Arkansas Nuclear One

Name: KEY-KEY-KEY-KEY Logon: ________

Exam Number: NRC 2017-1

Exam Title: 2017 Unit 2 NRC Exam

Qual:

SRO [■] **RO** [■]

Date: ________

Points: 75/25/100

% Grade: ________

1.	[A]	[B]	[C]	[■]	51.	[A]	[B]	[■]	[D]
2.	[■]	[B]	[C]	[D]	52.	[A]	[B]	[■]	[D]
3.	[A]	[B]	[C]	[■]	53.	[A]	[■]	[C]	[D]
4.	[A]	[B]	[C]	[■]	54.	[■]	[B]	[C]	[D]
5.	[A]	[B]	[■]	[D]	55.	[A]	[B]	[■]	[D]
6.	[A]	[B]	[C]	[■]	56.	[A]	[■]	[C]	[D]
7.	[A]	[B]	[■]	[D]	57.	[■]	[B]	[C]	[D]
8.	[■]	[B]	[C]	[D]	58.	[A]	[B]	[■]	[D]
9.	[A]	[B]	[■]	[D]	59.	[A]	[B]	[C]	[■]
10.	[A]	[B]	[■]	[D]	60.	[■]	[B]	[C]	[D]
11.	[A]	[B]	[C]	[■]	61.	[■]	[B]	[C]	[D]
12.	[A]	[■]	[C]	[D]	62.	[A]	[■]	[C]	[D]
13.	[A]	[B]	[■]	[D]	63.	[A]	[■]	[C]	[D]
14.	[A]	[B]	[■]	[D]	64.	[A]	[B]	[■]	[D]
15.	[A]	[B]	[C]	[■]	65.	[A]	[B]	[C]	[■]
16.	[A]	[B]	[■]	[D]	66.	[A]	[■]	[C]	[D]
17.	[A]	[■]	[C]	[D]	67.	[■]	[B]	[C]	[D]
18.	[■]	[B]	[C]	[D]	68.	[A]	[B]	[■]	[D]
19.	[A]	[■]	[C]	[D]	69.	[A]	[B]	[C]	[■]
20.	[■]	[B]	[C]	[D]	70.	[■]	[B]	[C]	[D]
21.	[A]	[B]	[■]	[D]	71.	[■]	[B]	[C]	[D]
22.	[A]	[■]	[C]	[D]	72.	[A]	[B]	[C]	[■]
23.	[■]	[B]	[C]	[D]	73.	[A]	[B]	[■]	[D]
24.	[A]	[■]	[C]	[D]	74.	[A]	[■]	[C]	[D]
25.	[■]	[B]	[C]	[D]	75.	[A]	[B]	[■]	[D]
26.	[A]	[B]	[C]	[■]	76.	[A]	[B]	[C]	[■]
27.	[A]	[■]	[C]	[D]	77.	[A]	[B]	[■]	[D]
28.	[A]	[B]	[C]	[■]	78.	[■]	[B]	[C]	[D]
29.	[■]	[B]	[C]	[D]	79.	[A]	[B]	[C]	[■]
30.	[A]	[B]	[C]	[■]	80.	[A]	[■]	[C]	[D]
31.	[A]	[B]	[■]	[D]	81.	[A]	[B]	[■]	[D]
32.	[A]	[B]	[C]	[■]	82.	[A]	[B]	[■]	[D]
33.	[A]	[B]	[C]	[■]	83.	[A]	[B]	[C]	[■]
34.	[A]	[B]	[■]	[D]	84.	[A]	[■]	[C]	[D]
35.	[A]	[■]	[C]	[D]	85.	[A]	[B]	[■]	[D]
36.	[A]	[B]	[C]	[■]	86.	[A]	[■]	[C]	[D]
37.	[A]	[B]	[■]	[D]	87.	[A]	[■]	[C]	[D]
38.	[A]	[B]	[■]	[D]	88.	[A]	[■]	[C]	[D]
39.	[■]	[B]	[C]	[D]	89.	[A]	[B]	[■]	[D]
40.	[A]	[■]	[C]	[D]	90.	[■]	[B]	[C]	[D]
41.	[■]	[B]	[C]	[D]	91.	[A]	[B]	[■]	[D]
42.	[■]	[B]	[C]	[D]	92.	[■]	[B]	[C]	[D]
43.	[A]	[■]	[C]	[D]	93.	[A]	[B]	[C]	[■]
44.	[A]	[B]	[C]	[■]	94.	[A]	[B]	[C]	[■]
45.	[■]	[B]	[C]	[D]	95.	[A]	[■]	[C]	[D]
46.	[■]	[B]	[C]	[D]	96.	[A]	[■]	[C]	[D]
47.	[A]	[B]	[■]	[D]	97.	[■]	[B]	[C]	[D]
48.	[■]	[B]	[C]	[D]	98.	[A]	[■]	[C]	[D]
49.	[A]	[■]	[C]	[D]	99.	[■]	[B]	[C]	[D]
50.	[■]	[B]	[C]	[D]	100.	[A]	[B]	[■]	[D]

Graded By: ________

Regraded By (As Required): ________

QUESTION: 1

Given the following:

* The plant is at full power.
* A 200 gpm Feedwater line break downstream of Main Feedwater Check valve (2FW-5A) occurs.
* Containment temperature, pressure and humidity start rising.
* The plant is manually tripped.
* EFAS is automatically actuated.
* SG 'A' level is 35% Narrow Range and lowering.

Based on these conditions Steam Generator 'A' will depressurize and start an uncontrolled cooldown when:

A. Steam Generator 'A' level drops below 10% Narrow Range level.

B. The Main Feedwater Regulating valves are closed to 'A' Steam Generator.

C. The DP between B and A Steam Generators is > 70 psid and rising.

D. A MSIS occurs and Emergency Feedwater to 'A' Steam Generator is secured.

QUESTION: 2

(REFERENCE PROVIDED)

Given the following at full power:

* A Loss of the 500 KV Mabelvale line has occurred.
* Main Generator Megawatts are 1050 MWe.
* Main Generator Hydrogen Pressure is 60 psig.
* Main Generator Reactive Load is 400 MVARS Out "+".
* Main Generator Field Exciter Current is 4750 Amps.
* Generator Voltage Regulator is in AUTO.

Based on these conditions, in accordance with OP-2106.009, Turbine Generator Operations, Main Generator _____________ are required to be reduced to within the required limits by bumping the _______________________________ COUNTER-CLOCKWISE (Lower) to prevent overheating of the Main Generator.

A. MVARS; Terminal Voltage Adjust handswitch

B. FIELD AMPS; Terminal Voltage Adjust handswitch

C. MVARS; Generator Field Voltage Adjust handswitch

D. FIELD AMPS; Generator Field Voltage Adjust handswitch

QUESTION: 3

Given the following:

* Unit 2 has tripped due to an Excess Steam Demand event.
* Containment Building pressure is 24 psia and trending up.
* Annunciator 2K01-B7 "LO RELAY TRIP" for 2A1 is in alarm.
* Neither Emergency Feedwater (EFW) pump is available.
* "A" Steam Generator (S/G) Level is 200" (WR) and trending down.
* "B" Steam Generator (S/G) Level is 21% (NR) and trending down slowly.
* SPTAs have been completed.
* CRS has entered OP-2202.009, Functional Recovery EOP.
* Standard Attachment 50, Condensate Pump Start, has been completed up to the step to start a Condensate Pump.

Based on these conditions, a Steam Generator feed source will be accomplished by starting Condensate Pump __________ from the Control Room.

A. 2P-2A

B. 2P-2B

C. 2P-2C

D. 2P-2D

QUESTION: 4

From 100% Power, which of the following describes the primary reason for the automatic initiation of a SIAS during an Excess Steam Demand event?

A. To inject a large amount of water inventory into the RCS to overcome the rapid drop in the Pressurizer level.

B. To restore RCS Pressure back to the normal operating pressure band to overcome the rapid drop in RCS Pressure.

C. To ensure continued long term cooling of the Reactor core after the Excess Steam Demand event has terminated due to loss of SG inventory.

D. To inject a large amount of boric acid to the core to overcome the positive reactivity added due to the negative MTC to ensure adequate SDM is maintained.

QUESTION: 5

Given the following conditions:

* The plant is at 100% power.
* PZR PRESS CHANNEL SELECT Switch 2HS-4626 is selected to CH "A".
* PZR Pressure Transmitter 2PT-4626A fails high.
* Annunciator "Control CH 1/2 Pressure HI/LO" comes into alarm.

Which of the following actions is procedurally required to be taken FIRST to restore RCS pressure control in accordance with OP-2203.028, PZR Systems Malfunction?

A. Place PZR PRESSURE CONTROLLER 2PIC-4626A in manual to restore RCS pressure control.

B. Take manual control of PZR proportional heaters and spray valves to restore RCS pressure control.

C. Verify PZR Pressure Transmitter 2PT-4626B NOT failed and transfer 2HS-4626 "PZR PRESS CHANNEL SELECT" to CH "B".

D. Energize all PZR backup heaters to restore RCS pressure that has been lost due to the excess amount of PZR spray flow.

QUESTION: 6

Given the following:

* A Loss of Offsite Power has occurred and the LOOP EOP has been implemented.
* The CRS directs the ATC to isolate Letdown per the LOOP EOP.

When isolating Letdown, ______________ is the LEAST preferred isolation to use because ________________________________.

A. Letdown Isolation 2CV-4820-2; it could cause the Regen Heat Exchanger to overpressurize.

B. Regen HX Outlet 2CV-4823-2; it could cause the Regen Heat Exchanger to overpressurize.

C. Letdown Isolation 2CV-4820-2; it is NOT designed to isolate letdown at normal pressure/flow.

D. Regen HX Outlet 2CV-4823-2; it is NOT designed to isolate letdown at normal pressure/flow.

QUESTION: 7

Given the following:

* The plant is at 100% power.
* CCW Pump 2P33B is running supplying the CCW System.
* CCW Pump 2P33A is in standby.
* CCW Pump 2P33C is tagged out for motor replacement.

NOW the following alarms come in:

* Annunciator 2K12-E7 "2P-33B DISCH PRESS HI/LO".
* Annunciators 2K11-A1/A3/A5/A7 "RCP CCW DISC FLOW LO" .
* RCP Controlled Bleedoff (CBO) temperatures are 145°F and rising.

Which of the following actions is required to be taken FIRST based on these alarms and conditions?

A. Ensure CCW Pump 2P33A starts automatically then place 2P33B in Pull to Lock.

B. Restore CCW to RCPs using one second modulations on Return Valve 2CV-5255-1.

C. Enter the RCP Emergencies AOP and attempt to restore CCW flow to the RCPs.

D. Trip the Reactor, stop ALL RCPs, isolate RCP CBO flow and then GO TO SPTAs.

QUESTION: 8

A plant pressure spike has caused a PZR Code Safety to lift with the following conditions:

* The unit tripped on high RCS pressure.
* Two HPSI Pumps are running due to subsequent low RCS pressure after the safety lifted.

* NOW the stuck open Code Safety valve has partially closed but is still leaking at 60 gpm.
* The CRS has entered the LOCA EOP Procedure OP-2203.003

* RCS press is 1340 psia and rising.
* PZR level is now 80% and rising slowly.
* T-hot is 533°F, T-cold is 520°F.
* Ave CET temp is 540°F and stable.
* RVLMS Level 3 indicates WET.
* A S/G is 23%.
* B S/G is 22%.
* Feed water flow to A S/G is 250 GPM and rising.
* Feed water flow to B S/G is 250 GPM and rising.
* All RCPs have been secured.

Based on the given plant conditions, which of the following is the procedurally required action to take at this time and the reason for the action?

A. Override and throttle HPSI flow as needed using HPSI Injection MOVs to control RCS pressure and inventory.

B. Override and throttle HPSI flow as needed using HPSI Injection MOVs to restore Margin to Saturation (MTS) below the MAXIMUM Limit.

C. Continue full flow injection to the RCS with both HPSI pumps and monitor RCS pressure and inventory.

D. Continue full flow injection to the RCS with both HPSI pumps to restore Margin to Saturation (MTS) above the MINIMUM limit.

QUESTION: 9

Given the following plant conditions:

* Plant trips from 100% power due to a Loss of Offsite Power (LOOP).
* The Power Supply Breaker from 2D01 to 2D23, 2D-31, has tripped open.
* All attempts to reclose Supply Breaker 2D-31 have failed.
* #1 Emergency Diesel 2DG1 does not automatically start.
* CRS directs crew to place 2DG1 in service on the Vital 4160 VAC 2A3 bus.

Which of the following describes the starting and stopping of 2DG1 with these conditions?

A. 2DG1 can be started from the Control Room but has to be secured locally in the #1 EDG Room.

B. 2DG1 has to be started locally in the #1 EDG Room but can be secured from the Control Room.

C. 2DG1 has to be started and stopped locally in the #1 EDG Room.

D. 2DG1 can be started and stopped remotely from the Control Room.

QUESTION: 10

Given the following conditions:

* The plant is shutdown to replace a failed RCP seal.
* OP-1015.008, Unit 2 SDC Control, Attachment B, Verification of SDC System Alignment, has been completed.
* LPSI Pump 2P60A has been placed in service through SDC HX 2E-35A with the flows established during completion of OP-1015.008 Attachment B.
* The RCS has been drained to lowered inventory for seal replacement.
* RCS Temperature is 115°F and steady.

* NOW a loss of 125 VDC power to the SDC Temperature Control Valve 2CV-5093 solenoid causes the temperature control valve to go to its failed position.
* All other components in the SDC System remain the same as before the failure.

Based on these conditions, the SDC HX 2E-35 Inlet and Outlet RCS temperatures would ____________ as compared to a loss of power to 2CV-5093 without OP-1015.008 Attachment B having been completed.

A. lower slower

B. rise faster

C. rise slower

D. lower faster

QUESTION: 11

(1 PAGE OF SPDS SAFETY FUNCTION DISPLAY (SFD) CHANNEL 1 SCREEN ATTACHED)

Given the following:

* The plant has tripped from 100% power due to a transient.
* The current indications in the attached SPDS SFD screen exist after SPTAs are complete.
* SPDS SFD Channel 2 indications are the same as Channel 1.
* The Thot and Tcold temperatures have a lowering trend.

Referring to the attached SPDS indications, the correct recovery procedure would be ______________________ and proper Natural Circulation conditions ___________ been established.

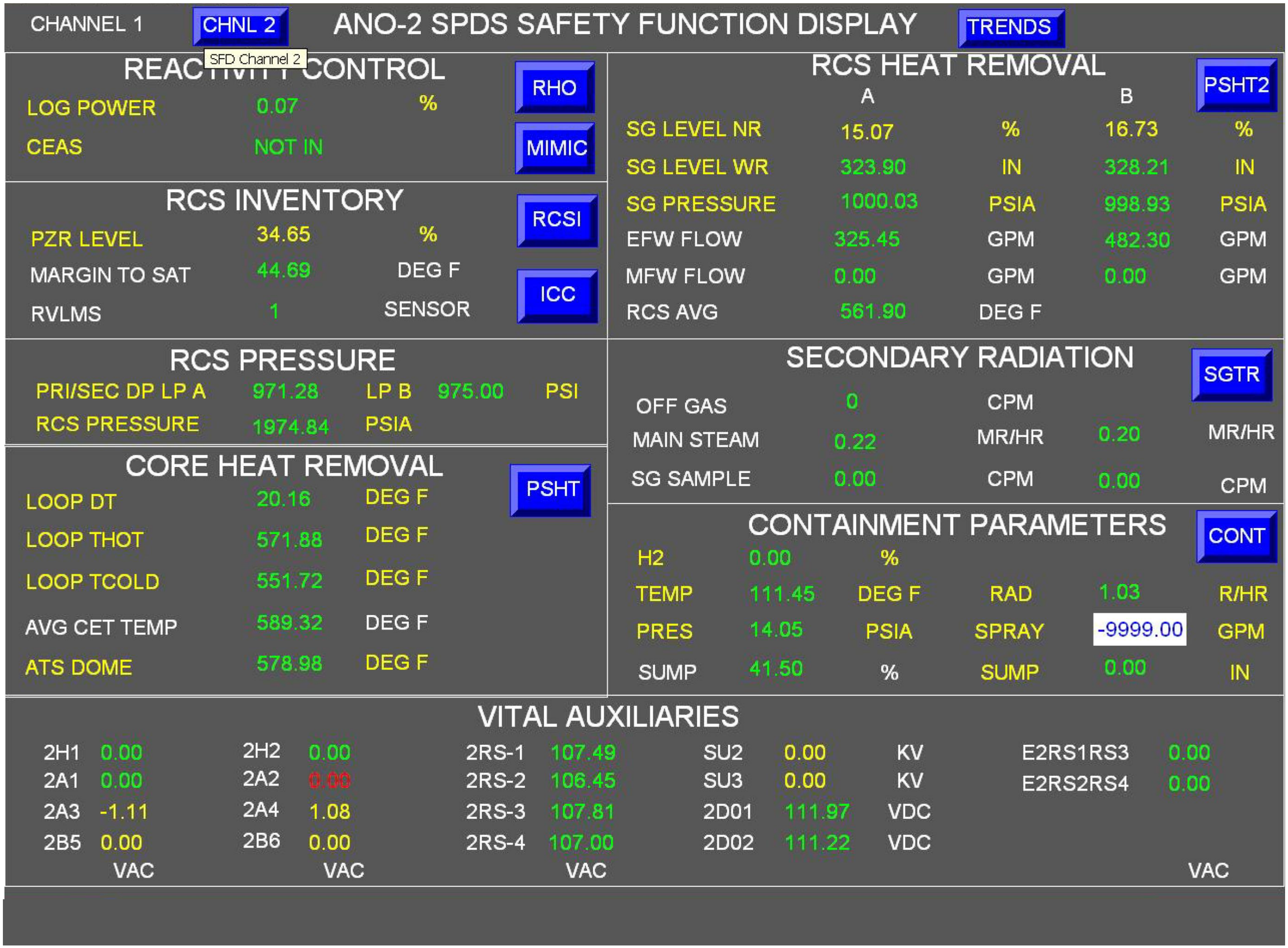
A. Natural Circulation Operations AOP; have not

B. Station Blackout EOP; have

C. Natural Circulation Operations AOP; have

D. Station Blackout EOP; have not

ATTACHMENT FOR QUESTION #11



QUESTION: 12

Given the following:

* Unit 2 has experienced a Station Blackout.
* The EOP Station Blackout OP-2202.008 has been implemented.
* Station Blackout conditions are expected to last for 2.6 hours
* The CRS directs Standard Attachment 25, Load Shedding of Vital Battery Loads," be performed.

Standard Attachment 25 requires that 120 VAC Vital Instrument Buses __________should be secured to ____.

A. 2RS3 & 2RS4; prevent ESF actuations upon restoration of AC power

B. 2RS3 & 2RS4; reduce the loads on the unit Vital 125 VDC batteries

C 2RS1 & 2RS2; prevent ESF actuations upon restoration of AC power

D. 2RS1 & 2RS2; reduce the loads on the unit Vital 125 VDC batteries

QUESTION: 13

Given the following at 100% Power:

* Annunciator 2K11-G3 "RCP BLEEDOFF FLOW HI/LO" comes into alarm.
* RCP 2P-32B Controlled Bleedoff (CBO) flow is 1.8 gpm on PMS and rising.
* RCP 2P-32B Seal Pressure indications are as follows:
  - Middle Seal Pressure is 1800 psia.
  - Upper Seal Pressure is 1575 psia.
  - Vapor Seal Pressure is 1530 psia.
* All other RCPs are operating with normal indications.
* OP-2203.025, RCP EMERGENCIES has been entered.

Which of the following is correct concerning RCP 2P-32B seals and the correct procedural action to take?

A. ONLY the upper seal has failed; Commence a plant shutdown and secure RCP 2P-32B after the Reactor is tripped.

B. The middle and upper seals have failed; Commence a plant shutdown and secure RCP 2P-32B after the Reactor is tripped.

C. ONLY the upper seal has failed; the reactor should be tripped from 100% power and then secure RCP 2P-32B.

D. The middle and upper seals have failed; the reactor should be tripped from 100% power and then secure RCP 2P-32B.

QUESTION: 14

Given the following:

* A small break LOCA has occurred inside Containment.
* The plant has been tripped.
* CIAS, SIAS, CCAS, CSAS, and MSIS have actuated.
* All RCPs have been secured.
* Total EFW flow to the SGs is currently 425 gpm.
* LOCA Recovery OP-2202.003 has been entered.
* Containment Temperature has risen to 250°F.
* Containment Pressure has risen to 30 PSIA.
* RCS Tcold is 550°F and stable.

Which of the following combinations of Steam Generator levels listed would be the LOWEST that would still meet the MINIMUM requirements to satisfy the LOCA EOP RCS Heat Removal Safety Function during this event?

A. SG 'A' 10% and SG 'B' 8%.

B. SG 'A' 15% and SG 'B' 13%.

C. SG 'A' 20% and SG 'B' 18%.

D. SG 'A' 23% and SG 'B' 22%.

QUESTION: 15

Given the following:

* Unit 2 is operating at 100% power.
* "A" Stator Water Cooling pump trips and the standby pump did NOT start.
* Main Turbine Runback is in progress.
* RCS pressure rises to 2450 psia and the Reactor trips.
* "A" and "B" Steam Generator levels are 14% and trending down.
* EFAS 1 and EFAS 2 have failed to actuate.
* NO operator action is taken.

The Reactor trip was caused by ________________and Emergency Feedwater Flow Control Valves (FCVs) will_________________________________.

A. CPC Auxiliary trip; cycle open and closed to maintain SG levels 22.2% to 25%

B. CPC Auxiliary trip; open and feed the SG levels to 80% and then reclose

C. DSS Trip; cycle open and closed to maintain SG levels 22.2% to 25%

D. DSS trip; open and feed the SG levels to 80% and then reclose

QUESTION: 16

Given the following:

* The plant is at full power.
* Annunciator 2K12-A8, INSTR AIR PRESS HI/LO, comes in.
* Instrument Air Header pressure has lowered to 55 psig and stabilizing.
* The Loss of Instrument Air AOP OP-2203.021 has been entered.
* CNTMT Chill Water Isolation Valves 2CV-3851-1 and 2CV-3852-1 have failed CLOSED.
* I&C has commenced monitoring CEA CEDM Coil Temperatures.
* Restoration of Instrument Air and CNTNT Chill water is not imminent.

Which of the listed CEA coil temperature would be the LOWEST temperature that would require commencement of a plant shutdown in accordance with the Loss of Instrument Air procedure OP-2203.021?

A. 411°F

B. 435°F

C. 474°F

D. 504°F

QUESTION: 17

Given the following:

* Plant is operating at 100% power at 450 EFPD.
* The Main Turbine Load Limit Pot Light is ON.
* Core Operating Limit Supervisory System (COLSS) is Operable.

Based on the above conditions, which of the following set of parameters would indicate a steam line rupture of the Main Steam line going to the Main Feedwater Pumps?

A. COLSS Indicated Reactor power lowering; Turbine First Stage Pressure remains the same.

B. COLSS Indicated Reactor power rising; Turbine First Stage Pressure lowering.

C. COLSS Indicated Reactor power lowering; Turbine First Stage Pressure lowering.

D. COLSS Indicated Reactor power rising; Turbine First Stage Pressure remains the same.

QUESTION: 18

Consider the following:

* Unit 2 is in Mode 6.
* GREEN TRAIN is protected.
* "B" Shutdown Cooling train is in service.
* Core reload is in progress.
* The Refueling Canal is at 401.2 feet and 2675 ppm boric acid concentration.
* Makeup to the Refuel Canal from the RWT in progress to raise level to 401.5 feet.
* While filling the Refuel Canal, the RWT level was noted to be at 7.2% level.
* Boric Acid MU Tank (BAMT) 2T-6A is tagged and drained for heater replacement.
* Batch additions of boric acid have been made to BAMT 2T-6B.
* BAMT 2T-6B is on recirc using BAM Pump 2P-39B for a sample.
* BAM Pump 2P-39B breaker trips on overcurrent due to a pump seal/bearing failure.
* BAMT 2T-6B level is currently at 34% and slowly trending down.

Which of the following actions is required to be performed FIRST in accordance with TRM 3.1.7 Borated Water Sources - Shutdown?

A. Suspend any operation that could add coolant to the RCS with less than the TS required boron concentration

B. Start BAM Pump 2P-39A and continue to recirc 2T-6B to obtain chemistry operability sample.

C. Add additional batches of boric acid solution to BAM Tank 2T-6B to restore 2T-6B to operable level status.

D. Perform blended MU to the Refueling Water Tank to restore RWT level to operable level status.

QUESTION: 19

Which of the following statements best describes the purpose and function of Main Feedwater Reactor Trip Override (RTO) signal after a Reactor Trip?

A. To rapidly add feedwater to the SGs to ensure an adequate RCS heat sink.

B. To slowly add feedwater to the SGs to prevent overcooling the RCS.

C. To rapidly add feedwater to the SGs to prevent an EFAS actuation.

D. To slowly add feedwater to the SG to limit thermal stresses to the feed rings.

QUESTION: 20

(REFERENCE PROVIDED)

Given the following:

* The plant is at 100% power steady state.
* Annunciator 2K11-A10 "SEC SYS RADIATION HI" comes in.
* Main Steam Line Rad Monitors start rising.
* SG Tube Leak N-16 Monitors start trending up.
* The CRS enters OP-2203.038, Primary to Secondary Leakage due to a calculated 29 gpm RCS leakrate.

If RCS temperature, and Pressurizer level remain constant during the next 10 minutes, the VCT will drop _________% and the procedural required action to mitigate the Steam Generator tube leak is to ____.

A. 8.6; commence a rapid plant shutdown to Mode 3

B. 5.4; commence a rapid plant shutdown to Mode 3

C. 8.6; trip the Reactor, actuate SIAS/CCAS, and commence SPTAs

D. 5.4; trip the Reactor, actuate SIAS/CCAS, and commence SPTAs

QUESTION: 21

Given the following plant conditions:

* The plant is at full power.
* Pressurizer Level Control System master controller is in AUTO REMOTE.
* Pressurizer Level Control is selected to "CHANNEL 4627-A".
* Pressurizer Heater Low Level Cutout Switch is selected to Both "A & B".
* Charging Pump Selector Switch, 2HS-4868, is in "A & B".
* Pressurizer Level Indicator 2LT-4627-1 fails high.
* No operator action is taken.

Based on these conditions Backup Charging Pumps 'A' and 'B' will ________________ and all the Pressurizer Heaters will ________________.

A. start; energize.

B. start; de-energize.

C. get a stop signal; energize.

D. get a stop signal, de-energize.

QUESTION: 22

Given the following:

* Plant power ascension is in progress following a refueling outage.
* Reactor power is at 64% and slowly rising.
* SDBCS is in its normal line up.
* Condenser vacuum has started degrading.
* Annunciators 2K03-A3/A4 "2E11A/B Pressure HIGH" come in.
* Condenser vacuum is 5.7 inches Hg absolute and rising.

In accordance with OP-2203.019, Loss of Condenser Vacuum, the _______________ is procedurally REQUIRED to be tripped if condenser vacuum exceeds a MAXIMUM of ___________ inches Hg absolute.

A. Turbine; 7.0

B. Reactor; 7.0

C. Turbine; 6.0

D. Reactor; 6.0

QUESTION: 23

Given the following:

* Power to Green Vital 4160 VAC ESF Bus 2A4 has been lost.
* The Green EDG fails to start and load Electrical Bus 2A4.
* The Plant subsequently trips and SPTAs are in progress.
* All but one (1) CEA rod bottom lights are illuminated Green.
* CEACs 1 and 2 agree with the rod bottom light indications.
* Reactor power is lowering.

Based on the above conditions, SPTAs directs ____.

A. commencing emergency boration using gravity feed valves

B. depressing BOTH "Reactor Trip" pushbuttons on 2C03

C. commencing emergency boration using either BAM Pump

D. opening the breakers for the MG sets 2B712 and 2B812

QUESTION: 24

Given the following:

* Unit 2 is in a refueling outage.
* A report comes in from a qualified operator that a Class 'B' fire is in progress in the basement of Containment.

What kind of hazards exist to the personnel during this fire and which action can be taken to reduce this hazard?

A. Smoke inhalation from burning trash; evacuate the containment building area.

B. Intense heat from burning liquid fuel; ensure fire water aligned to the containment.

C. Potential electrocution; de-energize the applicable power supply to containment.

D. Flying sparks from burning metal; evacuate the local area on the 335 elevation.

QUESTION: 25

The following plant conditions exist:

* Mode 6 with core re-load in progress
* The Control Room receives a report that as a spent fuel assembly was being inserted into the core, the refueling machine started moving on its own.
* Bubbles are rising around the fuel assemblies in the core.
* Low Range Rad Monitors on the 404 elevation are indicating a large rise in radiation.
* OP-2502.001, Refueling Shuffle, Att. M, Refueling Accident is in progress.

Per the procedure, to inform the required individuals to commence an evacuation, the Evacuation Warning Handswitch, 2HS-EVAC on panel ________ should be taken to ______________.

A. 2C22; "CTMT" ONLY

B. 2C20; "CTMT" ONLY

C. 2C22; "PLANT" and "CTMT"

D. 2C20; "PLANT" and "CTMT"

QUESTION: 26

Given the following:

* The plant is in Mode 6 with core reload in progress.
* I&C is performing the required 7 day Channel Functional Test on SU Channel 2 NI
* I&C reports that the power supply to SU Channel 2 NI was below the acceptable allowed voltage value during the surveillance.

Based on this report the SU Channel #2 NI would be __________________ and the count rate in the Control Room would read __________ counts than actual.

A. more sensitive to gammas; higher

B. more sensitive to neutrons; higher

C. less sensitive to gammas; lower

D. less sensitive to neutrons; lower

QUESTION: 27

Given the following:

* The plant is at 50% during power ascension coming out of a refueling outage.
* Annunciator "LETDOWN RADIATION HI/LO" (2K12-A1) comes in Alarm.
* Letdown Gross Activity Monitor (2RITS-4806-A) reads 5.5E+5 CPM and rising rapidly.
* Letdown I-131 Activity Monitor (2RITS-4806-B) reads normal at 350 cpm on the 1E3 scale.

Which of the following events occurred for the given plant conditions?

A. Failed fuel

B. RCS crud burst

C. RCS zinc addition

D. Letdown Demineralizer exhausted

QUESTION: 28

During a Large Break LOCA Design Basis Accident, the Containment Spray Pumps will provide cooling flow to the Containment from the RWT until the RWT drops below __________. At this point, the level of water in Containment should be no less than _______________________________ to ensure adequate NPSH for the Containment Spray Pumps post RAS.

A. 40%; 86% Containment Sump Level indication on 2C-33

B. 40%; 86 inches Containment Sump Flood Level on 2C-16/17

C. 6%; 86% Containment Sump Level indication on 2C-33

D. 6%; 86 inches Containment Sump Flood Level on 2C-16/17

QUESTION: 29

Following a Reactor Trip, a check of the Vital Auxiliaries Safety Function status indicates that:

* #1 Emergency Diesel Generator (2DG1) is Running
* 2DG1 Frequency is 60 Hz
* 2DG1 Voltage is 4165 V
* 4160V Vital AC Bus 2A3 is Deenergized
* Annunciator 2K08-B3 "2A3 LO RELAY TRIP" has actuated

2DG1 should be secured to prevent ____ .

