Draft Submittal
(Pink Paper)

Reactor Operator Written Exam

VOGTLE MAY 2006 RETAKE EXAM

05000424/2005302 AND 05000425/2005302

WRITTEN ONLY ON
MAY 1, 2006

2/9/06

U.S. Nuclear Regulatory Commission
Site-Specific RO Written Examination

Applicant Information	
Name:	
Date:	Facility/Unit: Vogtle Nuclear Plant
Region: II	Reactor Type: W
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination, you must achieve a final grade of at least 80.00 percent. Examination papers will be collected 6 hours after the examination begins.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results		
Examination Value	______	Points
Applicant's Score	______	Points
Applicant's Grade	______	Percent

1. 001A1.06 001/2/2/C RODS - RX POWER/C/A - 4.1/NEW/RO/NRC RO/TNT/DSS

The first reactor startup after a refueling outage is in progress, UOP-12004-C, "Power Operation (Mode 1) is in effect.

The UOP limitations reference procedure 88073-C, "Limitations and Conditions for Fuel Operation"

Which **ONE** of the following **CORRECTLY** describes the limits placed on power ascension and control rod withdrawal in accordance with procedure ~~88073-C~~ ?
87073 ✓

A.✓ 0 to 30 % power - no limit on target rate ramp.
30 to 100% power - target rate ramp limited to 3% per hour is desirable.
Control rod withdrawal - limited to 3 steps per hour above 50% power.

B. 0 to 30 % power - target rate ramp limited to 3% per hour is desirable.
30 to 100% power - target rate ramp limited to 8% per hour.
Control rod withdrawal - limited to 3 steps per hour above 50% power.

C. 0 to 30 % power - no limit on target rate ramp.
30 to 100% power - target rate ramp limited to 8% per hour is desirable.
Control rod withdrawal - no limit on rod withdrawal rate.

D. 0 to 30 % power - target rate ramp limited to 3% per hour is desirable.
30 to 100% power - target rate ramp limited to 8% per hour.
Control rod withdrawal - ~~no limit~~ on rod withdrawal rate.

C+D look implausible - licensee to confirm in a procedure, or change (6 steps per hour?) ✓

1. 001A1.06 001/2/2/C RODS - RX POWER/C/A - 4.1/NEW/RO/NRC RO/TNT/DSS

K/A

001 Control Rod Drive System

A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CRDS contols including:

Reactor Power

K/A MATCH ANALYSIS

Question gives a plausible scenario where the first reactor startup following a refueling outage is in progress. Procedure 88073 requires limits on withdrawal of control rods and reactor power ramp rate limits. First startup after refueling would require 3% per hour rate limit after power > 30%, no ramp rate limits apply for power < 30%. Control rod withdrawal rate is 3 steps per hour when power > 50%.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. No limit on ramp rate < 30%, 3% per hour from 30-100%, rods 3 steps per minute withdrawal limit.

B. Incorrect. Plausible, candidate may be familiar with the 3% ramp rate limit and think it applies to < 30% power level. 8% per hour is for conditioned fuel.

C. Incorrect. While there is no ramp rate limit < 30% power, and the 8% per hour is also correct for conditioned fuel but not for unconditioned fuel, there is a rod withdrawal limit of 3 steps per hour above 50% for unconditioned fuel.

D. Incorrect. Plausible, candidate may be familiar with the 3% ramp rate limit and think it applies to < 30% power level. 8% per hour is for conditioned fuel. There is also a 3 step per hour withdrawal limit for the control rods.

REFERENCES

UOP-12004-C, "Power Operations (Mode 1)" limitation 2.2.1.

88073-C, "Limitations and Conditions for Fuel Operation" limitations 2.2 and 2.6, also, steps 5.1.1, Caution prior step 5.1.7, step 5.1.7, step 5.3.1.

VEGP learning objectives:

LO-LP-61202-03 State the reasons for the power change restrictions called for in Procedure 87073-C, "Limitations and Conditions for Fuel Operation".

2.1.4 The average core thermal power level over any eight hours shall not exceed 3565 MWTH (3562 MWTH if excess letdown is in service).

It is permissible to briefly exceed 3565 MWTH by as much as 2% for as long as 15 minutes. In addition, lesser power excursions for longer periods are allowed (i.e., 1% excess for 30 minutes, 1/2% excess for one hour). In no case should 102% (full steady state) power be exceeded.

There are no limits on the number of times these excursions may occur, or the time interval that must separate such excursions, however, the recommendation regarding the eight hour average power will prevent abuse of this allowance.

2.1.5 If steam back leakage into Main Feed Water is suspected during operation of MFIVs as evidenced by water hammer, then the Main Feed Water Isolation valve to the affected Steam Generator should be immediately closed. (1985303297, 1991321521)

2.1.6 Core reactivity should be carefully controlled. If the reactor is driven below the Point of Adding Heat due to a significant negative reactivity insertion, the reactor should be placed in a stable shutdown condition.

2.1.7 Anytime the Power System Stabilizers (PSS) on both Units are out of service, and a system transmission line is out of service, Turbine load on both units should be reduced per 13830, "Main Generator Operation", Table 1.

2.2 **LIMITATIONS**

2.2.1 All power increases above 30% power shall be performed in accordance with 87073-C, "Limitations and Conditions for Fuel Operation." (1997335153)

2.2.2 The Axial Flux Difference (AFD) shall be maintained within the limits specified in the COLR. (TS LCO 3.2.3)

2.2.3 In Modes 1 and 2, the control banks shall be within the insertion, sequence and overlap limits specified in the COLR. (TS 3.1.6). (1985303130)

2.2.4 In Modes 1 and 2, all shutdown and control rods shall be operable with their individual indicated rod positions within ±12 steps of their group step counter demand position. (TS LCO 3.1.4) (1985303130)

1.0 PURPOSE

To provide limitations and conditions for fuel operation to reduce fuel cladding failures and the resulting increased reactor coolant system radionuclide activity.

2.0 PRECAUTIONS AND LIMITATIONS

2.1 The probability of fuel cladding failures during startup, as evidenced by increased reactor coolant activity, increases with the magnitude of startup ramp rates above a ramp rate of 3% per hour until the fuel is conditioned.

2.2 For unconditioned fuel, the ramp rate limitations apply when power is greater than or equal to 30% of full power. A target ramp rate of 3% per hour is desirable, but ramp rates in excess of 3% per hour sometimes unintentionally occur and the following requirements shall be met:

The rate of reactor power increase between 30% and 100% of full power should be less than or equal to 3% power in an hour, but shall not exceed an increase of:

4% over any 1 hour period
7% over any 2 hour period
10% over any 3 hour period

No single step increase in power shall exceed 3% full reactor power.

2.3 Nuclear instrument, Delta-T or heat balance indications shall be used to monitor ramp rates. The most limiting indication should be utilized to assure compliance with the limits.

2.4 These limitations and conditions for fuel operation are strictly applicable only to fuel supplied by Westinghouse Electric Corporation and are based on Reference 8.4.

2.5 Technical Specification and procedure changes that may affect the fuel shall be discussed with Westinghouse Nuclear Fuel Division and SNC Nuclear Fuels Department prior to implementation.

2.6 There are no ramp rate restrictions on conditioned or unconditioned fuel from 0 to 30% of full power.

2.7 If the RCS chemistry indicates that there is failed fuel, follow the guidance in procedure 87085-C, "Fuel Integrity Monitoring and Failed Fuel Action Plan".

3.0 TEST EQUIPMENT

NONE

4.0 PREREQUISITES AND INITIAL CONDITIONS

NONE

5.0 INSTRUCTIONS

5.1 FUEL LIMITATIONS AND CONDITIONS AT BEGINNING-OF-CYCLE

5.1.1 The rate of reactor power increase shall be limited to 3% of full power in an hour between 30% and 100% of full power. This ramp rate requirement applies during the initial startup of each cycle for that period of time until full power is achieved for 72 cumulative hours in a seven-day operating period at power. Time below 30% full power is not considered as operating time at power.

5.1.2 The requirement of Step 5.1.1 can be removed for reactor power levels at or below a given P ($30\% < P \leq 100\%$) provided the plant has operated at or above level P for at least 72 cumulative hours in a seven day operating period at power.

5.1.3 In Steps 5.1.1 and 5.1.2 it is preferable that the 72 hours of cumulative operation at power level P be continuous.

5.1.4 If continuous operation at power level P is impractical, with the permission of the Shift Manager, the 72 hours of cumulative operation at power level P should be obtained in a seven day operating period at some continuous power at or above 30% full power.

5.1.5 If a seven day operating period at some continuous power at or above 30% full power is impractical, the 72 hours of cumulative operation at power level P should be obtained in any seven day operating period at some power at or above 30% full power that need not be continuous but rather could be generated from a longer interrupted power history so long as the duration of the interrupted power history is reasonable as determined by the Engineering Supervisor.

5.1.6 On attaining some steady state power level for 72 hours, the plant may perform load follow operation up to that power level without fuel related limitations on ramp rates except that Subsections 5.2 and 5.3 must still be met where applicable.

NOTE

The purpose of the above requirements on ramp rate is to limit local power increases. Therefore, control rod motion during the initial return to power should be kept to a minimum.

5.1.7 The rate of rod withdrawal shall be limited to 3 steps per hour above 50% of full power where rod withdrawal may occur concurrently with power increases up to the 3% per hour requirement. Once the control rods have been withdrawn to some position at a given power level, during subsequent maneuvers there is no restriction on rod withdrawal to the previous position up to that power level.

5.1.8 Examples of ramp rate situations at beginning-of-cycle are shown in Figure 1. The vertical segments on the diagrams represent ramp rate limits for fully conditioned fuel.

5.2 FUEL LIMITATIONS AND CONDITIONS AFTER EXTENDED LOW-POWER OPERATION

5.2.1 Steps 5.2.2 and 5.2.3 only apply when power has been reduced for a period of time longer than 27 days.

5.2.2 After the beginning-of-cycle startup requirements in Subsection 5.1 are met, the rate of reactor power increases above the highest power level maintained for at least 72 cumulative hours during the preceding thirty days of operation at power shall be limited to 3% of full power in an hour, but shall not exceed an increase of 10% over any three hour period and no single step increase of 10% full reactor power. The rate of rod withdrawal specified in 5.1.8 shall be applied concurrently with the ramp rate limit. Time below 30% full power is not considered as operating time and is not counted toward the thirty day limit.

5.2.3 After the beginning-of-cycle startup requirements in Subsection 5.1 are met, the reactor power increase can be accomplished by a single step increase less than or equal to 10% of full power followed by a maximum ramp rate of 3% of full power in an hour beginning three hours after the step increase. Time below 30% full power is not considered as operating time and is not counted toward the thirty day limit.

5.2.4 If a thirty day operating period at some continuous power is impractical, the 72 cumulative hours of operation at the highest power should be obtained in a thirty day operating period at some power that need not be continuous but rather could be generated from a longer interrupted power history so long as the duration of the interrupted power history is reasonable as determined by the Engineering Supervisor.

5.2.5 Examples of ramp rate situations after extended low-power operation are shown in Figure 2. The vertical segments on the diagrams represent ramp rate limits for fully conditioned fuel.

5.3 FUEL LIMITATIONS AND CONDITIONS AFTER REACTOR SHUTDOWN, TRIP OR DERATE

NOTES

a. The following return-to-power ramp rates are applicable only after the fuel has been fully conditioned as specified in Subsection 5.1

b. The following return-to-power ramp rates are not applicable immediately after reduced power operation for a period of time longer than 27 days which are specified in Subsection 5.2.

c. There are no ramp rate restrictions from 0 to 30% of full power.

5.3.1 Return-to-power ramp rates shall be limited to 8% of full power per hour without specific approvals.

5.3.2 Return-to-power ramp rates shall be limited to 16% of full power per hour with the permission of the General Manager.

5.3.3 Return-to-power ramp rates, after two trips from greater than 50% of full power within a 24 hour period shall be limited to no more than 4% of full power per hour.

5.3.4 Return-to-power ramp rates, after any shutdown of duration greater than or equal to 21 days, shall be limited to 3% of full power per hour.

6.0 RESTORATION

NONE

7.0 ACCEPTANCE CRITERIA

NONE

8.0 REFERENCES

8.1 Fuel Contract Between Southern Nuclear Operating Company and Westinghouse Electric Corporation

2. 003K3.02 001/2/1/RCP - SG/C/A - 3.5/BANK/RO/NRC RO/TNT/DSS

The Unit is at 100% power:

* RCP # 1 trips on overcurrent due to an apparent shaft seizure.

* A reactor trip occurs.

Several minutes after the trip the BOP announces that SG # 1 level is different from the other 3 SGs ?

SG # 1 level is......

A. lower than the other 3 SGs due to SG # 1 shrink when the RCP tripped.

B.✓ higher than the other 3 SGs due to less heat removal from SG # 1.

C. higher than the other 3 SGs due to SG # 1 swell when the RCP tripped.

D. lower than the other 3 SGs due to the ~~rise~~ in Tcold on loop # 1. ← Implausible

decrease ✓

2. 003K3.02 001/2/1/RCP - SG/C/A - 3.5/BANK/RO/NRC RO/TNT/DSS

K/A

003 Reactor Coolant Pump System (RCPS)

K3.02 Knowledge of the effect that a loss or malfunction of the RCPS will have on the following:

S/G

K/A MATCH ANALYSIS

Question gives a plausible scenario where a reactor coolant pump trip from power has resulted in a reactor trip. Candidate must identify the proper SG level response on the affected versus the unaffected SG.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Effects of shrink are minimal.

B. Correct. Less heat input means less evaporation of SG contents, therefore with same feed, higher level.

C. Incorrect. Effects of SG swell are minimal, but in this case, there is no SG swell.

D. Incorrect. Delta T on the idle loop will go to 0, but Tcold will be at no load temp..

REFERENCES

Beaver Valley October 2004 NRC RO exam question # 2

VEGP learning objectives:

LO-LP-60305-09 Given the entire AOP, describe:

a. Purpose of selected steps.
b. How and why the step is being performed.
c. Expected response of the plant/parameter(s) for the step.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003 K3.02	
	Importance Rating	3.5	

Knowledge of the effect that a loss or malfunction of the RCPS will have on the following: S/G

Proposed Question: Common 2

Given the following conditions:

- The Unit is at 100% power with all systems in NSA.
- RC-P-1A, Reactor Coolant Pump has tripped on overcurrent due to an apparent shaft seizure.
- A reactor trip occurs.

Which ONE of the following describes SG response to the event 3 (THREE) minutes following the trip?

SG "1A" level is...

A. Lower than "1B" and "1C" SG due to SG shrink when the RCP tripped.

B. Higher than "1B" and "1C" SG due to less heat removal from "1A" SG.

C. Higher than "1B" and "1C" SG due to SG swell when the RCP tripped.

D. Lower than "1B" and "1C" SG due to the rise in Tcold on the idle loop.

Proposed Answer: **B**

Explanation (Optional):

A. Incorrect. Effects of shrink are minimal.
B. Correct. Less heat input means less evaporation of SG contents, therefore, same feed, higher level.
C. Incorrect. Effects of SG swell are minimal, but in this case, there is no SG swell.
D. Incorrect. Delta T on the idle loop will go to 0, but Tcold will be at no-load temperature.

Technical Reference(s): T&AA Simulator (Attach if not previously provided)

Goes 3 with Q# 2

3. 004A4.11 001/2/1/CVCS -RCP SEAL INJ/C/A - 3.4/NEW/RO/NRC RO/TNT/DSS

Which **ONE** of the following actions would be **CORRECT** to take using HC-0182 given the ~~following~~ provided seal injection flow indication ?

A. seal injection flows - 13.8 gpm per RCP, depress the red UP arrow to throttle open HV-0182 to lower seal injection flow.

B.✓ seal injection flows - 7.4 gpm per RCP, depress the red UP arrow to throttle closed HV-0182 to raise seal injection flow.

C. seal injection flows - 7.4 gpm per RCP, depress the green DOWN arrow to throttle open HV-0182 to raise seal injection flow.

D. seal injection flows - 13.8 gpm per RCP, depress the green DOWN arrow to throttle closed HV-0182 to lower seal injection flow.

3. 004A4.11 001/2/1/CVCS -RCP SEAL INJ/C/A - 3.4/NEW/RO/NRC RO/TNT/DSS

K/A

004 Chemical and Volume Control System

A4.11 Ability to manually operate and / or monitor in the control room

RCP seal injection

K/A MATCH ANALYSIS

Question gives an out of spec RCP seal injection flow rate and asks for corrective actions the RO should take to bring the flow rate back within limits.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. UP arrow raises flow by closing valve.

B. Correct. UP arrow raises flow by closing valve.

C. Incorrect. DOWN arrow lowers flow by opening valve.

D. Incorrect. DOWN arrow lowers flow by opening valve.

REFERENCES

Vogtle Text Chapter # 9 vor CVCS pages # 21 and # 26

SOP-13003-1/2, Reactor Coolant Pump Operation" limitation 2.2.3.

VEGP learning objectives:

LO-PP-16401-03 Describe the control room indications for a failure of a RCP seal.

2.1.5 When starting the first RCP with a bubble in the Pressurizer, the additional RCP heat input may cause an insurge of cooler RCS water into the pressurizer. Surge line temperature may be controlled by monitoring surge line temperature and adjusting RHR cooling and charging flow to ~~ensure~~verify a net outsurge from the pressurizer.

2.1.6 With Westinghouse and Operations management approval, RCPs may be started without ACCW flow to perform 30 second and 1 minute air sweeps per 13001, "Reactor Coolant System Filling and Venting" or to verify proper rotation following electrical maintenance (less than 1 minute). General Manager approval will be required for starting RCPs without ACCW for any other operation. Operation without ACCW in service for more than 10 minutes is prohibited.

2.1.7 Seal Injection flow should be maintained to coupled RCPs when RCS level is greater than the 190 foot elevation, however, if necessary, seal injection may be secured to RCPs above the 190 foot elevation provided RCS level is maintained constant.

2.1.8 RCPs should not be uncoupled and placed on their back seat until the RCS is depressurized and vented.

2.2 **LIMITATIONS**

2.2.1 If seal injection is not in service and the reactor coolant temperature is greater than 150°F, Auxiliary Component Cooling Water shall be supplied to the thermal barrier.

2.2.2 When the reactor coolant pressure is less than 100 psig, the No. 1 Seal Leakoff Valves should be closed.

2.2.3 The RCP seal injection flow should be maintained greater than 8 gpm and less than 13 gpm any time seal injection is required.

2.2.4 With the reactor coolant temperature greater than 400°F, the seal injection temperature should be maintained less than 135°F.

Charging Flow Control Valve FV-121

The discharge of the NCP and CCPs combine into a single charging flow path. Charging flow is controlled by the position of FV-121. In automatic, FV-121 controls the total flow directed toward the normal charging header. The position of FV-121 is determined by the output error signal from pressurizer level controller FIC-121. This error signal is determined by the difference between pressurizer program level(determined by auctioneered high Tavg) and actual pressurizer level. If an output error signal indicates that pressurizer level is below program level, FV-121 will open to provide more charging flow to eliminate that error signal. Conversely, if pressurizer level is above program level, then FV-121 valve position will throttle more closed to lower charging flow. Charging flow controller FIC-121 can be operated locally in the Auxiliary Building. Charging flow indication is provided in the control room on panel A and C.

Seal Flow Control Valve, HV-182

This hand controller air-operated valve in the charging header maintains sufficient backpressure in the charging header to ensure adequate flow of seal water to the reactor coolant pumps. The flow indicators, (FI-142, -143, -144, and -145) for each RCP seal injection are used to adjust the setting of this valve so that approximately 8-13 gpm seal injection flow is maintained to each RCP. The valve is manually controlled from the main control board. The valve fails open on loss of power or air. If more seal injection flow is required, the operator depresses the UP pushbutton on HC-182. This causes HV-182 to be in a more shut position, thus forcing more flow toward the RCP seal injection line. This has the immediate effect of lowering charging flow directed toward the normal charging header. With letdown flow the same and now less charging flow going through the normal charging header, letdown temperature out of the regenerative heat exchanger will increase. Consequently, anytime seal injection flow is adjusted, the effect on letdown parameters must be evaluated. Vice versa, anytime charging flow is changed, the effect on seal injection and letdown parameters must be evaluated also and appropriate actions to restore system parameters to their normal operating band.

Normal Charging header Isolation Valves, HV-8106, HV-8105

Charging flow that does not flow toward the seal package(based on the position of HV-182) flows past HV-182 toward the RCS penetration past series charging isolation valves HV-8106(Train A) and HV-8105(Train B). These valves are operated from the main control room and from the shutdown panels. Each valve is powered from a 1E MCC. On a Safety Injection actuation signal, these valves receive a CLOSE signal from their respective train related SI signal. This isolates the normal charging header and allows the safety-related CCPs to direct their flow through the BIT into all RCS cold legs.

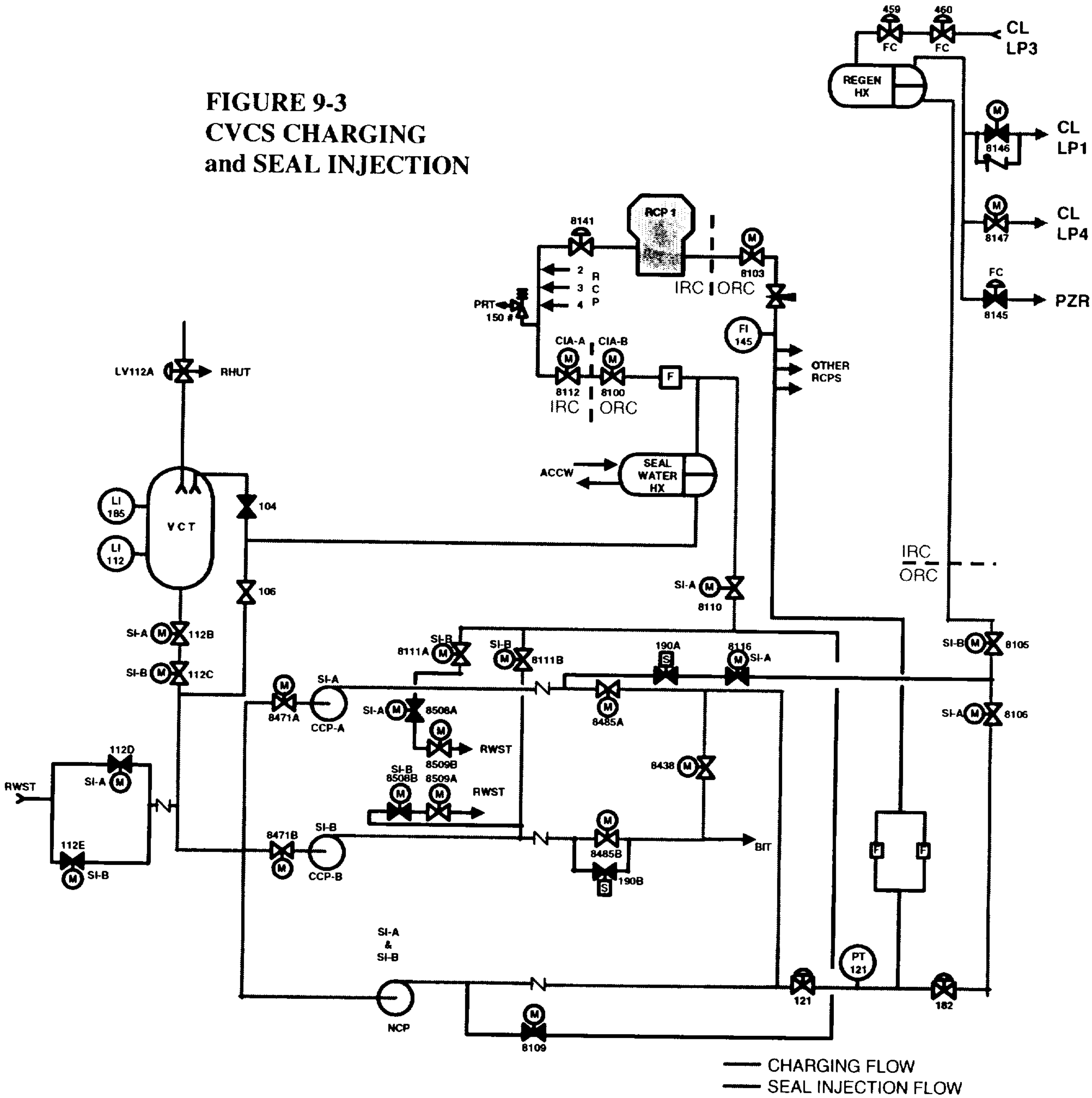
Normal/Alternate Charging to RCS Isolation Valve (HV-8146 and HV-8147)

Control switches for these motor-operated isolation valves are located on the QMCB and the Remote Shutdown panels. These switches are two position (Close/Open). An additional switch Control Room/Local transfer switch is located on the Remote Shutdown panel. This switch must be in the Control Room position to enable the QMCB switches. To equalize thermal stresses, SOP 13006-1/2 states that normal charging valve HV-8146 should be in service during even numbered fuel cycles and alternate charging valve HV-8147 should be used during odd-numbered fuel cycles. This should be performed at cold shutdown conditions to avoid thermal transients.

CVCS Pressurizer Auxiliary Spray Valve (HV-8145)

This air-operated isolation valve is operated from the QMCB and the Remote Shutdown panel. These switches are two position (Close/Open). There is also a through the seal leak off isolation valves (HV-8141A, B, C, and D), through a motor-operated isolation valve (HV-8112), and then exits the containment building. The seal return flow immediately passes through a second motor-operated isolation valve (HV-8100) upon exiting the containment. Both of these motor-operated isolation valves serve to isolate the containment upon receiving a Containment Isolation Actuation(CIA) signal. Seal water return flow next passes through the seal water return filter which removes any insoluble material picked up as the seal water passed through the reactor coolant pump seals. It is then reduced in temperature from approximately 175°F to 130°F as it passes through the tube side of the seal water heat exchanger before returning through an isolation valve to the suction header of the charging pumps.

FIGURE 9-3
CVCS CHARGING
and SEAL INJECTION



4. 004K6.17 001/2/1/CVCS - E. BOR PATHS/C/A - 4.4/BANK/RO/NRC RO/TNT/DSS

Emergency boration is required on Unit 1 and 1HV-8104 **CANNOT** be opened. An alternate Emergency Boration flow path must be established. Per procedure, which **ONE** of the following flow paths would **NOT** qualify ?

procedure title & #

A.✓ Through 1LV-112D & E, using a CCP, discharging through 1FCV-121 at 89 gpm.

B. Through 1FV110A & B, at a boric acid flowrate of 37 gpm, using the NCP at 86 gpm to the normal charging path.

C. Through 1LV-112D & E, using a CCP, discharging through the regenerative Hx at a flowrate of 103 gpm.

D. Through 1FV-110A & B, at a boric acid flowrate of 32 gpm, using the NCP at 51 gpm to the normal charging path.

4. 004K6.17 001/2/1/CVCS - E. BOR PATHS/C/A - 4.4/BANK/RO/NRC RO/TNT/DSS

K/A

004 Chemical and Volume Control System

K6.1.7 Knowledge of the effect of a loss or malfunction on the following CVCS components.

Flow paths for Emergency Boration

K/A MATCH ANALYSIS

Question gives several various emergency boration flow path possibilities and the candidate has to choose the one that does NOT meet requirements.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. Charging flow has to be >100 gpm through this flow path.

B. Incorrect. This path meets requirements.

C. Incorrect. This path meets requirements.

D. Incorrect. This path meets requirements.

REFERENCES

SOP-13009-1/2, "CVCS Reactor Makeup Control System", section 4.9 for Emergency Boration

Vogtle Initial Exam Bank question # LO-OR--09401-04-002

VEGP learning objectives:

LO-PP-09300-06 Describe for all emergency boration flow paths

a. borated water sources and discharge flow paths

4.9 **EMERGENCY BORATION**

NOTE

> Table 1 provides a convenient tool for checking emergency boration flow path alternatives.

4.9.1 **Emergency Boration Through 1-HV-8104**

4.9.1.1 Start one Boric Acid Transfer Pump. [ ]

4.9.1.2 Ensure a Charging Pump is running. [ ]

4.9.1.3 Open EMERGENCY BORATE valve 1-HV-8104. [ ]

NOTE

> The following step assumes that with 12 gpm of seal return, 30 gpm will be supplied to the RCS.

4.9.1.4 Place 1-FIC-0121 in MANUAL. [ ]

4.9.1.5 Adjust 1-FIC-0121 to maintain flow greater than 42 gpm. [ ]

4.9.1.6 Verify emergency boration flow 1-FI-0183A greater than 30 gpm. [ ]

4.9.1.7 If flow is less than 30 gpm, start the second Boric Acid Transfer Pump. [ ]

4.9.1.8 Operate the Pressurizer Backup Heaters as necessary in order to equalize boron concentration between the RCS and the Pressurizer. [ ]

4.9.1.9 Observe that plant conditions are consistent with the boration of the RCS: [ ]

RCS Tavg may be dropping,

NIS may be dropping.

4.9.1.10 Determine the amount of boric acid required to allow termination of emergency boration. [ ]

4.9.1.11 When the determined amount of boric acid has been added to the RCS, close 1-HV-8104. [ ]

4.9.1.12 Return the Boric Acid Transfer Pumps to the desired system configuration. [ ]

4.9.1.13 Restore 1-FIC-0121 to the AUTO position. [ ]

4.9.1.14 Direct Chemistry to sample and report the RCS boron concentration, or monitor the Boron Meter 1-AI-40134 if available. [ ]

4.9.2 **Emergency Boration Through The Normal Charging Flow Path**

4.9.2.1 Start one Boric Acid Transfer Pump. [ ]

4.9.2.2 Ensure a Charging Pump is running. [ ]

4.9.2.3 Open the following:

- 1-FV-0110A BA TO BLENDER, [ ]
- 1-FV-0110B BLENDER OUTLET TO CHARGING PUMPS SUCT. [ ]

NOTE

> The following step assumes that with 12 gpm of seal return, 30 gpm will be supplied to the RCS.

4.9.2.4 Place 1-FIC-0121 in MANUAL. [ ]

4.9.2.5 Adjust 1-FIC-0121 to maintain flow greater than 42 gpm. [ ]

4.9.2.6 Verify emergency boration flow greater than 30 gpm as indicated by 1-FI-0110A. [ ]

4.9.2.7 If flow is less than 30 gpm, start the second Boric Acid Transfer Pump. [ ]

4.9.2.8 Operate the Pressurizer Backup Heaters as necessary in order to equalize boron concentration between the RCS and the Pressurizer. [ ]

4.9.2.9 Observe that plant conditions are consistent with the boration of the RCS: [ ]

RCS Tavg may be dropping,

NIS may be dropping.

4.9.2.10 Determine the amount of boric acid required to allow termination of emergency boration. [ ]

4.9.2.11 When the determined amount of boric acid has been added to the RCS, close 1-FV-0110A and 1-FV-0110B. [ ]

4.9.2.12 Establish automatic makeup per Section 4.1 of this procedure. [ ]

4.9.2.13 Restore 1-FIC-0121 to the AUTO position. [ ]

4.9.2.14 Direct Chemistry to sample and report the RCS boron concentration or monitor the Boron Meter 1-AI-40134 if available. [ ]

4.9.3 **Emergency Boration From The RWST**

4.9.3.1 Ensure one Charging Pump is running and supplied with cooling water. [ ]

4.9.3.2 Open Charging Pump Suctions from the RWST 1-LV-0112D and 1-LV-0112E. [ ]

4.9.3.3 Close VCT Outlet Isolations 1-LV-0112B and 1-LV-0112C. [ ]

4.9.3.4 Place 1-LV-0112A to the HUT position. [ ]

4.9.3.5 If the charging flow path through the Regenerative Heat Exchanger is in service:

a. Place 1-FIC-0121 in MANUAL, [ ]

b. Adjust Charging Line Flow Controller 1-FIC-0121 to obtain Charging Flow 1-FI-0121C greater than 100 gpm, [ ]

c. Adjust Charging Seal Flow Control 1-HV-0182 as necessary to maintain RCP seal injection flow at approximately 40 gpm (between 8 and 13 gpm per pump). [ ]

4.9.3.6 If the charging flow path through the Regenerative Heat Exchanger is not in service:

a. Ensure open BIT DISCH ISOLATION valves 1-HV-8801A and 1-HV-8801B, [ ]

b. Ensure BIT Flow (1-FI-0917A) plus total seal injection flow, less total seal return flow is greater than 87.5 gpm, [ ]

c. Adjust Charging Line Flow Controller 1-FIC-0121 as necessary to maintain RCP seal injection flow at maximum flow less than 13 gpm per pump. [ ]

4.9.3.7 If required for RCS inventory control, place an additional letdown orifice in service. [ ]

4.9.3.8 Operate the Pressurizer Backup Heaters as necessary in order to equalize boron concentrations between the RCS and the Pressurizer. [ ]

4.9.3.9 Observe for indications consistent with boration of the RCS: [ ]

RCS Tavg may be dropping,

NIS may be dropping.

4.9.3.10 When boration is complete:

a. Open VCT OUTLET ISOLATION valves 1-LV-0112B and 1-LV-0112C, [ ]

b. Close Charging Pump Suctions from the RWST 1-LV-0112D and 1-LV-0112E, [ ]

c. Place 1-HS-0112A to the AUTO position, [ ]

d. Restore 1-FIC-0121 to the AUTO position if it was placed in MANUAL. [ ]

4.9.3.11 Direct Chemistry to sample and report the RCS boron concentration or monitor Boron Meter 1-AI-40134 if available. [ ]

QUESTIONS REPORT
for Questions

LO-OR-09401-04 002/LOLP09401/LO-TA-09029/004A4.09/////

Emergency boration is required on Unit 1, and 1HV-8104 can NOT be opened. An alternate Emergency Boration flowpath must be established. Per procedure, which ONE of the following flowpaths would NOT qualify?

A.✓ Through 1LV-112D & E, using a CCP, discharging through 1FCV-121 at greater than 87.5 gpm.

B. Through normal makeup, at a boric acid flowrate of greater than 42 gpm, using the NCP at 85 gpm charging.

C. Through 1LV-112D & E, using a CCP, discharging through the Regenerative Hx at a flowrate greater than 100 gpm.

D. Through normal makeup, at a boric acid flowrate of greater than 30 gpm, using the NCP at 50 gpm charging.

Goes with question # 4

5. 005K2.03 001/2/1/RHR - RCS ISOL/MEM - 2.7/MODIFIED/RO/NRC RO/TNT/DSS

Regarding the two following valves:

* HV-8701B, RHR PMP-A UPSTREAM SUCTION FROM HOT LEG LOOP 1

* HV-8702B, RHR PMP-B UPSTREAM SUCTION FROM HOT LEG LOOP 4

Which **ONE** of the following **CORRECTLY** describes the power supplies to the RHR upstream loop suction isolation valves ?

A.✓ 125V DC Inverter CD1I5 and 480V 1E MCC BBE

B. 480V 1E SWGR AB05 and 125V DC Inverter DD1I6

C. 480V MCC NBE and 125V DC Inverter DD1I4

D. 125V DC Inverter CD1I3 and 480V 1E SWGR BB07

5. 005K2.03 001/2/1/RHR - RCS ISOL/MEM - 2.7/MODIFIED/RO/NRC RO/TNT/DSS

K/A

005 Residual Heat Removal System

K2.03 Knowledge of bus power supplies to the following.

RCS pressure boundary motor operated valves

K/A MATCH ANALYSIS

Question asks power supplies to RHR Upstream Loop Suction Isolation valves HV-8701B and HV-8702B. HV-8701B is a Train A valve powered from Train A inverter CD1I5, HV-8702B is powered from a 480V 1E MCC BBE.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. HV-8701B powered from CD1I5 125V DC inverter, HV-8702A powered from 480V 1E MCC BBE.

B. Incorrect. This 480V SWGR is a power supply for larger loads such as pumps, DD1I6 is the correct inverter however.

C. Incorrect. This is a non-safety related MCC that does supply non-1E components in the containment but not the RHR valve, this is also an incorrect inverter.

D. Incorrect. This is an incorrect inverter and 480V SWGR power larger loads such as pump motors.

REFERENCES

SOP-13011-1, Residual Heat Removal System, pages 12, 13, 17, and 18.
{This is steps 4.1.4.b.(3), 4.1.7.m, 4.2.3.1.m, 4.2.4.b.(1)}

Arkansas Nuclear One (ANO) February 2005 RO NRC exam question # 30

VEGP learning objectives:

LO-PP-12101-12 Briefly describe the RHR system alignment during normal power operations and during RCS cooldown.

4.1.4 If RHR is being placed in standby for MODE 3 entry, perform the following; (IV REQUIRED):

a. Shutdown Inverter 1CD1I5 per 13405-1, "125V DC 1E Electrical Distribution System," [ ]

b. Open and lock the power supplies to the RHR Loop 1 Inlet Isolations:

(1) Turn off 1ABE-15 for 1-HV-8701A, [ ]

(2) Open the K2 links for breaker 1ABE-15, [ ]

(3) At 1CD1I5N open and lock the disconnect for 1-HV-8701B. [ ]

4.1.5 Verify RHR System Flow Control Valves remain in required standby positions:

a. Close INSTR AIR ISOLATION TO LINE 136 1-2420-U4-151 (RC-85 overhead), [ ]

b. Open INSTR AIR LINE 136 DRAIN 1-2420-U4-152 (RC-89) to bleed off air pressure. [ ]

4.1.6 If depressurization of the RHR System is required, either:

a. Open the following:

(1) SIS CHECK VALVE TEST CNMT ISO 1-HV-8964, [ ]

(2) SIS CHECK VALVE TEST CNMT ISO 1-HV-8871, [ ]

(3) SIS RHR PMP-A CHECK VALVE TEST 1-HV-8890A. [ ]

-OR-

b. Request Chemistry to open RHR TRAIN A SAMPLE VALVE 1-HV-3520. [ ]

4.1.7 If required, when RHR depressurization is complete, close all valves opened in Step 4.1.6; (IV REQUIRED). [ ]

CAUTION

> To preclude thermal binding of RHR PMP-A TO COLD LEG 1&2 ISO VLV 1-HV-8809A, this valve should NOT be closed if it is being throttled to increase the RHR cooldown. All valve throttling will be required to be performed locally and not electrically.

NOTE

> Due to leak-by of the RHR HEAT EXCHANGER OUTLET and BYPASS Valves, it may be necessary to manually throttle RHR PMP-A TO COLD LEG 1&2 ISO VLV 1-HV-8809A to achieve desired temperature during cooldown.

k. If required, slowly throttle RHR PMP-A TO COLD LEG 1&2 ISO VLV 1-HV-8809A to achieve desired RHR temperature, [ ]

l. When RHR Pump A is at desired temperature, stop RHR Pump A, and place 1-HS-0620 in PULL-TO-LOCK, [ ]

NOTE

> The following steps should be performed as soon as possible after stopping RHR PMP-A to prevent system heatup due to system remaining aligned to the RCS.

m. Close the **RHR** PMP-A UPSTREAM SUCTION FROM HOT LEG LOOP 1 Valve 1-HV-8701B, [ ]

n. Close the RHR PMP A DOWNSTREAM SUCTION FROM HOT LEG LOOP 1 1-HV-8701A, [ ]

o. Verify 1-HV-8811A and 1-HV-8804A are closed to satisfy interlocks associated with next step, [ ]

p. Open the RWST TO RHR PMP-A SUCTION 1-HV-8812A, [ ]

q. Place RHR Pump A 1-HS-0620 in AUTO, [ ]

r. If required, open RHR PMP-A TO COLD LEG 1&2 ISO VLV 1-HV-8809A. [ ]

4.1.3.2 Restore power to RHR Train A to CCP Suction as follows:

a. Remove tags and close K2 link for breakers 1ABB-05, [ ]

b. Close 1ABB-05 to Valve 1-HV-8804A. [ ]

4.1.3.3 Align RHR TRN-A for standby per Checklist 3. [ ]

CAUTION

> To preclude thermal binding of RHR PMP-B TO COLD LEG 3&4 ISO VLV 1-HV-8809B, this valve should NOT be fully closed if it is being throttled to increase the RHR cooldown. All valve throttling will be required to be performed locally and not electrically.

NOTE

> Due to leak-by of the RHR HEAT EXCHANGER OUTLET and BYPASS Valves, it may be necessary to manually throttle RHR PMP-B TO COLD LEG 3&4 ISO VLV 1-HV-8809B to achieve desired temperature during cooldown.

k. If required, slowly throttle RHR PMP-B TO COLD LEG 3&4 ISO VLV 1-HV-8809B to achieve desired RHR temperature, [ ]

l. When RHR Pump B is at desired temperature, stop RHR Pump B, and place handswitch 1-HS-0621 in PULL-TO-LOCK, [ ]

NOTE

> The following steps should be performed as soon as possible after stopping RHR PMP-B to prevent system heatup due to system remaining aligned to the RCS.

m. Close the RHR PMP-B UPSTREAM SUCTION FROM HOT LEG LOOP 4 Valve 1-HV-8702B, [ ]

n. Close the RHR PMP B DOWNSTREAM SUCTION FROM HOT LEG LOOP 4 1-HV-8702A, [ ]

o. Verify 1-HV-8811B and 1-HV-8804B are closed to satisfy interlocks associated with next step, [ ]

p. Open the RWST TO RHR PMP-B SUCTION 1-HV-8812B, [ ]

q. Place RHR Pump B Handswitch 1-HS-0621 in AUTO, [ ]

r. If required, open RHR PMP-B TO COLD LEG 3&4 ISO VLV 1-HV-8809B. [ ]

4.2.3.2 Restore power to RHR Train B to SIP Suction as follows:

a. Remove tags and close K2 link for breakers 1BBB-05, [ ]

b. Close 1BBB-05 to Valve 1-HV-8804B. [ ]

4.2.3.3 Align RHR TRN-B for standby per Checklist 4. [ ]

4.2.4 If RHR is being placed in standby for MODE 3 entry, perform the following:

a. Shutdown Inverter 1DD1I6 per 13405-1, "125V DC 1E Electrical Distribution System." ((IV REQUIRED)) [ ]

b. Open and lock the power supplies to the RHR Loop 4 Inlet Isolations:

(1) Open 1BBE-13 for 1-HV-8702B, ((IV REQUIRED)) [ ]

(2) Open K2 links for breaker 1BBE-13, ((IV REQUIRED)) [ ]

(3) At 1DD1I6N, open and lock the disconnect for 1-HV-8702A. ((IV REQUIRED)) [ ]

4.2.5 Verify RHR System Flow Control Valves remain in their required standby positions:

a. Close INSTR AIR ISOLATION TO LINE 136 1-2420-U4-151 (RC-85 overhead), [ ]

b. Open INSTR AIR LINE 136 DRAIN 1-2420-U4-152 (RC-89) to bleed off air pressure. [ ]

4.2.6 If depressurization of the RHR System is required, either:

a. Open the following:

(1) SIS CHECK VALVE TEST CNMT ISO 1-HV-8964, [ ]

(2) SIS CHECK VALVE TEST CNMT ISO 1-HV-8871, [ ]

(3) SIS RHR PMP-B CHECK VALVE TEST 1-HV-8890B. [ ]

<u>OR</u>

b. Have Chemistry open RHR TRAIN B SAMPLE VALVE 1-HV-3521. [ ]

4.2.7 If required, when RHR depressurization is complete, close all valves opened in Step 4.2.6; ((IV REQUIRED)) [ ]

Questions For 2005 ANO UNIT 2 RO/SRO Exam

BANK 0473 **Rev** 0 **Rev Date:** 10/12/2004 **RO Select:** Yes **SRO Select:** Yes **Points:** 1.00
Lic Level: RS **Difficulty:** 2 **Taxonomy:** K **Source:** NEW **Originator** COBLE
10CFR55_41: 41.7 **10CFR55_43:** NA **Section:** 3.4 **Type** RCS HEAT REMOVAL
System RESIDUAL HEAT REMOVAL SYSTEM **System** 005 **K/A:** k2.03
RO Tier: 2 **RO Group:** 1 **RO Imp:** 2.7 **SRO Tier:** 2 **SRO Group:** 1 **SRO Imp:** 2.8
Description Knowledge of bus power supplies to the following: RCS Pressure boundary motor operated valves.

Question # 30

The Shutdown Cooling Motor Operated Suction Isolation Valves, 2CV-5084-1 and 2CV-5086-2, are powered from:

A. Vital 480 Volt Load Centers 2B5 and 2B6.

B. Non Vital 480 Volt Load Centers 2B7 and 2B8.

C. Vital 480 Volt MCCs 2B51and 2B62.

D. Non Vital 480 Volt MCCs 2B31 and 2B41.

Answer:

C. Vital 480 Volt MCCs 2B51and 2B62.

Notes:

These valves are powered from 2B52 and 2B62 so the other distracters are incorrect.
Load centers are where the big 480 volt loads such as large 480 volt motors come from not MOVs.
2B31 and 41 are the Aux Building non vital MCC busses and these valves are safety related vital

References

STM 2-14, Shutdown Cooling System, Sections 2.1.1 and 2.1.2
A2LP-RO-SDC OBJ. 2, Describe the following shutdown cooling system components as stated in STM 2-14 including purpose and design features of components, including automatic features and purpose of automatic features: SDC suction motor operated valves.

Historical

This question has not been used on any previous NRC exams. BNC 10/12/2004.

Goes with #5

6. 006K5.09 001/2/1/ECCS - THERMODYNAMIC/MEM - 3.3/BANK/RO/NRC RO/TNT/DSS

Given the following plant conditions:

* A large break LOCA has occurred.

* Appropriate actions in accordance with 19000-C, "Reactor Trip or Safety Injection", 19010-C, "Loss of Reactor or Secondary Coolant", and 19013-C, "Transfer to Cold Leg Recirculation" have been completed.

* RCS pressure is stable at 50 psig
* All safety related equipment operates as designed
* ECCS is operating in the cold leg recirculation mode

Which **ONE** of the following statements describes the primary method of long term decay heat removal ?

A. Heat removal from Reflux boiling in the S/Gs.

B. Heat transfer between the RCS and the S/Gs due to natural circulation flow.

C. Heat transfer between the RCS and the S/Gs due to forced circulation flow.

D✓ Heat transfer from ECCS flow from CMNT sump and removal of steam/water out of the break.

6. 006K5.09 001/2/1/ECCS - THERMODYNAMIC/MEM - 3.3/BANK/RO/NRC RO/TNT/DSS

K/A

006 Emergency Core Cooling System (ECCS)

K5.09 Knowledge of the operational implications of the following concepts as they apply to ECCS.

Thermodynamics of water and steam, including subcooled margin, superheat, and saturation.

K/A MATCH ANALYSIS

Question poses a scenario after a large break LOCA has occurred and candidate has to identify the correct method of decay heat removal during cold leg recirculation mode.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. S/Gs would be a heat source during a large break LOCA event.

B. Incorrect. S/Gs would be a heat source during a large break LOCA event.

C. Incorrect. S/Gs would be a heat source during a large break LOCA event.

D. Correct. ECCS injection flow from CNMT sump and break flow are cooling the core.

REFERENCES

Watts Bar July 2004 NRC RO examination question # 31

WOG EOP background document for 19013, "Cold Leg Recirculation" pages 11, 13, and 17.

VEGP learning objectives:

LO-PP-13101-02 List the three major accidents that require the need for the ECCS.

CAUTION 1

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

PURPOSE: To alert the operator that SI flow to the RCS must be maintained at all times

BASIS:

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

ACTIONS:

N/A

INSTRUMENTATION:

N/A

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

N/A

STEP: Verify CCW Flow To RHR Heat Exchangers

PURPOSE: To ensure CCW flow to cool the recirculation fluid

BASIS:

This step assumes that the RHR heat exchangers are used for heat removal during the post accident recirculation phase and that either CCW flow has been automatically provided to the heat exchangers or the operator has manually established CCW flow prior to the switchover alarm. If CCW flow had not previously been established, then it should be established at this time.

ACTIONS:

o Determine if CCW flow to RHR heat exchangers has been established
o Establish CCW flow to RHR heat exchangers

INSTRUMENTATION:

Plant specific instrumentation to indicate CCW flow to RHR heat exchangers

CONTROL/EQUIPMENT:

CCW system controls for establishing flow to RHR heat exchangers

KNOWLEDGE:

If CCW cannot be established to one heat exchanger, the remaining guideline can be performed as listed provided that the uncooled recirculation fluid temperature and pressure do not exceed equipment design conditions.

PLANT-SPECIFIC INFORMATION:

Means to verify CCW flow to RHR heat exchanger

CAUTION: Switchover to recirculation may cause high radiation in the auxiliary building.

PURPOSE: To alert the operator of possible high radiation in the auxiliary building

BASIS:

During a LOCA, water from the RCS with higher than normal activity will be transferred from the break in the RCS to the containment sump. When the plant is switched over to the recirculation mode, these higher activity levels may cause higher than normal radiation in the auxiliary building.

ACTIONS:

N/A

INSTRUMENTATION:

N/A

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

N/A

006 K5.09 001

Given the following plant conditions:

- A iarge break LOCA has occurred.
- Appropriate actions in accordance with E-0, 'Reactor Trip or Safety Injection", E-1, "Loss or Reactor or Secondary Coolant", and ES-1.3, "Transfer To Containment Sump" have been completed.
- RCS pressure is stable at 50 psig.
- ECCS is operating in cold leg recirculation mode.

Which ONE of the following statements describes the primary method of long term decay heat removal?

A. Heat transfer from Reflux boiling in the *SIGs*.

B. Heat transfer between the RCS and the *SIGs* due to natural circulation flow.

C. Heat transfer between the RCS and the S/Gs due to forced circulation flow.

D✓ Heat transfer from the injection of water from the containment sump and the removal of steam/water out of the break.

The correct answer is D.

A Incorrect - S/G are heat source in large break LOCA event.

B. Incorrect - S/G are heat source in large break LOCA event

C. Incorrect - S/G are heat source in large break LOCA event.

D. *Correct*- Injection flow and break flow are cooling the core.

REFERENCES:

Lesson plan 3-OT-TAA013 p. 8
UFSAR section 15.4-2

10CFR55.41.5/45.7
Knowledge of the operational implications of the following concepts as they apply to ECCS: Thermodynamics of water and steam, including subcooled margin, superheat, and saturation.

RO-31 SRO-31

Reference:	3-OT-TAA013	K/A Number:	006 K5.09
WAvalue:	3.3	Last Used:	
Level:	2	Source:	RANK
Tier/Grp:	2/1	SRO Only:	

Goes with Q #6

7. 007K5.02 001/2/1/PRT - STEAM BUBBLE/C/A - 2.5/NEW/RO/NRC RO/TNT/DSS

The plant is coming out of a refueling outage and is in the process of performing a vacuum refill of the RCS in accordance with UOP-12009-C, "RCS Vacuum Refill".

Which **ONE** of the following would be **CORRECT** regarding drawing a bubble in the pressurizer after vacuum refill ?

A. PRZR level would be filled to solid conditions, heaters energized until steam space temperature at Tsat, PORVs used intermittently to "burp" the PRZR of non-condensible gases, a letdown/charging mismatch established to lower level to establish a bubble at 70%.

B.✓ PRZR level would be raised to 70%, heaters energized until steam space temperature at Tsat, "burping" of steam space not required to remove non-condensible gases, charging/letdown modulated to control PRZR level at 70%.

C. PRZR level would be raised to 70%, heaters energized until steam space temperature at Tsat, PORVs used to intermittently "burp" the PRZR of non-condensible gases, charging/letdown modulated to control PRZR level at 70%.

D. PRZR level would be filled to solid conditions, heater energized until steam space temperature at Tsat, "burping" of steam space not required to remove non-condensible gases, a letdown/charging mismatch established to lower level to establish a bubble at 70%.

mismatch

8. 008G2.1.11 001/1/1/PRZR VPR - 1 HR TS/C/A 3.0/MODIFIED/RO/NRC RO/TNT/DSS

The controlling pressurizer pressure channel has failed high and the associated PORV has opened. The RO while performing his IOAs was unable to close the affected PORV from the QMCB handswitch.

* The RO closed the associated block valve.

* The PORV remains open even with the PORV handswitch in the hard closed position.

delete

Which **ONE** of the following would be **CORRECT** regarding automatic operation of the PORV block valve and Tech Spec LCO actions required ?

A. The block valve would remain open until manually closed by the operator. Leave the block valve closed, remove power within 1 hour.

B. The block valve would shut automatically when PRZR pressure drops < 2185 psig. Leave the block valve closed, maintain power to the block valve.

C. The block valve would remain open until manually closed by the operator. Leave the block valve closed, maintain power to the block valve.

D✓ The block valve would shut automatically when PRZR pressure drops < 2185 psig. Leave the block valve closed, remove power within 1 hour.

8. 008G2.1.11 001/1/1/PRZR VPR - 1 HR TS/C/A 3.0/MODIFIED/RO/NRC RO/TNT/DSS

K/A

008 Pressurizer Vapor Space Accident (Relief Valve Stuck Open)

G2.1.11 Knowledge of less than one hour technical specification action statements for systems.

K/A MATCH ANALYSIS

Question gives a plausible scenario where a pressurizer pressure channel has resulted in a stuck open PORV. Candidate must have knowedge of block valve response to mitigate the event and 1 hour or less tech spec actions for the block valve.

Block valve would automatically shut when 2 of 3 PRZR pressure channels decreases to less than 2185 psig. Tech specs requires de-energizing the block valve in 1 hour.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. Candidate may think with channel failed high the block valve would have to be manually closed.

B. Incorrect. Plausible. Correct block valve response but block valve has to have power removed within 1 hour by Tech Specs if PORV can't be cycled.

C. Incorrect. Plausible. Candidate may think with channel failed high the block valve would have to be manually closed.

D. Correct. Block valve response and tech spec action both correct for this answer.

REFERENCES

Technical Specifications 3.4.11, Pressurizer Power Operated Relief Valves (PORVs) and the Tech Spec Bases for spec 3.4.11

Vogtle November 2005 SRO Retake Exam question # 11. This question significantly changes the plant conditions to where PORV is not capable of being cycled which changes the Tech Spec action required. Also, system response for block valve versus procedural step performance of an EOP is required.

VEGP learning objectives:

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

a. State the LCO

b. State any one hour or less required actions.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each PORV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable and capable of being manually cycled.	A.1 Close and maintain power to associated block valve.	1 hour
B. One PORV inoperable and not capable of being manually cycled.	B.1 Close associated block valve.	1 hour
	<u>AND</u>	
	B.2 Remove power from associated block valve.	1 hour
	<u>AND</u>	
	B.3 Restore PORV to OPERABLE status.	72 hours

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief pressurizer safety valves and PORVs. The PORVs are safety-related DC solenoid operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the block valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The power supplies to the PORVs, their block valves, and their controls are Class 1E. Two PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

The plant has two PORVs, each having a relief capacity of 210,000 lb/hr at 2385 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure — High reactor trip setpoint up to and including the design step-load decreases with steam dump. In addition, the PORVs minimize challenges to the pressurizer

(continued)

BASES

LCO

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

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

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

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

APPLICABILITY

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

(continued)

BASES

APPLICABILITY (continued)	requirements in MODES 4, 5, and 6 with the reactor vessel head in place.

ACTIONS

A Note has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis).

A.1

PORVs may be inoperable and capable of being manually cycled (e.g., excessive seat leakage, instrumentation problems, or other causes that do not create a possibility for a small break LOCA). In this condition, either the PORVs must be restored or the flow path isolated within 1 hour. The associated block valve is required to be closed, but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. The PORVs may be considered OPERABLE in either the manual or automatic mode. This permits operation of the plant until the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition.

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B.1, B.2, and B.3

If one PORV is inoperable and not capable of being manually cycled, it must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to

(continued)

1. 038EG2.1.11 001

PORV 455A has developed excessive seat leakage and the appropriate Tech Spec actions were taken. The PORV is capable of being manually cycled.

PORV 456A & block valve are in their normal alignments.

Subsequently a SGTR has occurred and 19030-C, Steam Generator Tube Rupture" is in progress. You are at the step to check PORV and Block Valve status

In accordance with Technical Specifications and 19030, which ONE of the following would be CORRECT for the Train A PORV and Block Valve ?

A. The block valve should have been closed within 1 hour with power maintained. WHEN PRZR pressure > 2185 psig, THEN open block valve.

B. The block valve should have been closed within 1 hour with power removed. Restore power to block valve, WHEN PRZR pressure > 2185 psig, do not open block valve.

C.✓ The block valve should have been closed within 1 hour with power maintained. WHEN PRZR pressure > 2185 psig, do not open block valve.

D. The block valve should have been closed within 1 hour with power removed. Restore power to block valve, WHEN PRZR pressure > 2185 psig, THEN open block valve.

Goes with Q#8

K/A

038 Steam Generator Tube Rupture

EG2.1.11 Conduct of Operations

Knowledge of less than one hour technical specification action statements for systems.

K/A MATCH ANALYSIS

Question gives a PORV with excessive seat leakage and requires the candidate to know the correct choice of 1 hour or less Tech Spec actions from memory. A SGTR develops and the candidate must choose the correct action from 19030-C.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Block valve should remain shut to isolate the PORV seat leakage.

B. Incorrect. The block valve should have remained energized per Tech Specs.

C. Correct. Do not open the block valve since Train B PORV & Block still operable.

D. Incorrect. The block valve should have remained energized per Tech Specs.

REFERENCES

19030-C, "Steam Generator Tube Rupture", step 23.

Tech Spec 3.4.11 for PORVs and Block valves.

Tech Spec Bases 3.4.11 for PORVs and Block valves.

VEGP learning objectives:

LO-LP-37311-07, "Using EOP 19030-C as a guide, briefly describe how each step is accomplished".

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

a. State the LCO

b. State any one hour or less required actions"

Goes with Q #8

9. 008K3.01 001/2/1/CCW - LOADS/MEM - 3.4/MODIFIED/RO/NRC RO/TNT/DSS

A loss of ACCW has occurred on Unit 1.

Which **ONE** of the following would **CORRECTLY** describe the loads affected **ONLY** in the containment building ?

A.✓ RCDT Hx, CVCS Excess Letdown Hx, RCP lube oil coolers.

B. RCDT Hx, RCP lube oil coolers, CVCS Normal Letdown Hx.

C. RCDT Hx, CVCS Normal Letdown Hx, CVCS Excess Letdown Hx.

D. CVCS Normal Letdown Hx, CVCS Excess Letdown Hx, RCP lube oil coolers.

9. 008K3.01 001/2/1/CCW - LOADS/MEM - 3.4/MODIFIED/RO/NRC RO/TNT/DSS

K/A

008 Component Cooling Water System (CCWS)

K3.01 Knowledge of the effect that a loss or malfunction of the CCWS will have on the following:

Loads cooled by CCWS.

K/A MATCH ANALYSIS

Question gives a plausible scenario where ACCW flow has been lost, candidate must determine which loads in the containment building have been lost.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. All are ACCW loads in containment.

B. Incorrect. Letdown Hx not in containment.

C. Incorrect. Letdown Hx not in containment.

D. Incorrect. Letdown Hx not in containment.

REFERENCES

Vogtle 2002 NRC RO license examination question # 15, modified to give different ctmt choices resulting from a loss of ACCW to fit KA more closely.

Vogtle Text Chapter # 4 page 8 for ACCW system loads.

VEGP learning objectives:

LO-PP-04101-01 From memory state the following for the ACCW system:

a. Heat loads

1. 008K1.02R 001

Which ONE of the following correctly describes the uses of ACCW in the containment building.

A. RCP motor coolers, Seal Water HX and RCP thermal barriers.

B. RCP motor coolers, RCP thermal barriers and CVCS Letdown Regen HX.

C. RCP motor coolers, RCP thermal barriers and Normal Letdown HX.

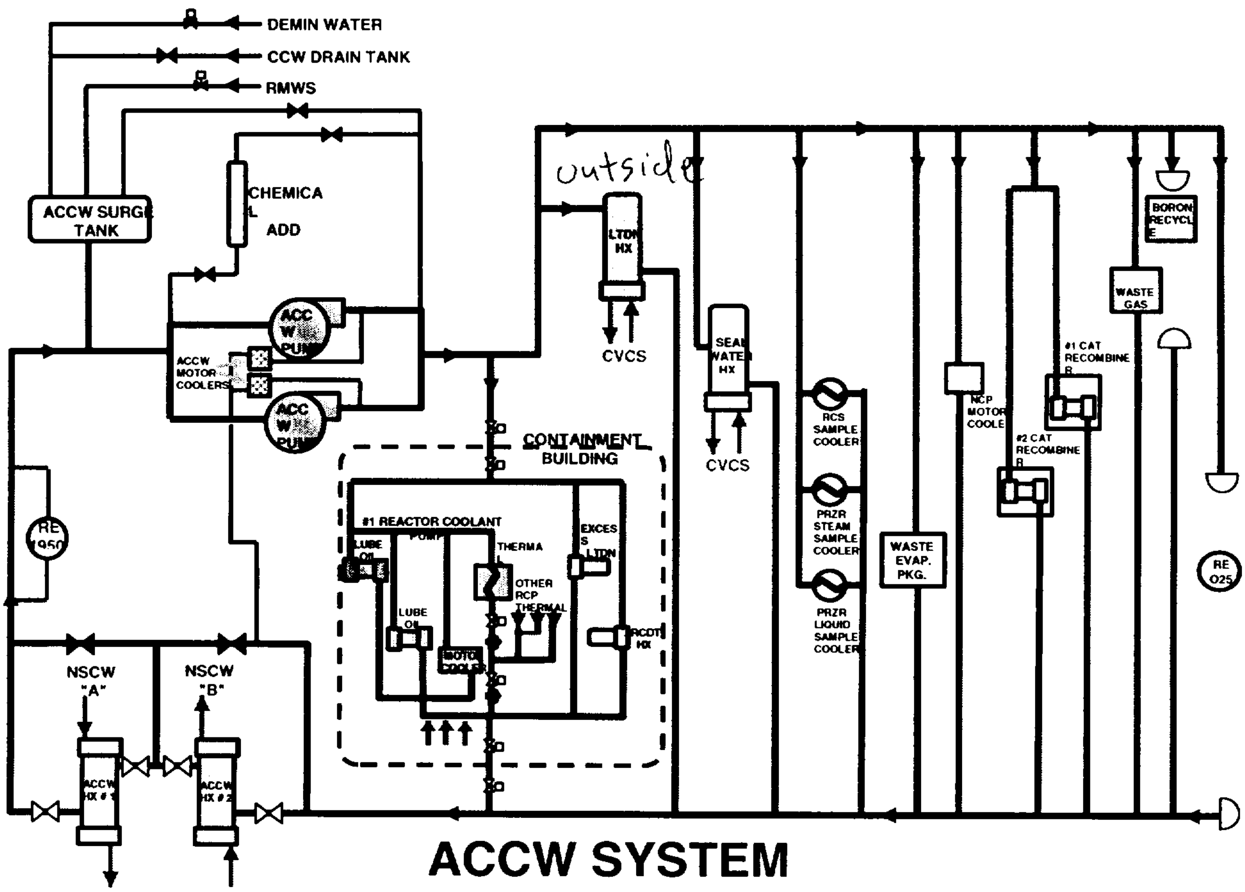
D.✓ RCP motor coolers, RCP thermal barriers and CVCS Excess Letdown HX.

Ref: WB bank, verified VG LP-04101-19-C

Distractor analysis:

a. incorrect - seal wtr hx not in containment
b. incorrect - regen hx not an ACCW load
c. Non-regenative hx not ACCW load
d. correct

Goes with Q# ~~10~~ 9 TNT 3-24-06



ACCW SYSTEM

FIGURE 04-1

The system is chemically treated with Calgon NCS. The nitrites in this chemical scavenge free oxygen while the borates in this chemical maintain a pH range of 8.5-10.0 for corrosion control. Chemicals are added to the system by the use of a chemical addition tank connected to the system by a 1/2 inch line. This line runs from the ACCW pump discharge header, to the tank, and then to the suction header to provide water flow for mixing the chemicals. Access into the chemical tanks is on the top of each tank.

A 3/4 inch recirculation line from the Auxiliary Component Cooling Water pump discharge header to the ACCW surge tank provides flow as desired to maintain the chemical concentration in the surge tank at the same concentration as the rest of the ACCW System (This line is normally isolated).

A one inch line from the ACCW heat exchanger outlet header supplies the system's off-line radiation monitor. The radiation monitor provides indication of a leak from one of the heat loads into the ACCW system. The radiation detector is a skid-mounted unit with a sample pump providing ACCW flow through the unit. The pump returns the sample fluid to the ACCW heat exchanger outlet header. Alarm and indication are provided in the control room to assist the control room operator during abnormal operations. The

10. 009EA2.38 001/1/1/SMALL LOCA - BUBBLE/MEM - 3.9/NEW/RO/NRC RO/TNT/DSS

A small break LOCA is in progress and the crew is implementing 19012-C, "Post LOCA Cooldown and Depressurization".

Which ONE of the following actions could possibly create an upper head void during performance of this procedure ?

A.✓ Depressuring RCS to refill the pressurizer with the RCPs stopped.

B. Sequentially stopping ECCS pumps based on RCS subcooling with RCPs stopped.

C. RCS cooldown at 100 degrees F per hour with the RCPs running.

D. Depressuring RCS to refill the pressurizer with normal sprays with RCPs running.

10. 009EA2.38 001/1/1/SMALL LOCA - BUBBLE/MEM - 3.9/NEW/RO/NRC RO/TNT/DSS

K/A

009 Small Break LOCA

EA2.38 Ability to determine or interpret the following as they apply to a small break LOCA:

Existence of head bubble.

K/A MATCH ANALYSIS

Question asks what indications are available for operators to determine a void in the head exists.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. Procedure notes / cautions warn of this condition.

B. Incorrect. Sequentially reducing ECCS flow should not form void with adequate subcooling.

C. Incorrect. This is normal cooldown rate per the procedure.

D. Incorrect. This is normal method to refill the pressurizer.

REFERENCES

19012-C, "Post LOCA Cooldown and Depressurization"

VEGP learning objectives:

LO-LP-37112-01 Using EOP 19012 as a guide, briefly describe how each step is accomplished.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15. Check if ECCS is in service: ☐ SI pumps - ANY RUNNING -OR- ☐ FLOW THROUGH BIT -OR- ☐ RHR Pumps - ANY RUNNING IN SI MODE	☐ 15. Go to Step 32.
☐ 16. Check Normal PRZR Spray - AVAILABLE	☐ 16. Go to Step 18.

NOTE: The Upper Head region of the vessel may void during RCS depressurization if RCPs are not running. This will result in a rapidly rising PRZR level.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. Depressurize RCS using Normal PRZR Spray to refill PRZR:	
☐ a. Spray PRZR with maximum available spray.	
☐ b. Normal PRZR Spray - EFFECTIVE AT REDUCING RCS PRESSURE	☐ b. Shut Normal PRZR Spray Valves. ☐ Go to Step 18.
☐ c. Go to Step 24.	
☐ 18. Check at least one PRZR PORV - AVAILABLE	☐ 18. Go to Step 22.
☐ 19. Check at least one PRZR PORV Block Valve - AVAILABLE	☐ 19. IF neither PRZR PORV Block Valve available, THEN go to Step 22.

11. 010G2.4.31 001/2/1/PRZR PC - ANNUN/C/A - 3.3/NEW/RO/NRC RO/TNT/DSS

The plant is at 100% power when the following annunciator illuminates.

* ALB11 window D02 for PRZR CONTROL LO PRESS AND HEATERS ON

Which **ONE** of the following would be **CORRECT** regarding the cause of this annunciator and the operators response to the PRZR pressure control system ?

A. Channel 456 failing low in the 455/456 control position could cause this alarm. The backup heaters should energize. Check RCS pressure stable or rising, if not, close sprays, close PORV, operate heaters as necessary.

