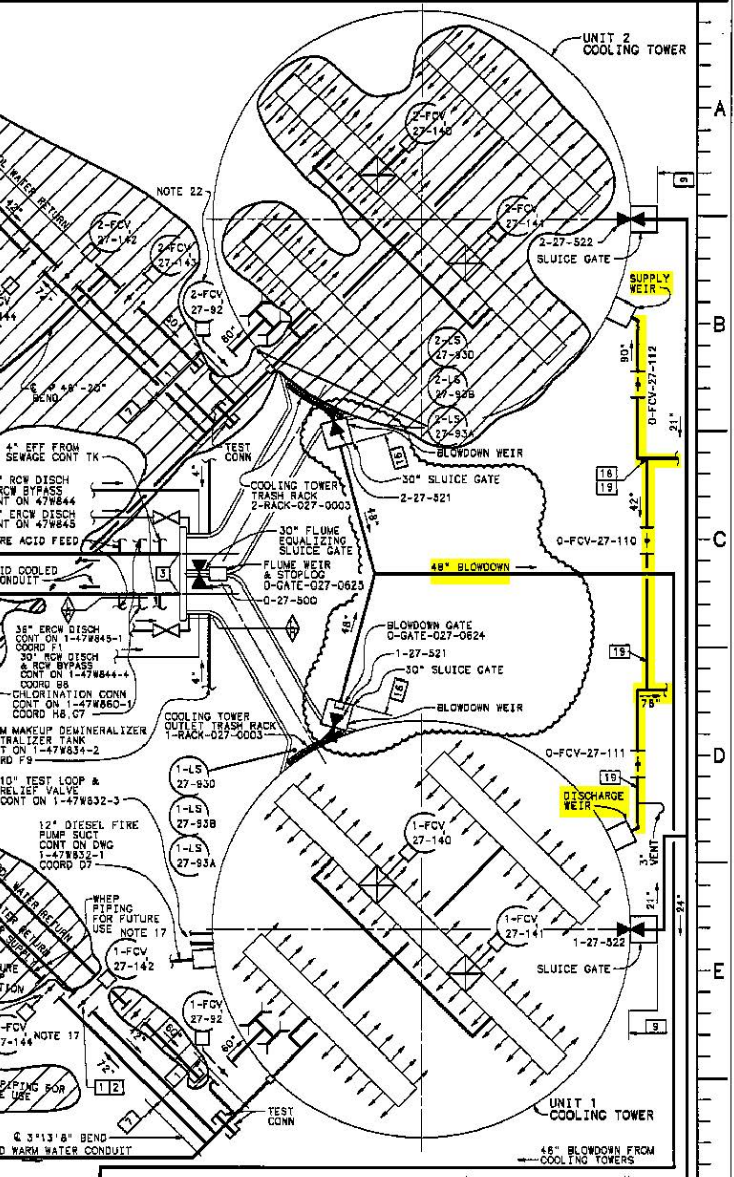
Higher ______
Lower X

Question Source:

New X
Modified Bank ______
Bank ______

Question History: New question for the 2015-301 NRC RO Exam

Comments:



UNIT 2
COOLING TOWER
A
B
C
D
E
9
2-FCV
27-140
2-FCV
27-141
2-27-522
SLUICE GATE
SUPPLY
WEIR
NOTE 22
WATER RETURN
42"
2-FCV
27-142
2-FCV
27-143
2-FCV
27-92
72"
60"
80"
CV
144
℄ 48"-20"
BEND
7
1
2-LS
27-930
2-LS
27-93B
2-LS
27-93A
BLOWDOWN WEIR
90"
0-FCV-27-112
21"
16
18
19
30" SLUICE GATE
2-27-521
4" EFF FROM
SEWAGE CONT TK
TEST
CONN
" RCW DISCH
RCW BYPASS
NT ON 47W844
" ERCW DISCH
NT ON 47W845
RE ACID FEED
ID COOLED
ONDUIT
COOLING TOWER
TRASH RACK
2-RACK-027-0003
48"
30" FLUME
EQUALIZING
SLUICE GATE
FLUME WEIR
& STOPLOG
0-GATE-027-0625
0-27-500
3
42"
0-FCV-27-110
48" BLOWDOWN
36" ERCW DISCH
CONT ON 1-47W845-1
COORD F1
30" RCW DISCH
& RCW BYPASS
CONT ON 1-47W844-4
COORD B6
CHLORINATION CONN
CONT ON 1-47W860-1
COORD H8,C7
BLOWDOWN GATE
0-GATE-027-0624
1-27-521
30" SLUICE GATE
BLOWDOWN WEIR
78"
M MAKEUP DEMINERALIZER
TRALIZER TANK
T ON 1-47W834-2
RD F9
COOLING TOWER
OUTLET TRASH RACK
1-RACK-027-0003
1-LS
27-930
1-LS
27-93B
1-LS
27-93A
0-FCV-27-111
10" TEST LOOP &
RELIEF VALVE
CONT ON 1-47W832-3
DISCHARGE
WEIR
12" DIESEL FIRE
PUMP SUCT
CONT ON DWG
1-47W832-1
COORD D7
1-FCV
27-140
3"
VENT
21"
24"
WATER RETURN
WHEP
PIPING
FOR FUTURE
USE NOTE 17
1-FCV
27-142
1-FCV
27-141
1-27-522
SLUICE GATE
1-FCV
27-92
FCV
7-144
NOTE 17
72"
60"
PIPING FOR
E USE
1 2
7
TEST
CONN
UNIT 1
COOLING TOWER
℄ 3°13'8" BEND
D WARM WATER CONDUIT
48" BLOWDOWN FROM
COOLING TOWERS

164-D

Source

1-LS-27-93B

Setpoint

el 730' 6" (6 inches below top of basin and 6 inches above normal level)

COOLING TOWER BASIN LEVEL HI

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Probable Cause:

A. CT blowdown terminated via closure of all the following valves:
- 0-FCV-27-97, HOLDING POND ISOL
- 0-FCV-27-100, 54" DIFFUSER A SUP
- 0-FCV-27-101, 48" DIFFUSER B SUP

B. Misalignment of SCCW supply or discharge valves
C. SCCW evolution in progress
D. Excessive debris on Trash rack 1-RACK-27-1 and/or 1-RACK-27-3

NOTE
If both HI level and LO level (window 164-E) are lit, serious blockages on trash racks could exist.

Corrective Action:

[1] **REFER** TO SOI-24.01 for operation of 1-FCV-24-191, RCW BYPASS FLOW CONTROL VALVE.

[2] **CHECK** position of 1-FCV-27-97, -100, and -101.

[3] **ENSURE** SCCW supply and discharge valves BOTH open OR BOTH closed:
- 0-FCV-27-112, SUPPLEMENTAL CCW SYSTEM SUPPLY FLOW CONTROL.
- 0-FCV-27-111, SUPPLEMENTAL CCW SYSTEM DISCHARGE FLOW CONTROL.