A. overheating the diesel engine

B. uneven lower crankshaft bearing wear

C. buildup of unburned fuel in the exhaust manifold

D. running the diesel engine without Jacket Cooling Water flow

QUESTION: 30

Loop 2 Component Cooling Water (CCW) is normally supplied by 2P-33C, "C" Component Cooling Water (CCW) pump, which is powered from ________ and its backup pump, 2P-33B, "B" CCW pump, is powered from ________ .

A. 2B-1; 2B-2

B. 2B-2; 2B-1

C. 2B-2; 2B-7

D. 2B-7; 2B-2

QUESTION: 31

Given the following at full power:

* Tags have been cleared on the "C" HPSI Pump, 2P-89C, Breaker 2A-407.
* The breaker has been racked up with the following indications on 2C-16:

Green light is ON
White light is OFF
Red Light is OFF
Amber light is ON

Based on these indications, the 2P-89C Green Train Breaker, 2A-407, closing springs are _____________ and the Kirk Key lock is _____________________.

A. charged; locked out/NOT available

B. charged; unlocked/available

C. discharged; locked out/NOT available

D. discharged; unlocked/available

QUESTION: 32

During a LOCA, if Containment pressure rises to a MINIMUM of __________ psia, then a SIAS signal is generated and the SIAS cannot be reset in accordance with Standard Attachment 13, SIAS RESET, until Containment Pressure drops below a MAXIMUM of __________ psia.

A. 23.3; 23.0

B. 23.3; 22.5

C 18.3; 18.0

D. 18.3; 17.5

QUESTION: 33

Given the following:

* The plant is at full power.
* Annunciator 2K01-D11 "BATTERY 2D12 NOT AVAIL" comes into Alarm.
* Annunciator 2K01-A11 "CONT CENTER 2D02 UNDER VOLT" comes into alarm.
* 2D02 Bus Voltage indicates "0 VDC" on SPDS.

Given these conditions, which of the following list the correct procedure to be used to mitigate the consequences of this event on the PPS system and the correct action to take on the PPS System?

A. Inadvertent SIAS OP-2203.018; Place PPS Channels 1 OR 3 in Bypass.

B. Inadvertent SIAS OP-2203.018; Place PPS Channels 2 OR 4 in Bypass.

C. Loss of 125V DC OP-2203.037; Place PPS Channels 1 OR 3 in Bypass.

D. Loss of 125V DC OP-2203.037; Place PPS Channels 2 OR 4 in Bypass.

QUESTION: 34

Given the following conditions:

* The plant is at 100% power.
* The CRS directs adding 10 gallons of boric acid to the RCS.
* BAM Pump Select Switch 2HS-4911-2 is selected to BAM Pump 2P-39A.
* The Boric Acid Flow Controller 2FIC-4926 has been set to a flowrate of 10 gpm.
* The Mode Select Switch 2HS-4928 is taken to "BORATE".
* The Boric Acid Batch Controller 2FQIS-4926 has been set to 10 gallons.

When the Mode Select switch is taken to BORATE, the BAM pump 2P-39A will be started __________ and when the 10 gallons of boric acid has been added to the RCS, 2P-39A will be stopped __________.

A. manually; manually

B. manually; automatically

C. automatically; manually

D. automatically; automatically

QUESTION: 35

Given the following:

* The plant is at full power.
* Annunciator 2K03-C9 "FW PUMP SUCT PRESS LO" comes in.
* The Main Feed Water Pump suction crossover line has ruptured.
* Steam is filling up the Turbine Building.
* Turbine Building Sumps LEVEL HIGH alarms 2K12-J8/J9 have come in.
* Main FW Pump Suction pressure is 444 psig and lowering rapidly.

Based on the above conditions, the correct action/procedure needed to mitigate the event would be to ____ and if the suction pressure to the MFW Pumps continues to drop, MFW Pumps will trip at a setpoint of _________ psig after a 30 second time delay.

A. commence a rapid Turbine load reduction using OP-2203.027 Loss of MFW Pump; 325

B. trip the Reactor and secure Feed/Condensate using OP-2203.051 Internal Flooding; 325

C. commence a rapid Turbine load reduction using OP-2203.027 Loss of MFW Pump; 425

D. trip the Reactor and secure Feed/Condensate using OP-2203.051 Internal Flooding; 425

QUESTION: 36

(1 PAGE OF PMS SCREEN OF CONTAINMENT PARAMETERS ATTACHED)

Given the following:

* The plant is at full power in the middle of April on dayshift at 0930.
* The running Main Chiller 2VCH-1A trips and will not restart.
* Main Chiller 2VCH-1B was started but trips after every start attempt.
* At 10:30 Annunciator 2K10 A-7 " CNTNT TEMP/HUMIDITY HI" comes in.
* All 4 Containment Fan Coolers are running.
* Current conditions in containment at 10:30 are as shown on the attached PMS screen.

Based on the current containment conditions, which of the following would be correct in accordance with Annunciator 2K10 A-7 " CNTNT TEMP/HUMIDITY HI"?

A. Restore the Containment internal pressure/temperature within the limits of T.S 3.6.1.4, Containment Internal Pressure/Temperature, within one hour.

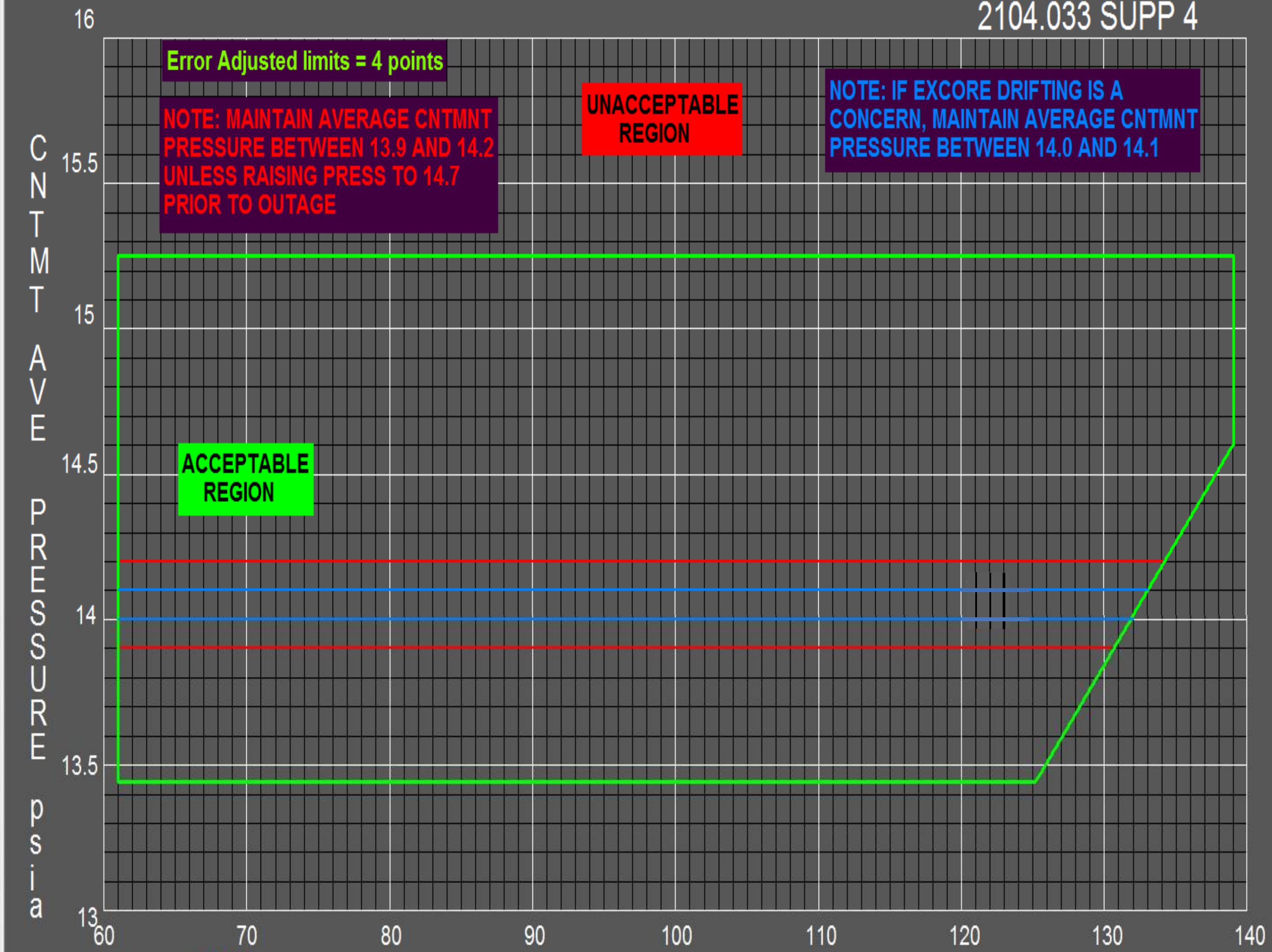
B. Raise Containment internal pressure to approximately 14.6 psia to provide margin to prevent exceeding TS. 3.6.1.4, Containment Internal Pressure/Temperature, limits.

C. Lower Containment internal temperature/humidity by starting all available CEDM cooling and Containment Recirculation Fans within one hour to prevent exceeding TS 3.6.1.4, Containment Internal Pressure/Temperature, limits.

D. Lower Containment internal temperature/humidity by aligning Service Water to Containment Fan Coolers to prevent exceeding TS. 3.6.1.4, Containment Internal Pressure/Temperature, limits.

ATTACHMENT FOR QUESTION # 36

2104.033 SUPP 4



CONTAINMENT AVG AIR TEMP °F

DATE/TIME VERIFIED BY ______ INIT.

	P5601-1	14.11	PSIA
	P5602-2	14.08	
	P5603-3	14.08	
	P5604-4	13.96	
AVG	P5601-A	14.06	
	T5605-5	129.6	°F
	T5606-6	129.8	
AVG	T5600	129.7	

T.S.3.6.1.4 - CIRCLE YES IF ALL 4 CNTMNT PRESS COMPUTER POINTS ARE AVAILABLE AND PLOTTED POINT IS WITHIN THE ACCEPTABLE REGION, OTHERWISE CIRCLE NO. IF NO CIRCLED, THEN PERFORM 2104.033 SUPP 4. YES NO

PERFORMED BY:________________ REVIEWED BY:________________

ATTACHMENT FOR QUESTION # 36

QUESTION: 37

Which of the following describes the main purpose of the Flywheel on the RCPs?

A. To prevent a pump overspeed transient during a cold leg break in the suction piping.

B. To prevent and/or mitigate vapor seal failures when securing RCPs with degraded seals.

C. To improve RCS flow characteristics through the core during a Loss of Offsite power.

D. To improve the cooling air flow to the upper part of the RCP Motor and thrust bearing.

QUESTION: 38

Given the following with the unit at full power:

* "INSTR AIR PRESS HI/LO" annunciator (2K12 A8) comes in alarm.
* OP-2203.021, Loss of IA AOP, has been implemented.
* Instrument Air pressure is 34 psig on 2PIS-3013 and slowly lowering .
* Unit 1 informs Unit 2 that Unit 1 IA Header Pressure is 70 psig and slowly lowering.
* D/P between IA HDR pressure and IA receiver pressure is 3 psid and steady.

Based on these conditions which of the following actions is REQUIRED to be taken as directed by the Loss of IA AOP?

A. Close the IA to Unit 1 Isolations 2CV-3004 and 2CV-3015.

B. Open the IA Air Dryer Bypass Isolation Valve 2IA-8.

C. Trip the Reactor and perform SPTA procedure 2202.001.

D. Align Service Water Return headers to Lake Dardanelle.

QUESTION: 39

Given the following:

* The plant is at full power with indications of a Pressurizer Safety Valve leaking.
* The Quench Tank temperature has risen above its alarm limit.
* The CRS directs the crew to cool the Quench Tank using the normal feed and bleed method.

To ensure the sparger in the Quench Tank remains covered during this evolution, Quench Tank level should be maintained greater than a MINIMUM of ________ with makeup water aligned while draining the Quench Tank to the _________________.

A. 75%; Reactor Drain Tank

B. 75%; Containment Sump

C. 87%; Reactor Drain Tank

D. 87%; Containment Sump

QUESTION: 40

Given the following:

* The plant is at full power.
* A Loss of Main Feed Water causes a plant trip.
* An EFAS signal was generated.
* EFW Pump 2P-7A tripped on overspeed during start.
* OP-2202.001 SPTAs has been entered.
* SG A level is 6% NR and slowly rising.
* SG B level is 4% NR and slowly rising.
* EFW Pump 2P-7B started on the EFAS signal.

Which of the following combinations of EFW flow would indicate that EFW Pump 2P-7B is adequately providing the LOWEST amount of flow required to satisfy the RCS Heat Removal Safety Function Requirements during SPTAs? (assume no EFW recirculation flow)

A. A SG 145 gpm; B SG 145 gpm

B. A SG 248 gpm; B SG 242 gpm

C. A SG 234 gpm; B SG 244 gpm

D. A SG 155 gpm; B SG 152 gpm

QUESTION: 41

Given the following:

* Plant has just shutdown and is cooling down for a refueling outage.
* Electrical 4160 Vital AC Bus 2A3 maintenance outage in progress.
* RCS Pressure is currently 80 psig.
* The running SDC Pump 2P-60B trips on breaker fault and cannot be restarted.

To restore SDC to service ___________________ should be aligned for SDC and placed in service after RCS pressure has been ____.

A. Spray Pump 2P-35B; reduced to less than 50 psig to prevent RCS inventory loss

B. Spay Pump 2P-35A; reduced to less than 50 psig to prevent RCS inventory loss

C. Spray Pump 2P-35B; raised to greater than 100 psig to ensure adequate pump NPSH

D. Spray Pump 2P-35A; raised to greater than 100 psig to ensure adequate pump NPSH

QUESTION: 42

During full power operations, if the VCT level drops to the setpoint of __________%, the RWT to Charging Pump Suction 2CV-4950-2 will open to provide a water source to the CVCS system and RCS T-ave will_____________.

A. 5; lower

B. 5; not change

C. 7; lower

D. 7; not change

QUESTION: 43

Given the following:

* The plant has tripped from 100% power.
* RCS Pressure is 1800 psia and dropping.
* Steam Generator pressures are 700 psia and dropping.
* Containment pressure is 19.3 psia and rising.

Based on the above conditions, when checking the status of the Containment Cooling Fans, the proper status would have Chilled Water _____________ and Service Water _______________.

A. aligned; aligned

B. isolated; aligned

C. aligned; isolated

D. isolated; isolated

QUESTION: 44

Given the following:

* The Reactor Trips and SPTAs are completed.
* The CRS has entered OP-2202.002 Reactor Trip Recovery (RTR) EOP.

NOW

* While performing RTR actions, an SIAS starts both trains of ESFAS Pumps.
* RCS Pressure is 2157 psia and slowly rising.
* Containment Pressure is 14.4 psia and stable.

Based on these conditions, which of the following is the required direction to take in accordance with OP-1015.021 ANO-2 EOP/AOP Users Guide to address the SIAS?
(LMFRP = Lower Mode Functional Recovery Procedure)

A. Exit RTR EOP Now and GO TO LMFRP and perform the applicable EOP actions.

B. Exit RTR EOP Now and GO TO Inadvertent SIAS and perform applicable AOP actions.

C. Continue RTR actions while REFERENCING LMFRP and perform applicable EOP actions.

D. Continue RTR actions while REFERENCING Inadvertent SIAS and perform applicable AOP actions

QUESTION: 45

During an EFAS automatic start of the 2P-7A EFW Pump, which of the following is the correct sequence for the Main Steam to 2P-7A EFW Pump Isolations 2CV-0205 and 2CV-0340 and the primary reason for the sequence?

A. 2CV-0205 will open first then 2CV-0340 after a time delay to prevent a turbine overspeed.

B. 2CV-0340 will open first then 2CV-0205 after a time delay to prevent a turbine overspeed.

C. 2CV-0205 will open first then 2CV-0340 after a time delay to reduce DP across 2CV-0340.

D. 2CV-0340 will open first then 2CV-0205 after a time delay to reduce DP across 2CV-0205.

QUESTION: 46

Given the following:

* Plant is at 100% Power.
* Annunciator 2K11-G5 "RCP BLEEDOFF PRESSURE HI" comes in.
* RCP Controlled Bleedoff Containment Isolation Valve 2CV-4847-2 has failed CLOSED.

Based on the given conditions, RCP Controlled Bleedoff (CBO) flow will be directed to the ___________.

A. Quench Tank

B. Reactor Drain Tank

C. Containment Sump

D. Volume Control Tank

QUESTION: 47

Consider the following:

* Unit 2 has tripped from 100% power due to a 400 gpm LOCA.
* SIAS has automatically initiated.
* 2 RCPs have been secured.
* Service Water was realigned to CCW HXs using Exhibit 5 during SPTAs.
* The LOCA gets larger over time.
* 45 minutes after the trip, a Recirculation Actuation Signal (RAS) comes in.

After the RAS, Service Water ______________ be aligned to the CCW Heat Exchangers and should be _______________ aligned to the Shutdown Cooling Heat Exchangers.

A. should; automatically

B. should; manually

C. should not; automatically

D. should not; manually

QUESTION: 48

When an EDG is paralleled with offsite power in an under excited condition, reactive load could be very large as indicated by _________________ and the EDG Normal Operating Procedure 2104.036 can be used to mitigate this condition by ensuring "Incoming Generator" voltage is ______________ than "System Running" voltage on the 2C-33 Indicators prior to closing the EDG output breaker.

A. leading (IN) KVARs; 100 volts higher

B. leading (IN) KVARs; 10 volts higher

C. lagging (OUT) KVARS; 100 volts higher

D. lagging (OUT) KVARS; 10 volts higher

QUESTION: 49

With the plant at full power the Pressurizer Spray Valves 2CV-4651 and 2CV-4652 can be operated from the Control Room only if the PZR Spray Valve Hand Switches 2HS-4651D and 2HS-4652D on the Remote Shutdown Panel are in the __________ position.

A. 2C03

B. 2C04

C. 2C09

D 2C80

QUESTION: 50

Reactor Trip Circuit Breakers, TCBs, 4 and 8 have two redundant tripping circuits that ensure the TCBs open when a signal is received from RPS. TCBs 4 and 8 tripping circuits are powered from ________________.

A. 2D36

B. 2D25

C. 2RS2

D. 2Y2

QUESTION: 51

The bus power supply to the 2P-7B Electric Emergency Feedwater Pump is:

A. 2A1 4160 VAC Bus

B. 2A2 4160 VAC Bus

C. 2A3 4160 VAC Bus

D. 2A4 4160 VAC Bus

QUESTION: 52

Given the following:

* Unit 2 has tripped from full power due to a Large break LOCA.
* RCS Pressure is 150 psia and slowly lowering.
* Containment pressure is 28 psia and lowering.
* Both Trains of ECCS Equipment are in operation as designed.

NOW

* A Recirculation Actuation Signal (RAS) is generated.

With no operator action, which of the following is the correct component positions to verify after the RAS actuations have completed?

A. LPSI pumps trip, HPSI and Containment Spray pumps remain running, RWT outlet isolations remain Open.

B. Containment Spray pumps trip, HPSI and LPSI pumps remain running, RWT outlet isolations Close.

C. LPSI pumps trip, HPSI and Containment Spray pumps remain running, RWT outlet isolations Close.

D. Containment Spray pumps trip, HPSI and LPSI pumps remain running, RWT outlet isolations remain Open.

QUESTION: 53

In accordance with AOP-2203.025, RCP Emergencies, which of the following is the primary reason for the time limit for operating RCPs during a Loss of CCW event?

A. Overheating of the RCP Motor.

B. Overheating of the RCP Seals.

C. Overheating of the RCP Hydrostatic BRG.

D. Overheating of the RCP Lube Oil.

QUESTION: 54

Given the following:

* Unit 2 is at full power operation.
* The WCO is tasked with quantifying a 2P-89B HPSI pump seal leak in the 'B' ESF room.
* The WCO has signed onto 2017-2002 Operations Activities RWP Task 1 Radiation Areas.
* The EAD setpoint limits for RWP 2017-2002 are 4 mrem dose and 40 mrem/hr dose rate.
* The WCO has just entered the 'B' ESF room and the calculated stay time is 30 minutes.

* NOW the CVCS Process Radiation Monitor 2RE-4806 indicates rising activity in the RCS.
* Based on Chemistry input, OPS has started additional Charging Pumps to clean up the RCS.

Based on these conditions, the stay time for the WCO will be ___________ before the RWP allowed dose limit is reached due to the change in radiation dose rates caused by ____________.

A. shorter; RCS N-16 gammas spend less time in the Letdown decay chamber

B. longer; RCS N-16 gammas spend more time in the Letdown decay chamber

C. shorter; more RCS radioactivity buildup in the Letdown Filter raising 'B' ESF room dose

D. longer, RCS radioactivity will be removed faster by the in-service Letdown Demineralizer

QUESTION: 55

Given the following:

* Unit 2 is at full power.
* The Main Turbine trips.
* Electrical loads transfer to offsite as designed.
* Offsite Grid Voltage starts to degrade.

Without any operator action, which of the following combination of voltages would be the MAXIMUM voltages on Electrical 480 Volt Buses 2B5 and 2B6 that would cause BOTH Emergency Diesel Generators to START and LOAD their respective Safety Bus. (assume all of the listed voltages have dropped to only the values listed and have been at these values for greater than 15 seconds)

A. 440 VAC on 2B5; 441 VAC an 2B6

B. 435 VAC on 2B5; 438 VAC an 2B6

C. 426 VAC on 2B5; 428 VAC an 2B6

D. 419 VAC on 2B5; 422 VAC an 2B6

QUESTION: 56

Given the following:

* The plant is at 100% Power.
* The Cooling Tower Basin Level Control Valve 2CV-1460 is in Automatic.

If the Circulating Water cooling tower basin lowers below its current setpoint of 80%, then the Cooling Tower Basin Level Control Valve 2CV-1460 will __________ to provide __________ pressure on the Service Water return header.

A. open; less

B. close; more

C. open; more

D. close; less

QUESTION: 57

Given the following:

* Plant is at 100% Power.
* The #1 MSIV fails closed.

Right after the MSIV closure and prior to the plant trip , the level in the ________ Steam Generator will lower, and the pressure in the ________ Steam Generator will lower.

A. 'A'; 'B'

B. 'B'; 'A'

C. 'A'; 'A'

D. 'B'; 'B'

QUESTION: 58

Given the following at full power.

* A severe fire has developed and completely engulfed the #1 EDG Room.
* The #1 EDG Fire Protection Valve 2UAV-3231 fails and CANNOT be opened manually.
* The Shift Manager has directed a Rapid Power Shutdown.

Based on the above conditions, boration should be commenced from the ______________________ and the procedure that is required to be entered for these conditions would be ____________________.

A. RWT directly to the Charging Pumps; OP-2203.014, Alternate Shutdown

B. BAMTs using the BAM Pumps; OP-2203.014, Alternate Shutdown

C. RWT directly to the Charging Pumps; OP-2203.049, Fires in Areas Affecting Safe Shutdown

D. BAMTs using the BAM Pumps; OP-2203.049, Fires in Areas Affecting Safe Shutdown

QUESTION: 59

Given the following plant conditions:

* Plant is in Mode 5 making preparations to refuel the Reactor.
* RCS is in lowered inventory preparing to install SG nozzle dams.
* Containment Purge System is in service.
* When the 1st set of SG Manways are removed, the Control Room receives Annunciator 2K11-D10 " Process Gas Radiation HI/LO".
* On 2C-25, the Gas Monitor for the Containment Purge, 2RITS-8233, is reading above setpoint.

Which of the following valves will automatically actuate to secure Containment Purge?

A. All three (3) Containment Purge Exhaust isolation valves will go closed ONLY.

B. All six (6) Containment Purge Supply AND Exhaust Isolation valves will go closed.

C. ONLY the Inside-Inside Containment Purge Supply AND Exhaust isolations go closed.

D. ONLY the Outside-Outside Containment Purge Supply AND Exhaust Isolations go closed.

QUESTION: 60

Given the following at full power:

* One Charging Pump is running.
* Annunciator "RRS TROUBLE" comes in.
* PZR level setpoint input to PZR Level Control System is reading 55%.
* Annunciator 2K12 C-1 "LETDOWN HX 2E29 OUTLET TEMP HI" comes in.
* Letdown HX Outlet temperature on 2TIC-4815 is reading 150°F and slowly lowering.

Based on these conditions, the Letdown Rad Monitor Valve 2CV-4804 would ________________________ Letdown Rad Monitor and after restoring Letdown temperature back to normal, the isolation/bypass valve will be restored ____.

A. isolate Letdown to the; manually using PZR Malfunction OP 2203.028

B. bypass Letdown around the; manually using CVCS Operations OP 2104.002

C. isolate Letdown to the; automatically and verified with CVCS Operations OP 2104.002

D. bypass Letdown around the; automatically and verified with PZR Malfunction OP 2203.028

QUESTION: 61

The High Range Area Radiation Monitor 2RITS-8912, located on the 404 CNMNT South West End Refuel Deck has a detector that is operated in the ______________ region of the Gas Amplification Curve and is designed to detect _____________ radiation events.

A. Geiger- Mueller; gamma only

B. Geiger- Mueller; gamma and neutron

C. Ion Chamber; gamma only

D. Ion Chamber; gamma and neutron

QUESTION: 62

Which of the following Radiation monitors would detect a small leak upstream of Letdown Isolation valve 2CV-4820-2 at 30% Reactor Power?

A. Letdown Line Radiation Monitors 2RE-4806.

B. Containment Atmosphere Monitor 2RITS-8231.

C. Containment Purge Radiation Monitor 2RITS-8233.

D. Loop 2 CCW Return Radiation Monitor 2RITS-5202.

QUESTION: 63

When performing OP-2106.009, Turbine Generator Operations, section 14.0 Turbine Roll, when 100 RPM is selected on the Speed Set controller, the proper Main Turbine Stop Valves opening sequence is:

A. #1 Stop Valve bypass valve opens to equalize pressure then #1 Stop Valve opens followed by #2, #3 and #4 Stop Valves simultaneously.

B. #2 Stop Valve bypass valve opens to equalize pressure then #2 Stop Valve opens followed by #1, #3 and #4 Stop Valves simultaneously.

C #1 Stop Valve bypass valve opens to equalize pressure then #1 Stop Valve opens followed by the opening of #2, #3 and #4 Stop Valves in sequence.

D. #2 Stop Valve bypass valve opens to equalize pressure then #2 Stop Valve opens followed by the opening of #1, #3 and #4 Stop Valves in sequence.

QUESTION: 64

Given the following:

* The plant is stabilized at 12% Reactor power following a forced outage at 250 EFPD.
* The Main Turbine has been tied to the grid currently at 50 MW electrical output.
* SDBCS 2CV-303 is in Automatic at approximately 50% open.
* RCS T-ave is 548°F and steady.
* The CRS directs raising turbine load and stabilize Reactor power at 18% in accordance with OP-2104.004 Power Operations Step 8.25.

If no additional operator action is taken in the Control Room EXCEPT for raising turbine load, which of the following is the correct response of Reactor power and T-ave as Turbine Load is raised?

A. Reactor Power will start rising and T-ave lowering as soon as Turbine Load starts rising.

B. Reactor Power will remain the same and T-ave will rise as soon as turbine Load starts rising.

C. Reactor Power and T-ave will remain the same until all SDBCS valves have gone closed.

D. Reactor Power and T-ave will start rising after all the SDBCS valves have gone closed.

QUESTION: 65

Given the following plant conditions:

* Unit 2 is at full power operation.
* Waste Condensate Tank (2T-21A) is on recirc using Pump 2P-53A
* A liquid release is in progress from Boric Acid Condensate Tank 2T-69A.
* Release Isolation 2CV-2330A is Open.
* Release Isolation 2CV-2330B is Open.
* Handswitch 2HS-2330 on 2C112 is in Position '3'.
* Handswitch 2HS-2331 on 2C112 is in Position "BOTH".

* NOW Waste Condensate Tank (2T-21A) Pump 2P-53A Discharge Valve 2CV-2122 is inadvertently opened.

At this point, which of the following is the correct status of the 2T-69A release and the valve positions?

A. 2CV-2330A Open and 2CV-2330B Open; 2P-53A Trips

B. 2CV-2330A Open and 2CV-2330B Closed; 2P-53A Running

C. 2CV-2330A Closed and 2CV-2330B Open; 2P-53A Trips

D. 2CV-2330A Closed and 2CV-2330B Closed; 2P-53A Running

QUESTION: 66

Which of the following conditions is correct in accordance with EN-OP-102 with regard to preparation and/or installation authorization of a common unit tagout?

A. Installation must be authorized by either the Unit 1 or the Unit 2 Operations Supervisor.

B. Preparer and reviewer from both units must be licensed operators.

C. Preparer and reviewer may be non-licensed if authorized by both Unit Operations Supervisors.

D. Preparer may be non-licensed as long as the opposite unit reviewer is licensed.

QUESTION: 67

Given the following:

* The plant is in Mode 6 with refueling core alterations in progress.

During the core alterations, the neutron count rate should be observed each time a fuel assembly is ______________ the core and the status of the neutron count rate should be reported to the Refueling Bridge. If the neutron count rate cannot be communicated to the refueling bridge, then ____.

A. inserted into; suspend core alterations immediately until communications have been restored

B. inserted into; suspend core alterations after 1 hour if communications have not been restored

C. removed from; suspend core alterations immediately until communications have been restored

D. removed from; suspend core alterations after 1 hour if communications have not been restored

QUESTION: 68

Which of the following describes the method of maintaining component configuration control when responding to an abnormal event using an AOP procedure?

A. The CRS keeps a handwritten list of components placed out of position and enters them in the component deviation log as time allows during the event and ensures the components are returned to normal prior to exiting the AOP.

B. Complete valve lineups for the affected systems are required to be performed after the event prior to exiting the AOP.

C. The AOP is reviewed by the CRS after the event to ensure that any equipment that was operated by the AOP procedure is returned to its required position or documented that it is out of its normal position.

D. The AOP procedures have restoration steps in them that will return all manipulated components to a normal configuration prior to exiting.

QUESTION: 69

Consider the following:

* Unit 2 is operating at 100% power.
* The Unit 2 Control Room Operating crew consists of the following:
  - A Shift Manager who holds a SRO License
  - A CRS who holds a SRO License
  - Three (3) Control Board Operators licensed as ROs
  - A STA who does not have a license

* The on watch At The Controls (ATC) operator has to leave due to an Emergency at home.

With the watchstander leaving due to an emergency, the Unit Staff ______ meet(s) the Shift Manning Requirements of EN-OP-115, Conduct of Operations, and IF the minimum Shift Manning is NOT met at any time the crew has a MAXIMUM requirement of _____ hours to reestablish minimum Shift Manning.

A. does not; 1.5

B. still ; 1.5

C. does not; 2

D. still; 2

QUESTION: 70

Given the following:

* A Plant transient has occurred causing several annunciators to come in.