B. Channel 458 failing low in the 455/458 control position would NOT cause this alarm. Spray valve malfunction for loop # 1 could cause this alarm. If spray won't shut. Trip the reactor, when reactor tripped, stop RCP # 1, go to E-0, Reactor Trip or SI.

C.✓ Channel 456 failing low in the 455/456 control position would NOT cause this alarm. Spray valve malfunction for loop # 1 could cause this alarm. If spray won't shut. Trip the reactor, when reactor tripped, stop RCP # 4, go to E-0, Reactor Trip or SI.

D. Channel 458 failing low in the 455/458 control position could cause this alarm. The backup heaters should energize. Check RCS pressure stable or rising, if not, close sprays, close PORV, operate heaters as necessary.

11. 010G2.4.31 001/2/1/PRZR PC - ANNUN/C/A - 3.3/NEW/RO/NRC RO/TNT/DSS

K/A

010 Pressurizer Pressure Control System (PZR PCS)

G2.4.31 Knowledge of annunciators, alarms, and indications, and use of the response instructions.

K/A MATCH ANALYSIS

Question gives a received annunciator for PRZR CONTRL LO PRESS AND HTRS ON. Candidate must determine correct plant response and actions to take.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Even numbered channels don't even control. System response correct for a controlling channel failure and IOAs correct if 18001 Primary instrument malfunction was entered.

B. Incorrect. Distractor is correct except IOA RNO for 18000 is to trip RCP # 4.

C. Correct. RCP # 4 should be tripped on spray valve failure.

D. Incorrect. Even numbered channels don't even control. System response correct for a controlling channel failure and IOAs correct if 18001 Primary instrument malfunction was entered.

REFERENCES

ARP-17011-1/2, for window D02 PRZR LO PRESS AND HEATERS ON

AOP-18001-C, Primary Instrumentation Malfunction section C for PRZR Pressure instrument failure. Step # 1 IOA and RNO.

AOP-18000-C, PRZR Spray, Safety, or Relief Valve Malfunction Step 1 IOA and RNO.

Vogtle Text Chapter 16 on Pressurizer Pressure Control.

VEGP learning objectives:

LO-PP-16303-01 Describe the response of the pressurizer pressure control system to variations in pressurizer pressure /level.

LO-PP-16303-02 Describe how the response of pressurizer pressure control to the following failures:

a. controlling (primary & secondary) channel fails low

LO-LP-60301-11 State the immediate operator action required for a pressurizer pressure instrument failure. Include RNO and substeps of the immediate action.

LO-LP-60300-01State from memory the immediate actions of procedure 18000-C

LO-LP-60300-02Given control room indications properly diagnose and describe the corrective actions necessary to mitigate a failed open:

a. PRZR Spray

WINDOW D02

ORIGIN	SETPOINT	
1-PIC-0445A output from selected channel: 1-PT-0455 OR 1-PT-0457	2210 psig	PRZR CONTROL LO PRESS AND HEATERS ON

1.0 PROBABLE CAUSE

1. Pressurizer Pressure Control System malfunction.

2. Pressurizer Spray or Relief Valve malfunction.

2.0 AUTOMATIC ACTIONS

Pressurize Backup Heaters will energize.

3.0 INITIAL OPERATOR ACTIONS

Check pressurizer pressure indications:

- ~~and i~~If an instrument failure is indicated, initiate 18001-C, "PRIMARY SYSTEMS INSTRUMENTATION MALFUNCTION".

- If a failed PRZR Spray Valve, Safety Valve or PORV is indicated, initiate 18000-C "PRESSURIZER SPRAY, SAFETY, OR RELIEF VALVE MALFUNCTION".

C. FAILURE OF PRZR PRESSURE INSTRUMENTATION

SYMPTOMS

- PRZR HI PRESS Annunciator.
- PRZR HI PRESS CHANNEL ALERT Annunciator.
- PV-455A (456) OPEN SIGNAL Annunciator.
- PRZR CONTROL LO PRESS AND HEATERS ON Annunciator.
- PRZR LO PRESS ALERT Annunciator.
- PRZR LO PRESS SI ALERT Annunciator.
- PRZR PRESS LO PORV BLOCK Annunciator.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
IMMEDIATE OPERATOR ACTIONS	
C1. Verify RCS pressure - STABLE OR RISING.	C1. Perform the following: • Close spray valves. • Close affected PRZR PORV. • Operate PRZR heaters as necessary. • Go to Step C3.

SUBSEQUENT OPERATOR ACTIONS

CAUTION: Failure of the controlling channel may saturate the Master Pressure Controller and cause inadvertent operation of the spray valves during recovery.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C2. Check channel selected for Pressurizer pressure control - OPERATING PROPERLY.	C2. IF controlling channel has failed, THEN perform the following: a. Place HS-455A in close. b. Place PRZR spray valve controllers in manual.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
IMMEDIATE OPERATOR ACTIONS	
☐1. Verify PRZR Spray Valves - CLOSED	1. IF PRZR pressure continues to lower in an uncontrolled manner, THEN: ☐a. Trip the Reactor. ☐b. WHEN Reactor is verified tripped, THEN stop RCP #4. ☐c. Go to 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.
SUBSEQUENT OPERATOR ACTIONS	
☐2. Operate PRZR Heaters as necessary.	
☐3. Verify PRZR PORVs - CLOSED	3. Perform the following: ☐a. Close affected PRZR PORV Block Valve(s). ☐b. IF PRZR pressure continues to lower, THEN go to 18004-C, REACTOR COOLANT SYSTEM LEAKAGE.
☐4. Check PRZR Safety Valves - CLOSED	☐4. IF PRZR pressure continues to lower, THEN go to 18004-C, REACTOR COOLANT SYSTEM LEAKAGE.
☐5. Check associated instrumentation - OPERATING PROPERLY	☐5. Initiate 18001-C, PRIMARY SYSTEMS INSTRUMENTATION MALFUNCTION.

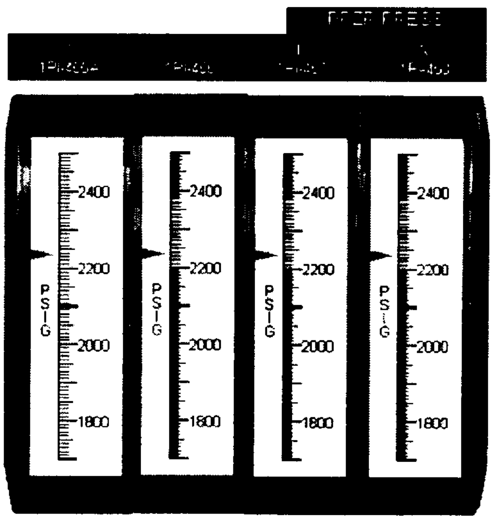
SECTION I

PRESSURIZER PRESSURE CONTROL AND PROTECTION

16-60 Pressurizer Pressure Control System

The purpose of the Pressurizer Pressure Control System is to maintain RCS pressure at 2235 psig for normal power operation. This prevents the reactor coolant from boiling in the RCS and limits the transient fluctuations of pressure so that the pressure does not exceed the design limitations of the system. The pressurizer pressure control system is designed to respond to both under pressure and over pressure conditions that may occur.

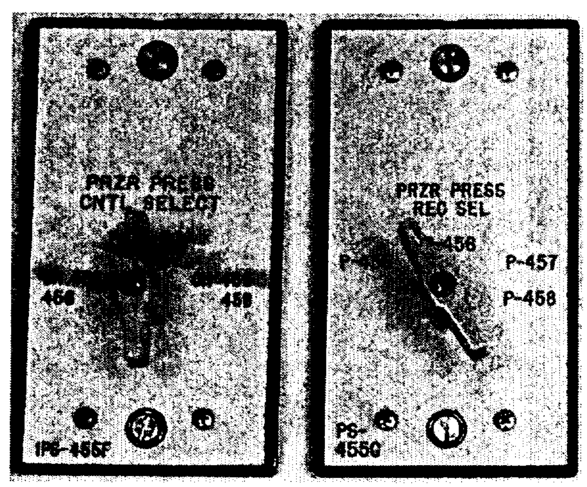
The pressurizer pressure control system uses four narrow-range pressure channels (PT-455, PT-456, PT-457, and PT-458) for control. The transmitters for these pressure channels use the same reference legs as the level channels.



Two channels are selected at any given time using a three-position selector switch (PS-455F). The possible switch selections are as follows:

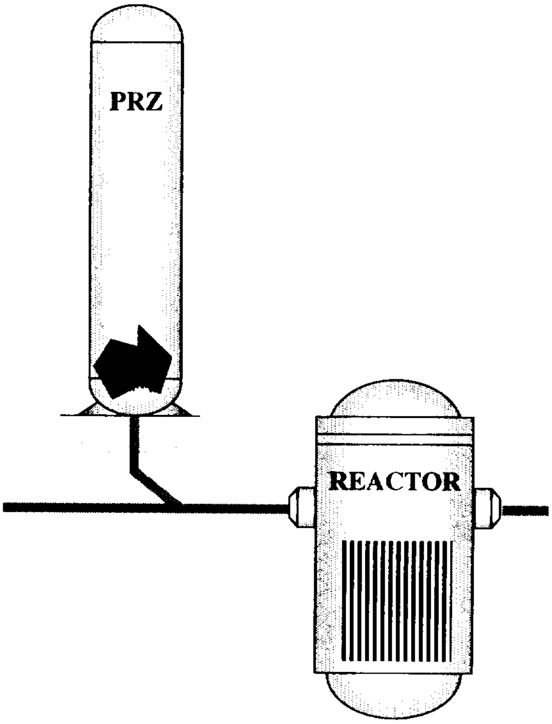
- channels PT-455 and PT-456 or
- channels PT-457 and PT-456 or
- channels PT-455 and PT-458

Top one
Odd # channel
always controls

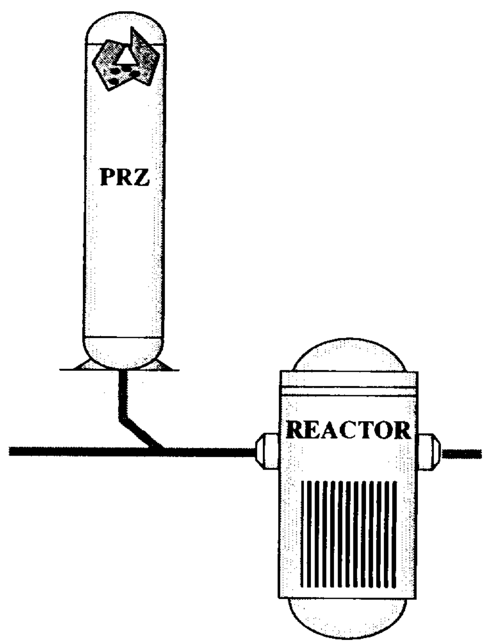


During plant operations, the pressurizer has a mixture of saturated water and steam. Approximately 60% of the total pressurizer volume is water and approximately 40% is steam at 100% reactor power. The pressurizer has heaters with a total heating capacity of 1800 kW. Backup heaters contain 1400 kW of the total capacity while the proportional heaters make up the remaining 400 kW. The backup heaters only have the capability of 0% or 100% output (i.e. on or off). However, the proportional heaters have a variable output range of 0 kW to 400 kW. The heaters are broken up into four groups depending on the power supply for each. They are supplied with 480 volts AC. Groups "A," "B," and "D" are identified as the "backup heaters". Group "C" is the proportional heaters. The backup heaters are used for large pressure changes in the pressurizer.

A decrease in RCS average temperature will result in an out surge from the pressurizer. As the water volume decreases the steam bubble expands to maintain pressure. The control heater current will increase as pressure decreases to add energy to the water and halt the pressure decrease. If pressure continues to decrease, the larger capacity backup heater banks will energize to add additional energy to the pressurizer water. This increase in energy to a saturated water volume will convert some of the water at the water-steam interface to steam. Remember, the specific volume of saturated steam is greater than that of the same temperature water. Since the steam volume expands faster than the water volume decreases, the net effect is to stop the pressure decrease. The heaters will continue to add energy to the water to return pressurizer pressure to normal conditions. As pressure rises, the backup heaters will turn off. As pressure continues to increase, control heater current will decrease to stabilize pressure at 2235 psig.



In surges into the pressurizer are more complex. An increase in RCS average temperature will cause an insurge into the pressurizer. This insurge will increase the pressure by compressing the steam bubble. This compression will cause some of the steam to condense. Since water has smaller specific volume than steam the pressure increase should be arrested. In addition, as pressure rises above 2235 psig, control group heater current decreases to reduce heat input into the pressurizer water. If the pressure increase continues, spray valves will open to spray cold leg water into the steam volume. This water, almost 100°F cooler than the pressurizer steam space temperature, will quench more of the steam bubble causing its volume to decrease rapidly. Since the steam volume decreases faster than the water volume increases from the insurge, pressurizer pressure will decrease. As pressure decreases, the spray valve will close and control group heater current will increase to stabilize pressure at 2235 psig. On large insurges, the relatively cooler water from the hot leg will decrease the pressurizer water temperature. This will tend to cool the water volume causing it to contract retarding the pressure increase. As the sprays respond to the increase in pressure caused by the insurge, the decrease in steam temperature coupled with the decrease in water temperature can result in pressure decreasing faster than the control and backup heaters can respond to arrest the decrease (there is a significant delay in the heat input from the heaters after they energize). To prevent large pressure decreases from an insurge, the backup heaters will energize if pressurizer level rises significantly. This action, in anticipation of the cool down expected from

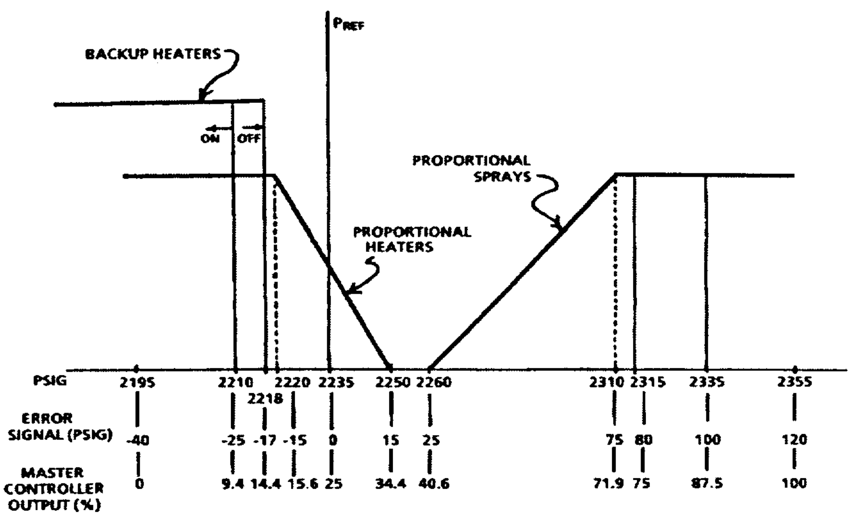


the insurge, reduces the time to feel the impact of the heaters in effect limiting the size of the pressure reduction.

If the pressure transients exceed the capability of the pressurizer sprays, two power-operated relief valve (PORVs) will open. The PORVs are solenoid piloted steam operated valves that relieve to the PRT.

The pressurizer pressure control uses a master controller that compares actual pressurizer pressure from one of the selected pressure channels to the reference pressure of 2235 psig. If there is a difference, the controller will energize the heaters or open the pressurizer spray. The master controller is a proportional plus integral controller. Because of this, its output is dependent on the magnitude of the difference and the integrated time that the difference is present.

Normally, the master controller has an output of approximately 25%, controlling pressure at 2235 psig. At 25% controller output, the proportional heaters are approximately 50% (200 kW) energized. This is necessary to account for the depressurizing effects from pressurizer bypass spray and ambient heat losses. As pressurizer pressure increases to 2250 psig, control heater power gradually decreases turning the proportional heaters off. As the error increases to 2260 psig, a controller output of 40.6%, pressurizer spray valves begin to open. If pressurizer pressure continues to increase, the spray valves will be fully open at 2310 psig with a controller output of 71.9%. Pressurizer pressure can be lowered manually by depressing the up arrow on PIC-455A (Pressurizer Master Pressure Controller) which will increase the controller output. This produces the same responses as the automatic control.

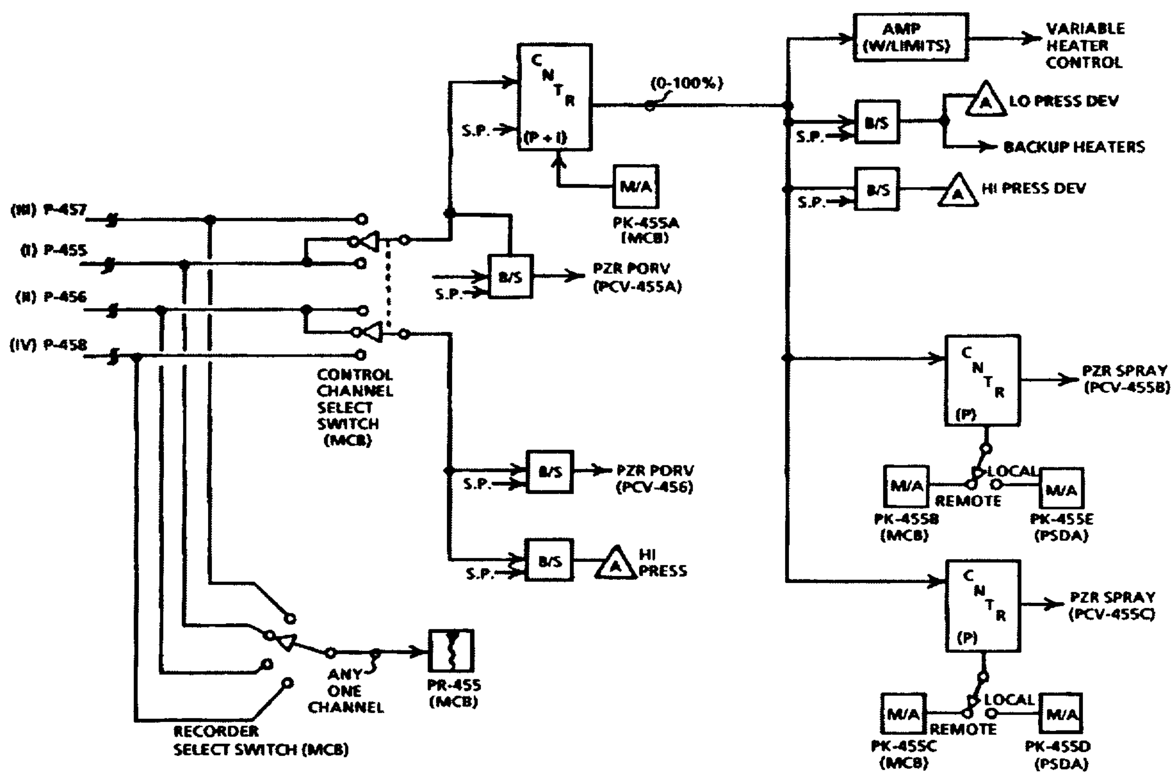


The controller output will decrease as pressurizer pressure decreases. As pressure decreases below 2235 psig, the controller output decreases below 25%. More power is supplied to the proportional heaters until they are fully energized at 2220 psig (controller output at 15.6%). If pressure continues to decrease, the backup heaters will energize at 2210 psig (9.4% output). When pressure returns to 2218 psig (14.4%percent output), the backup heaters de-energize. Pressurizer pressure can be raised manually by depressing the down arrow on 1-PIC-455A (Pressurizer Pressure Controller) to lower the controller output.

It is important to remember that the pressure set points discussed above may not be the exact set points that the respective pressurizer pressure control component will actuate. The integral portion of the master controller will modify the output signal for as the pressure error signal integrates (builds up). As the difference between the set point and actual pressure persists, the output of the controller continues to increase. In anticipation of the possible pressure transient that may occur, the pressurizer pressure controller actuates heaters and sprays before the pressure set points are reached.

Note that PORV PV-455 and PV-456 are not controlled by the master controller. PORV PV-455 responds to the primary pressure channel selected by PS-455F, (either PT-455 or PT-457). PORV PV-455 opens at 2345 psig and closes at 2325 psig. PORV PV-456 uses the secondary channel, (either PT-456 or PT-458) selected by PS-455F. PORV PV-456 opens at 2335 psig and closes at 2315 psig. However, it is important to note that the pressurizer control components will always energize or de-energize at the controller outputs discussed above. The master controller is also selected to control from PT-455 or PT-457 using hand switch PS-455F.

The pressurizer spray valves have separate controllers, one for each valve. These "slave controllers" are a proportional only controller and receive input from the master controller. The MANUAL/AUTO stations are located on the QMCB. Pressurizer Spray Valves can also be controlled from Shutdown Panel "A".



16-61 <u>Pressurizer Pressure Protection System</u>

The Pressurizer Pressure Protection System is designed to protect the pressurizer and RCS from overpressure and under pressure transients that the Pressurizer Pressure Control System is unable to correct.

The Pressurizer Pressure Protection System uses the same pressure channels as the Pressurizer Pressure Control System but is independent of the PS-455F. Pressurizer pressure information is supplied to the Reactor Protection System (RPS). The RPS will take appropriate safeguard actions to protect the plant when conditions warrant it. The RPS will not, however, take safeguard action based on one pressure channel. At least two channels must supply the same information before the RPS will act.

Pressurizer Pressure Protection System provides the following safeguard actions:

1. Pressurizer High Pressure Reactor Trip - 2/4 channels ≥ 2385 psig.

2. Pressurizer Low Pressure Reactor Trip - 2/4 channels ≤ 1960 psig > P-7 (> 10% power)

3. Pressurizer Low Pressure Safety Injection - 2/4 channels ≤ 1870 psig.

The pressurizer low pressure safety injection signal can be blocked (P-11) to allow for cool down and depressurization following a plant shutdown. This requires manual blocking at the P-11 set point of 2000 psig sensed by 2/3 channels. The permissive is automatically unblocked when pressure increases above 2000 psig and also sends a signal to open the SI accumulator outlet valves.

The pressurizer pressure protection system includes an interlock to close the PORVs and the block valves when 2/4 channels indicate a low pressure ≤ 2185 psig. This prevents depressurization from a failed open PORV.

12. 010K2.02 001/2/1/PRZR PC - SPRAY CONT/C/A - 2.5/NEW/RO/NRC RO/TNT/DSS

Which **ONE** of the following would be **CORRECT** regarding source of the power supplies ~~for~~ and effect on the **Controllers** for PRZR spray (PIC-0455B / PIC-0455C) ?

A. Primary power from ND11, 1E 125V DC distribution panel. On a loss of power, controller demand would fail to 100%, sprays would fail open.

B.✓ Primary power from NY4N, 120V AC Essential instrument panel. On a loss of power, controller demand would fail to 0%, sprays could not be opened.

C. Primary power from NY4N, 120V AC Essential instrument panel. on a loss of power, controller demand would fail to 100%, sprays would fail open.

D. Primary power from ND11, 1E 125V DC distribution panel. On a loss of power, controller demand would fail to 0%, sprays could not be opened.

12. 010K2.02 001/2/1/PRZR PC - SPRAY CONT/C/A - 2.5/NEW/RO/NRC RO/TNT/DSS

K/A

010 Pressurizer Pressure Control System (PZR PCS)

K2.02 Knowledge of bus power supplies to the following:

Controller for PZR spray valve.

K/A MATCH ANALYSIS

Question asks for power supplies to both PZR spray valve controllers and effects of a loss of power on spray valves.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Powered from NY4N. Controllers fail shut on loss of power.

B. Correct. Controller fails shut on loss of power.

C. Incorrect. Powered from NY4N. Controllers fail shut on loss of power.

D. Incorrect. Powered from NY4N. Controllers fail shut on loss of power.

REFERENCES

Various elementary drawings that Sim Shop and I & C instructor on security agreements researched to find. NY4N was determined to be main power supply with NYS as a backup. Controllers will fail to the shut position on loss of power.

Vogtle Text Chaper 28 for SSPS page # 7

Drawing 1X6AU01-00369-17, Interconnecting Wiring Diagram Cabinet 05

Drawing 1X6AU01-391-8, Interconnecting Wiring Diagram Cabinet 05

Drawing 1X6AU01-392-9, Interconnecting Wiring Diagram Cabinet 05

VEGP learning objectives:

This is the closest objective for this.

LO-LP-60301-10 Given that the channel selector switch is in the NORMAL position (455/456), describe how and why the plant will respond to the following pressurizer pressure instrument failures. Consider each separately and include effects on the Pressurizer Pressure Control System response, alarms, RPS, and ESF actuations.

b. 455 fails low
c. 456 fails high

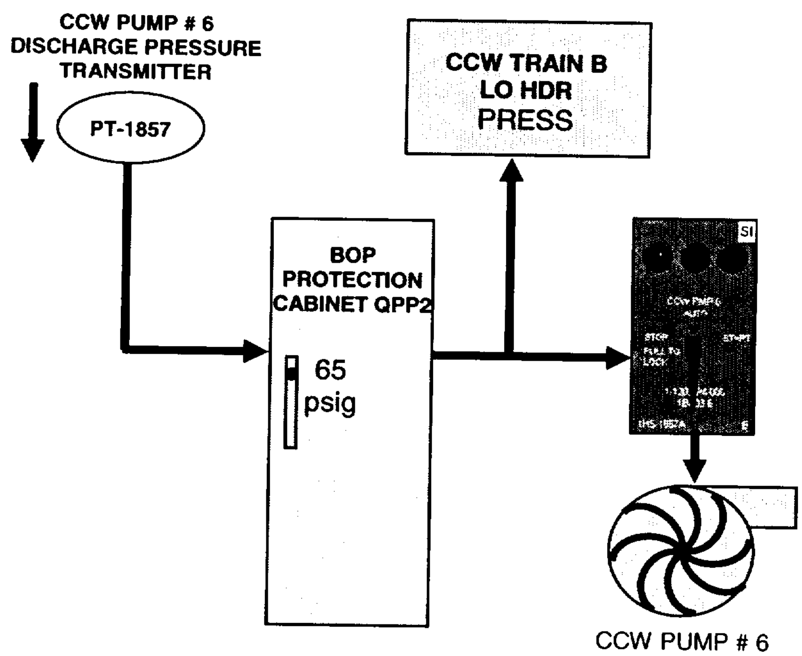
The NSSS control cabinet power supplies, unlike the protection cabinets, are powered by non safety-related essential 120 VAC. Each cabinet is powered by two different power sources; from an inverter (with battery backup) and from a regulated transformer. Listed below are the normal and alternate power supplies which should maintain uninterruptible power to the control cabinets should a loss of either the normal or alternate power source occur.

Cabinet	Normal	Alternate
NSSS Control Cabinet 1	1/2NY4N	1/2NYS
NSSS Control Cabinet 2	1/2NY2N	1/2NYRS
NSSS Control Cabinet 3	1/2NY1N	1/2NYRS
NSSS Control Cabinet 4	1/2NY2N	1/2NYR

Spray Controllers on Q# 12

BOP PROTECTION CABINETS

The BOP protection cabinets are located in the main control room, and are similar to the NSSS protection cabinets in that they are powered by safety-related vital 120 VAC. The difference between the two is how these cabinets are utilized. The BOP Protection cabinets receive field inputs from systems such as NSCW, CCW, ACCW, and SGBD. There is only one interface between the BOP protection cabinets and SSPS. It is the Main Turbine Low ETS pressure input which signals if the Main Turbine is tripped or not. We will review this again later on in the text. The BOP Protection cabinets are mostly used for things such as protective isolation of CVCS and SGBD pipe break protection, RCP thermal barrier return auto isolation, protective interlocks for NSCW, CCW, and ACCW, and (just like the NSSS protective cabinets) sending signals to control board meters, alarms, and the plant computer. Shown below is a drawing that illustrates one of the functions the BOP protection system. Another way that BOP protection differs from NSSS protection is that it performs some control functions. System controls for auxiliary feed water flow, essential chilled water flow, and Steam Generator atmospheric relief valve control are performed through BOP protection cabinets. These functions are safety-related, therefore they must be power from safety related buses. NSSS control cabinets are not powered from safety-related buses. Listed below are the power supplies for each cabinet.



13. 011K3.02 001/2/2/PRZR LC - RCS/C/A-3.5/BANK/RO/NRC RO/TNT/DSS

Unit 1 is operating at 50% power with all control systems in automatic, except Rod Control which is in manual.

Pressurizer level control selector switch in the 459 / 460 position.

Pressurizer level transmitter LT-459 fails as is at the level for 50% power. No operator action is taken for the failed instrument.

Reactor power is then ramped to 75% by the RO using control rods and ~~with the RO controlling rods in manual and the makeup system to control~~ boron concentration to keep Tave on program.

Which **ONE** of the following **CORRECTLY** describes the effect on CVCS charging flow and Pressurizer level when power raised to 75% with no action taken for the failure ?

A.✓ Charging flow increases. Actual pressurizer level will continue to increase from it's initial value until the reactor trips.

B. Charging flow decreases. Actual pressurizer level will continue to decrease from it's initial value until letdown isolates.

C. Charging flow modulates to increase slightly. Actual pressurizer level will be maintained at the programmed pressurizer level for 50%.

D. Charging flow modulates to decrease slightly. Actual pressurizer level will be maintained at the programmed pressurizer level for 50%.

13. 011K3.02 001/2/2/PRZR LC - RCS/C/A-3.5/BANK/RO/NRC RO/TNT/DSS

K/A

011 Pressurizer Level Control System (PZR LCS)

K3.02 Knowledge of the effect that a loss of malfunction of the PZR LCS will have on the following:

RCS

K/A MATCH ANALYSIS

Question gives a plausible scenario where a PZR LT fails as is during a ramp from 50% power to 75% power. As power increases, Tave will increase calling for the program PZR level to rise. Since actual level is failed below this value, charging flow will increase until the Rx. Trip setpoint of 92% on high PZR level is reached.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. Per KA match analysis above.

B. Incorrect. Charging flow will increase and PZR level would rise.

C. Incorrect. Charging flow will increase but not modulate to control at program level.

D. Incorrect. Charging flow increases and will not modulate to control at program level.

REFERENCES

Vogtle 2004 LORQ Biennial Written RO2 examination question # 1.

VEGP learning objectives:

LO-LP-60301-12 Given that the pressurizer level control selector switch is in the NORMAL position (459/460), describe how and why the Pressurizer Level Control System will respond to the following instrument failures. Consider each separately and include effects on pressurizer pressure control, alarms, RPS, and ESF actuations.

a. 459 fails high
b. 459 fails low
c. 460 fails high
d. 460 fails low

1. HL-AA-16000-00 008

Unit 1 is operating at 50 % power with all control systems in automatic, except Rod Control which is in manual.

Pressurizer level control selector switch is in the 461 / 460 postion.

Pressurizer level transmitter LT-461 fails as is at the level for 50% power. No operator action is taken for the failed instrument.

Reactor power is then ramped to 75% with the RO controlling rods in manual and the makeup system to control boron concentration to keep Tave on program.

Which of the following describes the effect on CVCS charging flow and Pressurizer level if reactor power is raised to 75% and no action taken for the instrument failure ?

A. Charging flow modulates to decrease slightly. Actual pressurizer level will be maintained at the programmed Pressurizer level for 50%.

B. Charging flow modulates to increase slightly. Actual pressurizer level will be maintained at the programmed Pressurizer level for 50%.

C. Charging flow decreases. Actual Pressurizer level will continue to decrease from it's initial value until Letdown isolates.

✓D. Charging flow increases. Actual Pressurizer level will continue to increase from it's initial value until the Rx trips.

Goes with Q#13

14. 012K4.07 002/2/1/RPS - 1ST OUT/MEM - 3.0/NEW/RO/NRC RO/TNT/DSS

A reactor protection system trip setpoint was exceeded and an automatic reactor trip has occurred.

first out

The initial trip condition has **NOT CLEARED**. The annunciator will initially _____**a**_____, following ACKNOWLEDGE the annunciator will begin to ____**b**_____.

however

	a	**b**
A.	"*fast* flash",	"*gallop* flash".
B. ✓	"*gallop* flash",	"*slow* flash".
C.	"*fast* flash",	"*gallop* flash".
D.	"*gallop* flash",	"*fast* flash".

14. 012K4.07 002/2/1/RPS - 1ST OUT/MEM - 3.0/NEW/RO/NRC RO/TNT/DSS

K/A

012 Reactor Protection System

K4.07 Knowledge of the RPS design feature(s) and/or interlock(s) which provide for the following:

First-out indication

K/A MATCH ANALYSIS

Question gives a plausible scenario with a reactor trip and the first out is acknowledged by the RO. Several other trip setpoints were exceeded with several trip limits cleared and several still exceeded. Candidate acknowledges first-out without noting. First out indication would clear and with other trip setpoints clear, he cannot determine the first-out, question tests this knowledge.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Initially gallop flashes and goes to slow flash on acknowledge.

B. Correct. Initially gallop flashes and goes to slow flash on acknowledge.

C. Incorrect. Initially gallop flashes and goes to slow flash on acknowledge.

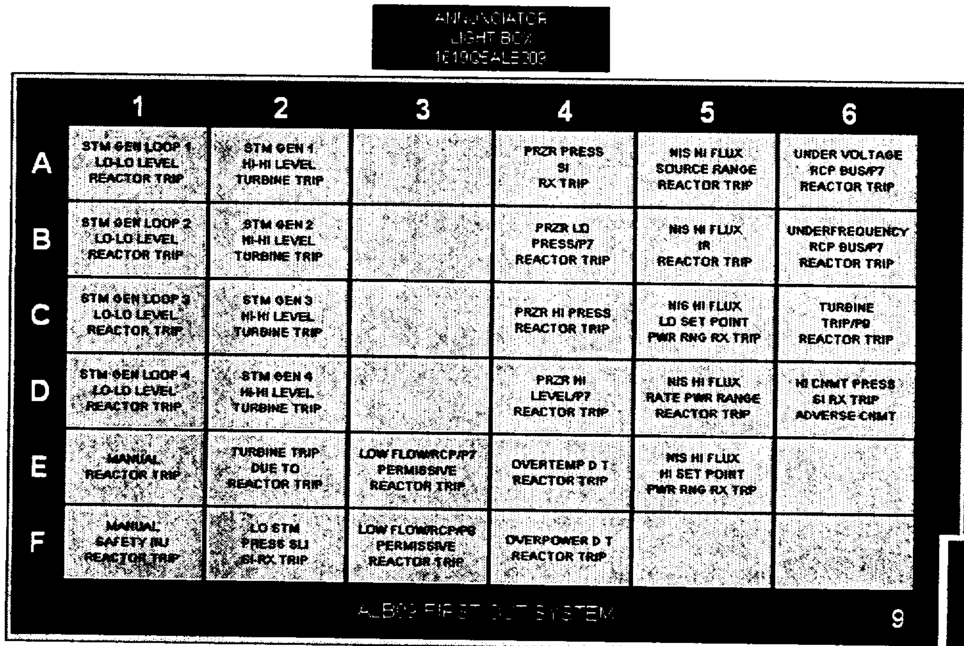
D. Incorrect. Initially gallop flashes and goes to slow flash on acknowledge.

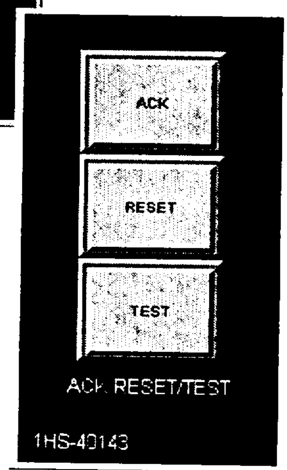
REFERENCES

Vogtle Text Chapter # 28 section 28.22 "First Out Alarm Panel"

VEGP learning objectives:

LO-LP-37011-09 Of multiple reactor trip alarms showing on the annunciator panel, state how the operator would be able to recognize which was the first one to be received.





28.22 FIRST OUT ALARM PANEL

The purpose in the "First Out" Alarm Panel is to aid the operator in diagnosing the cause of a transient. The term "First Out" can be also known as the first offender. The First Out alarm board located on the "C" main control board is electronically different from all other alarm boards. This ensures that the first reactor trip signal received is obvious to the operator.

There are three rates of blinking for the annunciator; Gallop, Fast and Slow rate. The "first out" (first alarm) received will cause its respective light to "gallop" flash. The gallop flash is a burst of fast blinks at a slow rate. The "second", "third", and later trip signals, termed "subsequent alarms", will cause their respective alarm light to fast-flash and continue sounding the horn until acknowledge. If the "first" or "subsequent alarm" returns to NORMAL before acknowledged, there is no change in the alarm window or horn. After the operator depresses the acknowledge button, the horn will silence and the first out annunciator will go to a slow-flash if the trip condition has not cleared. With the trip condition still present the alarm window will still continue to flash even after the reset button is depressed. The subsequent alarms windows remain solid until its condition has cleared. The trip condition has to clear before the operator can reset the associated alarm. Upon the reset of the "first out" alarm, the "next-out" is enabled. This makes it important to notice the first-out alarm prior to acknowledge of any alarms.

Caution: **Alarms on this panel should not be acknowledged until the operator is aware of the first to occur. (Peer check is always helpful)**

15. 013G2.1.32 001/2/1/ESFAS - PRE & LIM/C/A - 3.4/NEW/RO/NRC RO/TNT/DSS

The plant is at 100% power with the following conditions:

* SG # 2 NR level channel failed, the bistables are tripped per Tech Spec action.
* A surveillance is due on a second SG # 2 NR level channel.
* The Unit SS has requested you to bypass the second channel at the BTI panels.

In accordance with 13509-C, "Bypass Test Instrumentation (BTI) Panel Operation, precautions and limitations, going to bypass on the second channel would..........

A. NOT be allowed, a reactor trip would occur during testing.

B. be allowed, notify the Unit SS Tech Spec 3.0.3 entry is required during testing.

C. NOT be allowed, a reactor trip on SG # 2 Lo Lo level would not occur if required.

D.✓ be allowed, up to a specified time period for surveillance testing and maintenance.

Clean RO LO

15. 013G2.1.32 001/2/1/ESFAS - PRE & LIM/C/A - 3.4/NEW/RO/NRC RO/TNT/DSS

K/A

013 Engineered Safety Features Actuation System (ESFAS)

G2.1.32 Ability to explain and apply all system limits and precautions.

K/A MATCH ANALYSIS

Questions gives a plausible scenario where a SG NR level channel has failed and the bistables have been tripped in accordance with Tech Spec actions.Candidate has to determine whether bypassing another channel would be allowed, why or why not.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. With the channel bypassed, coincidence for reactor trip would not be made up. This is what panel is designed for.

B. Incorrect. Bypassing would be allowed by Tech Spec 3.0.3 not entered as long as allowed surveillance time not exceeded. A note at various parts of the procedure warns bypassing more than one channel could cause a 3.0.3 entry. Only one channel is bypassed here.

C. Incorrect. SG Lo Lo level trip is a 2 of 4 coincidence, reactor trip could still occur if required on Lo Lo levels.

D. Correct. Bypassing for surveillance testing or maintenance would be allowed for up to a specified time period.

REFERENCES

Technical Specifications 3.3.2 for ESFAS Instrumentation

SOP-13509-C, Bypass Test Intrumentation (BTI) Panel Operation

VEGP learning objectives:

LO-PP-28102-08 State the purpose of Bypass Test Instrumentation Panels for the Reactor Protection System and how a bi-stable output is bypassed.

LO-PP-28102-09 State the source of power to the BTI panels for a given 7300 Protection Cabinet.

2.1.5 The NIS BTI Annunciators will illuminate only if the Key Lock is in BYPASS ENABLE. NIS BTI switch LEDs will illuminate when the individual BTI switch is in BYPASS.

2.1.6 To provide a reasonable assurance that a channel has been bypassed, the BTI annunciator (on ALB04 or ALB05) and the BTI switch LEDs must be illuminated. If a channel will be tripped or is already tripped, the TSLB should remain extinguished while bypassed.

2.1.7 BTI panel breakers should remain on unless tagged out per 00304-C, "Equipment Clearance and Tagging".

2.2 **LIMITATIONS**

2.2.1 The ~~US~~SS must ensure action statements in Technical Specification LCO 3.3.1, Table 3.3.1-1 and LCO 3.3.2, Table 3.3.2-1 are met prior to performing bypassing manipulations. Bypassing a bistable renders that channel inoperable. A channel may be bypassed for surveillance testing or corrective maintenance (i.e. repair degraded or inoperable channel) for up to 12 hours, as allowed by Technical Specifications. If a problem is discovered during surveillance testing that renders a channel inoperable, the time clock for completing Tech Spec action per Tables 3.3.1-1 and 3.3.2-1 starts at the time the problem is discovered.

2.2.2 In Modes 5 and 6, I&C procedures will be used for controlling BTI Panel manipulations during SSPS response time testing. Additional bypass keys will be issued by the Outage Unit Superintendent ~~Unit~~ if required.

2.2.3 I&C personnel may be used to manipulate BTI switches and independently verify switches during channel restoration.

2.2.4 The BTI Key Lock is designed to prevent key removal while in the BYPASS ENABLE position.

2.2.5 Normally Step 3.2 or 3.3 in I&C ACOTs lists the bistables to be bypassed (i.e., bistables inoperable during the ACOT). It is left to ~~US~~SS discretion as to which bistables will be bypassed during ACOTs.

2.2.6 If a channel is inoperable and tripped per Technical Specification LCO 3.3.1 or LCO 3.3.2 Condition, the additional channel to be surveilled will be bypassed.

4.2.1.13 Notify ~~US~~SS to document the time of channel bypass in the ~~Unit~~ Shift Supervisor Logbook. [ ]

4.2.1.14 When surveillance testing is complete or plant conditions require, restore the BTI NSSS Channel Switch(es) to NORMAL per Section 4.2.4 ~~when surveillance testing is complete or plant conditions require~~. [ ]

4.2.2 **BTI NIS Channel Bypassing**

CAUTIONS

- **Bypassing more than one NIS channel may cause a Technical Specification LCO 3.0.3 entry.**
- Reactor trip and HFASA functions for Source Range NIS (N31 and N32) may be required in Mode 5.

4.2.2.1 Circle bistable(s) to be bypassed, on Checklist 5 (6, 7, 8) for the applicable channel. [ ]

For I&C surveillances, refer to I&C procedure Step 3.2 or 3.3. [ ]

4.2.2.2 For the bistable(s) circled on Checklist 5 (6, 7, 8), obtain the ~~US~~SS authorization to bypass, [ ]

AND

Document on Checklist 5 (6, 7, 8). [ ]

4.2.2.3 Obtain the BTI Enable Key from the ~~US~~SS. [ ]

NOTE

Precautions 2.1.1 and 2.1.2 should be reviewed prior to performing the next step.

4.2.2.4 Except as noted in 2.1.1, ensure no other channel NIS and BOP BTI Panels are in the bypass condition when in Modes 1-4. [ ]

NOTE

If surveillance testing is already in progress on this channel, all switches are not required to be in the NORMAL position.

4.2.2.5 On the BTI Panel on which bypassing is desired, verify all BTI Switches are in the NORMAL position. [ ]

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One train inoperable.	--------------------NOTE-------------------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. --	
	C.1 Restore train to OPERABLE status.	24 hours
	<u>OR</u>	
	C.2.1 Be in MODE 3.	30 hours
	<u>AND</u>	
	C.2.2 Be in MODE 5.	60 hours
D. One channel inoperable.	--------------------NOTE-------------------- A channel may be bypassed for up to 12 hours for surveillance testing. --	
	D.1 Place channel in trip.	72 hours
	<u>OR</u>	
	D.2.1 Be in MODE 3.	78 hours
	<u>AND</u>	
	D.2.2 Be in MODE 4.	84 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. One channel inoperable.	---NOTE--- A channel may be bypassed for up to 12 hours for surveillance testing. ---	
	I.1 Place channel in trip.	72 hours
	OR	
	I.2 Be in MODE 3.	78 hours
J. One Main Feedwater Pumps trip channel inoperable.	J.1 Restore channel to OPERABLE status.	48 hours
	OR	
	J.2 Be in MODE 3.	54 hours
K. One RWST Level - Low Low channel inoperable.	---NOTE--- One additional channel may be bypassed for up to 12 hours for surveillance testing. ---	
	K.1 Place channel in bypass.	72 hours
	OR	
	K.2.1 Be in MODE 3.	78 hours
	AND	
	K.2.2 Be in MODE 5.	108 hours

(continued)

Table 3.3.2-1 (page 5 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$^{(i)}$
5. Turbine Trip and Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1,2$^{(f)}$	2 trains	H	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. Low RCS T_{avg}	1,2$^{(f)}$	4	I	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 561.5 °F	564 °F
Coincident with Reactor Trip, P-4	Refer to Function 8a for all P-4 requirements.					
c. SG Water Level-High High (P-14)	1,2$^{(f)}$	4 per SG	I	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 SR 3.3.2.8	≤ 87.9%	86.0%
d. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. SG Water Level-Low Low	1,2,3	4 per SG	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 SR 3.3.2.8	≥ 35.9%	37.8%

(continued)

(f) Except when one MFIV or MFRV, and its associated bypass valve per feedwater line is closed and deactivated or isolated by a closed manual valve.

(i) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is readjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.

16. 013K5.01 001/2/1/ESFAS - SAFETY DEFIN/MEM - 2.8/NEW/RO/NRC RO/TNT/DSS

The ______a______ is divided into two distinct input, logic, and output bay ___b_____ with 3 or 4 _____c______ of process control equipment used for the signal processing of unit parameters measured by the field instruments.

This system initiates necessary safety systems such as CIA, SLI, SI based on the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

Which **ONE** of the following **CORRECTLY** describes the above mentioned instrumentation ?

A. a- ATWT Mitigation System Actuation Circuit (AMSAC)
b-Trains
c- Channels

B.✓ a- Solid State Protection System (SSPS)
b- Trains
c- Channels

C. a- ATWT Mitigation System Actuation Circuit (AMSAC)
b- Channels
c- Trains

D. a- Solid State Protection System (SSPS)
b- Channels
c- Trains

K/A

013 Engineered Safety Features Actuation System (ESFAS) Instrumentation

K5.01 Knowledge of the operational implications of the following concepts as they apply to the ESFAS.

Definitions of Safety Train and ESF channel.

K/A MATCH ANALYSIS

Quesstion is a fill in the blank question where the candidate has to choose the correct description of ESFAS, trains, and channels. Two logic bays, trains, and 3 or 4 input channels. CIA, SLI, SI given to differentiate between RPS and ESFAS.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. AMSAC is for turbine trip and AFW actuation on loss of FW.

B. Correct. SSPS (ESFAS) actuates CIA, SLI, SI, etc.

C. Incorrect. AMSAC is for turbine trip and AFW actuation on loss of FW. .

D. Incorrect. SSPS (ESFAS) actuates CIA, SLI, SI, etc. but testing circuitry divided into 2 logic trains with 3 or 4 channels of input instrumentation.

REFERENCES

Vogtle Text Chapter 28 for SSPS, RPS, and ESFAS selected material.

VEGP learning objectives:

LO-PP-28103-05 List all ESF actuation signals with applicable set points, coincidences, permissives, blocks, and discuss the systems response to each ESF actuation signal.

LO-PP-28101-08 What is the result of a General Warning condition of both **trains** of SSPS ?

CHAPTER 28

REACTOR PROTECTION SYSTEM

TABLE OF CONTENTS

Rev. No.	Date	Reason for Revision	Author's Initials	Supv's Initials
0.0	10/07/03	Initial Development	TLH	DS
1.0	1/07/04	Revised to incorporate Action Item # 2003202685	TLH	DS
2.0	3/25/04	Corrected minor technical errors from SME review and implementation of DCPs 01-V1N0021 and 01-V2N0022 (PASS Deletions)	GWG	DS
3.0	4/15/04	Instructor and HL-13 Student Feedback	TLH	DS
4.0	9/21/05	Incorporated P-14 setpoint change per DCP 1051602801 and 2051624801 (AI 2005201097)	TLH	DS

28.1 Introduction and Purpose of the Reactor Protection System

This chapter covers some of probably the most important subjects in your student text. The topics that will be discussed are the "meat and potatoes" of your licensed operator training. This is because of the importance the systems provide for protecting the health and safety of the public. There are safe operating limits that are designed to protect the three fission barriers. The reactor protection system ensures that the reactor does not exceed the limits that are set by law. There will be an enormous amount of information to remember coming from this text that will have to stay with you, not only to obtain your operating license, but also to maintain it.

28.2 Overview

The plant is designed to provide defense-in-depth to prevent the release of fission products to the environment. There are four subsystems that are used to ultimately protect the fission product barriers: (1) the **R**eactor **P**rotection **S**ystem (RPS), (2) **S**olid **S**tate **P**rotection **S**ystem ~~(SSPS)~~, (3) **S**afety **F**eatures **S**equencer **S**ystem (SFSS), and (4) **AT**WAT **M**itigation **S**ystem **A**ctuation **C**ircuit (AMSAC). The RPS provides varying levels of protection for these fission product barriers.

A reactor trip is the first level of automatic protection for the fission product barriers. The reactor trip inserts all of the control rods, which stops the fission process. Once the fission process has been stopped, heat generation from fission stops; however, the core continues to generate heat from the decay of fission products.

A safeguards actuation is the next level of protection. Safeguards actuations are provided for events of a serious nature. Upon receiving a safeguards actuation signal, a reactor trip signal is generated, and **E**ngineered **S**afety **F**eature **S**ystem (ESFAS) components are actuated.

The Engineered Safety Feature System is designed to remove decay heat which could cause serious core damage and the subsequent release of fission products. Safeguards actuation protects the clad and the Reactor Coolant System.

The last level of protection is the Containment Spray actuation, which is a subsystem of the Engineered Safety Feature System. The Containment Spray System protects the Containment structure from over pressurization. This over pressurization can be caused by a loss of coolant accident (LOCA) or by a loss-of-secondary coolant accident.

A LOCA is a leak or rupture of any Reactor Coolant System (RCS) piping. A LOCA can be as small as a leak in an instrument sensing line, or as large as the double-ended rupture of the largest RCS pipe.

The loss-of-secondary coolant accident can be caused by a pipe break in the Main Steam System or the Main Feed water System.

Either the LOCA or the loss of secondary coolant can release very large quantities of steam into Containment. The Containment Spray System sprays water into the steam, cooling and condensing the steam in an effort to prevent over pressurizing Containment.

The signal flow path from the analog instrumentation provides input to the RPS for protective features. A typical analog channel consists of a sensor and a protection channel set containing isolated outputs and bi-stable outputs. The isolated outputs are supplied to the Control System while the bi-stables supply both trains of the Protection Systems.

The sensors measure physical parameters such as temperature, pressure, level and flow. The resultant signal from the sensor is converted into a proportional output signal for use by the Control System and channel bi-stables. The bi-stable (signal comparator) compares the incoming signal to a predetermined set point and turns its output off or on if the set point is exceeded. Each bi-stable controls two separate relays, one for each train of protection. Most relays are designed to de-energize to actuate. This means that a loss of power would result in the relay output failing to its safety, or actuated, state. Some relays; however, are energize to actuate. Typically, these are actuations that would result in undesired consequences if they were to inadvertently actuate. For example, the bi-stables for Containment Spray System actuation must energize to actuate Containment Spray. This prevents actuation of the highly corrosive spray system from occurring due to an instrument bus power lost. Protection signals are isolated from control circuits to prevent a fault in a control circuit from affecting the protection circuit.

SECTION A

REACTOR PROTECTION SYSTEM

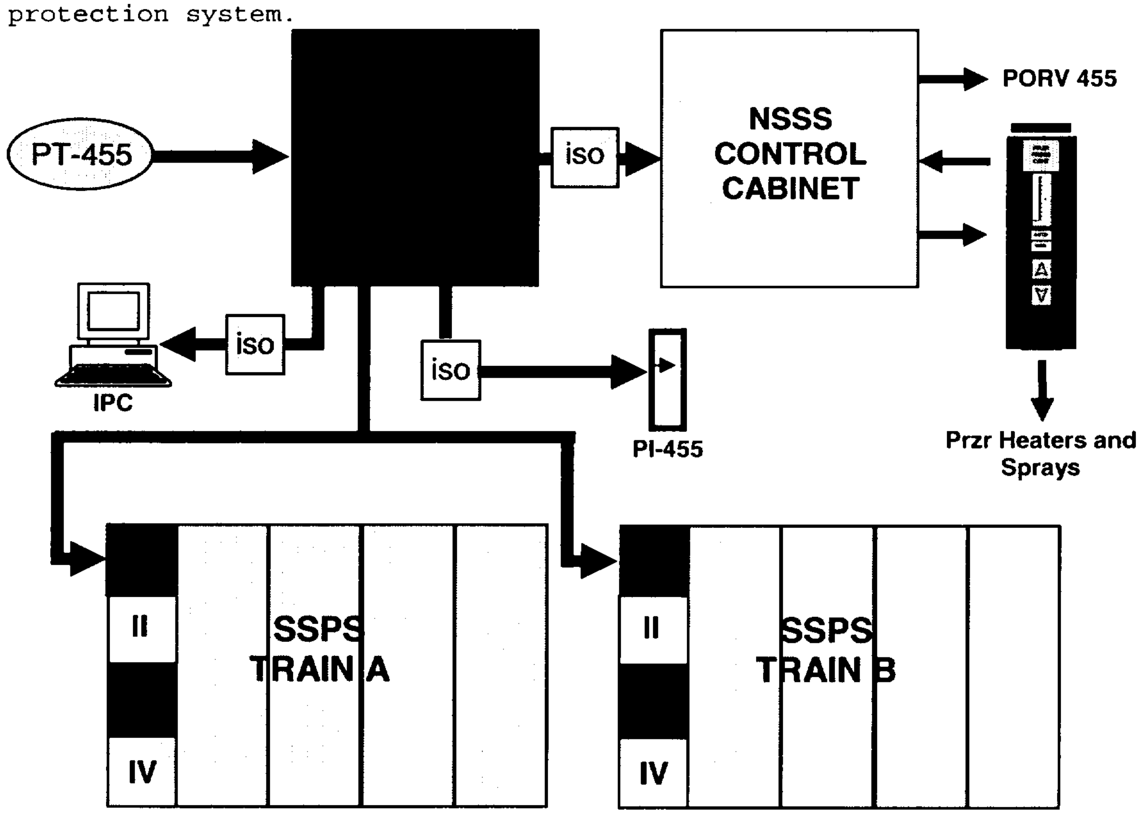
28.3 <u>DESCRIPTION AND LAYOUT</u>

The system for reactor protection is called the 7300 process system. The 7300 process system receives inputs from plant instrumentation such as flow, temperature, and pressure. These inputs are used to provide trip and safeguards actuation signals, automatic controls for certain parameters, MCB meter and alarm indication, and input to the plant computer. The 7300 process system is made up by four different sets of cabinets; (1) NSSS Protection Cabinets, (2) NSSS Control Cabinets, (3) BOP Protection Cabinets, and (4) BOP Control Cabinets.

28.4 <u>NSSS PROTECTION CABINETS</u>

The NSSS Protection Cabinets located in the main control room provide the interface between the safety related instrumentation and SSPS and/or NSSS control cabinets. The signals that are received by the NSSS protection cabinets are sent to; SSPS for Reactor trip and ESFAS actuation functions, to the NSSS control cabinets for parameter control, main control board meters and alarms for indication, and the integrated plant computer for trend and critical status monitoring.

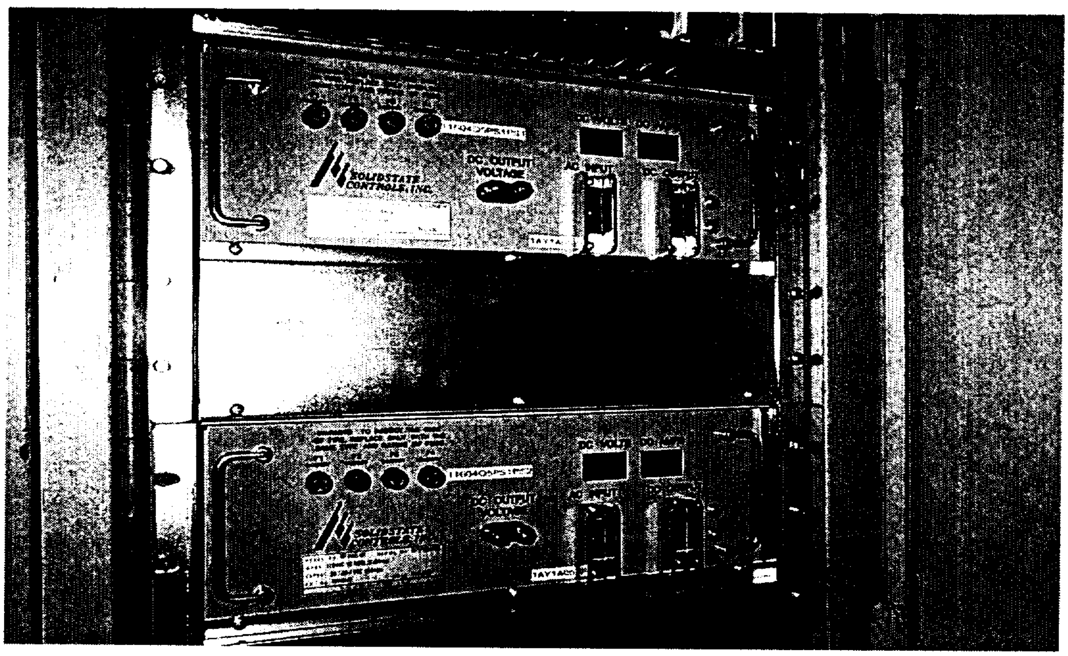
The interface between NSSS control cabinets, control board meters, alarms and plant computers are isolated. This isolation device prevents any electrical faults from interfering with the reactor protection system.



There are a total of 4 NSSS protection cabinets also known as channels. Each cabinet is powered from its associated train 120 volt Vital AC bus.

Cabinet	Power Supply
NSSS Protection Cabinet 1	1/2AY1A
NSSS Protection Cabinet 2	1/2BY1B
NSSS Protection Cabinet 3	1/2CY1A
NSSS Protection Cabinet 4	1/2DY1B

There are two 26 VDC power supplies located in each cabinet, each are capable of supplying adequate power but normally provide load sharing operation for reliability. Both 26 VDC power supplies are powered from its associated 120 VAC power source listed above.



26 VDC Power Supplies

28.5 NSSS CONTROL CABINETS

The NSSS 7300 control cabinets that are located in the main control room receive signals from both non-safety related field inputs and isolated output from NSSS 7300 protection cabinets. These cabinets provide control functions for the primary systems such as Pressurizer level, Pressurizer pressure, and rod control. It also provides control functions for other systems, such as Steam Generator Water Level Control, Steam Dumps, BTRS, and VCT makeup control.

Cabinet	Power Supply
BOP Protection Cabinet 1	1/2AY1A
BOP Protection Cabinet 2	1/2BY1B
BOP Protection Cabinet 3	1/2CY1A
BOP Protection Cabinet 4	1/2DY1B

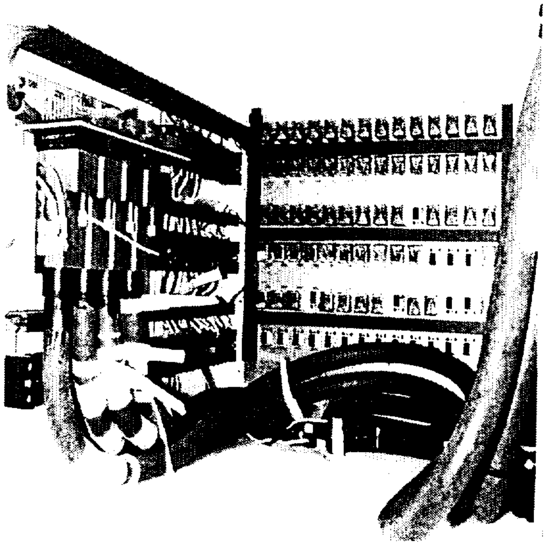
28.7 BOP CONTROL CABINETS

The 2 BOP control cabinets are located in the main control room. The majority of the inputs to these cabinets are from secondary systems like feedwater heaters, MSR's and the main turbine. Like the NSSS control cabinets, these inputs are used for system controls, main control board meter and alarm indications, and plant computer inputs. Listed below are the power supplies for each cabinet.

Cabinet	Normal	Alternate
BOP Control Cabinet 1	1/2NY4N	1/2NYS
BOP Control Cabinet 2	1/2NY1N	1/2NYR

28.8 INSTRUMENT LOOP

An instrument loop consists of a transmitter or sensor (for measuring a parameter), a power supply, a receiver, and connecting cabling. For the sake of discussion, we will use Pressurizer level channel I as an example. Transmitter LT-459 measures Pressurizer level by the use of a D/P type sensor. Its loop power supply comes from NSSS protection cabinet I. A current signal of 4-20 mA is generated based on the level measured in the Pressurizer. A constant voltage is sent to the transmitter via its loop power supply card. The resistance inside the transmitter is based on the level. This resistance changes the current signal from 4-20 mA. A signal converter changes this current signal to a 0-10 VDC signal. This 0-10 VDC signal is sent through isolated and un-isolated circuits. The isolated signal is sent to control board meters and alarms; NSSS control cabinet 2, and the integrated plant computer. One of the un-isolated signals is sent for reactor protection to an electronic on/off switch called a bi-stable. A bi-stable is a circuit that basically turns on or off its output based on a set point. Think of it as an electronic light switch. For instance, the "Pressurizer High Level Trip" set point is at 92%. When the bi-stable's input reaches a voltage level for 92% level (approx. 9.2 VDC), its output automatically turns off. The output of the bi-stable interfaces SSPS by an input relay. The SSPS input relay for level channel 459, which is located in the channel I input bay (shown like the one below), is normally energized when level is below the trip set point. When the trip set point is exceeded, the input relay will lose its input signal form its bi-stable. SSPS will process this signal through its logic bay. If the required coincidence and permissives are met, a reactor trip will occur. Some times it is necessary to remove an instrument channel from service due to maintenance or Tech Spec requirements. To place a channel in the trip condition, all the bi-stables associated with the channel must be tripped. This is performed by placing the associated bi-stables in the test position. The test relay removes the bi-stable's output from the associated SSPS input relay. This action places the channel in the tripped condition. The bi-stable test cards for NSSS protection cabinet I are shown below.



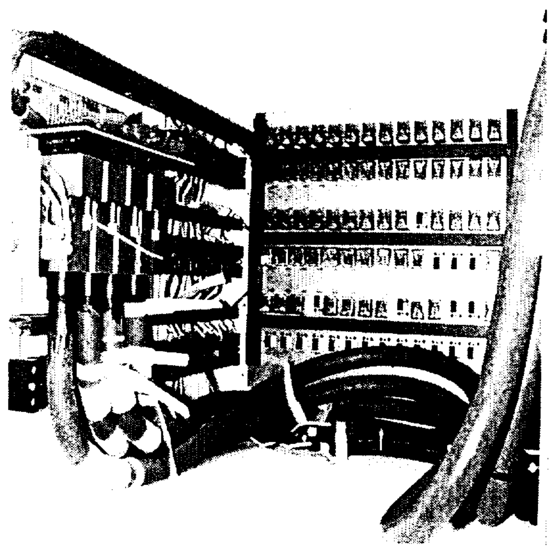
SECTION C

SOLID STATE PROTECTION SYSTEM

28.15 Layout and description of SSPS

Two separate trains of SSPS exist on each unit for reactor protection. SSPS is divided in the following manner: input relay bay, logic bay, output relay bay, and the test panels. This section will discuss the purpose, automatic and testing operation portions of SSPS.

28.16 Input Relay Bays



SSPS input relay bays acts as an isolation device between the various plant inputs and SSPS. It is divided into 4 compartments (one for each protection channel) to provide separation between each input channels. There are three different types of inputs to the input relay bays: (1) NSSS and BOP protection, (2) Nuclear Instrumentation System (NIS), and (3) Field Contacts. The input relays associated with NIS are supplied with 120 VAC from their respective NI channels. The field contacts are instruments that input directly into SSPS, such as Main Turbine Stop Valve position and RCP under frequency relays. The field contacts are powered directly from the input relay bay itself. Each SSPS input relay bay is supplied from its respective channel 120 VAC power source.

Input Relay Bay	Source
Channel I	1AY1A
Channel II	1BY1B
Channel III	1CY1A
Channel IV	1DY1B

Each logic card receives signals from its associated input relays. Most all inputs are de-energized to actuate (Fail Safe). When the input relay de-energizes this causes the input relay contact to close which completes the logic card 15VDC circuit. Some inputs like containment spray actuation and RWST suction auto swap over are energized to actuate to prevent their actuation due to a failed power supply. When its associated relay is energized its input relay contact closes which completes the logic card 15VDC circuit. There are also inputs that directly interface with SSPS bypassing the input bay completely and processing straight into the logic bay. These inputs include P-4 (reactor trip), Containment Ventilation Isolation (CVI), and Main Control Board (MCB) block and reset hand switches

17. 015K2.01 001/2/2/NIS - NIS CHANNELS/MEM - 3.3/BANK/RO/NRC RO/TNT/DSS

Which **ONE** of the following NIS channels power supplies are from 1E 120V AC buses AY1A and CY1A. ?

A.✓ Source Range N31, Intermediate Range N35, Power Ranges N41 and N43

B. Source Range N32, Intermediate Range N36, Power Ranges N42 and N43

C. Source Range N31, Intermediate Range N35, Power Ranges N41 and N44

D. Source Range N32, Intermediate Range N36, Power Ranges N42 and N44

17. 015K2.01 001/2/2/NIS - NIS CHANNELS/MEM - 3.3/BANK/RO/NRC RO/TNT/DSS

K/A

015 Nuclear Instrumentation System

K2.01 Knowledge of bus power supplies to the following.

NIS channels, components, and interconnections.

K/A MATCH ANALYSIS

Question asks which NIS channels are powered from 120V AC buses AY1A and CY1A. These buses supply power to SR N31, IR N35, and PR N41 and N43. The N43 and N44 choices are plausible due to physical location of these buses is inverted in the plant due to original NI cables being pulled too short. Power supplies left on the original designed channels.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. SR N31, IR N35, and PRs N41 and N43 are powered from these buses.

B. Incorrect. SR N31, IR N35, and PRs N41 and N43 powered from these buses.

C. Incorrect. SR N31, IR N35, and PRs N41 and N43 powered from these buses.

D. Incorrect. SR N31, IR N35, and PRs N41 and N43 powered from these buses.

REFERENCES

Prairie Island August 2005 NRC RO examination question # 58.

18032-1/2, "Loss of 120V AC Instrument Power"

VEGP learning objectives:

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

a. 1AY1A
e. CY1A

ATTACHMENT A (Cont'd)

TABLE 3 PANEL 1AY1A LOAD LIST

BRK	LOAD
03	NIS CHANNEL I CONTROL POWER
04	SPARE
05	PROCESS RACK PROTECTION SET I
06	SAFEGUARD TEST CABINET, TRAIN A
07	SOLID STATE PROTECTION SYSTEM OUTPUT CABINET 1, TRAIN A
08	NIS CHANNEL I INSTRUMENT POWER
09	DIESEL GENERATOR 1A WHM PULSE AMPLIFIER
10	SOLID STATE PROTECTION CHANNEL I, TRAIN B
11	BOP PROTECTION PANEL CHANNEL I
12	SPARE
13	HVAC PANEL 1ACQHVC2
14	SPARE
15	SPARE
16	SPARE
17	HVAC INSTRUMENT PANEL 1500-V7-001-CBA
18	SWITCHGEAR 1AD1 TRANSDUCER POWER
19	PLASMA DISPLAY A
20	SPARE
21	SPARE
22	SPARE
23	MISC EQUIPMENT PANEL QPCP XMFR OF ZLB-15
24	DISTRIBUTION PANELS 1AD11, 1AD12 CONTROL POWER

Sheet 4 of 4

ATTACHMENT E (Cont'd)

TABLE 2 PANEL 1CY1A LOAD LIST

BRK	LOAD
03	NIS CHANNEL III CONTROL POWER
04	SPARE
05	PROCESS RACK PROTECTION SET III
06	TERMINATION CABINET 1CCPT19 (lose MLB 13)
07	SOLID STATE PROTECTION SYSTEM CHANNEL III, TRAIN A
08	NIS CHANNEL III INSTRUMENT POWER
09	REMOTE PROCESSING UNIT A2, CHANNEL III
10	SOLID STATE PROTECTION SYSTEM CHANNEL III, TRAIN B
11	MCC 1CD1M SPACE HTR
12	TRAIN C SYSTEM STATUS MONITORING PANEL (MLB 13)
13	BATTERY CHARGER 1CD1CA SPACE HEATER
14	DISTRIBUTION PANEL 1CD11 CONTROL POWER & SPACE HEATER
15	SPARE
16	BOP PROTECTION CHANNEL III
17	SPARE
18	SWITCHGEAR 1CD1 TRANSDUCER POWER
19	BATTERY CHARGER 1CD1CB SPACE HEATER
20	SWGR 1CAC SPACE HEATER

END OF SUB-PROCEDURE TEXT

QUESTION: 057 (1.00)

The RO is increasing power from 20% to 100% with Control Bank D (CBD) rods currently at 100 steps.