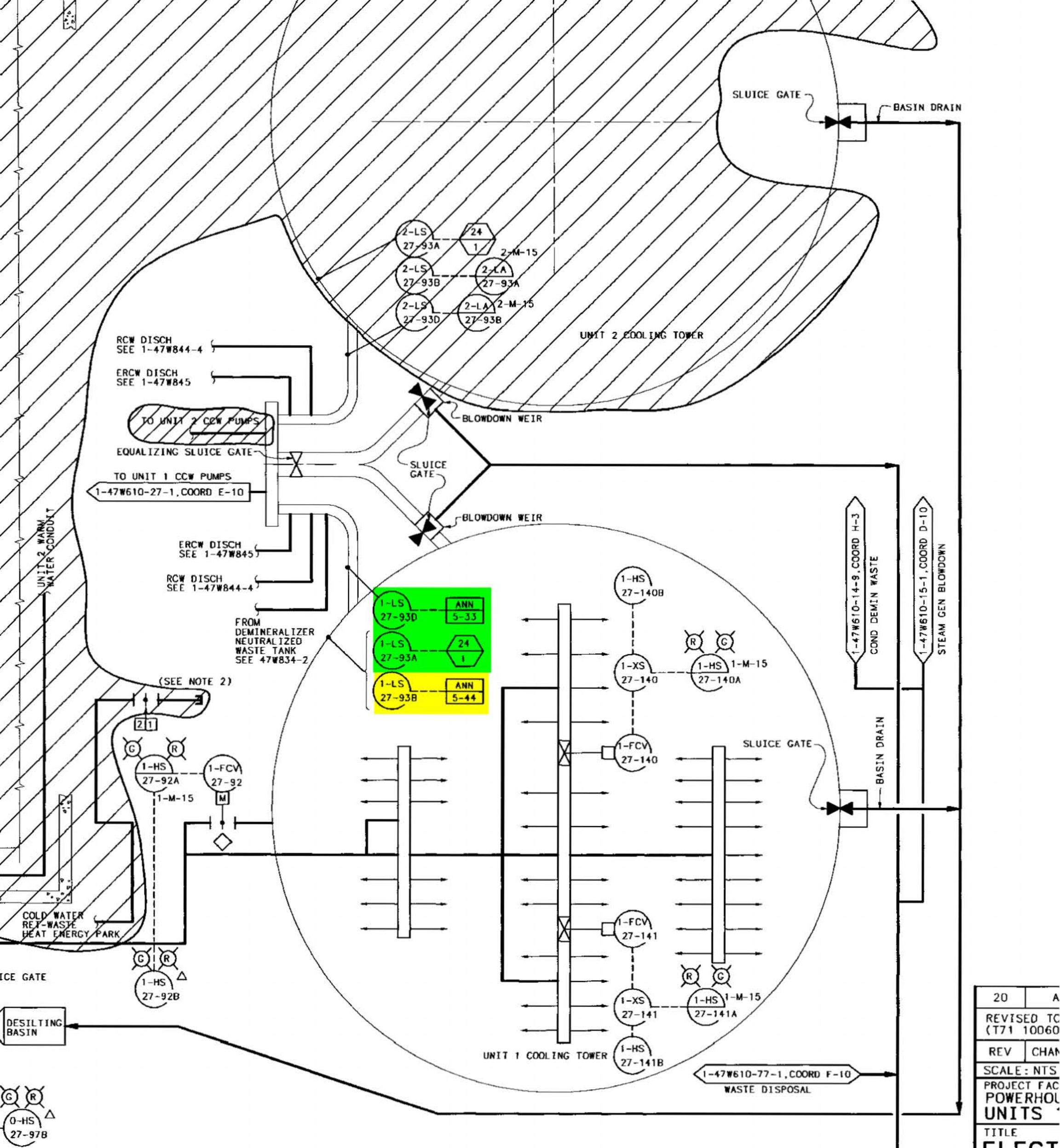
[4] **CHECK** Cooling Tower Trash Racks and **CONTACT** maintenance to clean trash racks if required.

[5] **ADJUST** 0-FCV-27-112, SUPPLEMENTAL CCW SYSTEM SUPPLY FLOW CONTROL, as necessary to control basin level:
- **REFER TO** SOI-27.03, Supplemental Condenser Circulating Water System.

[6] **ESTABLISH** CT blowdown per SOI-27.01, Condenser Circulating Water System.

References:

1-47W610-27-2
SOI-27.01
SOI-27.03



SLUICE GATE
BASIN DRAIN
2-LS 27-93A
24 1
2-M-15
2-LS 27-93B
2-LA 27-93A
2-LS 27-93D
2-LA 27-93B
2-M-15
UNIT 2 COOLING TOWER
RCW DISCH SEE 1-47W844-4
ERCW DISCH SEE 1-47W845
TO UNIT 2 CCW PUMPS
EQUALIZING SLUICE GATE
BLOWDOWN WEIR
SLUICE GATE
TO UNIT 1 CCW PUMPS
1-47W610-27-1, COORD E-10
BLOWDOWN WEIR
ERCW DISCH SEE 1-47W845
RCW DISCH SEE 1-47W844-4
UNIT 2 WARM WATER CONDUIT
FROM DEMINERALIZER NEUTRALIZED WASTE TANK SEE 47W834-2
1-LS 27-93D
ANN 5-33
1-LS 27-93A
24 1
1-LS 27-93B
ANN 5-44
(SEE NOTE 2)
2 1
1-HS 27-140B
R G
1-XS 27-140
1-HS 27-140A
1-M-15
1-FCV 27-140
1-47W610-14-9, COORD H-3
COND DEMIN WASTE
1-47W610-15-1, COORD D-10
STEAM GEN BLOWDOWN
SLUICE GATE
BASIN DRAIN
G R
1-HS 27-92A
1-M-15
1-FCV 27-92
M
COLD WATER RET-WASTE HEAT ENERGY PARK
ICE GATE
G R
1-HS 27-92B
DESILTING BASIN
1-FCV 27-141
R G
1-XS 27-141
1-HS 27-141A
1-M-15
UNIT 1 COOLING TOWER
1-HS 27-141B
1-47W610-77-1, COORD F-10
WASTE DISPOSAL
G R
0-HS 27-97B
20
REVISED TO
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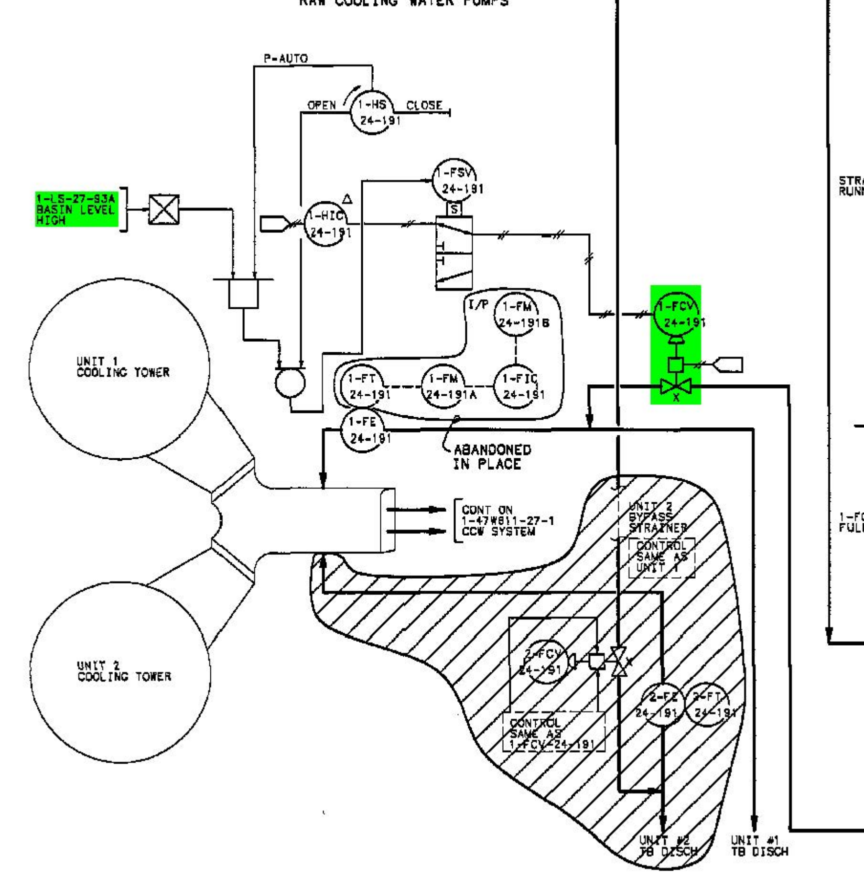
REV
CHAN
SCALE: NTS
PROJECT FAC
POWERHOU
UNITS
TITLE



RAW COOLING WATER PUMPS
P-AUTO
OPEN
1-HS 24-191
CLOSE
1-FSV 24-191
S
1-LS-27-93A BASIN LEVEL HIGH
1-HIC 24-191
1-FCV 24-191
I/P
1-FM 24-191B
1-FT 24-191
1-FM 24-191A
1-FIC 24-191
1-FE 24-191
X
UNIT 1 COOLING TOWER
ABANDONED IN PLACE
CONT ON 1-47W811-27-1 CCW SYSTEM
UNIT 2 BYPASS STRAINER
CONTROL SAME AS UNIT 1
UNIT 2 COOLING TOWER
2-FCV 24-191
2-FE 24-191
2-FT 24-191
CONTROL SAME AS 1-FCV-24-191
UNIT #2 TB DISCH
UNIT #1 TB DISCH