In accordance with the ANO-2 EOP/AOP Users Guide, 1015.021, the correct order to address and prioritize plant annunciators based on their color coding from the highest priority to the lowest priority would be:

A. Red, Green and White

B. Red, Amber and Green

C. Red, White and Green

D. Red, Amber and White

QUESTION: 71

Given the following:

* The plant has been tripped due to a Steam Generator Tube Rupture.
* Actions in Standard Attachment 19, Control of Secondary Contamination, have been completed.

After Standard Attachment 19 actions are taken, the ______________________________ would still be a contamination hazard and the __________________________ would continue to be a rising radiation source hazard.

A. Running Condensate pump coffer dams; SU/BD demineralizers

B. Running Condensate pump coffer dams; Condensate Inlet Filter 2F-807

C. EFW Pump 2P-7A bearing oil cooling water leak off; SU/BD Demineralizers

D. EFW Pump 2P-7A bearing oil cooling water leak off; Condensate Inlet Filter 2F-807

QUESTION: 72

During EOPs, the implementing procedure directs a step that the CBOT states is no longer appropriate for the given plant conditions. Which of the following is the correct process to handle the EOP procedure step in accordance with the ANO-2 EOP/AOP User Guide OP-1015.021?

A. The step MUST be performed as directed without reversal to mitigate the consequences of the accident.

B. The step MUST be performed but may be reversed if both the Shift Manager (SM) and CRS agree with the reversal.

C. The step MAY be skipped if the CRS and STA concur that the step is no longer appropriate for the given conditions.

D. The step MAY be skipped if authorized by the SM and the CRS concurs that the step is no longer appropriate.

QUESTION: 73

During assessment of annunciators at the end of SPTAs, what does it mean when a Control Room annunciator is in the "Slow Flash " mode?

A. An alarm condition has occurred and is still present and the Operator has NOT acknowledged the alarm.

B. An alarm condition has occurred and the Operator has acknowledged the alarm; however, the alarm condition is still present.

C. An alarm condition has occurred and then cleared and has NOT been acknowledged by the Operator.

D. An alarm condition has occurred and then cleared and then was acknowledged by the Operator.

QUESTION: 74

To minimize the radiological dose to refueling personnel in Mode 6 with core refueling in progress, the water level in the refueling canal must be a MINIMUM of 23 feet over the top of the _______________ and to protect the fuel assemblies from overheating and releasing radiation, _______ loop(s) of Shutdown Cooling shall be in operation.

A. reactor pressure vessel flange; 1

B. fuel assemblies seated in the vessel; 1

C. reactor pressure vessel flange; 2

D. fuel assemblies seated in the vessel; 2

QUESTION: 75

Per EN-HU-106, Procedure and Work Instructions Use and Adherence, steps identified as bulleted steps rather than numbered or lettered steps ____.

A. will be performed on a continuing basis after the step is presented in the procedure

B. are required to be performed in the sequence as written in the implementing procedure

C. can be performed in any sequence or in parallel when the step is presented in the procedure

D. can be performed at anytime the specified condition in the step exist when the procedure is in use

QUESTION: 76

Given the following:

* Unit 2 was operating at 4% reactor power.
* Main Steam line break on 'A' SG has occurred in Containment.
* Reactor trip and MSIS automatically initiated.
* 'A' SG level (WR) is off-scale low.
* Containment pressure is 20.5 psia and lowering.
* T-cold is 400°F and stabilizing.
* CET temperature is 420°F and stabilizing.
* Pressurizer pressure is 1950 psia and slowly rising.

Which procedure and required actions should be used?

A. Implement OP-2203.011, RCS Overcooling and stabilize RCS pressure.

B. Implement OP-2202.005, Excess Steam Demand and stabilize RCS pressure.

C. Implement OP-2203.011, RCS Overcooling and lower RCS pressure to maintain RCS MTS less than 200°F.

D. Implement OP-2202.005, Excess Steam Demand and lower RCS pressure to maintain RCS MTS less than 200°F.

QUESTION: 77

Given the following:

* Unit 2 is in Mode 6 with Core offload in progress.
* "B" train LPSI Pump (2P-60B) is in service through the "B" SDC Heat Exchanger.
* RCS temperature is 130°F.
* 2RITS-1456, 2E-35B Outlet Radiation monitor is in alarm.
* PROC LIQUID RADIATION HI/LO, 2K11-C10 is in alarm.
* RCS level is slowly lowering.

At this point the CRS _________________ and direct isolating ______________ to the "B" SDC Heat Exchanger (2E-35B).

A. should enter OP-2203.029, Loss of Shutdown Cooling; the Reactor Coolant System (RCS) ONLY

B. must enter OP-2202.011, Lower Mode Functional Recovery; the Reactor Coolant System (RCS) ONLY

C. should enter OP-2203.029, Loss of Shutdown Cooling; both the Reactor Coolant System (RCS) and Service Water (SW)

D. must enter OP-2202.011, Lower Mode Functional Recovery; both the Reactor Coolant System (RCS) and Service Water (SW)

QUESTION: 78

Consider the following:

* Unit 2 is operating at 100% power.
* 2D02 Green Battery disconnect has been opened for maintenance.
* Battery charger, 2D-32B, is in service.
* Now 2D-32B output breaker trips open.

The CRS would direct entry into ___________ and crew will be required to ___________.

A. OP-2203.037, Loss of 125V DC AOP; cycle Charging pumps to maintain PZR level

B. OP-2203.037, Loss of 125V DC AOP; place all of the HPSI, LPSI and CS Pump handswitches in PTL to prevent spurious starts

C. OP-2202.001, Standard Post Trip Action EOP; cycle Charging pumps to maintain PZR level

D. OP-2202.001, Standard Post Trip Action EOP; place all of the HPSI, LPSI and CS Pump handswitches in PTL to prevent spurious starts

QUESTION: 79

Given the following:

* SPTAs have been completed.
* OP-2202.007, Loss of Offsite Power has been entered.
* 3 CEAs stuck fully out on the reactor trip.
* ATC has commenced Emergency Boration per Exhibit 1.
* Charging Pump, 2P-36A is running with a discharge flow of 43 gpm.
* 4160v Vital AC bus 2A4 is de-energized and all attempts to start 2EDG2 have failed.
* T-cold is 557°F and stable.
* T-hot is 586°F and stable.
* CET temperature is 587°F and stable.
* RCS pressure is 1900 psia and slowly rising.
* Pressurizer level 45% and slowly lowering.
* SG levels are 20% NR and rising with EFW flow at 500 gpm per SG.
* SG pressures are 1102 psia and slowly lowering.
* Containment pressure is 16.2 psia and stable.
* Containment temperature is 145°F and slowly rising.

Which of the following is TRUE based on an assessment of the Safety Function Status Checks listed in the Loss of Offsite Power EOP for the above conditions?

A. Reactivity Control is NOT met.

B. Maintenance of Vital Auxiliaries is NOT met.

C. Core Heat Removal is NOT met.

D. CNTMT Temperature and Pressure Control is NOT met.

QUESTION: 80

Given the following:

* Unit 2 is operating at 100% power.
* PPS Channel 'C' Low SG 2 Level has been declared inoperable due to spurious trips and has been placed in trip channel bypass.
* Currently there is no low SG 2 Level trip in on Channel 'C' PPS.
* AC power is now lost to Low SG 2 level on Channel 'B' PPS.
* I&C has been contacted to initiate troubleshooting and repair on PPS Channel 'B' bistable.

Due to the loss of AC power and to comply with Tech Specs and allow I&C technicians to perform maintenance, the CRS should direct the CBOT to _______________, and if the inoperable channel is bypassed for greater than a MAXIMUM of ______ hours , the desirability of maintaining this channel in the bypassed condition shall be reviewed as soon as possible but no later than the next regularly scheduled OSRC meeting in accordance with the Quality Assurance Program Manual (QAPM).

A. place Channel B in bypass, place Channel C in trip, then remove Channel C from bypass; 48

B. remove Channel C from bypass, place Channel B in bypass, then place Channel C in trip; 48

C. place Channel B in bypass, place Channel C in trip, then remove Channel C from bypass; 72

D. remove Channel C from bypass, place Channel B in bypass, then place Channel C in trip; 72

QUESTION: 81

(REFERENCE PROVIDED)

Given the following sequence:

* Unit 2 is operating at 100% power.
* All 3 Charging pumps are running.
* Letdown is isolated.
* Containment pressure and humidity are slowly rising.
* Pressurizer level and pressure are lowering.
* A manual reactor trip is initiated.
* Pressurizer pressure stabilizes at 1500 psia.
* RCS T-cold is stable at 551°F.

This event should be classified as an ______ and the order of offsite notifications shall be ________ per OP-1903.010, EAL Classification.

A. NUE; State or Local agencies then the NRC

B. NUE; NRC then the State or Local agencies

C. Alert; State or Local agencies then the NRC

D. Alert; NRC then the State or Local agencies

QUESTION: 82

Given the following:

* Unit 2 is "heating up" following a refueling outage.
* RCS pressure is 800 psia.
* RCS Tave is 260 degrees.
* RCS Sample valve, 2SV-5833-1 has failed OPEN and been declared Inoperable.

LCO 3.6.3.1, Containment Isolation Valves, _____ applicable under current plant conditions and the following actions will be required to satisfy the LCO 3.6.3.1 if OR when it is required.

A. is; Verify Operability and close the associated containment isolation valve.

B. is NOT; Verify Operability and close the associated containment isolation valve.

C. is; Deactivate and secure the associated containment isolation valve in the isolated position.

D. is NOT; Deactivate and secure the associated containment isolation valve in the isolated position.

QUESTION: 83

Given the following:

* Unit 2 is stable at 100% power.
* "LETDOWN RADIATION HI/LO" annunciator (2K12-A1) alarms.
* OP-2203.020, High Activity in RCS, has been entered.
* Chemistry reports that Dose Equivalent Iodine 131 activity is 80 µCi/gram.
* Aux. Building Area Radiation monitors are 0.2 R/hr and rising.
* RCS Letdown Gross monitor (2RITS-4806A) activity is slowly rising.

This exceeds the LCO 3.4.8, Specific Activity, I-131 limit of ______ µCi/gram and based on these conditions, OP-2203.020 provides the Shift Manager direction to isolate _____________________.

A. 0.1; Letdown only

B. 1.0; Letdown only

C. 0.1; Letdown and RCP Bleedoff to VCT

D. 1.0; Letdown and RCP Bleedoff to VCT

QUESTION: 84

Given the following:

* Unit 2 is operating at 100% power.
* OP-2203.016, Excess RCS Leakage, has been entered due to indications of an RCS leak.
* Leakage is Pressure Boundary leakage into the Component Cooling Water system.
* Crew starts an additional Charging Pump.
* Letdown flow stabilizes at 60 gpm.

Per LCO 3.4.6.2 RCS Operational Leakage the unit must ____________. If a shutdown to HOT STANDBY is required, then AOP-2203.016 (Excess RCS Leakage) directs the use of __________ to perform the RCS boration IAW OP-2102.004, Power Operations.

A. be in at least HOT STANDBY within 6 hours; Exhibit 3, Normal RCS Boration at Power

B. be in at least HOT STANDBY within 6 hours; Attachment R, RCS Boration from the RWT or BAMT

C. reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours; Exhibit 3, Normal RCS Boration at Power

D. reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours; Attachment R, RCS Boration from the RWT or BAMT

QUESTION: 85

(REFERENCE PROVIDED)

Given the following:

* Unit 2 is at 100% power on March 15th.
* At 0800, LPSI Pump 'B', 2P-60B, is OOS for scheduled maintenance.
* HPSI Pump 'C', 2P-89C, is aligned to the Green train.
* At 1200, NLO reports he has inadvertently broken a sight glass on HPSI Pump 'A', 2P-89A.
* NLO also reports there is a large amount of oil spreading on the floor.

Assuming no Operator actions have been taken and using the applicable Tech Spec maximum allowable action times, Unit 2 must be in Hot Standby no later than ______________?

A. 1800 on March 15th

B. 1900 on March 15th

C. 1800 on March 18th

D. 1400 on March 22nd

QUESTION: 86

Given the following:

* Unit 2 is operating at 100% power.
* 'B' EFW Pump (2P-7B) is OOS for emergent maintenance.
* A seismic event occurs.
* Condenser vacuum rapidly degraded and indications are off scale high.
* A manual reactor trip is initiated.
* Bus 2A1 faults and cannot be re-energized.
* 'A' and 'B' Steam Generator pressures are being controlled by ADVs.
* Pressurizer pressure is 1380 psia and stable.
* Pressurizer level is 10% and slowly rising.
* RCS T-hot is 547°F.
* Containment radiation levels are rising.
* 'A' EFW Pump (2P-7A) trips on overspeed.
* SPTAs are complete.
* All 4 RCPs are running.

The crew should transition to ______________ and ___________________ .

A. OP-2202.009, Functional Recovery; secure ONLY 2 RCPs

B. OP-2202.009, Functional Recovery; secure all running RCPs

C. OP-2202.003, Loss of Coolant Accident; secure ONLY 2 RCPs

D. OP-2202.003, Loss of Coolant Accident; secure all running RCPs

QUESTION: 87

Consider the following

* Unit 2 is in Mode 3 preparing for a reactor startup when the following occurs.
* ATC reports RCS pressure is 2250 psia and rising.
* Crew is taking actions IAW OP-2203.028, Pzr Systems Malfunction.
* Pzr Spray valves 2CV-4651 and 2CV-4652 indicate closed.

Pressurizer Pressure LCO 3.2.8. will have to be entered if RCS pressure exceeds a MINIMUM of ______ psia and if PZR spray valves will NOT open manually, then OP-2203.028 directs using __________ to mitigate these conditions.

A. 2275; OP-2103.005, Pressurizer Operations

B. 2275; OP-2202.010, Attachment 48, RCS Pressure Control

C. 2340; OP-2103.005, Pressurizer Operations

D. 2340; OP-2202.010, Attachment 48, RCS Pressure Control

QUESTION: 88

Consider the following:

* Unit 2 entered a refueling outage 2 days ago.
* LPSI Pump 'B', 2P-60B, is aligned and providing SDC flow.
* RCS Tc is 220°F and is being slowly lowered.
* RCS pressure is 250 psia and stable.

NOW

* LPSI Pump 'B' motor faults and trips.
* SDC SUCTION PRESS HI, annunciator (2K07-D7) is in alarm.
* SDC FLOW HI/LO annunciator (2K07-C8) is in alarm.
* RCS temperature is 225°F and slowly rising.
* RCS pressure is 305 psia and slowly rising.
* SDC RCS Isolation MOV, 2CV-5084-1, is closed.

Based on these conditions the CRS should __________________ and this procedure will direct the CRS to _________________ in order to open SDC RCS Isolation valve 2CV-5084-1.

A. transition to the Lower Mode Functional, OP-2202.011; reduce RCS pressure with Aux. Spray

B. implement the Loss of Shutdown Cooling, OP-2203.029; reduce RCS pressure with Aux. Spray

C. transition to the Lower Mode Functional, OP-2202.011; ensure the 2CV-5084-1 handswitch is in CLOSE and remove the associated power supply fuse located in front of panel 2C21

D. implement the Loss of Shutdown Cooling, OP-2203.029; ensure the 2CV-5084-1 handswitch is in CLOSE and remove the associated power supply fuse located in front of panel 2C21

QUESTION: 89

Given the following:

* OP-2202.003, Loss of Coolant Accident has been entered.
* HPSI Pump 2P-89C is OOS for scheduled maintenance.
* HPSI Pump 2P-89B tripped and will not restart.
* HPSI Pumps 2P-89A is running.
* Cntmt Spray Pumps 2P-35A and 2P-35B are running.
* RAS has actuated, all automatic and manual actions are complete.

Standard Attachment 43, ECCS/CSS Pump monitoring states that Containment Sump blockage would be indicated by _________________ and ________________ handswitch(es) would be placed in PTL if Early Termination conditions are met.

A. changing containment sump level; either Cntmt Spray Pump 'A' or 'B' ONLY

B. changing containment sump level; either Cntmt Spray Pump 'A' or 'B' and HPSI Pump 'A'

C. unstable flow and discharge pressure on running ESF pumps; either Cntmt Spray Pump 'A' or 'B' ONLY

D. unstable flow and discharge pressure on running ESF pumps; either Cntmt Spray Pump 'A' or 'B' and HPSI Pump 'A'

QUESTION: 90

Given the following:

* Unit 2 is operating at 100% power.
* Service Water Loop 2 has been isolated due to a piping rupture.
* Pressurizer level and pressure are rapidly lowering.
* An automatic reactor trip occurs.
* RCS pressure is 1100 psia and lowering.
* Containment pressure is 24 psia and rising.
* SG pressures are 1000 psia and stable.
* OP-2202.003, LOCA has been entered.
* 'B' train Containment Spray flow is 1200 gpm.

NOW

* Bus 2A3 faults and cannot be re-energized.

Containment Coolers 2VSF-1C and 2VSF-1D are running with _______ and the CRS _________.

A. no cooling flow; must transition to OP-2202.009, FRP and implement CTPC-3, CNTMT Spray

B. normal chill water flow; must transition to OP-2202.009, FRP and implement CTPC-3, CNTMT Spray

C. no cooling flow; may remain in OP-2202.003, LOCA and implement Exhibit 13, Miscellaneous Containment Building Ventilation

D. normal chill water flow; may remain in OP-2202.003, LOCA and implement Exhibit 13, Miscellaneous Containment Building Ventilation

QUESTION: 91

Given the following:

* Unit 2 is operating at 100% power.
* Main Turbine net output is 975 megawatts.
* CEAs are being used to dampen ASI oscillations per OP-2102.004, Power Operation.
* Group 6 CEAs are 130 inches withdrawn.
* Group P CEAs are 136 inches withdrawn.
* Both CEACs are Operable.

NOW

* Annunciators 2K04-J5 and J6 "CEAC 1/2 CEA DEVIATION" come into alarm.
* Group 6, CEA-47, CEA Pulse Counter is 130 inches and CEAC positions are 121 inches withdrawn.
* Group P, CEA-23, CEA Pulse Counter and CEAC positions all indicate 0 inches.
* Crew has entered OP-2203.003, CEA Malfunction.

Prior to any Operator actions Main Turbine load ___________ and the CRS will enter 2203.003, CEA Malfunctions and direct a _____________________ for the given CEA positions.

A. remains constant; Plant shutdown using OP-2102.004, Power Operations

B. remains constant; Reactor trip, and performance of OP-2202.001, SPTAs

C. initially lowers; Plant shutdown using OP- 2102.004, Power Operations

D. initially lowers; Reactor trip, and performance of OP-2202.001, SPTAs

QUESTION: 92

Given the following:

* Unit 2 has tripped from full power due to a Steam Generator Tube Rupture.
* 'A' Steam Generator has been diagnosed as the ruptured SG.
* RCPs 2P-32A and 2P-32D are running.
* SG 'A' has been isolated.
* Cooldown and depressurization of the 'A' SG has commenced.
* All other system and components function as designed.

At this time, OP-2202.004, Steam Generator Tube Rupture, requires a level band of _______ % in the ruptured SG and the basis for this level is to ensure SG tubes are _______________.

A. 10 to 38; partially uncovered to cool the steam space of the 'A" SG

B. 10 to 38; covered to prevent release of gaseous activity from the RCS

C. 22.2 to 45; partially uncovered to cool the steam space of the 'A" SG

D. 22.2 to 45; covered to prevent release of gaseous activity from the RCS

QUESTION: 93

(Picture of Pzr level instruments attached to this question)

Given the following:

* Unit 2 is operating at 100% power.
* Pressurizer level instrument, 2LI-4627-2B, Channel 2, was declared Inoperable and removed from service three (3) shifts ago due to erratic readings.
* I & C has completed all required maintenance/rework activities.
* OPSLOG B18 = Post Accident Instrumentation.
* OPSLOG B19 = Remote Shutdown Instrumentation.
* 2LI-4627-1 = Channel 1 Pressurizer Level Instrument.
* 2LI-4627-2AN = Channel 2 Pressurizer Level Instrument.

In order to restore Operability of PZR Level Instrument 2LI-4627-2B, a channel check must be performed and documented using ____________________________ and the channel check must be performed using ________________________ PZR Level Instrument(s).

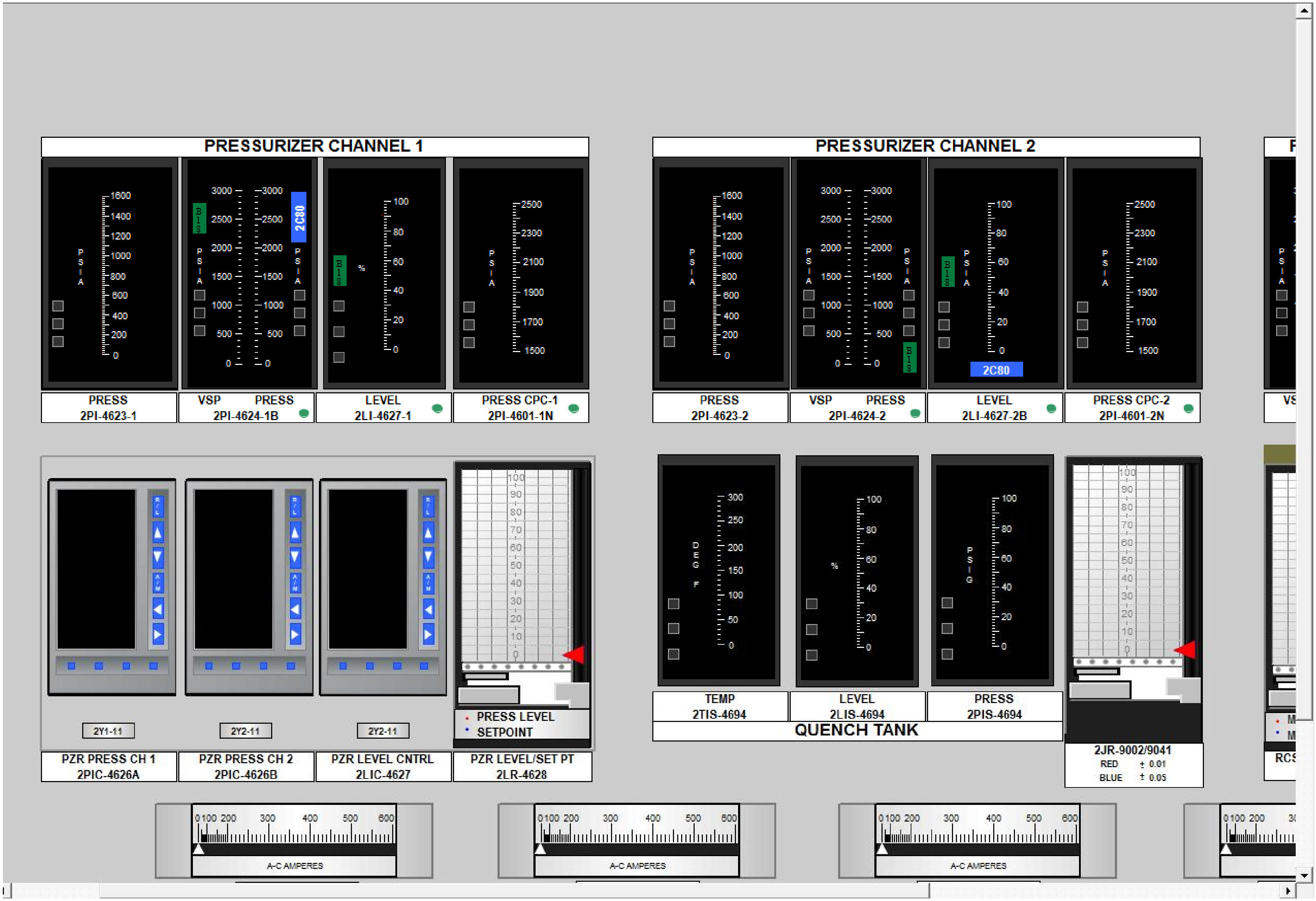
A. OPSLOG B18 ONLY; 2LI-4627-1 ONLY

B. OPSLOG B19 ONLY; 2LI-4627-2AN ONLY

C. OPSLOG B18 and OPSLOG B19; 2LI-4627-2AN ONLY

D. OPSLOG B18 and OPSLOG B19; 2LI-4627-1 and 2LI-4627-2AN

ATTACHMENT FOR QUESTION 93



QUESTION: 94

Consider the following:

* Core reload is in progress.
* A fuel assembly is having insertion difficulties.
* Reactor Fuel Bridge Operator requests a fuel shuffle sequence change.

Per OP-2502.001, Refueling Shuffle, both the "SRO in charge of fuel handling" and __________ permission is required to authorize this change?

A. Operations Manager's

B. Shift Manager's

C. Shift Outage Manager's

D. Reactor Engineering's

QUESTION: 95

Given the following:

* General Emergency has been declared.
* TSC has been activated.
* Repair activities need to be performed on LPSI Pump 'A', 2P-60A.
* LPSI 'A', 2P-60A, has been deemed valuable property.
* Dose readings in the repair area are 120 Rem/hr.

Per OP-1903.033, Protective Action Guidelines for Rescue/Repair & Damage Control Teams, the ____________ is responsible to AUTHORIZE the exposure for this repair and the MAXIMUM amount of time the repair team can remain in the area is _______________.

A. Emergency Plant Manager; 2.5 minutes

B. Emergency Plant Manager; 5.0 minutes

C. Radiological Coordinator; 2.5 minutes

D. Radiological Coordinator; 5.0 minutes

QUESTION: 96

Consider the following:

* System Engineering has determined that connecting a contaminated and non-contaminated system with a rubber hose would be a Temporary Modification

Before the hose is installed between these two systems, who must authorize installation AND what is the MINIMUM number of check valve(s) that must be installed in series to satisfy procedural requirements per EN-DC-136, Temporary Modifications?

A. Shift Manager; 1 check valve

B. Shift Manager; 2 check valves

C. System Engineering Manager; 1 check valve

D. System Engineering Manager; 2 check valves

QUESTION: 97

Given the following:

* Due to a fire, the Unit 2 CRS has entered OP 2203.034, Fire or Explosion.
* Unit 2 Control Room has become uninhabitable due to the fire.

Due to the subsequent Unit 2 Control Room evacuation, the _______ will be responsible to continue with OP 2203.034, Fire or Explosion procedure and if the Fire Brigade requires additional assistance the __________ Fire Department will be requested per OP-2203.034, Fire or Explosion.

A. Unit 1 SM; London

B. Unit 1 SM; Russellville

C. Unit 2 SM; London

D. Unit 2 SM; Russellville

QUESTION: 98

Given the following:

* Unit 2 has tripped from 100% power.
* OP-2202.009, Functional Recovery procedure has been entered.
* The assessment of safety functions has been completed and is provided below.
  - Maintenance of Vital Auxiliaries (AC) is Challenged.
  - RCS Inventory Control is Challenged.
  - Containment Isolation is Jeopardized.
  - RCS Pressure Control is Jeopardized.
  - All other Safety Function acceptance criteria are "Satisfied".

Per step 13 of the FRP, which of the following is the correct order for success path implementation?

A. RCS Pressure Control, Cntmt Isolation, RCS Inventory Control, MVAC (AC)

B. RCS Pressure Control, Cntmt Isolation, MVAC (AC), RCS Inventory Control

C. CNMT Isolation, RCS Pressure Control, RCS Inventory Control, MVAC (AC)

D. CNMT Isolation, RCS Pressure Control, MVAC (AC), RCS Inventory Control

QUESTION: 99

Protective Action Recommendations (PARs) are required to be provided during ____________ Emergencies and PARs must be reassessed within a MINIMUM of every _________________.

A. ONLY General; 15 minutes

B. ONLY General 30 minutes

C. General and Site Area; 15 minutes

D. General and Site Area; 30 minutes

QUESTION: 100

Given the following:

* Unit 2 is operating at 100% power.
* It is discovered that a 31 day surveillance was not performed on a Tech Spec component.
* The surveillance was due 10 days ago.

Which of the following actions is/are required per Surveillance Requirement 4.0.3?

Component must be declared Inoperable _________

A. at time of discovery.

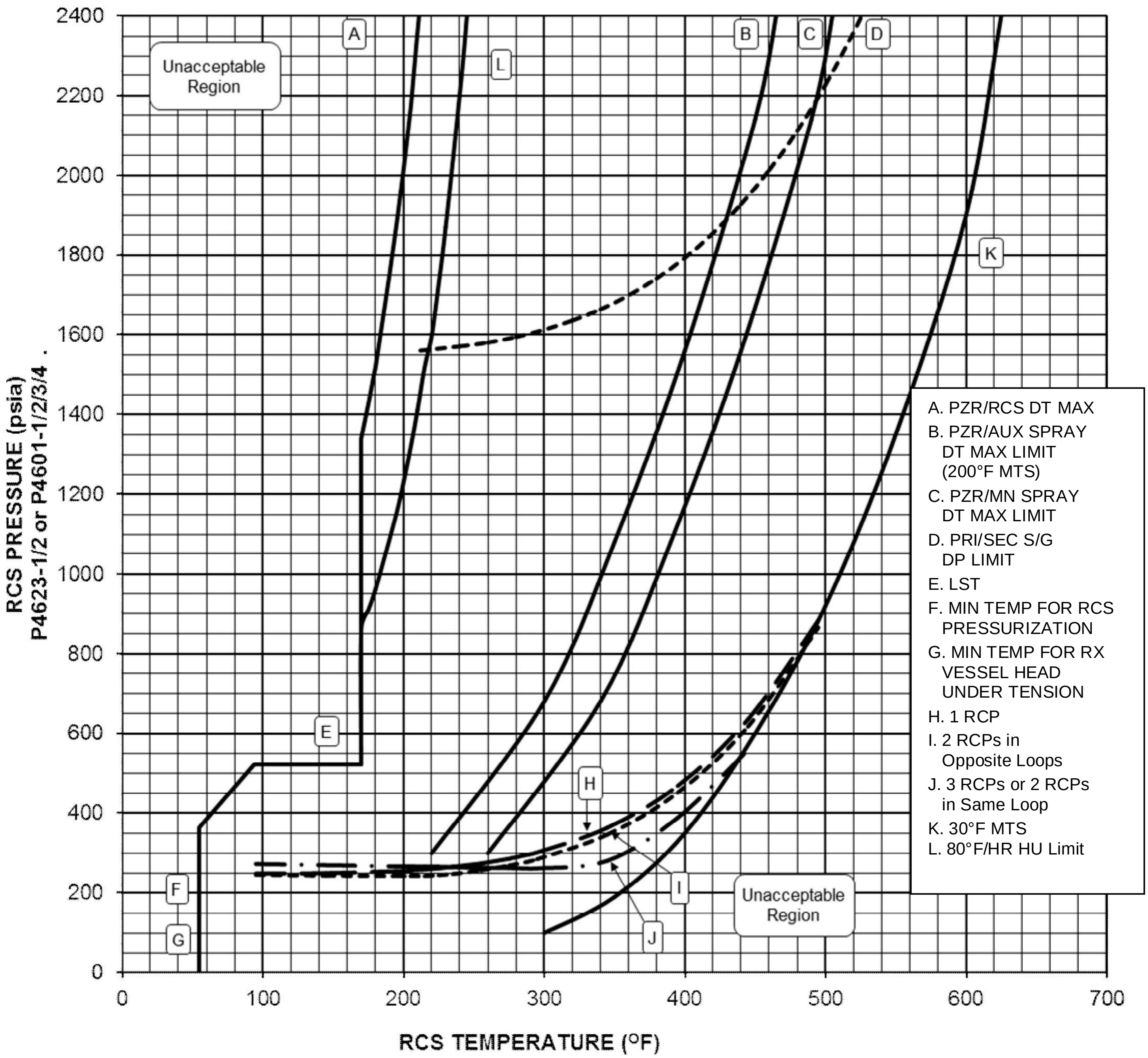
B. within a maximum 24 hours from time of discovery.