The lift coil fuse for CBD rod C-7 blows.

The RO begins to withdraw rods 2 steps at a time towards 218 steps as power is raised. Which alarm will FIRST alert the operator to the malfunction?

a. COMPUTER ALARM ROD DEVIATION/SEQUENCING

b. ROD AT BOTTOM

c. COMPUTER ALARM DELTA I CHECK TYPER

d. NIS POWER RANGE LOWER DETECTOR HI FLUX DEVIATION OR AUTO DEFEAT

QUESTION: 058 (1.00)

Which NIS channels will be available following a loss of Instrument Bus 112?

a. Source Range N31, Intermediate Range N35, Power Ranges N41, N43, and N44.

b. Source Range N32, Intermediate Range N36, Power Ranges N42, N43, and N44.

c. Source Range N31, Intermediate Range N35, Power Ranges N42, N43, and N44.

d. Source Range N32, Intermediate Range N36, Power Ranges N41, N43, and N44.

Goes with Q # ~~16~~ 17 3-24-06

18. 015/017AK2.07 001/1/1/RCP MALF - SEALS/C/A - 2.9/NEW/R0/NRC RO/TNT/DSS

The plant is at 100% power.

The following annunciator has illuminated for RCP # 3 on the QMCB

* ALB08 window C05 for RCP 3 CONTROLLED LKG HI/LO FLOW

Monitoring of RCP # 3 parameters notes the following:

* seal leakoff high range indicates 5.8 gpm

* seal injection flow slightly higher than previously noted

* excess letdown temperature and pressure have risen slightly

Which **ONE** of the following is **CORRECT** regarding the above indications and actions the crew should take ?

A. # 2 seal is in the abnormal operating range for this RCP, continue to monitor RCP operation, repair at the next outage.

B. # 1 seal is in the abnormal operating range for this RCP, notify duty engineering, shutdown the RCP within 8 hours if management so directs.

C. # 2 seal has failed for this RCP, continue to monitor RCP operation, contact duty engineering for further actions.

D.✓ # 1 seal is in the non-operating range for this RCP, immediately trip the reactor, stop the RCP, shut the seal leakoff valve.

K/A mismatch – revise to address K/A – make this "F" LOK

18. 015/017AK2.07 001/1/1/RCP MALF - SEALS/C/A - 2.9/NEW/R0/NRC RO/TNT/DSS

K/A

015/017 Reactor Coolant Pump (RCP) malfunctions

AK2.07 Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following:

RCP seals

K/A MATCH ANALYSIS

Question gives a plausible scenario where a reactor coolant pump trip from power has resulted in a reactor trip. Candidate must identify the proper SG level response on the affected versus the unaffected SG.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. # 2 seal would cause seal leakoff to indicate low, actions to take consistent with actions described in 13003, "RCP Operation" Figure 1 "Seal Abnormality Decision Tree.

B. Incorrect. # 1 seal is outside abnormal range and requires an immediate stop.

C. Incorrect. See description for A above.

D. Correct. RCP # 3 requires has a # 1 failed seal and requires immediate stop.

REFERENCES

ARP ALB08 window C05 for RCP 3 CONTROLLED LEAKAGE HI/LO FLOW

SOP-13003-1/2, Reactor Coolant Pump Operation

Figure 2 of SOP-13003-1/2, NO 1 Seal Normal Operating Range

Figure 1 of SOP-13303-1/2, RCP Seal Abnormalities Decision Tree

SOP-13003-1/2, Step 4.2.1.3 substeps b and e for RCP shutdown

VEGP learning objectives:

LO-PP-16401-03 Describe the control room indications for a failure of a RCP seal.

WINDOW C05

ORIGIN	SETPOINT
1-FT-0159	4.8 gpm
1-FT-0155	0.8 gpm

RCP 3
CONTROLLED LKG
HI/LO FLOW

1.0 PROBABLE CAUSE

1. High Flow:

   a. Flashing in the Seal Leakoff Line due to loss of seal injection flow or high seal injection temperature,
   b. Failure of Number 1 Seal.

2. Low Flow:

   a. Low differential pressure across Number 1 Seal,
   b. High Volume Control Tank (VCT) pressure,
   c. Excess letdown in service,
   d. Failure of Number 2 Seal.

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

NOTE

RCP 3 No. 1 seal water leakoff high range flow may be monitored using computer point F0159.

1. Observe seal injection flow and seal leakoff flow, as well as excess letdown temperature and pressure for indication of an actual seal anomaly.

2. If a seal problem is indicated, go to 13003-1, "Reactor Coolant Pump Operation".

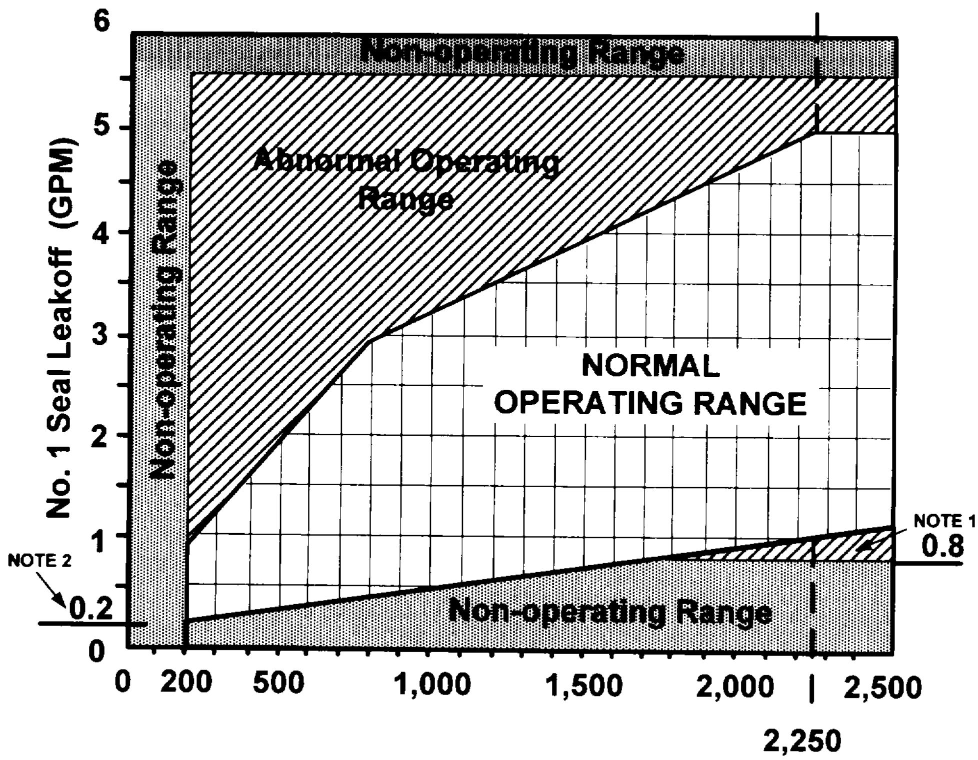
3. If an instrument problem is indicated, initiate maintenance as required.

4.0 SUBSEQUENT OPERATOR ACTIONS

NONE

FIGURE 2

NO. 1 SEAL NORMAL OPERATING RANGE

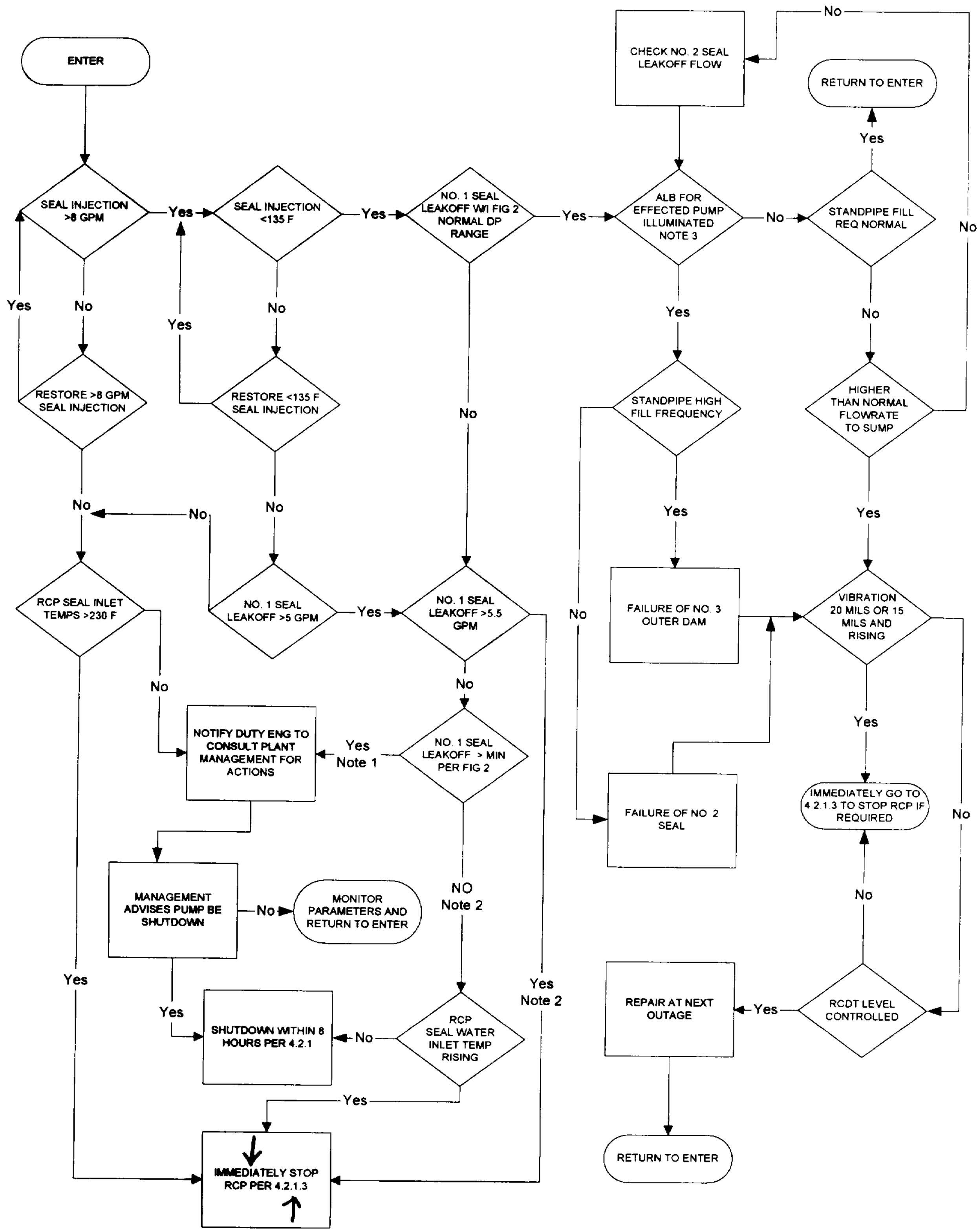


1. If the No. 1 seal leak rates are outside the normal (1.0-5.0 gpm) but within the operating limits ((0.8-5.5 gpm), continue pump operation. ~~ENSURE~~VERIFY that seal injection flow exceeds No. 1 seal leak rate for the affected RCP. Closely monitor pump and seal parameters and contact Engineering for further instructions.

2. Minimum startup requirements are 0.2 gpm at 200 PSID differential across the No. 1 seal. For startups at differential pressures greater than 200 PSID, the minimum No. 1 seal leak rate requirements are defined in the NO. 1 SEAL NORMAL OPERATING RANGE (e.g., at 1000 psi differential pressure, do not start the RCP with less than 0.5 gpm).

3. No.1 Seal Differential Press = RCS WR Press - VCT Press.

FIGURE 1 - RCP SEAL ABNORMALITIES DECISION TREE



Note 1: Abnormal Operating Range of Figure 2
Note 2: Non-operating Range of Figure 2
Note 3: ALB08 A-04, B-04, C-04 or D-04

CAUTION

> If necessary to stop RCP due to total loss of seal cooling, review limitation 2.2.11.

4.2.1.3 If required, perform an RCP shutdown as follows:

a. Start the RCP Oil Lift Pump for affected RCP, if available. [ ]

NOTE

> The affected RCP should be stopped by placing its Non-1E Control Switch in STOP and then placing its 1E Control Switch in STOP.

b. If Reactor Power is greater than 15% Rated Thermal Power:

(1) Trip the Reactor and initiate 19000-C, "E-0 Reactor Trip Or Safety Injection". [ ]

(2) When the immediate operator actions of 19000-C are complete, stop the affected RCP. [ ]

c. If Reactor Power is less than 15% Rated Thermal Power, initiate 18005-C, "Partial Loss Of Flow", and stop the affected RCP. [ ]

CAUTION

> If RCP #1 or #4 is to be stopped, the associated Spray Valve should be placed in manual and closed to prevent spray short cycling.

d. If RCP #1 or #4 is stopped, ~~ensure~~verify its associated spray valve is placed in MANUAL and CLOSED:

RCP 1: 1-PIC-0455C [ ]

RCP 4: 1-PIC-0455B [ ]

e. When the RCP comes to a complete stop (as indicated by reverse flow), close the RCP Seal Leakoff Isolation valve for the affected pump.

RCP 1: 1-HV-8141A [ ]

RCP 2: 1-HV-8141B [ ]

RCP 3: 1-HV-8141C [ ]

RCP 4: 1-HV-8141D [ ]

f. Secure oil lift pump. [ ]

4.3 **SHUTDOWN**

4.3.1 RCP Shutdown

CAUTION

> If RHR is in the Shutdown Cooling Mode, RCS Pressure shall be less than 365 psig prior to stopping a Reactor Coolant Pump (This is to preclude lifting a RHR Suction Relief).

4.3.1.1 Start the RCP Oil Lift Pump for the RCP to be stopped, if available. [ ]

CAUTION

> If RCP #1 or #4 is to be stopped, the associated Spray Valve should be placed in manual and closed to prevent spray short cycling.

4.3.1.2 If RCP #1 or #4 is to be stopped, ~~ensure~~verify its associated spray valve is placed in MANUAL and CLOSED:

RCP 1: 1-PIC-0455C [ ]

RCP 4: 1-PIC-0455B [ ]

4.3.1.3 Stop the RCP by placing its Non-1E Control Switch in STOP and then placing its 1E Control Switch in STOP. [ ]

RCP	Non-1E Control Switch	1E Control Switch
Loop 1	1-HS-0495B	1-HS-0495A
Loop 2	1-HS-0496B	1-HS-0496A
Loop 3	1-HS-0497B	1-HS-0497A
Loop 4	1-HS-0498B	1-HS-0498A

19. 022A3.01 001/2/1/CTMT COOL - ESFAS/C/A - 4.1/MODIFIED/RO/NRC RO/TNT/DSS

All containment coolers are in their normal at power configuration.

* A reactor trip occurs following an LOSP on both 1E busses.
* Both EDGs start, re-energize the busses and run the LOSP sequence.
* 5 minutes later, a Safety Injection occurs

Which **ONE** of the following would be **CORRECT** regarding:

a. - the initial response of the containment fan coolers to the LOSP sequence

b - their response to the Safety Injection sequence 5 minutes later ?

A.✓ a - at 30.5 seconds, 4 fans start in high speed, at 50.5 seconds, 4 fans start in high speed.

b - at 30.5 seconds, all 8 fans shift to slow speed.

B. a - at 30.5 seconds, 4 fans start in low speed, at 50.5 seconds, 4 fans start in low speed.

b - at 30.5 seconds, all 8 fans shift to high speed.

C. a - at 30.5 seconds, all 8 fans start in slow speed.

b - at 30. 5 seconds, 4 fans shift to high speed, at 50.5 seconds, 4 fans shift to high speed.

D. a - at 30.5 seconds, all 8 fans start in high speed.

b - at 30.5 seconds, 4 fans shift to slow speed, at 50.5 seconds, 4 fans shift to slow speed.

19. 022A3.01 001/2/1/CTMT COOL - ESFAS/C/A - 4.1/MODIFIED/RO/NRC RO/TNT/DSS

K/A

022 Containment Cooling System (CCS)

A3.01 Ability to monitor automatic operation of the CCS, including:

Initiation of safeguards mode of operation.

K/A MATCH ANALYSIS

Question gives a plausible scenario where the containment coolers are in the normal configuration (2 coolers in slow speed on each train), an ESFAS actuation signal (LOSP) for the coolers is generated follwed by an SI Actuation signal later. The candidate must pick the proper response of the coolers

ANSWER / DISTRACTOR ANALYSIS

A. Correct. Fans start high speed at 30.5 seconds and 50.5 seconds for LOSP, later shift to slow speed on SI at 30.5 seconds into sequence.

B. Incorrect. See A explanation.

C. Incorrect. See A explanation.

D. Incorrect. See A explanation.

REFERENCES

Vogtle 2002 NRC RO exam question # 21. Modified from this.

VEGP learning objectives:

LO-PP-29101-02 State the ESF function of Containment Coolers and state the backup for this system.

1. Vogtle test question

20. 022AK1.04 001/1/1/RX MU - LOSS MU/C/A - 2.9/NEW/RO/NRC RO/TNT/DSS

A loss of instrument air has occured while in Mode 3 just prior to a plant startup.

* Air pressure has been restored.
* The RO has just ~~placed~~ restored FIC-0121 ~~in~~ to automatic from manual

Which **ONE** of the following would be **CORRECT** regarding the effects of a loss of air on the dilution capability of the Reactor Makeup System and why ~~FIC-0121 was just placed in automatic~~? initially placed in manual

A. No effect on reactor makeup water capability to dilute from QMCB. FIC-0121 was in manual control to raise charging flow.

B.✓ Makeup capability to dilute from QMCB was lost, local manual dilution possible only as a last resort. FIC-0121 was in manual control to lower charging flow.

C. No effect on reactor makeup water capability to dilute from QMCB. FIC-0121 was in manual control to lower charging flow.

D. Makeup capability to dilute from QMCB was lost, local manual dilution is not procedurally addressed. FIC-0121 was in manual control to raise charging flow.

20. 022AK1.04 001/1/1/RX MU - LOSS MU/C/A - 2.9/NEW/RO/NRC RO/TNT/DSS

K/A

022 Loss of Reactor Coolant System Makeup

AK1.04 Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Pump Makeup.

Reason for changing from manual to automatic control of charging flow valve controller.

K/A MATCH ANALYSIS

Questions give a plausible scenario where air is lost causing a loss of Rx. Makeup Water capability to the blender. AOP-18007 directs manual dilution as a last resort. FV-0121 would fail open on loss of air necessitating taking to manual to lower charging.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Reactor water makeup valves to the blender fail closed. FV-0121 would fail open causing charging flow to need to be lowered.

B. Correct. Dilution capablity from QMCB lost but can be done with local valves as a last resort requiring coordination. FV-0121 taken to manual to lower charging flow.

C. Incorrect. Reactor water makeup valves to the blender fail closed. FV-0121 would fail open and this part of response is correct.

D. Incorrect. Dilution capablity from QMCB lost but can be done with local valves as a last resort requiring coordination. AOP-18007 for loss of VCT Makeup addresses this. FV-0121 would fail open and charging flow needs to be lowered.

REFERENCES

18007-C, "Chemical and Volume Control System Malfunction" section C for "Loss of VCT Makeup".

19001-C, "Reactor Trip Response"

18028-C, "Loss of Instrument Air, section B and C for Loss of Air in Mode 3 caution informs FV-0121 will fail open.

VEGP learning objectives:

LO-LP-60321-02 State the fail position of the following valves on loss of instrument air:

o. FV-121 (Charging Flow Control valve)

21. 025G2.1.23 001/1/1/LOSS RHR - PROCEDURE/MEM - 3.9/MODIFIED/RO/NRC RO/TNT/DSS

Unit 1 is in Mode 5, Cold Shutdown preparing to install a Hot Leg nozzle dam.

* RCS drain down is in progress.

* PRZR Cold Cal level indication is offscale low.

* RCS level is 193 ft 4 inches as indicated on the RCS sightglass.

* RCS level on the WR and NR level indicators agree with the sightglass level.

Suddenly, RCS level begins to lower and radiation alarms are received in containment.

* The RO is unable to raise level through the normal charging flow path.

Which **ONE** of the following procedures should the crew implement ?

A. AOP 18004-C, "Reactor Coolant System Leakage, section "A". Actuate SI and go to E-0, "Reactor Trip or Safety Injection" since level cannot be maintained.

B. AOP 18019-C, "Loss of Residual Heat Removal" section "A" for loss of RHR capability with the plant in Mode 5 since level cannot be maintained.

C. AOP 18004-C, "RCS Leakage", section "B" for RCS leakage in Modes 3, 4, or 5 from section "A" IOAs RNO since level cannot be maintained.

D✓ AOP-18019-C, "Loss of Residual Heat Removal", section "B" for RCS leakage with PRZR level below indicating range since level cannot be maintained.

21. 025G2.1.23 001/1/1/LOSS RHR - PROCEDURE/MEM - 3.9/MODIFIED/RO/NRC RO/TNT/DSS

K/A

025 Loss of Residual Heat Removal System (RHRS)

G2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.

K/A MATCH ANALYSIS

Question gives a plausible scenario where an RCS draindown to midloop to install hot leg nozzle dams has been initiated. An RCS leak develops and the candidate must determine whether entry conditions for Loss of RHR or RCS Leak should be entered. Section B of 18019-C for Loss of Residual Heat Removal is the appropriate section.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Entry conditions for 18004 specify 18019 should be entered if an RCS leak develops with PRZR level below indicating range.

B. Incorrect. This section is just for a loss of capability in Mode 5, not a leak.

C. Incorrect. Entry conditions for 18004 specify 18019 should be entered if an RCS leak develops with PRZR level below indicating range.

D. Correct. 18019 section B should be entered if RCS leak develops while PRZR level is not in indicating range.

REFERENCES

AOP-18004-C, "Reactor Coolant System Leakage" in particular symptoms and entry conditions and IOA step 1 and RNO step 1.

AOP-18019-C, "Loss of Residual Heat Removal" symptoms / entry conditions.

UOP-12008-C, page 9 substep b. for RCS WR / NR level indicators and PRZR cold cal.

HL-13 Audit RO/SRO exam question # 38, this question used as base for modification.

VEGP learning objectives:

LO-LP-60314-04 Given conditions and/or indications, determine the required AOP to enter (including subsections, as applicable).

Abnormal Operating Procedures

REACTOR COOLANT SYSTEM LEAKAGE

PURPOSE

PRB REVIEW REQUIRED

Section A is the entry point of this procedure. It specifies the actions to be taken for Reactor Coolant System leakage where the initial leakage rate is within the capabilities of the CVCS and Reactor Makeup System.

Section B of this procedure specifies the actions to be taken for Reactor Coolant System leakage where the leakage rates are GREATER than the capabilities of the CVCS and Reactor Makeup System AND the reactor is in Mode 3 with RCS less than 1000 psig, Mode 4 or 5.

CAUTION:

If RCS leakage is detected while operating with level below PRZR indication range, or with SG nozzle dams installed, go to 18019-C, LOSS OF RESIDUAL HEAT REMOVAL.

SYMPTOMS

- Unexplained change in charging flow.
- A rise in VCT makeup frequency.
- Unexplained lowering of PRZR level or pressure.
- PRT temperature, pressure or level rising.
- CNMT moisture alarm or activity rising.
- CNMT sump level rising.
- CNMT Air Cooler condensate flow rising alarm.
- 18019-C, LOSS OF RESIDUAL HEAT REMOVAL, Step A1.

Abnormal Operating Procedures

LOSS OF RESIDUAL HEAT REMOVAL

PURPOSE

PRB REVIEW REQUIRED

Section A of this procedure specifies actions to be taken for a loss of RHR capability while in:

- Mode 4 or,
- Mode 5 with RCS level within the PRZR indication range and with no SG nozzle dams installed.

Section B of this procedure specifies actions to be taken for a loss of RHR capability or RCS leakage while in:

- Mode 6 prior to Rx cavity flood up,
- Mode 5 with RCS level below the PRZR indication range or,
- With any SG nozzle dams installed.

Section C of this procedure specifies actions to be taken for a loss of RHR capability while the Rx head is removed, Rx cavity flooded, and transfer canal open.

Section D of this procedure specifies actions to be taken for a loss of RHR capability while the RCS is under vacuum pressure conditions (while performing 12009, RCS Vacuum Refill).

SYMPTOMS

- Unexplained change in RHR flow or discharge pressure.
- Detected RHR system or excessive RCS leakage while RHR is in operation.
- Any unexplained rise in RCS temperature while RHR is in operation.
- Any observed loss of RHR system capability while RHR is in operation.
- RHR motor amps fluctuating (Computer Points J9623 or J9624)

- SPDS alarm on:

  - RHR pump current
    (core cooling CSFST - Computer Points UD4623 or UD4624)
  - Both RHR loops not operating
    (core cooling or heat sink CSFST - Computer Point UM5626)
  - RHR trouble
    (core cooling CSFST - Computer Point UD0626)

INITIALS

b. RCS WR level indication, LIS-10405 (L4405), is scaled from 185'-9" to 205'-9". RCS NR level indication, LIS-10403 (L4403), is scaled from 185'-9" to 193'-9". The RCS sightglass, LG-10401, range is from 185'-7" to 207'. Pressurizer Cold Calibrated Level Instrument, LI-462 (L0479), goes off scale low at approximately 201'. ________

c. The following flow rate limitations should be followed during draindown to prevent drawing a partial vacuum in the head due to the limited vent capacity of 1201-U4-086.

(1) less than 100 gpm above 201 feet elevation,

(2) less than 40 gpm below 201 feet elevation

d. Obtain Chemistry concurrence that RCS chemistry is appropriate for draining the RCS, ________

e. Ensure IPC is selected to current mode and trend RHR Pump parameters (J9623 and J9624) for early detection of RHR Pump degradation due to vortexing. ________

37. HL-AW-16101-16 002

Pressurizer pressure is being controlled in AUTO at 2235 psig with the control channel selector switch in the "457-456" position. How will a high failure of PT-456 affect the PRZR power-operated relief valves, and how will the unit respond under these conditions, assuming no operator action over the next 15 minutes?

A. No effect on PZR PORV's with the selector switch in this position.

B. PV-456 would open and depressurize the RCS to the low PZR pressure SI setpoint.

C. PV-456 would open and depressurize the RCS until the PORV block valve closed, and then pressure would return to normal.

D✓ PV-456 would open and depressurize the RCS until its block valve closed, and then pressure would cycle around the block valve set and reset point.

38. HL-AW-60000-00 007

Unit 1 is in Cold Shutdown conditions and the crew has just taken the plant solid per UOP-12006-C. RCS pressure and temperature are 240 psig and 185 F, respectively with charging and letdown flows balanced. The 1B RHR pump is operating, with the 1A RHR pump in standby.

RCS pressure then begins to lower, as radiation alarms are received on 1RE-002 and 1RE-003. Pressurizer level is indicating 90% and is continuing to lower steadily. To properly respond to these changing plant conditions, the crew should implement:

A. AOP 18019-C, Loss of Residual Heat Removal, Section B. Indications of primary leakage are present.

B✓ AOP 18004-C, Reactor Coolant System Leakage, Section B from Section A IOA RNO. Indications of primary system leakage are present.

C. AOP 18019-C, Loss of Residual Heat Removal, Section A. Indications of primary leakage are present with pressurizer level in indicating range.

D. EOP 19000-C, Rx. Trip or Safety Injection. This procedure would be used after IOA RNO for AOP 18004-C is performed and level could not be maintained.

Goes with
Q #21

due to sensor failures

22. 026A4.01 002/2/1/CS - CSS CONTROLS/C/A/4.5/NEW/RO/NRC RO/TNT/DSS

Containment pressure is 32.7 psig following a LOCA, neither train of Containment Spray has automatically actuated. 1AA02 is de-energized. The RO manually actuates 1 of 2 Containment Spray (CS) actuation handswitches.

Which **ONE** of the following **CORRECTLY** describes the CS system response ?

A. CS pump "B" starts, discharge valves open on both trains.

B. CS pump "B" starts, discharge valve on Train B opens.

C. Neither CS pump starts, discharge valve on Train B opens.

D.✓ Neither CS pump starts, discharge valves remain closed on both trains.

22. 026A4.01 002/2/1/CS - CSS CONTROLS/C/A - 4.5/NEW/RO/NRC RO/TNT/DSS

K/A

026 Ability to manually operate and/ or monitor in the control room.

A4.01 CSS controls.

K/A MATCH ANALYSIS

Question gives a plausible scenario where a spray actuation signal is required and did not automatically actuate. The operator operates 1 of 2 C. Spray handswitches. BOTH handswitches are required to actuate spray, therefore the system does not respond.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. This would be true if 2 out of 2 handswitches were used to actuate and lost power only to the pump.

B. Incorrect. This would be true if 2 out of 2 handswitches were used to actuate with only train B power available.

C. Incorrect. The discharge valve on Train B would open on 2 of 2 handswitches used to actuate. However the pump would have started too.

D. Correct. Neither train pump would start or discharge valve would open unless 2 of 2 handswitches used to actuate.

REFERENCES

Vogtle Text Chapter # 15 sections 15.4 and 15.5 for Containment Spray actuation

VEGP learning objectives:

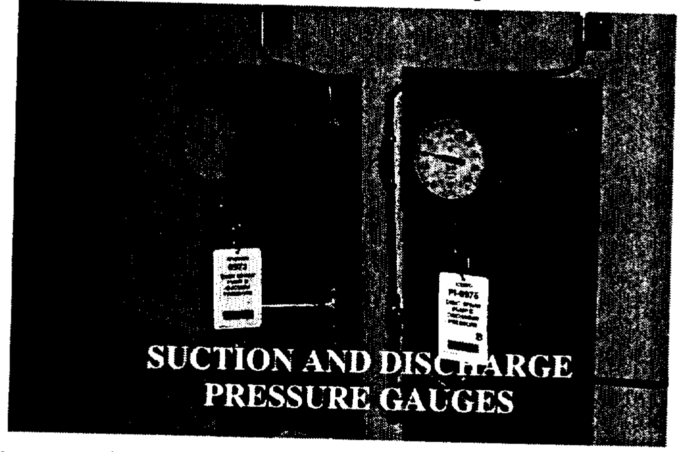
LO-PP-15101-04 List all components that receive a Containment Spray Actuation signal and their change in status.

isolation valve. The encapsulation vessel acts as an extension of the containment wall. The vessel prevents leakage of Containment sump water from Containment in the event of a leak in the sump piping.

[redacted]

15.4 INSTRUMENTION AND CONTROLS

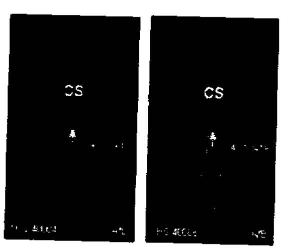
Containment Spray initiates automatically on high containment pressure of ≥ 21.5 psig (CNMT HI-3) sensed by 2-out-of-4 pressure channel coincidence. Instrument used are; PT-934, PT-935, PT-936, and PT-937. **T.S. (4) Containment pressure channels shall be operable in modes 1, 2, 3.** Bi-stables for containment spray actuation are "energize to actuate" to prevent actuation on a loss of instrument power. Containment Spray actuation is not desired unless absolutely needed. Cleanup from a spray actuation would be very costly and equipment damage could be irreparable. Having two separate slave relays for spray actuation gives additional protection from accidental spray initiation. Slave relay K-643 starts the Spray pumps and K-644 opens the discharge valves.


SUCTION AND DISCHARGE PRESSURE GAUGES

The main control room monitoring capability of the Containment Spray system is very limited at best. Containment pressure trend to verify effectiveness is the only parameter the operator can monitor. There are notes in the Emergency Operator Procedure that give direction for local monitoring, if radiation levels in the Aux Building permit.

Note:

Satisfactory CS pump operation is verified by verifying containment pressure is stable or decreasing and by local observation of CS pump suction and discharge pressure gauges, radiation levels permitting.



Radiation levels would be expected be above normal during accident conditions, especially during the recirculation phase.

The manual actuation function allows the operators to manually initiate Containment spray should automatic systems fail to respond. Two switches on the Main Control Board that must be turn simultaneously produces both a Containment Spray and a Containment Ventilation Isolation actuation. Simultaneous operation of the switches has been designed to prevent inadvertent spray actuation. There are two sets of Manual Containment Spray actuation hand switches located on "A" and "C" panels in the Main Control Room.

The pumps will auto start when a C.S. signal is received, unless the Safety Injection sequencer is running. In which case, an auto start will occur on steps 5, 6, 7, or 8 of the SI sequence. The sequencer start intervals for Containment Spray Pumps are 15.5-16.5, 20.5-21.5, 25.5-26.5, and 30.5-31.5 seconds. The pumps will start at anytime a start signal from ESFAS is received 36 seconds after the sequencer has run. The discharge valves open as soon as the Containment Spray signal is generated.

To reset a Containment Spray signal both "A" and "B" train CS reset switches must be placed momentarily in the "reset" position. Containment Spray actuation can be reset anytime regardless of the status containment pressure.

15.5 SYSTEM INTERFACE

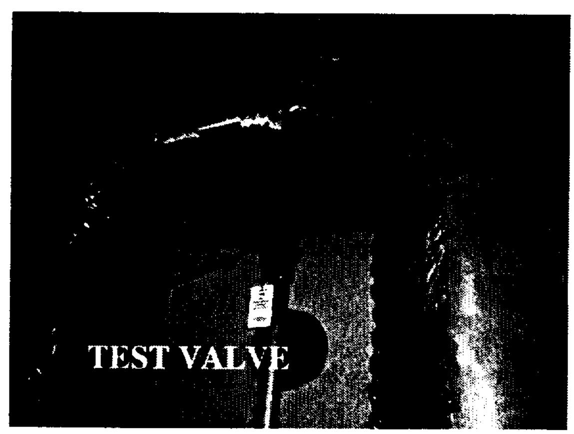
There is very few system interfaces associated with the Containment Spray system. The Containment Spray pump motors are cooled by the Nuclear Service Cooling Water system. NSCW flows through the motor coolers continuously while in operation and in standby alignment. The Containment Spray pump room coolers automatically start on either high room temperature or the start of its associated train CS Pump. The Essential Chilled Water system provides cooling for the CS pump room coolers.

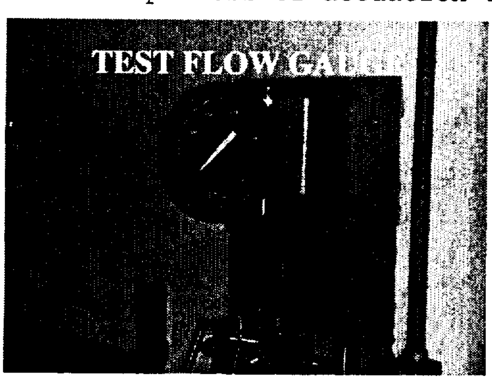
During the ECCS recirculation phase, the RHR system takes suction from the Containment sumps, cools the water in the RHR heat exchanger, and returns to the RCS. This essentially removes the heat from the Containment Spray system by regulating the sump temperature. The RHR system does not interface directly with the Containment Spray system but it some what interacts.

15.6 POWER SUPPLIES

Both Containment Spray Pumps are power from 1E 4160 VAC buses (1/2AA02 and 1/2BA03 respectively). All the Containment Spray motor operated valves are powered by 1E 480 VAC MCC. All the 1E buses are back up by their associated Emergency Diesel Generator.

15.7 TESTING





Just like all other safety related system, the Containment Spray system must be tested to prove reliability. There are two surveillance tests that prove that the Containment Spray system will both actuate and perform when called upon. Having two separate slave relays on each train helps in this process of actuation testing. Remember there are a separate slave relay for the pump and a slave relay for its associated discharge valve. Two

23. 026AA1.07 001/1/1/LOSS CCW - FLOW LOAD/C/A- - 2.9/NEW/RO/NRC RO/TNT/DSS

Which **ONE** of the following would be **CORRECT** regarding a direct automatic response of the ACCW system associated with each annunciator?

A. ALB07, window E03, LTDN HX OUT HI PRESS. Results from high failure of TE-0130 for ACCW cooling to Letdown Hx. causing TV-0130 to go fully shut.

B. ALB04, window B05, ACCW RCP 2 THRM BARRIER HX HI FLOW. Results from high pressure due to a thermal barrier leak causing HV-2041 to go fully shut.

C. ALB07, window B05 for EXCESS LTDN HX HI TEMP. Results from an air line break to Excess Letdown flow control valve causing HV-0123 to fail open.

D✓ ALB04, window B06, ACCW RCP THRM BARRIER HI PRESS. Results from high pressure/or flow due to a thermal barrier leak causing HV-2041 to go fully shut.

23. 026AA1.07 001/1/1/LOSS CCW - FLOW LOAD/C/A- - 2.9/NEW/RO/NRC RO/TNT/DSS

K/A

026 Loss of Component Cooling Water (CCW)

AA1.07 Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water:

Flow rates to the components that are serviced by the CCWS; interactions among the components.

K/A MATCH ANALYSIS

Question gives a plausible scenario with various ACCW Hx alarms. Candidate has to pick the correct auto response / failure that could have resulted in the annunciator.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. High failure of TE-0130 would cause TV-0130 to open for max cooling.

B. Incorrect. High flow due to leak is what shuts valve and HV-2041 does not shut from actions on this annunciator window, HV-19053 would close.

C. Incorrect. HV-0123 is a CVCS valve for Excess Letdown and would fail shut on loss of air. Excess Ltdn Hx cooled by ACCW.

D. Correct. High pressure / flow due to leak thermal barrier leak causes HV-2041 to shut.

REFERENCES

ARP windows ALB04 window B05 and B06. (ACCW RCP 2 THRM BARRIER HX HI FLOW and ACCW RCP THRM BARRIER HI PRESS)

ARP windows ALB07 windows B05 and D03 (EXCESS LTDN HX HI TEMP and LTDN HX OUT HI TEMP).

VEGP learning objectives:

LO-PP-04101-04 From memory describe the expected system response and operator corrective actions for each of the following:

g. Thermal barrier heat exchanger leak
k. TIC-130 failure

ALB-04

	(1)	(2)	(3)	(4)	(5)	(6)
A	ACCW SURGE TK HI/LO LVL	ACCW LO HDR PRESS	ACCW RCP 1 CLR LO FLOW	ACCW RCP 1 CLR OUTLET HI TEMP	ACCW RCP 1 THRM BARRIER HX HI FLOW	
B	BOP PROT GR I BYPASS	ACCW RX COOLANT DRN TK HX LO FLOW	ACCW RCP 2 CLR LO FLOW	ACCW RCP 2 CLR OUTLET HI TEMP	ACCW RCP 2 THRM BARRIER HX HI FLOW	ACCW RCP THRM BARRIER HI PRESS
C	BOP PROT GR II BYPASS	ACCW EXCESS LTDN HX LO FLOW	ACCW RCP 3 CLR LO FLOW	ACCW RCP 3 CLR OUTLET HI TEMP	ACCW RCP 3 THRM BARRIER HX HI FLOW	
D	BOP PROT GR III BYPASS	ACCW RTN HDR FROM RCP LO FLOW	ACCW RCP 4 CLR LO FLOW	ACCW RCP 4 CLR OUTLET HI TEMP	ACCW RCP 4 THRM BARRIER HX HI FLOW	
E	TRAIN A SYS STATUS MON PNL ALERT	TRAIN B SYS STATUS MON PNL ALERT		BOP PROCESS PROT CABINET DOORS OPEN	BOP PROCESS PROT DOOR OPEN >1 CABINET	BOP PCS CABS PWR SUPPLY FAILURE
F	TRAIN C SYS STATUS MON PNL ALERT		TRAIN A SHUTDOWN PNL ON LOCAL CNTL	TRAIN B SHUTDOWN PNL ON LOCAL CNTL	TRAIN C SHUTDOWN PNL ON LOCAL CNTL	"ASIS" TROUBLE

WINDOW B06

ORIGIN	SETPOINT
1-PSH-2041B	155 psig

ACCW RCP
THRM BARRIER
HI PRESS

1.0 **PROBABLE CAUSE**

Leak into Auxiliary Component Cooling Water (ACCW) from a Reactor Coolant Pump (RCP) Thermal Barrier Heat Exchanger (HX).

2.0 **AUTOMATIC ACTIONS**

ACCW RCP Thermal Barrier Outlet Header Isolation Valve 1-HV-2041 closes on high header pressure and/or flow.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

WINDOW B05

ORIGIN	SETPOINT
1-FSH-19054B	65 gpm

ACCW RCP 2
THRM BARRIER
HX HI FLOW

1.0 **PROBABLE CAUSE**

1. Leak into Auxiliary Component Cooling Water (ACCW) from Reactor Coolant Pump (RCP) 2 Thermal Barrier Heat Exchanger (HX).

2. Leak downstream of 1-FE-19054.

2.0 **AUTOMATIC ACTIONS**

RCP 2 Thermal Barrier HX Outlet Isolation Valve 1-HV-19053 closes on high flow of 69 gpm.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. Verify RCP 2 seal injection and return flows are normal.

2. Check Radiation Monitor 1-RE-1950 for signs of radiation leakage into ACCW.

3. Check computer point L2700 for increasing Surge Tank level.

4. If RCP 2 temperatures and seal flows are normal and there is no indication of radiation in the ACCW System, initiate maintenance as required to correct cause of the alarm.

5. If 1-HV-19053 closes, ~~ensure~~ verify the RCP is operated within the limits established in 13003-1, "Reactor Coolant Pump Operation" and ensure actions of Technical Requirement (TR) 13.7.4 are met.

6. If it becomes necessary to stop the pump and:

   a. Reactor power is above 15%:

      (1) Trip the reactor and initiate 19000-C, "E-0 Reactor Trip Or Safety Injection,"
      (2) Stop the pump.

   b. Reactor power is 15% or less, stop the pump and initiate 18005-C, "Partial Loss Of Flow."

	(1)	(2)	(3)	(4)	(5)	(6)
A	BA TANK 1 HI/LO LEVEL	BA TANK 1 LO-LO LEVEL	BA TANK 1 EMPTY	BA TANK 1 HI/LO TEMP	REGEN HX LTDN HI TEMP	NC PUMP LO FLOW
B				VOLUME CONTROL TANK OUTLET TEMP HI	EXCESS LTDN HX HI TEMP	CHARGING LINE HI/LO FLOW
C	BA BATCHING TK HI/LO TEMP	TOTAL MAKEUP FLOW DEVIATION	BTRS RETURN HDR HI TEMP		LP LTDN RELIEF HI TEMP	CHARGING PUMP OVERLOAD TRIP
D	BA BATCHING TK LO LEVEL	CHILLER SURGE TANK HI/LO LVL	LTDN HX OUT HI TEMP	BTRS DEMIN INLET HI TEMP	AUTO MAKE-UP START SIGNAL BLOCKED	TWO CNMT PRESS CHANNELS IN TEST
E			LTDN HX OUT HI PRESS	BTRS LTDN HI TEMP DEMIN DIVERT	VCT HI/LO LEVEL	
F	BA FLOW DEVIATION	CHLR TROUBLE PACKAGE	LTDN HX OUT HI FLOW	LTDN HX HI TEMP DEMIN DIVERT	VCT HI/LO PRESS	

WINDOW D03

ORIGIN	SETPOINT	
1-TE-0130	127.5°F	LTDN HX OUT HI TEMP

1.0 **PROBABLE CAUSE**

1. Low Auxiliary Component Cooling Water (ACCW) flow through the Letdown Heat Exchanger or malfunction of Temperature Control Valve 1-TV-0130.

2. Letdown flow greater than charging flow.

2.0 **AUTOMATIC ACTIONS**

1. 1-TV-0130 opens (ACCW to Letdown Heat Exchanger).

2. Letdown flow will divert to Volume Control Tank (VCT) on high temperature of 132.5°F.

3.0 **INITIAL OPERATOR ACTIONS**

1. Check letdown temperature using 1-TI-0130 on the QMCB.

2. If necessary, initiate 18007-C, "Chemical And Volume Control System Malfunction."

3. ~~Ensure~~ Verify 1-TV-0130 is open and ACCW is available as indicated by 1-PI-1977 on the QMCB.

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. ~~Ensure~~ Verify letdown flow diverts to the VCT using 1-HS-0129 on the QMCB if temperature reaches 132.5°F.

2. Adjust charging or letdown flow as required to reduce the letdown flow temperature.

3. If letdown temperature cannot be reduced, secure letdown flow and go to 18007-C, "Chemical And Volume Control System Malfunctions."

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB115, 1X4DB139, PLS

WINDOW B05

ORIGIN	**SETPOINT**	EXCESS LTDN HX HI TEMP
1-TE-0122	180°F	

1.0 **PROBABLE CAUSE**

1. Low Auxiliary Component Cooling Water (ACCW) flow through the Excess Letdown Heat Exchanger.

2. Large excess letdown flow.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

Check normal operation of ACCW and for ACCW EXCESS LTDN HX LO FLOW (ALB04-C02) and, if necessary, initiate 18022-C, "Loss of Auxiliary Component Cooling Water System."

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. Check Excess Letdown Heat Exchanger outlet temperature using 1-TI-0122 on the QMCB.

2. Reduce Excess Letdown flow by modulating closed 1-HV-0123 using 1-HC-0123 on the QMCB.

3. If Excess Letdown Heat Exchanger outlet temperature does not decrease to normal, remove Excess Letdown from service per 13008-1, "Chemical And Volume Control System Excess Letdown."

4. Return Excess Letdown to service, as necessary, when the cause of the alarm has been determined and corrected.

5. If equipment failure is indicated, initiate maintenance as required.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB114, PLS

24. 026K1.01 001/2/1/CS - ECCS/C/A - 4.2/MODIFIED/RO/NRC RO/TNT/DSS

~~A large break LOCA is in progress, in accordance with 19013-C, "Transfer to Cold Leg Recirculation", the crew has realigned the plant to cold leg recirculation.~~

Which **ONE** of the following is **CORRECT** regarding the connections between the Containment Spray suction sources and the ECCS ?

A. Containment spray and ECCS have separate suction headers from the RWST. Spray header is physically located lower on RWST than ECCS suction header, this is why RWST swapover value for Containment Spray is 10% versus 39% for ECCS.

B.✓ Containment spray and ECCS share a common suction header from the RWST. ECCS swapover must be initiated at 39% to allow adequate time for swapover due to interlocks on ECCS valves. Containment spray does not have these interlocks.

C. Containment spray and ECCS have separate suction headers from the RWST. ECCS and containment spray share the same sump suctions. ECCS suction is swapped at 39% RWST to make more water available for mixing the TSP.

D. Containment spray and ECCS share a common suction header from the RWST. ECCS and containment spray have separate sump suctions. Containment spray is swapped at 10% RWST level to ensure tank bottoms are not drawn into the pumps.

24. 026K1.01 001/2/1/CS - ECCS/C/A - 4.2/MODIFIED/RO/NRC RO/TNT/DSS

K/A

026 Containment Spray System (CSS)

K1.01 Knowledge of the physical connections and/or cause effect relationships between the CSS and the following systems.

ECCS

K/A MATCH ANALYSIS

Question asks about the physical relationships between Containment Spray suction header from RWST and ECCS. Also, Containment sump suctions for spray and ECCS. Both share the same suction header from RWST and ECCS swap initiated to give operators time to align ECCS due to various valve interlocks designed to prevent pumping containment sumps back to RWST. Also, spray and ECCS have different physical sumps in containment.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Have the same suction header which taps off the same place on the RWST and splits on various levels of the auxiliary building to supply each.

B. Correct. Swapover intiated at 39% on ECCS due to various valve interlocks to give time to perform swapover. CSS system does not have any interlocks.

C. Incorrect. Both share the same suction header as stated in A above but have separate sump suctions. ECCS suction is swapped at 39% RWST level. However, all the RWST water ends up in containment via ECCS or CSS irregardless of the level ECCS is swapped to recirculation.

D. Incorrect. Both do share the same suction header from RWST but have separate sump suctions. Spray suction is swapped at a 10% RWST level. However, sludge in the bottom of the RWST is not a concern.

REFERENCES

WOG Background page 14 for 19013-C, "Transfer to Cold Leg Recirculation"

Oconee June 2005 NRC RO examination question # 29

Vogtle Text Cptr 15 for Ctmt Spray pages 6 and 8

V-LO-PP-15101, Containment Spray System

1X4DE322 Containment Location Drawing

VEGP learning objectives:

LO-LP-37113-02 Using EOP 19013 as a guide, briefly describe how each step is accomplished.

LO-LP-37113-03 Describe the basic lineup of the ECCS during cold leg recirculation.

LO-LP-37113-04 State when cold leg recirculation lineup is performed.

LO-PP-15101-04 List all components that receive a Containment Spray Actuation signal and their change in status.

NOTE: Steps 1 through 4 should be performed without delay. FRGs should not be implemented prior to completion of these steps.

PURPOSE: To call attention to the fact that the operator actions to realign the SI System must be done in a rapid manner

BASIS:

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

ACTIONS:

N/A

INSTRUMENTATION:

N/A

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

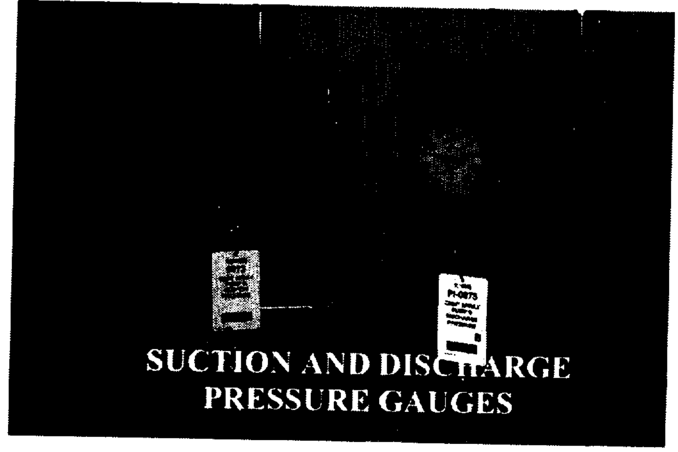
A suction source of water for the SI pumps must be maintained to provide
for core cooling. The actions of these first four steps must be completed even if challenges to a Critical Safety Function occur at this
time, since these steps relate to the maintenance of core cooling.

PLANT-SPECIFIC INFORMATION:

The actions included within this note may be extended to include aligning the containment spray system for recirculation, if necessary to meet plant-specific FSAR containment analysis isolation valve. The encapsulation vessel acts as an extension of the containment wall. The vessel prevents leakage of Containment sump water from Containment in the event of a leak in the sump piping.

15.4 INSTRUMENTION AND CONTROLS

Containment Spray initiates automatically on high containment pressure of ≥ 21.5 psig (CNMT HI-3) sensed by 2-out-of-4 pressure channel coincidence. Instrument used are; PT-934, PT-935, PT-936, and PT-937. **T.S. (4) Containment pressure channels shall be operable in modes 1, 2, 3.** Bi-stables for containment spray actuation are "energize to actuate" to prevent actuation on a loss of instrument power. Containment Spray actuation is not desired unless absolutely needed. Cleanup from a spray actuation would be very costly and equipment damage could be irreparable. Having two separate slave relays for spray actuation gives additional protection from accidental spray initiation. **Slave relay K-643 starts the Spray pumps and K-644 opens the discharge valves.**

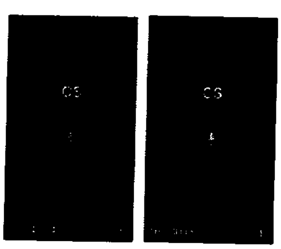


SUCTION AND DISCHARGE PRESSURE GAUGES

The main control room monitoring capability of the Containment Spray system is very limited at best. Containment pressure trend to verify effectiveness is the only parameter the operator can monitor. There are notes in the Emergency Operator Procedure that give direction for local monitoring, if radiation levels in the Aux Building permit.

Note:

Satisfactory CS pump operation is verified by verifying containment pressure is stable or decreasing and by local observation of CS pump suction and discharge pressure gauges, radiation levels permitting.



Radiation levels would be expected be above normal during accident conditions, especially during the recirculation phase.

The manual actuation function allows the operators to manually initiate Containment spray should automatic systems fail to respond. Two switches on the Main Control Board that must be turn simultaneously produces both a Containment Spray and a Containment Ventilation Isolation actuation. Simultaneous operation of the switches has been designed to prevent inadvertent spray actuation. There are two sets of Manual Containment Spray actuation hand switches located on "A" and "C" panels in the Main Control Room.

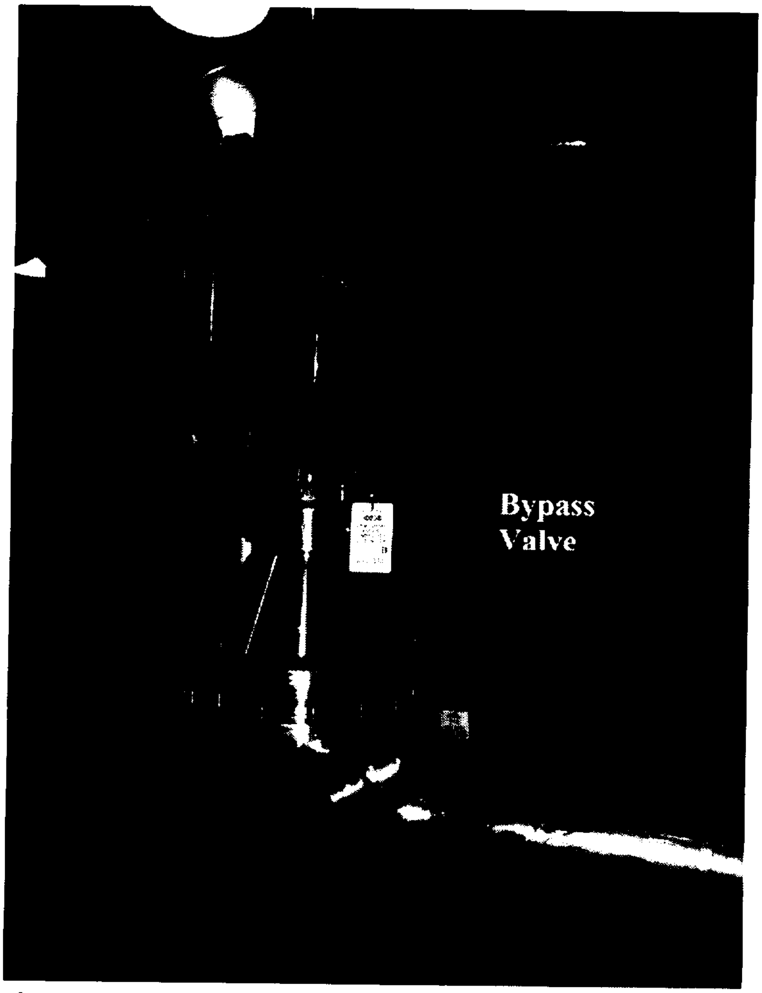
What Components receive a signal when actuated?

- Each pump (Train “A” and Train “B”)

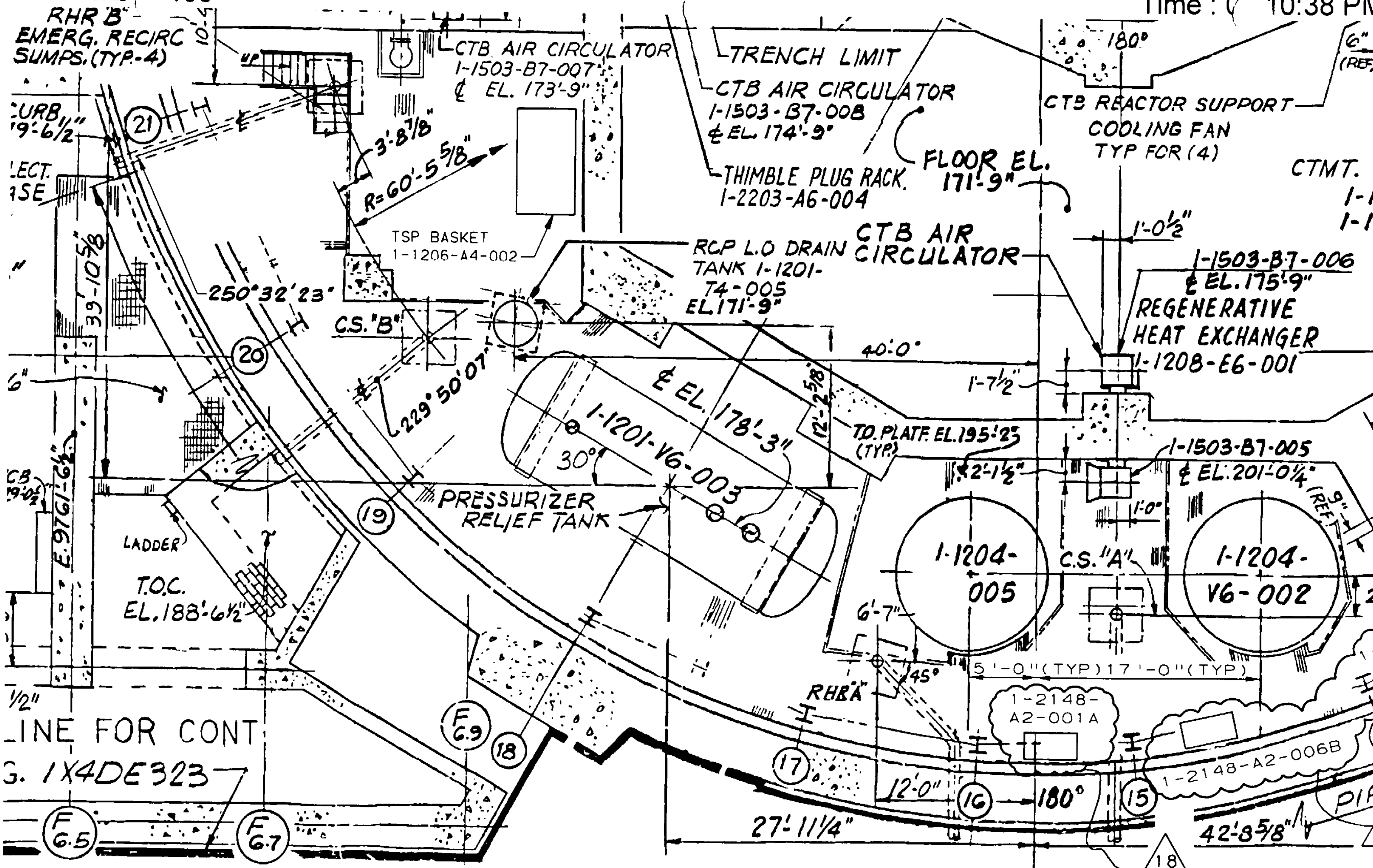
- Each pump discharge Valve (HV-9001A/B) different "Go Test" one of which starts the pump and the other test the discharge valve. The "go test" for the CS pump will not start the pump unless its discharge valve is closed. Both are tested separately to prevent actually flowing water to the spray header. To provide mini flow protection while running the CS pump for testing, a test line is locally aligned. Run time using the test line is limited to 1 ½ hours do to its size. The test line is a common line that can be used by either train that flows back to the RWST suction header. A flow gauge is permanently installed in the test line to allow operators to adjust the test flow to a set reference value. This is done to ensure the test conditions can be compared to a base line value during In Service Testing (IST). Both the IST and the actuation test are normally performed simultaneously, so close coordination is a must.

WARNING:

Containment Spray Pumps do not have mini flow protection in there normal alignment. Therefore, if a Containment Spray Actuation occurs and the pump's discharge valve fails to open, automatically or manually, the pump should be immediately secured by placing its hand switch in the PTL position.



Filling the Containment Spray sump suction is a common operation, coming out of a refueling outage. The key to success in this operation is a thorough pre-job brief and clear and concise communication between the operators in the field and the control room. The control room operator opens the RWST suction valve after ensuring both sump suction valves are closed. This is important because there isn't a check valve in the suction lines. The filling is performed by the coordination of three operators. While an operator is station with a flashlight at the associated containment sump, the control room operator opens the IRC sump suction valve (HV-9002). The operator in the auxiliary building opens HV-9003(ORC sump suction valve) bypass valve until the operator in the Containment verifies the suction line filled. Once filled, the operators will then close all containment sump suction valves and the


RHR "B" EMERG. RECIRC SUMPS. (TYP.-4)
CTB AIR CIRCULATOR 1-1503-B7-007 ℄ EL. 173'-9"
TRENCH LIMIT
CTB AIR CIRCULATOR 1-1503-B7-008 ℄ EL. 174'-9"
THIMBLE PLUG RACK 1-2203-A6-004
FLOOR EL. 171'-9"
CTB REACTOR SUPPORT COOLING FAN TYP FOR (4)
180°
CURB 19'-6½"
R=60'-5 5/8"
3'-8 7/8"
TSP BASKET 1-1206-A4-002
RCP L.O DRAIN TANK 1-1201-T4-005 EL 171'-9"
CTB AIR CIRCULATOR
1'-0½"
1-1503-B7-006 ℄ EL. 175'-9"
REGENERATIVE HEAT EXCHANGER 1-1208-E6-001
250°32'23"
C.S. "B"
40'-0"
1'-7½"
229°50'07"
℄ EL. 178'-3"
1-1201-V6-003
30°
12'-2 5/8"
T.O. PLATF. EL. 195'-2 5/8" (TYP.)
1-1503-B7-005 ℄ EL. 201'-0¼"
2'-1½"
1'-0"
PRESSURIZER RELIEF TANK
LADDER
T.O.C. EL. 188'-6½"
1-1204-005
C.S. "A"
1-1204-V6-002
6'-7"
45°
5'-0"(TYP) 17'-0"(TYP)
RHR "A"
1-2148-A2-001A
1-2148-A2-006B
LINE FOR CONT
G. 1X4DE323
27'-11¼"
12'-0"
180°
42'-8 5/8"
18
F 6.5
F 6.7
F 6.9
15
16
17
18
19
20
21

29. 026K1.01 001

Which ONE of the following describes the interelation of the Reactor Building Spray system and the Low Pressure Injection system?

A. Low pressure injection discharge supplies the Reactor Building Spray pump suction source when aligned to the RBES.

B. Reactor Building Spray pumps take a suction from the Low Pressure Injection suction line when aligned to the RBES.

C. Low Pressure injection coolers are aligned to cool the Reactor Building Spray pumps during extended periods of operation.

D. Reactor Building Spray pumps normally take a suction from the Low Pressure Injection coolers when the BWST is isolated.

New Question developed to match K/A. OP-OC-PNC-BS objective # 5.

A. Incorrect, this is no longer the case according to the lesson plan.

B. Correct, the low pressure injection suction line is connected to the RBES via LP -19 and LP-20.

C. Incorrect, the Low pressure injection coolers are no longer aligned to RBS.

D. Incorrect, the RBS takes a suction from the RBES when the BWST is isolated.

K/A: 026K1.01 Knowledge of the physical connections and or cause effect relationships between containment spray and the following: ECCS.
Utility reviewed no changes.

Goes with Q #24

25. 027AK2.03 005/1/1/PRZR PCS - CONT/POS/C/A 2.6/NEW/RO/NRC RO/TNT/DSS

A **HIGH** failure of the controlling pressurizer pressure channel has occurred. The crew is responding in accordance with AOP-18001-C, section C for "Failure of PRZR Pressure Instrumentation".

Which **ONE** of the following would be **CORRECT** regarding operation of the Pressurizer Master Pressure Controller and the PRZR PRESSURE CONTROL SELECTOR switch ?

A. Select an unaffected channel. Then take manual control of the Master controller and adjust output to 0% demand to prevent inadvertent PORV operation due to a saturated Master Controller.

B.✓ Take manual control of the Master Controller and adjust to output approximately 25% demand to prevent inadvertent spray operation due to a saturated Master Controller. Then select an unaffected channel.

C. Select an unaffected channel. Then take manual control of the Master Controller and adjust to 0% demand to prevent inadvertent spray operation due to a saturated Master Controller.

D. Take manual control of the Master Controller and adjust output to approximately 25% demand to prevent inadvertent PORV operation due to a saturated Master Controller. Then select an unaffected channel.

25. 027AK2.03 005/1/1/PRZR PCS - CONT/POS/C/A 2.6/NEW/RO/NRC RO/TNT/DSS

K/A

027 Pressurizer Pressure Control System (PZR PCS) Malfunction

AK2.03 Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following:

Controllers and positioners

K/A MATCH ANALYSIS

Question gives a plausible scenario where a pressurizer pressure channel has failed high. The candidate must know the failure would cause Master Controller demand to go high (100%) and the sequence of steps to swap to another controlling channel and reason for sequence of steps.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Steps out of sequence for channel swap. PORVs no longer have integrator function at Vogtle, have hard setpoints. Master controller output is set to 25%.

B. Correct. CAUTION in AOP warns of the possibility of inadvertent spray actuation and is why controller taken to 25% demand

C. Incorrect. Reason for spray inadvertent actuation is correct. Steps out of sequence. Master controller output is adjusted to 25%.

D. Incorrect. PORVs no longer have integrat or function at Vogtle, they have hard setpoints.

REFERENCES

18001-C, "Failure of Primary Instrumentation" section C for PRZR pressure intrument.

VEGP learning objectives:

LO-LP-60301-10 Given that the channel selector switch is in the NORMAL position (455/456), describe how and why the plant will respond to the following pressurizer pressure instrument failures. Consider each separately and include effects on the Pressurizer Pressure Control System response, alarms, RPS, and ESF actuations.

a. 455 fails high
b. 455 fails low
c. 456 fails high

C. FAILURE OF PRZR PRESSURE INSTRUMENTATION

SYMPTOMS

- PRZR HI PRESS Annunciator.
- PRZR HI PRESS CHANNEL ALERT Annunciator.
- PV-455A (456) OPEN SIGNAL Annunciator.
- PRZR CONTROL LO PRESS AND HEATERS ON Annunciator.
- PRZR LO PRESS ALERT Annunciator.
- PRZR LO PRESS SI ALERT Annunciator.
- PRZR PRESS LO PORV BLOCK Annunciator.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
IMMEDIATE OPERATOR ACTIONS	
C1. Verify RCS pressure - STABLE OR RISING.	C1. Perform the following: • Close spray valves. • Close affected PRZR PORV. • Operate PRZR heaters as necessary. • Go to Step C3.

SUBSEQUENT OPERATOR ACTIONS

CAUTION: Failure of the controlling channel may saturate the Master Pressure Controller and cause inadvertent operation of the spray valves during recovery.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C2. Check channel selected for Pressurizer pressure control - OPERATING PROPERLY.	C2. <u>IF</u> controlling channel has failed, <u>THEN</u> perform the following: a. Place HS-455A in close. b. Place PRZR spray valve controllers in manual.

C. FAILURE OF PRZR PRESSURE INSTRUMENTATION

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C3. Check pressurizer pressure between 2220 and 2250 psig.	C3. Operate heaters and sprays as necessary to control pressurizer pressure.
C4. Check Pressurizer Master Pressure Controller PIC-455A in AUTO with output signal approximately 25%.	C4. Take manual control of Pressurizer Master Pressure Controller PIC-455A and adjust controller output to approximately 25%.
C5. Check unaffected channels selected on PRZR PRESSURE CONTROL SELECTOR switch PS-455F. Go to Step C10.	C5. Select unaffected channels on PRZR PRESSURE CONTROL SELECTOR switch PS-455F: Failed channel — Select P455 — P457/456 P456 — P455/458 P457 — P455/456 P458 — P455/456
C6. Perform the following:	
a. Place PRZR heaters in AUTO.	
b. Place PRZR spray valve controllers in AUTO.	
c. Verify RCS pressure - STABLE <u>OR</u> RISING.	c. Return PRZR heaters and sprays to manual and operate as necessary to maintain PRZR pressure at 2235 psig. Go to Step C3.
C7. Place PORVs in AUTO and verify proper operation.	C7. Close any open PORV. Return to Step C3.
C8. Return PRZR pressure Master Controller to AUTO.	

26. 028AK1.01 001/1/2/PLC MALF - REF LEG/C/A - 2.8/MODIFIED/RO/NRC RO/TNT/DSS

The unit is operating at 100% with all control systems in automatic, except Rod Control which is in manual.

Pressurizer level control is selected to the 461/460 position when the following occurs:

* A small Pressurizer level reference leg leak occurs on the controlling channel.

Which **ONE** of the following would be **CORRECT** regarding the response of the Pressurizer level control system ?

A. LT-460 level would read lower than actual, charging flow would increase.

B. LT-461 level would read lower than actual, charging flow would increase.

C. LT-460 level would read higher than actual, charging flow would decrease.

D✓ LT-461 level would read higher than actual, charging flow would decrease.

26. 028AK1.01 001/1/2/PLC MALF - REF LEG/C/A - 2.8/MODIFIED/RO/NRC RO/TNT/DSS

K/A

028 Pressurizer (PZR) Level Control Malfunction

AK1.01 Knowledge of the operational implications of the following concepts as they apply to Pressurizer Level Control Malfunction.

PZR reference leak abnormalities.

K/A MATCH ANALYSIS

Question gives plant status with PZR level control selector switch in the 461/460 position. LT-461 would be the controling channel in this configuration. Candidate must be able to identify the controlling channel plus the effects of a small reference leg break on the controlling channel. A reference leg break would decrease the density of the reference leg relative to the impulse leg resulting in an artificially high level sensed on the controlling channel. PZR level control would reduce charging flow in response to the artificially sensed high level.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Level would read higher than actual if the controlling channel. This is not the controlling channel.

B. Incorrect. Level reads higher than actual. Charging flow would decrease.

C. Incorrect. Level would read higher than actual if the controlling channel. This is not the controlling channel.

D. Correct. Level reads higher than actual and charging flow would decrease.

REFERENCES

Vogtle Exam Bank question # LO-PP-16302-02-11

Vogtle Exam Bank question # LO-PP-16302-07-03

VEGP learning objectives:

LO-LP-60301-12 Given that the pressurizer level control selector switch is in the NORMAL position (459/460), describe how and why the Pressurizer Level Control System will respond to the following instrument failures. Consider each separately and include effects on pressurizer level control, alarms, RPS, and ESF actuations.

a. 459 fails high
b. 459 fails low

27. 029EK2.06 001/1/1/ATWS - BRKR RLY DISC/C/A 2.6/BANK/RO/NRC RO/TNT/DSS

During an ATWT event, locally opening the reactor trip and bypass breakers would **DIRECTLY** cause which **ONE** of the following to occur ?

A. A Main Turbine Trip, automatically starts the AFW pumps, and closes the MFRV and Bypass valves.

B. The rods to drop into the core, seals in the feedwater isolation signal with a LO-LO Tavg signal in, and trips the MFPTs.

C. The rods to drop into the core, automatically blocks the Safety Injection signal, and closes the MFRV and Bypass valves.

D.✓ A Main Turbine trip, arms the Steam Dumps, and places the plant trip controller into control of the Steam Dumps.

27. 029EK2.06 001/1/1/ATWS - BRKR RLY DISC/C/A 2.6/BANK/RO/NRC RO/TNT/DSS

K/A

029 Anticipated Transient Without Scram (ATWS)

EK2.06 Knowledge of the interrelations between the following and ATWS:

Breakers, relays, and disconnects

K/A MATCH ANALYSIS

Question gives a plausible scenario where an ATWS is in progress and Rx trip and bypass breakers opened locally. Candidate must be able to choose correctly effects of P-4 signal on plant and control systems.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Would trip the turbine but not auto start AFW pumps or shut FRV and Bypass valves unless Lo Tavg signal < 564 degrees F is present.

B. Incorrect. Rods would drop into the core but FWI does not seal in unless an SI signal or P-14 signal (Hi Hi SG level) is present, MFPTs do not trip on receipt of P-4.

C. Incorrect. Rods would drop into the core but allows operator to manually block SI, this does not automatically occur, MFRV and Bypass would shut if Lo Tavg present.

D. Correct. A turbine trip would occur, steam dumps would arm and shift to the plant trip mode of operation.

REFERENCES

Farley August of 2004 NRC RO examination question # 31

Vogtle Text Chapter 28 excerpt on P-4.

VEGP learning objectives:

LO-PP-28103-02 List all permissives with applicable set points, coincidences, and functions.

SECTION B

REACTOR TRIP AND ESFAS SIGNALS

28.11 PERMISSIVE INTERLOCKS

Permissive interlocks provide input to the protection systems to allow or prevent protective functions from occurring under certain plant conditions.

P-4 **Indicates reactor tripped**

Set point or conditions that give P-4

RTA and its bypass (BYA) both open give P-4 Train A
RTB and its bypass (BYB) both open give P-4 Train B

Function:

1) **Trips** the Main Turbine to limit the RCS cool down

- P-4 Train A generates a "Mechanical Turbine Trip"
- P-4 Train B generates an "Electrical Turbine Trip"

2) Steam Dumps

- P-4 Train A generates a Steam Dump Arming signal
- P-4 Train B transfers Steam Dump controllers from "Load reject" mode to the "Plant trip" mode

3) Feed Water Isolation (FWI)

- P-4 in conjunction with Lo Tavg of 564°F

plausible distrator

4) Seals in FWI if caused by a Safety Injection or Hi-Hi Steam Generator water level (P-14).

Not Lo-Lo

allows rest NOT auto block

5) SI reset logic

- After Safety Injection has been reset, P-4 blocks any future automatic safety injection signals.

P-6 **Source Range Block Permissive**

Set point:

2.0 x 10-5% POWER on any **1 / 2** IR NIS detector.

Function:

1) Allows the operator to manually block SR high flux trip.
(both TRN A and TRN B switches, QMCB-C)

2) Loss of P-6 (either train no SSPS) will automatically unblock the Source Range Trip Permissive status light on BPLP (QMCB-C) Illuminates when P-6 is present.

1. 029EK2.06 001

During an ATWT event, opening the reactor trip and bypass breakers locally causes which one of the following to occur?

A.✓ A Main Turbine trip, arms the Steam Dumps, and places the plant trip controller into control of the Steam Dumps.

B. The rods to drop into the core, automatically blocks the Safety Injection signal, and closes the FRV and Bypass valves.

C. The rods to drop into the core, seals in the feedwater isolation signal with a LO-LO Tavg signal in, and trips the SGFPs.

D. Resets the high steam flow setpoint to 40 percent, automatically starts the AFW pumps, and closes the FRV and Bypass valves.

A. Correct-**A Main Turbine trip, arms the Steam Dumps, and places the plant trip controller into control of the Steam Dumps.**
This is a few of the functions of P-4 permissive.

B. Incorrect- P-4 does not automatically block an SI signal but allows the operator to block the SI signal after a time delay, does not auto close the FRV and Bypass valves unless temp is < 554°F, Lo Tavg.

C. Incorrect - does not trip the SGFPs and does not technically seal in the FW isolation signal on a LO tavg, but definitely does not seal in on a Lo-Lo Tavg.

D. Incorrect - Does not auto start the AFW pumps. does not always close the FRV and bypass valves, only in conjunction with a LO Tavg.

<u>OPS--52201I</u>
P-4 Permissive
The P-4 permissive is generated when both the reactor trip breaker and the bypass breaker, which physically bypasses it, are open. Train A of the reactor protection system uses RTA and BYA, and train B uses RTB and BYB. The following are functions of P-4:
1. Causes a turbine trip
2. Closes main feedwater regulating valves and feedwater bypass valves if low Tavg (554°F) is also present
3. Seals in feedwater isolation signal from safety injection or steam generator high-high water level
4. Resets high steam flow setpoint to 40 percent
5. Arms steam dumps on a plant trip, defeats the output of the load rejection controller, and places the plant trip controller into control.
6. Allows operator block of the safety injection signal after a time delay
This last feature ensures that the reactor is tripped and that all emergency core coolant system (ECCS) loads are started before the operator overrides what could be a spurious actuation signal. The block does not prevent the operator from reinitiating safety injection through use of either manual safety injection actuation switch.

Goes with Q #27

28. 033G2.1.33 001/2/2/SFPC - TCH SP ENTRY/MEM - 3.4/NEW/RO/NRC RO/TNT/DSS

The following parameters are noted for the Spent Fuel Pool (SFP). during spent fuel movement

* Level - 216 ft, 4 inches

* Temperature - 132 degrees F

* Chemistry reports SFP boron - 2020 ppm Cb.

Which **ONE** of the following would be **CORRECT** regarding these indications ?

A.✓ Level is below the Tech Spec limit and an LCO entry required.

B. Temperature is above the high limit and an LCO entry required.

C. Boron concentration is below the Tech Spec Limit and an LCO entry required.

D. No Tech Spec LCO entry is required, all parameters listed within Tech Spec limits.

28. 033G2.1.33 001/2/2/SFPC - TCH SP ENTRY/MEM - 3.4/NEW/RO/NRC RO/TNT/DSS

K/A

033 Spent Fuel Pool Cooling System (SFPCS)

G2.1.33 Ability to recognize inidcations for system operating parameters which are entry-level conditions for technical specifications.

K/A MATCH ANALYSIS

Question gives 3 plausible indications associated with the SFP and the candidate has to choose which one is out of limits.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. Level is lower than the Tech Spec limit (217 ft 0 inches or 216 ft 5 inches with the gates open.

B. Incorrect. Above high temperature alarm setpoint but no Tech Specs associated.

C. Incorrect. Boron is above the low limit of 2000 ppm.

D. Incorrect. Level is lower than Tech Spec limit, not temperature.

REFERENCES

Technical Specifications 3.7.15 for Fuel Storage Pool Water Level.

Technical Specifications 3.7.17 for Fuel Storage Pool Boron Concentration.

ARP-17005-1/2, Window A06 for Spent Fuel Pit High Temperature

ARP-17005-1/2, Window E02 for Spent Fuel Pit Lo Level.

VEGP learning objectives:

LO-LP-39211-02 Given a set of Tech Specs and the bases, determine for a specific set of plant conditions, eqiupment availability, and operational mode:

a. Whether any Tech Spec LCOs of section 3.7 are exceeded.

b. The required actions for all section 3.7 LCOs.

3.7 PLANT SYSTEMS

3.7.15 Fuel Storage Pool Water Level

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the fuel storage pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.17 Fuel Storage Pool Boron Concentration

LCO 3.7.17 The fuel storage pool boron concentration shall be ≥ 2000 ppm.

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	------------------NOTE------------------ LCO 3.0.3 is not applicable. ---	
	A.1 Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	AND	
	A.2.1 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify the fuel storage pool boron concentration is within limit.	7 days

WINDOW A06

ORIGIN	**SETPOINT**	SPENT FUEL PIT HI TEMP
1-TISH-626	130°F	

1.0 **PROBABLE CAUSE**

1. Spent Fuel Pool Pump trip.
2. Loss of Component Cooling Water (CCW) flow to Spent Fuel Pool Heat Exchanger.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

Go to 18030-C, "Loss Of Spent Fuel Pool Level Or Cooling."

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NONE

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB130, PLS

WINDOW E02

ORIGIN	SETPOINT	
1-LSHL-625	217 feet elevation	SPENT FUEL PIT LO LEVEL

1.0 **PROBABLE CAUSE**

1. Insufficient inventory during filling or refueling operation.
2. Normal evaporation.
3. System leak.
4. Loss of air to the Fuel Transfer Canal and/or Cask Loading Pit Gate Seals.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. Dispatch an operator to determine actual level locally. (see Figure 1 in this procedure).
2. Refer to 13719-1, "Spent Fuel Pool Cooling And Purification" and return the Spent Fuel Pit to normal level (218.5 feet).
3. **If level cannot be maintained greater than 217 feet with fuel movement in containment in progress or 216.5 feet with the Spent Fuel Pool Gate Valve closed, then suspend movement of irradiated fuel assemblies in the Spent Fuel Pool and all crane operations over the Spent Fuel Pool. Initiate 18030-C, "Loss Of Spent Fuel Pool Level Or Cooling" and 18006-C "Fuel Handling Event."**
4. Check service air to gate seals and refer to 13710-1, "Service Air System" to restore service air if lost.
5. Refer to Technical Specification LCO 3.7.15.

29. 033G2.4.1 001/1/2/IR NIS - EOP ENTRY/C/A - 4.3/NEW/RO/NRC RO/TNT/DSS

A reactor startup is in progress, the operators have stabilized the plant to take critical data when the following sequence of events occurs:

* A loss of vital bus BY1B occurs, reactor power remains stable at the power level for taking critical data.

Which **ONE** of the following are the next **CORRECT** actions for the crew to take ?

A. Place rods in MANUAL, control SG NR levels in MANUAL from 60 - 70% NR.

B. Suspend all operations involving positive reactivity changes, maintain power stable.

C.✓ Manually trip the reactor and go to 19000-C, "E-0 Reactor Trip or Safety Injection"

D. Place rods in MANUAL, reduce reactor power to < P-6 within the next two hours.

29. 033G2.4.1 001/1/2/IR NIS - EOP ENTRY/C/A - 4.3/NEW/RO/NRC RO/TNT/DSS

K/A

033 Loss of Intermediate Range Nuclear Instrumentation

G2.4.1 Knowledge of EOP entry conditions and immediate action steps.

K/A MATCH ANALYSIS

Question gives a plausible scenario where IR NIS fails between P-6 and 10% power. A reactor trip should occur but does not. Operator should trip reactor in accordance with 18032-C, section C for Loss of Vital Instrument Panel BY1B.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Correctr IOAs of 18032 if you don't realize ATWT in progress. Reactor should be manually tripped per 18032, section C step # 1.

B. Incorrect. IOAs for SR/IR malfunction AOP, but an ATWT is in progress. Reactor should be manually tripped per 18032, section C step # 1.

C. Correct. Reactor should be manually tripped per 18032, section C step # 1.

D. Incorrect. Reactor should be manually tripped per 18032, section C step # 1. This is combination of IOAs for 18032 and Tech Spec actions for loss SR NIS.

REFERENCES

AOP-18032, "Loss of 120V AC Instrument Power, section C for loss of BYIB.

AOP 18002-C, "NIS Malfunction" section A forSource / Intermediate Range Channel Malfunction

Tech Specs 3.3.1 for Reactor Trip Instrumentation

VEGP learning objectives:

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

c. BY1B

Approval	Vogtle Electric Generating Plant NUCLEAR OPERATIONS	Procedure No. 18032-1
		Revision No. 23
Date	Unit 1	Page No. 1 of 90

Abnormal Operating Procedures

LOSS OF 120V AC INSTRUMENT POWER

PURPOSE

PRB REVIEW REQUIRED

This procedure specifies the response to a loss of a single 120V AC Vital or Essential Instrument Panel. The reactor is assumed to be at power.

Specific instructional steps will be found in the following subsections:

SYMPTOMS

Symptoms are identified in the individual sub-procedures.

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

SYMPTOMS

- All Channel II trip status lights (except P-6, CNMT HI-3 and RWST LO-LO LEVEL) energized.
- Verify AMSAC is in service if above 40% turbine power by observing permissive light AMSAC BYPASSED LO TURBINE LOAD C20 (BPLB 8-4) extinguished.
- Reactor trip due to loss of NI channel.
- 120V AC PANELS 1BY1B 1BY2B TROUBLE alarm at QEAB.
- INVERTERS 1BD1I2 1BD1I12 TROUBLE alarm at QEAB.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
IMMEDIATE OPERATOR ACTIONS	

NOTE: Below P-10 the reactor should trip due to loss of power to a Source and Intermediate range channel.

C1. CHECK reactor power - GREATER THAN P-10 SETPOINT.	C1. VERIFY reactor trip and go to 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.
C2. ENSURE rod bank select switch is in MANUAL.	
C3. CONTROL SG levels in MANUAL from 60% to 70% NR; • MFRVs in MANUAL. • MFP MASTER CONTROLLER in MANUAL.	Plausible distracton

A. SOURCE/INTERMEDIATE RANGE CHANNEL MALFUNCTION

SYMPTOMS

- SR/IR SIGNAL PROCESSOR TROUBLE
- SR/IR REMOTE SIG PROCESSOR - DPU-B TROUBLE
- SR/IR AMPLIFIER TROUBLE ALARM
- INCONSISTENT INDICATION BETWEEN THE TWO SOURCE RANGE (INTERMEDIATE RANGE) CHANNELS
- ERRATIC OPERATION
- INTMD RANGE HIGH FLUX LEVEL ROD STOP alarm
- Improper overlap of Source and Intermediate Range channel
- Improper overlap of Power Range and Intermediate Range channels

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
IMMEDIATE OPERATOR ACTIONS	
A1. Suspend all operations involving positive reactivity changes and stabilize count rate. Suspend core alterations if in MODE 6.	Plausibe distracton
SUBSEQUENT OPERATOR ACTIONS	
A2. Check power less than P-6.	A2. Go to Step A5.
A3. Select an operable Source and Intermediate Range channel on NR-45.	
A4. Verify valves 1208-U4-175, 1208-U4-176, 1208-U4-177, and 1208-U4-183 are locked closed if required by Technical Specifications 3.3.8 or 3.9.2.	

Table 3.3.1-1 (page 1 of 9)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT(n)
1.	Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA	NA
		3(a), 4(a), 5(a)	2	C	SR 3.3.1.13	NA	NA
2.	Power Range Neutron Flux						
	a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 111.3% RTP	109% RTP
	b. Low	1(b),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 27.3% RTP	25% RTP
3.	Power Range Neutron Flux High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.3% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec
4.	**Intermediate Range Neutron Flux**	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 41.9% RTP	25% RTP
		2(d)	2	H	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 41.9% RTP	25% RTP

(continued)

(a) With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(n) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is readjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	------------------NOTE------------------ A channel may be bypassed for up to 12 hours for surveillance testing. ---	
	E.1 Place channel in trip.	72 hours
	OR	
	E.2 Be in MODE 3.	78 hours
F. THERMAL POWER > P-6 and < P-10, one Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6.	24 hours
	OR	
	F.2 Increase THERMAL POWER to > P-10.	24 hours
G. THERMAL POWER > P-6 and < P-10, two Intermediate Range Neutron Flux channels inoperable.	G.1 Suspend operations involving positive reactivity additions.	Immediately
	AND	Reasonable distraction
	G.2 Reduce THERMAL POWER to < P-6.	2 hours
H. THERMAL POWER < P-6, one or two Intermediate Range Neutron Flux channels inoperable.	H.1 Restore channel(s) to OPERABLE status.	Prior to increasing THERMAL POWER to > P-6

(continued)

30. 034K1.01 001/2/2/FH EQUIP - RCS/C/A - 2.5/NEW/RO/NRC RO/TNT/DSS

Which **ONE** of the following would **CORRECTLY** describe the flow path for RHR Train "A" cooling during fuel movement and QMCB **indicated** flow rate that ensures Tech Spec Flow requirements are met ?

A. supply back into Hot Legs # 1 and # 2, > 3000 gpm

B.✓ supply back into Cold Legs # 1 and # 2, > 3200 gpm

C. supply back into Hot Legs # 1 and # 4, > 3200 gpm.

D. supply back into Cold Legs # 1 and # 4, > 3000 gpm.

K/A mismatch – licensee to develop new question

30. 034K1.01 001/2/2/FH EQUIP - RCS/C/A - 2.5/NEW/RO/NRC RO/TNT/DSS

K/A

034 Fuel Handling Equipment System (FHES)

K1.01 Knowledge of the physical connections and/or cause-effect relationships between the Fuel Handling System and the following system:

RCS

K/A MATCH ANALYSIS

Question asks straightforward the suction and discharge path for RHR Train "A" during refueling operations and the required indicated system flow to ensure Tech Spec flow requirements met. RHR Train "A" suction is Loop # 1 Hot Leg and injects into Loop # 1and # 2 Cold Legs to ensure flow through the core. >3200 gpm is required indicated flow rate to ensure >3000 gpm Tech Spec limit is met.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Injects back into Cold Legs. Plausible because during cold leg recirc some ECCS pumps continue to inject into Hot Legs, RHR has this capability. Flow must be > 3200 to ensure > 3000 gpm Tech Spec limit due to calibration characteristics of RHR flow instruments.

B. Correct. Suction from Hot Leg # 1 and injects back into Cold Legs # 1 and # 2. Indicated flow must be > 3200 gpm to insure Tech Spec limits on flow met.

C. Incorrect. Injects into Cold Legs # 1 and # 2 but flow rate is correct.

D. Incorrect. Suction from Hot Leg # 1, injects into Cold Legs # 1 and # 2. Flow must be >3200 gpm to ensure > 3000 gpm Tech Spec Limits.

REFERENCES

19014-C, "Transfer to Hot Leg Recirculation"

OSP-14000, "Control Room Tech Spec Rounds"

VEGP learning objectives:

Not applicable.

DATA SHEET 3 - MODE 5 & 6 MODE ________ Sheet 2 of 5

DATE ________

LCO METHOD OF VERIFICATION	TECH SPEC SURV REQ	PARAMETER		INSTRUMENT	INDICATION DAY	INDICATION NIGHT	LIMIT(S) TOLERANCE	LCO/PROC
AT LEAST 1 RHR TRAIN SHALL BE IN OPERATION AND THE REQUIRED RHR TRAINS OR THE REQUIRED SGS OPERABLE	3.4.7.1 3.4.7.2 (MODE 5, LOOPS FILLED)	RHR TRAINS (A, B)		ENTER TRAINS OPERABLE			***	3.4.7
		RHR FLOW (GPM)		1FIC-0618A			* ≥3200	
		RHR FLOW (GPM)		1FIC-0619A			* ≥3200	
VERIFY RHR CIRCULATION AND OR SG LEVELS	3.4.8.1 (MODE 5 LOOPS NOT FILLED)	STEAM GENERATOR LEVEL (%)	1	1LI-0501				3.4.8
			2	1LI-0502				
			3	1LI-0503				
			4	1LI-0504				
	SR 3.9.5.1 (MODE 6 ≥23' ABOVE FLANGE)	*** MODE 5, LOOPS FILLED - AT LEAST 1 RHR TRAIN IN OPERATION AND ONE ADDITIONAL RHR TRAIN SHALL BE OPERABLE OR THE SECONDARY SIDE WATER LEVEL OF AT LEAST TWO STEAM GENERATORS ≥63% WIDE RANGE. STEAM GENERATORS MAY NOT BE USED AS AN OPTION TO AN RHR TRAIN UNLESS THE RCS IS FILLED GREATER THAN 15% PRESSURIZER LEVEL AND RCS PRESSURE HAS BEEN MAINTAINED >100 PSIG SINCE THE MOST RECENT FILL & VENT. MODE 5, LOOPS NOT FILLED - AT LEAST 2 RHR TRAINS OPERABLE WITH 1 TRAIN IN OPERATION. MODE 6, ≥23 FT ABOVE FLANGE - AT LEAST 1 RHR TRAIN OPERATING WITH ≥3000 GPM# FLOW. MODE 6, <23 FT ABOVE FLANGE - AT LEAST 2 RHR TRAINS OPERABLE WITH 1 TRAIN IN OPERATION WITH >3000 GPM# FLOW.						3.9.5
	SR 3.9.6.1 (MODE 6 <23' ABOVE FLANGE)	* ≥3200 GPM ENSURES ≥3000 GPM ACTUAL FLOW AT ALL TEMPERATURES RHR FLOW OF ≥3000 GPM IS ONLY REQUIRED IN MODE 6 AND IS N/A IN MODE 5						3.9.6
STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATIONS MONITOR RCS AND SG PRESSURE	TRS 13.7.1.1	SG PRESSURE (PSIG)		1PI-514A			<200 PSIG IF SG OR RCS PRESSURE IS ≥200 PSIG, PERFORM THE NEXT SECTION TAKING SG/RCS TEMPERATURES	13.7.1 CRITERIA APPLICATION REPORT
				1PI-524A				
				1PI-534A				
				1PI-544A				
		RCS PRESSURE (PSIG)		1PI-408				
				1PI-418				
				1PI-428				
				1PI-438				
STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATIONS MONITOR RCS AND SG TEMPERATURE	TRS 13.7.1.1	SG TEMPERATURE (°F)		1TI-1175			>70°F	14915-1 DATA SHEET # 8
				1TI-1176				
				1TI-1177				
				1TI-1178				
		NOTES: IF SG TEMPERATURES ARE LESS THAN 200°F, USE COMPUTER POINTS T9883, T9884, T9885, T9886 IF BLOWDOWN IS NOT IN SERVICE OR TIS ARE NOT AVAILABLE, USE A CONTACT PYROMETER TO MEASURE SKIN TEMPERATURE.						
		RCS TEMPERATURE (°F)		1TI-0413B				
				1TI-0423B				
				1TI-0433B				
				1TI-0443B				
		IF REACTOR OR SECONDARY SIDE TEMPERATURE IS ≤70 DEGREES COMMENCE LOGGING REACTOR AND SECONDARY PRESSURES PER 14915-1 DATA SHEET 8 WITHIN 1 HOUR.						

This page may be signed off before contact skin temps. are completed.

COMPLETED BY: DAY: ________ TIME: ________ NIGHT: ________ TIME: ________

SS REVIEW: DAY: ________ TIME: ________ NIGHT: ________ TIME: ________

31. 035K6.03 001/2/2/SG - LEVEL DETECTOR/C/A - 2.6/NEW/RO/NRC RO/TNT/DSS

While at full power:

Which **ONE** of the following would be **CORRECT** regarding the Steam Generator Level Control System (SGWLC) response to a **HIGH** failure of the controlling SG NR level channel **assuming no operator actions** ?

A. MFRV opens to raise SG level, turbine trips on P-14.

B.✓ MFRV closes to lower SG level, reactor trips on SG lo lo level.

C. MFRV opens to raise SG level, other detector inputs (SG feed flow, SG steam flow, SG pressure) will offset the failure and return level to program.

D. MFRV closes to lower SG level, other detector inputs (SG feed flow, SG steam flow, SG pressure) will offset the failure and return level to program.

31. 035K6.03 001/2/2/SG - LEVEL DETECTOR/C/A - 2.6/NEW/RO/NRC RO/TNT/DSS

K/A

035 Steam Generator System (SGS)

K6.03 Knowledge of the effect that a loss or malfunction on the following will have on the SGs.

S/G level detector

K/A MATCH ANALYSIS

Question gives a plausible scenario where a SG level transmitter fails and candidate must determine the correct response(s) to the failure. In this scenario the detector fails high and the MFRV would close until the reactor trips on lo lo SG levels of 38.6%. SG level is predominant detector.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. MFRV would close lowering SG level.

B. Correct. MFRV would close lowering SG level to lo lo level trip setpoint.

C. Incorrect. MFRV would close lowering level. Also, level is predominant. Plausible since level could possibly overcome failure of one of other transmitters but this is reverse from actual.

D. Incorrect. MFRV would close lowering level. Plausible since level could possibly overcome failure of one of other transmitters but this is reverse from actual.

REFERENCES

LO-LP-60301-12-C excerpts on SG LI Failure.

VEGP learning objectives:

LO-LP-60301-13 Describe how and why the Steam Generator Level Control System will respond to a steam generator level control channel instrument failure. Include effects on SGFP speed control, alarms, and RPS actuations.

a. If PRZR level is not trending to program level take manual control of charging flow

1) Be mindful of letdown flashing if charging flow lowered too much

Maintain charging flow to prevent an Additional thermal cycle on the charging nozzle

2) Isolate letdown if flashing occurs

b. Maintain seal injection flow 8-13 gpm

1) Excess letdown would be needed only if normal charging or normal letdown is not available

c. Select an unaffected channel for control and on recorder

d. Restore letdown flow if required or initiate excess letdown if normal letdown unavailable

1) Refer to 18007-C, if cannot be restored timely

e. Restore PRZR heaters, it tripped off due to failure

f. Return charging flow to AUTO when stable and verify stable control

g. Contact Maintenance to initiate repairs.

h. If desired, bypass the affected channel per 13509-C (Bypass Test Instrumentation Panel Operation).

i. Within 72 hours, trip the affected channel bistables per table guidance. If unable to comply, then place the reactor in Hot Standby within 78 hours.

i. Place Master Test switch to Test per table guidance.

j. Restore normal configuration when repairs complete.

F. Steam Generator Level Instrument Failure

1. Symptoms, alarms and indications Objective 1

a. STM GEN 1 (2, 3, 4) HI/LO LVL DEVIATION; + 5% from program (controlling channel only)

b. STM GEN 1 (2, 3, 4) HI-HI LEVEL ALERT; 1/4 channels at 83% level (P-14)

1) Controlling channel failing high will not cause P-14 (P-14 requires 2/4 > 83%)

2) If controlling channel fails low and actual level increases to 83%, a P-14 turbine trip, FWI will occur

c. STM GEN 1 (2, 3, 4) LO LEVEL;< 44% level in controlling channel

d. STM GEN 1 (2, 3, 4) LO-LO LVL ALERT; 1/4 channels 37.8%

1) Controlling channel failing low will not cause direct Rx trip (Trip requires 2/4 < 37.8%)

2) If controlling channel fails high and no intervention, then other channels go to Rx trip setpoint

2. Response to controlling channel failure — **Objective 13**

a. A level channel failure will result in a level deviation alarm and a trip input

b. Control response will only occur if the selected control channel fails

c. Controlling channel failing high — **Objective 1**

1) The output from the control system will throttle down on the feed regulation valve attempting to lower level

2) The actual level will decrease

3) With no operator action level will fall to the low-low level setpoint resulting in a reactor trip and aux feedwater actuation

4) Feed pump speed control receives no direct input from SG level, however the feed pump will respond to the changes in the feedwater system caused by the reduction in feed flow to the affected steam generator.

This results in an increase Feed Flow to unaffected SG's

3. Actions — **Objective 20**

a. Immediate action - if steam flow/feed flow is not matched and level not in program — **Objective 14**

c. LT-460 FAILS HIGH

None

d. LT-460 FAILS LOW

Letdown isolates

Actual level increases

13. DESCRIBE HOW AND WHY THE STEAM GENERATOR LEVEL CONTROL SYSTEM WILL RESPOND TO A STEAM GENERATOR LEVEL CONTROL CHANNEL INSTRUMENT FAILURE. INCLUDE EFFECTS ON SGFP SPEED CONTROL, ALARMS, AND RPS ACTUATIONS

A level channel failure will result in a level deviation alarm and a channel trip input

Control responses will only occur if the selected control channel fails

Controlling channel failing high

The output from the control system will throttle down on the feed regulating valve attempting to lower level

The actual level will decrease

With no operator action level will fall to the low-low level setpoint resulting in a reactor trip and aux feedwater actuation

Feed pump speed control receives no direct input from SG level; however, the feed pump will slow down in respond to the changes in the feedwater system caused by the reduction in feedflow to the affected steam generator

14. STATE THE IMMEDIATE OPERATOR ACTION REQUIRED FOR A FAILED STEAM GENERATOR LEVEL INSTRUMENT. INCLUDE RNO AND SUBSTEPS OF THE IMMEDIATE ACTION

Verify all SGs:

Stm/Feed flow - MATCHED

Levels at 65% NR

RNO:

Manually control SG level between 60-70% NR

15. DESCRIBE HOW AND WHY THE STEAM GENERATOR LEVEL CONTROL SYSTEM WILL RESPOND TO A STEAM GENERATOR PRESSURE INSTRUMENT FAILING IN THE FOLLOWING DIRECTIONS. CONSIDER EACH SEPARATELY AND INCLUDE EFFECT ON SGFP SPEED CONTROL RESPONSE, ALARMS, AND RPS ACTUATIONS

32. 036AA2.01 001/1/2/FH EVNT - ARM INDICA/MEM - 3.2/NEW/RO/NRC RO/TNT/DSS

A dropped fuel assembly in the East Spent Fuel Pool has resulted in the following radiation monitor alarms:

* **1RE-0008**, "Fuel Handling Building Area Monitor" is in **HIGH** alarm.

* **ARE-2532A, 2533A and 2533B** are all in **INTERMEDIATE** alarm.

Which **ONE** of the following would be a **CORRECT** ~~regarding a FHB Isolation actuation~~ response to the ARM alarms?

A.✓ Neither train of the FHB HVAC Filter systems should have started.

B. Train A of the FHB HVAC Filter systems should have started.

C. Train B of the FHB HVAC Filter systems should have started.

D. Both trains of the FHB HVAC Filter systems should have started.

K/A mismatch as written

32. 036AA2.01 001/1/2/FH EVNT - ARM INDICA/MEM - 3.2/NEW/RO/NRC RO/TNT/DSS

K/A

036 Fuel Handling Incidents

AA2.01 Ability to determine and interpret the following as they apply to the Fuel Handling Incidents.

ARM system indications.

K/A MATCH ANALYSIS

Question gives plant status with a dropped fuel assembly occuring near 1RE-0008, 1RE-0008 has no automatic actuations associated with it. FHB HVAC will not start until ARE-2532A/B or ARE-2533A/B are in high alarm.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. None of the FHB HVAC units will start.

B. Incorrect. None of the FHB HVAC units will start.

C. Incorrect. None of the FHB HVAC units will start.

D. Incorrect. None of the FHB HVAC units will start.

REFERENCES

ARP-17005, window C03 HIGH radiation

V-LO-PP-23101, FHB HVAC Actuation Signals

ARP-17100, "Annunciator Response for Process and Effluent Rad Monitors"

VEGP learning objectives:

LO-PP-23101-09 Explain how the Fuel Handling Building HVAC System responds to a Fuel Handling Building Isolation Signal.

WINDOW C03

ORIGIN	SETPOINT	
K-1	Not Applicable	HIGH RADIATION ALARM

1.0 **PROBABLE CAUSE**

A high alarm on one or more of the Radiation Monitor Channels.

2.0 **AUTOMATIC ACTIONS**

The following actions will occur if a High Level Radiation Alarm is actuated on the associated monitor:

1. 1-RE-0002 or 1-RE-0003, Containment Low Range Area Monitor: Containment Ventilation Isolation (CVI).
2. A-RE-0014, Waste Gas Processing System Effluent Radiogas Monitor: Closes valve A-RV-0014 to the Waste Gas Processing System discharge.
3. 1-RE-0018, Waste Liquid Effluent Monitor: Closes 1-RV-0018 to isolate the Liquid Waste Discharge Line.
4. 1-RE-0021, Steam Generator Blowdown Liquid Process Monitor: Isolates Steam Generator Blowdown Processing System.
5. 1-RE-0848, Turbine Building Drain Effluent Monitor: Diverts Turbine Building Drains to Dirty Drains Tank.
6. A-RE-2532 A or B or A-RE-2533 A or B, Fuel Handling Building Effluent Radiogas Monitors: Fuel Handling Building Isolation (FHBI).
7. 1-RE-2565 A, B or C, Containment Ventilation Effluent Monitors: Containment Ventilation Isolation (CVI).

FHB Actuation Signals

- <u>Manual</u> (objective 2)
  - 1/2 switches on Unit 1 HVAC Panel
    - Note only one required to satisfy TRM instrumentation requirements.
- <u>Hi Radiation</u> on any of the following: (objective 2)
  - 1/4 exhaust duct radiation detectors
  - RE-2532A, B
  - RE-2533A, B
    - Note: 4 channels and only one required to satisfy TRM instrumentation requirements.

1-LO-PP-23101 FHB HVAC

ORIGIN

Area Monitor

SETPOINT

As determined by Chemistry Department

1-RE-0008
(High)

NOTE For other than HIGH conditions see Pages 4 and 5.

1.0 **PROBABLE CAUSE**

Increase in radiation level near Unit 1 Spent Fuel Pool in the Fuel Handling Building.

2.0 **AUTOMATIC ACTIONS**

On the south wall of the Fuel Handling Building Spent Fuel Pool Room near the door:

a. Alarm horn on 1-RA-0008 sounds,

b. Strobe light on 1-RA-0008 blinks.

3.0 **INITIAL OPERATOR ACTIONS**

Evacuate the Fuel Handling Building.

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. Check for elevated radiation levels on A-RE-2533A and A-RE-2533B on the SRDC.

2. Notify Health Physics to survey Spent Fuel Pool area to determine cause of the alarm.

3. Isolate the source of radioactivity if possible.

4. Refer to 91001-C, "Emergency Classification And Implementing Instructions".

5. Obtain detector trend data per 13508-1, "Radiation Monitoring Systems".

6. Monitor the channel for further changes.

7. If sampling and analysis determine the channel has malfunctioned, request Chemistry to deactivate the channel.

33. 037AK1.02 001/1/2/SGTR - LEAK VS DP/C/A - 3.5/BANK/RO/NRC RO/TNT/DSS

During operation at power SG tube leakage is detected and estimated to be 250 gpm by the Reactor Operator. The following plant conditions exist:

* RCS pressure - 2200 psig and lowering
* Reactor power - 80%
* SG pressures - 1000 psig
* PRZR level - 42% and lowering

The unit is then tripped and plant parameters following the trip are:

* RCS pressure - 1700 psig and lowering
* Reactor power - 0%
* SG pressures - 1100 psig
* PRZR level - 13%

Based on the two sets of given data, which **ONE** of the following describes the effect on primary-to-secondary leakage ?

Leakage following the trip is...

A. one third of the initial leak rate or about 83 gpm.

B✓ approximately 70% of the initial leak rate or about 175 gpm.

C. essentially equal to the intial leak rate or about 250 gpm.

D. one half of the intial leak rate of about 125 gpm.

33. 037AK1.02 001/1/2/SGTR - LEAK VS DP/C/A - 3.5/BANK/RO/NRC RO/TNT/DSS

K/A

037 Steam Generator (SG) Tube Leak

AK1.02 Knowledge of the interrelations between the Steam Generator Tube Leak and the following:

Leak Rate vs. pressure drop

K/A MATCH ANALYSIS

Question gives a plausible scenario where a steam generator tube leak rate is given while the plant is at power with a given dp between RCS and SG pressure. The plant is then tripped and the candidate must calculate the new leak rate with a different dp between RCS and SG pressure. Values used are approximate plant values for full power and post trip for SGs.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Inverse of distractor B.

B. Correct. Leak rate is proportional to the square root of the dP.

C. Incorrect. As dP changes, flow rate also changes.

D. Incorrect. The dP is approximately half.

REFERENCES

Beaver Valley October 2004 NRC RO exam question # 59

VEGP learning objectives:

LO-LP-60309-02 Describe the actions necessary after shutdown to reach minimum break flow to the affected SG.

LO-LP-60309-03 Desribe why RCS pressure is maintained slightly less than the leaking SG pressure for leaks requiring a unit shutdown.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	037 AK1.02	
	Importance Rating	3.5	

Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak Leak Rate versus pressure drop

Proposed Question: Common 59

During operation at power SG tube leakage is detected and estimated at 250 gpm by the Reactor Operator. The following plant indications exist:

- RCS pressure - 2200 psig and lowering
- Reactor power - 80%
- SG pressures - 1000 psig
- PRZR level - 42% and lowering

The Unit is then tripped and plant parameters following the trip are:

- RCS pressure - 1700 psig and lowering
- Reactor power - 0%
- SG pressures - 1100 psig
- PRZR level - 13%

Based on the two sets of given data, which ONE of the following describes the effect on primary-to-secondary leakage?

Leakage following the trip is...

A. one half of the initial leak rate or about 125 gpm.

B. essentially equal to the initial leak rate or about 250 gpm.

C. approximately 70% of the initial leak rate or about 175 gpm.

D. One third of the initial leak rate or about 83 gpm.

Goes with Q #33

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect. The DP is approximately half.
B. Incorrect. As DP changes, flow rate also changes.
C. Correct. Leak rate is proportional to the square root of the DP.
D. Incorrect. Inverse of distractor C.

Technical Reference(s): Thermo (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: N/A (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 X
55.43 ____

Comments:
Indian Point 2 2001 Audit Exam – Western Tech Bank Archive

34. 038EK3.04 001/1/1/SGTR - PRM AUTO ACT/C/A - 3.9/NEW/RO/NRC RO/TNT/DSS

A steam generator tube rupture event is occuring on Unit 1. The following radiation monitors are in alarm:

* RE-0019 "SG Sample Liquid" - HIGH radiation
* RE-0021, "Steam Generator Blowdown Liquid Process" - INTERMEDIATE radiation
* RE-0724, "N16 Rad Monitor" - INTERMEDIATE radiation
* RE-0810, "SJAE Exhaust Rad" - HIGH radiation
* RE-0848, "Turbine Building Drain Effluent" - HIGH radiation
* RE-12839C, "Condenser Air Ejector and SPE Effluent" - INTERMEDIATE radiation

Which **ONE** of the following automatic actions would be **CORRECT** for the above listed radiation monitor conditions ?

A. Steam Generator Blowdown Processing system isolates.

B. RV-0018 closes to isolate the Liquid Waste Discharge Line.

C.✓ Turbine Building Drain system diverts flow to the Dirty Drain Tank.

D. SJAE and SPE discharges divert to the filtration mode of operation.

34. 038EK3.04 001/1/1/SGTR - PRM AUTO ACT/C/A - 3.9/NEW/RO/NRC RO/TNT/DSS

K/A

038 Steam Generator Tube Rupture (SGTR)

EK3.04 Knowledge of the reasons for the following responses as they apply to SGTR.

Automatic actions provided by each PRM.

K/A MATCH ANALYSIS

Question gives a plausible scenario where a SGTR is in progress with radiation monitors associated with SGTR in various stages of alarm. Candidate must choose which automatic action would be correct for alarm conditions listed.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. This would isolate on RE-0021 high radiation but is currently intermediate. Plausible candidate could confuse with RE-0019 which is in high alarm.

B. Incorrect. RE-0018 would isolate a liquid release, not RE-0810. Plausible candidate could confuse the two radiation monitors as this is a Human Performance possibility. This is the correct auto action for RE-0018 high alarm.

C. Correct. This is the correct action for RE-0848 high radiation.

D. Incorrect. This would divert to filter mode on high radiation but is currently intermediate. Plausible candidate could confuse with RE-0810 SJAE which is in high radiation condition.

REFERENCES

ARP-17100-C, "Annunciator Response Procedure For The Process And Effluent Radiation Monitoring System"

Page 21 for RE-0019, Page 28 for RE-0021, Page 37 for RE-0810, Page 39 for RE-0848, Page 60 for RE-12839C, Page 20 for RE-0018.

ARP-17005, Window B03 for Intermediate Radiation, shows no auto actions.

ARP-17005, Window C05 for High Radiation lists auto actions for High Radiation too.

AOP-18009-C, Steam Generator Tube Leak (monitors listed consistent with SGTL)

EOP-19030-C, Steam Generator Tube Rupture (monitors listed consistent with SGTR)

VEGP learning objectives:

LO-PP-45311-02 Describe the automatic actions that occur when radiation levels exceed the setpoint of the turbine building radiation monitor.

LO-LP-60309-07 Describe the response of the following parameters to a primary to secondary leak: (include in the discussion the response at power, and during a reactor startup)

g. Main steam line radiation monitors
h. Steam generator blowdown radiation monitors
i. Steam jet air ejectors and steam packing exhauster radiation monitor

ORIGIN	SETPOINT	
Liquid Effluent Monitor	As determined by Chemistry Department	1-RE-0018 (High)

NOTE

> For other than HIGH conditions see Pages 3—4 and 45.

1.0 **PROBABLE CAUSE**

Increase in radiation level in waste disposal system liquid being released.

2.0 **AUTOMATIC ACTIONS**

Isolates Waste Monitor Tank Discharge Header via valve 1-RV-0018.

3.0 **INITIAL OPERATOR ACTIONS**

ENSURE 1-RV-0018 closed.

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. Dispatch an operator to take action per 17213-1, "Annunciator Response Procedures For ALB On Waste Processing Panel - Liquid (PLPP)".
2. Notify Health Physics of the alarm.
3. Notify Chemistry of the alarm and that the release has terminated.
4. Refer to 91001-C, "Emergency Classifications and Implementing Instructions".
5. If sampling and analysis determine the channel has malfunctioned:
   a. Comply with ODCM Requirements,
   b. Request Chemistry to deactivate the channel.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB126, 1X5DS3C06

ORIGIN	SETPOINT	
Liquid Monitor	As determined by Chemistry Department	1-RE-0019 (High)

NOTE For other than HIGH or intermediate conditions see Pages ~~3~~4 and 45.

1.0 **PROBABLE CAUSE**

Increase in radiation in the Steam Generator Blowdown from:

a. Steam Generator tube leak,
b. Blowdown processing system malfunction.

2.0 **AUTOMATIC ACTIONS**

1. At the Steam Generator Blowdown Instrument Rack PSGI

   a. Alarm horn on 1-RA-0019 sounds,
   b. Strobe light on 1-RA-0019 blinks.

2. In the Component Cooling Water Train B Pump Room, 1-RX-1950 indicates the high radiation alarm.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

ORIGIN

Liquid Process

SETPOINT

As determined by Chemistry Department

1-RE-0021 (High)

NOTE

For other than HIGH conditions see Pages ~~3-~~4 and ~~4~~5.

1.0 **PROBABLE CAUSE**

Increase in radiation in the Steam Generator Blowdown from:

a. Steam Generator tube leak,

b. Blowdown processing system malfunction.

2.0 **AUTOMATIC ACTIONS**

1. Isolates Blowdown Heat Exchanger discharge to Blowdown Demineralizers via 1-FV-1150.
2. At the Steam Generator Blowdown Process Panel PSBP: 1-RI-0021 indicates current radiation level.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

ORIGIN

SJAE Rad
Monitor

SETPOINT

As determined by
Chemistry
Department

1-RE-0810
(High)

NOTES

- For other than HIGH conditions see Pages ~~3~~4 and 45.
- The IPC calculates primary-to-secondary leakage using the output of RE-0810 and SJAE exhaust flow, and displays GPD and GPD/HR on the IPC. For this indication to be valid IPC constant K6422 (SJAE in-service flag) must be updated for any change in SJAE status.
- The IPC indication of GPD and GPD/HR from RE-0810 will be "BAD" when the mechanical vacuum pumps are in-service.

1.0 PROBABLE CAUSE

Steam Generator Tube leakage

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

NONE

4.0 SUBSEQUENT OPERATOR ACTIONS

1. Evaluate plant parameters to determine if a Steam Generator Tube leak is indicated:
   a. VCT makeup frequency and/or Charging flow has increased.
   b. Pressurizer level and/or pressure has decreased.
   c. Steam Flow/Feed Flow mismatch and SG level response.

ORIGIN

Skid mounted
Liquid Effluent
Monitor

SETPOINT

As determined by
Chemistry
Department

1-RE-0848
(High)

NOTE

For other than HIGH conditions see Pages ~~3~~4 and 45.

1.0 **PROBABLE CAUSE**

High radiation level in the Turbine Building Drain effluent.

2.0 **AUTOMATIC ACTIONS**

1. Isolates Turbine Building Drain Header via 1-HV-877A.
2. Isolates the clean drain tank via 1-HV-844A.
3. Aligns the Turbine Building Drain Header to the Dirty Drain Tank via 1-HV-877B.
4. Aligns the Clean Drain Tank to the Dirty Drain Tank via 1-HV-844B.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

ORIGIN	**SETPOINT**	
Post Accident Wide Range Air Ejector Radiogas Monitor (Low Range) (Beta Scintillation)	As determined by Chemistry Department	1-RE-12839C (High)

NOTES

- For other than HIGH conditions see Pages ~~3~~4 and 45.
- This detector monitors activity in the range 10^{-7} to 10^{-2} micro Ci/cc, for the same gas as 1-RE-12839D and E.

1.0 **PROBABLE CAUSE**

Radioactive gas in the exhaust from SJAE and SPE from a primary to secondary system leak.

2.0 **AUTOMATIC ACTIONS**

1. Diverts SJAE Exhaust from Turbine Building Roof Vent to SJAE Exhaust Filtration unit via 1-HV-2875A, B and C.
2. Diverts SPE Exhaust from Turbine Building Roof Vent to SPE Exhaust Filtration Unit via 1-HV-2876A, B and C.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. ~~ENSURE~~ Verify SJAE and SPE discharge flow diverted to filtration banks.
2. Evaluate plant parameters to determine if a Steam Generator Tube leak is indicated:
   a. VCT makeup frequency and/or Charging flow has increased.
   b. Pressurizer level and/or pressure has decreased.
   c. Steam Flow/Feed Flow mismatch and SG level response.

WINDOW C03
(Continued)

8. 1-RE-12116 and 1-RE-12117, Control Room Intake Airborne Monitors: Control Room Ventilation Isolation (CRI).

9. 1-RE-12839 C, Condenser Air Ejector and Steam Packing Exhauster Effluent Monitor: Diverts air ejector discharge to filtration.

10. A-RE-50003, Technical Support Center Air Intake Monitor: Technical Support Center Ventilation Isolation.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

a. Check the Safety Related Display Console (QRM2), the RMS Communications Console (QRM1) and the Plant Computer to determine the monitor in alarm and go to 17100-1, "Annunciator Response Procedure For The Process And Effluent Radiation Monitor System (RMS)" or 17102-1, "Annunciator Response Procedure For The Safety Related Display Console QRM2" as appropriate.

b. Initiate a CR documenting Alarm condition.

5.0 **COMPENSATORY OPERATOR ACTIONS**

Monitor Plant Computer for radiation alarms if annunciator is inoperable or invalid.

END OF [illegible]-PROCEDURE

REFERENCES: AX4DB123-2, AX4DB129, AX4DB181, AX4DB20[illegible]-2, AX4DB235, 1X4DB126, 1X4DB142-1, 1X4DB179-2, 1X4DB180-1, 1X4DB206-1, 1X4DB213-1, 1X4DB213-2, 1X4DB229-3

35. 039A1.06 002/2/1/M STEAM - MS PRESSUR/C/A - 3.0/NEW/RO/NRC RO/TNT/DSS

The plant is stable in Mode 3 with RCS Tave at 557 degrees F following a reactor trip. Steam dumps are in automatic in the Steam Pressure Mode.

Suddenly, UI-500, steam dump demand indicates 100%.

Which **ONE** of the following **CORRECTLY** describes the system response and the reason ?

A. No change in steam dumps since this meter indicates demand based on Tave to Tref deviation.

B.✓ All 12 steam dumps fully open due to a high steam header pressure failure (PT-507).

C. The 3 cooldown valves fully open due to low failure of main steam header pressure (PT-507).

D. All 12 steam dumps fully open due to the low failure of turbine 1st stage pressure (PT-505)

35. 039A1.06 002/2/1/M STEAM - MS PRESSUR/C/A - 3.0/NEW/RO/NRC RO/TNT/DSS

K/A

039 Main and Reheat Steam System (MRSS)

A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including:

Main steam pressure

K/A MATCH ANALYSIS

Question gives plant status with plant being in Mode 3 when UI-507 suddenly reads high. Candidate must determine possible causes of the indication.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Indicates a high failure of PT-507 and all 12 dumps should open.

B. Correct. PT-507 failing high would give this demand indication and all 12 steam dumps should open.

C. Incorrect. Demand would not go to 100% on a low failure of PT-507.

D. Incorrect. Demand would not go to 100% on a low failure of PT-505 turbine impulse pressure while in steam pressure mode. This would be true for Tave mode.

REFERENCES

Vogtle Text Chapter 21 pages # 66 and # 67

VEGP learning objectives:

LO-PP-21201-04 Given that any of the following instrument/component inputs into the Steam Dump System has failed, determine how the system will respond and any required operator actions:

Initial conditions, Steam dumps in the "Steam Pressure Mode" Reactor Startup in progress (current power at 25%), waiting the roll the Main Turbine:

a. Main Steam Crosstie Pressure Instrument PT-507 fails HIGH

INFREQUENT OPERATIONS:

When a plant cooldown is to be completed, the Steam Dump Control System is normally utilized. With the system in the steam pressure mode of control, the header pressure controller is adjusted to control steam demand and thus, the cooldown rate. When reactor coolant average temperature reaches 550°F, the two interlock selector switches (Trains A and B) are momentarily placed in the BYPASS INTERLOCK position to allow the 3 cooldown valves to be utilized for further cooldown. Cooldown is then continued until the desired temperature is reached.

Steam Dump Instrumentation Failure response:

Steam Dumps in the **"TAVG Mode"** of operation:

1. **T_{avg} fails high;** The steam dumps demand meter will demand full open on the steam dump valves; however, with no arming signal the steam dump valves will remain closed. When the operators place the T_{avg} defeat switch to the failed loop the Steam Dump System will return to normal operation (minus the failed loop input).

2. **T_{avg} fails low;** The steam dump system will not be affected by this failure because the steam dump system uses auctioneered high T_{avg} for control functions. The Abnormal Operating Procedure (AOP) will still defeat the failed channel.

3. **Turbine Power (PT-505) fails high;** The steam dumps will remain closed with no demand on the demand meter, however, the steam dump system will not function in the "TAVG mode" with this failure on a loss of Main Turbine load. This will require the Steam dump system be placed in the "Steam Pressure Mode" of operation.

4. **Turbine Power (PT-505) fails low;** The steam dump valves will remain closed with full output demand on the demand meter (no arming signal present). If the steam dump were to be armed by loss of Main Turbine load (PT-506), in this condition all the steam dump valves would go open. This failure would have the operators place the Steam Dump System in the "Steam Pressure Mode" of operation.

5. **Turbine Power (PT-506) fails high;** The steam dump valves will remain closed. With PT-506 failed high the Steam Dump System will not function in the TAVG Mode on a loss of Main Turbine load due to no arming signal, so the Abnormal Operating

Procedure would have the operators place the Steam Dump System in the Steam Pressure Mode of operation.

6. **Turbine Power (PT-506) fails low;** The Steam Dump System will be armed but with no demand signal on the output demand meter the steam dump valves will remain closed. Note that if T_{avg} happened to be 2°F above T_{ref} the steam dumps valves will modulate open (how much would depend on the error signal). Abnormal Operating Procedure would have the operators place the Steam Dump System in the Steam Pressure Mode of operation.

7. **Reactor Trip with the Train "A" Reactor Trip Breaker failing to open;** The Train "A" Reactor Trip Breaker normally arms the Steam Dump System during a reactor trip, however, the Steam Dump System should still receive an arming signal due to the loss of Main Turbine load (PT-506). Therefore, the net affect would be that the Steam Dumps should function normally.

8. **Reactor Trip with the Train "B" Reactor Trip Breaker failing to open;** The Train "B" Reactor Trip Breaker normally swaps the steam dump controllers from the "Load Rejection Controller" to the "Plant Trip Controller", with this failure the swap will not take place. With the "Plant Trip Controller" still operating following the reactor trip the final primary T_{avg} will stabilize at 559°F rather than the normal 557°F (T_{ref} would be 557°F with the two degree deadband)

Steam Dumps in the **"Steam** Pressure **Mode"** of operation:

1. **Main Steam Crosstie Pressure instrument (PT-507) fails high;** The steam dump valves will go open the lower pressure with the steam pressure controller (PIC-507) in automatic control. The operators must place PIC-507 in manual control.

2. **Main Steam Crosstie Pressure instrument (PT-507) fails low;** The steam dump valves will go closed and not operate in automatic control. The operators must place PIC-507 in manual control.

Operating Note: Remember the Steam Dump System does **NOT** lower Primary T_{avg} but replaces the Main Turbine as the steam load on the reactor. The Rod Control System is what lowers primary temperature following a loss of Main Turbine Load (match reactor power with turbine power).

36. 040AK1.04 001/1/1/SLB - NIL DUCT TEMP/MEM - 3.2/BANK/RO/NRC RO/TNT/DSS

An unisolable steamline break has led to an excessive cooldown of the RCS and an orange path on RCS Integrity.

Which **ONE** of the following **CORRECTLY** describes the area of most concern for the propagation of an existing flaw in the RCS during this event ?

A. plastic deformation due to tensile stresses in the reactor vessel upper head area

B. brittle fracture due to compressive stresses in the reactor vessel hot leg area

C. plastic deformation due to compressive stresses in the reactor vessel cold leg area

D.✓ brittle fracture due to tensile stresses in the reactor vessel downcomer beltline area

36. 040AK1.04 001/1/1/SLB - NIL DUCT TEMP/MEM - 3.2/BANK/RO/NRC RO/TNT/DSS

K/A

040 Steam Line Rupture - Excessive Heat Transfer

AK1.04 Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture.

K/A MATCH ANALYSIS NDT

Question asks which part of the RCS would be the area of most concern during an excessive cooldown event and why.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Area of most concern is the reactor vessel beltine area. Plausible with all the focus on Rx. vessel head leaks lately in the industry.

B. Incorrect. Plausible due to focus on recent hot leak weld problem at V. C. Summer.

C. Incorrect. Plausible since CL temps are most referred to as the best indicator of cooldown in the beltline area.

D. Correct. Area of most concern per WOG background due to tensile stresses causing a possbile pre-existing flaw to propagate.

REFERENCES

WOG Background documents for PTS - pages # 1 and # 3.

VEGP learning objectives:

LO-LP-37071-03 State the difference between the following:

a. pressurized thermal shock event
b. overpressure event
c. cold overpressure event

1. INTRODUCTION

The Function Restoration Guideline (FRG) FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, provides guidance in the event of an unexpectedly severe RCS cooldown (i.e., pressurized thermal shock) or an unexpected overpressure condition at a low temperature during a controlled cooldown (i.e., cold overpressure). The operator is provided with instructions to attempt to prevent further cooldown and to minimize the pressure in the RCS in order to respond to a challenge to reactor vessel integrity. Guidance is also provided on any subsequent RCS cooldown restrictions required to safely achieve cold shutdown conditions. Refer to the document PRESSURIZED THERMAL SHOCK in the Generic Issues section of the Executive Volume for additional information on pressurized thermal shock and cold overpressure conditions.

This guideline is entered from three separate branches of the Integrity Status Tree, one having a RED priority signifying a potentially damaging thermal shock condition independent of RCS pressure, and two having an ORANGE priority signifying an imminent approach to the RED priority, depending on subsequent increases in pressure or decreases in temperature. Scenarios related to single or multiple failures involving LOCA, secondary breaks or steam generator tube rupture, may lead to a severe cooling of the entire RCS (e.g., secondary breaks or LOCA) or a local cooling of the cold leg/downcomer area (e.g., LOCA or steam generator tube rupture). Scenarios related to excessive charging (or safety injection) may lead to a system overpressure condition.

Instruction is provided in the Optimal Recovery Guidelines (ORGs) to address Integrity concerns, but if a cooldown of unexpected severity or overpressure still occurs, this FRG provides additional guidance to be used to address any challenges to the Integrity Critical Safety Function.

Guideline FR-P.1 is exited when all actions have been completed. At this time the operator is instructed to return to the

2. DESCRIPTION

An event or series of events which leads to a relatively rapid and severe reactor vessel downcomer cooldown can result in a thermal shock to the vessel wall that may lead to a small flaw, which may already exist in the vessel wall, growing into a larger crack. The growth or extension of such a flaw may lead, in some cases (where propagation is not stopped within the wall), to a loss of vessel integrity. The objective of Function Restoration Guideline FR-P.1 is to prevent the growth of a flaw and, in the event the limits set forth are exceeded, provide specific actions which appropriately restrict operation to prevent further challenges to vessel integrity.

Two separate types of events lead to entry into this guideline:

- Pressurized Thermal Shock Events

Several possible transients can be hypothesized which will produce rapid and extensive temperature decreases in the RCS cold leg(s) and, by inference, also the reactor vessel downcomer region. The rate and extent of cooldown determine whether entry into this guideline is on a RED or ORANGE priority. The actions in this guideline attempt to stop the cooldown, i.e., stabilize temperature, and also decrease RCS pressure to reduce the pressure stress component of total stress in the reactor vessel wall, partially offsetting the large thermal stress created by the rapid cooldown.

- Cold Overpressure Events

For this type of event, entry on an ORANGE priority is warranted if RCS pressure has exceeded the Cold Overpressure Protection Limit, and RCS cold leg temperature is sufficiently low that vessel ductility is reduced. There is little or no thermal stress associated with this event, so the benefit in using this guideline comes from the prompt RCS pressure reduction actions which supplement the Cold Overpressure Protection System.

37. 041A3.05 001/2/2/STM DUMP - MS PRESS/C/A - 2.9/NEW/RO/NRC RO/TNT/DSS

A failure of Turbine Impulse Pressure Instrument PT-506 at power required the BOP to place the steam dumps in Steam Pressure Mode with a setpoint to control no-load Tave at 557 degrees F.

Subsequently the reactor has tripped, steam dumps should be controlling pressure on PT-507 at approximately....................

A. 1060 psig and a setting of approximately 7.08 on the potentiometer.

B.✓ 1092 psig and a setting of approximately 7.28 on the potentiometer.

C. 1107 psig and a setting of approximately 7.73 on the potentiometer.

D. 1125 psig and a setting of approximately 7.48 on the potentionmeter.

37. 041A3.05 001/2/2/STM DUMP - MS PRESS/C/A - 2.9/NEW/RO/NRC RO/TNT/DSS

K/A

041 Steam Dump System (SDS) / Turbine Bypass Control

A3.05 Ability to monitor automatic operation of the SDS, including:

Main Steam Pressure

K/A MATCH ANALYSIS

Question gives plant status with plant being stabilized in Mode 3 following a reactor trip after the AOP for Failure of a Turbine Impulse Pressure instrument has them set to control Tave at 557 degrees. Candidate must determine correct pressure and pot setting for steam dumps.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. This is 100 lbs below 1160 setpoint in SGTR procedure and would be controlling temperature just over 550 degrees F.

B. Correct. 7.28 called for in AOP (.728 X 1500 = 1092 psig)

C. Incorrect. This is pot setting for SGTR of 7.73 for ARV and would control at 1160 psig. 1107 psig corresponds to psia setting for 557 but meter is psig for calc.

D. Incorrect. 7.48 would be setting for 1125 psig which is an ARV lift setpoint.

REFERENCES

AOP-18001-C, "Primary Systems Instrumentation Malfunction", section H for Failure of Turbine Impulse Pressure Instrumentation step H-3.

19030-C, Steam Generator Tube Rupture step 6.

VEGP learning objectives:

LO-PP-21201-04 Given that any of the following instrument/component inputs into the Steam Dump System has failed, determine how the system will respond and any required operator actions:

Initial conditions, Steam dumps in the "Steam Pressure Mode" Reactor Startup in progress (current power at 25%), waiting the roll the Main Turbine:

a. Main Steam Crosstie Pressure Instrument PT-507 fails HIGH

H. FAILURE OF TURBINE IMPULSE PRESSURE INSTRUMENTATION

SYMPTOMS

- TAVG/TREF DEVIATION.
- RCS LOOP TAVE/AUCT TAVG HI-LO DEV Annunciator.
- TURB PWR P13 CHI(II) PB-505(6)A status light - OFF.
- LO TURB IMP PRESS ROD STOP C5 status light - ON.
- LOSS OF TURB LOAD INTLK C7 status light - ON.
- Turbine impulse pressure indication mismatch.
- AMSAC TROUBLE annunciator.
- AMSAC LO FW FL AFW ACTUATION TURBINE TRIP.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
IMMEDIATE OPERATOR ACTIONS	
H1. Verify - NO ROD MOTION.	H1. Place rods in MANUAL.
SUBSEQUENT OPERATOR ACTIONS	
H2. Restore TAVG to program band.	
H3. Perform the following:	
a. Verify steam dumps controller at setpoint to maintain no-load Tavg of 557°F during normal operation (approximately 7.28).	
b. Place the steam dump control mode select switch in steam pressure mode of operation.	
H4. Within one hour, verify the P-7 and P-13 status lights indicate correctly for plant condition.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 5. **Identify ruptured SG(s) by any of the following conditions:** ☐ Unexpected rise in any SG NR level. ☐ High radiation from any SG sample. ☐ High radiation from any SG steamline. ☐ High radiation from any SG blowdown line.	☐ * 5. Continue with Steps 22 through 29. ☐ WHEN ruptured SG(s) identified, THEN perform Steps 6 through 21.

CAUTION: At least one SG should be maintained available for RCS cooldown.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 6. **Isolate ruptured SG ARV(s):**	
☐ a. Adjust ruptured SG ARV(s) controller setpoint to 1160 psig (pot setting 7.73).	Plausible distractor #
b. Check ruptured SG ARV(s) - CLOSED ☐ PV-3000 (SG 1) ☐ PV-3010 (SG 2) ☐ PV-3020 (SG 3) ☐ PV-3030 (SG 4)	☐ b. **WHEN ruptured SG(s) pressure less than 1160 psig, THEN verify SG ARV is closed.** IF SG ARV(s) can NOT be closed, THEN locally unlock and close associated SG ARV INLET isolation valve: ☐ 1301-U4-136 (SG 1) ☐ 1301-U4-137 (SG 2) ☐ 1301-U4-138 (SG 3) ☐ 1301-U4-139 (SG 4)

38. 045A4.01 001/2/2/TG - TURB VALVE IND/C/A - 3.1/NEW/RO/NRC RO/TNT/DSS

Following a reactor trip from power, the Unit SS has asked you to verify Turbine Trip during the performance of E-0, "Reactor Trip or Safety Injection".

The following is the status of the QMCB indications:

* Annunciator ALB13, window E02 for TURBINE STOP VLVS CLOSE / AUTO OIL is illuminated.
* All 4 Turbine Stop Valve Closed bistable lights illuminated on TSLB-2 - BOP panel.
* All 3 ETS Ch I, II, III Turbine Trip bistable lights illuminated on TSLB-2 - BOP panel.
* Stop valves # 1 and # 3 indicate 0% open on Main Turbine panel.
* Stop valves # 2 and # 4 indicate 35% and 45% open respectively on the Main Turbine panel.

Which **ONE** of the following would be **CORRECT** regarding the status of the Main Turbine and the next action(s) the BOP should take ?

A. TRIPPED, 4 of 4 stop valve closed bistable lights is preferred method to verify. Check AC Emergency Buses - At least one energized, 4160V AC1E bus.

B. NOT TRIPPED, all stop valves on main turbine panel do not indicate fully closed. Manually trip the Main Turbine, if turbine won't trip, isolate the Main Steam Lines.

C. TRIPPED, 3 of 3 ETS Ch I, II, III Trip bistable lights is preferred method to verify. Check AC Emergency Buses - At least one energized, 4160V AC 1E bus.

D✓ NOT TRIPPED, all stop valves on main turbine panel do no indicate fully closed. Manually trip the Main Turbine, if turbine won't trip, runback the Main Turbine.

38. 045A4.01 001/2/2/TG - TURB VALVE IND/C/A - 3.1/NEW/RO/NRC RO/TNT/DSS

K/A

045 Main Turbine Generator (MT/G) System

A4.01 Ability to manually operate and/or monitor in the control room:

Turbine valve indicators (throttle, governor, control, stop, intercept), alarms, and annunciators.