C. within a maximum of 31 days from time of discovery provided a risk evaluation has been performed and risk impact is managed.

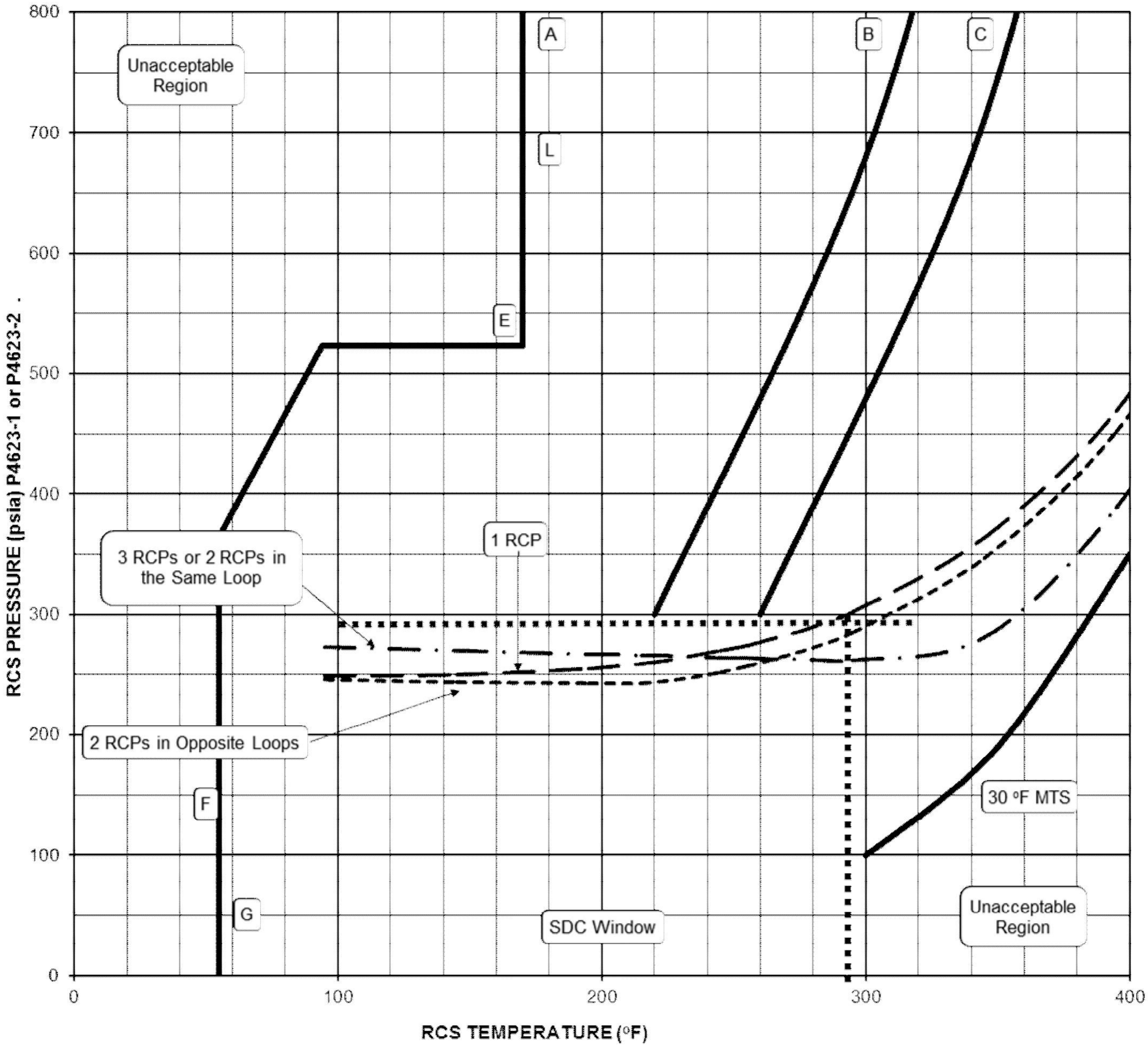
D. within a maximum of 38.75 days from time of discovery provided a risk evaluation has been performed and risk impact is managed.

ATTACHMENT 1
P-T LIMITS

- Use T4615 (2TR-4615) or T4715 (2TR-4715) if plotting RCS cooldown.
- If plotting cooldown with no RCPs in operation and SDC in service, Tcold should be SDC return temp T5095 (2TR-5097).
- Use Control Board indication for RCP Operating Curves - Includes 12°F and 19 psi instrument errors.
- Use 200°F MTS line for PTS limit for uncontrolled RCS cooldown below 500°F T_C.
- Use RCS TH in forced circulation to determine RCS MTS.
- Use average CETs in natural circulation to determine RCS MTS.
- Stay to right of primary to secondary Δ P curve during controlled cooldown. (ER010861E201)
- Minimum RCS temperature with fuel in the Reactor Vessel is 68°F.



ATTACHMENT 1
P-T LIMITS



- Use T4615 (2TR-4615) or T4715 (2TR-4715) if plotting RCS cooldown.
- If plotting cooldown with no RCPs in operation and SDC in service, Tcold should be SDC return temp T5095 (2TR-5097).
- Use Control Board indication for RCP Operating Curves - Includes temperature error of 12°F and pressure error of 19 psi (ER010861E201).
- Use 200°F MTS line for PTS limit for uncontrolled RCS cooldown below 500°F T_{C}.
- Use RCS T_{H} in forced circulation to determine RCS MTS.
- Use average CETs in natural circulation to determine RCS MTS.
- Minimum RCS temperature with fuel in the Reactor Vessel is 68°F.

ATTACHMENT C

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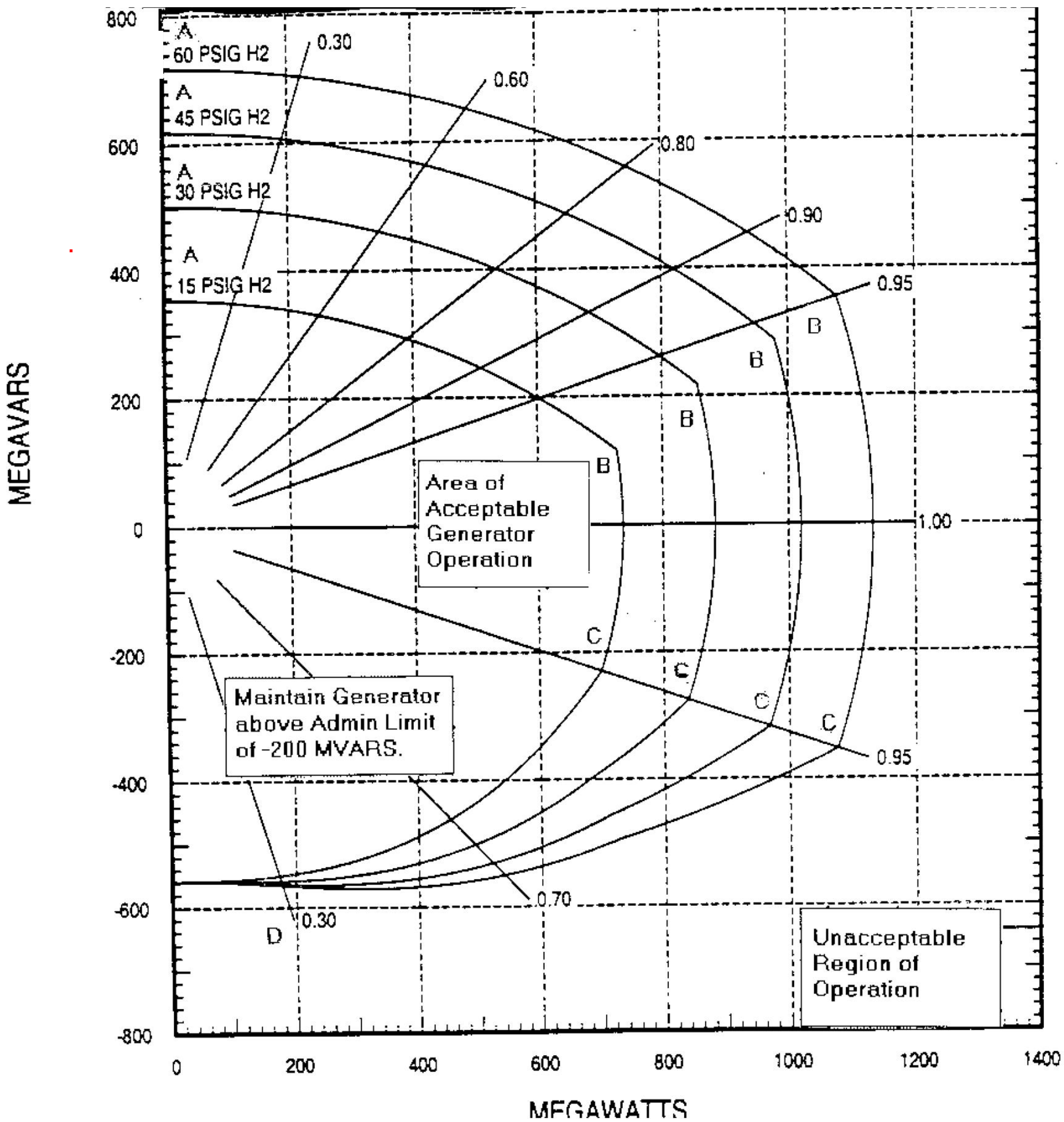
REACTIVE CAPABILITY CURVES



CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
CURVE CD LIMITED BY ARMATURE CORE END HEATING

2305.002 EXHIBIT 1 REVISED 06/30/16

COMPONENT VOLUME VS LEVEL

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PRESSURIZER	53.5 gal/%
VCT	33.8 gal/%
CNTMT SUMP	39.0 gal/%
QUENCH TANK & RDT	17.6 gal/%
CCW SURGE TANK (CCW Loops Split)	9.3 gal/%
CCW SURGE TANK (CCW Loops Cross-Connected)	18.6 gal/%
SIT	120.6 gal/%
RWT	4787.8 gal/%
ABS	9.48 gal/%
2T-20	57.0 gal/%
2T-12	425 gal/%

ATTACHMENT 1

INDEX OF EMERGENCY ACTION LEVELS

INDEX OF EMERGENCY ACTION LEVELS

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TAB A

Abnormal Radiation Levels / Radiological Effluents

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

ABNORMAL RADIOLOGICAL EFFLUENTS

GENERAL EMERGENCY

AG1 1 2 3 4 5 6 D

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 1000 mR TEDE or 5000 mR child thyroid CDE for the actual or projected duration of the release using actual meteorology

Emergency Action Level(s):

NOTE:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.

1. VALID reading on Channel 9 on any of the following radiation monitors > the reading shown for ≥ 15 minutes:

MONITORS – Unit 1		LIMIT
RX-9820	Containment Purge	1.18E+2 µCi/cc
RX-9825	Radwaste Area	1.07E+2 µCi/cc
RX-9830	Fuel Handling Area	9.08E+1 µCi/cc
RX-9835	Emerg. Penetration Room	1.91E+3 µCi/cc
MONITORS – Unit 2		**LIMIT**
2RX-9820	Containment Purge	8.92E+1 µCi/cc
2RX-9825	Radwaste Area	6.64E+1 µCi/cc
2RX-9830	Fuel Handling Area	8.92E+1 µCi/cc
2RX-9835	Emerg. Penetration Room	1.77E+3 µCi/cc
2RX-9840	PASS Building	8.84E+2 µCi/cc
2RX-9845	Aux. Building Extension	2.53E+2 µCi/cc

OR

SITE AREA EMERGENCY

AS1 1 2 3 4 5 6 D

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 100 mR TEDE or 500 mR child thyroid CDE for the actual or projected duration of the release

Emergency Action Level(s):

NOTE:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.

1. VALID reading on Channel 9 on any of the following radiation monitors > the reading shown for ≥ 15 minutes:

MONITORS – Unit 1		LIMIT
RX-9820	Containment Purge	1.18E+1 µCi/cc
RX-9825	Radwaste Area	1.07E+1 µCi/cc
RX-9830	Fuel Handling Area	9.08E0 µCi/cc
RX-9835	Emerg. Penetration Room	1.91E+2 µCi/cc
MONITORS – Unit 2		**LIMIT**
2RX-9820	Containment Purge	8.92E0 µCi/cc
2RX-9825	Radwaste Area	6.64E0 µCi/cc
2RX-9830	Fuel Handling Area	8.92E0 µCi/cc
2RX-9835	Emerg. Penetration Room	1.77E+2 µCi/cc
2RX-9840	PASS Building	8.84E+1 µCi/cc
2RX-9845	Aux. Building Extension	2.53E+1 µCi/cc

OR

ALERT

AA1 1 2 3 4 5 6 D

Any release of gaseous or liquid radioactivity to the environment > 200 times the ODCM limits for ≥ 15 minutes

Emergency Action Level(s):

NOTE:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

1. VALID reading on Channel 7 on any of the following radiation monitors > the reading shown for ≥ 15 minutes:

MONITORS – Unit 1		LIMIT
RX-9820	Containment Purge	1.18E0 µCi/cc
RX-9825	Radwaste Area	1.07E0 µCi/cc
RX-9830	Fuel Handling Area	9.08E-1 µCi/cc
RX-9835	Emerg. Penetration Room	1.91E+1 µCi/cc
MONITORS – Unit 2		**LIMIT**
2RX-9820	Containment Purge	8.92E-1 µCi/cc
2RX-9825	Radwaste Area	6.64E-1 µCi/cc
2RX-9830	Fuel Handling Area	8.92E-1 µCi/cc
2RX-9835	Emerg. Penetration Room	1.77E+1 µCi/cc
2RX-9840	PASS Building	8.84E0 µCi/cc
2RX-9845	Aux. Building Extension	2.53E0 µCi/cc
2RX-9850	LLRW Storage Building	3.54E0 µCi/cc

OR

UNUSUAL EVENT

AU1 1 2 3 4 5 6 D

Any release of gaseous or liquid radioactivity to the environment > 2 times the ODCM limits for ≥ 60 minutes

Emergency Action Level(s):

NOTE:

The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

1. VALID reading on Channel 7 on any of the following radiation monitors > the reading shown for ≥ 60 minutes:

MONITORS – Unit 1		LIMIT
RX-9820	Containment Purge	1.18E-02 µCi/cc
RX-9825	Radwaste Area	1.07E-02 µCi/cc
RX-9830	Fuel Handling Area	9.08E-03 µCi/cc
RX-9835	Emerg. Penetration Room	1.91E-01 µCi/cc
MONITORS – Unit 2		**LIMIT**
2RX-9820	Containment Purge	8.92E-3 µCi/cc
2RX-9825	Radwaste Area	6.64E-3 µCi/cc
2RX-9830	Fuel Handling Area	8.92E-3 µCi/cc
2RX-9835	Emerg. Penetration Room	1.77E-1 µCi/cc
2RX-9840	PASS Building	8.84E-2 µCi/cc
2RX-9845	Aux. Building Extension	2.53E-2 µCi/cc
2RX-9850	LLRW Storage Building	3.54E-2 µCi/cc

OR

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

ABNORMAL RADIOLOGICAL EFFLUENTS

AG1 *(continued)*

2. Dose assessment using actual meteorology indicates doses > 1000 mR TEDE or 5000 mR child thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 minutes; or analyses of field survey samples indicate child thyroid CDE > 5000 mR for one hour of inhalation, at or beyond the site boundary.

AS1 *(continued)*

2. Dose assessment using actual meteorology indicates doses > 100 mR TEDE or 500 mR child thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates > 100 mR/hr expected to continue for ≥ 60 minutes; or analyses of field survey samples indicate child thyroid CDE > 500 mR for one hour of inhalation, at or beyond the site boundary.

AA1 *(continued)*

2. **EITHER** VALID reading on any of the following radiation monitors > 200 times the alarm setpoint established by a current release permit for ≥ 15 minutes **OR** VALID reading greater than the value listed for ≥ 15 minutes:

MONITORS – Unit 1		LIMIT
RX-9820	Cont. Purge (Ch. 7 or 9)	N/A
RX-4830	Waste Gas Monitor	9.5E7 cpm
RX-4642	Liquid Radwaste Monitor	9.5E7 cpm
RX-9835	Emerg. Penetration Room	N/A
MONITORS – Unit 2		**LIMIT**
2RX-9820	Cont. Purge (Ch. 7 or 9)	N/A
2RX-2429	Waste Gas Monitor	9.5E5 cpm
2RX-2330	BMS Discharge Monitor	9.5E5 cpm
2RX-4423	LRW Discharge Monitor	9.5E5 cpm
2RX-4425	SG BD to Flume Monitor	9.5E5 cpm

OR

3. Confirmed grab sample analyses for gaseous or liquid releases indicates concentrations or release rates > 200 times the applicable values of the ODCM for ≥ 15 minutes.

AU1 *(continued)*

2. VALID reading on any of the following radiation monitors > 2 times the alarm setpoint established by a current release permit for ≥ 60 minutes:

MONITORS – Unit 1	
RX-9820	Cont. Purge (Ch. 7 or 9)
RX-4830	Waste Gas Monitor
RX-4642	Liquid Radwaste Monitor
RX-9835	Emerg. Penetration Room
MONITORS – Unit 2	
2RX-9820	Cont. Purge (Ch. 7 or 9)
2RX-2429	Waste Gas Monitor
2RX-2330	BMS Discharge Monitor
2RX-4423	LRW Discharge Monitor
2RX-4425	SG BD to Flume Monitor

OR

3. Confirmed grab sample analyses for gaseous or liquid releases indicates concentrations or release rates > 2 times the applicable values of the ODCM for ≥ 60 minutes.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

ABNORMAL RADIATION LEVELS

AA2 (ALERT)

1 2 3 4 5 6 D

Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel

Emergency Action Level(s):

1. A water level drop in the refueling canal or spent fuel pool that will result in irradiated fuel becoming uncovered.

   OR

2. VALID alarm on any of the following radiation monitors due to damage to irradiated fuel or loss of water level:

MONITORS – Unit 1	
RX-9820	Containment Purge (Channel 7 or 9)
RX-9825	Radwaste Area (Channel 7 or 9)
RX-9830	Fuel Handling Area (Channel 7 or 9)
RE-8060	Containment High Range Monitor
RE-8061	Containment High Range Monitor
RE-8009	Spent Fuel Area
RE-8017	Fuel Handling
MONITORS – Unit 2	
2RX-9820	Containment Purge (Channel 7 or 9)
2RX-9825	Radwaste Area (Channel 7 or 9)
2RX-9830	Fuel Handling Area (Channel 7 or 9)
2RE-8905	Containment Equipment Hatch Area
2RE-8909	Containment Personnel Hatch Area
2RE-8925-1/2	Containment High Range Monitors
2RE-8914/15/16	Spent Fuel Area Monitors
2RE-8912	Containment Incore Instruments

AU2 (UNUSUAL EVENT)

1 2 3 4 5 6 D

UNPLANNED rise in plant radiation levels

Emergency Action Level(s):

1. a. UNPLANNED lowering of water level in the refueling canal or spent fuel pool as indicated by:

   - Personnel observation, refueling crew report, indication on area security camera, borated water source (BWST or RWT) level drop due to makeup demands.

   AND

   b. VALID Area Radiation Monitor reading rise on any of the following:

MONITORS – Unit 1	
RE-8009	Spent Fuel Area
RE-8017	Fuel Handling Area
MONITORS – Unit 2	
2RE-8914	Spent Fuel Area
2RE-8915	Spent Fuel Area
2RE-8916	Spent Fuel Area
2RE-8912	Containment Incore Instrumentation

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
ABNORMAL RADIATION LEVELS			
		AA3 1 2 3 4 5 6 D Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions. **<u>Emergency Action Level(s):</u>** 1. Dose rate > 15 mR/hr in any of the following areas requiring continuous occupancy to maintain plant safety functions: • Unit 1 Control Room • Unit 2 Control Room • Central Alarm Station	**<u>OR</u>** **AU2** *(continued)* 2. UNPLANNED VALID Area Radiation Monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels. **NOTE:** *For area radiation monitors with ranges incapable of measuring 1000 times normal* levels, classification shall be based on VALID full scale indication unless surveys confirm that area radiation levels are below 1000 times normal* within 15 minutes of the Area Radiation Monitor indications going to full scale indication.* * Normal can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

TAB C

Cold Shutdown / Refueling System Malfunction

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of RCS / Reactor Vessel Inventory

GENERAL EMERGENCY

CG1 5 6

Loss of RCS / reactor vessel inventory affecting fuel clad integrity with containment challenged

Emergency Action Level(s):

NOTE:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

1. a. Core exit thermocouples indicate superheat for ≥ 30 minutes.

   AND

   b. Any of the following containment challenge indications:

   - CONTAINMENT CLOSURE not established
   - Explosive mixture inside containment
   - UNPLANNED rise in containment pressure

   OR

2. a. RCS / reactor vessel level cannot be monitored for ≥ 30 minutes with a loss of RCS/ reactor vessel inventory as indicated by any of the following:

SITE AREA EMERGENCY

CS1 5 6

Loss of RCS / reactor vessel inventory affecting core decay heat removal capability

Emergency Action Level(s):

NOTE:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

1. With CONTAINMENT CLOSURE **not** established:

   Loss of RCS / reactor vessel level as indicated by:

   Unit 1: RVLMS Levels 1 through 9 indicate DRY

   Unit 2: RVLMS Levels 1 through 6 indicate DRY

   OR

2. With CONTAINMENT CLOSURE established, core exit thermocouples indicate superheat.

   OR

ALERT

CA1 5 6

Loss of RCS / reactor vessel inventory

Emergency Action Level(s):

NOTE:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

1. Loss of RCS / reactor vessel inventory as indicated by:

   Unit 1: RVLMS Levels 1 through 8 indicate DRY

   Unit 2: RVLMS Levels 1 through 5 indicate DRY

   OR

   Unit 1: Reactor vessel level <368 ft., 0 in. (bottom of the hot leg)

   Unit 2: Reactor vessel level < 369 ft., 1.5 in. (bottom of the hot leg)

   OR

UNUSUAL EVENT

CU1 5

RCS leakage

Emergency Action Level(s):

NOTE:

The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

1. RCS leakage results in the inability to maintain or restore level within Pressurizer or RCS level target band for ≥ 15 minutes.

CU2 6

UNPLANNED loss of RCS / reactor vessel Inventory

Emergency Action Level(s):

NOTE:

The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

1. UNPLANNED RCS / reactor vessel level drop as indicated by either of the following:

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of RCS / Reactor Vessel Inventory			
CG1 *(continued)* • Containment High Range Radiation Monitor reading >10 R/hr • Erratic source range monitor indication • Unexplained level rise in Reactor Building Sump, Reactor Drain Tank, Quench Tank, Aux. Building Equipment Drain Tank, or Aux. Building Sump **AND** b. Any of the following containment challenge indications: • CONTAINMENT CLOSURE not established • Explosive mixture inside containment • UNPLANNED rise in containment pressure	**CS1** *(continued)* 3. RCS / reactor vessel level cannot be monitored for ≥ 30 minutes with a loss of RCS / reactor vessel inventory as indicated by any of the following: • Containment High Range Radiation Monitor reading > 10 R/hr • Erratic source range monitor indication • Unexplained level rise in Reactor Building Sump, Reactor Drain Tank, Quench Tank, Aux. Building Equipment Drain Tank, or Aux. Building Sump	**CA1** *(continued)* 2. RCS / reactor vessel level cannot be monitored for ≥ 15 minutes with a loss of RCS / reactor vessel inventory as indicated by an unexplained level rise in the Reactor Building Sump, Reactor Drain Tank, Aux. Building Equipment Drain Tank, Aux. Building Sump, or Quench Tank.	**CU2** *(continued)* a. RCS / reactor vessel water level drop below the reactor vessel flange for ≥15 minutes when the RCS / reactor vessel level band is established above the reactor vessel flange. **OR** b. RCS / reactor vessel water level drop below the RCS / reactor vessel level band for ≥ 15 minutes when the RCS / reactor vessel level band is established below the reactor vessel flange. **OR** 2. RCS / reactor vessel level cannot be monitored with a loss of RCS / reactor vessel inventory as indicated by an unexplained level rise in (as applicable) the Reactor Building Sump, Reactor Drain Tank, Aux. Building Equipment Drain Tank, Aux. Building Sump, or Quench Tank.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of Decay Heat Removal

ALERT

CA3 5 6

Inability to maintain plant in Cold Shutdown

Emergency Action Level(s):

1. An UNPLANNED event results in RCS temperature > 200 °F > the specified duration in Table C1.

Table C1
RCS Reheat Duration Thresholds

RCS	Containment Closure	Duration
Intact (but not RCS lowered inventory)	N/A	60 minutes*
Not intact or RCS lowered inventory	Established	20 minutes*
	Not Established	0 minutes

*If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.

OR

NOTE: *EAL #2 does not apply in solid plant conditions.*

2. An UNPLANNED event results in RCS pressure rise > 10 psi due to a loss of RCS cooling.

UNUSUAL EVENT

CU3 5 6

UNPLANNED loss of decay heat removal capability with irradiated fuel in the reactor vessel

Emergency Action Level(s):

NOTE:

The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

1. UNPLANNED event results in RCS temperature exceeding 200 °F.

OR

2. Loss of all RCS temperature and RCS / reactor vessel level indication for ≥ 15 minutes.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of AC Power			
		CA5 5 6 D Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 15 minutes **Emergency Action Level(s):** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.* 1. Loss of all offsite and all onsite AC power to Vital 4.16KV busses ≥ 15 minutes.	**CU5** 5 6 AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes such that any additional single power source failure would result in station blackout **Emergency Action Level(s):** **NOTE:** *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.* 1. a. AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes. **AND** b. Any additional single power source failure will result in station blackout.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of DC Power			
			CU6 5 6 Loss of required DC power ≥ 15 minutes **<u>Emergency Action Level(s):</u>** **NOTE:** *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.* 1. < 105 volts on required Vital DC bus ≥ 15 minutes.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Inadvertant Criticality			
			CU7 5 6 Inadvertent criticality **<u>Emergency Action Level(s):</u>** 1. UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of Communications			
			CU8 (5 6 D) Loss of all onsite or offsite communications capabilities **Emergency Action Level(s):** 1. Loss of all Table C2 onsite communication methods affecting the ability to perform routine operations. **Table C2 Onsite Communications Equipment**: Station radio system; Plant paging system; In-plant telephones; Gaitronics **OR** 2. Loss of all Table C3 offsite communication methods affecting the ability to perform offsite notifications. **Table C3 Offsite Communications Equipment**: All telephone lines (commercial and microwave); ENS

TAB E

Independent Spent Fuel Storage Installation (ISFSI) Malfunction

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
ISFSI MALFUNCTION – Cask Damage			
			E-HU1 1 2 3 4 5 6 D **Note:** Security Events are bounded by the Hazards EALs. Damage to a loaded cask CONFINEMENT BOUNDARY **Emergency Action Level(s):** 1. Damage to a loaded cask CONFINEMENT BOUNDARY.

TAB F

Fission Product Barrier Degradation

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
FISSION PRODUCT BARRIER DEGRADATION – Barriers			
FG1 1 2 3 4	**FS1** 1 2 3 4	**FA1** 1 2 3 4	**FU1** 1 2 3 4
Loss of ANY two barriers AND loss or potential loss of third barrier	Loss or potential loss of ANY two barriers	ANY loss or ANY potential loss of EITHER fuel clad or RCS	ANY loss or ANY potential loss of containment

Note: Determine which combination of the three barriers are lost or have a potential loss and use the above key to classify the event. Also, multiple events could occur which result in the conclusion that exceeding the loss or potential loss EALs is IMMINENT. In this IMMINENT loss situation use judgment and classify as if the EALs are exceeded.