K/A MATCH ANALYSIS

Question gives a plausible scenario where a signal for a Turbine Trip has been generated but the operator has conflicting information on the QMCB. Candidate must decide which action would be correct. The ETS bistables I, II, and III indicate EHC pressure has dropped below setpoint indicating Turbine Trip required or called for. Main Turbine Stop valve indicator bistables light up if any < 96.7% open but not necessarily shut. Stop valve indication on turbine panel indicates valves still open. Operator should continue IOAs for Verify Turbine Trip RNO and next steps would be to manually trip the turbine and then, runback the main turbine. Isolate main steam lines comes after attempt at runback. Training has operators verify stop valve position on the main turbine panel.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Turbine trip not verified by this indication. Check AC buses is the next IOA to perform if you think turbine is tripped.

B. Incorrect. Turbine trip not verified, manually trip turbine and runback turbine are next IOAs. Isolate steam lines is not next in IOA sequence.

C. Incorrect. Turbine trip not verified by this indication. Check AC buses is the next IOA to perform if you think the turbine is tripped.

D. Correct. Turbine trip not verified, manually trip turbine and runback turbine are the next IOAs to perform.

REFERENCES

ARP-17013, window E02 for Turbine Stop Vlvs Close / Auto Oil.

19000-C, E-0 Reactor Trip or Safety Injection step # 2 and # 3 IOAs and RNOs.

VEGP learning objectives:

LO-LP-37011-06 State from memory the immediate action steps form 19000.

	(1)	(2)	(3)	(4)	(5)	(6)
A	STM GEN 1 FLOW MISMATCH		STM GEN 1 LO LEVEL	STM GEN 1 LO STEAMLINE PRESS ALERT	STM GEN 1 LO-LO LVL ALERT	STM GEN 1 HI/LO LVL DEVIATION
B	STM GEN 2 FLOW MISMATCH		STM GEN 2 LO LEVEL	STM GEN 2 LO STEAMLINE PRESS ALERT	STM GEN 2 LO-LO LVL ALERT	STM GEN 2 HI/LO LVL DEVIATION
C	STM GEN 3 FLOW MISMATCH		STM GEN 3 LO LEVEL	STM GEN 3 LO STEAMLINE PRESS ALERT	STM GEN 3 LO-LO LVL ALERT	STM GEN 3 HI/LO LVL DEVIATION
D	STM GEN 4 FLOW MISMATCH		STM GEN 4 LO LEVEL	STM LO STEAMLINE GEN 4 PRESS ALERT	STM GEN 4 LO-LO LVL ALERT	STM GEN 4 HI/LO LVL DEVIATION
E	AMSAC LO FW FL AFW ACTUATION TURBINE TRIP	TURBINE STOP VLVS CLOSE/ AUTO OIL	LO TAVG AND REACTOR TRIP FW VLVS CLOSE	FWI SI OR P-14 SG HI-HI LVL		
F	HI STM PRESS RATE (ANY LOOP)					

WINDOW E02

ORIGIN	SETPOINT	TURBINE STOP VLVS CLOSE/ AUTO OIL
1-PT-6161 1-PT-6162 1-PT-6163	580 psig	
SV Lim Sw	any valve less than 96.7% open	

1.0 PROBABLE CAUSE

1. Turbine Stop Valve testing.
2. Instrument malfunction.

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

1. If Turbine has tripped and reactor has tripped, go to 19000-C, "E-0, Reactor Trip Or Safety Injection".
2. If Turbine has tripped and reactor has not tripped, go to 18011-C, "Turbine Trip Below P-9".

4.0 SUBSEQUENT OPERATOR ACTIONS

1. Check TSLB-2 and determine which channel caused the alarm.
2. If Turbine Stop Valve testing is in progress, no further action is required.
3. If an instrument malfunction is indicated, initiate maintenance as required.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
IMMEDIATE OPERATOR ACTIONS	
1. Verify Reactor trip: ☐ • Rod Bottom Lights - LIT ☐ • Reactor Trip and Bypass Breakers - OPEN ☐ • Neutron Flux - LOWERING	☐ 1. Trip Reactor using both Reactor trip handswitches. ☐ IF Reactor NOT tripped, THEN go to 19211-C, FR-S.1 RESPONSE TO NUCLEAR POWER GENERATION/ATWT.
2. Verify Turbine trip: ☐ • All Turbine Stop Valves - CLOSED	☐ 2. Trip Turbine. ☐ IF Turbine will NOT trip, THEN run back Turbine. ☐ IF Turbine can NOT be run back, THEN close Main Steamline Isolation and Bypass Valves.
3. Verify power to AC Emergency Busses:	
a. AC Emergency Busses - AT LEAST ONE ENERGIZED: ☐ • 4160V AC 1E Busses	☐ a. Go to 19100-C, ECA-0.0 LOSS OF ALL AC POWER.
b. AC Emergency Busses - ALL ENERGIZED: ☐ • 4160V AC 1E Busses ☐ • 480V AC 1E Busses	☐ b. Try to restore power to de-energized AC Emergency Bus while continuing with Step 4.

39. 054AA2.04 001/1/1/LOSS MFW - AFW OPS/C/A - 4.2/NEW/R/NRC RO/TNT/DSS

While at full power the following occurs:

* Steam Generator Hi Hi level setpoint exceeded due to overfeed
* Reactor trips and is stable in Mode 3
* Main Feed Reg Valves (MFRVs) currently shut
* Motor Driven AFW (MDAFW) pumps are running
* ALB13, window E04 for FWI SI or P-14 SG HI-HI LVL is still present

Which **ONE** of the following **CORRECTLY** describes the response of the MDAFW pumps and the MFRVs ?

typo

A. MFRVs shut on SG Hi Hi level and can be re-opened by resetting feedwater isolation (FWI). MDAFW pumps started ~~when~~ on trip of both MFPTs when the Hi Hi level setpoint was exceeded.

B. MFRVs shut on P-4 coincident with Tavg < 564 and can be re-opened by resetting FWI. MDAFW pumps started on SG Lo Lo level after the reactor tripped.

C.✓ MFRVs shut on SG Hi Hi level and can be re-opened by resetting FWI after SG Hi Hi level clears. MDAFW pumps started on trip of both MFPTs when the Hi Hi level setpoint was exceeded.

D. MFRVs shut on P-4 coincident with Tavg < 564 and can be re-opened by resetting FWI after the SG Hi Hi level clears. MDAFW pumps started on SG Lo Lo Level after the reactor tripped.

39. 054AA2.04 001/1/1/LOSS MFW - AFW OPS/C/A - 4.2/NEW/R/NRC RO/TNT/DSS

K/A

054 Loss of Main Feedwater (MFW)

AA2.04 Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW):

Proper operation of AFW pumps and regulating valves

K/A MATCH ANALYSIS

Question gives a plausible scenario where a SG Hi Hi level setpoint (P-14) is exceeded at power. This would result in a FWI (seals in), Main Turbine and MFPT trips. The reactor would immediately trip on Turbine Trip giving a P-4 seal in immediately. The MDAFW pumps would immediatley auto start on the trip of both MFPTs versus post reactor trip when SG levels decrease below the Lo Lo level setpoint for actuation.

The FWI seal in would have to be broken by clearing the SG Hi Hi level setpoint, cycling the reactor trip breakers, then resetting the FWI.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible that candidate may not realize the Hi Hi level (P-14) seals in and would have to be cleared to reset FWI. MDAFW pump starts is correct.

B. Incorrect. Plausible candidate may not realize FWI occurred on SG Hi Hi level, P-4 with Lo Tave < 564 is what normally gives FWI post trip. MDAFW pumps would have started on Hi Hi level when both MFPTs tripped.

C. Correct. FWI seal in would have to be broken after SG Hi Hi level clears. MDAFW pumps would have started on trip of both MFPTs at Hi Hi level setpoint.

D. Incorrect. Plausible candidate may think FWI occurred on P-4 with Lo Tavg < 564 but may think seal in would still have to be broken. MDAFW pumps would have started on SG Hi Hi level when both MFPTs tripped..

REFERENCES

ARP-17013, window E04, for FWI SI OR P-14 SG HI-HI LVL

ARP-17015, window D03 for MFPT A TRIPPED

ARP-17016, window D01 for MFPT B TRIPPED

11886-1/2, "Recovery From ESF Actuations" section 4.8 for Recovery From Feedwater Isolation (FWI)

Tech Spec Bases for 3.3.2 ESFAS Instrumentation Funtions 5 and 6 for Turbine Trip and FWI on SG Hi Hi Level (P-14) and AFW Actuation on Trip of Both MFPTs.

VEGP learning objectives:

LO-PP-28103-02 List all permissives with applicable set points, coincidences, and functions.

WINDOW E04

ORIGIN	SETPOINT
SAFETY INJECTION	SSPS OUTPUT
1-LT-517/518/519/551 1-LT-527/528/529/552 1-LT-537/538/539/553 1-LT-547/548/549/554	8~~6~~3% - P14 OUTPUT

FWI
SI or P14
SG HI-HI LVL

1.0 PROBABLE CAUSE

Safety Injection, SGs overfed, or SG level intentionally high.

2.0 AUTOMATIC ACTIONS

Feedwater Isolation - Seal-in with P-4.

3.0 INITIAL OPERATOR ACTIONS

If an SI has occurred go to 19000-C, "E-0 Reactor Trip Or Safety Injection".

4.0 SUBSEQUENT OPERATOR ACTIONS

1. If equipment failure is indicated, initiate maintenance.
2. When the cause of the FWI has been corrected and it has been determined that the FWI actuated equipment is ready to return to service, initiate FWI recovery action per 11886-1, "Recovery From ESF Actuations".

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB159-1, 1X4DB159-3, 1X6AA02-237, PLS, Technical Specification LCO 3.3.2 (Table 3.3.2-1)

WINDOW D03

ORIGIN	SETPOINT	
1-PS-5346	75 psig	MFPT A TRIPPED

1.0 **PROBABLE CAUSE**

Main Feedwater Pump Turbine (MFPT) A tripped due to one or more of the following:

a. Manual trip using 1-HS-3169 on the QMCB,

b. Low oil pressure to MFPT Bearings,

c. Low oil pressure to MFP Bearings,

d. Low suction pressure to MFP,

e. Low vacuum in Condenser,

f. MFPT Steam Exhaust Valve has closed,

g. MFPT Thrust Bearing wear,

h. Steam Generator HI-HI level or Safety Injection actuation,

i. Manual operation of Local Trip Lever,

j. Turbine overspeed.

2.0 **AUTOMATIC ACTIONS**

None

3.0 **INITIAL OPERATOR ACTIONS**

Initiate 18016-C, "Condensate And Feedwater Malfunction."

4.0 **SUBSEQUENT OPERATOR ACTIONS**

None

5.0 **COMPENSATORY OPERATOR ACTIONS**

None

END OF SUB-PROCEDURE

REFERENCES: 1X3D-BC-N50F, 1X3D-BC-N50H, 1X5DN550-4, CX5DT1101-26C

WINDOW D01

ORIGIN	SETPOINT	
1-PS-5347	75 psig	MFPT B TRIPPED

1.0 **PROBABLE CAUSE**

Main Feedwater Pump Turbine (MFPT) B tripped due to one or more of the following:

a. Manual trip using 1-HS-3170 on QMCB,
b. Low oil pressure to MFPT Bearings,
c. Low oil pressure to Main Feedwater Pump (MFP) Bearings,
d. Low suction pressure to MFP,
e. Low vacuum in condenser,
f. MFPT Steam Exhaust Valve has closed,
g. MFPT Thrust Bearing wear,
h. Steam Generator Hi-Hi level or safety injection actuation,
i. Manual operation of local trip lever,
j. Turbine overspeed.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

Initiate 18016-C, "Condensate And Feedwater Malfunction".

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NONE

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X3D-BC-N51F, 1X3D-BC-N51H, CX5DT1101-26C

4.8 Recovery From Feedwater Isolation (FWI)

4.8.1 If a reactor trip has occurred, GO TO 19000-C, "E-0 Reactor Trip Or Safety Injection". ______

4.8.2 CHECK that the following conditions which could cause a FWI have cleared:

4.8.2.1 CHECK CNMT pressure - LESS THAN 3.8 PSIG (HI-1), ______

4.8.2.2 CHECK PRZR pressure - GREATER THAN 1970 PSIG, ______

4.8.2.3 CHECK all steamline pressures - GREATER THAN 585 PSIG, ______

4.8.3 If a Hi Hi SG level condition, CONTROL feedwater flow to restore SG level in the program band. ______

4.8.4 CHECK status of annunciator, FWI SI OR P-14 SG HI-HI LVL, ALB13 E04.

4.8.4.1 If ALB13 E04 is illuminated, PERFORM the following:

a. RESET SI if actuated. ______

NOTE

Cycling the Reactor Trip Breakers will generate a Regulatory Event for the IPC. If more than one hour has elapsed since the last event, data collection will start over; previous event history may be lost.

b. CYCLE RTBs. ______

c. RESET FW Isolation. ______

d. ENERGIZE 1NB01 and 1NB10 if required. ______

4.8.4.2 If ALB13 E04 is not illuminated, RESET FWI. ______

4.8.5 VERIFY the IPC is collecting desired data. If not, INITIATE data collection per 13505-1, "Integrated Plant Computer". ______

4.8.6 WHEN RCS pressure is less than P-11 setpoint, Then BLOCK SI signals. ______

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

(2) Steam Line Pressure — Negative Rate — High (continued)

the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless one MSIV and associated bypass valve in each steam line is closed. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have an SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

While the transmitters may experience elevated ambient temperatures due to an SLB, the trip function is based on rate of change, not the absolute accuracy of the indicated steam pressure. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

5. Turbine Trip and Feedwater Isolation

The primary functions of the Turbine Trip and Feedwater Isolation signals are to prevent damage to the turbine due to water in the steam lines, and to stop the excessive flow of feedwater into the SGs. These Functions are necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

This Function is actuated by SG Water Level — High High, or by an SI signal. The RTS also initiates a turbine trip signal whenever a reactor trip (P-4) is generated. In the event of SI, the unit is taken off line and the turbine generator must be tripped. The MFW System is also taken out of operation and the AFW System is automatically started. The SI signal was discussed previously.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Turbine Trip and Feedwater — Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b. Under specific conditions, a single inoperable actuation relay does not require that the affected automatic actuation logic function be declared inoperable. Specific guidance is provided in this section under the heading "Actuation Relays."

b. Feedwater Isolation — Low RCS T_{avg} Coincident with Reactor Trip

Since T_{avg} is used as an indication of bulk RCS temperature, this Function meets redundancy requirements with one OPERABLE channel in each loop. Thus, this function is specified as a total of four channels and not on a per loop basis. The channels are used in a two-out-of-four logic. The Low RCS T_{avg} signal is interlocked with P-4 to avert or reduce the continued cooldown of the RCS following a reactor trip. An excessive cooldown of the RCS following a reactor trip could cause an insertion of positive reactivity with a subsequent increase in generated power. The P-4 interlock is discussed in Function 8.a.

c. Turbine Trip and Feedwater Isolation — Steam Generator Water Level — High High (P-14)

LOOP 1	LOOP 2	LOOP 3	LOOP 4
LI-0517	LI-0527	LI-0537	LI-0547
LI-0518	LI-0528	LI-0538	LI-0548
LI-0519	LI-0529	LI-0539	LI-0549
LI-0551	LI-0552	LI-0553	LI-0554

NOTE: Steam Generator Water Level channels are required OPERABLE by the Post Accident Monitoring Technical Specification.

The setpoints for this Function on Table 3.3.2-1 are in % of narrow range instrument span.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

c. Turbine Trip and Feedwater Isolation — Steam Generator Water Level — High High (P-14) (continued)

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand both 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. Thus, four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic.

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause an adverse environment in containment. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

d. Turbine Trip and Feedwater Isolation — Safety Injection

Turbine Trip and Feedwater Isolation is also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

Turbine Trip and Feedwater Isolation Functions must be OPERABLE in MODES 1 and 2 except when one MFIV or MFRV and associated bypass valve per feedwater line are closed and deactivated or isolated by a closed manual valve when the MFW System is in operation and the turbine generator may be in operation. In MODES 3, 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Auxiliary Feedwater (continued)

remove decay heat or sufficient time is available to manually place either system in operation.

d. Auxiliary Feedwater-Trip Of All Main Feedwater Pumps

A Trip of all MFW pumps is an indication of a loss of MFW and the subsequent need for some method of decay heat and sensible heat removal to bring the reactor back to no load temperature and pressure. Each turbine driven MFW pump is equipped with a pressure switch on the control oil header. A low pressure signal from this pressure switch indicates a trip of that pump. A trip of all MFW pumps starts the motor driven AFW pumps to ensure that at least one SG is available with water to act as the heat sink for the reactor.

Function 6.d must be OPERABLE in MODES 1 and 2 when the MFW system is operating and supplying the SGs. This ensures that at least one SG is provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. In MODE 2, when the MFW system is not supplying the SGs, this function is not required as the AFW system is operating to supply the SGs and does not require the auto start from this function. In MODES 3, 4, and 5, the RCPs and MFW pumps may be normally shut down, and thus neither pump trip is indicative of a condition requiring automatic AFW initiation.

7. Semi-Automatic Switchover to Containment Sump

At the end of the injection phase of a LOCA, the RWST will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is switched to the containment recirculation sump. The low head residual heat removal (RHR) pumps and containment spray pumps draw the water from the containment recirculation sump, the RHR pumps pump the water through the RHR

(continued)

40. 055EK1.01 001/1/1/LOSS AC - BATT DISCH/MEM - 3.3/BANK/RO/NRC RO/TNT/DSS

Unit 2 has tripped and a Loss of All AC power event is in progress.

The battery chargers to the 125V DC buses have been de-energized for approximately one hour. It is estimated another 4 hours until AC power is restored.

* DC bus loads are consistent with design bases

In response to a prolonged loss of all AC power, operators can expect the 125V DC bus voltages to drop __________ ~~at first~~, then later drop ____________.

after the first five minutes

A. slowly, faster due to high load.

B. quickly, more slowly due to battery depletion

C. quickly, more slowly due to low load.

D. ✓ slowly; faster due to battery depletion.

40. 055EK1.01 001/1/1/LOSS AC - BATT DISCH/MEM - 3.3/BANK/RO/NRC RO/TNT/DSS

K/A

055 Loss of Offsite and Onsite Power (Station Blackout)

EK1.01 Knowledge of the operational implications of the following concepts as they apply to the station blackout:

Effect of battery discharge rates on capacity. voltage + capacity related

K/A MATCH ANALYSIS

Question gives a plausible during a loss of All AC Power (Station Blackout) where the 1E 125V DC buses are being supplied from the batteries only. The candidate must choose whether the discharge rates of the batteries will get faster or slower over the course of the event and why.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Rate of decay slow at first then increases due to battery depletion.

B. Incorrect. Rate of decay slow at first then increases due to battery depletion.

C. Incorrect. Rate of decay slow at first then increases due to battery depletion.

D. Correct. Rate of decay slow at first then increases due to battery depletion.

REFERENCES

North Anna June 2004 NRC RO examination question # 50. not used for same K/A

VEGP learning objectives:

Not applicable.

Unit 1 tripped from 100% power following a loss of several electrical busses. The following conditions exist:

- Operating crew has completed 1-ES-0.1, "Reactor Trip Response"
- All electrical busses have been restored with the exception of the 1J emergency bus
- Repairs to the bus are expected to take 6 hours
- The 1J bus has been dead for one hour
- All AC vital busses are being supplied from their respective inverters.

In response to these conditions, operators can expect the 1-III DC bus voltage to drop __________ at first, then later drop __________.

A.✓ slowly; faster due to cell reversal

B. quickly; more slowly due to cell reversal

C. slowly; faster due to high load

D. quickly; more slowly due to low load

A. Correct. The battery voltage will drop slowly at first. The longer the battery supplies the bus without a battery charger, the faster the battery voltage will drop. This is due to the individual battery voltages being affected by cell reversal on any weak, or weakening cells.

B. Incorrect. Both parts of answer are incorrect. Candidate may choose this answer based on the mistaken idea that the battery voltage will drop faster at the beginning and if he only remembers that cell reversal causes the rate of voltage drop to change.

C. Incorrect. The first part of the answer is correct. The candidate may choose this answer based on the mistaken idea that high load is what causes the battery voltage discharge rate to increase.

D. Incorrect. Both parts of this answer are incorrect. The candidate could choose this answer based on the mistaken assumption that the discharge rate will be faster at the beginning and the knowledge that some DC loads (such as turbine oil pumps) are removed from service when possible.

Goes with Q #40

41. 057AA1.04 001/1/1/LOSS VITAL -RWST/VCT/C/A - 3.5/NEW/RO/NRC RO/TNT/DSS

A loss of 120V AC vital bus 1AY1A at power resulted in a reactor trip and safety injection due to improper feedwater control by the operating crew.

Which **ONE** of the following would be **CORRECT** regarding the operation of the Train A charging pump suction valves after the safety injection ?

A. VCT outlet valve (112B) would automatically fully shut, then RWST suction (112D) would automatically open.

B.✓ VCT outlet valve (112B) would remain open, RWST suction (112D) would remain closed. The crew could reposition the valves if desired.

C. VCT outlet valve (112B) would remain open, RWST suction (112D) would remain closed. The crew would NOT be able to reposition the valves.

D. VCT outlet valve (112B) would remain open, RWST suction (112D) would automatically open. The crew should shut the VCT outlet valve.

III. LESSON OUTLINE: NOTES

12. Loss of Regulated Instrument Panel 1NYJ
13. Loss of Regulated Instrument Panel 1NYR
14. Loss of Regulated Instrument Panel 1NYS
15. Loss of Regulated Instrument Panel 1NYRS
16. Loss of Regulated Instrument Panel 1NY01

D. Present Lesson Objectives

II. PRESENTATION — LO-TP-60324-001

A. Loss of Vital Instrument Panel 1AY1A

1. Symptoms

   a. All Channel I trip status lights energized.

      1) Probably the most significant and easiest method of determining the loss of 1AY1A.

   b. Loss of N31, N35, N41 simultaneously.

   c. 1AY1A Trouble Alarm

   d. Inverter 1AD1I1 Trouble Alarm

2. Concerns and major effects of loss of 1AY1A — **Objective 1**

   a. SG 1 and 4 ARV will not operate from the control room or remote shutdown panel.

   b. May lose letdown and have max charging if LT-459 is selected for control.

   c. Tref (PT-505) will fail causing AUTO Rods to step in at 72 steps/min

   d. Steam Generator levels and steam flows fail causing significant transient on all the steam generators.

   e. If unable to stabilize plant conditions may require plant shutdown or trip. — **Objective 3**

   f. Plant will trip if below P-10 due to N35 trip signal when de-energized.

   g. General Warning on Train A SSPS. — **Objective 4**

      Without power from 1AY1A there is no power to Train A SSPS slave relays.

Any actuation signal requiring operation of slave relays will be blocked.

If SI signal present, only Train B equipment will Auto align.

Train A equipment must be manually re-aligned in this case.

h. Plant will trip if another channel instrument bistable is tripped and/or channel is in test.

3. Actions

Objective 2

a. Verify reactor power > P-10 otherwise verify Reactor trio and go to 19000-C.

b. Rods in MANUAL

1) Prevents rod motion due to loss of PT-505 (Tref)

2) Rods would step in to match Tref of 557°F

c. Control SG levels between 60 - 70%

1) Some or all of the controlling channels of steam flow, feed flow and SG levels will fail to zero.

2) Placing all FRV's in Manual may increase the difficulty of controlling the levels

3) Utilize the trend charts for each SG.

The selected control channels for each parameter is displayed on the chart.

If SG level, steam flow and feed flow are all above), then that FRV should be left in AUTO.

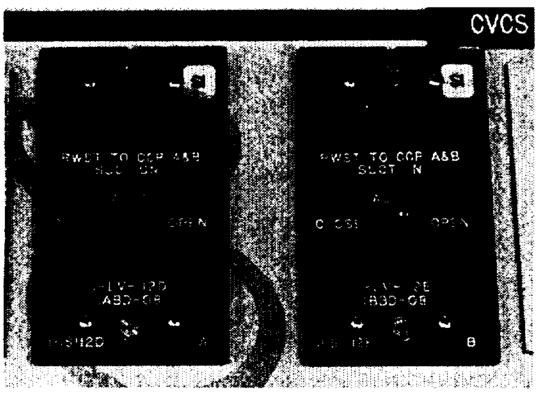
4) Place Main Feed Pump Speed controller in MANUAL since some or all selected steam flows have failed

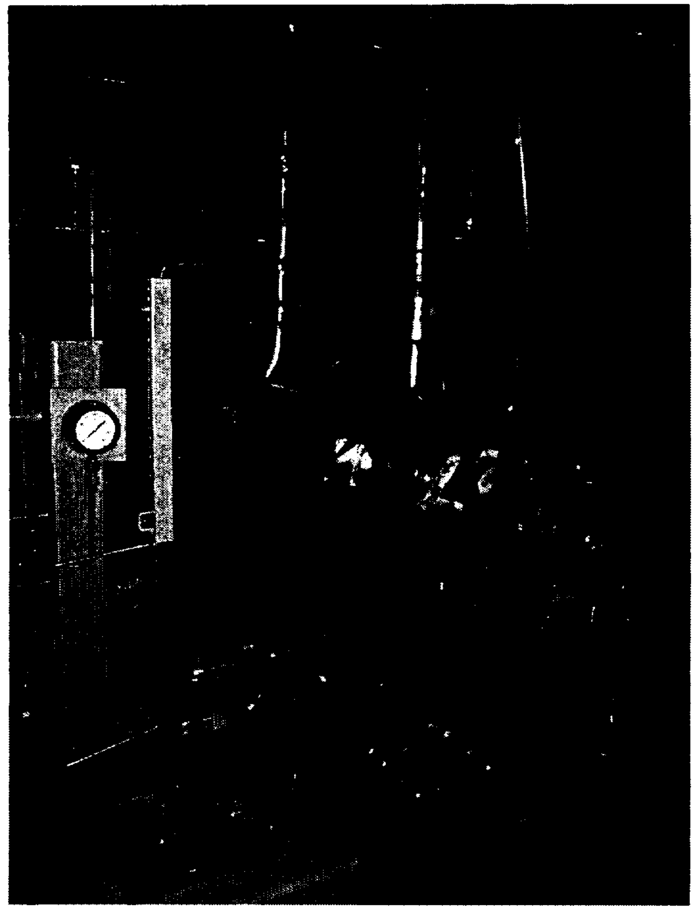
d. Position Feed Control selector switches to unaffected channels as listed in the procedure.

1) If FRV not in MANUAL and you select a different channel it will induce a

13.4 HIGH HEAD SAFETY INJECTION

The High Head system response to a safety injection signal contains the most component actuations than the intermediate and the low head injection system. The SI signal generates a Containment Isolation Actuation Signal (CIA) that isolates CVCS normal letdown and the seal return line from the RCPs. The main components in the CVCS system that are actuated are listed below.

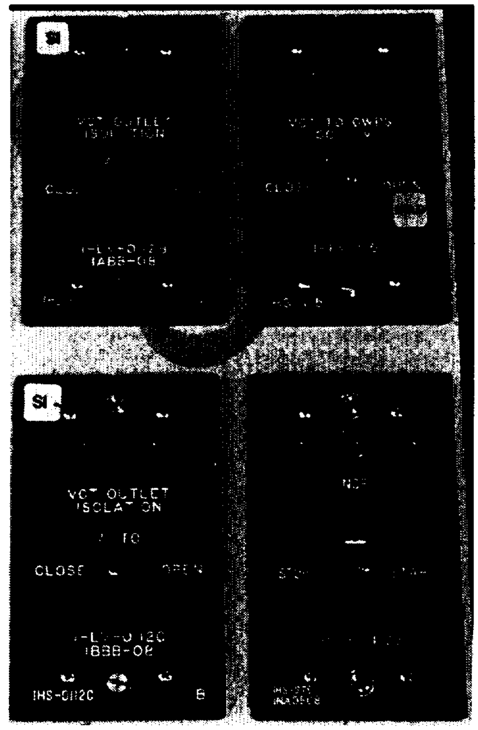




Component Action

1.	Centrifugal charging pumps Start	
2.	Refueling water storage tank (RWST) suction valves to charging pumps	Open
3.	Discharge header (parallel) valves for Unit 1 boron injection tank (BIT)	Open
4.	Normal charging path valves	Close
5.	Charging pump mini flow valves	Close
6.	Charging pump alternate mini flow valves	Enabled
7.	Volume control tank outlet isolation valves	Close
8.	Normal Charging Pump	Trips
9.	Safety Grade Charging Isolation Valve	Close

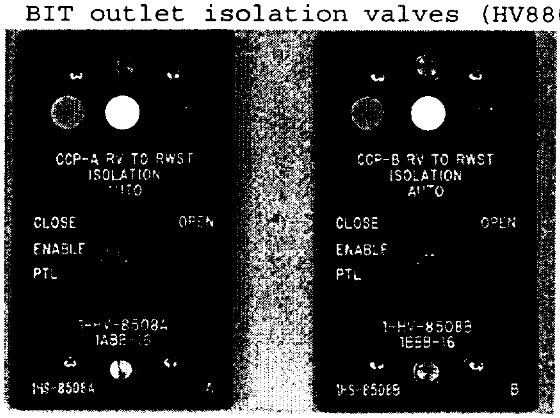
The simplified drawing on page 10 depicts the alignment of the CVCS system after receipt of a SI signal. As shown, the CCPs are taking suction from the RWST, through isolation valves, LV-112 D/E, which are interlocked with LV-112B/C, VCT outlet isolation valves. The interlock works in the following manner: when the respective train-related RWST valve is fully open, then the train-related VCT outlet valve shuts. If the RWST suction valve fails to open, then the train-related VCT valve remains open.



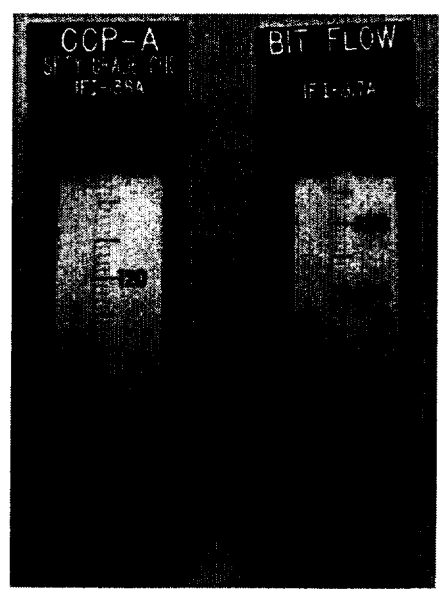
Notice the design feature in the valve arrangement. To ensure that the CCPs have a supply under SI conditions, the RWST outlet valves are in parallel. In the same fashion, to ensure positive isolation of the VCT from the suction flow path, the VCT outlet isolation valves are arranged in series.

Since the normal mini flow path for the Normal Charging pump is isolated, the NCP receives a trip signal. This is an item checked in the first few steps of the emergency operating procedure in response to a SI actuation.

As the table shows, other valves in the discharge line reposition to direct the CCP discharge flow to all four RCS cold legs. The normal charging header isolates (HV-8105 & HV-8106), as well as the Safety Grade Charging isolation valve HV-8116. Completing the discharge flow path are the BIT outlet isolation valves (HV8801A/B), which receive an open signal.

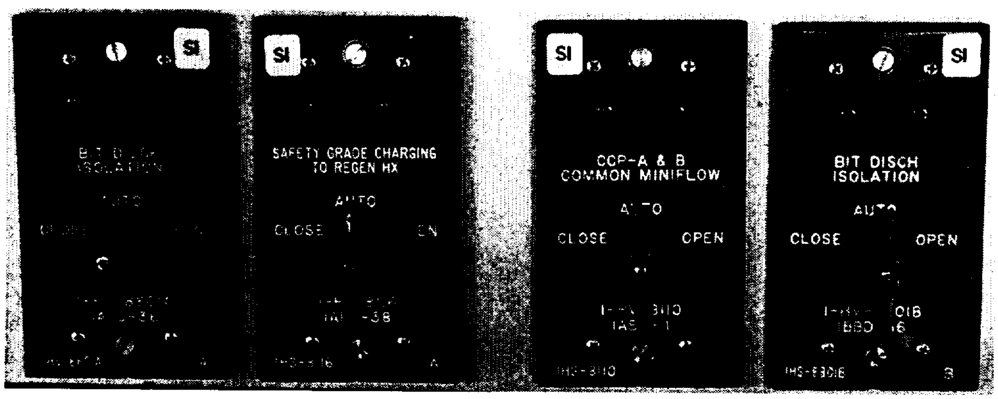


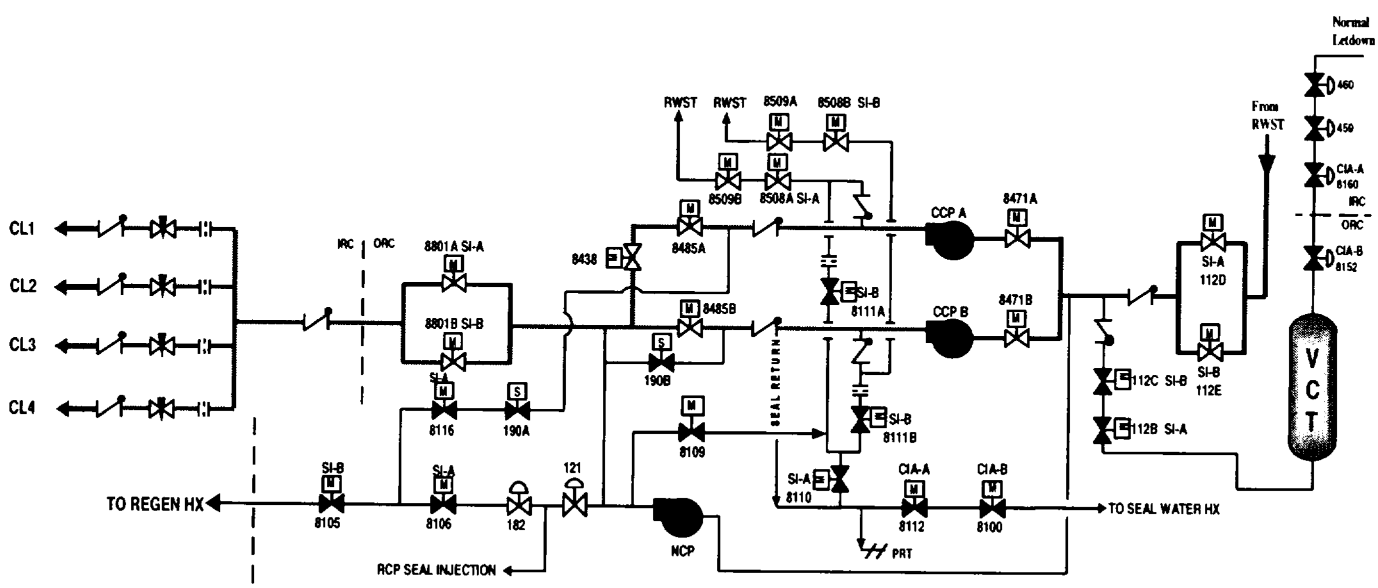
The CCP mini flow path is swapped from the VCT to the RWST. Normal mini flow valves HV-8111A/B and HV-8110 close. The new flow path is based on individual CCP discharge pressure switches. One valve in the flow path is normally open - HV-8509A/B. The upstream valve in each pump's alternate mini flow path (HV-8508A/B) receives a SI signal to be enabled in the pressure control mode (PCM). When pump discharge pressure reaches 1935 psig, the valve automatically opens, thus providing a mini flow path to the RWST. When pump discharge pressure lowers to 1805 psig, the valve closes. When the CCP alternate mini flow isolation valves are enabled, the center light on the hand switch illuminates. The other means of placing in the PCM is placing the HS in the "ENABLE PTL" position. After the SI signal is reset, the PCM can be disabled by taking the HS to the CLOSE or OPEN position.



Indication of high head injection flow can be verified by checking BIT flow indicator FI-0917A. Flow values will vary based on the drop in RCS pressure. From fundamentals training, the flow rate for a CCP at near shutoff head is very small as opposed to near runout conditions. The CCPs are rated for 150 gpm each at TDH of 2510.3 psid. The Pressurizer Safeties will open before the CCP's reach their shut off head of 2660 psig. So if there is a flow path available to the RCS, the CCPs should always be able to deliver flow.

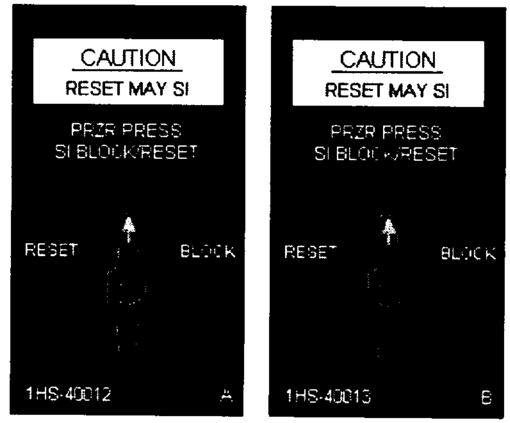
Indication of BIT flow would mean that one of the train-related BIT isolation valves was open. Unit 1 has a Boron Injection Tank with inlet valves that are de-energized in the OPEN position. Unit 2 does not have a BIT, but has the outlet isolation valves (HV-8801A/B) like Unit 1. See picture below of BIT outlet isolation valves. From these valves, high head injection line enters the containment penetration, at which point it separates into individual penetrations for each RCS cold leg.





This simplified drawing shows the final alignment of both trains of the high head safety injection system after receiving a SI signal.

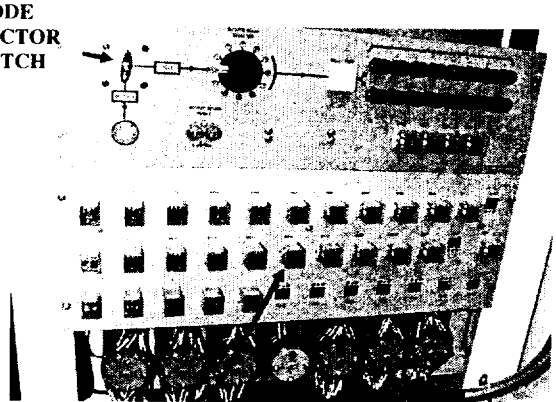
The switches shown are typical for all block and reset switches found on the main control board. Each hand switch will reset and block its associated train of SSPS only. This noted by the train designator in the bottom right hand corner.

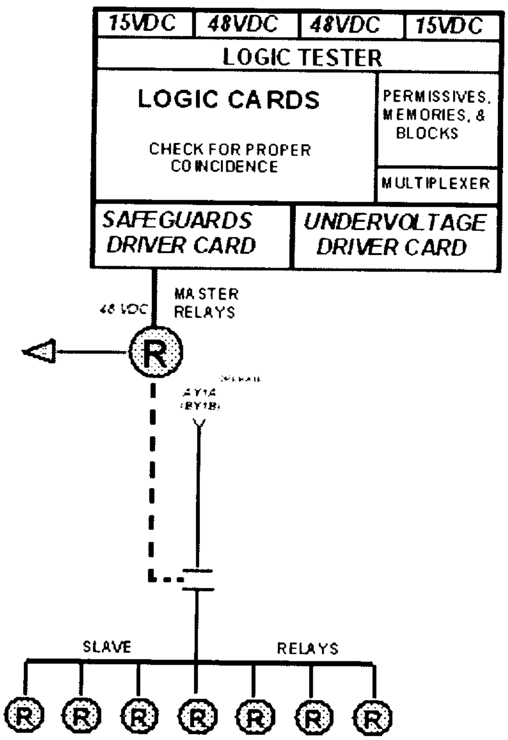


The multiplexer portion of the logic bay monitors the input and output of the logic cards and transfers the status of each parameter to a de-multiplexer for the main control board or computer. Identical information from both train "A" and "B" SSPS is transmitted over a common "OR" cable. The multiplexer illuminates the appropriate Trip Status Light Box (TSLB) when its parameter bi-stable trips. Also it indicates the status of all permissives and control interlocks on the Bypass Permissive Light Panel (BPLP).

28.18 OUTPUT RELAYS BAYS

There are two types of output relays; (1) Master relays, and (2) Slave relays. The master relays can be energized from either the Safeguard Driver Card from the Logic Bay or directly from the actuation switch located on the main control board. The master relays must be energized to actuate which require 48 VDC that comes from the logic bay. Each master relay can control up to 4 slave relays which are dedicated for a given actuation.





The slave relays are used to send actual start/stop signals to specific plant equipment when actuated. Slave relays require 120 VAC to actuate. Many are arranged with dual operating coils. The "latching coil" is used to actuate, which occurs when its master relay energizes. This type relay will remain in the latch position even when de-energized. The second coil called the "unlatch coil" energizes when the operator reset the actuation from the control board. If both coils are actuated at the same time the relay will remain in the Latch (actuate) position.

42. 058AA1.01 002/1/1/LOSS DC - XTIE BUS/MEM - 3.4/BANK/RO/NRC RO/TNT/RLM and

Which **ONE** of the following **CORRECTLY** describes how simultaneous feed of 120V AC bus 1AY2A powered from 1E 125V DC switchgear 1BD1 ~~or~~ the regulated transformer is prevented when ~~swapped~~ swapping to the alternate power supply ?

A. Mechanical interlock prevents simultaneous closing of both supply breakers.

B. Electrical interlock trips both supply breakers if they are simultaneously closed.

C. Alarm sounds in control room if both supply breakers are simultaneously closed.

D. Admin controls only (procedure) prevent simultaneous closing of both breakers.

42. 058AA1.01 002/1/1/LOSS DC - XTIE BUS/MEM - 3.4/BANK/RO/NRC RO/TNT/RLM

K/A

058 Loss of DC Power

AA1.01 Ability to operate and / or monitor the following as they apply to the Loss of DC Power:

Cross-tie of the affected dc bus with the alternate supply

K/A MATCH ANALYSIS

Question asks how simultaneous closure of supply breakers to 1AY2A is prevented.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. A mechanical interlock prevents closing in the normal supply and the alternate breaker simultaneously.

B. Incorrect. Mechanical interlock prevents.

C. Incorrect. Mechanical interlock prevents.

D. Incorrect. Mechanical interlock prevents.

REFERENCES

SOP-13431-1/2, "120V AC 1E Vital Instrument Distribution Panel" limitation 2.2.4.

Beaver Valley October 2004 RO NRC exam question # 52.

VEGP learning objectives:

Not applicable

2.2 **LIMITATIONS**

2.2.1 The 120V AC Vital Instrument Distribution Panels shall be energized in Modes 1, 2, 3, and 4 per Technical Specification LCO 3.8.9.

2.2.2 The 120V AC Vital Instrument Distribution Panels shall be energized in Modes 5 and 6 per Technical Specification LCO 3.8.10.

2.2.3 The Class 1E 120V inverters shall be OPERABLE in Modes 1 through 6 per Technical Specification LCO 3.8.7 and LCO 3.8.8.

2.2.4 **The 120V AC Vital Instrument Distribution Panel Supply Breakers are interlocked so that only one breaker at a time can be closed.**

2.2.5 The inverter DC input voltage shall be between 105 and 140V DC.

2.2.6 The inverter AC input voltage shall be between 414 and 506V AC.

2.2.7 While in Mode 1, 2, 3, or 4; no more than one 120V AC Vital Instrument Distribution Panel shall be powered from the regulated-transformer backup power supply at any one time during routine preventative maintenance on the associated inverter.

2.2.8 The doors to Control Building Rooms B26, B29, B31, and B36 should be opened if cooling is lost, and required actions per 00310-C "Standard For Use Of Doors" initiated.

3.0 **PREREQUISITES OR INITIAL CONDITIONS**

3.1 The 125V DC power is available to the inverters.

3.2 The 480V AC power is available to the regulated transformers.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	058 AA1.01	
Importance Rating	3.4	

Ability to operate and / or monitor the following as they apply to the Loss of DC Power: Cross-tie of the affected dc bus with the alternate supply

Proposed Question: Common 52

How is it ensured that loads powered off of two different 125VDC buses ("swing") loads do not inadvertently cross-tie the affected DC buses?

A. A mechanical interlock prevents simultaneous closure of both DC supply breakers.

B. An electrical interlock trips both DC supply breakers if they are simultaneously closed.

C. An alarm sounds in the control room to warn the operators if both DC supply breakers are simultaneously closed.

D. Administrative controls only (procedure requirements) prevent simultaneous closure of both DC supply breakers.

Proposed Answer: **A**

Explanation (Optional):

A. Correct. A physical barrier in the design of the distribution buses ensure loads cannot be cross-tied.

B. Incorrect for reason above.

C. Incorrect for reason above.

D. Incorrect for reason above.

Technical Reference(s): ______________ (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source: Bank # X

Modified Bank # ______ (Note changes or attach parent)

43. 059A2.04 001/2/1/MFW - HOT/DRY SG/C/A - 2.9/MODIFIED/R/NRC RO/TNT/DSS

A MFW malfunction has led to a reactor trip and the following sequence of events;

* The crew was been unable to establish feed to the S/Gs via any method.
* 19231, "Loss of Secondary Heat Sink" is in progress.
* RCS bleed and feed has been established.
* Feedwater capability is now restored.

Steam generator WR water levels are as follows:

* SG # 1 - 8.2%
* SG # 2 - 8.6%
* SG # 3 - 7.8%
* SG # 4 - 8.9%

The Reactor Operator reports that CETCs are rising.

Which **ONE** of the following is **CORRECT** regarding establishing feedwater flow to the SGs under the current conditions ?

A. The SGs are Non-dry, feed selected SG(s) at any flow rate.

B.✓ The SGs are Dry, feed one selected SG at any flow rate.

C. The SGs are Non-dry, feed selected SG(s) at 30-100 gpm.

D. The SGs are Dry, feed one selected SG at 30-100 gpm.

43. 059A2.04 001/2/1/MFW - HOT/DRY SG/C/A - 2.9/MODIFIED/R/NRC RO/TNT/DSS

K/A

059 Main Feedwaer (MFW) System

A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

Feeding a dry SG

K/A MATCH ANALYSIS

Question gives various combinations of SG levels and CETC parameters and candidate must determine which SG to feed and the rate. SG will be hot / dry and with CETCs rising it can be fed at any rate to protect the core with CETCs rising and bleed and feed established.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. All SGs below adverse containment parameter of 31% WR.

B. Correct. All SGs considered dry, one SG to be fed at any rate with CETCs rising.

C. Incorrect. All SGs below adverse containment parameter of 31% WR.

D. Incorrect. 30-100 gpm is proper rate UNLESS CETCs rising with feed and bleed.

REFERENCES

FRP-19231-C, "Response to Loss of Secondary Heat Sink"

* steps 49a thru f

* steps 53a thru f

* steps 57a thru 61b

* steps 64 thru 65c.

Modified from Vogtle HL-13 audit SRO retake question # 6 to correlate with new procedure flow path that changed on 12-17-2005 after the audit and to make RO level.

WOG background for FR-H.1, "Loss of Secondary Heat Sink" pages 65 thru 67

VEGP learning objectives:

steam generator following recovery from a loss of heat sink accident.

6. HL-AW-37000-00 016

RCS bleed and feed has NOT been initiated per 19231-C (FR-H.1), "Response to Loss of Secondary Heat Sink".

* AFW capability to SG # 2 only, is finally restored.

* SGs # 2 and # 4 WR levels < 9%

* All four RCS hot legs are > 550 F and rising

* SGs # 1 and # 3 levels are both rapidly lowering toward 29% WR.

The strategy the crew should use to re-establish feed under these conditions would be to........

A. Feed the SG at maximum rate to ensure RCS subcooling is maintained.

B✓ Feed the SG with no limit on flow since feed and bleed appears to be imminent.

C. Feed the SG at a rate of 30-100 gpm, to prevent excessive cooldown of the RCS.

D. Feed the SG at a rate of 30-100 gpm, to limit the possibility of a SGTR to this SG.

7. HL-AW-37000-00 017

19100-C, ECA-0.0 "Loss of All AC Power", has been implemented following a Loss of All AC power on Unit 2. The crew is presently depressurizing all intact Steam Generators using local operation of the SG ARVs.

Which of the following would be TRUE concerning rapid depressurization of the intact SGs during performance of 19100-C ?

A. If PRZR level is lost or RVLIS shows upper head voiding occurring, depressurization should be immediately halted.

B✓ The SGs should be depressurized at maximum rate (within capacity of the TDAFW pump) to minimize RCS inventory loss.

C. RCS subcooling is NOT expected to be lost during this evolution. If subcooling falls to less than 24 degrees F, depressurization should immediately be halted.

D. To prevent a PTS concern, depressurization may continue until RCS WR Cold Leg temperatures reach 200 degrees F at which time it should be immediately halted.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
CAUTION: Feed flow rates should be controlled to prevent excessive RCS cooldown.	
***49. Try to establish MDAFW flow to at least one SG:**	
a. Check MDAFW Pump - AVAILABLE: ☐ • Power available ☐ • Suction pressure ☐ • Discharge pressure	a. Perform the following: ☐ • Initiate actions to restore a MDAFW Pump. ☐ • WHEN MDAFW Pump is started, THEN go to Step 49b. ☐ • Go to Step 53.
b. Select SG(s) to feed: ☐ 1) All SG WR levels - LESS THAN 9% (31% ADVERSE)	b. Perform the following: ☐ • Restore feed flow to Non-Dry SG(s) by going to Step 50.
☐ c. Check Core Exit TCs - STABLE OR LOWERING	☐ c. Do NOT limit feed flow to the selected SG if Core Exit TCs are rising and go to step 49f.
☐ d. Restore feed flow to selected SG - BETWEEN 30 GPM AND 100 GPM	
☐ e. Check Dry SG WR level - GREATER THAN 9% (31% ADVERSE)	☐ e. WHEN Dry SG WR level is greater than 9% (31% ADVERSE) THEN raise feed flow to restore NR level greater than 10% (32% ADVERSE). ☐ Go to Step 70.
☐ f. Raise feed flow to restore NR level greater than 10% (32% ADVERSE) and go to Step 70.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Feed flow rates should be controlled to prevent excessive RCS cooldown.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***53. Try to establish TDAFW flow to at least one SG:**	
a. Check TDAFW Pump - AVAILABLE:	a. Perform the following:
☐ • Steam admission valve HV-5106 - OPEN ☐ • Trip & Throttle valve PV-15129 - OPEN (HS-15111) ☐ • Governor valve SV-15133 - OPERATING PROPERLY (PDIC-5180A)	☐ • Initiate 13610, AUXILIARY FEEDWATER SYSTEM to operate TDAFW Pump as necessary. ☐ • WHEN TDAFW Pump is started, THEN go to Step 53b. ☐ • Go to Step 57.
b. Select SG(s) to feed: ☐ 1) **All SG WR levels - LESS THAN 9% (31% ADVERSE)**	b. Perform the following: ☐ • Restore feed flow to Non-Dry SG(s) by going to Step 54.
☐ c. Check Core Exit TCs - STABLE OR LOWERING	☐ c. Do NOT limit feed flow to the selected SG if Core Exit TCs are rising and go to 53f.
☐ d. Restore feed flow to selected SG - BETWEEN 30 GPM AND 100 GPM	
☐ e. Check Dry SG WR level - GREATER THAN 9% (31% ADVERSE)	☐ e. WHEN Dry SG WR level is greater than 9% (31% ADVERSE) THEN raise feed flow to restore NR level greater than 10% (32% ADVERSE). ☐ Go to Step 70.
☐ f. Raise feed flow to restore NR level greater than 10% (32% ADVERSE) and go to Step 70.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
54. Verify TDAFW pump throttle valves open for selected SG(s): ☐ HV-5122 TDAFW Pump to SG 1 ☐ HV-5125 TDAFW Pump to SG 2 ☐ HV-5127 TDAFW Pump to SG 3 ☐ HV-5120 TDAFW Pump to SG 4	
☐ 55. Verify adequate feed flow to raise SG levels.	55. IF feed flow to at least one SG verified, THEN perform the following: ☐ a. Maintain flow to restore NR level to greater than 10% [32% ADVERSE]. ☐ b. Go to Step 70. ☐ IF feed flow to at least one SG can NOT be verified, THEN go to Step 57.
☐ 56. Go to Step 70.	
***57. Try to establish main FW flow to at least one SG:**	
☐ a. Check condensate system - IN SERVICE	☐ a. Place condensate system in service by initiating 13615, CONDENSATE AND FEEDWATER SYSTEM. ☐ WHEN Condensate system in service, THEN go to Step 58. ☐ Return to Step 49.
58. Verify the following: ☐ • MFRVs CLOSED AND CONTROLLERS AT 0% DEMAND IN MANUAL ☐ • BFRVs CLOSED AND CONTROLLERS AT 0% DEMAND IN MANUAL	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

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

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
59. Reset/bypass SI signals as necessary:	
☐ a. Check CNMT pressure - LESS THAN 3.8 PSIG	☐ a. Bypass the CNMT HI-1 pressure inputs in 2 of 3 NSSS protection channels (Channels 2, 3, 4, Bistables PB-936B, PB-935B, PB-934B) by initiating 13509-C, BYPASS TEST INSTRUMENTATION (BTI) PANEL OPERATION.
☐ b. Check PRZR pressure - GREATER THAN 2000 PSIG	b. Block SI signals: ☐ • Low Steamline Pressure SI ☐ • Low PRZR Pressure SI ☐ • Go to Step 59d.
☐ c. Check SG pressures - GREATER THAN 585 PSIG	c. Bypass the SG LOW PRESS inputs for 2 channels of any depressurized SG by initiating 13509-C, BYPASS TEST INSTRUMENTATION (BTI) PANEL OPERATION: ☐ SG1:PB514A, PB515A, PB516A ☐ SG2:PB524A, PB525A, PB526A ☐ SG3:PB534A, PB535A, PB536A ☐ SG4:PB544A, PB545A, PB546A
d. Perform the following: ☐ 1) Reset SI. ☐ 2) Close RTBs. ☐ 3) Reset FW Isolation. ☐ 4) Energize Stub Busses.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
60. Start one MFP:	☐ 60. Go to Step 64.
a. Check MFP - AVAILABLE	
☐ • Main Steam to MFP(s) - AVAILABLE	
☐ • Condenser vacuum established.	
b. Select one MFP and perform the following:	
☐ 1) Lower GE pot setting to zero.	
☐ 2) Reset MFP.	
☐ 3) Open MFP discharge valve.	
☐ 4) Slowly raise MFP speed using GE pot as necessary.	
☐ 5) Maintain MFP differential pressure approximately 50 psid.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
61. Select SG(s) to feed:	
☐ a. All SG WR levels - LESS THAN 9% (31% ADVERSE)	☐ a. Restore feed flow to Non-Dry SG(s) by going to Step 62.
☐ b. Check Core Exit TCs - STABLE OR LOWERING	☐ b. Do NOT limit feed flow to the selected SG if Core Exit TCs are rising.
☐ c. Open BFIV for selected SG.	☐ c. Open MFIV for the selected SG. ☐ IF MFIV will NOT open, THEN dispatch an operator to locally open the BFIV. ☐ IF neither BFIV or MFIV will open, THEN return to Step 49.
☐ d. Slowly open BFRV for the selected SG to establish feed flow - BETWEEN 30 GPM AND 100 GPM	☐ d. Open MFRV for the selected SG. ☐ IF MFRV will NOT open, THEN dispatch an operator to locally open the BFRV. ☐ IF neither BFRV or MFRV will open, THEN return to Step 49.
☐ e. Check Dry SG WR level - GREATER THAN 9% (31% ADVERSE)	☐ e. WHEN Dry SG WR level is greater than 9% (31% ADVERSE) THEN raise feed flow to restore NR level greater than 10% (32% ADVERSE). ☐ Go to Step 70.
☐ f. Raise feed flow to restore NR level greater than 10% (32% ADVERSE) and go to Step 70.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Feed flow rates should be controlled to prevent excessive RCS cooldown.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***64. Try to establish feed flow from the condensate system to one SG:**	
a. Select SG(s) to feed:	a. Perform the following:
☐ • All SG WR levels - LESS THAN 9% (31% ADVERSE)	☐ • Restore feed flow to Non-Dry SG by going to Step 66.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
65. Establish condensate flow to Dry SG:	
a. Depressurize selected SG to less than 550 psig:	☐ a. Actuate Main Steamline Isolation.
☐ 1) Check MSIVs and BSIVs - ANY OPEN	☐ Dump steam using selected SG ARV.
☐ 2) Close all MSIVs and BSIVs except on selected SG.	☐ IF unable to dump steam, THEN return to Step 49.
☐ 3) Depressurize selected SG using Steam Dumps.	
☐ b. Open Main Feed Pump discharge valves.	☐ b. IF discharge valves can NOT be opened, THEN locally open MFP bypass valve 1305-U4-655. (TB-Lvl 2)
☐ c. Check Core Exit TCs - STABLE OR LOWERING	☐ c. Do NOT limit feed flow to the selected SG if Core Exit TCs are rising.
☐ d. Open BFIV for selected SG.	☐ d. Open MFIV for the selected SG. ☐ IF MFIV will NOT open, THEN dispatch an operator to locally open the BFIV. ☐ IF neither BFIV or MFIV will open, THEN return to Step 49.
☐ e. Slowly open BFRV for the selected SG to establish feed flow - BETWEEN 30 GPM AND 100 GPM	☐ e. Open MFRV for the selected SG. ☐ IF MFRV will NOT open, THEN dispatch an operator to locally open the BFRV. ☐ IF neither BFRV or MFRV will open, THEN return to Step 49.
☐ f. Check Dry SG WR level - GREATER THAN 9% (31% ADVERSE)	☐ f. WHEN Dry SG WR level is greater than 9% (31% ADVERSE) THEN raise feed flow to restore NR level greater than 10% (32% ADVERSE). ☐ Go to Step 70.
☐ g. Raise feed flow to restore NR level greater than 10% (32% ADVERSE) and go to Step 70.	

system would begin to boil after a short period of time (see Period 6, subsection 2.1). Once boiling began, depressurization of the RCS using PORVs without having core uncovery would be highly unlikely. Core uncovery would be necessary to reduce the steam generation rate to a rate that permitted RCS depressurization using pressurizer PORVs. Thus, the use of feed and bleed precludes the use of bleed and feed without core uncovery and possible core damage. Therefore, based on the above arguments, feed and bleed is not recommended to provide an alternative heat removal method during a loss of secondary heat sink condition.

2.4 Feeding a Dry Steam Generator

If bleed and feed has been initiated, during restoration of secondary heat sink, feeding a dry steam generator may be necessary. If the event was initiated from high temperature and high decay heat conditions it is likely that feedwater flow will have to be established to a hot, dry steam generator. A hot, dry steam generator is defined as a steam generator in which the primary side of the steam generator is above 550°F* and the secondary side has no liquid inventory. Reestablishment of feedwater is the more desirable mode of recovery from a loss of secondary heat sink than remaining on bleed and feed and establishing cold leg recirculation for long term cooling because this will be more likely to avoid core uncovery. However, care must be taken when re-establishing feedwater flow to minimize the effects of thermal shock consistent with the urgency of the need to restore the secondary side heat sink.

Since the heat removal capability of one steam generator is always greater than decay heat, it is advisable to reestablish feedwater to only one steam generator regardless of the size of the plant or number of loops. Thus, if a failure in an SG occurs due to excessive thermal stresses, the failure is isolated to one steam generator.

* 550°F is a temperature evaluated to be low enough that thermal stress would
not lead to a failure when feedwater is established to any remaining dry
steam generator.

If bleed and feed has been initiated and RCS temperature is increasing, the re-establishment of feedwater flow should be limited to one steam generator and the flow rate used should be as high as can be made available due to the urgency of the situation. If RCS temperatures are stable or decreasing when feedwater flow is restored the flow should be directed to one steam generator and the rate should be limited to the plant-specific equivalent of 25 - 100 gpm until wide range level is established. With stable or decreasing RCS temperatures, the feedwater flow rate is limited to minimize the potential impact of excessive thermal stresses since a direct measure of the steam generator temperature is not available. Once an indicated wide range level is achieved in the affected steam generator, feedwater flow can be adjusted as necessary to restore level into the narrow range and thereby satisfying the requirements for a secondary heat sink.

Once feedwater is established, the feeding process should continue until the RCS temperature indications are decreasing. At that time the active steam generator should be checked for symptoms indicating a faulted or ruptured condition. If the active steam generator is faulted or ruptured, then feedwater should be established to another intact steam generator. If an intact steam generator does not exist, then a decision should be made to use the best available steam generator, which may be the active steam generator. Once the heat load has been transferred to a backup steam generator, the original steam generator should be isolated to prevent further radiation releases.

Thus, the process of initiating feedwater to a dry steam generator, as described here, is one that accounts for the fact that the steam generator temperature may be above 550 F. The number of steam generators that may be fed in a hot, dry condition are limited and if RCS temperature is decreasing the flow rate is also limited so as to limit the thermal shock to the steam generator being fed. Subsequent to securing SI and exiting FR-H.1 the remaining dry steam generators may have their levels recovered at the direction of the plant engineering staff in a manner that will minimize thermal shock to the steam generators.

44. 059AK2.02 001/1/2/ACC LIQ - GAS MNTR/C/A - 2.7/NEW/RO/NRC RO/TNT/DSS

A transfer of water to fill the Spent Fuel Pool Transfer canals was in progress when the Reactor Operator diverted CVCS letdown.

High airborne radiation alarms are received in the Spent Fuel Area.

Which **ONE** of the following **CORRECTLY** identifies an evolution which ~~could have resulted~~ would result in the airborne radiation alarms ?

A. RWST was being used to fill the transfer canal, RWST fill and letdown divert lines share a common path.

B. CVCS letdown divert and Waste Monitor Tank # 9 (WMT) to transfer canal fill line share a common path, letdown was diverted into the SFP transfer canal.

C. RMWST was being used to fill the transfer canal, RMWST fill and letdown divert lines share a common path.

D. In service RHUT was being used to fill the transfer canal, CVCS letdown went into the in service RHUT then into the SFP area.

44. 059AK2.02 001/1/2/ACC LIQ - GAS MNTR/C/A - 2.7/NEW/RO/NRC RO/TNT/DSS

K/A

059 Accidental liquid release radwaste release.

AK2.02 Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following:

Radioactive gas - monitors

K/A MATCH ANALYSIS

Question asks plausible scenario where fill of the SFP Transfer canals is in progress and radiation monitors in SFP go into alarm. Candidate must diagnose which condition would have resulted in the event. Question based on a couple of events in past. Procedure now has only Out Of Service RHUT transferred to SFP Transfer canal and letdown divert HS caution tagged in on QMCB.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. RWST not used to make up to transfer canals. Only BRS system. RWST makeup goes directly into discharge of Spent Fuel Pool Cooling System.

B. Incorrect. WMT # 9 is not used to fill the transfer canal.

C. Incorrect. RMWST not used to make up to transfer canals. Only BRS system. RMWST makeup goes directly into discharge of Spent Fuel Pool Cooling System.

D. Correct. Based on past plant event, diverting letdown to inservice RHUT during fill of transfer canal will cause gas problems in the SFP area..

REFERENCES

SOP-13703-C, "Boron Recycle System" section 4.4.14, "Transfering a Recycle Holdup Tank To A Spent Fuel Pit Transfer Canal".

VEGP learning objectives:

LO-PP-25102-11 Describe when the different sources of makeup to the spent fuel pool would be used.

a. For evaporation
b. For leakage

LO-LP-44101-02 Given a diagram of the BRS, show the flowpath through the following components:

j. System Interfaces

4.4.14 Transferring A Recycle Holdup Tank To A Spent Fuel Pit Transfer Canal

CAUTION

> Transfer only an OUT OF SERVICE Recycle Holdup Tank to the selected Spent Fuel Pit Transfer Canal.

4.4.14.1 If applicable, place the Recycle Holdup Tank that will NOT be transferred in service by performing one of the following:

a. To place RHT 001 in service, ensure open BRS RHT 001 INLET FROM RECYCLE EVAP FD FLTRS, A-1210-U6-069 (RC-83,RR) and ensure closed BRS RHT 002 INLET FROM RECYCLE EVAP FILTERS, A-1210-U6-070, (RC-49,RR)

b. To place RHT 002 in service, ensure open BRS RHT 002 INLET FROM RECYCLE EVAP FILTERS, A-1210-U6-070 (RC-49,RR) and ensure closed BRS RHT 001 INLET FROM RECYCLE EVAP FD FLTRS, A-1210-U6-069. (RC-83,RR)

4.4.14.2 If there is no available RHT to place in service, then place Letdown Divert 1-HS-0112A and 2-HS-0112A to VCT and install Caution Tags.

4.4.14.3 If there is no available RHT to place in service and it becomes necessary to divert CVCS to the RHT, perform the following prior to diverting:

a. Stop the running RECYCLE EVAPORATOR FEED PUMP #1 or #2 using Handswitch A-HS-270A or A-HS-270B,

b. Open BRS RHT 001 INLET FROM RECYCLE EVAP FD FLTRS, A-1210-U6-069 (RC-83,RR) or BRS RHT 002 INLET FROM RECYCLE EVAP FILTERS, A-1210-U6-070, (RC-49,RR)

c. Close BRS RECYCLE EVAP FEED PUMP DISCH HDR ISOL A-1210-U6-061, (RD-42,RR)

d. Realign the RHT per Step 4.4.14.9c or 4.4.14.10c.

4.4.14.4 Place the Recycle Holdup Tank to be transferred on recirculation per Section 4.2.1.

4.4.14.5 Notify Chemistry to obtain a sample and analyze it for compatibility with the Spent Fuel Pool water, into which the transfer will be made.

4.4.14.6 Ensure the following;

a. Sample analysis is complete and chemistry has determined water quality is compatible with Spent Fuel Pool water.

b. Shift Supervisor's permission has been obtained.

4.4.14.7 Notify Health Physics of impending transfer.

4.4.14.8 Align the Boron Recycle System RHT for transfer to a Spent Fuel Pit Transfer Canal by performing the following:

a. Close WPSL BRS EVAP XCONN FROM W EVAPS 1&2, A-1901-U4-375, (RC-63)

b. Close BRS EVAP FD PMP DISCH TELL TALE, A-1210-U4-064, (RD-42,RR)

c. Open BRS RECYCLE EVAP FEED PMPS DISCH HDR ISO, A-1210-U6-061, (RD-42,RR)

d. Open BRS RECYCLE EVAP FEED PMPS ALT RECIRC ISO, A-1210-U6-062, (RD-42,RR)

e. Open BRS RECYCLE EVAP FEED PMPS DISCH TO SFPCPS CANALS ISO, A-1210-U6-063. (RD-42,RR)

4.4.14.9 Establish communications with the operator monitoring Spent Fuel Pit Transfer Canal level.

45. 061K1.10 001/2/1/AFW - DG FUEL/MEM - 2.6/NEW/RO/NRC RO/TNT/DSS

A control room fire has resulted in an LOSP and evacuation to the remote shutdown panels. The crew has taken local control of the plant.

* TDAFW pump OOC

Which **ONE** of the following is **CORRECT** regarding local operation during this event ?

A. DG1B has local remote transfer switches for DG Fuel Oil Transfer pump control. MDAFW 1B has local remote switches to control its motor operated mini-flow valve.

B. DG2B has local remote transfer switches for DG Fuel Oil Transfer pump control. MDAFW 2B does **NOT** have local control switches for it's mini-flow valve. The valve cycles on discharge flow.

C.✓ DG2B has local remote transfer switches for DG Fuel Oil Transfer pump control. MDAFW 2B has local remote switches to control its motor operated mini-flow valve.

D. DG1B has local remote transfer switches for DG Fuel Oil Transfer pump control. MDAFW 1B does **NOT** have local control switches for it's mini-flow valve. The valve cycles on discharge flow.

45. 061K1.10 001/2/1/AFW - DG FUEL/MEM - 2.6/NEW/RO/NRC RO/TNT/DSS

K/A

061 Auxiliary / Emergency Feedwater (AFW) System

K1.10 Knowledge of the physical connections and/or cause effect-relationships between the AFW and the following systems:

Diesel Fuel Oil

K/A MATCH ANALYSIS

Question gives a plausible scenario where a CR fire has resulted in an LOSP and evacuation to the Remote Shutdown Panels. The DGs would be running and need Fuel Oil Transfer system to keep running. MDAFW pumps are powered from DGs. A unit difference has DG2B FO Transfer Pumps capable of being locally controlled from a 1E 280V MCC in the DG2B room. MDAFW pump B miniflow is also powered from this MCC in the diesel room and has local control of the miniflow at Shutdown Panel B. Candidate must determine which DG has this capability and proper MDAFW miniflow ops in evacuation scenario.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. DG2B has local control switches for DG Fuel Oil Transfer pumps and MDAFW B on both units has local control switches for the mini flow valve.

B. Incorrect. DG2B has local control switches for DG Fuel Oil Transfer pumps and MDAFW B on both units has local control switches for the mini flow valve..

C. Correct.

D. Incorrect. DG2B has local control switches for DG Fuel Oil Transfer pumps and MDAFW B on both units has local control switches for the mini flow valve.

REFERENCES

AOP-18038-2, "Operation From Remote Shutdown Panels"

LO-LP-61300-07C, "Unit Differences" page 10

VEGP learning objectives:

LO-LP-61300-03 Given a design or operational difference, be able to describe why it exists and the impact on the plant and its operation.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: The AFW pumps and throttle valves will NOT respond to automatic signals after control has been transferred. If a SG fed by the MDAFW pump is also being fed from the TDAFW pump, one of the MDAFW pump's throttle valves or its miniflow valve must remain fully open to ensure minimum miniflow requirements.

* 12. Control SG WR level(s) at 65% to 70% on all SGs:

a. On Shutdown Panel A:

1) Start MDAFW Pump A.

2) Throttle 2-HV-5139 and 2-HV-5137.

3) Operate miniflow 2-FV-5155 as follows;
OPEN - flow less than 175 gpm
CLOSE - flow greater than 330 gpm.

b. On Shutdown Panel B:

1) Start MDAFW Pump B.

2) Throttle 2-HV-5134 and 2-HV-5132

3) Operate miniflow 2-FV-5154 as follows;
OPEN - flow less than 175 gpm
CLOSE - flow greater than 330 gpm.

RESPONSE NOT OBTAINED:

* 12. IF MDAFW pumps not available, THEN initiate ATTACHMENT A, Turbine Driven AFW Pump Operation From Panel C.

applicable to **Fans #1** on **Unit 1** and **Fans #3** on **Unit 2.** NSCW motor coolers have orifices on Unit 2.

4. Diesel Generator

a. Fuel oil transfer switches

The Unit 2 "B" Train DG fuel oil transfer pumps have local/remote transfer switches (2HS-9045A and 9047A) to transfer control of the pumps out of the control room in case of a control room fire. The switches are located on the front of the breaker cubes (2BBF 27)

In remote the pumps run only in auto. There is no control switch for the pumps on the front of the breaker cube or anywhere outside the control room

1) "B" Train only

2) On breaker cubicle 2BBF27

3) Fire analysis report

4) Auto operation only

b. Day tank drain

Unit 2 has a drain line installed on the day tank so that it can be drained back to the fuel oil storage tank. The line has a locked closed isolation valve

No plans to change Unit 1

1) Drain entire tank

2) Lock closed drain valve

2) Human factors problem

5. RHR Cold Leg Injection Flow Orifices

The RHR discharge valves (HV-606 & 607) 100% demand signal moves the valve to a degrees limit open. Both Unit's RHR discharge valves have valve stops.

The Unit 1 RHR valve travel is limited to prevent pump runout because the flow orifices are sized larger than Unit 2's. The valve stroke is reduced to 50 degrees (Train A) to 55 degrees (Train B) open. The defining pressure drop in the system impacting RHR flow is across the discharge valves.

46. 062A3.04 001/2/1/AC DIST - INVERTER/C/A - 2.7/MODIFIED/RO/NRC RO/TNT/DSS

Unit 1 plant conditions:

* Startup of 1AD1I11 inverter is in progress following maintenance.
* The Control Building Operator has depressed and released the inverter precharge pushbutton.
* The precharge light is illuminated.

Which **ONE** of the following is correct ?

The DC input breaker.....

A. can be closed after the precharge light extinguishes, an interlock prevents closing with the precharge light illuminated.

B. cannot be closed because the light indicates an inverter input fuse has blown.

C.✓ can be closed within 3 seconds of releasing the precharge pushbutton, if the light extinguishes prior to closing, another precharge will have to be performed.

D. cannot be closed until subsequent steps place load on the inverter.

46. 062A3.04 001/2/1/AC DIST - INVERTER/C/A - 2.7/MODIFIED/RO/NRC RO/TNT/DSS

K/A

062 AC Electrical Distribution System

A3.04 Ability to monitor automatic operation of the AC distribution system, including:

Operation of inverter (e.g., precharging, synchronizing ligt, static transfer)

K/A MATCH ANALYSIS

Question asks plausible scenario for starting up a 120V AC 1E vital inverter with status of the precharge light. Candidate must choose whether or not start of the inverter could be performed in the given conditions and why / why not.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. There is no interlock for closing with the precharge light.

B. Incorrect. Precharge light indicates capacitor bank charged.

C. Correct. DC breaker must be closed within 3 seconds or precharge light may extinguish, if this occurs, another precharge must be performed.

D. Incorrect. DC breaker should be closed within 3 seconds once precharged.

REFERENCES

Oconee June 2004 NRC RO exam question # 45

SOP-13431-1/2, section 4.1.2 for 120V AC 1E Vital Inverter startup.

VEGP learning objectives:

Not applicable

1 POINT

Question 45

Unit 1 plant conditions:
- Startup of 1DIA inverter in progress following maintenance
- Precharge light is extinguished

Which ONE of the following is correct?

The DC input breaker...

A. cannot be closed because the input filter capacitors are not charged.

B. cannot be closed because the inverter input fuse is blown.

C. can be closed because the inverter is not connected to any loads.

D. can be closed because the input filter capacitors are indicating a full charge.

Goes with Q # 46

Question 45
T2/G1

062A3.04, AC Electrical Distribution System **Operation of inverter (e.g., precharging synchronizing light, static transfer)** (2.7/2.9)

Answer: A

A. Correct, the input filter capacitors have not been charged and could result in a blown fuse if the DC breaker were closed at this time.

B. Incorrect, this would be the result of closing the breaker under the given conditions.

C. Incorrect, the inverter may be connected to downstream loads via the manual transfer selector switch. However, the input filter capacitors must be charged (light lit) prior to closing the DC input breaker to prevent fuse damage.

D. Incorrect, this would be correct if the precharge light were illuminated indicating a full charge on the input filter capacitors.

Technical Reference(s):

Proposed references to be provided to applicants during examination:

Learning Objective: **EL-VPC, R2**

Question Source: **NEW**

Question History: Last NRC Exam ___________

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

Contacted Gerry Laska on 4/5/04 to discuss the selected KA. At ONS the vital AC busses do not have automatic actions based on bus amperage. Mr. Laska randomly selected another KA. The new KA is 062A3.04.

Replaced question with a better question. Changed to Memory or Fundamental Knowledge level.

NOTES

- The Precharge light must be lit to allow closing the DC INPUT Breaker. Holding the Precharge Pushbutton for an additional five seconds after the Precharge light illuminates will allow time to fully charge the capacitor bank.
- The Precharge Pushbutton has a strong spring and will require a firm push to operate.
- The DC INPUT breaker should be closed immediately (within 3 seconds) after releasing the Precharge Pushbutton. If the Precharge light extinguishes prior to closing the DC INPUT Breaker, Step 4.1.2.5 will need to be repeated.

4.1.2.5 Press and hold the Precharge Pushbutton, maintain depressed for at least five seconds after the Precharge light illuminates, then release. [ ]

4.1.2.6 Close the inverter DC INPUT Breaker (within 3 seconds after releasing the Precharge Pushbutton). [ ]

4.1.2.7 Verify proper inverter operation by observing approximately 120 VAC on the AC OUTPUT Voltmeter and 60 Hz on the AC OUTPUT frequency meter. [ ]

4.1.2.8 Close the INVERTER OUTPUT Breaker. [ ]

47. 062AA2.02 001/1/1/LOSS NSCW - CAUSE/C/A - 2.9/NEW/RO/NRC RO/TNT/DSS

The plant is at full power with DG1B monthly surveillance in progress.

The following annunciators illuminate on the QMCB:

* NSCW TRAIN B LO HDR PRESS
* NSCW TRAIN B DG CLR LO FLOW
* NSCW TRAIN B RHR PMP & MTR CLR LO FLOW
* CNMT CLR 3 & 4 LO FLOW
* CNMT CLR 7 & 8 LO FLOW
* RX CVTY CLG COIL LOW FLOW

Approximately 15 seconds later the crew notes the following:

* 3 NSCW pumps on Train B red lights are illuminated
* NSCW Train B supply and return flows **BOTH** indicate approximately 12,500 gpm

Which **ONE** would be the **CORRECT** initiating event and action to take?

A. Pump sheared shaft, trip ~~any~~ one of the running NSCW pumps. ← not credible to trip any pump

B. Catastrophic leak occurred, place all 3 NSCW Train B pumps in PTL.

C. Catastrophic leak occurred, depress BOTH DG1B Emergency Stop pushbuttons.

D.✓ Pump sheared shaft, allow standby pump discharge valve to stroke open.

47. 062AA2.02 001/1/1/LOSS NSCW - CAUSE/C/A - 2.9/NEW/RO/NRC RO/TNT/DSS

K/A

062 Loss of Nuclear Svc Water

AA2.02 Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water:

The cause of possible SWS loss.

K/A MATCH ANALYSIS

Question gives a plausible scenario where NSCW alarms are received on the main control board. The candidate must determine the action to take and the possible cause of the NSCW annunciators. Actions are from AOP-18020-C, Loss of NSCW.

The scenario is a sheared shaft of a running pump. The standby pump would start but discharge valve does not begin to open until approximately 45 seconds after the pump starts and not fully open until approximately 65-70 seconds. The stem states 30 seconds has passed. Therefore, standby pump would not be injecting into system yet.

Also, supply and discharge flows are low (18,000-19000 gpm normally) but balanced. This is not indicative of a catastrophic break where there would be a large mismatch in these flows.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Standby pump not supplying system yet so low header pressure alarm would not be clear yet. Cannot tell from QMCB which pump would have the sheared shaft and a local inspection would have to determine this. Would be improper to stop one of the running pumps until this could be determined.

B. Incorrect. Catastrophic leakage not indicated since supply / discharge flows are balanced. Al 3 pumps should not be stopped. Leak detection alarms would be expected for this situation too.

C. Incorrect. Catastrophic leak not indicated. As soon as standby pump discharge valve opens, DG cooling flow will be normal again.

D. Correct. As soon as standby pump discharge valve opens system parameters will return to normal. Then a local determination of which pump has sheared shaft could be made.

REFERENCES

ARP-17003, windows B01, C03, C04, D02, E02, F02 for the various annunciators.

AOP-18020-C. "Loss of Nuclear Service Cooling Water (NSCW).

VEGP learning objectives:

LO-LP-60316-04 Given the entire AOP, describe:

a. Purpose of selected steps
b. How and why the step is performed
c. Expected response of the plant/parameter(s) for the step

	(1)	(2)	(3)	(4)	(5)	(6)
A	NSCW TRAIN B F-1 HI VIB	NSCW TRAIN B F-2 HI VIB	NSCW TRAIN B F-3 HI VIB	NSCW TRAIN B F-4 HI VIB	CCW TRAIN B SURGE TK LO-LO LVL	CCW TRAIN B LO HDR PRESS
B	NSCW TRAIN B LO HDR PRESS	NSCW TRAIN B TRANSF PMP LO DISCH PRESS			CCW TRAIN B SURGE TK HI/LO LVL	CCW TRAIN B LO FLOW
C	NSCW TRAIN B CLG TWR BASIN HI/LO LVL		NSCW TRAIN B DG CLR LO FLOW	NSCW TRAIN B RHR PMP & MTR CLR LO FLOW	CCW TRAIN B SURGE TK MAKE UP LVL	CCW TRAIN B RHR HX HI FLOW
D		NSCW TRAIN B CNMT CLR 3 & 4 LO FLOW	NSCW INTERTIE TRN B TO TRN A HI FLOW			CCW TRAIN B RHR HX LO FLOW
E		NSCW TRAIN B CNMT CLR 7 & 8 LO FLOW	NSCW TRAIN B NORM/BYP VLV MISPOSITIONED		CCW TRAIN B RHR PMP SEAL LO FLOW	
F		NSCW TRN B RX CVTY CLG COIL LOW FLOW				

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. VERIFY only 2 NSCW pumps running normally in the AFFECTED train.	1. PERFORM the following as applicable: If catastrophic leakage is present: a. PLACE all NSCW pump handswitches in the AFFECTED train in PULL-TO-LOCK, b. PERFORM the following for the AFFECTED Emergency Diesel Generator: • DEPRESS BOTH Emergency Stop pushbuttons, -AND- • DISABLE automatic operation by initiating 13145, DIESEL GENERATORS section 4.4.11. c. VERIFY proper operation of UNAFFECTED NSCW train and go to Step 4.

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

(Step 1 continued from previous page)

IF no pumps or only one pump can be placed in service, THEN:

a. PLACE all NSCW pump handswitches in the AFFECTED train in PULL-TO-LOCK,

b. PERFORM the following for the AFFECTED Emergency Diesel Generator:

- DEPRESS BOTH Emergency Stop pushbuttons,

-AND-

- DISABLE automatic operation by initiating 13145, DIESEL GENERATORS section 4.4.11.

c. INVESTIGATE cause for trip of running pump(s).

d. VERIFY proper operation of UNAFFECTED NSCW train and go to Step 4.

IF three NSCW pumps are running, and the low header pressure annunciator is clear, THEN TRIP one NSCW pump and go to Step 2.

Won't clear until standby disch valve opens.

IF power is not available per status light indication, THEN INITIATE 18031-C, LOSS OF CLASS 1E ELECTRICAL SYSTEMS.

Can't determine which is sheared until local inspection, even if alarm does clear.

48. 062K3.01 001/2/1/AC ELEC - MAJOR LOAD/C/A - 3.5/NEW/RO/NRC RO/TNT/DSS

The plant has tripped from 100% power.

The following sequence of events occurs:

* Safety Injection actuates and has **NOT** been reset

* 2 minutes later, LOSP train "A" occurs due to loss of RAT "A"

* DG1A re-energizes the bus and the appropriate sequence runs.

Which **ONE** of the following **CORRECTLY** describes a pump which would be running assuming all systems respond as ~~expected~~ designed ?

A. Normal Charging Pump

B.✓ Safety Injection Pump Train A

C. Nuclear Service Cooling Water Pump # 5 Train A

D. Auxiliary Component Cooling Water Pump Train A

48. 062K3.01 001/2/1/AC ELEC - MAJOR LOAD/C/A - 3.5/NEW/RO/NRC RO/TNT/DSS

K/A

062 AC Electrical Distribution System

K3.01 Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following:

Major system loads.

K/A MATCH ANALYSIS

Question asks plausible scenario following a plant trip where a Safety Injection followed by an LOSP sequence on 1 train occurs with the DG1A re-energizing the bus. Candidate must determine which of several listed pumps would still be running.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Normal charging pump receives a trip signal on SI and cannot be started until SI is reset.

B. Correct. Without SI reset, SI pump A would get a start signal when LOSP / SI sequence is run. SI sequence would be the pre-dominant sequence. Plausible to confuse this because we have notes in procedures which require starting this pump after LOSP sequence run but AFTER SI is RESET.

C. Incorrect. NSCW pump # 5 would remain running on the SI sequence if already running. But, is load shed and not run on the LOSP sequence unless one of the other running NSCW pumps trips or fails to start on LOSP sequence.

D. Incorrect. ACCW pump # 1 (Train A) would remain running if running on the SI sequence. BUT, would load shed and is not restarted on an SI sequence which would be the predominant sequence run when LOSP occurs.

REFERENCES

Vogtle Text Chapter 28 pages 11 through 18 on sequencer operation.

19000-C, "Reactor Trip or SI step # 11 and # 29

V-LO-PP-04101 for ACCW pump start (2 pages)

V-LO-PP-06101 for NSCW pump start (2 pages)

V-LO-PP-13101 for SIP pump start (1 page)

VEGP learning objectives:

LO-PP-28201-03 Describe sequencer operation, including load shedding, load sequencing, and diesel generator operation under the following conditions:

a. Undervoltage (UV)
b. Safety Injection
d. SI followed by UV

LO-PP-06101-09 Describe the NSCW system response to an SI or LOSP signal.
LO-PP-04101-04 From memory describe the expected system response and operator corrective actions for each of the following:

a. SI
b. LOSP
c. SI followed by LOSP
d. LOSP followed by SI

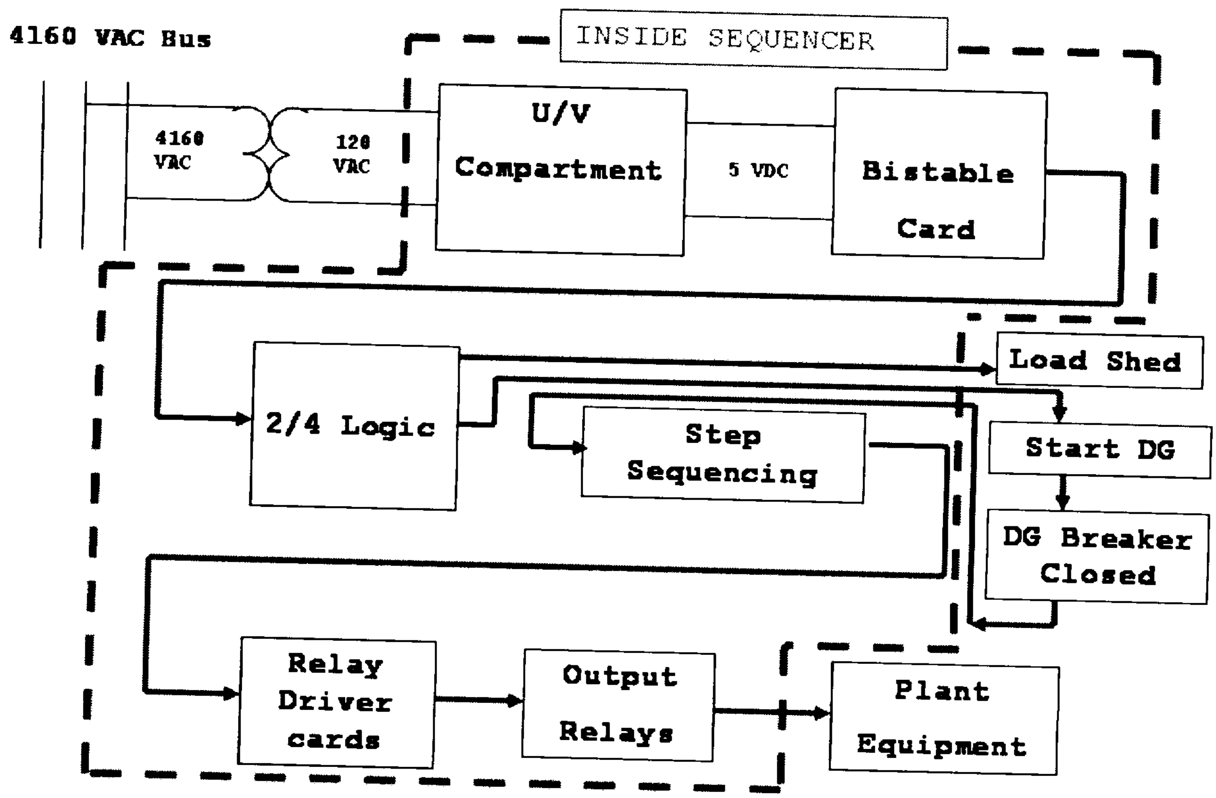
LO-PP-09200-03 List the trips associated with the NCP.

LO-PP-13101-05 Given the following components, describe the response of each on receipt of a SI signal and their alignment during cold leg and hot leg recirculation phases.

Sequencer Operation in Response to an UV Condition.

The Sequencer is constantly monitoring for a loss of or degraded voltage on its 4160 VAC bus. The following diagram shows how the signal is processed from the bus to the plant loads.

UV Processing Block Diagram



Before discussing sequencer operation on an U/V or SI signal, there are two functions that occur during sequenced operation that should be explained. These are the Block Auto/Manual signal and non sequenced load relays. These functions occur during SI and U/V sequencing.

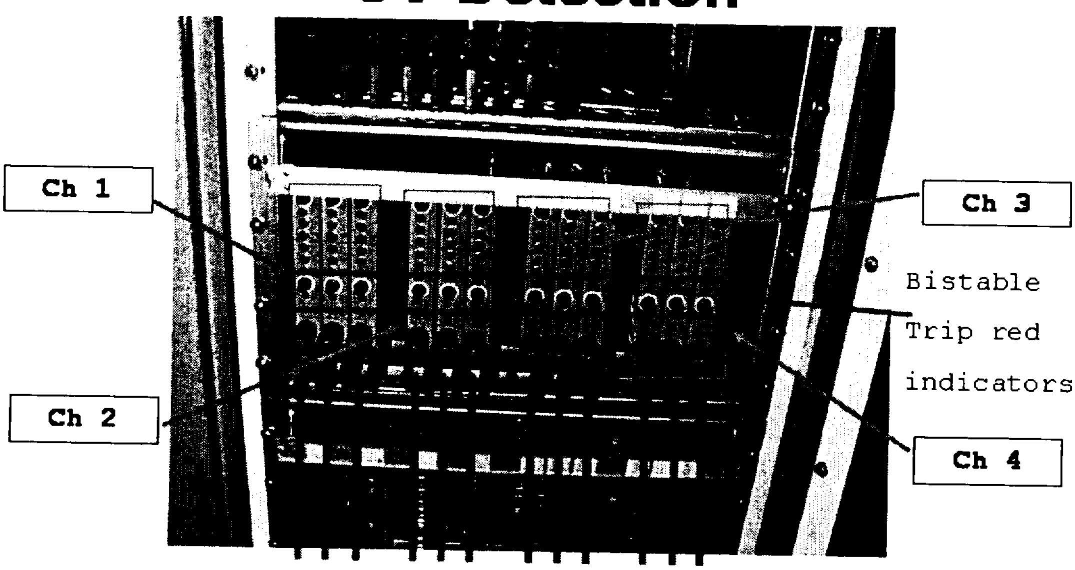
The Block Auto/Manual signal is a signal that prevents a change in equipment operation in two cases. First, the Block Auto/Manual signal blocks equipment from automatically starting due to an auto start signal caused by a normal system process such as a low header pressure or high temperature. A good example of this is the prevention of the start of the standby NSCW pump on low header pressure during an SI signal. In the second case, the Block Auto/Manual signal blocks the operator from manually stopping the equipment. Both of these blocks are their to prevent other load changes on the bus during load sequencing to ensure voltage and load limits on the bus are maintained. Manual starting of equipment by the operator is not blocked in any way to allow starts of equipment, if equipment failures require it. If load sequencing is in progress, operators should allow this to complete prior to starting this equipment. The Block Auto/Manual signal is generated on an U/V or SI signal. The signal is removed 36 seconds after receipt of an SI only. For an U/V signal or an U/V and SI signal, the Block Auto/Manual signal is removed 36 seconds after the DG output breaker is closed.

Non sequenced load relays are relays that are actuated immediately by the sequencer on the actuation signal (U/V or SI). The UV non sequenced relays actuate loads such as valves, generate trip signals to breakers, or provide start permissive to start on load sequencing. These loads generally do not need power from the emergency AC bus to actuate or are small loads; therefore load sequencing is not required. The SI non sequenced relays provide block signals to loads that should not start with an SI signal present.

Some examples of non-sequenced loads are the TDAFWP steam admission valve HV-5106), AFW discharge motor operated valves, steam generator blow down isolation valves, tripping of the incoming feeder breakers (U/V only), and tripping of the normal charging pump (SI only).

The Sequencer System monitors for three decreased levels of bus voltage (U/V). The U/V Bistable cards in Logic Bay 2 perform this function. If the bistable trip setpoint is reached, the "Red Trip" lamps on the corresponding bistable modules are lit. These trip indicators are sealed in and must be manually reset by depressing the indicator.

UV Detection



1st Level $\leq$ **71.5% for** $\geq$ **0.8 sec**
2nd Level $\leq$ **90% for** $\geq$ **20 sec**
3rd Level $\leq$ **93.1% for** $\geq$ **10 sec**

The U/V Schemes are as follows:

First level voltage- (INSTANTANEOUS Trip) <71.5 %(2975 VAC) for >0.8 sec.
Coincidence is 2/4.

Second level voltage- DEGRADED(Trip) <90 % (3746 VAC) for >20 sec.
Coincidence is 2/4.

Third level voltage- (Alarm only) <93.1 % (3873 VAC) for >10 sec.
Coincidence is 2/4.

The operation of the Vogtle Safety Features Sequencer System is automatic. Upon receiving a bus undervoltage (U/V) signal, the sequencer will automatically shed loads from the power bus, provide a start signal to the diesel generator, and, when the diesel generator is on line, sequentially return selected loads to the bus.

On an undervoltage actuation (First or second Level) the following actions are performed by the sequencer: (There is figure 1, Sequencer Manual Test Panel at the end of this section that can be used with the discussions of Sequencer operation)

Time=0 sec

Emergency start signal sent to Diesel Generator.

Load shed occurs: ACCW would be shed here

1 sec trip signal sent to the Normal and Emergency feeder breakers to the 4160 VAC bus.

1 sec trip signal sent to pump breakers to the 4160 VAC bus.

1 sec trip signal sent to 480 VAC secondary side (low side) Switchgear breakers for all 1E and Non 1E loads.

NOT loaded back on SI

Auto/Manual Block circuit is enabled.

U/V non sequenced load relays are energized to actuate those loads. This same signal generates the UNDERVOLTAGE light on the sequencer panel.

Signal sent to Loss of Power (LOP) monitor circuit.

Reset and stop signal sent to ATI subsystem. ATI step counter is reset to 00 and stopped.

Reset and inhibit signal sent to Manual test circuits.

Lights **U/V SIGNAL** (red), **U/V RELAYS ACTUATED** (red), **BLOCK AUTO/MNL SIG** (red), **SEQ LOGIC FAILURE** (amber), **UNDERVOLTAGE** (amber) generated on Sequencer panel.

Sequencer Trouble and **AA02 (BA03)SWGR Trouble** alarms received in the control room on the QEAB.

Time=0.5 sec

Sends DG Breaker Auto Closure permissive to DG output breaker closure circuit if not blocked by LOP monitor circuit.

Time=6.0 to 11.5 secs	When DG ready to load, DG output breaker closes. **D-G BRKR CLOSED** (red) light generated on Sequencer panel. Sequencer elapsed time display begins running.
Brkr CL +0.5 to 30.5 secs	**SEQ STEPS INDICATION** (red) for steps 1A-9A and 1C-9C will begin flashing in the intervals specified in the list below as the components are sequence on.

UV LOAD SEQUENCE

Train A only (Train B loads are similar)

TIME	LOAD
0.5 secs	CCP A, 480 VAC Secondary side feeder breakers
5.5 sec	NONE
10.5 sec	NB01 (Stub bus secondary side feeder breaker closes)
15.5 sec	ACCW Pump 1
20.5 sec	CCW Pumps 1 and 3 MDAFW Pump A
25.5 sec	NSCW Pumps 1 and 3 CCW Pump 5 (if CCW Pumps 1 or 3 breaker did not close)
30.5 sec	CTMT Cooling Units 5 and 6 (Fast Speed) NSCW pump 5 (if NSCW Pumps 1 or 3 breaker did not close) CTMT Cooling Units 1 and 2 start contact closed (Fast Speed)

CTMT cooling units must not all be started at the same time to prevent bus voltage transients. Analysis has shown that if all four were allowed to be simultaneously, DG voltage could drop below 80 percent. The sequencer provides all CTMT cooling units with a start signal at the 30.5 second step. Coolers 1, 2, 7, and 8 start at 50.5 seconds due to an additional time delay of 20 seconds by an agastat time delay relay in the auto-start circuit. This is for a UV condition only.

Time=32 secs	**SEQ STEPS INDICATION** (red) flashing lights extinguish. **SAF EQPT FAIL TO START** (amber) light to indicate that Cnmt Coolers 1 and 2 have not started. Audible alarm is sounded on the sequencer panel. Alarm generated on QEAB. ATI stop removed and ATI restarted
Time=36 secs	**BLOCK AUTO/MNL SIG** (red) light extinguishes.
Time=50.5 sec	CTMT Cooling Units 1 and 2 start (Fast Speed) *This is not a sequencer function but internal to the start logic of the cooler high speed motors* **SAF EQPT FAIL TO START** (amber) light extinguishes. QEAB alarm clears.

SEQ LOGIC FAILURE (amber) light and **UNDERVOLTAGE** (amber)light can be reset by depressing their indicator lights.

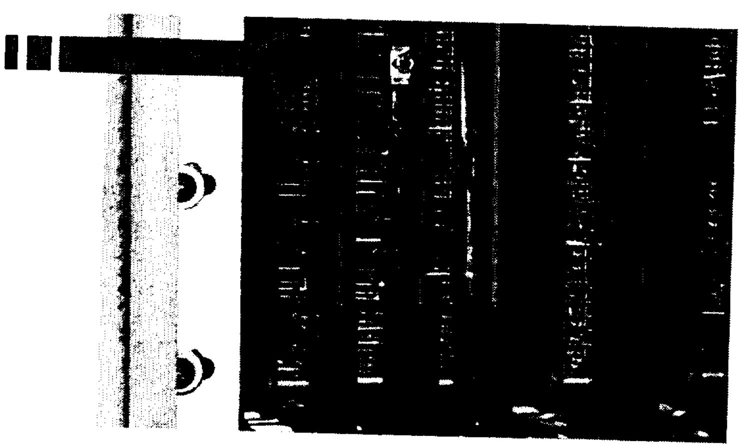
The undervoltage bistable trip indicator lights that were actuated will still be lit. Depressing the indicator lights will rest the indication.

The LOP monitor circuit discussed above counts the number of INSTANTANEOUS and DEGRADED undervoltage signals received in a two hour sliding window.

The purpose is to prevent exceeding the starting duties of the ESF equipment. The third UV signal received in a two hour period causes:

1. Start D/G
2. Load Shed Bus
3. Auto/Manual Block to Sequencer
4. Blocks DG Breaker Auto Closure permissive to DG output breaker
5. Generates a **SEQ LOGIC FAILURE** (amber) on Sequencer panel and Sequencer Trouble alarm in the control room on the QEAB.

The block of the DG Auto closure permissive prevents the DG output breaker from closing when the DG ready to load permissive is received, thus preventing another loading sequence. The ARP for the Sequencer Trouble alarm provides guidance locating the LOP monitor card. This card has a toggle switch to reset the counting circuit. Manual Sequencer Testing bypasses counting circuit.



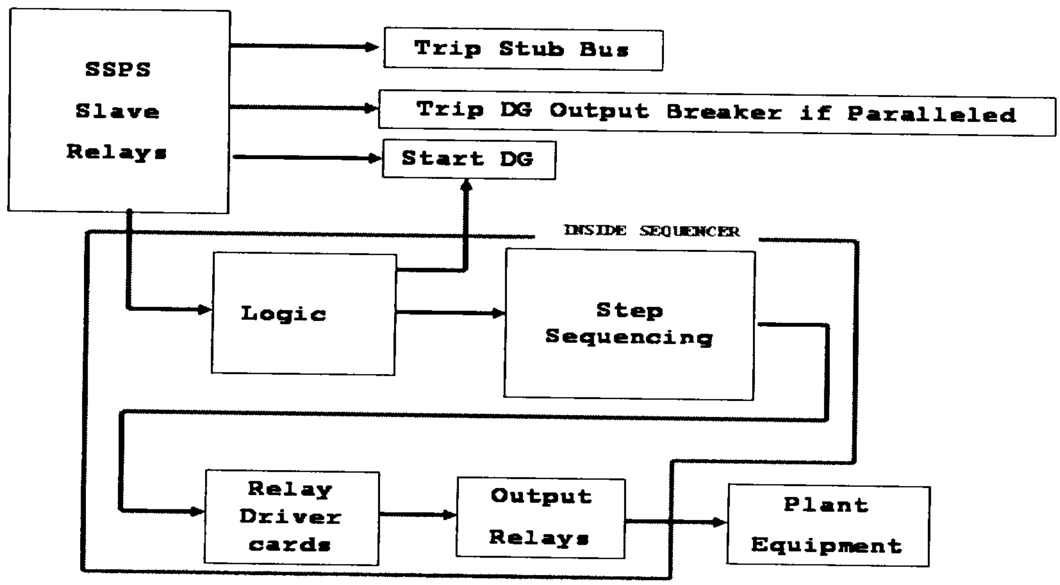
Sequencer Card Rack - Slot A4-2

Toggle Switch to reset UV Counting Circuit.

Sequencer Operation in Response to an SI Condition.

The Sequencer is also awaiting an SI signal. The following diagram shows how the SI signal is processed from the SSPS to the plant loads.

SI Signal Processing Block Diagram



On receipt of an SI signal, the following actions are performed by the sequencer:

Time=0 sec	Emergency start signal sent to Diesel Generator.
	Auto/Manual Block circuit is enabled.
	ATI step counter is reset to 00 and stopped.
	Reset and inhibit signal sent to Manual test circuits.
	Non sequenced SI maintained relays actuated.
	Non sequenced SI Momentary relays actuated for 1 sec
	Lights **SI SIGNAL** (red), **SI MAIN RELAYS ACTUATED** (amber), **BLOCK AUTO/MNL SIG** (red), **SI MOM RELAYS ACTUATED** (amber) generated on Sequencer panel.
Time≈0.1 secs	SI sequence timing starts. Sequencer elapsed time display begins running. This is set at 90 millisecs.
Time=1.0 sec	Light **SI MOM RELAYS ACTUATED** (amber) extinguishes.
Time=0.5 to 30.5 secs	Lights **SEQ STEPS INDICATION** (red) will begin flashing in the intervals specified in the list below. Steps 1-9, 1A-9A, 1B-9B, and 1C-9C steps are sequenced on.

SI LOAD SEQUENCE

Train A only (Train B loads are similar)

TIME	LOAD
0.5 sec	CCP A
5.5 sec	SIP A
10.5 sec	RHR pump A
15.5 sec	Containment Spray Pump A (W/ CSAS)
20.5 sec	CCW pumps 1 and 3 MDAFW Pump A
25.5 sec	NSCW pumps 1 and 3 CCW 5 (if CCW Pumps 1 or 3 breaker did not close)
30.5 sec	CTMT Cooling Units 1, 2, 5 and 6 (Slow Speed) NSCW 5 (if NSCW Pumps 1 or 3 breaker did not close)

ACCW NOT on SI sequence, NCP trips E-O Reference

Note that the Containment spray pump sequence is unique. At 15.5 secs the sequencer will send a 1 sec start signal. IF the CNMT Spray Actuation slave relay is energized the pump will start. This one sec start signal is sent on all the following steps to start CNMT spray if the actuation signal occurs. Therefore it is possible for the CNMT Spray pump to start at 15.5, 20.5, 25.5, or 30.5 secs depending on when a CNMT Spray actuation signal is received.

Time=6.0 to 11.5 secs	When DG ready to load, **D-G READY FOR LOADING** (red) light generated on Sequencer panel.
Time=32 secs	**SEQ STEPS INDICATION** (red) for 1B-9B and 1C-9C steps flashing lights extinguish. The 1-9 and 1A-9A steps will continue to flash until the SI signal is reset.
Time=36 secs	**BLOCK AUTO/MNL SIG** (red) light extinguishes.

Note that if CCW pumps 5 or 6 or NSCW pumps 5 or 6 are in service before an SI signal occurs, then all three pumps will be running in those trains after the SI sequence is complete.

ATI must be manually reset after the SI signal is reset.

Sequencer Operation in Response to an SI Condition with a U/V Condition.

There are five separate combinations to consider on operation with an SI and an U/V condition. They are:

SI signal and U/V simultaneously
SI signal following U/V (before sequencing is complete)
SI signal following U/V (after sequencing is complete)
U/V following SI signal (before SI is reset)
U/V following SI signal (after SI is reset)

The general rule of operation in these conditions is the sequencer operation will be a combination of the U/V and SI sequences. By understanding priority system of the sequencer, each of the above combinations can be evaluated. The U/V sequence will predominate until the ESF bus is energized. With the bus energized, the SI signal will predominate.

If a SI and U/V signal are received simultaneously, the SI sequence will be initiated after the completion of the load shed and subsequent re-energization of the 1E bus by the EDG.

If a U/V signal is received after SI actuation, the sequencer will initiate a load shed and generate the permissive for the EDG to re-energize the 1E bus (the EDG would have previously been started by the initiating SI signal). After the 1E bus is re-energized, the loading sequence will be a function of the status of the SI signal. If the SI signal is still present, the SI sequence will be initiated at step 1.

If the SI signal is no longer present (i.e. SI has been reset) when the EDG re-energizes the 1E bus, then the U/V sequence will be initiated at step 1. After completion of the loading sequence, any SI loads required to be in service that were not started during the U/V sequence will have to be manually placed in service (i.e. SI pumps, RHR pumps, Containment Cooler Low Speed motors, ESF Chilled Water pumps and ESF Chillers).

If an SI signal is received after the U/V sequence has been initiated, the sequencer will suspend the U/V sequence upon receipt of the SI signal and restart at step 1 of the SI sequence. If the SI signal is received after completion of the U/V sequence, then the sequencer will begin at step 1 of the SI sequence. For these conditions, since the 1E bus was energized at the time the SI signal was received, no additional load shed will occur. Any U/V loads started from the initiating U/V signal will remain in operation.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. Verify ECCS Pumps and NCP status:	
☐ a. CCPs - RUNNING SI actuated Step 4	a. Perform the following for available CCP(s): 1) Place alternate miniflow valve handswitch in ENABLE PTL, ☐ HS-8508A ☐ HS-8508B ☐ 2) Start CCP(s).
☐ b. SI Pumps - RUNNING	☐ b. Start Pumps.
☐ c. RHR Pumps - RUNNING	☐ c. Start Pumps.
☐ d. NCP - TRIPPED	☐ d. Stop the NCP.
☐ 12. Verify CCW Pumps - ONLY TWO RUNNING EACH TRAIN	
13. Verify proper NSCW system operation:	
☐ a. NSCW Pumps - ONLY TWO RUNNING EACH TRAIN #1 + #3	
b. NSCW TOWER RTN HDR BYPASS BASIN handswitches - IN AUTO: ☐ • HS-1668A ☐ • HS-1669A	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
28. Check if RCPs should be stopped:	
a. ECCS Pumps - AT LEAST ONE RUNNING:	☐ a. Go to Step 29.
☐ • CCP or SI Pump	
☐ b. RCS pressure - LESS THAN 1375 PSIG	☐ b. Go to Step 29.
☐ c. Stop all RCPs.	
☐ 29. Check ACCW Pumps - AT LEAST ONE RUNNING	☐ 29. Try to start one ACCW Pump. ☐ <u>IF</u> an ACCW Pump can <u>NOT</u> be started within 10 minutes of loss of ACCW, <u>THEN</u> stop all RCPs. <u>IF</u> an ACCW Pump can <u>NOT</u> be started within 30 minutes of loss of ACCW, <u>THEN</u> close ACCW Containment isolation valves: ☐ • ACCW SPLY HDR ORC ISO VLV HV-1979 ☐ • ACCW SPLY HDR IRC ISO VLV HV-1978 ☐ • ACCW RTN HDR IRC ISO VLV HV-1974 ☐ • ACCW RTN HDR ORC ISO VLV HV-1975

step 4 → 18 are the SI loads, Since possible ACCW load shed + did not Restart

PUMP AUTO STARTS

2. Loss of Offsite Power on the train related bus

   a. This start is blocked anytime a Safety Injection signal is present

V-LD-PP-04101-ACCW

PUMP AUTO STARTS

1. Low discharge pressure(<130 psig on Unit 1)
   Low discharge pressure(<110 psig on Unit 2)

   a. Pressure switch that starts pump generates the alarm on the QMCB

   b. This start is blocked while the sequencer is running during a Loss of Offsite Power

   c. This start is blocked anytime a Safety Injection signal is present

V-LO-PP-04101 ACCW

AUTO START SIGNALS

1. **Safety Injection**

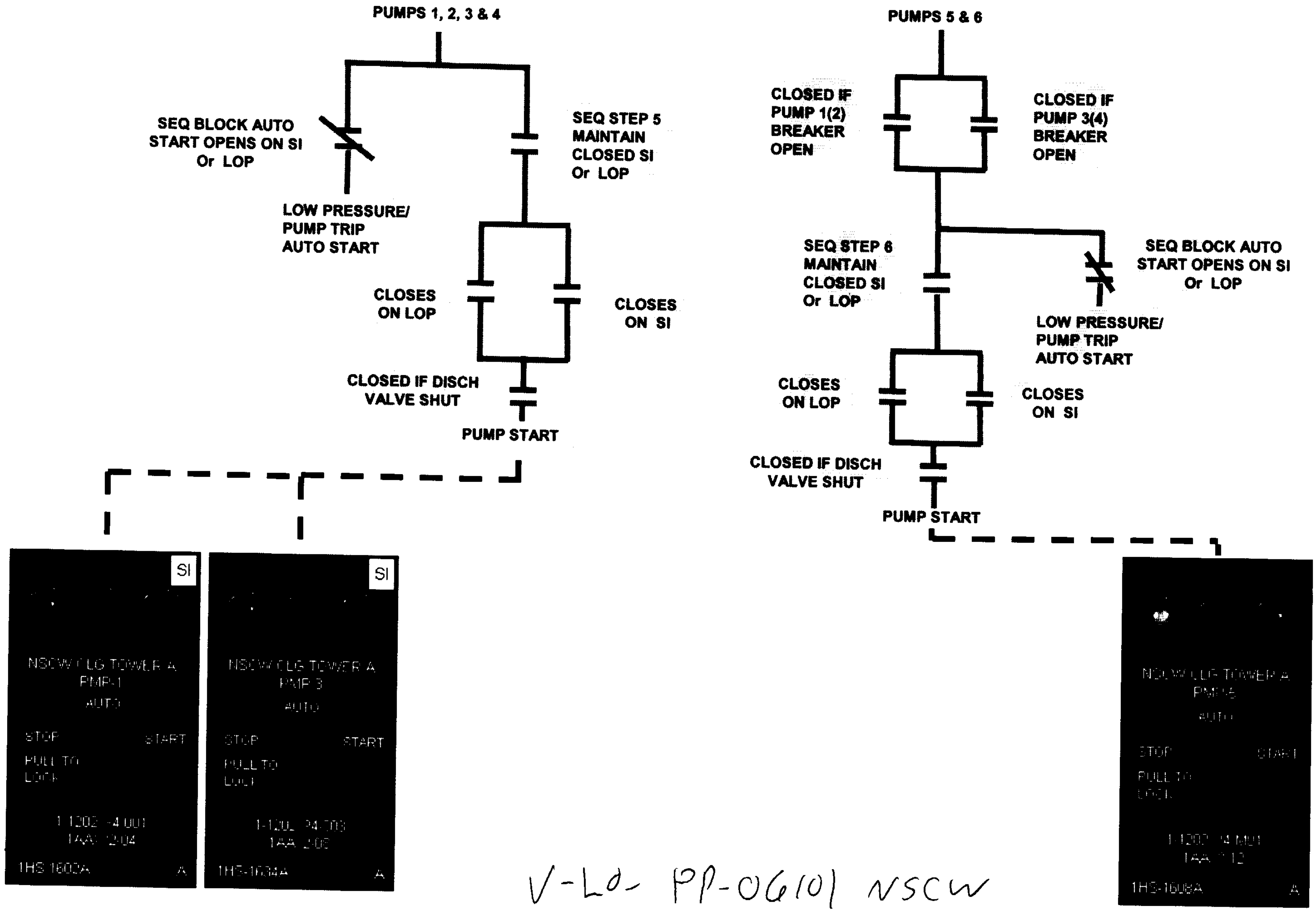
   Pumps 1 & 3 @ step 6(25.5 seconds)
   Pumps 2 & 4 @ step 6(25.5 seconds)
   Pump 5 or 6 @ step 7(30.5 seconds if needed)

2. **Loss of Offsite Power**

   Pumps 1 & 3 @ step 6(25.5 seconds)
   Pumps 2 & 4 @ step 6(25.5 seconds)
   Pump 5 or 6 @ step 7(30.5 seconds if needed)

V-LO-PP-06101 NSCW

START SEQUENCE



V-LO- PP-06101 NSCW

Intermediate Head Cold Leg Injection

- Intermediate Head Safety Injection (IHSI) System injects water into the RCS at pressures below 1625 psig.
- Main Function is in ECCS capacity
  - Also used to Fill SI accumulators

49. 063A2.01 001/2/1/DC DIST - GROUNDS/C/A - 2.5/NEW/RO/NRC RO/TNT/DSS

While at 100% power, the following annunciator illuminates:

* ALB34, window B01 for 125V DC SWGR BD1 TROUBLE

* The Control Building Operator (CBO) has been dispatched to investigate

Which **ONE** of the following is **CORRECT** regarding indications / actions to take if a **GROUND** has occurred ?

A. There are no local ground detection targets associated with BD1, the CBO would have no indication of a ground, maintenance would have to determine the cause.

B. A bus ground detection target would be dropped on BD1, de-energize panels BD1M, BD11, and BD12 one at a time to locate the panel with the ground.

C. There are no bus ground detection targets associated with BD1, de-energize all loads on BD1M, BD11, and BD12 one at a time ~~to locate the source of the ground.~~ until alarms clear

D.✓ A bus ground detection target would be dropped on BD1, de-enegize selected loads on BD1M, BD11, and BD12 one at a time to locate the source of the ground.

not credible with no ground detection targets

49. 063A2.01 001/2/1/DC DIST - GROUNDS/C/A - 2.5/NEW/RO/NRC RO/TNT/DSS

K/A

063 DC Electrical Distribution System

A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct , control, or mitigate the consequences of those malfunctions or operations.

Grounds

K/A MATCH ANALYSIS

Question asks plausible scenario where a ground has appeared on a 125V DC electrical bus. The operator must choose which of the indications would be available to determine the source of the ground and corrective action to take.Ground fault indication is available on BD1 but not the panels fed from BD1. The panels fed from BD1 should have loads selectively de-energized one at a time to determine the source of the ground. Not all loads can be de-energized. MSIVs and MFIVs in particular are powered from BD11 / BD12 and would result in a plant trip if de-energized.

ANSWER / DISTRACTOR ANALYSIS

A. Inorrect. There is ground fault indication available on BD1. Even though maintenance may have to be eventually called, selectively de-energizing loads on BD1 and buses fed from BD1 may find the ground.

B. Incorrect. Ground fault indication is available for BD1. However, de-energzing the BD11 or BD12 panel would result in a reactor trip due to MSIVs / MFIVs shutting when their solenoids de-energized.

C. Incorrect. There is ground fault indication available for BD1. However, ALL loads on BD11 or BD12 cannot be de-energized without resulting in a reactor trip. See B above.

D. Correct. Ground fault indication is available for BD1 and BD1M, BD11, and BD12 should have loads selectively de-enrgized to try to find the ground.

REFERENCES

ARP-17034, window B01 for 125V DC SWGR 1BD1 TROUBLE

VEGP learning objectives:

Not applicable.

WINDOW B01

ORIGIN	SETPOINT	
1BD1 Prot Rly	Not Applicable	125V DC SWGR 1BD1 TROUBLE

1.0 **PROBABLE CAUSE**

1. One of the breakers on Switchgear 1BD1 tripped.
2. Bus ground fault.
3. Bus undervoltage or loss of voltage.
4. Loss of control power.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. Check for associated alarms and indications.
2. If necessary, dispatch an operator to the switchgear to check for:
   a. Ground fault indications,
   b. Loss of control power, (Battery Charger shutdown with corresponding switchgear breaker open, or control power breaker open)
   c. Tripped breakers,
   d. Bus undervoltage or loss of voltage,
   e. Other abnormal conditions.
3. If alarm is due to a breaker tripping on fault:
   a. Determine what loads are affected,
   b. If necessary, dispatch an operator to the affected switchgears to manually operate the breakers, under the direction of Control Room,
   c. Notify Maintenance and return affected equipment to service once the cause has been corrected.

WINDOW B01
(Continued)

NOTE

The DC distribution panels do not have ground detection relays.

4. If a bus ground protection alarm is indicated, selectively de-energize components on 1BD1, 1BD1M, 1BD11, and 1BD12 to locate the ground:

   a. Using "120V AC/125V DC Panel Load Data Base" 1X3D-AA-M01C, determine which circuits can be momentarily de-energized by evaluating effect on plant systems,
   b. Momentarily de-energize selected circuits to locate the source of the ground,
   c. Initiate maintenance to clear the ground.

5. If the alarm is due to an undervoltage condition or loss of voltage:

   a. Return battery and/or charger to service if possible,
   b. If the chargers cannot be returned to service, monitor battery voltage and selectively strip the bus loads,
   c. If battery 1BD1B voltage drops to 109.7, notify Maintenance to perform 28912-C, "Class 1E Quarterly Battery Inspection And Maintenance,"
   d. If the bus cannot be returned to service, determine the affected loads and dispatch an operator to manually operate the breakers under direction of the Control Room,
   e. Notify Maintenance of the problem and return the bus to service when the cause is corrected.

6. If the alarm is due to loss of control power:

   a. Determine which breakers are affected,
   b. If loss is due to breakers being open, reclose the breakers,
   c. If loss is due to charger shutdown with corresponding breakers open restore bus to normal if possible,
   d. If the bus cannot be restored, notify Maintenance of the problem and return to service when the cause is corrected.

7. Refer to Technical Specification LCO 3.8.9, and 3.8.10.

50. 064A4.03 001/2/1/DG - SYNC SCOPE/C/A 3.2/NEW/RO/NRC RO/TNT/DSS

The following has occurred on Unit 1:

* RAT 1A de-energized due to a faulty relay
* DG1A auto starts and re-energizes 4160 1E bus 1AA02
* The faulty relay has been repaired.
* The crew is ready to parallel RAT 1A to 4160 1E bus 1AA02 being carried by DG1A

The operator should adjust DG1A SPEED CONTROL pushbuttons (RAISE or LOWER) until the Sync Scope needle is rotating slowly in the ___________ direction at great than 10 seconds per revolution. The incoming breaker should be closed as close to possible to the 12 o'clock position on the ____________ of the sync scope.

greater

A. CLOCKWISE, 11 o'clock side

B. COUNTERCLOCKWISE, 1 o'clock side

C. CLOCKWISE, 1 o'clock side

D✓ COUNTERCLOCKWISE, 11 o'clock side

2.0 **PRECAUTIONS AND LIMITATIONS**

2.1 **PRECAUTIONS**

2.1.1 When paralleling two AC sources, the following guidelines must be followed:

a. The speed of rotation of the sync scope should be relatively slow (at least 10 seconds per revolution), indicating the frequencies of the two sources are close to each other.

b. Normally, the incoming source frequency should be slightly higher than the bus frequency to ensure some immediate load pickup by the incoming source when the breaker is closed. However, when paralleling the grid to the 1E bus being carried by a diesel, the grid frequency cannot be adjusted. The DG frequency must be adjusted. The DG frequency is adjusted to slightly higher than grid frequency to prevent motoring the DG when the incoming breaker is closed.

c. Normally, when paralleling two AC sources, the incoming breaker should be closed as close as possible to the 12 o'clock position to minimize the phase difference between the two sources. The slower the sync scope is rotating, the closer to the 12 o'clock position the sync scope needle should be before trying to close the breaker. In all cases, the breaker should be closed as close as possible to the 12 o'clock position on the 11 o'clock side of the sync scope.

d. Since grid frequency cannot be adjusted, to prevent motoring the Diesel Generator, its frequency is adjusted to slightly higher than grid frequency causing the sync scope to rotate counterclockwise. The breaker should be closed as close to the 12 o'clock position as possible. (It is preferred for the breaker to close slightly on the 11 o'clock side of 12 rather than the 1 o'clock side because it produces less of a transient on the DG to attain synchronization with the grid, however, the intent is to close the incoming breaker at the 12 o'clock position.)

2.1.2 The Diesel Generator load should not exceed its rated capacity of 7000 kW. A 10% overload to 7700 kW is allowed for 2 hours during emergency conditions or when specifically required by test procedure.

4.2.2.6 If in the UNIT Mode, place the DSL GEN 1A UNIT/PARALLEL Switch 1HS-4414B momentarily to PARALLEL position AND check the blue UNIT MODE/FAST START light is not lit. [ ]

4.2.2.7 Using the Diesel Generator 1A VOLTAGE CONTROL pushbutton, adjust generator voltage by momentarily depressing the RAISE or LOWER pushbutton until it is slightly higher than the Normal Incoming Source voltage. [ ]

NOTE

> Since the Normal Incoming Source frequency, (the grid), cannot be adjusted, the following steps will increase Diesel Generator frequency to slightly greater than grid frequency to ensure the DG picks up load when the Normal Incoming Breaker is closed.

4.2.2.8 While observing the Sync Scope, adjust Diesel Generator speed using the Diesel Generator 1A SPEED CONTROL pushbuttons (RAISE or LOWER) until the Sync Scope needle is rotating slowly in the counterclockwise (Slow) direction [greater than 10 seconds per revolution]. [ ]

4.2.2.9 Set DSL GEN 1A LOADING SET PT CONTROL 1SE-4915 to the current Diesel Generator load. [ ]

Pot setting is calculated using the following formula:

$$\frac{\text{D / G LOAD [kW]}}{700} = \text{LOAD POT SETTING}$$

4.2.2.10 Review Precaution 2.1.1 on indication and operation of the synchroscope prior to proceeding. [ ]

CAUTION

> Lead time for closing the Normal Incoming breaker should be selected to ensure breaker closure as close as possible to the 12 o'clock position, to minimize the phase difference between the DG and the incoming power source.

4.2.2.11 When the Sync Scope needle reaches 12 o'clock, close NORM INCM BRKR 1AA0205. [ ]

4.2.2.12 Place BRKR 1AA0205 SYNCHRONIZING SWITCH to OFF. [ ]

2) Normally, the incoming source frequency should be slightly higher than the bus frequency to ensure some immediate load pickup by the incoming source when the breaker is closed. However, when paralleling the grid to the 1E bus being carried by a diesel, the grid frequency cannot be adjusted. The DG frequency must be adjusted. The DG frequency is adjusted to slightly higher than grid frequency to prevent motoring the DG when the incoming breaker is closed.

3) Normally, when paralleling two AC sources, the incoming breaker should **be closed as close as possible to the 12 o'clock position to minimize the phase difference between the two sources.** The slower the sync scope is rotating; the closer to the 12 o'clock position the sync scope needle should be before trying to close the breaker. In all cases, **the breaker should be closed as close as possible to the 12 o'clock position on the 11 o'clock side of the sync scope.**

4) Since grid frequency cannot be adjusted, to prevent motoring the Diesel Generator, its frequency is adjusted to slightly higher than grid frequency causing the sync scope to rotate counterclockwise. **The breaker should still be closed as close to the 12 o'clock position as possible and still on the 11o'clock side.**

In AUTO the DG output breaker automatically closes when the following are met:

- No 186 LO relays
- DG READY to LOAD Voltage and Frequency are met
- TRS-LR selected to Control Room
- Auto sync check relay senses proper phase rotation
- Auto Sync pushbutton is held until breaker closes

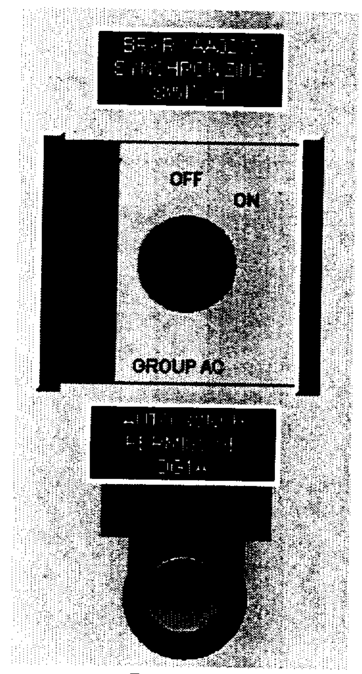


Figure 26

Synchronizing Lights are bright at the 6 o'clock position, and are dark at the 12 o'clock position. The Red AUTO SYNC PERMISSIVE LIGHT illuminates near the 12 o'clock position.

In MANUAL the operator manually closes breaker. (Always **11 o'clock side)**

i. SYNCHRONIZATION SWITCH

There is a synchronizing interlock for the breakers that have the ability to cause different sources to be paralleled. This interlock helps ensure proper phase rotation between the two sources prior to closing the breakers. This switch also enables the synch scope and the sync lights.

j. UNIT PARALLEL SWITCH (Figure 23)

This is a 3 Position Spring return to center position switch which is used to select voltage regulator and speed control operating modes, (if LRS in LOCAL). There is no procedure guidance for parallel operations from the LOCAL panel.

51. 064K6.07 001/2/1/DG - AIR RECEIVERS/C/A - 2.7/NEW/RO/NRC RO/TNT/DSS

A Loss of All AC power has occurred on Unit 1 with the following conditions.

* Both DGs fail to emergency start due to a common failure.
* Both DGs taken to LOCAL control to investigate the problem.
* DG1A air receiver pressures read 145 and 148 psig respectively.
* DG1B air receiver pressures read 158 and 162 psig respectively.
* An alignment problem found fuel valves to both DGs manually isolated.
* Alignment problem has been corrected.

Which **ONE** of the following is **CORRECT** regarding starting the DGs if control is transferred back to REMOTE ?

A. Both DGs would accept an automatic Emergency Start signal and start.

B. DG1B would accept an automatic Emergency Start signal and attempt to start. DG1A could NOT be started under any circumstances.

C.✓ DG1B would accept an automatic Emergency Start signal and atttempt to start. DG1A could still be started using other methods.

D. Neither DG1A or DG1B accept an auto Emergency Start signal. Both DGs could be locally "Emergency Started" on their local control panels.

51. 064K6.07 001/2/1/DG - AIR RECEIVERS/C/A - 2.7/NEW/RO/NRC RO/TNT/DSS

K/A

064 Emergency Diesel Generator (ED/G) System

K6.07 Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system.

Air receivers

K/A MATCH ANALYSIS

Question gives plant status with a Loss of All AC power in progress and both EDGs have failed to start due to a alignment problem isolating fuel oil. Air receiver pressure for DG1A is < 150 psig so it will not accept an emergency start signal in REMOTE. DG1B has 1 air receiver > 150 psig will accept an emergency start signal in REMOTE. DG1A could be started with the special guidance for ALB35 Window F05 for DG1A FAILED TO START by other methods until approximately 90 psig remains in receivers.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Only DG1B would accept REMOTE Emergency Start signal.

B. Incorrect. DG1B would accept REMOTE Em. Start signal. DG1A could be started.

C. Correct. DG1B would accept REMOTE Em. Start signal. DG1A could be started.

D. Incorrect. DG1B would accept REMOTE Em. Start signal. DG1A would need special guidance but could be started until approximately 90 psig in the receivers.

REFERENCES

OSP-13145-1, "Diesel Generators" limitation 2.2.1

ARP-17035-1, window F05 for DG1A FAILED TO START

VEGP learning objectives:

LO-PP-11101-10 Describe the operation of each of the system components listed below as related to: 1) Engine Normal Start Signal; 2) Engine Emergency Start Signal; 3) Low Air Start System pressure.

b. air receivers
c. solenoid operated air start admission valves

2.1.16 If Control Air pressure is depleted during ~~D/G~~DG operation, the ~~D/G~~DG will continue to run. Restoration of Control Air pressure while ~~the D/G~~DG is running may impact ~~the~~ load carrying capability of the ~~D/G~~DG. ~~Do NOT restore~~ Control Air pressure shall NOT be restored until ~~D/G~~DG shutdown is desired.

2.1.17 In the event of an abnormal trip of the ~~D/G~~DG, all personnel should avoid the Left Bank Side of the Engine for at least 15 minutes due to the risk of a Crankcase Explosion.

2.1.18 Any time a ~~D/G~~DG is operated greater than six hours in any 24-hour period, Environmental Services must be notified to ~~ensure~~verify compliance with EPA Regulations. Notification should be made by calling ~~the~~ Environmental Services on-call Pager Number 1-800-522-2246. Pin Number 0444071 will be required.

2.1.19 Independent Verifications performed in this procedure should be documented on Checklist 5(6), "Train A(B) Independent Verification."

2.2 **LIMITATIONS**

2.2.1 A Diesel Generator will NOT accept an Emergency Start signal from the Control Room if any of the following conditions exist:

a. LOCAL-REMOTE switch 1HS-4516(4517) at PDG1(PDG3) in LOCAL,

b. Starting air pressure in both air headers less than 150 psig,

c. Engine controls in the Maintenance Mode,

d. Emergency Stop circuit energized,

e. Overspeed trip NOT reset.

NOTE

> A Diesel Generator Emergency Start is initiated by closure of the Train A or B Engineered Safety Feature Safety Injection contacts, Loss of Offsite Power, or operation of the manual Emergency Start Switch at the Engine Control Panel. All other Diesel Generator start signals are considered to be a Normal Start.

2.2.2 The following Diesel Engine shutdown signals are bypassed during an Emergency Start:

a. High Crankcase Pressure,

b. High Engine/Turbocharger Vibration,

c. Low Turbocharger Oil Pressure,

d. High Engine Bearing Temperature,

e. High Engine Lube Oil Temperature,

f. Low Jacket Water Pressure,

g. Loss Of Field And Phase Overcurrent 186B (SI only), ALB 35

	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)
A	DG1A LOW TEMP LUBE OIL-IN	DG1A LOW TEMP LUBE OIL-OUT	DG1A HI TEMP LUBE OIL-IN	DG1A HI TEMP LUBE OIL-OUT	DG1A TRIP HIGH TEMP LUBE OIL	DG1A LOW LEVEL LUBE OIL	DG1A TRIP HIGH TEMP ENGINE BRG	DG1A TRIP HI CRANKCASE PRESS	DG1A VIBRATION TRIP	DG1A TRIP OVERSPEED
B	DG1A LOW PRESS LUBE OIL	DG1A TRIP LOW PRESS LUBE OIL	DG1A LOW PRESS TURBO OIL RIGHT	DG1A LOW PRESS TURBO OIL LEFT	DG1A TRIP LOW PRESS TURBO OIL	DG1A HI DIFF PRESS LUBE OIL FILTER	DG1A LOW OIL PRESS SENSOR MALFUNCTION	DG1A ENGINE CNTL POWER A FAILURE	DG1A ENGINE CNTL POWER B FAILURE	DG1A DISABLED GEN CONTROL PWR FAILURE
C	DG1A LOW TEMP JACKET WATER IN	DG1A LOW TEMP JACKET WATER OUT	DG1A HI TEMP JACKET WATER IN	DG1A HI TEMP JACKET WATER OUT	DG1A TRIP HI TEMP JACKET WATER	DG1A LOW PRESS JACKET WATER	DG1A TRIP LOW PRESS JACKET WATER	DG1A LOW LEVEL JACKET WATER		DG1A DISABLED NONRESET OF EMERGENCY TRIP
D	DG1A LOW PRESS FUEL OIL	DG1A HI DIFF PRESS FUEL OIL FILTER	DG1A FUEL OIL INJECTION LINE BREAK	DG1A HIGH LEVEL MAIN TANK	DG1A LOW LEVEL MAIN TANK	DG1A GEN UNDER FREQ	DG1A HIGH OR LOW LEVEL DAY TANK			DG1A DISABLED DG CKT BRKR INOPERABLE
E	DG1A GENERATOR TROUBLE	DG1A LOW VOLTAGE	DG1A HI TEMP GEN CNTL PNL		DG1A DISABLED ENGINE CONTROL IN LOCAL	DG1A TRIP GENERATOR FAULT	DG1A TRIP GEN DIFF	DG1A HIGH TEMP GEN BEARINGS	DG1A * LOCAL ANN PNL PWR FAILURE	DG1A DISABLED MAINTENANCE LOCK OUT
F	DG1A LOW PRESS CONTROL AIR	DG1A LOW PRESS STARTING AIR	DG1A HIGH PRESS STARTING AIR	DG1A DISABLED DC START POWER FAILURE	DG1A FAILED TO START	DG1A SWITCH NOT IN AUTO	DG1A BARRING DEVICE ENGAGED	DG1A PANEL INTRUSION	DG1A HI TEMP PANEL	DG1A EMERGENCY START

* CONTROL ROOM ALARM ONLY

WINDOW F05
(Continued)

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NOTE

Alarm does not prevent ~~diesel~~ the Diesel Generator from starting.

1. If the engine starts and continues to run, press the Start Pushbutton, 1-HS-4569B, to reset the Keep-Warm System, stopping the pumps and reinstating alarms that are bypassed during shutdown.

2. If the engine fails to start from a Normal Start:

   a. Notify Engineering Support.

   b. Dispatch an operator to check for proper operation and line up of Fuel Oil and Starting Air Systems.

   c. Correct any malfunctions or improper lineups.

   d. Reset alarm and attempt to restart engine, per 13145-1, "Diesel Generators," after cause of alarm has been corrected.

3. If the engine fails to start from an Emergency Start:

   a. Dispatch an operator to check for proper operation and line up of Fuel Oil and Starting Air Systems.

   b. If both air receivers are <150 psig and still >90 psig:

      (1) Contact Engineering and Maintenance for assistance.

      (2) Place the ~~diesel~~ Diesel Generator in the LOCAL position and depress the Reset From LOCA push-button located on the Engine Control Panel.

      (3) Isolate one of the air receivers by closing the outlet isolation valve for that receiver.

      (4) Verify Day Tank has fuel and is aligned to the engine properly.

WINDOW F05
(Continued)

(5) Verify that the shutdown cylinder located to the right of the mechanical governor and under the left bank turbocharger is fully retracted.

NOTE

> For verifying air butterfly valves are open, the scribe mark located on the bottom of the vertical shaft extending through the Intake Manifold should be in-line with the Intake Manifold when the valve is open. If not in the correct position, Engineering and Maintenance assistance will probably be required.

(6) Verify that the combustion air butterfly valves in the Intake Manifold are open.

(7) Perform Local Manual Start per 13145-1.

4. Refer to Technical Specifications LCO 3.8.1 or 3.8.2.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4AK01-48, 1X3D-BH-G03E, 1X3D-BH-G03F

52. 065AK3.03 001/1/1/LOSS AIR - EFFECT EQ/C/A - 2.9/NEW/RO/NRC RO/TNT/DSS

Which **ONE** of the following **CORRECTLY** describes the status of the RHR Hx Outlet (HV-0606/0607) and Hx Bypass Valves (FV-0618/0619) while at power ?

A. Air is supplied to the RHR Hx Outlets and isolated to the RHR Hx Bypass valves. RHR Hx Outlets in manual and fully open, Hx Bypass valves failed shut.

B. Air is supplied to both the RHR Hx Outlets and RHR Hx Bypass valves. RHR Hx Outlets in manual and fully open, Hx Bypass valves in manual and shut.

C. Air is isolated to the RHR Hx Outlets and supplied to the RHR Hx Bypass valves. RHR Hx Outlets failed fully open, Hx Bypass valves in manual and shut.

D.✓ Air is isolated to both the RHR Hx Outlets and RHR Hx Bypass valves. RHR Hx Outlets failed fully open, Hx Bypass valves failed shut.

outlet valves
vice outlets

52. 065AK3.03 001/1/1/LOSS AIR - EFFECT EQ/C/A - 2.9/NEW/RO/NRC RO/TNT/DSS

K/A

065 Loss of Instrument Air

AK3.03 Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air.