Fuel Clad Barrier EALs		RCS Barrier EALs		Containment Barrier EALs	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. <u>Primary Coolant Activity Level (FCB1)</u>		**1. <u>RCS Leak Rate (RCB1)</u>**		**1. <u>Containment Pressure (CNB1)</u>**	
1. Coolant activity > 300 µCi/gm dose equivalent I-131 activity by Chemistry sample **<u>OR</u>** 2. Radiation levels > 1000 MR/hr **Unit 1:** at SA-229 **Unit 2:** at 2TCD-19	None	RCS leak rate > available makeup capacity as indicated by: **Unit 1:** Loss of adequate subcooling margin **Unit 2:** RCS subcooling (MTS) can NOT be maintained at least 30 °F	**Unit 1:** UNISOLABLE RCS leak > 50 gpm with Letdown isolated **Unit 2:** UNISOLABLE RCS leak > 44 gpm with Letdown isolated	1. Rapid unexplained drop in containment pressure following an initial rise in containment pressure **<u>OR</u>** 2. Containment pressure or sump level response not consistent with LOCA conditions	1. **Unit 1:** Containment pressure 73.7 PSIA (59 PSIG) and rising **Unit 2:** Containment pressure 73.7 PSIA and rising **<u>OR</u>** 2. Explosive mixture exists inside Containment **<u>OR</u>** 3. a. Containment Pressure > containment spray actuation setpoint **Unit 1:** 44.7 PSIA (30 PSIG) **Unit 2:** 23.3 PSIA **<u>AND</u>** b. LESS THAN one full train of spray operating

Fuel Clad Barrier EALs		RCS Barrier EALs		Containment Barrier EALs	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
2. Core Exit Thermocouple Readings (FCB2)		**2. SG Tube Rupture (RCB2)**		**2. Core Exit Thermocouple Readings (CNB2)**	
> 1200 °F CET temperature	**Unit 1:** ICC exists as evidenced by CETs indicating superheated conditions **Unit 2:** Average CETs indicate superheat for current RCS pressure	SGTR that results in an ECCS (SI) actuation	None	None	1. a. CETs indicate > 1200 °F **AND** b. Restoration procedures not effective within 15 minutes **OR** 2. a. CETs indicate > 700 °F **AND** b. RVLMS indicates **Unit 1:** Levels 1 through 9 DRY **Unit 2:** Levels 1 through 7 DRY **AND** c. Restoration procedures not effective within 15 minutes

Fuel Clad Barrier EALs		RCS Barrier EALs		Containment Barrier EALs	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
3. Reactor Vessel Water Level (FCB3)		**3. Containment Radiation Monitoring (RCB3)**		**3. SG Secondary Side Release With Primary-to-Secondary Leakage (CNB3)**	
None	**Unit 1:** RVLMS Levels 1 through 9 indicate DRY **Unit 2:** RVLMS Levels 1 through 7 indicate DRY	Containment high range radiation monitor reading > 100 R/hr	None	1. RUPTURED steam generator is also FAULTED outside of containment **OR** 2. a. Primary to secondary leakrate > 10 gpm **AND** b. UNISOLABLE steam release from affected steam generator to the environment	None
4. Containment Radiation Monitoring (FCB4)		**4. Emergency Director Judgment (RCB4)**		**4. Containment Isolation Failure or Bypass (CNB4)**	
Containment high range radiation monitor reading > 1000 R/hr	None	Any condition in the opinion of the SM / ED that indicates Loss or Potential Loss of the RCS barrier		1. UNISOLABLE breach of containment **AND** 2. Direct downstream pathway to the environment exists after containment isolation signal	None

Fuel Clad Barrier EALs		RCS Barrier EALs		Containment Barrier EALs	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
5. Core Damage Assessment (FCB5)				**5. Containment Radiation Monitoring (CNB5)**	
At least 5% fuel clad damage as determined from core damage assessment	None			None	Containment high range radiation monitor reading > 4000 R/hr
6. Emergency Director Judgment (FCB6)				**6. Other Indications (CNB6)**	
Any condition in the opinion of the SM/ED that indicates Loss or Potential Loss of the fuel clad barrier				Elevated readings on the following radiation monitors that indicate loss or potential loss of the Containment barrier: (see monitor table below)	
				7. Emergency Director Judgment (CNB7)	
				Any condition in the opinion of the SM / ED that indicates Loss or Potential Loss of the containment barrier	

MONITORS – Unit 1	
RX-9820	Containment Purge
RX-9825	Radwaste Area
RX-9830	Fuel Handling Area
RX-9835	Emergency Penetration Room
MONITORS – Unit 2	
2RX-9820	Containment Purge
2RX-9825	Radwaste Area
2RX-9830	Fuel Handling Area
2RX-9835	Emergency Penetration Room
2RX-9840	Post Accident Sampling Building
2RX-9845	Auxiliary Building Extension

TAB H

Hazards and Other Conditions Affecting Plant Safety

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY – Security			
HG1 1 2 3 4 5 6 D HOSTILE ACTION resulting in loss of physical control of the facility **Emergency Action Level(s):** 1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions. **OR** 2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.	**HS1** 1 2 3 4 5 6 D HOSTILE ACTION within the PROTECTED AREA **Emergency Action Level(s):** 1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by ANO Security Shift Supervision.	**HA1** 1 2 3 4 5 6 D HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat **Emergency Action Level(s):** 1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by ANO Security Shift Supervision. **OR** 2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.	**HU1** 1 2 3 4 5 6 D Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant **Emergency Action Level(s):** 1. A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by ANO Security Shift Supervision. **OR** 2. A credible site specific security threat notification. **OR** 3. A validated notification from NRC providing information of an aircraft threat.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY – Discretionary			
HG2 1 2 3 4 5 6 D Other conditions exist which in the judgment of the SM / ED warrant declaration of General Emergency **Emergency Action Level(s):** 1. Other conditions exist which in the judgment of the SM / ED indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS2** 1 2 3 4 5 6 D Other conditions exist which in the judgment of the SM / ED warrant declaration of a Site Area Emergency **Emergency Action Level(s):** 1. Other conditions exist which in the judgment of the SM / ED indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA2** 1 2 3 4 5 6 D Other conditions exist which in the judgment of the SM / ED warrant declaration of an Alert **Emergency Action Level(s):** 1. Other conditions exist which in the judgment of the SM / ED indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU2** 1 2 3 4 5 6 D Other conditions exist which in the judgment of the SM warrant declaration of an NUE **Emergency Action Level(s):** 1. Other conditions exist which in the judgment of the SM indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY – Control Room Evacuation			
	HS3 1 2 3 4 5 6 D Control Room evacuation has been initiated and plant control cannot be established **<u>Emergency Action Level(s):</u>** 1. a. Control Room evacuation has been initiated. **<u>AND</u>** b. Control of the plant cannot be established in accordance with the following procedures within 15 minutes: **Unit 1:** 1203.002, “Alternate Shutdown” **Unit 2:** 2203.014, “Alternate Shutdown”	**HA3** 1 2 3 4 5 6 D Control Room evacuation has been initiated **<u>Emergency Action Level(s):</u>** 1. Alternate Shutdown procedure requires Control Room evacuation: **Unit 1:** 1203.002, “Alternate Shutdown” **Unit 2:** 2203.014, “Alternate Shutdown”	

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY			
		Fire	
		[1]HA4 1 2 3 4 5 6 D FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown **<u>Emergency Action Level(s):</u>** 1. FIRE or EXPLOSION resulting in VISIBLE DAMAGE to any **Table H1** structure or area containing safety systems or components <u>or</u> Control Room indication of degraded performance of those safety systems:	**[1]HU4** 1 2 3 4 5 6 D FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection **<u>OR</u>** EXPLOSION within the PROTECTED AREA **<u>Emergency Action Level(s):</u>** **NOTE:** *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. FIRE in any **Table H1** structure or area not extinguished: a. within 15 minutes of Control Room notification **<u>OR</u>** b. within 15 minutes of [2]verification of a Control Room FIRE alarm (i.e. Alarm valid until disproved) **<u>OR</u>** 2. EXPLOSION within the PROTECTED AREA.

[1]The HA4 and HU4 EALs apply to any Table H1 structure or area whether in service or tagged out for maintenance.

[2]Verification of a fire detection system alarm/actuation includes actions that can be taken within the Control Room or other nearby site specific location to ensure that it is not spurious.

Table H1

Unit 1

Reactor Building

All Elevations

Aux Building

All Elevations Including Penthouse/MSIV Room
Exceptions: Boric Acid Mix Tank Room (Chem Add Area) 404’ (157-B)
EDG Exhaust Fan area on 386’ (1-E and 2-E)

Turbine Building

All Elevations
Including:
Pipechase under ICW Coolers
CRD Pump Pit / T-28 Room / Area under ICW Pumps

Outside Areas

Manholes adjacent to Startup #2 XFMR (MH-03/MH-04)
Manholes adjacent to Intake Structure (MH-05/MH-06)
Intake Structure (354’ and 366’)
Diesel Fuel Vault
Diesel Fuel Vault Pump Manholes MH-09 and MH-10 (Manhole, MH-09, is located approximately 15 feet northeast of the Unit 1 QCST , Manhole, MH-10, is located approximately 5 feet west of Unit 2 Condensate Storage Tank, 2T-41A)

Table H1

Unit 2

Reactor Building

All Elevations

Aux Building

All Elevations including Aux Extensions

Turbine Building

All Elevations

Outside Areas

Intake Structure (354’ and 366’)
Concrete Manhole East, NE of intake
Concrete Manhole East of Turbine building next to train bay
Diesel Fuel Vault
Diesel Fuel Vault Pump Manholes MH-09 and MH-10 (Manhole, MH-09, is located approximately 15 feet northeast of the Unit 1 QCST , Manhole, MH-10, is located approximately 5 feet west of Unit 2 Condensate Storage Tank, 2T-41A)

<table>
<tr><th>GENERAL EMERGENCY</th><th>SITE AREA EMERGENCY</th><th>ALERT</th><th>UNUSUAL EVENT</th></tr>
<tr><td colspan="4">HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY</td></tr>
<tr><td colspan="2" rowspan="2"></td><td colspan="2">Toxic Gas</td></tr>
<tr><td>HA5 1 2 3 4 5 6 D

Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant, or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor

<u>Emergency Action Level(s):</u>

NOTE:
If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

1. Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant, or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor.</td><td>HU5 1 2 3 4 5 6 D

Release of toxic, corrosive, asphyxiant, or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS

<u>Emergency Action Level(s):</u>

1. Toxic, corrosive, asphyxiant, or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS.

<u>OR</u>

2. Report by Local, County or State officials for evacuation or sheltering of site personnel based on an offsite event.</td></tr>
</table>

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

Toxic Gas

HA5 *(continued)* 1 2 3 4 5 6 D

Unit 1

VITAL AREA	APPLICABLE MODES
A-4 Switchgear Room	3, 4
Upper North Electrical Penetration Room	3, 4
Lower South Electrical Equipment Room	3, 4
Control Room	ALL

Unit 2

VITAL AREA	APPLICABLE MODES
Auxiliary Building 317' Emergency Core Cooling Rooms	3, 4
Auxiliary Building 317' Tendon Gallery Access	3, 4
Auxiliary Building 335' Charging Pumps/ 2B-52	3, 4
Auxiliary Building 354' 2B-62 Area	3, 4
Emergency Diesel Generator Corridor	3, 4
Lower South Piping Penetration Room	3, 4
Auxiliary Building 386' Containment Hatch	3, 4
Control Room	ALL

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY			
		Natural or Destructive Phenomena	
		HA6 1 2 3 4 5 6 D Natural or destructive phenomena affecting VITAL AREAS **Emergency Action Level(s):** 1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by annunciation of the 0.1g acceleration alarm. **AND** b. Earthquake confirmed by ANY of the following: • Earthquake felt in plant • National Earthquake Center • Control Room indication of degraded performance of systems required for the safe shutdown of the plant **OR**	**HU6** 1 2 3 4 5 6 D Natural or destructive phenomena affecting the PROTECTED AREA **Emergency Action Level(s):** 1. Seismic event identified by any 2 of the following: • Seismic event confirmed by annunciation of the 0.01g acceleration alarm • Earthquake felt in plant • National Earthquake Center **OR** 2. Tornado striking within PROTECTED AREA boundary or high winds > 67 mph. (**2 minute average**) **OR** 3. Internal flooding that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode in any of the structures or areas in **Table H1. (Page 47)** **OR** 4. Turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY			
		Natural or Destructive Phenomena	
		HA6 *(continued)* 2. Tornado striking or winds > 67 mph (2 minute average) resulting in VISIBLE DAMAGE to any of the following structures/equipment containing safety systems or components <u>or</u> Control Room indication of degraded performance of those safety systems: • Reactor Building • Intake Structure • Ultimate Heat Sink • BWST/RWT • Auxiliary Building • Turbine Building • QCST • Control Room • Startup Transformers • Diesel Fuel Vault **<u>OR</u>** 3. Internal flooding in any of the following areas resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment <u>or</u> Control Room indication of degraded performance of those safety systems: • Intake Structure • Auxiliary Building • Turbine Building	**HU6** *(continued)* 5. Lake Dardanelle level < 335 feet. **<u>OR</u>** 6. Lake Dardanelle level > 345 feet.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY			
		Natural or Destructive Phenomena	
		HA6 *(continued)* **<u>OR</u>** 4. Turbine failure-generated PROJECTILES resulting in VISIBLE DAMAGE to or penetration of any of the structures/equipment containing safety systems or components <u>or</u> Control Room indication of degraded performance of those safety systems: • Auxiliary Building • Turbine Building • Control Room • Startup Transformers **<u>OR</u>** 5. Lake Dardanelle level < 335 feet and Emergency Cooling Pond inoperable. **<u>OR</u>**	

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY			
		Natural or Destructive Phenomena	
		HA6 *(continued)* 6. Vehicle crash resulting in VISIBLE DAMAGE to any of the structures/equipment containing safety systems or components or Control Room indication of degraded performance of those safety systems: • Reactor Building • Intake Structure • Ultimate Heat Sink • BWST/RWT • Auxiliary Building • Turbine Building • QCST • Startup Transformers • Diesel Fuel Vault	

TAB S

System Malfunction

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Loss of AC Power			
SG1 1 2 3 4	**SS1** 1 2 3 4	**SA1** 1 2 3 4	**SU1** 1 2 3 4
Prolonged loss of all offsite and all onsite AC power to Vital 4.16 KV busses **Emergency Action Level(s):** 1. a. Loss of all offsite and all onsite AC power to Vital 4.16 KV busses. **AND** b. Either of the following: • Restoration of at least one Vital 4.16 KV bus in < 4 hours is not likely. **OR** • Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by CETs ≥ 700 °F.	Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 15 minutes **Emergency Action Level(s):** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 15 minutes.	AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes such that any additional single power source failure would result in station blackout **Emergency Action Level(s):** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. a. AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes. **AND** b. Any additional single power source failure will result in station blackout.	Loss of all offsite AC power to Vital 4.16 KV busses ≥ 15 minutes **Emergency Action Level(s):** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. Loss of all offsite AC power to Vital 4.16 KV busses ≥ 15 minutes.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Failure of Reactor Protection System			
SG3 1 2 Automatic trip and all manual actions fail to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists **Emergency Action Level(s):** 1. a. An automatic trip failed to shutdown the reactor. **AND** b. All manual actions do not shutdown the reactor as indicated by reactor power ≥ 5%. **AND** c. Either of the following exist or have occurred due to continued power generation: • CET temperatures at or approaching 1200 °F. **OR** • Feedwater flow rate less than: **Unit 1:** 430 gpm **Unit 2:** 485 gpm	**SS3** 1 2 Automatic trip fails to shutdown the reactor and manual actions taken from the reactor control console are not successful in shutting down the reactor **Emergency Action Level(s):** 1. a. An automatic trip failed to shutdown the reactor. **AND** b. Manual actions taken at panel C03 (Unit 1) or panels 2C03/2C14 (Unit 2) do not shutdown the reactor as indicated by reactor power ≥ 5%.	**SA3** 1 2 Automatic trip fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor **Emergency Action Level(s):** 1. a. An automatic trip failed to shutdown the reactor as indicated by reactor power ≥ 5%. **AND** b. Manual actions taken at panel C03 (Unit 1) or panels 2C03/2C14 (Unit 2) successfully shutdown the reactor as indicated by reactor power < 5%.	

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Loss of DC Power			
	SS4 1 2 3 4 Loss of all Vital DC power ≥ 15 minutes **<u>Emergency Action Level(s):</u>** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. < 105 volts on all Vital DC busses ≥ 15 minutes.		

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Loss of Annunciators			
	SS6 1 2 3 4 Inability to monitor a SIGNIFICANT TRANSIENT in progress **Emergency Action Level(s):** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. a. UNPLANNED loss of > approximately 75% of the following ≥ 15 minutes: • Control Room annunciators associated with safety systems. **OR** • Control Room safety system indication. **AND** b. A SIGNIFICANT TRANSIENT in progress. **AND** c. Compensatory indications are unavailable.	**SA6** 1 2 3 4 UNPLANNED loss of safety system annunciation or indication in the Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) compensatory indicators unavailable **Emergency Action Level(s):** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. a. UNPLANNED loss of > approximately 75% of the following ≥ 15 minutes: • Control Room annunciators associated with safety systems. **OR** • Control Room safety system indication. **AND** b. Either of the following: • A SIGNIFICANT TRANSIENT is in progress **OR** • Compensatory indications are unavailable	**SU6** 1 2 3 4 UNPLANNED loss of safety system annunciation or indication in the Control Room for ≥ 15 minutes **Emergency Action Level(s):** **NOTE:** *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. UNPLANNED loss of > approximately 75% of the following ≥ 15 minutes: a. Control Room annunciators associated with safety systems. **OR** b. Control Room safety system indication.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – RCS Leakage			
			SU7 1 2 3 4 RCS leakage **Emergency Action Level(s):** 1. Unidentified or pressure boundary leakage > 10 gpm. **OR** 2. Identified leakage > 25 gpm.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Loss of Communications			
			SU8 1 2 3 4 Loss of all onsite or offsite communications capabilities **Emergency Action Level(s):** 1. Loss of all Table M1 onsite communications methods affecting the ability to perform routine operations. **Table M1 Onsite Communications Methods** Station radio system Plant paging system In-plant telephones Gaitronics **OR** 2. Loss of all Table M2 offsite communications methods affecting the ability to perform offsite notifications. **Table M2 Offsite Communications Methods** All telephone lines (commercial and microwave) ENS

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Fuel Clad Degradation			
			SU9 1 2 3 4 Fuel clad degradation **Emergency Action Level(s):** 1. Failed Fuel Iodine radiation monitor reading indicates fuel clad degradation > Technical Specification allowable limits: **Unit 1:** RI-1237S reads > 1.3 x 10^5 cpm **Unit 2:** 2RITS-4806B reads > .65 x 10^5 cpm **OR** 2. RCS sample activity value indicating fuel clad degradation > Technical Specification allowable limits: • > 1.0 uCi/gm Dose Equivalent I-131 for more than 48 hours **OR** • **Unit 1:** ≥ 60 uCi/gm Dose Equivalent I-131 **Unit 2:** > 60 uCi/gm Dose Equivalent I-131 **OR**

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Fuel Clad Degradation			
			SU9 *(continued)* • **Unit 1:** > 2200 µCi/gm Dose Equivalent Xe-133 for more than 48 hours • **Unit 2:** > 3100 µCi/gm Dose Equivalent Xe-133 for more than 48 hours
SYSTEM MALFUNCTION – Inadvertant Criticality			
			SU10 [3 4] Inadvertent criticality **Emergency Action Level(s):** 1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.
SYSTEM MALFUNCTION – Failure to Shutdown			
			SU11 [1 2 3 4] Inability to reach required operating mode within Technical Specification limits **Emergency Action Level(s):** 1. A Plant is not brought to required operating mode within Technical Specifications LCO action statement time.

Attachment 2

Arkansas Nuclear One

EAL Basis Document

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ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AU1

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

Any release of gaseous or liquid radioactivity to the environment >2 times the ODCM limits for ≥60 minutes

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2 or 3)

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.*

1. VALID reading on Channel 7 on any of the following radiation monitors > the reading shown for ≥ 60 minutes:

MONITORS – UNIT 1		LIMIT
RX-9820	Containment Purge	1.18E-02 µCi/cc
RX-9825	Radwaste Area	1.07E-02 µCi/cc
RX-9830	Fuel Handling Area	9.08E-03 µCi/cc
RX-9835	Emergency Penetration Room	1.91E-01 µCi/cc
MONITORS – UNIT 2		**LIMIT**
2RX-9820	Containment Purge	8.92E-03 µCi/cc
2RX-9825	Radwaste Area	6.64E-03 µCi/cc
2RX-9830	Fuel Handling Area	8.92E-03 µCi/cc
2RX-9835	Emergency Penetration Room	1.77E-01 µCi/cc
2RX-9840	Post Accident Sampling Building	8.84E-02 µCi/cc
2RX-9845	Aux. Building Extension	2.53E-02 µCi/cc
2RX-9850	Low Level Radwaste Storage Bldg.	3.54E-02 µCi/cc

<u>OR</u>

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AU1

2. VALID reading on any of the following radiation monitors >2 times the alarm setpoint established by a current release permit for ≥60 minutes.

EFFLUENT MONITORS – Unit 1	
RX-9820	Containment Purge (Channel 7 or 9)
RE-4830	Waste Gas Radiation Monitor
RE-4642	Liquid Radwaste Monitor
EFFLUENT MONITORS – Unit 2	
2RX-9820	Containment Purge (Channel 7 or 9)
2RE-2429	Waste Gas Decay Tank Vent Line Radiation Monitor
2RE-2330	BMS Liquid Discharge Monitor
2RE-4423	Regenerative Waste Discharge Monitor
2RE-4425	SG Blowdown to Flume Radiation Monitor

OR

3. Confirmed grab sample analyses for gaseous or liquid releases indicates concentrations or release rates >2 times the applicable values of the ODCM for ≥60 minutes.

Basis:

The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This IC addresses a potential reduction in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

ANO incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The ODCM multiples are specified in AU1 and AA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, NOT the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged over 60 minutes. For example, a release exceeding 4 times ODCM limits for 30 minutes does not meet the threshold for this IC.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AU1

This Initiating Condition includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

EAL #1

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the EAL.

This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

EAL #2

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in this Initiating Condition established by the release permit. This value may be associated with a planned batch release, or a continuous release path.

EAL #3

This EAL addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, lake, etc.

EAL #1 and #2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the ODCM and is used in calculating the alarm setpoints.

Reference Documents:

1. 1604.051, “Eberline Radiation Monitor System”
2. Offsite Dose Calculation Manual

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AU2

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

UNPLANNED rise in plant radiation levels

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2)

1. a. UNPLANNED lowering of water level in the refueling canal or spent fuel pool as indicated by:

   - Personnel observation, refueling crew report, indication on area security camera, borated water source (BWST or RWT) level drop due to makeup demands.

   AND

   b. VALID Area Radiation Monitor reading rise on any of the following:

Unit 1	
RE-8009	Spent Fuel Area
RE-8017	Fuel Handling Area
Unit 2	
2RE-8914	Spent Fuel Area
2RE-8915	Spent Fuel Area
2RE-8916	Spent Fuel Area
2RE-8912	Containment Incore Instrumentation

OR

2. UNPLANNED VALID Area Radiation Monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels.

Note: *For area radiation monitors with ranges incapable of measuring 1000 times normal* levels, classification shall be based on VALID full scale indication unless surveys confirm that area radiation levels are below 1000 times normal* within 15 minutes of the Area Radiation Monitor indications going to full scale indication.*

* Normal can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AU2

Basis:

This IC addresses elevated radiation levels as a result of lowered water level above irradiated fuel or events that have resulted, or may result, in UNPLANNED rises in radiation dose rates within plant buildings. These radiation rises represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

EAL #1

The refueling pathway is a site specific combination of cavities, tubes, canals and pools. While a radiation monitor could detect a rise in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, a refueling bridge ARM reading may rise due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Also, a monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Generally, elevated radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss.

For refueling events where the water level drops below the RPV flange classification would be via CU2. This event escalates to an Alert per AA2 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Matrix for events in operating Modes 1-4.

EAL #2

This EAL addresses rises in plant radiation levels that represent a loss of control of radioactive material resulting in a potential degradation in the level of safety of the plant.

This EAL excludes radiation level rises that result from planned activities such as use of radiographic sources and movement of radioactive waste materials. A specific list of ARMs is not required as it would restrict the applicability of the Threshold. The intent is to identify loss of control of radioactive material in any monitored area.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AA1

Initiating Condition - ALERT

Any release of gaseous or liquid radioactivity to the environment >200 times the ODCM limits for ≥15 minutes

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2 or 3)

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.*

1. VALID reading on Channel 7 on any of the following radiation monitors > the reading shown for ≥ 15 minutes:

MONITORS – UNIT 1		LIMIT
RX-9820	Containment Purge	1.18E+00 µCi/cc
RX-9825	Radwaste Area	1.07E+00 µCi/cc
RX-9830	Fuel Handling Area	9.08E-01 µCi/cc
RX-9835	Emergency Penetration Room	1.91E+01 µCi/cc
MONITORS – UNIT 2		**LIMIT**
2RX-9820	Containment Purge	8.92E-01 µCi/cc
2RX-9825	Radwaste Area	6.64E-01 µCi/cc
2RX-9830	Fuel Handling Area	8.92E-01 µCi/cc
2RX-9835	Emergency Penetration Room	1.77E+01 µCi/cc
2RX-9840	Post Accident Sampling Building	8.84E+00 µCi/cc
2RX-9845	Aux. Building Extension	2.53E+00 µCi/cc
2RX-9850	Low Level Radwaste Storage Bldg.	3.54E+00 µCi/cc

<u>**OR**</u>

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AA1

2. **<u>EITHER</u>** VALID reading on any of the following radiation monitors > 200 times the alarm setpoint established by a current release permit for ≥ 15 minutes **<u>OR</u>** VALID reading greater than the value listed for ≥ 15 minutes.

MONITORS – UNIT 1		LIMIT
RX-9820	Containment Purge (Channel 7 or 9)	N/A
RE-4830	Waste Gas Radiation Monitor	9.5E7 cpm
RE-4642	Liquid Radwaste Monitor	9.5E7 cpm
MONITORS – UNIT 2		**LIMIT**
2RX-9820	Containment Purge (Channel 7 or 9)	N/A
2RE-2429	Waste Gas Monitoring System	9.5E5 cpm
2RE-2330	BMS Liquid Discharge Monitor	9.5E5 cpm
2RE-4423	Regenerative Waste Discharge Monitor	9.5E5 cpm
2RE-4425	SG Blowdown to Flume Radiation Monitor	9.5E5 cpm

<u>OR</u>

3. Confirmed grab sample analyses for gaseous or liquid releases indicates concentrations or release rates > 200 times the applicable values of the ODCM for ≥ 15 minutes.

Basis:

The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This IC addresses an actual or substantial potential reduction in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. ANO incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The ODCM multiples are specified in AU1 and AA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an offsite dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, NOT the magnitude of the associated dose or dose rate.

Releases should <u>not</u> be prorated or averaged. For example, a release exceeding 600 times ODCM limits for 5 minutes does not meet the threshold for this IC.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AA1

This Initiating Condition includes any release for which a release permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

EAL #1

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the Initiating Condition.

This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

EAL #2

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in this Initiating Condition established by the radioactivity discharge permit. This value may be associated with a planned batch release, or a continuous release path. The limit values provided are for those cases in which the maximum monitor range is less than the release permit value multiplied by 200.

EAL #3

This EAL addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, lake, etc.

EAL #1 and #2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the ODCM and is used in calculating the alarm setpoints.

Reference Documents:

1. 1604.051, “Eberline Radiation Monitor System”
2. Offsite Dose Calculation Manual

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AA2

Initiating Condition - ALERT

Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2)

1. A water level drop in the refueling canal or spent fuel pool that will result in irradiated fuel becoming uncovered.

   OR

2. VALID alarm on any of the following radiation monitors due to damage to irradiated fuel or loss of water level.

Unit 1	
RX-9820	Containment Purge (Channel 7 or 9)
RX-9825	Radwaste Area (Channel 7 or 9)
RX-9830	Fuel Handling Area (Channel 7 or 9)
RE-8060	Containment High Range Radiation Monitors
RE-8061	Containment High Range Radiation Monitors
RE-8009	Spent Fuel Area
RE-8017	Fuel Handling
Unit 2	
2RX-9820	Containment Purge (Channel 7 or 9)
2RX-9825	Radwaste Area (Channel 7 or 9)
2RX-9830	Fuel Handling Area (Channel 7 or 9)
2RE-8905	Containment Equipment Hatch Area
2RE-8909	Containment Personnel Access Area
2RE-8925-1	Containment High Range Radiation Monitors
2RE-8925-2	Containment High Range Radiation Monitors
2RE-8914	Spent Fuel Area
2RE-8915	Spent Fuel Area
2RE-8916	Spent Fuel Area
2RE-8912	Containment Incore Inst.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AA2

Basis:

This IC addresses rises in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent an actual or substantial potential degradation in the level of safety of the plant.

These events escalate from AU2 in that fuel activity has been released, or is anticipated due to fuel heatup. This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

EAL #1

Indications may include instrumentation such as water level and local area radiation monitors, and personnel (e.g., refueling crew) reports. Depending on available level indication, the declaration may be based on indications of water makeup rate or drop in applicable borated water storage tank level. Video cameras (Security or outage-related) may allow remote observation of level.

EAL #2

This EAL addresses radiation monitor indications of fuel uncovery and/or fuel damage.

Elevated ventilation monitor readings may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Elevated background at the ventilation monitor due to water level drop may mask elevated ventilation exhaust airborne activity and needs to be considered.

While a radiation monitor could detect a rise in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

For example, a refueling bridge ARM reading may rise due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Also, a monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Generally, elevated radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss.

Escalation of this emergency classification level, if appropriate, would be based on AS1 or AG1.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AA3

Initiating Condition - ALERT

Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions

Operating Mode Applicability: All

Example Emergency Action Level(s):

Dose rate > 15 mR/hr in any of the following areas requiring continuous occupancy to maintain plant safety functions:

- Unit 1 Control Room
- Unit 2 Control Room
- Central Alarm Station

Basis:

This IC addresses elevated radiation levels that impact continued operation in areas requiring continuous occupancy to maintain safe operation or to perform a safe shutdown.

The cause and/or magnitude of the rise in radiation levels is <u>not</u> a concern of this IC. The SM/ED must consider the source or cause of the elevated radiation levels and determine if any other IC may be involved.

This IC is not meant to apply to rises in the containment dome radiation monitors as these are events which are addressed in the fission product barrier matrix EALs.

Areas requiring continuous occupancy include the Control Rooms and the Central Alarm Station.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AS1

Initiating Condition -- SITE AREA EMERGENCY

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity >100 mR TEDE or 500 mR child thyroid CDE for the actual or projected duration of the release

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2 or 3)

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.*

1. VALID reading on Channel 9 on any of the following radiation monitors > the reading shown for ≥ 15 minutes:

MONITORS – UNIT 1		LIMIT
RX-9820	Containment Purge	1.18E+01 µCi/cc
RX-9825	Radwaste Area	1.07E+01 µCi/cc
RX-9830	Fuel Handling Area	9.08E+00 µCi/cc
RX-9835	Emergency Penetration Room	1.91E+02 µCi/cc
MONITORS – UNIT 2		**LIMIT**
2RX-9820	Containment Purge	8.92E+00 µCi/cc
2RX-9825	Radwaste Area	6.64E+00 µCi/cc
2RX-9830	Fuel Handling Area	8.92E+00 µCi/cc
2RX-9835	Emergency Penetration Room	1.77E+02 µCi/cc
2RX-9840	Post Accident Sampling Building	8.84E+01 µCi/cc
2RX-9845	Aux. Building Extension	2.53E+01 µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses > 100 mR TEDE or 500 mR child thyroid CDE at or beyond the site boundary.

OR

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AS1

3. Field survey results indicate closed window dose rates >100 mR/hr expected to continue for ≥60 minutes; or analyses of field survey samples indicate child thyroid CDE >500 mR for one hour of inhalation, at or beyond the site boundary.

Basis:

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

EAL #1

The monitor list in EAL #1 includes monitors on all potential release pathways (plant stack, primary-secondary leak, fuel handling accident).

EAL #2

Since dose assessment in EAL #2 is based on actual meteorology, whereas the monitor readings in EAL #1 are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.

EAL #3

Field team surveys in EAL #3 should be performed at or beyond the SITE BOUNDARY and at the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. The assumed release duration is one hour. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Sampling of radioiodine by adsorption on a charcoal cartridge should determine the iodine value.

Reference Documents:

1. 1604.051, “Eberline Radiation Monitor System”
2. Offsite Dose Calculation Manual

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AG1

Initiating Condition -- GENERAL EMERGENCY

Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity >1000 mR TEDE or 5000 mR child thyroid CDE for the actual or projected duration of the release using actual meteorology

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2 or 3)

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.*

1. VALID reading on Channel 9 on any of the following radiation monitors > the reading shown for ≥ 15 minutes:

MONITORS – UNIT 1		LIMIT
RX-9820	Containment Purge	1.18E+02 (µCi/cc)
RX-9825	Radwaste Area	1.07E+02 (µCi/cc)
RX-9830	Fuel Handling Area	9.08E+01 (µCi/cc)
RX-9835	Emergency Penetration Room	1.91E+03 (µCi/cc)
MONITORS – UNIT 2		**LIMIT**
2RX-9820	Containment Purge	8.92E+01 (µCi/cc)
2RX-9825	Radwaste Area	6.64E+01 (µCi/cc)
2RX-9830	Fuel Handling Area	8.92E+01 (µCi/cc)
2RX-9835	Emergency Penetration Room	1.77E+03 (µCi/cc)
2RX-9840	Post Accident Sampling Building	8.84E+02 (µCi/cc)
2RX-9845	Aux. Building Extension	2.53E+02 (µCi/cc)

OR

2. Dose assessment using actual meteorology indicates doses >1000 mR TEDE or 5000 mR child thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates >1000 mR/hr expected to continue for ≥60 minutes; or analyses of field survey samples indicate child thyroid CDE >5000 mR for one hour of inhalation, at or beyond the site boundary.

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS
AG1

Basis:

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

EAL #1

The monitor list in EAL #1 includes monitors on all potential release pathways (plant stack, primary-secondary leak, fuel handling accident).

EAL #2

Since dose assessment in EAL #2 is based on actual meteorology, whereas the monitor readings in EAL #1 are not, the results from these assessments may indicate that the classification is not warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.

EAL #3

Field team surveys in EAL #3 should be performed at or beyond the SITE BOUNDARY and at the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. The assumed release duration is one hour. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Sampling of radioiodine by adsorption on a charcoal cartridge should determine the iodine value.

Reference Documents:

1. 1604.051, “Eberline Radiation Monitor System”

2. Offsite Dose Calculation Manual

Cold Shutdown / Refueling System Malfunction

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU1

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

RCS leakage

Operating Mode Applicability: Cold Shutdown (Mode 5)

Example Emergency Action Level(s):

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. RCS leakage results in the inability to maintain or restore level within Pressurizer or RCS level target band for ≥15 minutes.

Basis:

This IC is considered to be a potential degradation of the level of safety of the plant. The inability to maintain or restore level is indicative of loss of RCS inventory.

Relief valve normal operation should be excluded from this IC. However, a relief valve that operates and fails to close per design should be considered applicable to this IC if the relief valve cannot be isolated.

Prolonged loss of RCS Inventory may result in escalation to the Alert emergency classification level via either CA1 or CA3.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU2

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

UNPLANNED loss of RCS / reactor vessel inventory

Operating Mode Applicability: Refueling (Mode 6)

Example Emergency Action Level(s): (1 or 2)

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. UNPLANNED RCS / reactor vessel level drop as indicated by either of the following:

   a. RCS / reactor vessel water level drop below the reactor vessel flange for ≥15 minutes when the RCS / reactor vessel level band is established above the reactor vessel flange

   OR

   b. RCS / reactor vessel water level drop below the RCS / reactor vessel level band for ≥15 minutes or longer when the RCS / reactor vessel level band is established below the reactor vessel flange.

OR

2. RCS / reactor vessel level cannot be monitored with a loss of RCS / reactor vessel inventory as indicated by an unexplained level rise in (as applicable) the Reactor Building Sump, Reactor Drain Tank, Aux. Building Equipment Drain Tank, Aux. Building Sump, or Quench Tank.

Basis:

This IC is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that lower RCS water level below the reactor vessel flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level dropping below the reactor vessel flange, or below the planned RCS water level for the given evolution (if the planned RCS water level is already below the reactor vessel flange), warrants declaration of an NUE due to the lowered RCS inventory that is available to keep the core covered.

The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists.

Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either CA1 or CA3.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU2

EAL #1

This EAL involves a drop in RCS level below the top of the reactor vessel flange that continues for 15 minutes due to an UNPLANNED event. This EAL is not applicable to drops in flooded reactor cavity level, which is addressed by AU2 EAL1, until such time as the level drops to the level of the vessel flange.

If reactor vessel level continues to drop and reaches the Bottom ID of the RCS Loop then escalation to CA1 would be appropriate.

EAL #2

This EAL addresses conditions in the refueling mode when normal means of core temperature indication and RCS level indication may not be available. Redundant means of reactor vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that reactor vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

Escalation to the Alert emergency classification level would be via either CA1 or CA3.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU3

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

UNPLANNED loss of decay heat removal capability with irradiated fuel in the reactor vessel

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s): (1 or 2)

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. UNPLANNED event results in RCS temperature exceeding 200 °F.

OR

2. Loss of all RCS temperature and RCS/reactor vessel level indication for ≥15 minutes.

Basis:

This IC is a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered.

During refueling the level in the reactor vessel will normally be maintained above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at lowered inventory may result in more rapid rises in RCS/reactor vessel temperatures depending on the time since shutdown.

Normal means of core temperature indication and RCS level indication may not be available in the refueling mode. Redundant means of reactor vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown of refueling modes, EAL 2 would result in declaration of an NUE if both temperature and level indication cannot be restored within 15 minutes from the loss of both means of indication.

Escalation to Alert would be via CA1 based on an inventory loss or CA3 based on exceeding its temperature criteria.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU5

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

AC power capability to Vital 4.16 KV busses reduced to a single power source ≥15 minutes such that any additional single power source failure would result in station blackout

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s):

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. a. AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes.

   AND

   b. Any additional single power source failure will result in station blackout.

Basis:

The condition indicated by this IC is the degradation of the offsite and onsite AC power systems such that any additional single power source failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of all but one emergency generator to supply power to its emergency busses. The subsequent loss of this single power source would escalate the event to an Alert in accordance with CA5.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

The EAL allows credit for operation of the Alternate AC Diesel Generator.

Reference Documents:

1. 1202.007, "*Degraded Powe*r"
2. 1202.008, "*Blackout*"
3. 2202.007, "*Loss of Off-Site Power*"
4. 2202.008, "*Station Blackout*"
5. 2104.037, "*Alternate AC Diesel Generator Operations*"

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU6

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Loss of required DC power ≥15 minutes

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s):

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. < 105 volts on required Vital DC bus ≥15 minutes.

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations.

It is intended that the loss of the operating (operable) train is to be considered. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will be per CA3.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU7

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Inadvertent criticality

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s):

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

Basis:

This IC addresses criticality events that occur in Cold Shutdown or Refueling modes such as fuel mis-loading events and inadvertent dilution events. This IC indicates a potential degradation of the level of safety of the plant, warranting an NUE classification.

This condition can be identified using the startup rate meter. The term “sustained” is used in order to allow exclusion of expected short term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

Escalation would be by SM judgment.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CU8

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Loss of all onsite or offsite communications capabilities

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)
Defueled

Example Emergency Action Level(s): (1 or 2)

1. Loss of all Table C2 onsite communication methods affecting the ability to perform routine operations.

   OR

2. Loss of all Table C3 offsite communication methods affecting the ability to perform offsite notifications.

Table C2 Onsite Communications Methods
Station radio system
Plant paging system
In-plant telephones
Gaitronics

Table C3 Offsite Communications Methods
All telephone lines (commercial and microwave)
ENS

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with offsite authorities. The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform federal, state, and local authorities of plant issues. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being utilized to make communications possible.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CA1

Initiating Condition - ALERT

Loss of RCS / reactor vessel inventory

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s): (1 or 2)

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. Loss of RCS / reactor vessel inventory as indicated by:

   Unit 1: RVLMS Levels 1 through 8 indicate DRY

   Unit 2: RVLMS Levels 1 through 5 indicate DRY

   OR

   Unit 1: Reactor vessel level < 368 ft., 0 in. (bottom of the hot leg)

   Unit 2: Reactor vessel level < 369 ft., 1.5 in. (bottom of the hot leg)

   OR

2. RCS / reactor vessel level cannot be monitored for ≥15 minutes with a loss of RCS / reactor vessel inventory as indicated by an unexplained level rise in (as applicable) the Reactor Building Sump, Reactor Drain Tank, Aux. Building Equipment Drain Tank, Aux. Building Sump, or Quench Tank.

Basis:

These EALs serve as precursors to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further reactor vessel level lowering and potential core uncovery. This condition will result in a minimum emergency classification level of an Alert.

EAL #1

The bottom of the RCS hot leg penetration into the reactor vessel is approximately RLVMS Level 8 (Unit 1) or RVLMS Level 5 (Unit 2). However, RVLMS may not be available in mode 6. Redundant means level indication is provided in this mode and included in EAL #1. The bottom of the RCS hot leg penetration into the reactor vessel is 368 ft., 0 in. (Unit 1) or 369 ft., 1.5 in. (Unit 2). Below this level, reactor vessel level indication will be lost and loss of suction to decay heat removal systems will occur. The inability to restore and maintain level after reaching this setpoint would be indicative of a failure of the RCS barrier.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CA1

EAL #2

In the cold shutdown mode, normal RCS level and reactor vessel level instrumentation systems will usually be available. In the refueling mode, normal means of reactor vessel level indication may not be available. Redundant means of reactor vessel level indication will usually be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that reactor vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

If reactor vessel level continues to lower then escalation to Site Area Emergency will be via CS1.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CA3

Initiating Condition - ALERT

Inability to maintain plant in Cold Shutdown

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s): (1 or 2)

1. An UNPLANNED event results in RCS temperature >200 °F > the specified duration in Table C1.

Table C1 **RCS Reheat Duration Thresholds**		
RCS	**Containment Closure**	**Duration**
Intact (but not RCS Lowered Inventory)	N/A	60 minutes*
Not intact or RCS Lowered Inventory	Established	20 minutes*
	Not Established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

OR

Note: *EAL #2 does not apply in solid plant conditions.*

2. An UNPLANNED event results in RCS pressure rise >10 psi due to a loss of RCS cooling.

Basis:

EAL #1

The RCS Reheat Duration Threshold table addresses complete loss of functions required for core cooling for greater than 60 minutes during refueling and cold shutdown modes when RCS integrity is established. RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CA3

The RCS Reheat Duration Threshold table also addresses the complete loss of functions required for core cooling for greater than 20 minutes during refueling and cold shutdown modes when CONTAINMENT CLOSURE is established but RCS integrity is not established or RCS inventory is lowered (e.g., mid-loop operation). As discussed above, RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible.

Finally, the EAL addresses complete loss of functions required for core cooling during refueling and cold shutdown modes when neither CONTAINMENT CLOSURE nor RCS integrity are established.

The (*) indicates that this EAL is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the specified time frame.

EAL #2

The 10 psi pressure rise addresses situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes. The RCS pressure setpoint chosen should be 10 psi or the lowest pressure that the site can read on installed Control Board instrumentation that is equal to or greater than 10 psi.

Escalation to Site Area Emergency would be via CS1 should boiling result in significant reactor vessel level loss leading to core uncovery.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available.

The SM / ED must remain alert to events or conditions that lead to the conclusion that exceeding the EAL is IMMINENT. If, in the judgment of the SM / ED, an IMMINENT situation is at hand, the classification should be made as if the threshold has been exceeded.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CA5

Initiating Condition - ALERT

Loss of all offsite and all onsite AC power to Vital 4.16KV busses ≥15 minutes

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)
Defueled

Example Emergency Action Level(s):

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. Loss of all offsite and all on-site AC power to Vital 4.16KV busses ≥ 15 minutes.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including DHR/shutdown cooling, emergency core cooling, containment cooling, spent fuel pool cooling and the ultimate heat sink.

The event can be classified as an Alert when in cold shutdown, refueling, or defueled mode because of the significantly reduced decay heat and lower temperature and pressure, which allow raising the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL.

Escalating to Site Area Emergency, if appropriate, is by Abnormal Radiation Levels/ Radiological Effluent (TAB A) ICs.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CS1

Initiating Condition - SITE AREA EMERGENCY

Loss of RCS / reactor vessel inventory affecting core decay heat removal capability

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s): (1 or 2)

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. With CONTAINMENT CLOSURE **not** established:

   Unit 1: RVLMS Levels 1 through 9 indicate DRY

   Unit 2: RVLMS Levels 1 through 6 indicate DRY

   OR

2. With CONTAINMENT CLOSURE established, core exit thermocouples indicate superheat.

   OR

3. RCS / reactor vessel level cannot be monitored for ≥ 30 minutes with a loss of RCS / reactor vessel inventory as indicated by any of the following:
   - Containment High Range Radiation Monitor reading > 10 R/hr
   - Erratic source range monitor indication
   - Unexplained level rise in Reactor Building Sump, Reactor Drain Tank, Quench Tank, Aux. Building Equipment Drain Tank, or Aux. Building Sump.

Basis:

Under the conditions specified by this IC, continued lowering in RCS / reactor vessel level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via CG1 or AG1.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CS1

EAL #3

In the cold shutdown mode, normal RCS level and reactor vessel level instrumentation systems will usually be available. In the refueling mode, normal means of reactor vessel level indication may not be available. Redundant means of reactor vessel level indication will usually be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that reactor vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

The 30-minute duration allows sufficient time for actions to be performed to recover inventory control equipment.

As water level in the reactor vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in site specific monitor indication and possible alarm.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CG1

Initiating Condition - GENERAL EMERGENCY

Loss of RCS / reactor vessel inventory affecting fuel clad integrity with containment challenged

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Example Emergency Action Level(s):

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. a. Core exit thermocouples indicate superheat for ≥ 30 minutes.

   AND

   b. Any of the following containment challenge indications:
   - CONTAINMENT CLOSURE not established
   - Explosive mixture inside containment
   - UNPLANNED rise in containment pressure

   OR

2. a. RCS / reactor vessel level cannot be monitored for ≥ 30 minutes with a loss of RCS / reactor vessel inventory as indicated by any of the following:
   - Containment High Range Radiation Monitor reading > 10 R/hr
   - Erratic source range monitor indication
   - Unexplained level rise in Reactor Building Sump, Reactor Drain Tank, Quench Tank, Aux. Building Equipment Drain Tank, or Aux. Building Sump

   AND

   b. Any of the following containment challenge indications:
   - CONTAINMENT CLOSURE not established
   - Explosive mixture inside containment
   - UNPLANNED rise in containment pressure

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION
CG1

Basis:

This IC represents the inability to restore and maintain reactor vessel level to above the top of active fuel with containment challenged. Fuel damage is probable if reactor vessel level cannot be restored, as available decay heat will cause boiling, further reducing the reactor vessel level. With the CONTAINMENT breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE. The GE is declared on the occurrence of the loss or IMMINENT loss of function of all three barriers.

A number of variables can have a significant impact on heat removal capability challenging the fuel clad barrier. Examples include mid-loop, reduced level / flange level, head in place, cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, and steam generator U-tube draining.

Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, 30 minutes was conservatively chosen.

If CONTAINMENT CLOSURE is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to GE would not occur.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gasses in Containment. However, Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists.

Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

In the cold shutdown mode, normal RCS level and reactor vessel level instrumentation systems will usually be available. In the refueling mode, normal means of reactor vessel level indication may not be available. Redundant means of reactor vessel level indication will usually be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that reactor vessel inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

As water level in the reactor vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in site specific monitor indication and possible alarm.

<u>Reference Documents</u>:

1. ULD-1-SYS-24, *“Unit 1 Inadequate Core Cooling”*
2. ULD-2-SYS-24, *“Unit 2 Inadequate Core Cooling”*

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) MALFUNCTION

ISFSI MALFUNCTION
E-HU1

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Note: *Security Events are bounded by the Hazards EALs.*

Damage to a loaded cask CONFINEMENT BOUNDARY

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. Damage to a loaded cask CONFINEMENT BOUNDARY.

Basis:

An NUE in this IC is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes classification based on a loaded fuel storage cask CONFINEMENT BOUNDARY loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

This EAL addresses a dropped cask, a tipped over cask, EXPLOSION, PROJECTILE damage, FIRE damage or natural phenomena affecting a cask (e.g., seismic event, tornado, etc.).

FISSION PRODUCT BARRIER DEGRADATION

General Bases

The logic used for these initiating conditions reflects the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier (See Sections 3.4 and 3.8). NUE ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction (S) ICs.

- At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier “Loss” EALs existed, that, in addition to off-site dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier “Potential Loss” EALs existed, the SM / ED would have more assurance that there was no immediate need to escalate to a General Emergency.

- The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.

- The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment Barrier status is addressed by Technical Specifications.

FISSION PRODUCT BARRIER
FUEL CLAD

Fuel Clad Barrier Emergency Action Levels: FCB1 **OR** FCB2 **OR** FCB3 **OR** FCB4 **OR** FCB5 **OR** FCB6

The Fuel Clad barrier consists of the zircalloy or stainless steel fuel bundle tubes that contain the fuel pellets.

1. Primary Coolant Activity Level (FCB1)

Loss:

1. Coolant activity > 300 µCi/gm dose equivalent I-131 activity by Chemistry sample

OR

2. Radiation levels > 1000 MR/hr

   Unit 1: at SA-229

   Unit 2: at 2TCD-19

Potential Loss: None

Basis:

Loss

The site specific value corresponds to 300 µCi/gm I-131 equivalent. Assessment by the EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

A reading of greater than 1000 mR/hr within at one foot from the RCS sample lines (SA-229 for Unit 1, 2TCD-19 for Unit 2) has been determined to correspond to fuel clad failure of approximately 2-5%, and thus the fuel clad barrier is considered lost. This reading is well above that expected for iodine spikes and thus indicates significant clad damage and thus the fuel clad barrier is considered lost.

Potential Loss

There is no Potential Loss EAL associated with this item.

Reference Documents

1. ANO Calculation 03-E-0002-01, *"Radiation Monitor EAL Setpoints for Fission Product Barrier Degradation"*

FISSION PRODUCT BARRIER
FUEL CLAD

2. Core Exit Thermocouple Readings (FCB2)

Loss: > 1200 °F CET temperature.

Potential Loss:

Unit 1: ICC exists as evidenced by CETs indicating superheated conditions

Unit 2: Average CETs indicate superheat for current RCS pressure

Basis:

Loss

The Loss EAL of > 1200 °F is consistent with NEI 99-01 and corresponds to significant superheating of the coolant.

Potential Loss

The Potential Loss EAL corresponds to a loss of subcooling margin.

Note that the loss or potential loss EAL for this category will occur after a loss of adequate sub-cooling margin, which represents a loss of the RCS barrier in EAL RCB1, and therefore represents the loss of two barriers, resulting in a Site Area Emergency per FS1. Any loss or potential loss of the containment barrier at that point would escalate to a General Emergency.

Reference Documents

1. Unit 1 EOP 1202.005, “Inadequate Core Cooling”
2. Unit 1 EOP 1202.013, “EOP Figures”
3. Unit 2 OP 2202.009, “Functional Recovery”
4. ANO Procedure OP 1302.022, “Core Damage Assessment”
5. CE-NPSD-241, “Development of the Comprehensive Procedure Guideline for Core Damage Assessment,” Task 467
6. BWOG EOP Technical Bases Document, Vol. 3, Chapter III.F

FISSION PRODUCT BARRIER
FUEL CLAD

3. Reactor Vessel Water Level (FCB3)

Loss: None

Potential Loss:

Unit 1: RVLMS Levels 1 through 9 indicate DRY

Unit 2: RVLMS Levels 1 through 7 indicate DRY

Basis:

Loss

There is no Loss EAL associated with this item.

Potential Loss

The Reactor Vessel Level Monitoring Systems at ANO do not provide positive indication of core uncovery. The above core level indication provided is used to monitor the approach to and recovery from ICC conditions, but the CETs are used to identify core uncovery, and are the only positive indication of core uncovery.

Per reference document #1, the reactor vessel level indicators installed in Unit 1 extend from the top of the reactor vessel to the fuel alignment plate, and information in reference document #2 indicates that the lowest sensor is greater than 2 feet above the top of active fuel. If any of the 4 RCPs are running, flow induced turbulence produced by the pumps renders the reactor vessel level indicator readings invalid.

Per reference document #3, only the reactor vessel level indicators above the core are considered part of the ICC monitoring system. Per reference document #4, the lowest sensor above the core, RVLMS LVL 6 on the ICC monitoring panel 2C388, is 47 inches above the top of the core. If any of the 4 RCPs are running, flow induced turbulence produced by the pumps renders the reactor vessel level indicator readings invalid.

For either unit then, should CET indication be unavailable and reactor vessel level indication be unavailable due to RCP operation or any other cause, a degraded ability to monitor the barrier would exist.

Reference Documents:

1. ULD-1-SYS-24, *"Unit 1 Inadequate Core Cooling System"*
2. Calculation 84-EQ-0080-02, *"Loop Error Analysis for Reactor Vessel Level Monitoring System"*
3. ULD-2-SYS-24, *"Unit 2 Inadequate Core Cooling Monitoring System"*
4. Calculation 90-E-0116-01, *"Unit 2 EOP Setpoint Document,"* Setpoint R.3

FISSION PRODUCT BARRIER
FUEL CLAD

4. Containment Radiation Monitoring (FCB4)

Loss: Containment high range radiation monitor reading > 1000 R/hr

Potential Loss: None

Basis:

Loss

The 1000 R/hr reading on the containment high range radiation monitors (RE-8060 or RE-8061 for Unit 1, 2RE-8925-1 or 2RE-8925-2 for Unit 2) is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment.

Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage.

This radiation monitor value is higher than that specified for RCS barrier Loss EAL RCB3. Thus, this EAL indicates a loss of both the Fuel Clad barrier and RCS barrier that appropriately escalates the emergency classification to a Site Area Emergency per FS1.

Potential Loss

There is no Potential Loss EAL associated with this item.

Reference Documents:

1. NUREG 1228, *"Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents"*

2. ANO Calculation 03-E-0002-01, *"Radiation Monitor EAL Setpoints for Fission Product Barrier Degradation"*

FISSION PRODUCT BARRIER
FUEL CLAD

5. Core Damage Assessment (FCB5)

Loss: At least 5% fuel clad damage as determined from core damage assessment

Potential Loss: None

Basis:

Loss

This level is consistent with other fuel clad barrier loss EALs indicative of significant fuel clad damage, but uses core damage assessment evaluations by Technical Support personnel. The fuel clad barrier is considered lost.

If this determination is made from the high range containment radiation monitor readings, or if accompanied by other indications of a loss or potential loss of the RCS barrier, this EAL condition represents a Site Area Emergency per FS1.

Potential Loss

There is no potential loss EAL associated with this item.

Reference Documents:

1. ANO Procedure OP-1302.022, *"Core Damage Assessment"*

6. Emergency Director Judgment (FCB6)

Any condition in the opinion of the SM / ED that indicates Loss or Potential Loss of the Fuel Clad barrier.

Basis:

This EAL addresses any other factors that are to be used by the SM / ED in determining whether the Fuel Clad barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in SM / ED judgment that the barrier may be considered lost or potentially lost.

FISSION PRODUCT BARRIER
RCS

RCS Barrier EALs: RCB1 **OR** RCB2 **OR** RCB3 **OR** RCB4

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

1. RCS Leak Rate (RCB1)

Loss: RCS leak rate > available makeup capacity as indicated by:

Unit 1: Loss of adequate subcooling margin

Unit 2: RCS subcooling (MTS) can NOT be maintained at least 30 °F

Potential Loss:

Unit 1: UNISOLABLE RCS leak > 50 gpm with Letdown isolated

Unit 2: UNISOLABLE RCS leak > 44 gpm with Letdown isolated

Basis:

Loss

This EAL addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak.

Potential Loss

This EAL is based on the apparent inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Makeup and Purification System (Unit 1) or the Chemical and Volume Control System (Unit 2).

Isolating letdown is a standard abnormal operating procedure action and may prevent unnecessary classifications when a non-RCS leakage path such as a Makeup and Purification System or CVCS leak exists. The intent of this condition is met if attempts to isolate Letdown are NOT successful. Additional charging pumps being required is indicative of a substantial RCS leak.

Reference Documents:

1. Unit 1 EOP 1202.013, Figure 1, *"Saturation and Adequate SCM"*
2. Unit 1 EOP Setpoint Document, Calculation 90-E-0116-07, Setpoint B.19
3. Unit 2 EOP 2202.009, *"Functional Recovery"*
4. Unit 2 EOP Setpoint Document, Calculation 90-E-0116-01
5. Unit 2 SAR Table 9.3-14, Charging Pumps Design Data

FISSION PRODUCT BARRIER
RCS

2. SG Tube Rupture (RCB2)

Loss: SGTR that results in an ECCS (SI) actuation

Potential Loss: None

Basis:

Loss

This EAL addresses the full spectrum of Steam Generator (SG) tube rupture events in conjunction with Containment barrier Loss EALs. It addresses RUPTURED SG(s) for which the leakage is large enough to cause actuation (either automatic or manual) of ECCS (SI). This is consistent to the RCS leak rate barrier Potential Loss EAL.

By itself, this EAL will result in the declaration of an Alert. However, if the SG is also FAULTED (i.e., two barriers failed), the declaration escalates to a Site Area Emergency per Containment barrier Loss EAL CNB3.

Potential Loss

There is no Potential Loss EAL associated with this item.

3. Containment Radiation Monitoring (RCB3)

Loss: Containment high range radiation monitor reading > 100 R/hr.

Potential Loss: None

Basis

Loss

The 100 R/hr reading on the containment high range radiation monitors (RE-8060 or RE-8061 for Unit 1, 2RE-8925-1 or 2RE-8925-2 for Unit 2) is a value which indicates the release of reactor coolant to the containment.

This reading is less than that specified for Fuel Clad barrier EAL FCB4. Thus, this EAL is indicative of a RCS leak only. If the radiation monitor reading rose to that specified by Fuel Clad barrier EAL, fuel damage would also be indicated.

During the initial fifteen minutes after a thermal event inside containment, the high range radiation monitor readings are considered invalid due to possibility of a transient thermally-induced current.

FISSION PRODUCT BARRIER
RCS

Potential Loss

There is no Potential Loss EAL associated with this item.

Reference Documents:

1. ANO Calculation 03-E-0002-01**, "***Radiation Monitor EAL Setpoints for Fission Product Barrier Degradation"*

4. Emergency Director Judgment (RCB4)

Any condition in the opinion of the SM / ED that indicates Loss or Potential Loss of the RCS Barrier.

Basis:

This EAL addresses any other factors that are to be used by the SM / ED in determining whether the RCS barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in SM / ED judgment that the barrier may be considered lost or potentially lost.

FISSION PRODUCT BARRIER
CONTAINMENT

Containment Barrier EALs: CNB1 **OR** CNB2 **OR** CNB3 **OR** CNB4 **OR** CNB5 **OR** CNB6 **OR** CNB7

The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve.

1. Containment Pressure (CNB1)

Loss:

1. Rapid unexplained drop in containment pressure following an initial rise in containment pressure

   OR

2. Containment pressure or sump level response not consistent with LOCA conditions

Potential Loss:

1. **Unit 1:** Containment pressure > 73.7 PSIA (59 PSIG) and rising

   Unit 2: Containment pressure > 73.7 PSIA (59 PSIG) and rising

   OR

2. Explosive mixture exists inside containment.

   OR

3. a. Containment Pressure > containment spray actuation setpoint

      UNIT 1: 44.7 PSIA (30 PSIG)

      UNIT 2: 23.3 PSIA (8.6 PSIG)

      AND

   b. LESS THAN one full train of spray operating

FISSION PRODUCT BARRIER
CONTAINMENT

Basis:

Loss

Rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure rise from a primary or secondary high energy line break indicates a loss of containment integrity. Containment pressure and sump levels should rise as a result of mass and energy release into containment from a LOCA. Thus, sump level or pressure not rising indicates containment bypass and a loss of containment integrity.

This indicator relies on operator recognition of an unexpected response for the condition and therefore, does not have a specific value associated with it. The unexpected response is important because it is the indicator for a containment bypass condition.

Potential Loss 1.

The site specific pressure is based on the containment design pressure.

Potential Loss 2.

Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. The hydrogen concentration of 4% has been recognized by the NRC staff as a well-established lower flammability limit in air or steam-air atmospheres that is adequately conservative for protecting against an H2 explosion. Hydrogen control systems at ANO are designed and operated as to maintain the containment hydrogen concentration below this level, so that indications of hydrogen concentrations above this are considered a potential challenge to the containment integrity.

Potential Loss 3.

This EAL represents a potential loss of containment in that the containment heat removal/depressurization system (e.g., containment sprays, ice condenser fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner, as indicated by containment pressure greater than the setpoint at which the equipment was supposed to have actuated.

Reference Documents:

1. Unit 1 OP-1105.003, *“Engineering Safeguards Actuation System”*
2. Unit 1 SAR Sections 1.4.43, 5.2.1.2.1, 14.2.2.5.5.1 (reactor building design pressure)
3. Unit 1 SAR Section 6.6 (Post-Loss of Coolant Accident Hydrogen Control)
4. Unit 1 TS Table 3.3.5-1
5. Unit 2 SAR Section 6.2.5 (Combustible Gas Control In Containment)
6. Unit 2 SAR Section 3.8.1.3.1.D (Containment Design Pressure)
7. Unit 2 TS Table 3.3-4
8. Regulatory Guide 1.7, *“Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident, Rev. 2 1978”*

FISSION PRODUCT BARRIER
CONTAINMENT

2. Core Exit Thermocouple Readings (CNB2)

Loss: None

Potential Loss:

1. a. CETs indicate > 1200 °F

   AND

   b. Restoration procedures not effective within 15 minutes.

OR

2. a. CETs indicate > 700 °F

   AND

   b. RVLMS indicates:

   Unit 1: Levels 1 through 9 DRY

   Unit 2: Levels 1 through 7 DRY

   AND

   c. Restoration procedures not effective within 15 minutes.

Basis:

Loss

There is no Loss EAL associated with this item.

Potential Loss

The conditions in these EALs represent an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and a higher potential for containment failure. In conjunction with the Core Cooling and RCS Leakage criteria in the Fuel and RCS barrier columns, this threshold would result in the declaration of a General Emergency, i.e., loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path.

The function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is dropping or if the vessel water level is rising.

FISSION PRODUCT BARRIER
CONTAINMENT

Whether or not the procedures will be effective should be apparent within 15 minutes. The SM / ED should make the declaration as soon as it is determined that the procedures have been, or will be ineffective.

3. SG Secondary Side Release With Primary-to-Secondary Leakage (CNB3)

Loss:

1. RUPTURED steam generator is also FAULTED outside of containment

   OR

2. a. Primary-to-secondary leakrate > 10 gpm

      AND

   b. UNISOLABLE steam release from affected steam generator to the environment

Potential Loss: None

Basis:

This loss EAL recognizes that SG tube leakage can represent a bypass of the containment barrier as well as a loss of the RCS barrier.

This EAL results in a NUE for smaller breaks that; (1) do not exceed the Normal Makeup Capacity for Unit 1 or the capacity of one charging pump in the normal charging lineup for Unit 2 EAL in RCS leak rate barrier Potential Loss , or (2) do not result in ECCS actuation in RCS SG tube rupture barrier Loss. For larger breaks, RCS barrier threshold criteria would result in an Alert. For SG tube ruptures which may involve multiple steam generators or UNISOLABLE secondary line breaks, this condition would exist in conjunction with RCS barrier conditions and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

Loss 1.

This EAL addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the RCS and containment barriers and is a subset of the second threshold. In conjunction with RCS leak rate barrier loss EAL RCB2, this would always result in the declaration of a Site Area Emergency.

FISSION PRODUCT BARRIER
CONTAINMENT

Loss 2.