Knowing the effects on plant operation of isolating certain equipment from instrument air.

K/A MATCH ANALYSIS

Question asks status of air alignment to RHR Hx Outlet and RHR Hx Bypass valves at power. Air normally isolated to both sets of valves to ensure Hx Outlets failed open and Hx Bypasses failed shut to ensure full flow through RHR Hx in injection or recirc mode.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Air failed to both sets of valves.

B. Incorrect. Air failed to both sets of valves.

C. Incorrect. Air failed to both sets of valves..

D. Correct. Air failed to both sets of valves to ensure Hx Outlets failed open and Hx Bypass failed shut.

REFERENCES

SOP-13011-1, Residual Heat Removal System, Precaution 2.1.9.

VEGP learning objectives:

LO-LP-60321-02 State the fail position of the following valves on loss of instrument air:

i. RHR heat exchanger outlet valve
j. RHR heat exchanger bypass valve

2.1.8 The following valves have a vent hole drilled in the valve disk to prevent pressure locking:

a. Valves 1-HV-8716A AND 1-HV-8716B have a vent hole drilled on the RHR pump side of the disk. The valves are not capable of a leak tight seal when pressurized from the opposite train RHR pump,

b. Valves 1-HV-8840, 1-HV-8809A/B AND 1-HV-8811 A/B have a vent hole drilled on the RCS side of the valve disk. Pressure from the RCS side will force the valve disk to seal on the upstream seat. The ability of the valves to provide containment isolation has NOT been changed.

2.1.9 **When both trains of RHR are in standby readiness, air is isolated to Valves 1-HV-0606, 1-HV-0607, 1-FV-0618, and 1-FV-0619. Air must be restored to these valves prior to RHR Pump operation for RCS cooldown.**

2.1.10 Ensure HP is informed to perform surveys when changes are made to the RHR operational configuration. Changes in area condition levels can occur when a train is placed in service, when trains are swapped, or when letdown is established or changed. In addition, ensure operators in the Auxiliary Building are informed that radiation levels may change in their workspaces due to system changes.

2.1.11 RHR Pump Miniflow Valves 1-FV-610 and 1-FV-611 are designed to open if flow is <780 gpm and close if flow is >1841 gpm.

2.1.12 Independent Verifications performed in this procedure should be documented on Checklist 7, "Independent Verification."

53. 073K4.01 001/2/1/PRM - RELEASE TERM/MEM - 4.0/NEW/RO/NRC RO/TNT/DSS

Given the following plant conditions:

* Liquid release is in progress from Waste Monitor Tank (WMT) # 10 to the blowdown sump.

* Two ~~circulating~~ river water dilution valves to the blowdown sump are open.

Which **ONE** of the following is **CORRECT** regarding the response of the Liquid Waste Process System to isolate a release if **HIGH** radiation is detected ?

A. Both dilution valves shut, WMT # 10 transfer pump would trip.

B. RV-0018 liquid process isolation shuts, WMT # 10 transfer pump continues to run.

C. Both dilution valves shut, WMT # 10 transfer pump continues to run.

D. RV-0018 liquid process isolation shuts, WMT # 10 transfer pump would trip.

53. 073K4.01 001/2/1/PRM - RELEASE TERM/MEM - 4.0/NEW/RO/NRC RO/TNT/DSS

K/A

073 Process Radiation Monitoring (PRM) System

K4.01 Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following.

Release termination when radiation exceeds setpoint.

K/A MATCH ANALYSIS

Question gives a plausible scenario where a liquid release is in progress through the normal release flowpath via RV-0018 to the Blowdown Sump and then to the Savannah River. Candidate must choose how release is isolated on high rad to RE-0018. The dilution valves to blowdown sump close on low dilution flow of 12,000 gpm with a time delay but do not shut on high rad. The WMT # 10 transfer pump would continue to run on miniflow since it does not get a trip signal on high rad.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. RV-0018 is only valve used to isolate a liquid release on HIGH rad.

B. Correct. RV-0018 is only valve used to isolate a liquid release on HIGH rad.

C. Incorrect. RV-0018 is only valve used to isolate a liquid release on HIGH rad.

D. Incorrect. RV-0018 is only valve used to isolate a liquid release on HIGH rad.

REFERENCES

ARP-17005-1/2, window C03 for High Radiation Alarm, Auto action # 3 for RE-0018.

SOP-13216-1/2, "Liquid Waste Release"

VEGP learning objectives:

LO-PP-47101-08 Describe the major steps required for Operations to release a WMT.

LO-PP-47101-09 State the conditions that require immediate termination of a Liquid waste release.

WINDOW C03

ORIGIN	SETPOINT	HIGH RADIATION ALARM
K-1	Not Applicable	

1.0 **PROBABLE CAUSE**

A high alarm on one or more of the Radiation Monitor Channels.

2.0 **AUTOMATIC ACTIONS**

The following actions will occur if a High Level Radiation Alarm is actuated on the associated monitor:

1. 1-RE-0002 or 1-RE-0003, Containment Low Range Area Monitor: Containment Ventilation Isolation (CVI).
2. A-RE-0014, Waste Gas Processing System Effluent Radiogas Monitor: Closes valve A-RV-0014 to the Waste Gas Processing System discharge.
3. 1-RE-0018, Waste Liquid Effluent Monitor: Closes 1-RV-0018 to isolate the Liquid Waste Discharge Line.
4. 1-RE-0021, Steam Generator Blowdown Liquid Process Monitor: Isolates Steam Generator Blowdown Processing System.
5. 1-RE-0848, Turbine Building Drain Effluent Monitor: Diverts Turbine Building Drains to Dirty Drains Tank.
6. A-RE-2532 A or B or A-RE-2533 A or B, Fuel Handling Building Effluent Radiogas Monitors: Fuel Handling Building Isolation (FHBI).
7. 1-RE-2565 A, B or C, Containment Ventilation Effluent Monitors: Containment Ventilation Isolation (CVI).

4.2.8 Verify the "Batch Liquid Release Permit" for WMT 010 has been approved by the SS. [ ]

4.2.9 Start WASTE MONITOR TANK PUMP 010 using 1-HS-1083. [ ]

4.2.10 If 1-RE-0018 is operable, perform 4.2.12 And mark 4.2.13 n/a.

4.2.11 If 1-RE-0018 is inoperable, perform 4.2.13 and mark 4.2.12 n/a.

4.2.12 If 1-RE-0018 is operable:

a. PULSE CHECK 1-RE-0018 as follows:

(1) Verify Blowdown Sump dilution flow is at least 12,000 gpm and greater than flow required by the "Batch Liquid Release Permit". [ ]

ALB05-B3 INTMD RADIATION ALARM
ALB05-C3 HIGH RADIATION ALARM

(2) Notify the Control Room to expect an alarm on 1-RE-0018 on the Digital Radiation Monitor System. [ ]

(3) Open LWPS UNIT 1 CLEAN WASTE DISCH HI-RAD ISOL, 1-RV-0018 using 1-HS-0018. [ ]

(4) Request Chemistry to activate the pulse test on channel 1-RE-0018. [ ]

(5) Verify LWPS UNIT 1 CLEAN WASTE DISCH HI-RAD ISOL, 1-RV-0018 closes and Hi Radiation alarm annunciates. [ ]

(6) Position handswitch 1-HS-0018 to OPEN and verify LWPS UNIT 1 CLEAN WASTE DISCH HI-RAD ISOL, 1-RV-0018 remains closed. [ ]

(7) Request Chemistry to restore channel 1-RE-0018 to normal. [ ]

b. If Chemistry request to lower background, flush 1-RE-0018 with demin water per Section 4.8 [ ]

4.2.19 Throttle LWPS CLEAN WASTE DISCH RE-0018 DIVERT 1-1901-U4-329 (RD59) to obtain between 3 and 6 gpm as indicated on 1-FI-0018. [ ]

4.2.20 Record the following on the "Batch Liquid Release Permit":

a. Release start time and date [ ]

b. Release flow rate [ ]

c. 1-RI-0018 activity. [ ]

4.2.21 Notify the Control Room which tank is being released and the release start time. [ ]

4.2.22 Complete the Flow Channel Check Form attached to the release permit. [ ]

NOTE

Waste Monitor Tank 010 level as indicated on 1-LIS-1083 is 52 gallons per percent level.

4.2.23 Calculate Pump Discharge Flow and compare to indicated Flow by performing the following:

a. Monitor Tank Level Indicator 1-LIS-1083 and time. multiply change in level by 52 then divide by time required for level to change: [ ]

____ %Level change X $\frac{\text{52 gallons}}{\text{percent}}$ ÷ ___ minutes = ____ gpm

b. Record indicated flow from 1-FI-1085B and calculated flow rates in the SO Logbook [ ]

c. If Indicated flow is 25% greater than or less than Calculated flow, notify SS and write a WRT. [ ]

4.2.24 Upon completion of the release

a. Close LWPS UNIT 1 CLEAN WASTE DISCH HI-RAD ISOL, 1-RV-0018 using 1-HS-0018, (IV REQUIRED). [ ]

b. Stop WASTE MONITOR TANK PUMP 010 using 1-HS-1083 [ ]

54. 076A2.01 001/2/1/SWS - LOSS SWS/C/A - 3.5/NEW/R/NRC RO/TNT/DSS

The plant is at 100% power when:

* Train A standby NSCW pump # 5 is tagged out.

* Train A NSCW pump # 3 trips

Which **ONE** of the following conditions would **CORRECTLY** describe when use of NSCW single pump operations would be appropriate per direction of ~~18022~~-C, Loss of Nuclear Service Cooling Water (NSCW)? 18021

A. The other NSCW train is available, plant to be maintained at 100%. ← also correct?

B. The other NSCW train is **NOT** available, plant to be maintained at 100%.

C. The other NSCW train is available, after a reactor trip has been performed.

D.✓ The other NSCW train is **NOT** available, after a reactor trip has been performed..

Licensee to rework - right now looks like two correct answers

✓

54. 076A2.01 001/2/1/SWS - LOSS SWS/C/A - 3.5/NEW/R/NRC RO/TNT/DSS

K/A

076 Service Water System

A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.

Loss of SWS

K/A MATCH ANALYSIS

Question gives a plausible scenario where an NSCW pump trips with the standby pump tagged out. Candidate must choose what conditions would warrent placing the train in single pump operations. At power, per AOP direction of 18022, the only time single pump operations would occur is if no train available and following a reactor trip.

ANSWER / DISTRACTOR ANALYSIS

A. Incorect. Reactor trip and single pump operations directed if other train unavailable.

B. Incorrect. Reactor trip and single pump operations directed if other train unavailable.

C. Incorrect. Reactor trip and single pump operations directed if other train unavailable.

D. Correct. Single pump ops directed if other train unavailable, after performing a reactor trip.

REFERENCES

SOP-13150-1/2, "Nuclear Service Cooling Water System" section 4.4.10, "Single Pump Operations - Abnormal"

AOP-18021-C, "Loss of Nuclear Service Cooling Water" RNO for step # 2

VEGP learning objectives:

LO-PP-06101-14 Describe the operation of the NSCW system for single pump operations during refueling outages and in response to a loss of NSCW per the AOP.

LO-LP-60316-04 Given the entire AOP, describe:

a. Purpose of selected steps
b. How and why the step is performed
c. Expected response of the plant/parameter(s) for the step

LO-LP-60317-02 Describe the operator action(s) required if NSCW is lost and neither

NSCW Train can be placed into operation.

CAUTION

> **THIS SECTION IS ONLY PERFORMED WHEN NEITHER TRAIN OF NSCW IS CAPABLE OF SUPPORTING TWO PUMP OPERATION AND WHEN DIRECTED BY 18021-C.** This section will establish single pump operation to support ESF equipment when normal two-pump operation of Train A NSCW is prohibited.

NOTES

> - While low flow alarms may be present during single pump operation, the following actions result in adequate flow to all pump motor coolers and lube oil coolers supplied by NSCW.
> - With an SI signal present the CNMT Cooler isolation valves cannot be closed from the control room. Sending an operator to the appropriate penetration rooms now will minimize the time needed to establish single pump operation.

4.4.10 **NSCW Train A Single Pump Operation (Abnormal)**

NOTE

> Checklist 8 should be used for Independent Verification in this section.

4.4.10.1 If an SI signal is present and cannot be reset, dispatch operators to the penetration rooms in preparation for isolating NSCW to one pair of CNMT Coolers, [ ]

CNMT Coolers 001 & 002 - 1-HV-1806 (AB-B08)

CNMT Coolers 005 & 006 - 1-HV-1808 (AB-B08)

4.4.10.2 Unlock and throttle NSCW to the ACCW HX (or Bypass if HX is bypassed) to approximately 15% to 20% open:

ACCW HX A NSCW INLET ISO - 1-HV-11708 Key (1OP2-123) [ ]

OR

ACCW HX A Bypass - 1-HV-11709 Key (1OP2-124) [ ]

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. CHECK AFFECTED NSCW train operation:	
a. VERIFY the following: • Supply header pressure - GREATER THAN 70 PSIG. Train A: PI-1636 Train B: PI-1637 • Supply header temperature computer indication - LESS THAN 90°F. Train A: TE-1642 Train B: TE-1643 • Supply header flow - APPROXIMATELY 17,000 GPM. Train A: FI-1640B Train B: FI-1641B	a. ENSURE the opposite NSCW train in operation by initiating 13150, NUCLEAR SERVICE COOLING WATER SYSTEM. -OR- <u>IF</u> neither NSCW train can be placed in normal, two pump operation, <u>THEN</u>: • TRIP the reactor and go to 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION. • TRIP all reactor coolant pumps. • ISOLATE letdown • ATTEMPT to place one train of NSCW in single pump operation by initiating 13150, NUCLEAR SERVICE COOLING WATER SYSTEM. <u>WHEN</u> single pump NSCW operation has been established, <u>THEN</u> VERIFY RCP No. 1 seal temperatures less than 220°F, and ENSURE the train-related CCP is running and seal injection flow established per 13006, CHEMICAL AND VOLUME CONTROL SYSTEM.

55. 076G2.1.30 001/1/2/HI RCS ACT - CONTROL/MEM - 3.9/NEW/R/NRC RO/TNT/DSS

The plant has been operating at 100% with a 60 gpd SG tube leak on SG # 2.

* Subsequently, a confirmed loose part causing high RCS activity has resulted in a unit shutdown to Mode 3.

Shortly after entering Mode 3, a fire in Control Room has resulted in implementation of AOP-18038 "Operation From Remote Shutdown Panels"

* The TDAFW pump is the only available pump for feeding the SGs.

While at the Remote Shutdown Panels, Health Physics requests closing of the Loop 2 steam supply valve (HV-3019) to the TDAFW pump to limit release via TDAFW exhaust to atmosphere.

Which **ONE** of the following would be **CORRECT** regarding closing the TDAFW steam supply valve HV-3019 ?

A. valve could be closed from Remote Shutdown Panel "A".

B. valve could be closed from Remote Shutdown Panel "B".

C✓ valve could be closed from Remote Shutdown Panel "C".

D. valve would have to be locally shut manually, radiation levels permitting.

55. 076G2.1.30 001/1/2/HI RCS ACT - CONTROL/MEM - 3.9/NEW/R/NRC RO/TNT/DSS

K/A

076 High Reactor Coolant Activity

G2.1.30 Ability to locate and operate components, including local controls.

K/A MATCH ANALYSIS

Question gives a plausible scenario where a reactor trip has occurred with a SGTL on SG # 2 in progress. Control room has to be evacuated post trip to the shutdown panels. Reactor coolant activity has been determined to be high by chemistry department. TDAFW pump is only pump available, and chemistry asks to isolate the steam supply to limit release. Candidate must determine proper location to perform this evolution.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible candidate may think valve is controlled from Remote Shutdown panel A or B since 125V DC power supply is from Train A or Train B power.

B. Incorrect. Plausible candidate may think valve is controlled from Remote Shutdown panel A or B since 125V DC power supply is from Train A or Train B power.

C. Correct. Valve is operated from Train C shutdown panel.

D. Incorrect. Valve would not have to be locally shut. Can be operated from Train C Remote Shutdown Panel.

REFERENCES

18038-1/2, "Operation From Remote Shutdown Panels" attachment "A" for TDAFW Pump Operation From Shutdown Panel "C"

AOP-18034-1/2, Loss of Class 1E 125V DC Power" pages # 11 and # 45 for power supplies to HV-3009 and HV-3019.

VEGP learning objectives:

LO-LP-60326-06 Given the entire AOP, describe:

a. Purpose of selected steps
b. How and why the step is being performed
c. Expected response of the plant/parameter(s) for the step

Sheet 1 of 2

ATTACHMENT A

TURBINE DRIVEN AFW PUMP OPERATION FROM SHUTDOWN PANEL C

CAUTION:

- The TDAFW pump steam supply valve, 1-HV-5106, will not automatically open on a pump start signal after control has been transferred to the Shutdown Panel.
- If a SG fed by the MDAFW pump is also being fed from the TDAFW pump, one of the MDAFW pump's throttle valves or its miniflow valve must remain fully open to ensure minimum miniflow requirements.

A.1 STARTING TDAFW PUMP

A.1.1 TRANSFER control of the TDAFW pump steam supply isolation valves, 1-HV-3009 and 1-HV-3019 to LOCAL.

A.1.2 ENSURE at least one steam supply isolation valve is open.

A.1.3 TRANSFER control of the TDAFW Pump Room Outside Air Damper, 1-HV-12010 to LOCAL.

A.1.4 ENABLE differential pressure controller 1-PDIC-5180B by placing transfer switch 1-HS-5180 in LOCAL.

A.1.5 TRANSFER control of the TDAFW Pump AFW Throttle Valves 1-HV-5120, 1-HV-5122, 1-HV-5125 and 1-HV-5127 to LOCAL.

A.1.6 ENSURE TDAFW pump Trip/Throttle Valve 1-PV-15129 is reset by verifying the motor operator is OPEN and the valve is latched and OPEN.

A.1.7 ENSURE differential pressure controller 1-PDIC-5180B is in SPEED Control and ADJUST its CONTROL signal to desired speed.

A.1.8 START the TDAFW pump by placing 1-HS-5106C in LOCAL and opening 1-HV-5106.

A.1.9 VERIFY that pump speed and discharge pressure begin rising.

Abnormal Operating Procedures

LOSS OF CLASS 1E 125V DC POWER

PURPOSE

PRB REVIEW REQUIRED

This procedure provides operator actions to be followed in the event that power is lost to one of the 125V DC Vital Busses (1AD1, 1BD1, 1CD1, or 1DD1).

Specific instructional steps will be found in the following sections:

A. Loss of 125V DC Bus 1AD1

B. Loss of 125V DC Bus 1BD1

C. Loss of 125V DC Bus 1CD1

D. Loss of 125V DC Bus 1DD1

SYMPMTOMS

Symptoms are identified in the individual sections.

Sheet 1 of 25

ATTACHMENT A

LOSS OF 125V DC BUS 1AD1
EQUIPMENT RESPONSE DUE TO LOSS OF TRAIN A 125V DC POWER

- Main Feedwater Isolation, Bypass Feedwater Isolation, Main Feedwater Regulation, and Bypass Feedwater Regulation Valves close resulting in Feedwater Isolation.
- Main Steam Isolation and Bypass Steam Isolation Train A Valves close resulting in steamline isolation.
- Above P10, Reactor and Turbine trip occurs from loss of main feedwater.
- Below P10, Reactor trip occurs from Intermediate Range Instrumentation.
- Control Power is lost to 1AA02, 1AB04, 1AB05, and 1AB15 SWGR Breakers.
- Diesel Generator DG1A control power to Generator Control Panel PDG1 and Engine Control Panel PDG2 is lost rendering the Diesel Generator inoperable; if running, it will fail as is with a loss of electrical protective trips, frequency, and voltage control. Due to loss of power to the Low Speed Relay, the generator space, Engine Lube Oil and Jacket Water Heaters and Lube Oil and Jacket Water Keep-Warm Pumps will come on.
- Loss of Train A DG AUTO sequencer reset.
- Power to Inverters 1AD1I1 and 1AD1I11 is lost causing 120V AC Vital Busses 1AY1A and 1AY2A to de-energize.
- Instrument Air Containment Isolation Valve 1-HV-9378 closes resulting in loss of instrument air inside Containment.
- Power To Isolation Panel 1ACQIP1 is lost rendering the annunciators in Train A inoperable.
- Pressurizer PORV 1-PV-455A fails closed.
- TDAFW Steam Supply 1-HV-3019 fails as is.

<u>NOTE</u>: Feeder Breakers must be locally controlled in the event the transfer to an alternate power supply is required. <u>IF</u> the Diesel Generator (DG1A) is not running, it may not be selected as an alternate power source.

Plausible

NOTE: HV-3009 "B" power and 3019 "A" power and vice versa,

ATTACHMENT B

LOSS OF 125V DC BUS 1BD1
EQUIPMENT RESPONSE DUE TO LOSS OF TRAIN B 125V DC POWER

- Main Feedwater Isolation, Bypass Feedwater Isolation, Main Feedwater Regulation, and Bypass Feedwater Regulation Valves close resulting in Feedwater Isolation.
- Main Steam Isolation and Bypass Steam Isolation Train B Valves close resulting in steamline isolation.
- Above P10, Reactor and Turbine trip occurs from loss of main feedwater.
- Below P10, Reactor trip occurs from Intermediate Range Instrumentation.
- Control Power is lost to 1BA03, 1BB06, 1BB07, and 1BB16 Breakers.
- Diesel Generator DG1B control power to Generator Control Panel PDG3 and Engine Control Panel PDG4 is lost rendering the Diesel Generator inoperable; if running, it will fail as is with a loss of electrical protective trips, frequency, and voltage control. Due to loss of power to the Low Speed Relay, the generator space, Engine Lube Oil and Jacket Water Heaters and Lube Oil and Jacket Water Keep-Warm Pumps will come on.
- Loss of Train B DG AUTO Sequencer Reset.
- Power to Inverters 1BD1I1 and 1BD1I12 is lost causing 120V AC Vital Busses 1BY1B and 1BY2B to de-energize.
- Instrument Air Containment Isolation Valve 1-HV-9378 closes resulting in loss of instrument air inside Containment.
- Power To Isolation Panel 1BCQIP2 is lost rendering the annunciators in Train B inoperable.
- Pressurizer PORV 1-PV-456A fails closed.
- TDAFW Steam Supply 1-HV-3009 fails as is.

NOTE: Feeder Breakers must be locally controlled in the event the transfer to an alternate power supply is required. IF the Diesel Generator (DG1B) is not running, it may not be selected as an alternate power source.

56. 078K4.03 001/2/1/AIR - LOSS COOLING/MEM - 3.1/NEW/R/NRC RO/TNT/DSS

The unit is at 100% power with all systems in normal alignment:

Which **ONE** of the following **CORRECTLY** describes the affect of a loss of cooling water to the Instrument Air Compressors ?

A.✓ Rotary air compressors trip on low TPCCW pressure.

B. Reciprocating air compressors trip on low Utility Water pressure.

C. Rotary air compressors trip on low TPCW Water pressure.

D. Reciprocating air compressors trip on low Demin Water pressure.

56. 078K4.03 001/2/1/AIR - LOSS COOLING/MEM - 3.1/NEW/R/NRC RO/TNT/DSS

K/A

078 Instrument Air System

K4.03 Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following:

Securing of SAS upon loss of cooling water.

K/A MATCH ANALYSIS

Question asks which air compressors trip on a loss of cooling water. Candidate has a choice between TPCW and TPCCW. Rotary air compressors trip on low TPCCW pressure of 20 psig.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. TPCCW low pressure of 20 psig trips rotary compressors.

B. Incorrect. TPCCW low pressure of 20 psig trips rotary compressors.

C. Incorrect. TPCCW low pressure of 20 psig trips rotary compressors.

D. Incorrect. TPCCW low pressure of 20 psig trips rotary compressors.

REFERENCES

SOP-13710-1/2, "Service Air System" limitations 2.2.1 and 2.2.2

ARP-17210-1/2, ARP for PMEC Air Compressor Control Panel, windows C01 and C02

VEGP learning objectives:

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

LO-PP-02101-06 List the conditions which will trip the rotary air compressors.

2.2 **LIMITATIONS**

2.2.1 Rotary Air Compressors 1-2401-C4-501 and 502 will trip on:

- High discharge temperature of 235°F,
- High discharge pressure of 135 psig,
- Low Turbine Plant Closed Cooling Water (TPCCW) pressure of 20 psig,
- Emergency Stop handswitch.

2.2.2 Reciprocating Air Compressors 1-2401-C4-503 and A-2401-C4-504 will trip on:

- Low-low oil pressure of 12 psig,
- High-high lube oil temperature of 190°F,
- High-high discharge temperature of 400°F,
- High-high discharge pressure of 130 psig,
- High-high Intercooler Condenser level equal to 1/2 full,
- Emergency Stop handswitch.

3.0 **PREREQUISITES OR INITIAL CONDITIONS**

If temporary cooling is being supplied, verify the applicable portions of the Turbine Plant Closed Loop Cooling Water System or Utility Water System is in service. [ ]

Plausibe distractor
Not used "Normally"

WINDOW C01

ORIGIN	SETPOINT	
1-PSL-17273	20 psig	1.2401.C4.501 NO. 1 COMPRESSOR LO WATER PRESS.

1.0 **PROBABLE CAUSE**

1. Coolant Heat Exchanger 1-2401-C4-501-E01 or Aftercooler 1-2401-E4-501 leakage.
2. Loss of Turbine Plant Closed Cooling Water.

2.0 **AUTOMATIC ACTIONS**

Compressor Motor trips.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. Ensure standby compressor starts as required to maintain system pressure.
2. Ensure Compressor Motor trips using AIR COMPRESSOR 1 ROTARY 1-HS-19338 on the QMCB.
3. Check Turbine Plant Closed Cooling Water System pressure.
4. Check Coolant Heat Exchanger for possible leaks.
5. Check Aftercooler for possible leaks.
6. If equipment failure is indicated, initiate maintenance as required.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB154-2, 1X4DB175-1, 1X3D-BH-R50A, 1X5DN077-2, CX5DT1101-72B

WINDOW C02

ORIGIN	SETPOINT	
1-PSL-17274	20 psig	1.2401.C4.502 NO. 2 COMPRESSOR LO WATER PRESS.

1.0 **PROBABLE CAUSE**

1. Coolant Heat Exchanger 1-2401-C4-502-E01 or Aftercooler 1-2401-E4-502 leakage.
2. Loss of Turbine Plant Closed Cooling Water.

2.0 **AUTOMATIC ACTIONS**

Compressor Motor trips.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. Ensure standby compressor starts as required to maintain system pressure.
2. Ensure Compressor Motor trips using AIR COMPRESSOR 2 ROTARY 1-HS-9383 on the QMCB.
3. Check Turbine Plant Closed Cooling Water System pressure.
4. Check Coolant Heat Exchanger for possible leaks.
5. Check Aftercooler for possible leaks.
6. If equipment failure is indicated, initiate maintenance as required.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB154-2, 1X4DB175-1, 1X3D-BH-R50B, 1X5DN077-2(-7), CX5DT1101-72B

57. 079K4.01 001/2/2/SERV AIR - XTIE IAS/C/A - 2.9/NEW/RO/NRC RO/TNT/DSS

Unit 2 has lost 2 air compressors, the running air compressor is not able to keep up with the system demand.

* Unit 2 air pressure is 92 psig and slowly lowering for both instrument and service air.

* The "Swing" compressor is currently aligned to Unit 1

* All air compressors on Unit 1 are available.

Which **ONE** of the following would be **CORRECT** regarding crosstie of the Unit 1 Instrument Air system to Unit 2 ?

A.✓ **ALLOWED**, swing compressor would be isolated from Unit 1 and aligned to Unit 2. Unit 1 ZLBs will indicate which unit the swing compressor is aligned to.

B. **NOT ALLOWED**, Unit 2 has to have 2 compressors available in order to align the swing compressor to Unit 2. The swing compessor can be operated from either units main control room.

C. **ALLOWED**, swing compressor would be isolated from Unit 1 and aligned to Unit 2. The swing compessor is now operated from the Unit 2 main control room.

D. **NOT ALLOWED**, Unit 2 has to have 2 compressors available in order to align the swing ~~compressors~~ compressor to Unit 2. Unit 1 and Unit 2 ZLBs will indicate which unit the swing compressor is aligned to.

Revise

57. 079K4.01 001/2/2/SERV AIR - XTIE IAS/C/A - 2.9/NEW/RO/NRC RO/TNT/DSS

K/A

079 Station Air System (SAS)

K4.01 Knowledge of SAS design feature(s) and/or interlocks which provide for the following:

Cross-connect with IAS.

K/A MATCH ANALYSIS

Questions asks a plausible scenario where Unit 2 is losing instrument air and the swing compressor is to be aligned to unit 2. Candidate has to determine if this is allowed and whether or not indications on QMCB available that this has occurred.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. Crosstie allowed per AOP and a ZLB available on unit 1.

B. Incorrect. Cross tie allowed. Swing compressor only operated from Unit 1.

C. Incorrect. Cross tie allowed. Swing compressor only operated from Unit 1.

D. Incorrect. Crosstie allowed per AOP and ZLB avaiable on Unit 1 only.

REFERENCES

AOP-18028-C, "Loss of Instrument Air".

VEGP learning objectives:

LO-LP-60321-11 Given the entire AOP, describe:

a. Purpose of selected steps
b. How and why the step is being performed
c. Expected response of the plant/parameter(s) for the step

A. LOSS OF INSTRUMENT AIR AT POWER

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Steps A4 and A5 provide different options for re-establishing instrument air to the affected unit.

A4. Check if control of swing compressor is set to affected unit.

IF control is set to the affected unit, THEN verify swing compressor is running.

Response Not Obtained:

A4. IF the swing compressor is not required to support unaffected unit, THEN:

a. Dispatch operator to perform for the affected unit:

UNIT 1

- Stop swing air compressor.
- Service Air Receiver 504 to Air Dryer Isolation Valve 1-2401-U4-510 (TB-A-TC11) open.
- Service Air Unit 1 to Unit 2 Header Isolation Valve 2-2401-U4-510 (TB-A-TC11) shut.
- Unit 1/Unit 2 Control Transfer Switch A-HS-19458 in UNIT 1 position.

UNIT 2

- Stop swing air compressor
- Valve 1-2401-U4-510 (TB-A-TC11) shut.
- Valve 2-2401-U4-510 (TB-A-TC11) open.
- Switch A-HS-19458 in UNIT 2 position.

b. Start swing compressor.

A. LOSS OF INSTRUMENT AIR AT POWER

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:

- UNIT 1 SERV AIR HDR TIED TO UNIT 2 annunciator C05 of ALB01 will annunciate in the Unit 1 control room when instrument air header cross-tie to Unit 1 and Unit 2 is established.
- If the pressure in the instrument air header common to both units lowers below 80 psig, the Unit 1 and Unit 2 headers should be reisolated from each other.

A5. Check air compressors running on affected unit.

IF two or more, THEN go to Step A6.

A5. Perform the following:

a. Verify all available air compressors for both units are running.

b. IF the total number of running Unit 1 and Unit 2 air compressors is 4 or more, THEN dispatch operator equipped with a radio (or use local phone) to establish an open instrument air header between Unit 1 and Unit 2 by opening the following valves:

- 1-2401-U4-510, Service Air Receiver 504 to Air Dryer Isolation Valve (TB-A-TC11)
- 2-2401-U4-510, Service Air Unit 1 to Unit 2 Header Isolation Valve (TB-A-TC11)

58. 086K5.03 001/2/2/FIRE PRT - ELC COMP/MEM - 3.1/NEW/R/NRC RO/TNT/DSS

* The unit is operating at 100% power.

* A fire breaks out in the cable spreading room.

* The fire brigade has responded and is fighting the fire.

* They report heavy fire / water damage to cable spreading room wiring is occurring.

The control room operators have referenced Annuciator Response Procedure 17103-A/B-C For the Fire Alarm Computer Table # 3 for "CR OPER ACTIONS" to obtain additional information regarding...............

A. safety hazards / protective actions the fire brigade may need to take for this zone.

B.✓ potential spurious actions/inactions and for safe shutdown operator actions.

C. additional suppression system references the fire brigade may need for this zone.

D. evacuation route if evacuation to the remote shutdown panels is required.

58. 086K5.03 001/2/2/FIRE PRT - ELC COMP/MEM - 3.1/NEW/R/NRC RO/TNT/DSS

K/A

086 Fire Protection System

K5.03 Knowledge of the operational implication of the following concepts as they apply to the Fire Protection System.

Effects of water spray on electrical components

K/A MATCH ANALYSIS

Question presents a scenario where a fire is occurring in the cable spreading room with major fire and water damage to electrical wiring. This could lead to spurious actions/inactions of components. ARP-17103-A/B-C lists CR operator actions to take in response to spurious actions and for safe shutdown operator actions.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Addresses operator response to spurious actions / inactions. Fire protection procedures do list hazards but not this ARP table.

B. Correct. Addresses operator responses to spurious actions / inactions.

C. Incorrect. Detection and suppression system references are on the sheet that directs operators to table 3. Table 3 is for spurious actions response however.

D. Incorrect. Addresses operator response to spurious actions / inactions. Fire protection procedures do list safe shutdown evacuation routes but not this ARP table. Cable spreading rooms are adjacent to Shutdown Panel B and just above and just below the main control room.

REFERENCES

92005-C, "Fire Response Procedure" step 3.5.4.4

ARP-17103A/B-C, Annunciator Response Procedures for Fire Alarm Computer, Tables # 1 and # 3 in particular.

VEGP learning objectives:

Not applicable

3.5.4 **Actions After Notification**

3.5.4.1 After notification to the Control Room, the Responder may attempt to extinguish the fire provided he/she is confident, based on training received, that he/she can do so safely.

3.5.4.2 If the individual is a qualified person in Fire Brigade Training, he should perform the following duties in order:

a. Evaluate conditions;

b. Call the Control Room and report confirmation;

c. Evaluate the use of a fire extinguisher and if safety permits, attempt extinguishment;

d. If conditions allow, check for exposures;

e. Deploy hoses (minimum of 2) and CHARGE hose lines;

f. Unless directed otherwise by the Control Room, stand by the area to report to the Fire Team Captain what was done in preparation for fire fighting.

3.5.4.3 If the alarm is from a location outside the protected area, the SM/SS shall contact the Security Department.

3.5.4.4 Refer to Annunciator Response Procedure 17103A/B-C, "Annunciator Response Procedure For Fire Alarm Computer", to obtain information regarding potential spurious actions/inactions and for safe shutdown operator actions.

NOTE

Anyone not directly involved in the fire fighting activities and needing information from the FTC should go through his designee if available or Security.

3.6 **SECURITY PERSONNEL**

3.6.1 Upon receiving notification from SM/SS or page announcement, the Security Department will dispatch an officer to the appropriate Fire Brigade Dress Out Locker unless directed to another location by the Control Room.

3.6.2 The Security Department officer shall make contact with the Fire Team Captain or a member of the Fire Team to obtain information and instructions for crowd control.

TABLE 1 - UNIT 1 AND COMMON

DETECTION AND SUPPRESSION SYSTEMS REFERENCE INFO FOR FIRE ALARM RESPONSE

FIRE ZONE	FIRE ZONE DESCRIPTION	LZIP NO	LZIP LOC BLDG-LVL-RM	LDC NO	LDC LOC	LSIP NO	LSIP LOC BLDG-LVL-RM	SUPPR SYS NO	SUPPR SYS TYPE	PREPLAN NO.	AVAIL S/D TRN	CR OPER ACTIONS (SEE TABLE 3)
53	AUX BLDG - LEVELS 2 & 3 R-212, LVL 2 HVAC EQUIP RM	1S18	AUX-2-02			1F23	AUX-2-12	041	PREACTION	92753-1	A	NONE
54	AUX BLDG - LEVEL 2 R-203 CCW HX RM TRN A	1S17	AUX-2-12			1F23	AUX-2-12	085	PREACTION	92754-1	B	5a, 5d, 6a, 7, 12a, 15, 18a, 18b, 23
55	AUX BLDG - LEVEL 2, 3 R-201, 202, 302, CCW HX RM TRN B	1S17	AUX-2-12			1F23	AUX-2-12	083	PREACTION	92755-1	A	9,25
56 A	CNTL BLDG - LEVEL B R-B48, LVL B SWGR & BATT TRN D RM CH 4	1S25	CB-B-70							92756A-1	A	NONE
56 B	CNTL BLDG - LEVEL B R-B44, LVL B SWGR & BATT TRN D RM CH 4	1S25	CB-B-70			1F28	CB-B-42	106	MAN DELUGE	92756B-1	A	NONE
57B	CNTL BLDG - LEVEL B, A, 1, 2, R-B40, A66, 167, 253, LVL A MECH SHAFT RM TRN B	1S31	CB-A-58							92757B-1	A	NONE
58	CNTL BLDG - LEVEL B R-B38, LVL B NORM HVAC RM	1S23	CB-B-70							92758-1	B	1a, 1c, 4, 15, 27
59	CNTL BLDG - LEVEL B R-B75	1S21	CB-B-68							92759-1	B	1e, 2b, 17a, 28
60	CNTL BLDG - LEVEL B R-B74	1S21	CB-B-68			1F25	CB-B-70	061	PREACTION	92760-1	A	1f,2a, 4, 5b, 5c, 12b, 13a, 14, 15, 17b
61	CNTL BLDG - LEVEL B R-B78	1S21	CB-B-68			1F26	CB-B-70	062	PREACTION	92761-1	B	1a, 1c, 2b, 13a, 13b, 14, 15, 17a
62	CNTL BLDG - LEVEL B R-B65	1S22	CB-B-70			1F24	CB-B-70	060	PREACTION	92762-1	A	1f,2a, 4, 5b, 5c, 12b, 13a, 14, 15, 17b
63	CNTL BLDG - LEVEL B R-B66	1S22	CB-B-70							92763-1	A or B	NONE

TABLE 3

OPERATOR ACTIONS FOR A CONFIRMED FIRE IN A SAFETY RELATED AREA

NOTES

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

CRITERIA FOR IMPLEMENTING TABLE 3 ACTIONS

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

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

a. On QMCB, place Pressurizer PORV Block Valve HS-8000A in CLOSE. [ ]

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

a. On QMCB, place Pressurizer PORV Block Valve HS-8000B in CLOSE. [ ]

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

a. Dispatch operator to locally open 125V DC MCC breaker to close PV-0455A. [ ]

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

59. 103A1.01 001/2/1/CTMT - PRES TEMP HUM/MEM - 3.7/MODIFIED/RO/NRC RO/TNT/DSS

Given the following plant condtions:

* Unit 1 is in Mode 4 with RCS cooldown in progress.

* Containment Mini-Purge supply fan motor is tagged out for maintenance to add oil.

* A small air leak inside containment is causing a slow rise in containment pressure.

* Containment pressure is currently 1.7 psig.

In order to ensure containment pressure is maintained below Technical Specification maximum pressure, containment pressure will have to be reduced by ....

A.✓ opening mini-purge exhaust dampers to allow containment to vent.

B. opening main purge exhaust dampers to allow containment to vent.

C. opening containment sample line vent for 1RE-2562 in FHB Bldg penetration room.

D. opening POST LOCA PURGE CTB ISO VALVE 1-1508-U4-012 on Equip Bldg roof.

59. 103A1.01 001/2/1/CTMT - PRES TEMP HUM/MEM - 3.7/MODIFIED/RO/NRC RO/TNT/DSS

K/A

103 Containment System

A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including:

Containment pressure, temperature, and humidity.

K/A MATCH ANALYSIS

Question gives a plausible scenario with Ctmt mini-purge tagged out and Ctmt pressure approaching Tech Spec limits. Candidate must choose correct method to prevent exceeding the Tech Spec limits.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. Normal method for venting if containment > 0.3 psig per procedure.

B. Incorrect. Main purge shut / sealed per Tech Spec 3.6.3 and procedure.

C. Incorrect. Not allowed per 13130-1 but plausible. Farley does this per their test.

D. Incorrect. Plausible, especially if candidate doesn't know how done in "A" above.

REFERENCES

Farley August 2004 NRC RO exam question # 59

SOP-13125-1/2, "Containment Purge System" section 4.4.1 for Ctmt Pressure Relief.

SOP-13125-1/2, Limitation 2.2.5.

SOP-13130-1/2, "Post Accident Hydrogen Control"

VEGP learning objectives:

LO-PP-29101-08 Describe routine actions taken to adjust Containment pressure and temperature.

1. 103A1.01 001

Given the following plant conditions:

- Unit 1 is at 100% power
- Containment Mini-Purge supply and exhaust fans are tagged out for maintenance.
- A small air leak inside containment is causing a slow rise in containment pressure.
- Containment pressure is currently 1.7 psig.

In order to ensure containment pressure is maintained below Technical Specification maximum pressure, containment pressure will have to be reduced.
Containment should be vented...

A.✓ using the Post-LOCA vent system. NOT at Vogtle.

B. into the piping penetration room using Containment Purge Exhaust Bleed MOV-2788A.

C. using a containment sample point lineup to a filtered poly bottle in the 139' electrical penetration room.

D. into the piping penetration room using Containment Mini-Purge exhaust manual fill line bleed valve, V285B and a filtered poly bottle.

Farley Aug 2004 #59

Goes with Q#59

CAUTION

The pressure relief should not be initiated until the current approved Containment Gaseous Release Permit is obtained.

NOTES

- Heater will not energize until CTB Mini-Purge Fan is started and pressure in Filter Housing is negative.
- Annunciator ALB-52-B07, CNMT PURGE EXH FLTR HI MSTR alarm may come in when pressure relief is initiated. It should clear after approximately 5 minutes of CTB Mini-Purge Fan operation.
- Containment Pressure should be maintained within limits of 2.2.3.

4.4.1.6 If containment pressure is greater than +0.3 psig and less than or equal to +4.4 psig, initiate pressure relief to zero ±0.1 psig as follows:

NOTE

The following pressure relief is via Flow Orifice 1-FO-12593.

a. Ensure CTB MINI-PURGE EXH DMPR 1-HV-12592 is closed (C34), [ ]

b. Open CTB MINI PURGE EXH ORC ISO VLV-MINI 1-HV-2629B (B34), [ ]

c. Open CTB NORM PURGE EXH IRC ISO VLV-MINI 1-HV-2628B (A34), [ ]

d. Log the Initial Containment pressure, START TIME, and DATE on the Containment Gaseous Release Permit. [ ]

4.4.1.7 Notify Chemistry that pressure relief has commenced. RECORD the name of the person contacted in the Unit Control Log. [ ]

4.4.1.8 When CTB pressure drops below +0.3 psig, perform the following:

a. Open CTB MINI-PURGE EXH DMPR 1-HV-12592 (C34), [ ]

b. Start the CTB MINI-PURGE EXH FAN using 1-HS-2631B (D34), [ ]

c. Place the CTB MINI-PURGE EXH DMPR 1-HS-12592 in AUTO (C34). [ ]

2.1.5 If a valid Alert is received on 1RE-2565C while a Containment Purge or Pressure Relief is ongoing, the release/vent must be terminated immediately.

2.2 **LIMITATIONS**

2.2.1 The ODCM Section 3.1.1 Table 3-1 specifies Plant Vent Radiation Monitor operability requirements.

2.2.2 Technical Specification LCO 3.6.4 requires containment pressure to be maintained between -0.3 psig and +1.8 psig in Modes 1, 2, 3 and 4.

2.2.3 Containment pressure should be maintained between -0.1 psig and 1.0 psig. Before containment pressure reaches 1.0, initiate pressure relief. If containment pressure is less than -0.1 psig, raise pressure.

2.2.4 When monitoring and changing containment pressure under this procedure, monitor containment pressure using 1-PI-10945 (QHVC) or P-9871 (plant computer point). These are the only containment pressure instruments that will indicate a negative pressure.

2.2.5 Technical Specification LCO 3.6.3 applies in Modes 1, 2, 3 and 4 and requires the following surveillance requirements:

a. Each Main (24") Purge Supply and Exhaust Valve (1-HV-2626A, 1-HV-2627A and 1-HV-2628A, 1-HV-2629A) shall be closed and sealed closed,

b. Each Mini (14") Purge Supply and Exhaust Valve (1-HV-2626B, 1-HV-2627B and 1-HV-2628B, 1-HV-2629B) shall be operable,

c. The Mini (14") Purge Valves shall be maintained closed except when in the opinion of the Unit Shift Supervisor or Shift Superintendent they need to be opened for pressure control, for ALARA and respirable air quality considerations for personnel entry and for surveillance and maintenance testing that require the valves to be open.

2.2.6 For ALARA and respirable air quality, the Mini-Purge System should be placed in service approximately 48 hours prior to planned containment entries. After work is complete and all personnel have exited containment, the Mini-Purge System should be shut down.

2.2.7 Technical Specification LCO 3.6.3 applies in Modes 1, 2, 3 and 4 and requires each Containment Isolation Valve be operable.

4.4.2.6 Monitor containment pressure 1-PI-0934, 1-PI-0935, 1-PI-0936, and 1-PI-0937. If pressure rises to 40 psig or to the value specified by the Emergency Director, terminate dilution per Step 4.4.2.7:

4.4.2.7 When containment hydrogen concentration falls to 3.5%, terminate dilution as follows:

a. Close SERVICE AIR CNMT HDR ISOL 1-HV-9385 using either 1-HS-9385A or 1-HS-9385B on Control Room Panel QPCP, [ ]

b. Verify closed both Service Air Containment Post-LOCA Purge Valves using their Control Switches on Panel QPCP:

(1) 1-HV-9380A, [ ]

(2) 1-HV-9380B. [ ]

4.4.2.8 Periodically monitor containment hydrogen concentration and Repeat this section as required to maintain the concentration below 4.0%. [ ]

4.4.3 Post-LOCA Containment Hydrogen Purge System Operation

CAUTIONS

- **The Post-LOCA Containment Hydrogen Purge System is to be operated <u>ONLY</u> if the containment hydrogen concentration cannot be maintained below 4% by other means.**
- The Post-LOCA Containment Hydrogen Purge System is designed to operate with a maximum pressure of 3 psi downstream of CNMT POST LOCA PURGE EXH DUCT CONTROL VLV 1-FV-2693.

NOTE

If plant conditions warrant, the Emergency Director may waive the Gaseous Release Permit requirement.

4.4.3.1 Initiate a Gaseous Release Permit. [ ]

4.4.3.2 Verify containment atmosphere is sampled and analyzed. [ ]

4.4.3.3 Verify the Service Air System is operating. [ ]

4.4.3.4 Verify compliance with the ODCM Section 3.1.1 Table 3-1 for the gaseous effluent monitoring requirements. [ ]

4.4.3.5 Verify the Auxiliary Building Heating Ventilation And Air Conditioning System is operating. [ ]

4.4.3.6 Place disconnect switch at local Heater Control Panel 1-1508-N7-001-H01 to on. [ ]

4.4.3.7 Push RESET button at local Heater Control Panel 1-1508-N7-001-H01 and verify that reset red light is ON. [ ]

4.4.3.8 Due to high radiation area potential, verify Containment Inside Isolation Valves 1-HV-2624A and 1-HV-2624B are closed and remain closed during the performance of the next step and until personnel have exited the area. [ ]

4.4.3.9 Unlock and open POST LOCA PURGE CTB ISO VALVE 1-1508-U4-012. [ ]

4.4.3.10 Verify all conditions of the Gaseous Release Permit that must be satisfied prior to the release are met, unless the permit requirement has been waived by the Emergency Director. [ ]

NOTE

> The Containment Ventilation Isolation Train A signal must be reset to open 1-HV-2624A. The Train B signal must be reset to open 1-HV-2624B.

4.4.3.11 Open one CTB POST LOCA PURGE EXH IRC ISO VLV using its Control Switch on Main Control Room Panel QHVC:

~~a.~~ 1-HV-2624A, [ ]

OR

~~b.~~ 1-HV-2624B. [ ]

4.4.3.12 Place CNMT POST LOCA PURGE EXH DUCT CONTROL VLV 1-HS-2693 to the MOD position to initiate containment venting. [ ]

4.4.3.13 Verify Post-LOCA Purge flow rises to between 450 and 500 standard cubic feet per minute using 1-UI-2693B. [ ]

4.4.3.14 Monitor 1-UI-2693B, plant vent stack flow (using IPC Computer point F5106 or F6417), and vent stack radiation. VERIFY compliance with the Gaseous Release Permit, if required. [ ]

60. 103G2.4.31 001/2/1/CTMT - ANNUNC PROC/C/A - 3.3/NEW/RO/NRC RO/TNT/DSS

The unit is at 100% when the following annunciators / indications occur:

* ALB01, window F06 for CNMT HI MSTR

* ALB62, window E05 for CNMT CLR COND LEAK

* Containment South Sump level is slowly rising.

* Containment relative humidity, pressure, and temperatures are slowly rising.

* No other alarms are present

Which **ONE** of the following is **CORRECT** regarding annunciator / indications and actions to take ?

A. Insufficient Containment Cooling Units running, start an additional pair of coolers per 13120, "Containment Building Cooling Systems".

B. Service water leak from Containment Coolers, initiate 11121-C for a Containment Cooler Condensate Collection Calculation.

C. Reactor Coolant System leaks in vicinity of loops 1 and 4, initiate OSP-14905 for an RCS Leak Rate Calculation, enter AOP-18004 for RCS leakage.

D.✓ Steam leak or feedwater leak, continue monitoring containment pressure and temperature, enter AOP-18009 for Secondary Leakage.

60. 103G2.4.31 001/2/1/CTMT - ANNUNC PROC/C/A - 3.3/NEW/RO/NRC RO/TNT/DSS

K/A

103 Containment System

G2.4.31 Knowledge of annunciators, alarms, and indications, and use of the response instructions.

K/A MATCH ANALYSIS

Question gives a plausible scenario with.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Secondary leakage indicated by the various parameters and no radiation. Plausible actions directed from ALB01, window E06.

B. Incorrect. Secondary leakage indicated by the various parameters and no radiation. Plausible, humidity and temperature rising rules this out.

C. Incorrect. Secondary leakage indicated by the various parameters and no radiation. Plausible, no radiation rulles out primary leak. Sump is in loop 1 and 4 vicinity.

D. Correct. No radiation, sumps levels, humidity, pressure, temperature and indicate a secondary coolant leak.

REFERENCES

ALB01, window E06 for CNMT HI TEMP, not in but action to start coolers listed.

ALB01, window F06 for CNMT HI MSTR.

ALB62, window E05, CNMT CLR COND LEAK.

VEGP learning objectives:

LO-LP-60308-04 Discuss the parameters that distinguish primary coolant leakage from secondary coolant leakage.

WINDOW E06

CNMT
HI TEMP

ORIGIN	SETPOINT
1-TSH-2563	120°F
1-TSH-2612	
1-TSH-2613	

1.0 PROBABLE CAUSE

Insufficient number of Containment Building Cooling Units operating.

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

NONE

4.0 SUBSEQUENT OPERATOR ACTIONS

1. Start an additional pair of Containment Cooling Units or a Containment Auxiliary Cooling Unit per 13120-1, "Containment Building Cooling Systems".

2. Verify Nuclear Service Cooling Water flow to coolers, and if necessary, dispatch an operator to inspect the Containment Heat Removal System.

3. Refer to Technical Specification LCO 3.6.5 and 3.6.6.

4. If equipment failure is indicated, initiate maintenance as required.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB212, CX5DT101-66, CX5DT101-71

WINDOW F06

ORIGIN	SETPOINT
1-MTSH-2564 1-MTSH-2614 1-MTSH-2615	85% RH

CNMT
HI MSTR

1.0 PROBABLE CAUSE

Reactor Coolant System, Steam System, Feedwater System, or other moisture source leaking in containment.

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

NONE

4.0 SUBSEQUENT OPERATOR ACTIONS

1. Check Radiation Monitors for possible increase indicating Reactor Coolant System leakage.
2. Monitor Containment pressure and temperature for trends indicating system leakage.
3. Check other Reactor Coolant, Steam, and Feedwater System parameters for indications of source of leakage.
4. Refer to the appropriate AOP:
   a. 18004-C, "Reactor Coolant System Leakage",
   b. 18008-C, "Secondary Coolant Leakage".
5. If equipment failure is indicated, initiate maintenance as required.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF PROCEDURE TEXT

REFERENCES: 1X4DB212, CX5DT101-37C

WINDOW E05

ORIGIN	SETPOINT
1-LIS-17090	8%
1-LIS-17094	8%

CNMT CLR
COND LEAK

1.0 PROBABLE CAUSE

NOTE

Numbers in parenthesis refer to Train B.

1.0 PROBABLE CAUSE

1. High containment moisture.
2. CTB CLG UNIT TO CNMT SUMP SOUTH (NORTH) 1-HV-17091(1-HV-17095) closed.
3. Reactor Coolant System (RCS) or secondary system leakage inside containment.
4. Shifting of Containment Coolers.

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

NONE

4.0 SUBSEQUENT OPERATOR ACTIONS

1. CHECK 1-HV-17091 and 1-HV-17095 open.

2. MONITOR Containment Sump level, containment relative humidity, Radionuclide Monitors and the condensate measuring system.

   a. If sump level is increasing with humidity stable and condensate level stable, a low temperature leak is indicated,
   b. If condensate is increasing with a stable humidity and no increased radionuclides are detected, a service water leak from the Containment Coolers is indicated,
   c. If condensate and humidity are increasing, an RCS leak can be distinguished from a secondary leak due to increasing radionuclides.

3. If blockage of the cooler drain line is suspected then flush the line as follows:

   NOTE

   The standpipe may require multiple flushes to clear the alarm.

   a. CLOSE the associated standpipe drain valve 1-HS-17091 or 1-HS-17095 to fill the standpipe.
   b. When the standpipe is full, OPEN the associated standpipe drain valve 1-HS-17091 or 1-HS-17095.

4. To quantify the leak rate perform 11121-C, "Containment Coolers Condensate Collection Calculation".

5. If RCS leak is suspected, PERFORM 14905-1, "RCS Leakage Calculation (Inventory Balance)".

6. If necessary, DISPATCH an operator to containment to check for leaks.

61. G2.1.19 001/3//CONDUCT - CMPTR INFO/C/A - 3.0/NEW/RO/NRC RO/TNT/DSS

After a system perturbation, the RO checks the IPC computer to determine system status for a standby pump and its discharge valve and observes the following:

The pump indicates BLINKING HOLLOW RED.
The discharge valve indicates BLINKING SOLID GREEN

Which **ONE** of the following **CORRECTLY** describes the status of the components ?

A.✓ Discharge valve shut, pump is running and pumping against a shutoff head.

B. Discharge valve open, pump should be running but is tripped.

C. Discharge valve open, pump is running and could be approaching runout condition.

D. Discharge valve shut, pump should be running but is tripped.

61. G2.1.19 001/3//CONDUCT - CMPTR INFO/C/A - 3.0/NEW/RO/NRC RO/TNT/DSS

K/A

G2.1.19 Ability to use plant computer to obtain and evaluate parametric information on a system or component status.

K/A MATCH ANALYSIS

Question gives IPC computer status on a system which just underwent a perturbation. Candidate has to determine system status from given IPC computer information. BLINKING HOLLOW RED indicates a piece of equipment that is NOT in its normally de-energized state. Such as a standby pump which has started. BLINKING SOLID GREEN would indicate a piece of equipment that is not in its normally energized state. Such as a normally open valve that is closed.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. This discharge valve is shut and should normally be open. Pump is a standby pump which has auto started. With no discharge flow path pump could be at shutoff head.

B. Incorrect. Discharge valve is shut and pump is running.

C. Incorrect. Discharge valve is shut and pump is running.

D. Incorrect. Discharge valve status is correct but pump is running.

REFERENCES

SOP-13505-1/2, "Integrated Plant Computer".

VEGP learning objectives:

LO-LP-05210-06 State the indications available to Control Room Operators that the IPC is suspect or inoperable.

4.2.9.2 Symbols Color Convention:

a. HOLLOW RED - Symbols for equipment that is in its normally energized state. Example a normally running pump that is in service or a normally open valve which is open,

b. BLINKING HOLLOW RED - Symbol for equipment that is not in its normally de-energized state. Example a standby pump that is running or a normally closed valve that is open,

c. SOLID GREEN - Symbols for equipment that is in its normally de-energized state. Example a standby pump that is not running, or a normally closed valve which is closed,

d. BLINKING SOLID GREEN - Symbol for equipment that is not in its normally energized state. Example a non-running pump that is normally in service or a normally open valve which is closed,

e. SOLID MAGENTA - Indicates either a bad or questionable condition. Example the point ID for a pump has been removed from service, or a valve with limit switch input of a simultaneous open and closed position.

4.2.9.3 All Displays Conform to the following Color Convention:

a. Operator input pages, including pop-up menus have a blue background. Text will usually be displayed in white. Where operator input pages contain Point ID information shown in the Point ID quality color, the background is black to improve readability,

b. Display pages with dynamic values (excluding the top line information) have a black background,

c. The top line on all displays, whether dynamic or operator input, have a black background,

d. Messages will usually be displayed on the bottom of the screen in green or white.

62. G2.1.23 001/3/1/SYS INTG PLNT OPS/C/A - 3.9/BANK/RO/NRC RO/TNT/DSS

While performing a procedure, the Reactor Operator comes to a step which states:

* "Request Chemistry to sample for boron concentration"

The RO believes the step is **NOT** essential to achieving the purpose for which the procedure is being used and that omission of the step does not result in ommision of required work, violate a committment or intent of the procedure or create an unsafe plant condition. However, it is **NOT** obvious in the context of the procedure why he would N/A the step.

Which **ONE** of the following is the **MINIMUM** requirement(s) that must be met to allow marking the step "N/A" ?

A. Mark step N/A prior to performance, continue with the procedure.

B. Obtain supervisor approval, mark step N/A, continue with procedure.

C. Mark step N/A prior to performance, provide justification, continue with the procedure.

D.✓ Obtain supervisor approval, mark step N/A, provide justification, continue with procedure.

62. G2.1.23 001/3/1/SYS INTG PLNT OPS/C/A - 3.9/BANK/RO/NRC RO/TNT/DSS

K/A

G2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.

K/A MATCH ANALYSIS

Question presents a situation where an operator does not feel a step needs to be performed but it is not clear in the procedure why he would not perform the step. Candidate must choose correct action to take regarding N/A of the step.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Supervisor approval and justification required.

B. Incorrect. Supervisor approval and justification required.

C. Incorrect. Supervisor approval and justification required.

D. Correct. Supervisor approval and justification required.

REFERENCES

00054-C, "Rules For Performing Procedures"

Harris February 2004 NRC RO examination question # 13

VEGP learning objectives:

LO-LP-63054-01 Describe the following as they relate to procedural compliance:

a. proper usage of "N/As"

4.1.3.2 WHEN an activity allows Information Use of a procedure, the user(s):

a. MUST review the procedure often enough to:

- Be familiar with the requirements of the document.
- Ensure that the activity is being performed correctly or the information gained is utilized correctly.
- Ensure that the effects of procedure revisions have not been overlooked.

b. Is not required to keep the procedure in the area of the work activity.

4.1.3.3 Each person using an Information Use procedure is responsible for ensuring that actions performed or information utilized complies with the requirements of the procedure.

4.2 CLARIFICATIONS

4.2.1 Use of N/A

4.2.1.1 Procedures shall be completed consistent with the work performed. Sections of a procedure that are not required to be performed do not have to be completely filled-in (i.e., each space need not be marked N/A).

4.2.1.2 Unless the justification to N/A a section or step is obvious in the context of the procedure, the user should provide a justification for the N/A. If it is not obvious, supervisor approval is required for use of N/A and justification will need to be provided.

4.2.1.3 Prior to marking a step N/A, the user will ensure the non-performance of the step will not:

a. Result in omission of required work,

b. Violate a commitment or the intent of the procedure, or

c. Create an unsafe plant condition.

4.2.1.4 Normally, procedural steps are marked N/A when:

a. They are not applicable to the type of equipment, scope, or conditions under which the activity is performed,

b. When a procedure provides multiple actions (or a single action on many different possible items) to be completed, or

c. Allows the user alternate methods and allows the user to select one.

QUESTION: 13

While performing an Operating Procedure, the Reactor Operator comes to a step which states:

> "Request Chemistry to sample the RHT for boron concentration."

The Reactor Operator believes the step is **NOT** essential to achieving the purpose for which the procedure is being used and that the omission of the step does **NOT** violate the precautions and limitations of the Operating Procedure.

Which of the following is the **MINIMUM** requirement(s) that must be met to allow marking the step "N/A"?

a. • Step must be initialed by the Reactor Operator prior *to* performance

b. • Step must be initialed by the Reactor Operator prior to performance
   • A written explanation of why the step is N/A must be provided in the Comments section of the procedure

c. • Step must be initialed by the SCO prior to performance

d. • Step must be initialed by the SCO prior to performance
   • A written explanation **of** why the step is N/A must be provided in the Comments section of the procedure

ANSWER:

d. • Step must be initialed by the SCO prior to performance
   • A written explanation of why the step is N/A must be provided in the Comments section **of** the procedure

Goes with question #62

63. G2.1.25 001/3/1/REF GRPHS TBLES ETC/C/A - 2.8/MODIFIED/R0/NRC RO/TNT/DSS

The plant is operating at 100% power with the following conditions:

TIME	OUTSIDE AIR TEMPERATURE	CIRC WATER SYSTEM TEMP
1400	40 degrees F	62 degrees F
1800	25 degrees F	59 degrees F
2200	12 degrees F	56 degrees F

Using the **REFERENCE PROVIDED** which **ONE** of the following describes the **CORRECT** Cooling Tower Central Shutoff Valve alignment for these conditions ?

	1800	**2200**
A.	OPEN	OPEN
B.✓	OPEN	SHUT
C.	SHUT	OPEN
D.	SHUT	SHUT

63. G2.1.25 001/3/1/REF GRPHS TBLES ETC/C/A - 2.8/MODIFIED/R0/NRC RO/TNT/DSS

K/A

G2.1.25 Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data.

K/A MATCH ANALYSIS

Question presents a scenario where cold weather operation of the Circulating Water Cooling Tower Central Shutoff Valve position would have to be determend. Central shutoff valves would be open at start of scenario and move into the AS IS section of Figure 1 of 13724-1/2 "Circulating Water System" procedure. This means valves would remain open. As temperatute drops, valves would need to be shut.

NOTE - SOP-13724-1 should be provided to the candidate as they will need Figure 1 of this procedure to determine proper cold weather operations.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Valves should be open at 1800 and shut at 2200.

B. Correct. Valves should be open at 1800 and shut at 2200.

C. Incorrect. Valves should be open at 1800 and shut at 2200.

D. Incorrect. Valves should be open at 1800 and shut at 2200.

REFERENCES

SOP-13724-1/2, "Circulating Water System"

Harris February 2004 NRC RO examination question # 28

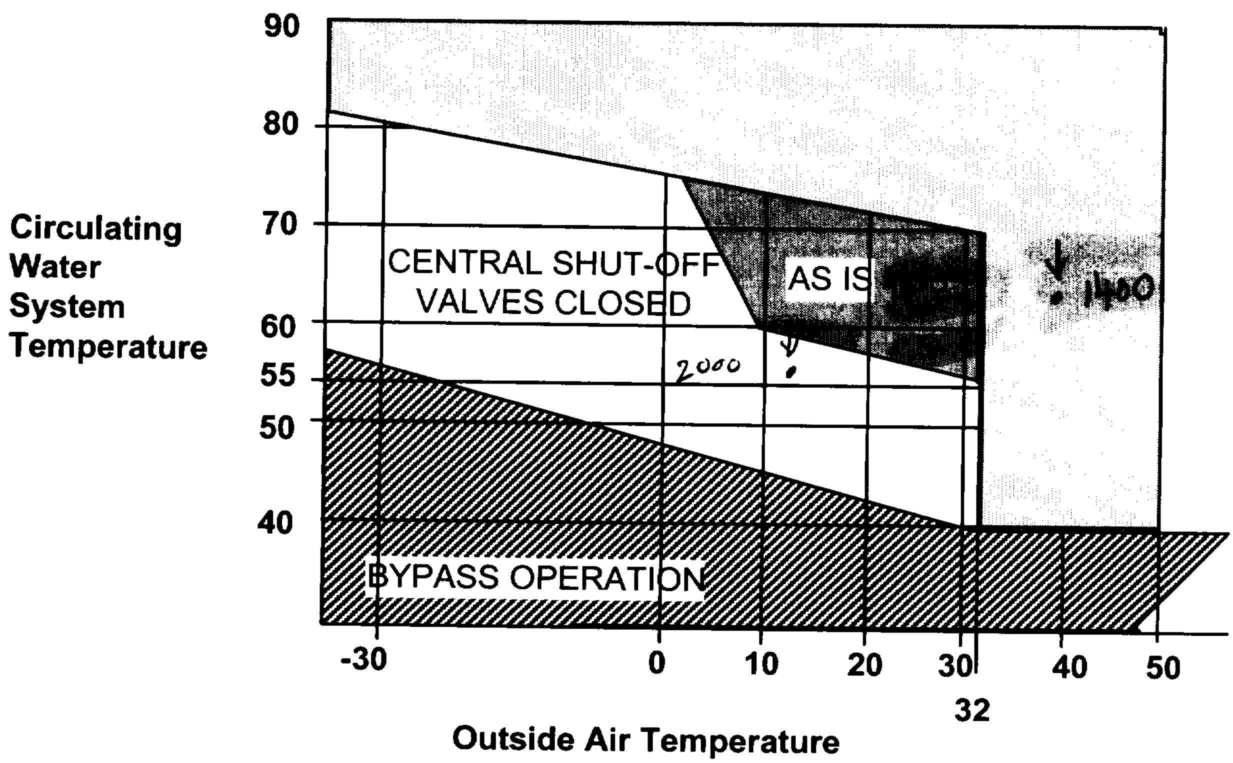
VEGP learning objectives:

LO-PP-07101-06 Describe how we prevent the accumulation of ice on the cooling tower, what the detrimental affects of ice would be, and the conditions that could lead to a rapid buildup of ice.

LO-PP-07101-07 Describe how we maintain the basin temperature above 32 degrees F during cold weather.

FIGURE 1

COLD WEATHER OPERATION OF THE CIRCULATING WATER COOLING TOWER



NOTE

Following a unit trip, with outside air temperature less than or equal to 32°F., rapid ice formation may occur in the Cooling Tower fill. As soon as plant conditions allow, proceed to BYPASS OPERATION by performing step 4.4.6.2.

Goes with Q # 63

QUESTION: 28

The plant is operating at 100%power with the following conditions:

Time	Ambient Temp	CT Basin Temp
1500	35 °F	64 °F
1900	20 °F	60 °F
2300	10 °F	58 °F

Which of the following describes the correct CT Deicing Gate Valve alignment for these conditions?

	1900	2300
a.	Full Open	Full Open
h.	Full Open	Half Open
c.	Half Open	Full Open
d.	Half Open	Half Open

ANSWER:

h.	Full Open	Half Open

Goes with question #63

64. G2.2.25 001/3/1/TS BASES - LCO & SL/MEM - 2.5/MODIFIED/RO/NRC RO/TNT/DSS

The Safety Injection system, along with other ECCS subsystems, ensures that the ECCS Acceptance Critera of 10CFR50.46 is met.

All of the following are part of this criteria **EXCEPT**:

A. The maximum fuel cladding temperature shall not exceed 2200 degrees F.

B. Changes in core geometry shall be such that the core remains amenable to cooling.

C.✓ The total oxidation of the cladding shall nowhere exceed 0.27 times the total cladding thickness before oxidation.

D. The total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount generated if all the metal in the cladding were to react.

64. G2.2.25 001/3/1/TS BASES - LCO & SL/MEM - 2.5/MODIFIED/RO/NRC RO/TNT/DSS

K/A

G2.2.25 Knowledge of the bases in technical specifications for limiting conditions for operations and safety limits.

K/A MATCH ANALYSIS

Question asks the bases for Safety Injection System and ECCS system, ensures the ECCS Acceptance Criteria of 10CFR50.46 is met and candidate must pick out the incorrect one.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. This is an acceptance criteria.

B. Incorrect. This is an acceptance criteria.

C. Correct. The acceptance criteria is 0.17 for cladding oxidation. This exceeds limit.

D. Incorrect. This is an acceptance criteria.

REFERENCES

Technical Specification 3.5.2 for ECCS and bases.

Watts Bar July 2004 NRC RO examination question # 70.

VEGP learning objectives:

LO-LP-39209-02 Given a set of the Tech Specs and the bases, determine for a specific set of plant conditions, equipment availability, and operational mode:

a. Whether any Tech Spec LCOs of section 3.5 are exceeded.

b. The required actions for all section 3.5 LCOs.

LO-LP-39209-03 Describe the bases for any given Tech Spec in section 3.5.

BASES

BACKGROUND (continued)

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power and a single failure disabling one ECCS train; and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

(continued)

G 2.2.25 001

The Safety Injection System, along with other ECCS subsystems, ensures that the ECCS Acceptance Criteria of 10CFR50.46 is met.

All of the following are part of this criteria EXCEPT:

A. The maximum fuel pellet centerline temperature shall not exceed 2000 °F.

B. Changes in core geometry shali be such that the core remains amenable to cooling.

C. The total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

D. The total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount generated if all of the metal in the cladding were to react.

The correct answer is A.

A. ***Correct*** - ECCS acceptance criteria states centerline temperature shall not exceed 2200 oF.

B. Incorrect - This is an acceptance criteria.

C. Incorrect - This is an acceptance criteria.

D. incorrect - This is an acceptance criteria.

REFERENCES:

3-OT-SYS063A p. 14
Tech Spec 3.5.1 basis

10CFR55.43.2

Knowledge of basis in technical specification for limiting conditions for operations and safety limits.

RQ-70 SRO-70

Reference:	3-OT-SYSO63	**K/A Number:**	G 2.2.25
K/A value:	2.5	**Last Used:**	
Level:	1	**Source:**	NEW
TieriGrp:	3/3	**SRO Only:**	

WATTS BAR JULY 2004 RO

Goes with Q #64

65. G2.2.27 002/3/1/RF - KNOL PROCESS/MEM - 2.6/MODIFIED/RO/NRC RO/TNT/DSS

During a refueling outage, the refueling canal lower cavity area (transfer cart area) has been filled in preparation for raising the refueling cavity area. Reactor cavity level is currently 196 ft elevation.

preferred IAW procedure

The ~~primary~~ method used to fill the refueling cavity to 217 foot elevation is ________

A. centrifugal charging pump or NCP injection into cold legs

B. refueling water purification pump into transfer canal area

RHR through the Hot Leg

C. ✓ ~~residual heat removal~~ via gravity feed from the RWST

D. safety injection pump injection into the hot legs

65. G2.2.27 002/3/1/RF - KNOL PROCESS/MEM - 2.6/MODIFIED/RO/NRC RO/TNT/DSS

K/A

G2.2.27 Knowledge of the refueling process.

K/A MATCH ANALYSIS

Question poses a plausible scenario where the lower refueling cavity is filled and asks which pump is used to finish raising cavity level to the 220 foot elevation. RHR is used.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. RHR pump would be used.

B. Incorrect. RWST purification used to fill lower cavity. RHR to raise to 220.

C. Correct. RHR pump is used to fill to 220 foot elevation.

D. Incorrect. RHR pump is used.

REFERENCES

UOP-12007-C, "Refueling Operations (Entry Into Mode 6"

SOP-13011-1/2, "Residual Heat Removal".

North Anna June 2004 NRC RO examination question # 66.

VEGP learning objectives:

LO-PP-25102-11 Describe when the different sources of makeup to the spent fuel pool would be used.

a. For evaporation
b. For leakage

INITIALS

NOTE

> The preferred path for cavity fill is via the Hot Legs, plant conditions permitting, for water clarity.

f. Coordinate with the Outage Area Coordinator to fill the Refueling Cavity to the 217 foot elevation from RWST via RHR System per SOP 13011, "Residual Heat Removal System". ________

g. Using the IPC, periodically monitor CTMT Sump Level instrumentation during Refueling Cavity fill to determine if unexpected leakage is occurring. (Rx Cavity-L7501; South Sump-L7502; North Sump-L7503) ________

NOTE

> The CTMT Personnel Airlock Interlock System should be enabled for the initial lift of the reactor vessel head, control rod unlatching, and initial removal of the upper internals.

h. When Refueling Cavity Level reaches 207 feet, coordinate with the FHS to initiate and complete Control Rod unlatching, as follows:

(1) Prior to unlatching rods, complete the applicable steps of Checklist 2. ________

NOTE

> The preferred level for performing control rod unlatching is 212'. This will result in reduced radiation exposure and provide lubrication for the latching/unlatching tool.

(2) Within 2 hours prior to unlatching Control Rods, verify the Refueling Cavity water level is at or above 207 feet, (23 feet above irradiated fuel). (LCO 3.9.7)

Refueling Cavity Level ________ ft

________/________ ________
date time

1.0 PURPOSE

This procedure provides the necessary instructions for operation of the Residual Heat Removal System (RHR). This procedure also provides instructions for filling the Reactor Coolant System (RCS) and the Refueling Cavity. Instructions are included in the following steps:

4.0 **INSTRUCTIONS**

4.1 Placing TRN-A RHR In Standby Readiness

4.2 Placing TRN-B RHR In Standby Readiness

4.3 Placing TRN-A RHRS In Service For RCS Cooldown from Standby Readiness.

4.4 Placing TRN-B RHRS In Service For RCS Cooldown from Standby Readiness.

4.5 RHR Letdown Operations

5.0 **NORMAL OPERATIONS**

5.1 Shifting From RHR TRN-A To TRN-B When RCS Temperature Is Less Than 250 Degrees Or RHR Cooldown Is Not Required

5.2 Shifting From RHR TRN-B To TRN-A When RCS Temperature Is Less Than 250 Degrees Or RHR Cooldown Is Not Required

5.3 Two Train RHR Operation During RCS Recirculation

5.4 TRN-A RHR To Fill The RCS And The Refueling Cavity

5.5 TRN-B RHR To Fill The RCS And The Refueling Cavity

5.6 Shifting From RHR TRN-A To TRN-B When RCS Temperature Is Greater Than 250 Degrees And RHR Cooldown Is Required

5.7 Shifting From RHR TRN-B To TRN-A When RCS Temperature Is Greater Than 250 Degrees And RHR Cooldown Is Required

6.0 **INFREQUENT OPERATIONS**

6.1 Filling And Venting The TRN-A RHR

6.2 Filling And Venting The TRN-B RHR

6.3 RHR Pump A Operating With Discharge Aligned Through The TRN-B Cold Legs (3&4)

6.4 RHR Pump B Operating With Discharge Aligned Through The TRN-A Cold Legs (1&2)

6.5 Operating the RHR TRN-A Pump on Mini-Flow

6.6 Operating the RHR TRN-B Pump on Mini-Flow

6.7 Partial Fill And Vent Of An RHR Pump And Associated Piping Following Maintenance

5.4 TRN-A RHR TO FILL THE RCS AND THE REFUELING CAVITY

CAUTIONS

- RCP seal injection shall be in service if the water level in the RCS is above the level of the seals in the RCP (190 feet elevation) with the RCP(s) coupled. This prevents crud infiltration into the seal chamber. RCP seal injection may be terminated with RCS level greater than 190 feet and the RCP(s) coupled provided the RCS level is maintained constant.
- Airborne activity should be monitored when filling the Refueling Cavity from RHR.

NOTES

- With the water level in the Refueling Cavity less than 217 feet 0 inches elevation (23 feet above the vessel flange), both trains of the RHR are required to be operable with one train in operation.
- Performance of this step assumes use of the RHR train that is operable but not operating.
- When the RCS level is less than 195 feet elevation or for minor level adjustments it is desirable to use an available centrifugal charging pump.
- Gravity Approximate Reactor Cavity level versus gallons:
  Cavity only approximately 11,272 gal/ft
  Cavity and fuel pool approximately 23,000 gal/ft.

5.4.1 Establish RCS or Refueling Cavity level monitoring per the applicable Unit Operating Procedure in effect. [ ]

5.4.2 Verify RHR PUMP "A" in PULL-TO-LOCK: 1-HS-0620 [ ]

5.4.3 Place Lockout Selector Switch 1-HS-8809C to ON. [ ]

5.4.4 Close the RHR PUMP A TO COLD LEG 1&2 ISO VLV 1-HV-8809A. [ ]

5.4.5 Place Lockout Selector Switch 1-HS-8809C to LOCKOUT. [ ]

5.4.6 Verify RHR HEAT EXCH BYPASS for Train A 1-FV-0618 is in MANUAL and closed. [ ]

5.4.7 Close the RHR HEAT EXCH OUTLET for Train A 1-HV-0606. [ ]

5.4.8 Close the RHR PMP A SUCTION FROM HOT LEG 1 1-HV-8701A and/or 1-HV-8701B. [ ]

5.4.9 Open RWST TO RHR PMP A SUCTION 1-HV-8812A. [ ]

NOTES

- In order to maintain water clarity, the preferred method of filling is through the HOT LEGS.
- Fill The Refueling Cavity slowly to prevent airborne activity and to maintain water clarity.
- If filling the RCS with the Reactor Vessel Cover installed, the table following Step 5.4.~~11~~ 12 should be used to determine level during fill.

5.4.10 Determine if the Reactor Cavity is to be filled through the RCS Hot Legs or the RCS Cold Legs and perform the following:

a. If filling through the RCS Hot Legs:

(1) Verify closed RHR TRAIN B TO HOT LEG CROSSOVER ISO 1-HV-8716B [ ]

(2) Open RHR TRAIN A TO HOT LEG CROSSOVER ISO 1-HV-8716A on the associated RHR train used to fill the reactor cavity, [ ]

(3) Place Lockout Selector Switch 1-HS-8840A to ON, [ ]

(4) Open the RHR TO HL ISO VLV 1-HV-8840. [ ]

b. If filling through the RCS Cold Legs:

(1) Place Lockout Selector Switch 1-HS-8809C to ON, [ ]

(2) Open the RHR PUMP A TO COLD LEG ISO VLV 1-HV-8809A. [ ]

5.4.11 Slowly throttle open the RHR HEAT EXCH OUTLET for Train A 1-HV-0606 to obtain an initial flow rate of 200 gpm to 500 gpm and perform the following:

a. Monitor water clarity, [ ]

b. At the discretion of the SS, establish a flow rate not to exceed 1000 gpm. [ ]

5.4.12 At the desired level in the cavity or from the table below, close the RHR TRN-A HEAT EXCH OUTLET 1-HV-0606. [ ]

RCS Elevation (ft)	1LI-0462 (%)	RCS Elevation (ft)	1LI-0462 (%)
201	0	211	23.0
202	2.3	212	25.3
203	4.6	213	27.6
204	6.9	214	29.9
205	9.2	215	32.1
206	11.5	216	34.4
207	13.8	217	36.7
208	16.1	218	39.0
209	18.4	219	41.3
210	20.7	220	43.6

5.4.13 Verify RCS Sightglass is placed in service per 13005-1 Checklist 2. [ ]

5.4.14 Place the RHR train in a configuration appropriate for plant conditions as follows:

a. If RHR flow was through the RCS Cold Legs:

(1) Close the RHR PUMP A TO COLD LEG 1&2 ISO VLV 1-HV-8809A, [ ]

(2) Place Lockout Selector Switch 1-HS-8809C to LOCKOUT. [ ]

b. If RHR flow was through the RCS Hot Legs:

(1) Close the RHR TO HL ISO VLV 1-HV-8840, [ ]

(2) Place Lockout Selector Switch 1-HS-8840A to LOCKOUT, [ ]

(3) Close the applicable RHR TRAIN A TO HOT LEG CROSSOVER ISO 1-HV-8716A. [ ]

5.4.15 Close RWST TO RHR PMP-A SUCTION 1-HV-8812A. [ ]

5.4.16 If the Reactor Vesssel Cover is installed, notify Maintenance that the RCS Loops have been filled and the Reactor Vessel Cover can be removed. [ ]

5.4.17 Open the RHR PMP-A SUCTION FROM HOT LEG LOOP 1 1-HV-8701A and 1-HV-8701B. [ ]

5.4.18 Place Lockout Selector Switch 1-HS-8809C to ON. [ ]

5.4.19 Open the RHR PMP-A TO COLD LEG 1 & 2 ISO VLV 1-HV-8809A; (IV REQUIRED). [ ]

5.4.20 Place Lockout Selector Switch 1-HS-8809C to LOCKOUT. [ ]

G2.2.27 001

The refueling cavity is **normally** filled with water prior to refueling operations using a __________ pump.

A✓ low-head safety injection

B. charging

C. refueling purification

D. residual heat removal

A. Correct. The cavity is filled by first gravity feeding from the RWST and then starting a low head SI pump, flowing to the hot legs, and overflowing the vessel into the cavity.

B. Incorrect. A charging pump is not used for this purpose as the flow rate would be too low. A candidate could choose this answer because charging pumps are the normal makeup method for the RCS.

C. Incorrect. The RP system is used to fill the transfer canal from the RWST (by backflowing through the canal drain line) and also to clean up the water in the cavity. It can also be used to pump the cavity back to the RWST. A candidate could choose this answer based on RP pumps being used for many refueling tasks.

D. Incorrect. The RHR system can be used to pump the cavity back to the RWST, although this does not clean up the water like using the RP system does. A candidate could choose this answer by confusing the methods for emptying the cavity versus filling the cavity.

Equipment Control
Knowledge of the refueling process

North Anna bank question 751

References:
Objective 9025 from study guide on Fuel Handling
1-OP-4.1, "Controlling Procedure for Refueling."

Level (RO/SRO):	RO	Tier:	3
Group:		Importance Rating:	2.6/3.5
Type (Bank/Mod/New):	BANK	Cog (Knowledge/Comp):	KNOWLEDGE
Reference (Y/N):	N	Last Exam(Y/N):	N

Goes with question # 65

66. G2.3.9 001/3/1/CTMT PURGE PROCESS/MEM - 2.5/BANK/RO/NRC RO/TNT/DSS

During a shutdown for a refueling outage, HP requests that Containment Main Purge System be placed in service "as soon as possible" to reduce dose to workers.

Which **ONE** of the following is the **EARLIEST** plant mode that will allow for placing Containment Main Purge in service ?

A. Entry into MODE 3 Hot Standby

B. Entry into MODE 4 Hot Shutdown

C.✓ Entry into MODE 5 Cold Shutdown

D. Entry into MODE 6 Refueling

66. G2.3.9 001/3/1/CTMT PURGE PROCESS/MEM - 2.5/BANK/RO/NRC RO/TNT/DSS

K/A

G2.3.9 Knowledge of the process for performing a containment purge.

K/A MATCH ANALYSIS

Question poses a scenario where HP requests Containment Main Purge be placed into service. Candidate must determine the correct Mode to allow.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Mode 5 is earliest per Tech Specs and procedure.

B. Incorrect. Mode 5 is earliest per Tech Specs and procedure.

C. Correct. Mode 5 / 6 is when Main Purge may be placed in service. Mode 5 earliest.

D. Incorrect. Mode 5 is earliest per Tech Specs and procedure.

REFERENCES

Technical Specifications 3.6.3, "Containment Isolations Valves"

SOP-13125-1/2, "Containment Purge System"

Prairie Island August 2005 NRC RO examination question # 71

VEGP learning objectives:

LO-LP-39210-02 Given a set of Tech Specs and the bases, determine for a sepcific set of plant conditions, equipment availability, and operational mode:

a. Whether any Tech Spec LCOs of section 3.6 are exceeded.

b. The required actions for all section 3.6 LCOs.

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

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

ACTIONS

--NOTES--

1. Penetration flow path(s) (except for 24 inch purge valves) may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one containment isolation valve inoperable except for purge valve leakage not within limit.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. <u>AND</u>	4 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	-----------NOTE-------------- Isolation devices in high radiation areas may be verified by use of administrative means. -------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment <u>AND</u> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
B. One or more penetration flow paths with two containment isolation valves inoperable except for purge valve leakage not within limit.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each 24 inch purge valve is sealed closed, except for one purge valve in a penetration flow path while in Condition C of this LCO.	31 days
SR 3.6.3.2	Verify each 14 inch purge valve is closed, except when the associated penetration(s) is (are) permitted to be open for purge or venting operations and purge system surveillance and maintenance testing under administrative control.	31 days
SR 3.6.3.3	-------------------------NOTE------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. --	
	Verify each containment isolation manual valve and blind flange that is located outside containment and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	31 days

(continued)

2.1.5 If a valid Alert is received on 1RE-2565C while a Containment Purge or Pressure Relief is ongoing, the release/vent must be terminated immediately.

2.2 LIMITATIONS

2.2.1 The ODCM Section 3.1.1 Table 3-1 specifies Plant Vent Radiation Monitor operability requirements.

2.2.2 Technical Specification LCO 3.6.4 requires containment pressure to be maintained between -0.3 psig and +1.8 psig in Modes 1, 2, 3 and 4.

2.2.3 Containment pressure should be maintained between -0.1 psig and 1.0 psig. Before containment pressure reaches 1.0, initiate pressure relief. If containment pressure is less than -0.1 psig, raise pressure.

2.2.4 When monitoring and changing containment pressure under this procedure, monitor containment pressure using 1-PI-10945 (QHVC) or P-9871 (plant computer point). These are the only containment pressure instruments that will indicate a negative pressure.

2.2.5 Technical Specification LCO 3.6.3 applies in Modes 1, 2, 3 and 4 and requires the following surveillance requirements:

a. **Each Main (24") Purge Supply and Exhaust Valve (1-HV-2626A, 1-HV-2627A and 1-HV-2628A, 1-HV-2629A) shall be closed and sealed closed,**

b. Each Mini (14") Purge Supply and Exhaust Valve (1-HV-2626B, 1-HV-2627B and 1-HV-2628B, 1-HV-2629B) shall be operable,

c. The Mini (14") Purge Valves shall be maintained closed except when in the opinion of the Unit Shift Supervisor or Shift Superintendent they need to be opened for pressure control, for ALARA and respirable air quality considerations for personnel entry and for surveillance and maintenance testing that require the valves to be open.

2.2.6 For ALARA and respirable air quality, the Mini-Purge System should be placed in service approximately 48 hours prior to planned containment entries. After work is complete and all personnel have exited containment, the Mini-Purge System should be shut down.

2.2.7 Technical Specification LCO 3.6.3 applies in Modes 1, 2, 3 and 4 and requires each Containment Isolation Valve be operable.

QUESTION: 071 (1.00)

You are the RO during a shutdown for a refueling outage. The Containment HP requests that Containment In-Service Purge be placed in service "as soon as possible" to reduce dose to workers.

Which of the following is the EARLIEST plant mode reached that will allow for establishment of Containment In-service Purge?

a. Entry into MODE 3 Hot Standby.

b. Entry into MODE 4 Hot Shutdown.

c. Entry into MODE 5 Cold Shutdown.

d. Entry into MODE 6 Refueling.

Goes with Q # 66

QUESTION: 072 (1.00)

11 Steam Generator has known primary to secondary leakage and has been isolated per 1C4 AOP2, STEAM GENERATOR TUBE LEAK.

What action is taken to limit the spread of contamination to the Turbine Building?

a. Draining the 11 SG via the SGB system to the river and refilling 11 SG with clean water.

b. Isolating the Turbine Building Sump, then draining the Main Condenser Hotwell to the sump and refilling the Main Condenser Hotwell with clean water.

c. Realigning the 11 SG Safety Relief Header drains from the Unit 2 Turbine Building sump to the Aerated Sump Tank.

d. Realigning the 11 SG Safety Relief Header drains from the Unit 2 Condenser to the Unit 2 Turbine Building Sump.

67. G2.3.11 001/3/1/CNTRL RAD RELEASES/MEM - 2.7/BANK/R/NRC RO/TNT/DSS

Chemistry reports RCS activity is exceeding the Tech Spec limits and the Unit SS has ordered a unit shutdown.

is a TS reqmt

In the event of a subsequent SGTR, which **ONE** of the following actions is designed to limit the release of radioactivity ?

A. Main Steam Isolation Valves (MSIVs) are closed.

B. All condensate polisher demineralizers are placed in service.

C.✓ Reactor Coolant System (RCS) is cooled down to below 500 degrees F.

D. SG atmospheric relief valve for a ruptured SG would be placed in manual and shut.

67. G2.3.11 001/3/1/CNTRL RAD RELEASES/MEM - 2.7/BANK/R/NRC RO/TNT/DSS

K/A

G2.3.11 Ability to control radiation releases.

K/A MATCH ANALYSIS

Question gives a scenario where a unit shutdown is directed based on high RCS activity. Candidate must know bases of cooling RCS to < 500 degrees F to limit release in event of a subsequent SGTR.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. RCS cooldown to < 500 degrees F performed per Tech Specs & bases.

B. Incorrect. RCS cooldown to < 500 degrees F performed per Tech Specs & bases.

C. Correct. RCS cooldown to < 500 degrees F performed per Tech Specs & bases.

D. Incorrect. RCS cooldown to < 500 degrees F performed per Tech Spec & bases.

REFERENCES

Technical 3.4.16 and Bases for RCS Specific Activity.

Watts Bar July 2004 NRC RO examination question # 73.

VEGP learning objectives:

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

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

b. The required actions for all section 3.4 LCOs.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

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

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

ACTIONS

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

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

(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/Ē μCi/gm.	7 days
SR 3.4.16.2	-----------------------NOTE----------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

LCO

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

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

APPLICABILITY

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

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

(continued)

G 2.3 11 001

The Unit Supervisor has directed a Unit shutdown based on RCS activity exceeding Tech Spec limits.

In the event of a subsequent SGTR, which ONE of the following actions is designed to limit the release of radioactivity?

A. MSIVs are closed.

B. RCS is cooled down below 500 0F.

C. Condensate polishers are in service at full flow

8. S/G atmospheric dump valve for the affected S/G is placed in OFF

The correct answer is B.

A. Incorrect - Closing MSIVs would contribute to Rad release through S/G PORVs if cooldown not performed in a timely manner.

B. *Correct* - Basis states that the cooldown to < 500 ºF will lower thr RCS temperature below the saturation pressure of the S/G reliefs.

C. Incorrect -While the polishing system would help clean the secondary, it is not an action contained in the procedure.

8. Incorrect - PORV setpoint are raised to 90% to limit release of radioactivity.

REFERENCES:

3-OT-T/S0304 p. 8
Tech Spec 3.4.16 basis
E-3 'S/G Tube Rupture'

Beaver Valley 12/01/2002
10CFR55.43.4/45.10

Ability to control radiation releases.

RO-73 SRO-73

Reference:	3-OT-SYSO63	K/A Number:	G 2.3.11
WA value:	2.1	Last Used:	12/01/2002
Level:	1	Source:	BANK
Tier/Grp:	3	SRO Only:	

5/27/2004

WATTS BAR JULY 2004 RO Q73

Goes with Question #67

68. G2.4.22 001/3/1/BASES CSFTS PRIORITY/C/A - 3.0/BANK/RO/NRC RO/TNT/DSS

Unit 1 has tripped and Safety Injection has actuated due to a Large Break Loss of Coolant Accident (LOCA).

Many complications have occurred.

The crew has exited 19000-C, "Reactor Trip or Safety Injection". The person monitoring Critical Safety Function Status Trees reports:

* Subcriticality - Orange Path

* Heat Sink - Yellow Path

* Core Cooling - Orange Path

* Containment - Red Path

Which **ONE** of the following states the **CORRECT** procedure transition and what it is based on ?

A. 19212-C, "Response to Loss of Core Shutdown", based on an Severe Challenge to the subcriticality Orange Path.

B. 19235-C, "Response to Steam Generator Low Level", based on Heat Sink Yellow Path not being satisfied.

C. 19222-C, "Response to Degraded Core Cooling", based on a Severe Challenge to the Core Cooling Orange Path.

D✓ 19251-C, "Response to High Containment Pressure, based on an Extreme Challenge to the Containment Red Path

68. G2.4.22 001/3/1/BASES CSFTS PRIORITY/C/A - 3.0/BANK/RO/NRC RO/TNT/DSS

K/A

G2.4.22 Knowledge of the bases for prioritizing safety functions during abnormal / emergency operations.

K/A MATCH ANALYSIS

Question poses a credible scenario where the candidate has to prioritize between multiple CSFST indications and choose the correct procedure based on status tree priority chain. Red, Orange, Yellow, Green.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Orange Path is lower priority than Red Path.

B. Incorrect. Yellow Path is lower priority than Red and Orange Path.

C. Incorrect. Orange Path is lowe priority than Red Path.

D. Correct. Red Path is highest priority even though lower on status trees.

REFERENCES

19200-C, "Critical Safety Function Status Trees"

Surry March 2004 NRC RO examination question # 21.

VEGP learning objectives:

LO-LP-37002-08 State the rules of usage for functional restoration procedures in terms of:

a. when they must be used
b. when they may be terminated
c. transition criteria for higher priority CSF challenge.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:

- If SPDS display of the Plant Computer is not operable or questionable, manual monitoring of CSFSTs should be performed by a licensed operator.
- CSFSTs should be monitored continuously if a RED or ORANGE condition is present or each 10 to 15 minutes if the highest priority CSFST is no higher than YELLOW.
- CSFSTs should be checked in order listed.
- Priority of operator action is given by the following:
  - Red (Solid) Path - Extreme challenge, in Tree Order per Step 1.
  - Orange (Dashed) Path - Severe challenge, in the Tree Order per Step 1.
  - Yellow (Dotted) Path - Not satisfied, in Tree Order per Step 1.
  - Green (Outlined) Path - Satisfied.
- If using the Plant Computer (if available) to monitor CSFSTs:
  - The mode indication of the Plant Computer CSFSTs should be indicating zero.
  - RCP breakers should be opened for RCPs NOT running in order to provide proper RVLIS indication.
  - If SPDS is operable, CSFSTs may be checked by scanning the display console for alarm conditions.
  - Color status of CSFSTs will also be indicated by letter R for red, O for orange, Y for yellow, G for green, and M for magenta.
  - CSFSTs will indicate active (alarming) paths as solid lines and non-active paths as empty or hollow lines.

1. Check CSFSTs - SATISFIED:
   - ☐ a. Subcriticality (F-0.1)
   - ☐ b. Core Cooling (F-0.2)
   - ☐ c. Heat Sink (F-0.3)
   - ☐ d. Integrity (F-0.4)
   - ☐ e. Containment (F-0.5)
   - ☐ f. Inventory (F-0.6)

☐ 1. IF a Red condition exists, THEN immediately go to FRP.

☐ IF an Orange condition exists, THEN go to FRP after completion of present pass thru CSFSTs.

☐ IF a Yellow condition exists, THEN initiate FRP after evaluating plant conditions with Shift Supervisor's approval.

21. 02202.4.22 001/2/1/SAFETY FUNTIONS/C/A 3.0/4.0/B/SR04301/R/MAB/SDR

Unit 1 has tripped and Safety Injection has actuated due to a Large Break Loss of Coolant Accident (LOCA).

Many complications have occurred.

The crew has exited E-0, Reactor Trip or Safety Injection. The Shift Technical Advisor has started to monitor Critical Safety Function Status Trees and reports:
- Subcriticality - Orange Path
- Heat Sink - Yellow Path
- Core Cooling - Orange Path
- Containment - Red Path

Which ONE of the following states the correct procedure transition?

A. FR-S.1, Response to Nuclear Power Generation/ATWS, based on Subcriticality Orange Path.

B. FR-H.1, Response to Secondary Heat Sink, based on Heat Sink Yellow Path.

C. FR-C.1, Response to Inadequate Core Cooling, based on Core Cooling Orange Path.

D.✓ FR-Z.1, Response to High Containment Pressure, based on Containment Red Path.

Goes with question #68

Surry

References:
ND-95.3-LP-26, Critical Safety Function Status Trees, Rev. 5

Distractor Analysis:

A. Incorrect based on the rules of use for safstv function status trees (ND-95.3-LP-26 Page 15). The Subcriticality Orange Path does not take priority over any Red Path.

B. Incorrect based on the rules of use for safety function status trees (ND-95.3-LP-26 Page 15). The Heat Sink Yellow Path does not take priority over Containment Red Path.

C. Incorrect based on the rules of use for safety function status trees (ND-35.3-LP-26 Page 15). Core Cooling Orange Path does not take priority over Containment Red Path.

D. Correct based on the rules of use for safety function status trees (ND-95.3-LP-26 Page 15). The Containment Red Path takes priority over the other paths. Only knowledge of safety function priority rules are needed to answer this question.

022 Containment Cooling
G2.4.22: Knowledge of the bases for prioritizing safety functions during abnormal and emergency operations.

Turkey Point Bank Question TP03301

Goes with question #68

69. G2.4.24 001/3/1/LOSS CW PROC/C/A - 3.3/BANK/RO/NRC RO/TNT/DSS

The crew is in 19100-C, "Loss of All AC Power". Prior to the step that the crew places equipment in PTL, the procedure ensures that 2 NSCW pumps should be available to load on each AC emergency bus.

These pumps are required to provide cooling for the.......

A.✓ EDG

B. SI pump

C. ACCW pump

D. MDAFW pump

69. G2.4.24 001/3/1/LOSS CW PROC/C/A - 3.3/BANK/RO/NRC RO/TNT/DSS

K/A

G2.4.24 Knowledge of loss of cooling water procedures.

K/A MATCH ANALYSIS

Question asks the bases for placing all equipment in PTL in 19100-C, "Loss of All AC Power" except for the 2 NSCW pumps. This is to provide cooling to EDG when power is restored.

ANSWER / DISTRACTOR ANALYSIS

A. Correct.

B. Incorrect. SI pump will not be started until electrical power is restored.

C. Incorrect. ACCW pumps are self cooled but Hx cooled by NSCW.

D. Incorrect. MDAFW pumps not cooled by NSCW.

REFERENCES

Plant Alvin W. Vogtle 2002 NRC RO examination question # 62

19100-C, "Loss of All AC Power" step # 13

WOG EOP Background for ECA-0.0 pages 114 and 116.

VEGP learning objectives:

LO-LP-37031-08 Using EOP 19100 as a guide, briefly describe how each step is accomplished.

LO-LP-37031-09 Given a NOTE or CAUTION statement from the EOP, state the bases for that NOTE or CAUTION statement.

1. 062AG2.4.24 002

The crew is in 19100-C, "Reactor Trip or Safety Injection". Prior to the step that the crew places equipment in PTL, the procedure cautions that 2 NSCW pumps should be available to load on each AC Emergency Bus.

These pumps are required to provide cooling for the ....

A. SI pump

B. MDAFW pump

C. ACCW pump

D✓ EDG

Ref: 19100-C Caution Before Step 7

Vogtle 2002

Goes with Q # 69

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 13. Verify at least two NSCW Pumps per train - HANDSWITCHES IN AUTO	
14. Place the following equipment handswitches in the PULL-TO-LOCK position: ☐ • CCPs ☐ • RHR Pumps ☐ • SI Pumps ☐ • Containment Spray Pumps ☐ • CCW Pumps ☐ • ACCW Pumps ☐ • MDAFW Pumps ☐ • Containment Coolers ☐ • ESF Chillers (STOP position)	
15. Check if AC Emergency Busses should be energized locally:	
☐ a. At least one 4160V AC Emergency Bus - CAN BE ENERGIZED FROM CONTROL ROOM	☐ a. Dispatch personnel to restore AC Emergency Bus(s) using 13427, 4160V AC ELECTRICAL DISTRIBUTION. ☐ Go to Step 16.
☐ b. Restore AC Emergency Bus(s) from the Control Room using 13427, 4160V AC ELECTRICAL DISTRIBUTION.	
☐ 16. Dispatch Operator to initiate isolation of RCP Seals by performing ATTACHMENT E.	

CAUTION: A service water pump should be kept available to automatically load on its ac emergency bus to provide diesel generator cooling.

PURPOSE: To alert the operator that the service water pumps should not be pulled to lock so that they are available to automatically load on the ac emergency bus when power is restored.

BASIS:

Automatic start of service water pumps is permitted to provide cooling to the diesel generator should the diesel generator start as a result of local operator actions. The reference plant requires service water for diesel generator cooling.

ACTIONS:

N/A

INSTRUMENTATION:

N/A

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

Evaluate cooling provisions for diesel generator. Automatic start of service water pumps is not necessary if the diesel generator does not require service water cooling.

STEP: Place Following Equipment Switches In PULL-TO-LOCK Position

PURPOSE: To defeat automatic loading of large loads on the ac emergency bus

BASIS:

If ac power cannot be restored from the control room, it is necessary to dispatch personnel to locally restore ac power. Local actions to restore ac power may result in ac power restoration by means of a temporary jury-rigged power supply. The longer the duration of the ac power outage, the more likely a plant condition may develop wherein automatic start of equipment upon ac power restoration may have detrimental affects on there stored ac emergency bus, the automatically started equipment or other plant equipment, such as the RCP. Consequently, Step 6 instructs the operator to defeat automatic loading of all large loads, except for the service water pump, prior to dispatching personnel to locally restore ac power. For the reference plant, automatic loading is defeated by placing the equipment switches in the pull to lock position.

Defeating automatic blackout or SI loading of as many large loads as practical is intended to avoid potential overload of the energized ac emergency bus. This action permits the operator to evaluate the status of the restored emergency bus and sequence loads onto the bus consistent with bus status and plant conditions. A service water pump is permitted to automatically load on the ac emergency bus to provide diesel cooling should the diesel have started. Other small loads, such as the 480 volt busses, are also permitted to automatically load on the restored ac emergency bus. These small loads will aid the operator in diagnosing plant status. Restricting automatic loading to this equipment limits the initial demand on the ac emergency bus. Equipment that are permitted to automatic load are common to both guideline ECA-0.1 and ECA-0.2 recovery and should not result in damage to any plant equipment if they automatically load upon ac power restoration.

Defeating automatic loading of the charging/SI pumps in Step 6 also functions to protect the RCP from damage when ac power is restored. This action prevents the automatic delivery of relatively cold seal injection flow into the RCP number 1 seal chamber and shaft area which has the potential for thermal shock

70. G2.4.7 001/3/1/EVNT EOP STRATEGIES/C/A - 3.1/BANK/RO/NRC RO/TNT/DSS

The following conditions exist on Unit 1:

* The control room team is responding to a LOCA outside containment.

* The reactor was tripped and an SI manually actuated.

* B Train has lost all AC power.

* 1A RHR pump has tripped.

While in 19010-C, "Response to Loss of Primary or Secondary Coolant" , the crew was transitioned to EOP 19111-C, "ECA-1.1, Loss of Emergency Coolant Recirculation".

Which **ONE** of the following describes the **CORRECT** **strategies** to use in ECA-1.1 under these conditions ?

A.✓ Start makeup to the RWST, initiate an RCS cooldown at < 100 degrees F per hour, reduce / minimize ECCS flow, depressurize the RCS.

B. Verify all containment coolers running ~~is~~ slow speed, initiate an RCS cooldown at the maximum rate to 200 degrees F, depressurize the RCS, place RHR in service.

C. Immediately initiate an RCS cooldown at maximum rate possible, start makeup to the RWST, establish one train of ECCS flow to maintain subcooling > 74 degrees F.

D. Verify all containment coolers running in fast speed, verify only one containment spray pump is running, maximize ECCS flow to ensure adequate core cooling.

70. G2.4.7 001/3/1/EVNT EOP STRATEGIES/C/A - 3.1/BANK/RO/NRC RO/TNT/DSS

K/A

G2.4.7 Knowledge of event based EOP mitigation strategies.

K/A MATCH ANALYSIS

Question gives a plausible scenario where the crew has to enter 19111-C, "Loss of Emergency Coolant Recirculation". Candidate has to determine the correct actions to take under the conditions.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. These are actions in proper sequence to be taken for this EOP.

B. Incorrect. Containment coolers status is checked in EOP but a maximum rate cooldown of RCS is not perfored in this EOP. RHR is also not available due to loss of power to one train and pump tripped in the other.

C. Incorrect. Maximum rate cooldown of RCS is not performed in this EOP. Step for making up to RWST is also out of sequence.

D. Incorrect. Containment coolers should not be in fast speed during a LOCA, containment spray pump requirements would be based on multiple parameters such as coolers available, RWST level, etc., ECCS should be minimized not max.

REFERENCES

Farley August 2004 NRC SRO examination question # 25.

19111-C, "Loss of Emergency Coolant Recirculation"

VEGP learning objectives:

LO-LP-37114-11 State the indications of a loss of emergency recirculation capability.

LO-LP-37114-12 State the intent of EOP 19111, Loss of Emergency Coolant Recirculation.

1. W/E11EA2.2 001

The following conditions exist on Unit 1:

- The Control room team is responding to a LOCA outside containment.
- The reactor was tripped and an SI manually actuated.
- B Train has lost all AC power.
- 1A RHR pump has tripped.

At the step to verify cold leg recirculation capability available in EEP-1.0, Loss of Reactor or Secondary Coolant, the team transitioned to ECP-1.1, Loss Of Emergency Coolant Recirculation. Which one of the following describes the correct actions to take in ECP-1.1 under these conditions?

A.✓ Start makeup to the RWST, initiate an RCS cooldown at less than 100°F/hr, minimize ECCS flow, and reduce RCS pressure.

B. Immediately initiate an RCS cooldown at the maximum rate possible, start makeup to the RWST, and establish one train of ECCS flow to maintain subcooling >66°F.

C. Verify all containment cooling units running in slow speed, initiate an RCS cooldown at the maximum rate possible to 200°F, and depressurize the RCS to allow RHR to be placed on service.

D. Verify all containment cooling units are running in fast speed, verify one containment spray pump is running, and control RCS temperature and pressure to maintain subcooling >66°F.

Goes with Q # 70

Farley August 2004
NRC SRO Exam.

Another plant RO exam
Referenced this as base
For their RO question.
Modified to be more RO.

RO exam was a G2.4.7
For this

EMERGENCY OPERATING PROCEDURE

ECA-1.1 LOSS OF EMERGENCY COOLANT RECIRCULATION

PRB REVIEW REQUIRED

PURPOSE

This procedure provides actions to restore emergency coolant recirculation capability, to delay depletion of the RWST by adding makeup and reducing outflow, and to depressurize the RCS to minimize break flow. (Applicable in Modes 1, 2, and 3)

ENTRY CONDITIONS

- 19010-C, E-1 LOSS OF REACTOR OR SECONDARY COOLANT
- 19012-C, ES-1.2 POST LOCA COOLDOWN AND DEPRESSURIZATION
- 19013-C, ES-1.3 TRANSFER TO COLD LEG RECIRCULATION
- 19112-C, ECA-1.2 LOCA OUTSIDE CONTAINMENT
- 19005-C, ES-0.0 REDIAGNOSIS
- 19251-C, FR-Z.1 RESPONSE TO HIGH CONTAINMENT PRESSURE

MAJOR ACTIONS

- ♦ Continue Attempts to Restore ECR.
- ♦ Increase/Conserve RWST Level.
- ♦ Initiate Cooldown to Cold Shutdown.
- ♦ Depressurize RCS to Minimize RCS Subcooling.
- ♦ Try to Add Makeup to RCS from Alternate Source.
- ♦ Depressurize SGs to Cool Down and Depressurize RCS.
- ♦ Maintain RCS Heat Removal.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. Determine Containment Spray requirements:	
a. Check CS Pump suction - FROM RWST	☐ a. IF CS Pump suction from Sump, THEN go to Step 9.
☐ • HV-9017A - CNMT SPRAY PMP-A RWST SUCT ISO VLV - OPEN	
☐ • HV-9017B - CNMT SPRAY PMP-B RWST SUCT ISO VLV - OPEN	
☐ b. Determine number of CS Pumps required from Table:	

RWST LEVEL	CONTAINMENT PRESSURE	FAN COOLERS IN SLOW	SPRAY PUMPS REQUIRED
GREATER THAN 39%	GREATER THAN 52 PSIG	N/A	2
	BETWEEN 21.5 PSIG AND 52 PSIG	0	2
		4	1
		8	0
	LESS THAN 21.5 PSIG	N/A	0
BETWEEN 10% AND 39%	GREATER THAN 52 PSIG	N/A	2
	BETWEEN 21.5 PSIG AND 52 PSIG	3	1
		6	0
	LESS THAN 21.5 PSIG	N/A	0
LESS THAN 10%	N/A	N/A	0

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ c. Check CS Pumps running - EQUAL TO NUMBER REQUIRED	☐ c. Reset Containment Spray.
	☐ Operate CS Pumps and discharge valves as required.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ *11. **Check CST level - GREATER THAN 15%**	☐ *11. Swap to alternate CST by initiating 13610, AUXILIARY FEEDWATER SYSTEM.
*12. **Initiate RCS cooldown to cold shutdown:**	
☐ a. Monitor shutdown margin by initiating 14005, SHUTDOWN MARGIN AND KEFF CALCULATIONS.	
☐ b. Maintain cooldown rate in RCS cold legs - LESS THAN 100° F/HR	
☐ c. Dump steam to Condenser from intact SG(s) using Steam Dumps.	☐ c. Dump steam from intact SG(s) using SG ARV(s). ☐ IF no intact SG(s) available, THEN use faulted SG(s).

71. WE04EK3.2 001/1/1/LOCA OUTSIDE - PROCS/C/A - 3.4/BANK/RO/NRC RO/TNT/DSS

A small break LOCA has occurred outside containment.

Actions of EOP 19112-C, "LOCA Outside Containment", have been completed and RCS pressure continues to decrease. A transition is made to 19111-C, "Loss of Emergency Coolant Recirculation".

Which **ONE** of the following describes the reason a transition is made to the Loss of Recirculation EOP ?

A. To terminate any possible offsite release.

B. To recover RCS pressure after the break was isolated.

C. To secure all pumps that are running in order to preserve inventory.

D✓ To take compensatory actions for lack of inventory in the containment sump.

71. WE04EK3.2 001/1/1/LOCA OUTSIDE - PROCS/C/A - 3.4/BANK/RO/NRC RO/TNT/DSS

K/A

WE04 LOCA Outside Containment

EK3.2 Knowledge of the reasons for the following responses as the apply to the LOCA Outside Containment.

Normal, abnormal, and emergency operating procedures associated with LOCA Outside Containment.

K/A MATCH ANALYSIS

Question gives a plausible scenario where a LOCA Outside Containment has occurred and actions in 19112-C obviously did not stop the leak. Candidate has to determine reason for entering 19111-C Loss of Recirc procedure.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. Candidate may think Loss of Recirc takes actions to mitigate possible release from LOCA Outside CTMT.

B. Incorrect. Plausible. Candidate must release that pressure still lowering indicates the leak was not isolated.

C. Incorrect. Plausible. Candidate may realize the need to conserve inventory but stopping all pumps would not be a correct action.

D. Correct. This is reason for entering 19111.

REFERENCES

Farley December 2004 NRC RO examination question # 69.

WOG Background Document for ECA-1.1, "Loss of Emergency Coolant Recirc"

VEGP learning objectives:

Not Applicable

1. W/E04EK3.2 001

A small break LOCA has occurred outside containment.

Actions of ECP-1.2, LOCA Outside Containment, have been completed and RCS pressure continues to decrease. A transition is made to ECP-1.1, Loss of Emergency Coolant Recirculation.

Which one of the following describes the reason a transition is made to ECP-1.1?

A. To terminate the offsite release.

B. To recover after the break was isolated.

C. To secure all pumps that are running in order to preserve inventory.

D.✓ To take compensatory actions for lack of inventory in the containment sump.

Farley #69
Goes with Q # 71

1. INTRODUCTION

Guideline ECA-1.2, LOCA OUTSIDE CONTAINMENT, provides procedural guidance for a LOCA that occurs outside containment. Specifically, the objective of this guideline is to provide actions to identify and isolate a LOCA outside containment.

There are two explicit transitions to this guideline. One is contained in E-0, REACTOR TRIP OR SAFETY INJECTION, Step 29, on abnormal radiation in the auxiliary building due to a loss of RCS inventory outside containment. The other is contained in E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 11, if it is determined that the cause of abnormal radiation is due to a loss of RCS inventory outside containment.

There are two transitions from this guideline. First, in Step 3, if the break is isolated based upon increasing RCS pressure, the operator is transferred to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1. Second, if the break cannot be isolated, the operator is sent to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1, since there will not be any fluid inventory in the containment sump to provide recirculation capability.

72. WE05G2.4.49 001/1/1/LOSH - IMM OP ACTION/C/A - 4.0/NEW/RO/NRC RO/TNT/DSS

While at full power at EOL, the following sequence of events occurs:

* MFPT "A" trips

* AOP-18016-C, "Condensate and Feedwater Malfunction" entered

Crew has just transitioned to 19231-C, "Loss of Secondary Heat Sink" due to inability to establish AFW flow from any source.

AFW flow is currently 0 gpm to all Steam Generators.

Steam Generator Wide Range (WR) levels are as follows:

SG # 1 - 27.9%
SG # 2 - 28.8%
SG # 3 - 27.2%
SG # 4 - 28.4%

licensee to revise - as worded it appears convoluted - & would not enter FRH1 until in E-0

Which **ONE** of the following would be **CORRECT** regarding:

a. - IOA RNOs to be taken prior to the plant trip after checking both MFPs - RUNNING.
b. - First actions to take now regarding the LOHS in progress.

A. a - check reactor power > 75% , start Turbine Setback, start 3rd condensate pump, ensure rapid rod insertion, TRIP Reactor if SG NR levels < 40% and go to E-0.

b - continue attempts to establish AFW, depressurize a SG, feed with condensate.

B. a - check reactor power > 75%, TRIP reactor and go to E-0.

b - trip all RCPs, actuate SI, establish feed and bleed of the RCS

C✓ a - check reactor power > 75%, start Turbine Setback, start 3rd condensate pump, ensure rapid rod insertion, TRIP reactor if SG NR levels < 40% and go to E-0.

b - trip all RCPs, actuate SI, establish feed and bleed of the RCS.

D. a - check reactor power > 75%, TRIP reactor and go to E-0.

b - continue attempts to establish AFW, depressurize a SG, feed with condensate.

if Rx tripped power is D

72. WE05G2.4.49 001/1/1/LOSH - IMM OP ACTION/C/A - 4.0/NEW/RO/NRC RO/TNT/DSS

K/A

WE05 Inadequate Heat Transfer - Loss of Secondary Heat Sink

G2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

K/A MATCH ANALYSIS

Question gives a plausible scenario where a Main Feed Pump Turbine (MFPT) trips at 100% power at EOL. Since power uprates it is uncertain whether or not a plant trip can be prevented on loss of a single MFPT.

Candidate must be able to recognize the RNOs to step 1 Immediate Operator Action that should have been performed. IF, power > 75%, at turbine setback should be started, 3rd condensate pump started, control rods inserted rapidly. **IF**, SG NR levels drop to < 40% a reactor trip is required.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. IOAs RNO are correct, however, feed and bleed criteria have been met and feed and bleed should be established.

B. Incorrect. IOAs RNO for trip of BOTH MFPTs, feed and bleed criteria have been met so this part of answer is right.

C. Correct. IOAs are correct, feed and bleed criteria have been met..

D. Incorrect. IOAs RNO for trip of BOTH MFPTs, feed and bleed criteria have been met and feed and bleed should be established.

REFERENCES

AOP-18016-C, "Condensate and Feedwater Malfunction"

EOP-19231-C, "Loss of Secondary Heat Sink" step # 6, and # 35 thru 38.

VEGP learning objectives:

LO-LP-60314-02 Describe the operator actions required if during the performance of AOP 18016-C "Condensate and Feedwater Malfunction" a loss of SG level is imminent.

LO-LP-60314-05 Given the entire AOP, describe:

a. Purpose of selected steps
b. How and why the step is being performed
c. Expected response of the plant/parameter(s) for the step

A. MFP(s) MALFUNCTION

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
IMMEDIATE OPERATOR ACTIONS	
A1. CHECK BOTH MFPs - RUNNING.	A1. PERFORM required actions for the loss of MFP(s): a. CHECK reactor power - GREATER THAN 75%. IF reactor power - LESS THAN 75%, THEN GO TO Step A3. b. IF no MFP running, THEN TRIP the reactor and GO TO 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION. c. PERFORM the following: • PRESS START SETBACK Pushbutton on the Turbine Control Panel. -OR- REDUCE generator output to less than 850 MWe. • START third condensate pump • ENSURE rapid insertion of control rods to match Tave and Tref. d. MAINTAIN SG NR levels greater than 40%. IF SG NR levels cannot be maintained greater than 40%, THEN TRIP the reactor and GO TO 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 5. Check CCP status - AT LEAST ONE AVAILABLE	☐ 5. Stop all RCPs. ☐ Go to Step 35.
* 6. **Check if RCS bleed and feed is required:**	
a. Check the following: ☐ **WR level in any 3 SGs - LESS THAN 29% [44% ADVERSE]** -OR- ☐ RCS pressure - GREATER THAN 2335 PSIG DUE TO LOSS OF SECONDARY HEAT SINK	☐ a. WHEN criteria for bleed and feed are met, THEN perform Steps 6b and 6c. ☐ Go to Step 7.
☐ b. Trip all RCPs.	
☐ c. Go to Step 35 and perform bleed and feed actions.	
☐ 7. Place Containment Hydrogen Monitors in service by initiating 13130, POST-ACCIDENT HYDROGEN CONTROL.	
☐ * 8. **Check CST level - GREATER THAN 15%**	☐ * 8. Swap to alternate CST by initiating 13610, AUXILIARY FEEDWATER SYSTEM.
9. Verify SG Blowdown isolated: ☐ • SG Blowdown Isolation Valves - CLOSED WITH HANDSWITCHES IN CLOSE POSITION ☐ • SG Sample Isolation Valves - CLOSED	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Steps 35 thru 38 should be performed quickly in order to establish RCS heat removal by RCS bleed and feed.

☐ 35. Verify SI actuated.

36. Verify RCS feed path:

a. Verify ECCS Pump status:

☐ CCPs - AT LEAST ONE RUNNING

-OR-

☐ SI Pumps - AT LEAST ONE RUNNING

☐ b. Verify ECCS valve alignment - PROPER INJECTION LINEUP INDICATED ON MLBs

Response Not Obtained:

☐ 36. Start pumps and align valves as necessary to establish injection flow using ATTACHMENT A or B.

☐ IF a feed path can NOT be established, THEN continue attempts to establish feed flow.

☐ Return to Step 10.

CAUTION: During bleed and feed operation the PRT may rupture.

37. Establish RCS bleed path:

☐ a. Place all PRZR Heaters in OFF/PTL.

☐ b. Check power to PRZR PORV Block Valves - AVAILABLE

Response Not Obtained: ☐ b. Restore power to block valves.

☐ c. Arm COPS and check PRZR PORV Block Valves - BOTH OPEN

Response Not Obtained: ☐ c. Open both PRZR PORV Block Valves.

☐ d. Open both PRZR PORVs.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
38. Verify adequate RCS bleed path: ☐ • COPS - ARMED ☐ • PRZR PORV Block Valves - BOTH OPEN ☐ • PRZR PORVs - BOTH OPEN	38. Perform the following: a. Open Reactor Vessel Head Vent Valves: ☐ • HV-8095A - RX HEAD VENT TO LETDOWN ISOLATION VLV ☐ • HV-8095B - RX HEAD VENT TO LETDOWN ISOLATION VLV ☐ • HV-8096A - RX HEAD VENT TO LETDOWN ISOLATION VLV ☐ • HV-8096B RX HEAD VENT TO LETDOWN ISOLATION VLV ☐ • HV-0442A - REACTOR HEAD VENT TO PRT ☐ • HV-0442B - REACTOR HEAD VENT TO PRT b. Align an available low pressure water source to at least one intact SG: ☐ Initiate Attachment C. ☐ <u>IF</u> low-pressure water source can <u>NOT</u> be aligned, <u>THEN</u> go to Step 39.
☐ 39. Initiate ATTACHMENT D while continuing with this procedure.	
*40. **Maintain RCS heat removal:** ☐ • ECCS flow. ☐ • PRZR PORVs - BOTH OPEN	☐ • Maintain Reactor Vessel Head Vent Valves open.

73. WE06EA1.3 001/1/2/ICC - DESIRED RESULT/MEM - 3.7/BANK/RO/NRC RO/TNT/DSS

FR-C.2, "Response to Degraded Core Cooling" is being performed. Which **ONE** of the following is the reason RCPs are stopped prior to depressuring the SGs to atmospheric pressure during the degraded core cooling event ?

A. To reduce the potential for pressurized thermal shock as the result of the rapid cooldown.

B.✓ The SG depressurization will lead to low RCP seal # 1 D/Ps possibly damaging the RCPs.

C. The SGs will depressurize more quickly if no forced circulation RCS flow exists.

D. RCP operation with the SGs at atmospheric pressure is prohibited due to excessive hydraulic stress on the SG U-tubes.

73. WE06EA1.3 001/1/2/ICC - DESIRED RESULT/MEM - 3.7/BANK/RO/NRC RO/TNT/DSS

K/A

WE06 Degraded Core Cooling

EA1.3 Ability to operate and/or monitor the following as they apply to the (Degraded Core Cooling).

Desired operating results during abnormal and emergency conditions.

K/A MATCH ANALYSIS

Question asks the reasons RCPs are stopped prior to depressurization of SGs during performance of 19222-C, "Response to Degraded Core Cooling". WOG background has RCPs stopped due to impending loss of support conditions. This will ensure RCPs are available for later use if necessary.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Basis for securing RCPs is not associated with PTS.

B. Correct. Losing RCP # 1 seal support conditions could damage the RCPs.

C. Incorrect. Basis for securing RCPs is not associated with heat input or forced flow.

D. Incorrect. Basis to due to hydraulic stresses on SGs but loss of support conditions.

REFERENCES

Surry March 2004 NRC RO examination question # 71.

19222-C, "Response to Degraded Core Cooling" steps # 13 and # 14.

WOG Background document for FR-C.2 step # 13 (page # 43)

VEGP learning objectives:

Not appllicable, no objectives on Degraded Core Cooling.

71. WE06EK3.1 001/1/2/CORE COOLING/MEM 3.4/3.8/B/SR04301/R/MAB/SDR

1-FR-C.1, Response to Inadequate Core Cooling, is being performed. Which ONE of the following is the reason RCPs are stopped prior to depressurizing the SGs to less than 150 psig during an inadequate core cooling event?

A. RCP operation with the SGs at atmospheric pressure is prohibited due to excessive hydraulic stress on the SG U-tubes.

5. The SGs will depressurize more quickly if no Forced Circulation RCS flow exists.

C. To minimize heat input *to* the RCS.

D.✓ The SG depressurization will lead to a loss of RCP support conditions.

Surry

References:
ND-95.3-LP-38, Response to Inadequate Core Cooling, Rev. 8
FR-C.1, Response to Inadequate Core Cooling, Rev. 18

Bistractor Analysis:

A. Incorrect because securing RCPs is necessary because the depressurization will result in losing the RCP seal support conditions, which could damage the WCPs.
5. Incorrect because the basis for securing RCPs is not associated with heat input into the RCS or forced flow.
C. Incorrect because the basis for securing RCPs is not associated with heat input into the RCS.
D. Correct because this is the stated reason in NB-95.3-LP-38. Losing #1 Seal support conditions could result in damage to the RCPs.

074 Inad. Core Cooling
E06EK3.1: Knowledge of the reasons for the following responses as they apply to (Degraded Core Cooling): Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

Surry Requal Exam Bank Question #467

Goes with Q # 73

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

(Step 12 continued from previous page)

d. Isolate ACCUM ISO VLVs - SHUT:

- HV-8808A
- HV-8808B
- HV-8808C
- HV-8808D

Response not obtained:

d. Vent any non-isolable accumulators:

1) Ensure N2 supply valve HV-8880 is shut.

2) Open ACCUM N2 SUPPLY/VENT VLVs:

TRAIN A

- HV-8875A
- HV-8875B
- HV-8875C
- HV-8875D

TRAIN B

- HV-8875E
- HV-8875F
- HV-8875G
- HV-8875H

3) Open common vent valve HV-0943A or HV-0943B.

4) IF an accumulator can NOT be isolated or vented, THEN consult the TSC to determine contingency actions.

e. Open ACCUM ISO VLV MOV breakers.

CAUTION: F-0.2 Core Cooling CSFST should be closely monitored during the subsequent steps.

13. Stop all RCPs.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*14. **Depressurize all intact SGs to atmospheric pressure:**	
a. Maintain cooldown rate in RCS cold legs - LESS THAN 100° F/HR.	
b. Dump steam to condenser from intact SG(s) using steam dumps.	b. Dump steam from the intact SG(s) using SG ARV(s).
*15. Verify ECCS flow: • CCP flow indicators - CHECK FOR BIT FLOW. -OR- • SI pump flow indicators - CHECK FOR FLOW. -OR- • RHR pump flow indicators - CHECK FOR FLOW.	*15. Continue efforts to establish ECCS flow. (CCPs through BIT, SIPs, RHR pumps) <u>IF</u> CCP flow not verified, <u>THEN</u> perform the following: a. Reset SI, if necessary. b. Start the NCP. c. Return to Step 14.

<u>NOTE</u>: COPS should be armed when RCS WR cold leg temperature is less than 350° F.

*16. Check core cooling: • RVLIS full range indication - GREATER THAN 62%. • At least two RCS WR hot leg temperatures - LESS THAN 350° F.	*16. Return to Step 14.
17. Go to 19010-C, E-1 LOSS OF REACTOR OR SECONDARY COOLANT, Step 13.	

END OF PROCEDURE TEXT

STEP DESCRIPTION TABLE FOR FR-C.2 Step 13

STEP: Stop All RCPs

PURPOSE: To verify all RCPs have been stopped

BASIS:

In preparation for the subsequent depressurization of the SGs to atmospheric pressure, the RCPs are stopped due to the anticipated loss of Number 1 seal requirements. Continued operation may result in damage to the RCPs.

ACTIONS:

Stop all RCPs

INSTRUMENTATION:

RCP status indication

CONTROL/EQUIPMENT:

RCP switches

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

N/A

74. WE15EK3.3 001/1/2/CTMT FLOOD-MANIP CTL/MEM - 2.9/BANK/RO/NRC RO/TNT/DSS

Given the following conditions:

* The unit has tripped from 100% power

* 19252-C (FR-Z.2), "Response to Containment Flooding" has been entered.

Which **ONE** of the following identifies the main reason it is important to identify and isolate the source of flooding that has caused sump level to exceed 105 inches ?

A. Will cause excessive backpressure on recirculation sump isolation valves and hinder or prevent sump swapover actions.

B.✓ Plant components necessary for recovery could be damaged / rendered inoperable.

C. Will lead to an overpressure condition that could challenge containment integrity.

D. Will lead to failure of containment "C" level penetrations due to low sump pH.

74. WE15EK3.3 001/1/2/CTMT FLOOD-MANIP CTL/MEM - 2.9/BANK/RO/NRC RO/TNT/DSS

K/A

WE15 Containment Flooding

knowledge of the reasons for the following responses ...

EK3.3 Manipulation of controls required to obtain desired operating results during abnormal and emergency conditions.

K/A MATCH ANALYSIS

Question gives a plausible scenario where a reactor trip has occurred and containment flooding is present. Candidate must be able to identify the main reason for entry into the containment flooding procedure.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Head pressure caused by excessive flooding may be greater than expected, however, not enough to interfere with operation of these valves.

B. Correct. Safety related componenets required for safe shutdown are above design flood level. Flooding above this level could impact these components. From WOG EOP background.

C. Incorrect. Since level will reduce containment volume in containment, some increased containment pressure may be expected. Examinee may think pressure rise excessive due to volume reduction.

D. Incorrect. Penetrations not expected to fail due to containment flooding.

REFERENCES

WOG EOP background document for FR-Z.1, "Response to Containment Flooding"

19252-C, "Response to Containment Flooding"

Watts Bar July 2004 NRC RO exam question # 27

VEGP learning objectives:

Not applicable.

2. DESCRIPTION

Guideline FR-Z.2, RESPONSE TO CONTAINMENT FLOODING, provides actions to respond when the containment level is greater than design flood level. This level is significant since the critical systems and components, which are necessary to ensure an orderly safe plant shutdown and provide feedback to the operator regarding plant conditions, are normally located above the design flood level. Therefore, the guideline FR-Z.2 is entered from the Containment Status Tree on an ORANGE priority when this design flood level is exceeded.

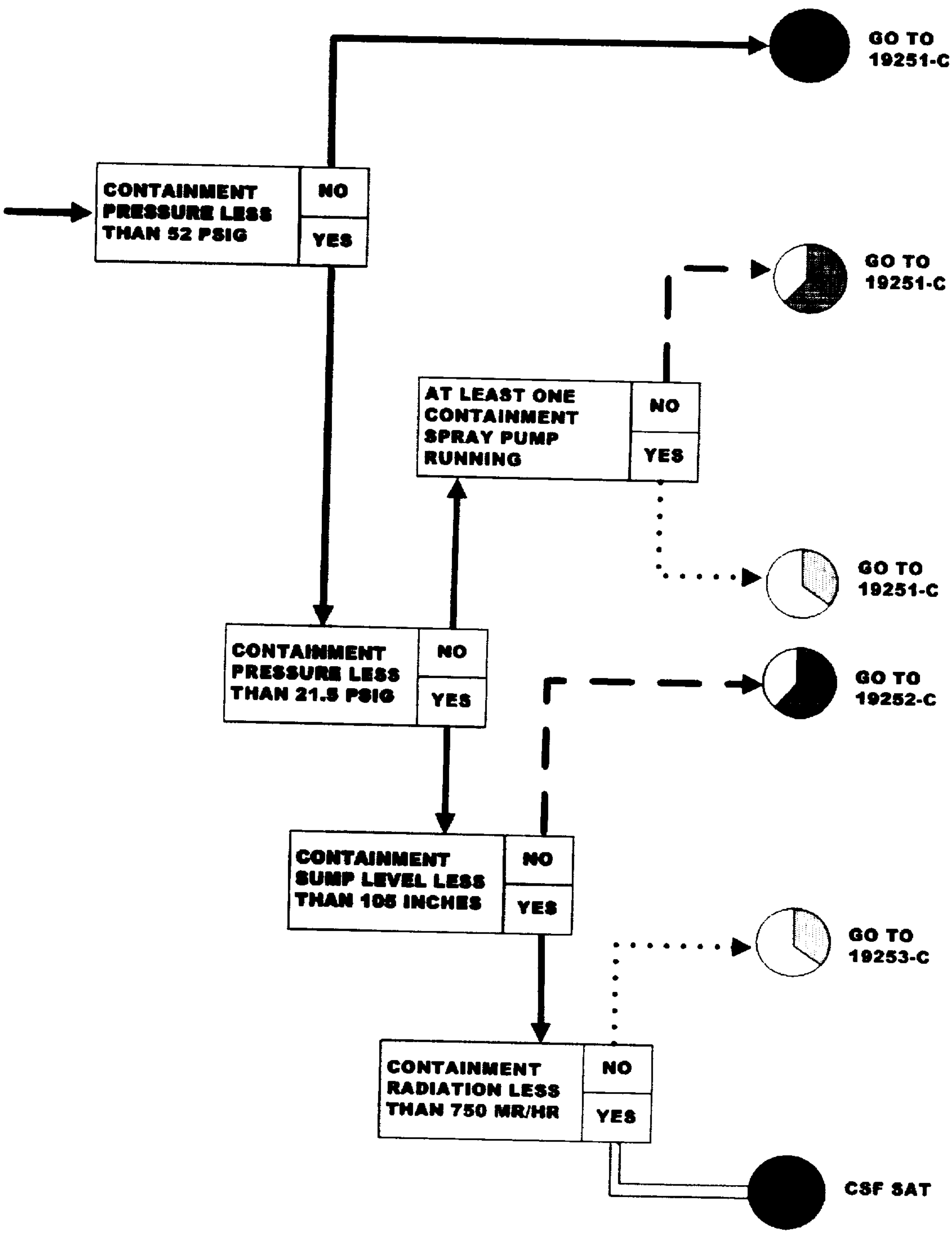
The primary purpose of the containment sump area is to collect the water injected into the containment or spilled from the reactor coolant system following an accident. The water collected in the containment sump is then available for long term core and/or containment cooling via the emergency core cooling or containment spray recirculation systems. In addition, the containment sump collects the injected or spilled water into areas such that vital systems or components will not be flooded and thus rendered inoperable.

The maximum level of water in the containment following a major accident generally is based upon the entire water contents of the reactor coolant system, refueling water storage tank, condensate storage tank, and SI accumulators. This water volume approximates the maximum water volume introduced into the containment following a LOCA plus a steamline or feedline break inside containment.

An indicated water level in the containment greater than the maximum expected volume (design basis flood level) is an indication that water volumes other than those represented by the above noted volumes have been introduced into the containment. Also, the high water level provides an indication that potential flooding of critical systems and components needed for plant recovery may occur.

The actions in this guideline attempt to identify any unexpected

F- 0.5
CONTAINMENT



E15 EK3.3 001

Given the following conditions:

- The unit has tripped from 100% power.
- Phase B Containment Isolation has been initiated.
- FR-Z.2, "Containment Flooding", has been entered

Which **ONE** of the following identifies the reason it is important to identify and isolate the source of flooding that has caused level to exceed 83%.

A. The high level will lower the effectiveness of the ice condenser to remove heat.

B✓ Plant components necessary for recovery could be damaged or rendered inoperable.

C. Will lead to an overpressure condition that could challenge containment integrity.

D. Will cause excessive backpressure on recirculation sump isolation valves and hinder or prevent sump swapover actions.

The correct answer is B.

A. Incorrect - containment level should not approach a point where it would block flow through the ice condenser. An Examinee that is not familiar with physical design of the ice condenser and containment may select this distracter.

B. *Correct* - components and instrumentation needed for orderly and safe plant shutdown are above design flood level. Flooding above that level could impact those components.

C. Incorrect - since level will reduce total volume in containment pressure some increased pressure may be expected. Examinee may think pressure rise would be excessive due to the volume reduction.

D. Incorrect - head pressure caused by the excessive flooding may be greater that expected however: not enough to interfere with operation of these valves.

Goes with
Q # 74

REFERENCES:

Lesson Plan 3-OT-FRZ0001 p. 12
FR-Z.2
WOG background document p.2 FR-Z.2

10CFR55.41.5/41.10/45.6/45.13
Knowledge of the reasons for the following responses as they apply to the (Containment Flooding): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

RO-27 SRO-27

Reference:	FR-Z.2	K/A Number:	W/E15 nK3.3
K/A value:	2.9	last Used:	
Level:	I	Source:	BANK
Tier/Grp:	1/2	SRO Only	

WATTS BAR JULY 2004 RO Q#27

Goes with question #74

75. WE16EK3.1 001/1/2/HI CTMT RAD-FAC/C/A - 2.9/NEW/RO/NRC RO/TNT/DSS

Which **ONE** of the following would be **CORRECT** regarding entry of 19253-C, "Response to Containment High Radiation Level" ? yellow Path

A. Entry at 750 mR / hr at containment outer wall due to significant core damage. Major procedure action(s) ensures evacuation of personnel in Auxiliary Building Penetration areas.

B. Entry at 100 mR / hr at containment outer wall due to fuel failure. Major procedure action(s) verifies Containment Ventilation Isolation and attempts to reduce radiation levels by containment filtration systems.

C. Entry at 100 mR / hr CNMT radiation levels requiring the use of Adverse CNMT values. Major procedure action(s) ensures evacuation of personnel at high radiation levels.

D. ✓ Entry at 750 mR / hr CNMT radiation levels after an RCS leak. Major procedure action(s) verifies Containment Ventilation Isolation and attempts to reduce radiation levels by containment filtration systems.