This EAL addresses SG tube leaks that exceed 10 gpm in conjunction with an UNISOLABLE release path to the environment from the affected steam generator. The threshold for establishing the UNISOLABLE secondary side release is intended to be a prolonged release of radioactivity from the RUPTURED steam generator directly to the environment. This could be expected to occur when the main condenser is unavailable to accept the contaminated steam (i.e., SG tube rupture with concurrent loss of off-site power and the RUPTURED steam generator is required for plant cooldown or a stuck open relief valve). The time it takes to isolate a SG with tube leakage > 10 gpm in accordance with plant specific EOPs is not considered a prolonged release. In this case the SG with tube leakage > 10 gpm with a concurrent loss of offsite power is normally steamed to the environment in a controlled manner to achieve and maintain a RCS Hot Leg temperature below that which corresponds to the Main Steam Safety Valve relief settings. However, if the SG cannot be isolated or if both SGs have tube leakage > 10 gpm, a prolonged release will likely be necessary to support plant cooldown. If the main condenser is available, there may be releases via air ejectors, gland seal exhausters, and other similar controlled, and often monitored, pathways. These pathways do not meet the intent of an UNISOLABLE release path to the environment. These minor releases are assessed using Abnormal Radiation Levels / Radiological Effluent ICs (TAB A).

Potential Loss

There is no Potential Loss EAL associated with this item.

4. Containment Isolation Failure or Bypass (CNB4)

Loss:

1. UNISOLABLE breach of containment

   AND

2. Direct downstream pathway to the environment exists after containment isolation signal

Potential Loss: None

FISSION PRODUCT BARRIER
CONTAINMENT

Basis:

Loss

This EAL addresses incomplete containment isolation that allows a direct release to the environment. A breach of containment has also occurred if an inboard and outboard pair of isolation valves fails to close on an automatic actuation signal or from a manual action in the Control Room and opens a release path to the environment.

The breach is not isolable from the Control Room if an attempt for isolation from the Control Room has been made and was unsuccessful. An attempt for isolation should be made prior to the accident classification. If isolable upon identification, then this Initiating Condition is not applicable.

The use of the modifier “direct” in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in-line charcoal filter does not make a release path indirect since the filter is not effective at removing fission product noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur.

In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

Potential Loss

There is no Potential Loss EAL associated with this item.

FISSION PRODUCT BARRIER
CONTAINMENT

5. Containment Radiation Monitoring (CNB5)

Loss: None

Potential Loss:

Containment high range radiation monitor reading > 4000 R/hr

Basis:

Loss

There is no Loss EAL associated with this item.

Potential Loss

The 4000 R/hr reading on the containment high range radiation monitors (RE-8060 or RE-8061 for Unit 1, 2RE-8925-1 or 2RE-8925-2 for Unit 2) is a value which indicates significant fuel damage well in excess of the EALs associated with both loss of Fuel Clad and loss of RCS barriers. A major release of radioactivity requiring off-site protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.

Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted.

Because the monitor reading exceeds the readings for Fuel Clad Barrier loss in **FCB4** and RCS Barrier loss in **RCB3**, the SM/ED should declare a General Emergency when this value on the Containment High Range Rad Monitor is exceeded as a loss of two barriers (fuel clad and RCS) and potential loss of the third (containment).

Reference Documents:

1. ANO Calculation 03-E-0002-01, *"Radiation Monitor EAL Setpoints for Fission Product Barrier Degradation"*
2. NUREG 1228, *"Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents"*

FISSION PRODUCT BARRIER
CONTAINMENT

6. Other Indications (CNB6)

Elevated readings on the following radiation monitors that indicate loss or potential loss of the Containment barrier:

MONITORS – UNIT 1	
RX-9820	Containment Purge
RX-9825	Radwaste Area
RX-9830	Fuel Handling Area
RX-9835	Emergency Penetration Room
MONITORS – UNIT 2	
2RX-9820	Containment Purge
2RX-9825	Radwaste Area
2RX-9830	Fuel Handling Area
2RX-9835	Emergency Penetration Room
2RX-9840	Post Accident Sampling Building
2RX-9845	Aux. Building Extension

Basis:

This EAL covers other indications that may unambiguously indicate the loss or potential loss of the containment barrier.

7. Emergency Director Judgment (CNB7)

Any condition in the opinion of the SM / ED that indicates Loss or Potential Loss of the Containment Barrier.

Basis:

This EAL addresses any other factors that are to be used by the SM / ED in determining whether the Containment barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in SM / ED judgment that the barrier may be considered lost or potentially lost.

The Containment barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment barrier status is addressed by Technical Specifications

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU1

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2 or 3)

1. A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by ANO Security Shift Supervision.

   OR

2. A credible site specific security threat notification.

   OR

3. A validated notification from NRC providing information of an aircraft threat.

Basis:

NOTE: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under HA1, HS1 and HG1.

A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. Consideration shall be given to upgrading the emergency response status and emergency classification in accordance with the Safeguards Contingency Plan and Emergency Plan.

EAL #1

The Security Shift Supervisor is the designated individual on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Safeguards Contingency Plan.

This EAL is based on the Safeguards Contingency Plan. The Safeguards Contingency Plan is based on guidance provided in NEI 03-12.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU1

EAL #2

This EAL is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the NUE.

The determination of “credible” is made through use of information found in the Safeguards Contingency Plan.

EAL #3

The intent of this EAL is to ensure that notifications for the aircraft threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

This EAL is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the NUE.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert via HA1 would be appropriate if the threat involves an airliner within 30 minutes of the plant.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU2

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Other conditions exist which in the judgment of the SM warrant declaration of an NUE

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. Other conditions exist which in the judgment of the SM indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the SM to fall under the NUE emergency classification level.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU4

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection or EXPLOSION within the PROTECTED AREA

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2)

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the duration has exceeded, or will likely exceed, the applicable time.*

1. FIRE in any **Table H1** structure or area not extinguished:

   1) within 15 minutes of Control Room notification

   OR

   2) within 15 minutes of verification of a Control Room FIRE alarm (i.e. Alarm valid until disproved).

OR

2. EXPLOSION within the PROTECTED AREA.

Basis:

This IC addresses the magnitude and extent of FIRES or EXPLOSIONS that may be potentially significant precursors of damage to safety systems. It addresses the FIRE / EXPLOSION, and not the degradation in performance of affected systems that may result.

As used here, detection is visual observation and report by plant personnel or sensor alarm indication.

EAL #1

The 15-minute time period begins with a credible notification that a FIRE is occurring or indication of a fire detection system alarm/actuation. Verification of a fire detection system alarm/actuation includes actions that can be taken within the Control Room or other nearby site specific location to ensure that it is not spurious. An alarm is assumed to be an indication of a FIRE unless it is disproved within the 15-minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm.

The intent of this 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket).

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU4

EAL #2

This EAL addresses only those EXPLOSIONS of sufficient force to damage permanent structures or equipment within the PROTECTED AREA.

No attempt is made to assess the actual magnitude of the damage. The occurrence of the EXPLOSION is sufficient for declaration.

The SM also needs to consider any security aspects of the EXPLOSION, if applicable.

Escalation of this emergency classification level, if appropriate, would be based on HA4.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU4

Table H1
Unit 1

Reactor Building

All Elevations

Aux Building

All Elevations Including Penthouse/MSIV Room
Exceptions: Boric Acid Mix Tank Room (Chem Add Area) 404’ (157-B)
EDG Exhaust Fan area on 386’ (1-E and 2-E)

Turbine Building

All Elevations
Including:
Pipechase under ICW Coolers
CRD Pump Pit / T-28 Room / Area under ICW Pumps

Outside Areas

Manholes adjacent to Startup #2 XFMR (MH-03/MH-04)
Manholes adjacent to Intake Structure (MH-05/MH-06)
Intake Structure (354’ and 366’)
Diesel Fuel Vault
Diesel Fuel Vault Pump Manholes MH-09 and MH-10 (Manhole, MH-09, is located approximately 15 feet northeast of the Unit 1 QCST , Manhole, MH-10, is located approximately 5 feet west of Unit 2 Condensate Storage Tank, 2T-41A)

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU4

Table H1
Unit 2

Reactor Building

All Elevations

Aux Building

All Elevations including Aux Extensions

Turbine Building

All Elevations

Outside Areas

Intake Structure (354’ and 366’)
Concrete Manhole East, NE of intake
Concrete Manhole East of Turbine building next to train bay
Diesel Fuel Vault
Diesel Fuel Vault Pump Manholes MH-09 and MH-10 (Manhole, MH-09, is located approximately 15 feet northeast of the Unit 1 QCST , Manhole, MH-10, is located approximately 5 feet west of Unit 2 Condensate Storage Tank, 2T-41A)

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU5

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Release of toxic, corrosive, asphyxiant, or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS.

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2)

1. Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS.

   OR

2. Report by Local, County or State officials for evacuation or sheltering of site personnel based on an offsite event.

Basis:

This IC is based on the release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect NORMAL PLANT OPERATIONS.

The fact that SCBAs may be worn does not eliminate the need to declare the event.

This IC is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

Escalation of this emergency classification level, if appropriate, would be based on HA5.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU6

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Natural or destructive phenomena affecting the PROTECTED AREA

Operating Mode Applicability: All

Example Emergency Action Level: (1 or 2 or 3 or 4 or 5 or 6)

1. Seismic event identified by any 2 of the following:
   - Seismic event confirmed by annunciation of the 0.01g acceleration alarm
   - Earthquake felt in plant
   - National Earthquake Center

   OR

2. Tornado striking within PROTECTED AREA boundary or high winds > 67 mph (**2 minute average**).

   OR

3. Internal flooding that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode in any of the structures or areas in **Table H1 (see Table H1 located in HU4).**

   OR

4. Turbine failure resulting in casing penetration or damage to turbine or generator seals.

   OR

5. Lake Dardanelle level < 335 feet.

   OR

6. Lake Dardanelle level > 345 feet.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU6

Basis:

These EALs are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

EAL #1

Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate.

As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a “felt earthquake” is *an earthquake of sufficient intensity such that the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time.*

The National Earthquake Center can confirm if an earthquake has occurred in the area of the plant.

EAL #2

This EAL is based on a tornado striking (touching down) or high winds within the PROTECTED AREA.

The high wind value in EAL #2 is conservatively based on the SAR design basis for Unit 1 of 67 mph. Unit 2 Design basis is 80 mph.

Escalation of this emergency classification level, if appropriate, would be based on VISIBLE DAMAGE, or by other in plant conditions, via HA6.

EAL #3

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps.

Escalation of this emergency classification level, if appropriate, would be via HA6, or by other plant conditions.

EAL #4

This EAL addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant.

Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIRES and flammable gas build up are appropriately classified via HU4 and HU5.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HU6

This EAL is consistent with the definition of an NUE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.

Escalation of this emergency classification level, if appropriate, would be to HA6 based on damage done by PROJECTILES generated by the failure or in conjunction with a steam generator tube rupture. These latter events would be classified by the radiological (A) ICs or Fission Product Barrier (F) ICs.

EALs #5 and #6

EALs #5 and #6 are based on the levels of Lake Dardanelle at which the site will take specific action to reduce the impact of the lake level on plant safety by initiating plant shutdown.

Reference Documents:

1. OP-1203.025, *"Natural Emergencies"*
2. OP-2203.008, *"Natural Emergencies"*
3. Unit 1 FSAR
4. Unit 2 FSAR

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA1

Initiating Condition - ALERT

HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2)

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by ANO Security Shift Supervision.

   OR

2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.

Basis:

NOTE: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

These EALs address the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).

EAL #1

This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the OWNER CONTROLLED AREA. Those events are adequately addressed by other EALs.

Note that this EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes Independent Spent Fuel Storage Installations that may be outside the PROTECTED AREA but still in the OWNER CONTROLLED AREA.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA1

EAL #2

This EAL addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.

The intent of this EAL is to ensure that notifications for the airliner attack threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

This EAL is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA2

Initiating Condition - ALERT

Other conditions exist which in the judgment of the SM / ED warrant declaration of an Alert

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. Other conditions exist which in the judgment of the SM / ED indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere, but that warrant declaration of an emergency because conditions exist which are believed by the SM / ED to fall under the Alert emergency classification level.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA3

Initiating Condition - ALERT

Control room evacuation has been initiated

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. Alternate Shutdown procedure requires Control Room evacuation:

   Unit 1: 1203.002, "Alternate Shutdown"

   Unit 2: 2203.014, "Alternate Shutdown"

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facilities may be necessary.

Inability to establish plant control from outside the Control Room will escalate this event to a Site Area Emergency.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA4

Initiating Condition - ALERT

FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. FIRE or EXPLOSION resulting in VISIBLE DAMAGE to any Table H1 structure or area containing safety systems or components or Control Room indication of degraded performance of those safety systems.

Basis:

VISIBLE DAMAGE is used to identify the magnitude of the FIRE or EXPLOSION and to discriminate against minor FIRES and EXPLOSIONS.

The reference to structures or areas containing safety systems or components is included to discriminate against FIRES or EXPLOSIONS in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the FIRE or EXPLOSION was large enough to cause damage to these systems.

The use of VISIBLE DAMAGE should not be interpreted as mandating a lengthy damage assessment prior to classification. The declaration of an Alert and the activation of the Technical Support Center will provide the SM/ED with the resources needed to perform detailed damage assessments.

The SM / ED also needs to consider any security aspects of the EXPLOSION.

Escalation of this emergency classification level, if appropriate, will be based on System Malfunction (S), Fission Product Barrier Degradation (F) or Abnormal Radiation Levels / Radiological Effluent (A) ICs.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA4

Table H1
Unit 1

Reactor Building

All Elevations

Aux Building

All Elevations Including Penthouse/MSIV Room
Exceptions: Boric Acid Mix Tank Room (Chem Add Area) 404’ (157-B)
EDG Exhaust Fan area on 386’ (1-E and 2-E)

Turbine Building

All Elevations
Including:
Pipechase under ICW Coolers
CRD Pump Pit / T-28 Room / Area under ICW Pumps

Outside Areas

Manholes adjacent to Startup #2 XFMR (MH-03/MH-04)
Manholes adjacent to Intake Structure (MH-05/MH-06)
Intake Structure (354’ and 366’)
Diesel Fuel Vault
Diesel Fuel Vault Pump Manholes MH-09 and MH-10 (Manhole, MH-09, is located approximately 15 feet northeast of the Unit 1 QCST , Manhole, MH-10, is located approximately 5 feet west of Unit 2 Condensate Storage Tank, 2T-41A)

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA4

Table H1
Unit 2

Reactor Building

All Elevations

Aux Building

All Elevations including Aux Extensions

Turbine Building

All Elevations

Outside Areas

Intake Structure (354’ and 366’)
Concrete Manhole East, NE of intake
Concrete Manhole East of Turbine building next to train bay
Diesel Fuel Vault
Diesel Fuel Vault Pump Manholes MH-09 and MH-10 (Manhole, MH-09, is located approximately 15 feet northeast of the Unit 1 QCST , Manhole, MH-10, is located approximately 5 feet west of Unit 2 Condensate Storage Tank, 2T-41A)

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA5

Initiating Condition - ALERT

Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor.

Unit 1

VITAL AREA	APPLICABLE MODES
A-4 Switchgear Room	3, 4
Upper North Electrical Penetration Room	3, 4
Lower South Electrical Equipment Room	3, 4
Control Room	ALL

Unit 2

VITAL AREA	APPLICABLE MODES
Auxiliary Building 317’ Emergency Core Cooling Rooms	3, 4
Auxiliary Building 317’ Tendon Gallery Access	3, 4
Auxiliary Building 335’ Charging Pumps / 2B-52	3, 4
Auxiliary Building 354’ 2B-62 Area	3, 4
Emergency Diesel Generator Corridor	3, 4
Lower South Piping Penetration Room	3, 4
Auxiliary Building 386’ Containment Hatch	3, 4
Control Room	ALL

Operating Mode Applicability: As stated in above tables.

Example Emergency Action Level(s):

Note: *If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.*

1. Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA5

Basis:

Gases in a VITAL AREA can affect the ability to safely operate or safely shutdown the reactor. The fact that SCBAs may be worn does not eliminate the need to declare the event.

Declaration should not be delayed for confirmation from atmospheric testing if the atmosphere poses an immediate threat to life and health or an immediate threat of severe exposure to gases. This could be based upon documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards.

If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL assumes concentrations of flammable gasses which can ignite/support combustion.

Escalation of this emergency classification level, if appropriate, will be based on System Malfunction (S), Fission Product Barrier Degradation (F) or Abnormal Radiation Levels / Radioactive Effluent (A) ICs.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA6

Initiating Condition - ALERT

Natural or destructive phenomena affecting VITAL AREAS

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2 or 3 or 4 or 5 or 6)

1. a. Seismic event > Operating Basis Earthquake (OBE) as indicated by annunciation of the 0.1g acceleration alarm.

   AND

   b. Earthquake confirmed by any of the following:

   - Earthquake felt in plant
   - National Earthquake Center
   - Control Room indication of degraded performance of systems required for the safe shutdown of the plant

OR

2. Tornado striking or high winds > 67 mph (2 minute average) resulting in VISIBLE DAMAGE to any of the following structures/equipment containing safety systems or components or Control Room indication of degraded performance of those safety systems:

Reactor Building	Turbine Building
Intake Structure	Q Condensate Storage Tank (QCST)
Ultimate Heat Sink	Control Room
Startup Transformers	Auxiliary Building
Diesel Fuel Vault	Borated Water Storage Tank (BWST)
Refueling Water Tank (RWT)	

OR

3. Internal flooding in any of the following areas resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment or Control Room indication of degraded performance of those safety systems:

   Intake Structure
   Turbine Building
   Auxiliary Building

OR

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA6

4. Turbine failure-generated PROJECTILES resulting in VISIBLE DAMAGE to or penetration of any of the structures/equipment containing safety systems or components or Control Room indication of degraded performance of those safety systems:

Control Room	Turbine Building
Startup Transformers	Auxiliary Building

OR

5. Lake Dardanelle level < 335 feet and Emergency Cooling Pond inoperable.

OR

6. Vehicle crash resulting in VISIBLE DAMAGE to any of the structures/equipment containing safety systems or components or Control Room indication of degraded performance of those safety systems:

Reactor Building	Turbine Building
Intake Structure	QCST
Ultimate Heat Sink	RWT
Startup Transformers	Auxiliary Building
Diesel Fuel Vault	BWST

Basis:

These EALs escalate from HU6 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by Control Room indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction (S) ICs.

EAL #1

Seismic events of this magnitude can result in a VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

The National Earthquake Center can confirm if an earthquake has occurred in the area of the plant.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HA6

EAL #2

This EAL is based on a tornado striking (touching down) or high winds that have caused VISIBLE DAMAGE to structures containing functions or systems required for safe shutdown of the plant. The high wind value in EAL #2 is conservatively based on the SAR design basis for Unit 1 of 67 mph. Unit 2 Design basis is 80 mph.

EAL #3

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

EAL #4

This EAL addresses the threat to safety related equipment imposed by PROJECTILEs generated by main turbine rotating component failures. Therefore, this EAL is consistent with the definition of an ALERT in that the potential exists for actual or substantial potential degradation of the level of safety of the plant.

EAL #5

EAL #5 addresses site specific phenomena which has the potential for the loss of primary and secondary heat sink.

EAL #6

This EAL addresses vehicle crashes within the PROTECTED AREA that result in VISIBLE DAMAGE to VITAL AREAS or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant.

Reference Documents:

1. OP-1203.025, “Natural Emergencies”
2. OP-2203.008, “Natural Emergencies”
3. Unit 1 FSAR
4. Unit 2 FSAR

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HS1

Initiating Condition - SITE AREA EMERGENCY

HOSTILE ACTION within the PROTECTED AREA

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by ANO Security Shift Supervision.

Basis:

This condition represents an escalated threat to plant safety above that contained in the Alert in that a HOSTILE FORCE has progressed from the OWNER CONTROLLED AREA to the PROTECTED AREA.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires Offsite Response Organization readiness and preparation for the implementation of protective measures.

This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the PROTECTED AREA. Those events are adequately addressed by other EALs.

Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HS2

Initiating Condition - SITE AREA EMERGENCY

Other conditions exist which in the judgment of the SM / ED warrant declaration of a Site Area Emergency

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. Other conditions exist which in the judgment of the SM / ED indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the SM / ED to fall under the emergency classification level description for Site Area Emergency.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HS3

Initiating Condition - SITE AREA EMERGENCY

Control Room evacuation has been initiated and plant control cannot be established

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. a. Control room evacuation has been initiated

   AND

   b. Control of the plant cannot be established in accordance with the following procedures within 15 minutes:

   Unit 1: 1203.002, “Alternate Shutdown”

   Unit 2: 2203.014, “Alternate Shutdown”

Basis:

The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. In this case, expeditious transfer of control of safety systems has not occurred (although fission product barrier damage may not yet be indicated).

The intent of the EAL is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions such as reactivity control (ability to shutdown the reactor and maintain it shutdown), RCS inventory (ability to cool the core), and decay heat removal (ability to maintain a heat sink).

The determination of whether or not control is established is based on SM / ED judgment. The SM / ED is expected to make a reasonable, informed judgment within 15 minutes that the plant staff has control of the plant .

Escalation of this emergency classification level, if appropriate, would be by Fission Product Barrier Degradation (F) or Abnormal Radiation Levels/Radiological Effluent (A) EALs.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HG1

Initiating Condition - GENERAL EMERGENCY

HOSTILE ACTION resulting in loss of physical control of the facility

Operating Mode Applicability: All

Example Emergency Action Level(s): (1 or 2)

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.

   OR

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.

Basis:

EAL #1

This EAL encompasses conditions under which a HOSTILE ACTION has resulted in a loss of physical control of VITAL AREAS (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location. These safety functions are reactivity control (ability to shut down the reactor and keep it shutdown) RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).

Loss of physical control of the Control Room or remote shutdown/alternate shutdown capability alone may not prevent the ability to maintain safety functions per se. Design of the remote shutdown/alternate shutdown capability and the location of the transfer switches should be taken into account. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions.

If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met.

EAL #2

This EAL addresses failure of spent fuel cooling systems as a result of HOSTILE ACTION if IMMINENT fuel damage is likely, such as when a freshly off-loaded reactor core is in the spent fuel pool. At ANO, the term “freshly off-loaded reactor core” refers to fuel that has been discharged from the core and stored in the spent fuel pool for a period of LESS THAN one year.

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
HG2

Initiating Condition - GENERAL EMERGENCY

Other conditions exist which in the judgment of the SM / ED warrant declaration of a General Emergency

Operating Mode Applicability: All

Example Emergency Action Level(s):

1. Other conditions exist which in the judgment of the SM / ED indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the SM / ED to fall under the emergency classification level description for General Emergency.

SYSTEM MALFUNCTION

SYSTEM MALFUNCTION
SU1

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Loss of all offsite AC power to Vital 4.16 KV busses ≥ 15 minutes

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. Loss of all offsite AC power to Vital 4.16 KV busses ≥ 15 minutes.

Basis:

Prolonged loss of offsite AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power to emergency busses.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of off-site power.

Reference Documents:

1. 1202.007, “Degraded Power”
2. 1202.008, “Blackout”
3. 2202.007, “Loss of Off-Site Power”
4. 2202.008, “Station Blackout”

SYSTEM MALFUNCTION
SU6

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

UNPLANNED loss of safety system annunciation or indication in the Control Room ≥ 15 minutes

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

Note: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. UNPLANNED Loss of > approximately 75% of the following ≥ 15 minutes:

   a. Control Room annunciators associated with safety systems.

   OR

   b. Control Room safety system indication.

Basis:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered e.g., SPDS, plant computer, etc.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the NUE is based on SU11 "Inability to reach required operating mode within Technical Specification limits."

SYSTEM MALFUNCTION
SU6

Indicators associated with safety systems are those indicators for reactivity control, core cooling, maintaining reactor coolant system integrity or maintaining containment integrity.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

This NUE will be escalated to an Alert based on a concurrent loss of compensatory indications or if a SIGNIFICANT TRANSIENT is in progress during the loss of annunciation or indication (SA6).

Reference Documents:

1. 1203.043, “Loss Control Room Annunciators”

2. 2203.042, “Loss of Control Room Annunciators”

SYSTEM MALFUNCTION
SU7

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

RCS leakage

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s): (1 or 2)

1. Unidentified or pressure boundary leakage > 10 gpm.

   OR

2. Identified leakage > 25 gpm.

Basis:

With respect to this IC, RCS leakage is defined as a loss of RCS inventory due to a leak in the RCS or a supporting system that is not or cannot be isolated within 10 minutes. For example, isolation of the RCS Letdown (purification) system is a standard abnormal operating procedure action and may prevent unnecessary classifications when a non-RCS leakage path leak exists. However, the intent of this condition is met if attempts to isolate the RCS leak are NOT successful.

This IC is included as an NUE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified or pressure boundary leakage was selected as it is observable with normal Control Room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances).

Relief valve normal operation should be excluded from this IC. However, a relief valve that operates and fails to close per design should be considered applicable to this IC if the relief valve cannot be isolated.

The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. Steam generator tube leakage is identified leakage. In either case, escalation of this IC to the Alert level is via Fission Product Barrier Degradation (F) ICs.

SYSTEM MALFUNCTION
SU8

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Loss of all onsite or offsite communications capabilities

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s): (1 or 2)

1. Loss of all Table M1 onsite communications methods affecting the ability to perform routine operations.

   OR

2. Loss of all Table M2 offsite communications methods affecting the ability to perform offsite notifications.

Table M1 Onsite Communications Methods
Station radio system
Plant paging system
In-plant telephones
Gaitronics

Table M2 Offsite Communications Methods
All telephone lines (commercial and microwave)
ENS

Basis:

The purpose of this IC and its associated EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with offsite authorities.

The availability of one method of ordinary offsite communications is sufficient to inform federal, state, and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from non-routine radio transmissions, individuals being sent to off-site locations, etc.) are being used to make communications possible.

Reference Documents:

1. 1903.062, “Communications System Operating Procedure”

SYSTEM MALFUNCTION
SU9

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Fuel clad degradation

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s): (1 or 2)

1. Failed Fuel Iodine radiation monitor reading indicates fuel clad degradation > Technical Specification allowable limits:

   Unit 1:

   RI-1237S reads > 1.3 x 10^5 counts per minute

   Unit 2:

   2RITS-4806B reads > .65 x 10^5 counts per minute

   OR

2. RCS sample activity value indicating fuel clad degradation > Technical Specification allowable limits:

   - > 1.0 uCi/gm Dose Equivalent I-131 for more than 48 hours

     OR

   - **Unit 1:**

     ≥ 60 uCi/gm Dose Equivalent I-131

     Unit 2:

     > 60 uCi/gm Dose Equivalent I-131

     OR

   - **Unit 1:**

     > 2200 µCi/gm Dose Equivalent Xe-133 for more than 48 hours

     Unit 2:

     > 3100 µCi/gm Dose Equivalent Xe-133 for more than 48 hours

SYSTEM MALFUNCTION
SU9

Basis:

This IC is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

EAL #1

This threshold addresses the Letdown Radiation Monitor readings that provide indication of a degradation of fuel clad integrity.

EAL #2

This EAL addresses coolant samples exceeding coolant technical specifications for transient iodine spiking limits and coolant samples exceeding coolant Technical Specifications for nominal operating limits for the time period specified in the Technical Specifications.

Escalation of this IC to the Alert level is via the Fission Product Barriers (F).

Reference Documents:

1. ANO1 Technical Specifications
2. ANO2 Technical Specifications

SYSTEM MALFUNCTION
SU10

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Inadvertent criticality

Operating Mode Applicability: Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

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

Basis:

This IC addresses inadvertent criticality events. This IC indicates a potential degradation of the level of safety of the plant, warranting an NUE classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated).

This condition can be identified using the startup rate meter. The term “sustained“ is used in order to allow exclusion of expected short term positive startup rates from planned control rod movements for (such as shutdown bank withdrawal). These short term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

Escalation would be by the Fission Product Barrier Table (F), as appropriate to the operating mode at the time of the event.

Reference Documents:

1. 1203.012G, “Annunciator K08 Corrective Action”
2. 2203.012D, “Annunciator 2K04 Corrective Action”

SYSTEM MALFUNCTION
SU11

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

Inability to reach required operating mode within Technical Specification limits

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

1. Plant is not brought to required operating mode within Technical Specifications LCO Action Statement time.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required operating mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the site Technical Specifications requires a four hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate NUE is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of an NUE is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed.

<u>Reference Documents</u>:

1. ANO2 Technical Specifications
2. ANO1 Technical Specifications

SYSTEM MALFUNCTION
SA1

Initiating Condition - ALERT

AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes such that any additional single power source failure would result in station blackout

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. a. AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes.

   AND

   b. Any additional single power source failure will result in station blackout.

Basis:

The condition indicated by this IC is the degradation of the offsite and onsite AC power systems such that any additional single power source failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of all but one emergency generator to supply power to its emergency busses. Another related condition could be the loss of all offsite power and loss of onsite emergency generators with only one train of emergency busses being backfed from the unit main generator, or the loss of onsite emergency generators with only one train of emergency busses being backfed from offsite power. The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with **SS1**.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

The EAL allows credit for operation of the Alternate AC Diesel Generator.

Reference Documents:

1. 1202.007, “Degraded Power”
2. 1202.008, “Blackout”
3. 2202.007, “Loss of Off-Site Power”
4. 2202.008, “Station Blackout”
5. 2104.037, “Alternate AC Diesel Generator Operations”

SYSTEM MALFUNCTION
SA3

Initiating Condition - ALERT

Automatic trip fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)

Example Emergency Action Level(s):

1. a. An automatic trip failed to shutdown the reactor as indicated by reactor power ≥ 5%.

   AND

   b. Manual actions taken at the reactor control console successfully shutdown the reactor as indicated by reactor power < 5%.

Basis:

Manual trip actions taken at the reactor control console are any set of actions by the Reactor Operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor. Any action taken to trip the reactor from any location other than panel C03 (Unit 1) or 2C03/2C14 (Unit 2) constitutes a failure of the manual trip function. Failure of manual trip would escalate the event to a Site Area Emergency (**SS3**).

This condition indicates failure of the automatic protection system to trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient. Thus the plant safety has been compromised because design limits of the fuel may have been exceeded. An Alert is indicated because conditions may exist that lead to potential loss of fuel clad or RCS and because of the failure of the Reactor Protection System to automatically shutdown the plant. This EAL applies whether or not a mode change has occurred. (Reference "**Operating Mode Applicability"** page 72)

If manual actions taken at the reactor control console fail to shutdown the reactor, the event would escalate to a Site Area Emergency.

SYSTEM MALFUNCTION
SA6

Initiating Condition - ALERT

UNPLANNED loss of safety system annunciation or indication in the Control Room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) compensatory indicators unavailable

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

Note: *The SM/ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. a. UNPLANNED loss of > approximately 75% of the following ≥ 15 minutes:

    - Control Room annunciators associated with safety systems

      OR

    - Control Room safety system indication

    AND

   b. Either of the following:

    - A SIGNIFICANT TRANSIENT is in progress

      OR

    - Compensatory indications are unavailable.

Basis:

This IC is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a SIGNIFICANT TRANSIENT.

Recognition of the availability of computer based indication equipment is considered (e.g., SPDS, plant computer, etc.).

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

SYSTEM MALFUNCTION
SA6

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the NUE is based on SU11 "Inability to reach required operating mode within Technical Specification limits."

Indicators associated with safety systems are those indicators for reactivity control, core cooling, maintaining reactor coolant system integrity or maintaining containment integrity.

"Compensatory indications" in this context includes computer based information such as SPDS, QSPDS, COLSS, etc. If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress due to a concurrent loss of compensatory indications with a SIGNIFICANT TRANSIENT in progress during the loss of annunciation or indication.

<u>Reference Documents</u>:

1. 1015.037, “Post Transient Review”
2. 1203.043, “Loss of Control Room Annunciators”
3. 2203.042, “Loss of Control Room Annunciators”

SYSTEM MALFUNCTION
SS1

Initiating Condition - SITE AREA EMERGENCY

Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 15 minutes

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 15 minutes.

Basis:

Loss of all AC power to emergency busses compromises all plant safety systems requiring electric power including Shutdown Cooling, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power to emergency busses will lead to loss of Fuel Clad, RCS, and Containment, thus this event can escalate to a General Emergency.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of offsite power.

Escalation to General Emergency is via Fission Product Barrier Degradation (F) or IC SG1, "Prolonged loss of all offsite and all onsite AC power to Vital 4.16 KV busses."

Reference Documents:

1. 1202.007, “Degraded Power”
2. 1202.008, “Blackout”
3. 2202.007, “Loss of Off-Site Power”
4. 2202.008, “Station Blackout”
5. 2104.037, “Alternate AC Diesel Generator Operations”

SYSTEM MALFUNCTION
SS3

Initiating Condition - SITE AREA EMERGENCY

Automatic trip fails to shutdown the reactor and manual actions taken from the reactor control console are not successful in shutting down the reactor

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)

Example Emergency Action Level(s):

1. a. An automatic trip failed to shutdown the reactor.

   AND

   b. Manual actions taken at the reactor control console do not shutdown the reactor as indicated by reactor power ≥ 5%.

Basis:

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful. A Site Area Emergency is warranted because conditions exist that lead to IMMINENT loss or potential loss of both fuel clad and RCS.

Manual trip actions taken at the reactor control console are any set of actions by the Reactor Operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

Manual trip actions are not considered successful if action away from panel C03 (Unit 1) or panels 2C03/2C14 (Unit 2) is required to trip the reactor. This EAL is still applicable even if actions taken away from panel C03 (Unit 1) or panels 2C03/2C14 (Unit 2) are successful in shutting the reactor down because the design limits of the fuel may have been exceeded or because of the gross failure of the Reactor Protection System to shutdown the plant. This EAL applies whether or not a mode change has occurred. (Reference "**Operating Mode Applicability"** page 72)

Escalation of this event to a General Emergency would be due to a prolonged condition leading to an extreme challenge to either core-cooling or heat removal.

SYSTEM MALFUNCTION
SS4

Initiating Condition - SITE AREA EMERGENCY

Loss of all vital DC power ≥15 minutes

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. <105 volts on all Vital DC busses ≥15 minutes.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation to a General Emergency would occur by Abnormal Radiation Levels/Radiological Effluent (A), Fission Product Barrier Degradation (F).

SYSTEM MALFUNCTION
SS6

Initiating Condition - SITE AREA EMERGENCY

Inability to monitor a SIGNIFICANT TRANSIENT in progress

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

Note: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.*

1. a. Loss of > approximately 75% of the following ≥ 15 minutes:

   - Control Room annunciators associated with safety systems

     OR

   - Control Room safety system indication

   AND

   b. A SIGNIFICANT TRANSIENT is in progress.

   AND

   c. Compensatory indications are unavailable.

Basis:

This IC is intended to recognize the threat to plant safety associated with the complete loss of capability of the control room staff to monitor plant response to a SIGNIFICANT TRANSIENT.

"Planned" and “UNPLANNED” actions are not differentiated since the loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

SYSTEM MALFUNCTION
SS6

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the NUE is based on SU11 "Inability to reach required operating mode within Technical Specification limits."

A Site Area Emergency is considered to exist if the Control Room staff cannot monitor safety functions needed for protection of the public while a significant transient is in progress.

Site specific indications needed to monitor safety functions necessary for protection of the public must include Control Room indications, computer generated indications and dedicated annunciation capability.

Indicators associated with safety systems are those indicators for reactivity control, core cooling, maintaining reactor coolant system integrity or maintaining containment integrity.

"Compensatory indications" in this context includes computer based information such as SPDS, QSPDS, COLSS, etc. This should include all computer systems available for this use depending on specific plant design and subsequent retrofits.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

<u>Reference Documents</u>:

1. 015.037, “Post Transient Review”
2. 1203.043, “Loss of Control Room Annunciators”
3. 2203.042, “Loss of Control Room Annunciators”

SYSTEM MALFUNCTION
SG1

Initiating Condition - GENERAL EMERGENCY

Prolonged loss of all offsite and all onsite AC power to Vital 4.16 KV busses

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s):

1. a. Loss of all offsite and all onsite AC power to Vital 4.16 KV busses.

   AND

   b. Either of the following:

   - Restoration of at least one Vital 4.16 KV bus in <4 hours is not likely.

     OR

   - Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by CETs ≥700 °F.

Basis:

Loss of all AC power to Vital 4.16 KV busses compromises all plant safety systems requiring electric power including Shutdown Cooling, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power to Vital 4.16 KV busses will lead to loss of fuel clad, RCS, and containment, thus warranting declaration of a General Emergency.

This IC is specified to assure that in the unlikely event of a prolonged station blackout, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.

The likelihood of restoring at least one Vital 4.16 KV bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

In addition, under these conditions, fission product barrier monitoring capability may be degraded.

SYSTEM MALFUNCTION
SG1

Although it may be difficult to predict when power can be restored, it is necessary to give the SM / ED a reasonable idea of how quickly (s)he may need to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of Fission Product Barriers is IMMINENT?

2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on SM / ED judgment as it relates to IMMINENT loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.

Reference Documents:

1. Unit 1 Calculation 85-E-0072-02, “Time from Loss of All AC Power to Loss of Subcooling”

2. Unit 2 Calculation 85-E-0072-01, “Time from Loss of All AC Power to Loss of Subcooling”

SYSTEM MALFUNCTION
SG3

Initiating Condition - GENERAL EMERGENCY

Automatic trip and all manual actions fail to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)

Example Emergency Action Level(s):

1. a. An automatic trip failed to shutdown the reactor

   AND

   b. All manual actions do not shutdown the reactor as indicated by reactor power ≥5%.

   AND

   c. Either of the following exist or have occurred due to continued power generation:

   - CET temperatures at or approaching 1200 °F

     OR

   - Feedwater flow rate less than:

     Unit 1: 430 gpm

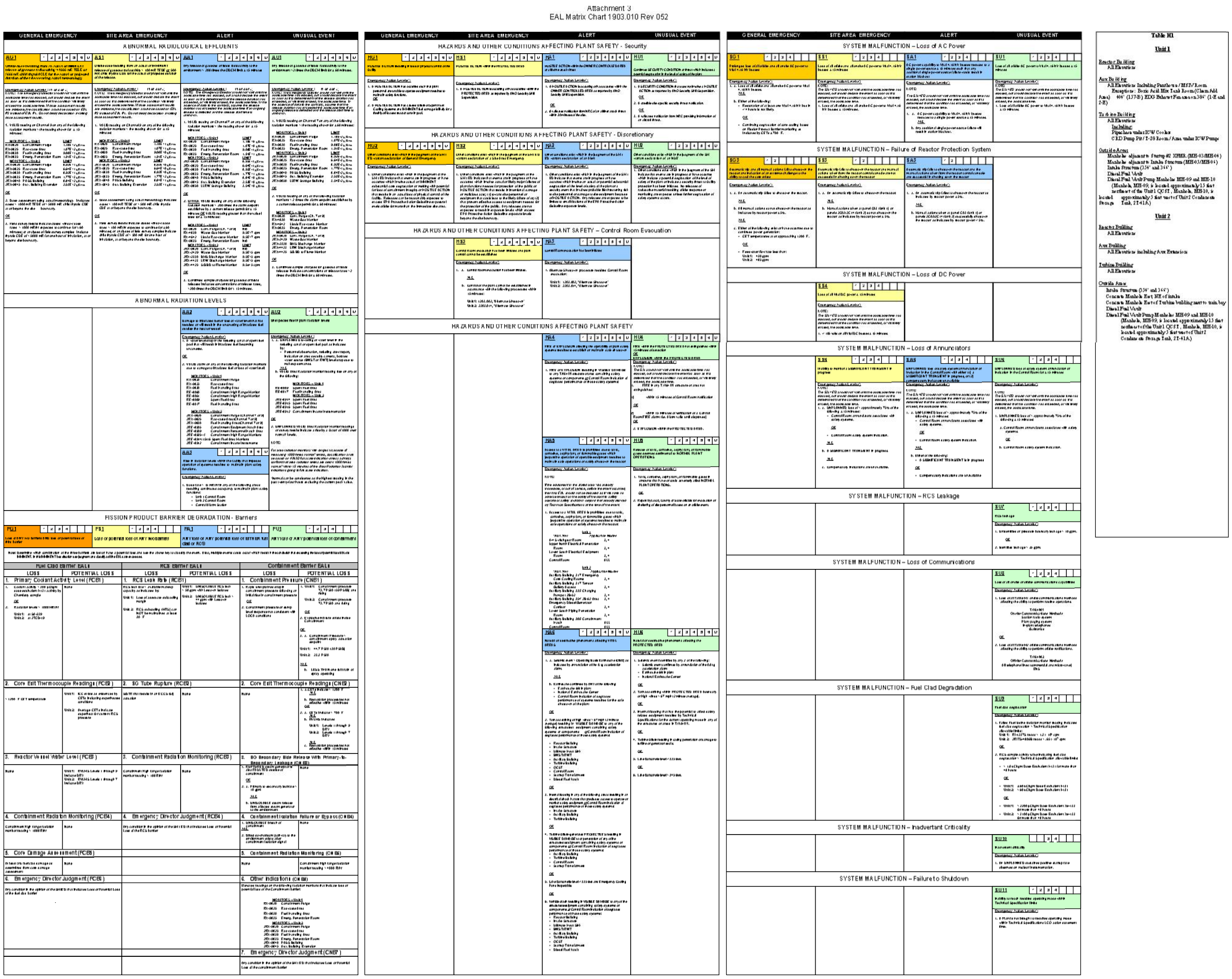
     Unit 2: 485 gpm

Basis:

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful.

In the event either of these challenges exists at a time that the reactor has not been brought below the power associated with the safety system design a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier table declaration to permit maximum off-site intervention time.

Attachment 3
EAL Matrix Chart



Attachment 3
EAL Matrix Chart

Attachment 3
EAL Matrix Chart Procedure 1903.010 Rev 052

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of RCS / Reactor Vessel Inventory

CG1 (5, 6) – Loss of RCS / reactor vessel inventory affecting fuel clad integrity with containment challenged

Emergency Action Level(s):

NOTE: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. a. Core exit thermocouples indicate superheat for ≥ 30 minutes.

   AND

   b. Any of the following containment challenge indications:
   - CONTAINMENT CLOSURE not established
   - Explosive mixture inside containment
   - UNPLANNED rise in containment pressure

OR

2. a. RCS / reactor vessel level cannot be monitored for ≥ 30 minutes with a loss of RCS / reactor vessel inventory as indicated by any of the following:
   - Containment High Range Radiation Monitor reading >10 R/hr
   - Erratic source range monitor indication
   - Unexplained level rise in Reactor Building Sump, Reactor Drain Tank, Quench Tank, Aux. Building Equipment Drain Tank, or Aux. Building Sump

   AND

   b. Any of the following containment challenge indications:
   - CONTAINMENT CLOSURE not established
   - Explosive mixture inside containment
   - UNPLANNED rise in containment pressure

CS1 (5, 6) – Loss of RCS / reactor vessel inventory affecting core decay heat removal capability

Emergency Action Level(s):

NOTE: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. With CONTAINMENT CLOSURE **not** established:

   Loss of RCS / reactor vessel level as indicated by:
   Unit 1: RVLMS Levels 1 through 9 indicate DRY
   Unit 2: RVLMS Levels 1 through 6 indicate DRY

   OR

2. With CONTAINMENT CLOSURE established, core exit thermocouples indicate superheat.

   OR

3. RCS / reactor vessel level cannot be monitored for ≥ 30 minutes with a loss of RCS / reactor vessel inventory as indicated by any of the following:
   - Containment High Range Radiation Monitor reading > 10 R/hr
   - Erratic source range monitor indication
   - Unexplained level rise in Reactor Building Sump, Reactor Drain Tank, Quench Tank, Aux. Building Equipment Drain Tank, or Aux. Building Sump

CA1 (5, 6) – Loss of RCS / reactor vessel inventory

Emergency Action Level(s):

NOTE: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. Loss of RCS / reactor vessel inventory as indicated by:

   Unit 1: RVLMS Levels 1 through 8 indicate DRY

   Unit 2: RVLMS Levels 1 through 5 indicate DRY

   OR

   Unit 1: Reactor vessel level < 368 ft., 0 in. (bottom of the hot leg)

   Unit 2: Reactor vessel level < 369 ft., 1.5 in. (bottom of the hot leg)

   OR

2. RCS / reactor vessel level cannot be monitored for ≥ 15 minutes with a loss of RCS / reactor vessel inventory as indicated by an unexplained level rise in the Reactor Building Sump, Reactor Drain Tank, Aux. Building Equipment Drain Tank, Aux. Building Sump, or Quench Tank.

CU1 (5) – RCS leakage

Emergency Action Level(s):

NOTE: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. RCS leakage results in the inability to maintain or restore level within Pressurizer or RCS level target band for ≥ 15 minutes.

CU2 (6) – UNPLANNED loss of RCS / reactor vessel inventory

Emergency Action Level(s):

NOTE: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. UNPLANNED RCS / reactor vessel level drop as indicated by either of the following:

   a. RCS / reactor vessel water level drop below the reactor vessel flange for ≥15 minutes when the RCS / reactor vessel level band is established above the reactor vessel flange.

   OR

   b. RCS / reactor vessel water level drop below the RCS / reactor vessel level band for ≥ 15 minutes when the RCS / reactor vessel level band is established below the reactor vessel flange.

   OR

2. RCS / reactor vessel level cannot be monitored with a loss of RCS / reactor vessel inventory as indicated by an unexplained level rise in (as applicable) the Reactor Building Sump, Reactor Drain Tank, Aux. Building Equipment Drain Tank, Aux. Building Sump, or Quench Tank.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of Decay Heat Removal

CA3 (5, 6) – Inability to maintain plant in Cold Shutdown

Emergency Action Level(s):

1. An UNPLANNED event results in RCS temperature > 200 °F for > the specified duration in Table C1.

Table C1
RCS Reheat Duration Threshold

RCS	Containment Closure	Duration
Intact (but not at reduced inventory)	N/A	60 minutes*
Not intact or RCS reduced inventory	Established	20 minutes*
	Not established	0 minutes

*If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable

OR

Note: EAL #2 does not apply in solid plant conditions.

2. An UNPLANNED event results in RCS pressure rise > 10 psi due to a loss of RCS cooling.

CU3 (5, 6) – UNPLANNED loss of decay heat removal capability with irradiated fuel in the reactor vessel

Emergency Action Level(s):

NOTE: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. UNPLANNED event results in RCS temperature exceeding 200 °F.

   OR

2. Loss of all RCS temperature and RCS / reactor vessel level indication for ≥ 15 minutes.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of AC Power

CA5 (5, 6, D) – Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 15 minutes

Emergency Action Level(s):

NOTE: *The SM / ED should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. Loss of all offsite and all onsite AC power to Vital 4.16 KV busses ≥ 15 minutes.

CU5 (5, 6) – AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes such that any additional single power source failure would result in station blackout

Emergency Action Level(s):

NOTE: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. a. AC power capability to Vital 4.16 KV busses reduced to a single power source ≥ 15 minutes.

   AND

   b. Any additional single power source failure will result in station blackout.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of DC Power

CU6 (5, 6) – Loss of required DC power ≥ 15 minutes

Emergency Action Level(s):

NOTE: *The SM should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. < 105 volts on required Vital DC bus ≥ 15 minutes.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Inadvertent Criticality

CU7 (5, 6) – Inadvertent criticality

Emergency Action Level(s):

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

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION – Loss of Communications

CU8 (5, 6, D) – Loss of all onsite or offsite communications capabilities

Emergency Action Level(s):

1. Loss of all Table C2 onsite communication methods affecting the ability to perform routine operations.

Table C2
Onsite Communications Equipment
Station radio system
Plant paging system
In-plant telephones
Gaitronics

OR

2. Loss of all Table C3 offsite communication methods affecting the ability to perform offsite notifications.

Table C3
Offsite Communications Equipment
All telephone lines (commercial and microwave)
ENS

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

ISFSI MALFUNCTION – Cask Damage

E-HU1 (1, 2, 3, 4, 5, 6, D) – Damage to a loaded cask CONFINEMENT BOUNDARY

Note: Security Events are bounded by the Hazards EALs.

Emergency Action Level(s):

1. Damage to a loaded cask CONFINEMENT BOUNDARY.

Definitions

CONFINEMENT BOUNDARY: The barrier(s) between areas containing radioactive substances and the environment

CONTAINMENT CLOSURE: The actions to secure primary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment closure must be capable of being set within 30 minutes. Containment Closure is Set when the penetrations are isolated by manual or automatic isolation valve, blind flange, or equivalent.

EXPLOSION: A rapid, violent, unconfined combustion or catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

FAULTED: In a steam generator, the existence of secondary side leakage that results in an uncontrolled drop in steam generator pressure or the steam generator being completely depressurized.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIREs.

HOSTILE ACTION: An act toward a Nuclear Power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur. Where IMMINENT timeframes are specified, they shall apply.

PROTECTED AREA: An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

RUPTURED: In a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.\

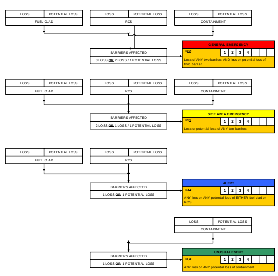
SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) automatic turbine runback > 25% thermal reactor power, (2) electrical load rejection > 25% full electrical load, (3) Reactor Trip, (4) Safety Injection Activation, or (5) thermal power oscillations > 10%.

UNISOLABLE: A breach or leak that cannot be promptly isolated

UNPLANNED: A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions

VALID: An indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

VISIBLE DAMAGE: Damage to equipment or structures that is readily observable without measurement, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.



ECCS SUBSYSTEMS – $T_{avg} \geq 300°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each sub-system comprised of:

a. One OPERABLE high-pressure safety injection (HPSI) train,

b. One OPERABLE low-pressure safety injection (LPSI) train, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 1, 2 and 3 with pressurizer pressure ≥ 1700 psia.

ACTION:

a. With one ECCS subsystem inoperable due to an inoperable LPSI train, restore the inoperable train to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

b. With one or more ECCS subsystems inoperable due to conditions other than "a" above and 100% of ECCS flow equivalent to a single OPERABLE HPSI and LPSI train is available, restore the inoperable train(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

c. With less than 100% ECCS flow equivalent to either the HPSI or LPSI trains within both ECCS subsystems, restore at least one HPSI train and one LPSI train to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

d. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the NRC within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

Question 01

Bank:	2331	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	1	**Author:**	Hatman
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC Exam Bank 1500		
Search	000054K101	**10CFR55:**	41.10	**Safety Function**	4				
Title:	Loss of Main Feedwater (MFW)	**System Number**	054	**K/A**	AK1.01				
Tier:	1	**Group:**	1	**RO Imp:**	4.1	**SRO Imp:**	4.3	**L. Plan:**	A2LP-RO-ELOSF
OBJ	5								

Description: Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): - MFW line break depressurizes the S/G (similar to a steam line break)

Question:

Given the following:

* The plant is at full power.
* A 200 gpm Feedwater line break downstream of Main Feedwater Check valve (2FW-5A) occurs.
* Containment temperature, pressure and humidity start rising.
* The plant is manually tripped.
* EFAS is automatically actuated.
* SG 'A' level is 35% Narrow Range and lowering.

Based on these conditions Steam Generator 'A' will depressurize and start an uncontrolled cooldown when:

A. Steam Generator 'A' level drops below 10% Narrow Range level.

B. The Main Feedwater Regulating valves are closed to 'A' Steam Generator.

C. The DP between B and A Steam Generators is > 70 psid and rising.

D. A MSIS occurs and Emergency Feedwater to 'A' Steam Generator is secured.

Answer:

D. A MSIS occurs and Emergency Feedwater to 'A' Steam Generator is secured.

Notes:

D is correct. As long as a feed source exists to "A" Steam Generator - Main or EFW, the feed source will be at a higher pressure than the Steam Generator, therefore the feed source (MFW/EFW) will be going out the leak. Once all feed is secured by MSIS and EFW due to the DP between 'A' and 'B' SG being > 90 psid, then steaming of the generator through the break will occur and cause an uncontrolled cooldown as long as the level in the SG is below the Feed Ring level (49% Narrow Range) (See the FW drawing in the reference package)

A is incorrect because EFW is feeding the "A" SG at this point if pressure is high enough (> 751 psia) but plausible as this is considered a very low SG level below the feed ring which would allow steam to exit the FW inlet if no feed was available and if pressure is too low will not feed the broken SG. 10% is the minimum SG level to meet the EOP Heat Removal safety function. Above the feed ring level (49% Narrow Range SG level will not act like an excess steam demand as water will exit the SG instead of Steam if all Feedwater is secured,

B is incorrect because EFAS has initiated and will feed the SG to prevent Excess Leakage from the SG but plausible because Main Feedwater will cause an excess steam demand when isolated if EFAS has not actuated.

C is incorrect because EFW will be feeding the SGs at this point if SG pressure is >751 psia and depending on the DP between SGs, will feed the "A" SG until DP goes above 90 psid with a higher pressure in "B" SG but plausible as 70 psid

is too low a DP. 70 psid was chosen as 70% Narrow Range level is the normal operating level for the SGs.

This question matches the K&A because the candidate must recognize the operational implications of a change in conditions from a feed line break to an excess steam demand if the crew secures feeding the SG. Feed must be secured at some point for repair but depending on the size, a plant cooldown could be performed prior to securing feed.

References:

STM_2-19_15-1 Main Feedwater SYS Overview (Verified reference updated 11/10/16); STM_2-15_19-1 SG and Main Steam Systems, SG level Drawing (Verified reference updated 11/10/16);
STM_2-19-2_39-1 EFW and AFW SYS Section 2.3.3.1 (Verified reference updated 11/10/16).

Historical Comments:

Used on the 2008 NRC Exam
To be used on the 2017 NRC Exam: Altered distractor C to be more plausible based on OPS Representative feedback and altered the order of the correct answer from the last time the QID was given. Altered the level in distractor A based on Facility Representative feedback to 10 % as this is below the EFAS setpoint and stem states that EFAS has actuated.

REV. 1 based on NRC Chief Examiner Feedback BNC, Added EFW secured by DP to the distractor analysis for the correct answer D.   Enhanced distractor A analysis to help explain why SG level affect the characteristics of the line break based on SG level. Changed distractor B to the Main Feedwater Regulating Valves

STM 2-19

Main Feedwater System

1.0 Introduction

1.1 Functions

The functions of the Main Feedwater system are as follows;

1) Provide continuous feedwater supply to the Steam Generators during normal plant operations between 2% and 100% power.

2) Provide isolation to the Steam Generators during a main steam or feedwater line break event or Containment Building overpressure condition.

1.2 System Overview

The Feedwater System is comprised of two trains of Main Feedwater pumps, pump Recirculation valves, Feedwater flow control valves, associated piping and controls. Refer to the figure on page 30. The Feedwater pumps take suction from the combined discharge headers of the Heater Drain Pumps and the Condensate system. A cross connect valve, 2CV-0742, in the common suction header to the Feedwater pumps allows all of the condensate sources to be connected to either Feedwater pump at all times. There are two centrifugal Main Feedwater Pumps (MFP) each rated to deliver 90% of the feedwater flow required at 100% plant power. The pumps are driven by steam turbines. In the suction line to each feedwater pump is a flow element, 2FE-0735 and 2FE-0742 for the “A” and “B” pump respectively, that provides control and alarm functions.

Recirculation valves are provided on the discharge of each MFP to maintain a minimum flow through the MFPs. This prevents the pumps from overheating and seizing when flow through the system is low. Also the Recirculation valves in conjunction with the high discharge pressure trip of the MFP’s provide for over pressure protection of the system. The recirculation flow is routed to the main Condenser, 2E-11A. There are two cross-connect lines downstream of the MFPs. One of these lines is an unisolable 24” line at the discharge of the MFPs. The other is a 24” line downstream of the high pressure feedwater heaters which has a normally closed isolation valve, 2FW-4A.

Located at each MFP discharge are the Main Feedwater (flow) Regulating valve (MFRV) and Main Feedwater Regulating Bypass valve (MFRBV). These valves are normally automatically positioned by signals from the Feedwater Control System in order to maintain a desired Steam Generator water level (Refer to STM 2-69, Feedwater Control System for more details). The 16” FRVs will ramp open as power is increased based on flow demand signals from the Feedwater Control System. At 100% power these valves are normally about 65-75% open. The FRBVs, which are also positioned by the FWCS, are used to control the feedwater flowrate when plant power is between

approximately 2% (when a MFP is started) and 15% power. These valves will ramp open with the FWCS in automatic. (Refer to STM 2-69 FWCS)

The Auxiliary Feedwater (AFW) pump, 2P-75, supplies auxiliary feedwater from the Condensate Storage tank to the Steam Generators during periods when the plant is shutdown or during plant startups when power below approximately 2% power. This low power feedwater supply can also be provided by the Emergency Feedwater pumps 2P7A (turbine driven) or 2P7B (motor driven). (Refer to STM 2-19-2, Emergency Feedwater System and Auxiliary Feedwater System for more details regarding the AFW system).

Two High Pressure Feedwater Heaters are provided for each feedwater train. These heaters are designated 2E1A/B and 2E2A/B. The heaters are used to improve the efficiency of the secondary system by preheating the feedwater prior to it entering the Steam Generators. Feedwater heating is provided by utilizing extraction steam from the Main Turbine to increase feedwater temperature in the heaters.

Downstream of the high pressure feedwater heaters are feedwater flow elements 2FE-1029 and 2FE-1129. These instruments provide flow signals to the Feedwater Control System which uses them to calculate the desired flow rate to maintain Steam Generator water level (Refer to STM 2-69, Feedwater Control System for more details). These flow instruments via the FWCS also provide flow input to a combined Steam flow/Feedwater flow recorder located on Control Room panel 2C02.

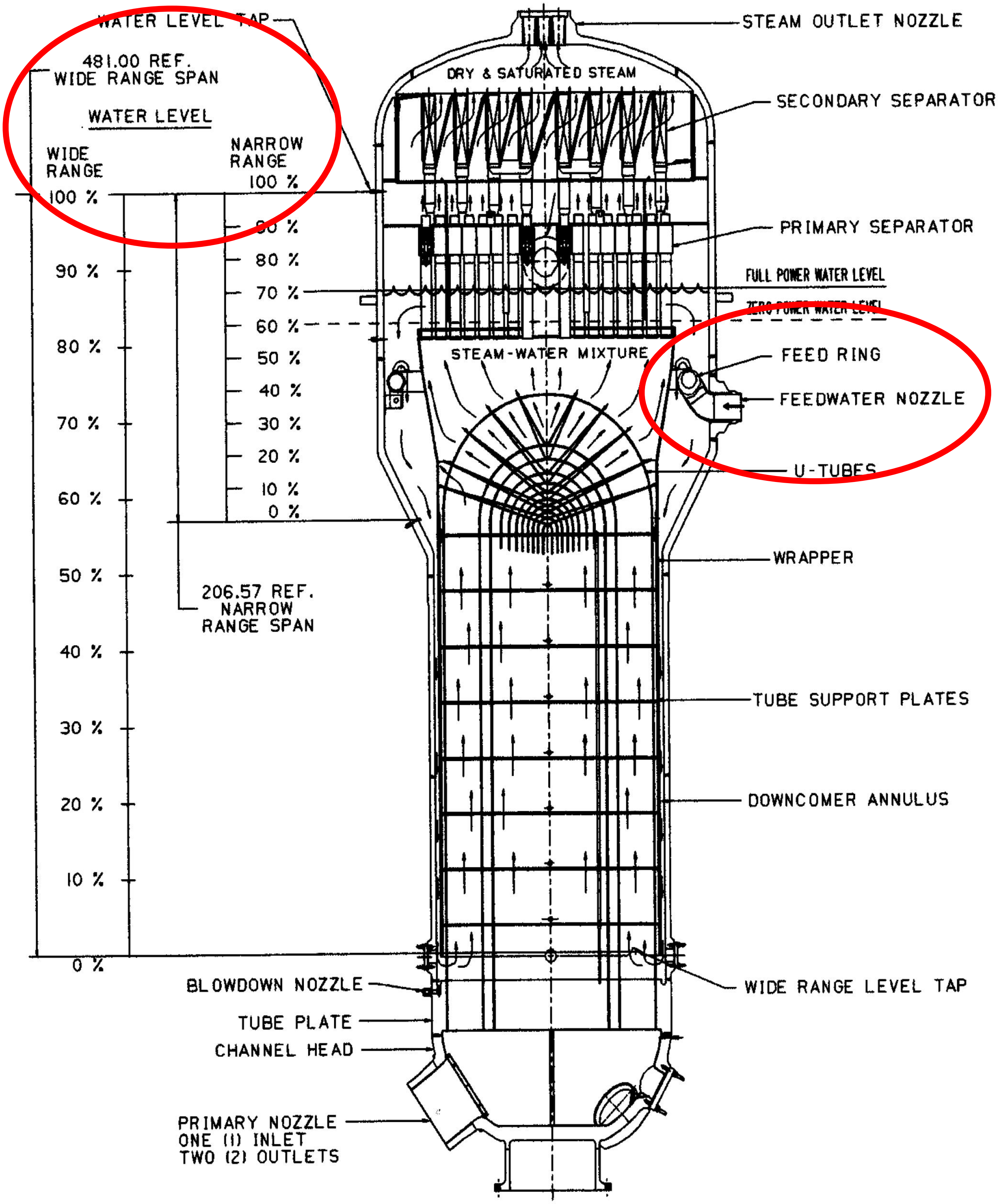
Two sets of motor operated Main Feedwater Isolation Valves (MFWIV) (commonly referred to as Block Valves) are provided on the Feedwater lines for isolation capability. These valves are located upstream (outside) of the Containment building. The MFWIVs are safety related valves that will automatically close upon receipt of a Main Steam Isolation Signal (MSIS) or a Containment Spray Actuation Signal (CSAS) from the Engineered Safety Features Actuation System (ESFAS).

A check valve is also provided in each Feedwater header. These valves, 2FW-5A and B, are located inside the Containment. If a line break should occur between the check valves and the MFWIVs the check valves will prevent the Steam Generator inventories from emptying into the Containment.

All piping downstream of the MFWIVs including the MFWIVs are Seismic Category I.

The Emergency Feedwater line for each Steam Generator combines with the Main Feedwater lines to the Steam Generators inside the Containment just downstream of the Main Feedwater line check valves.

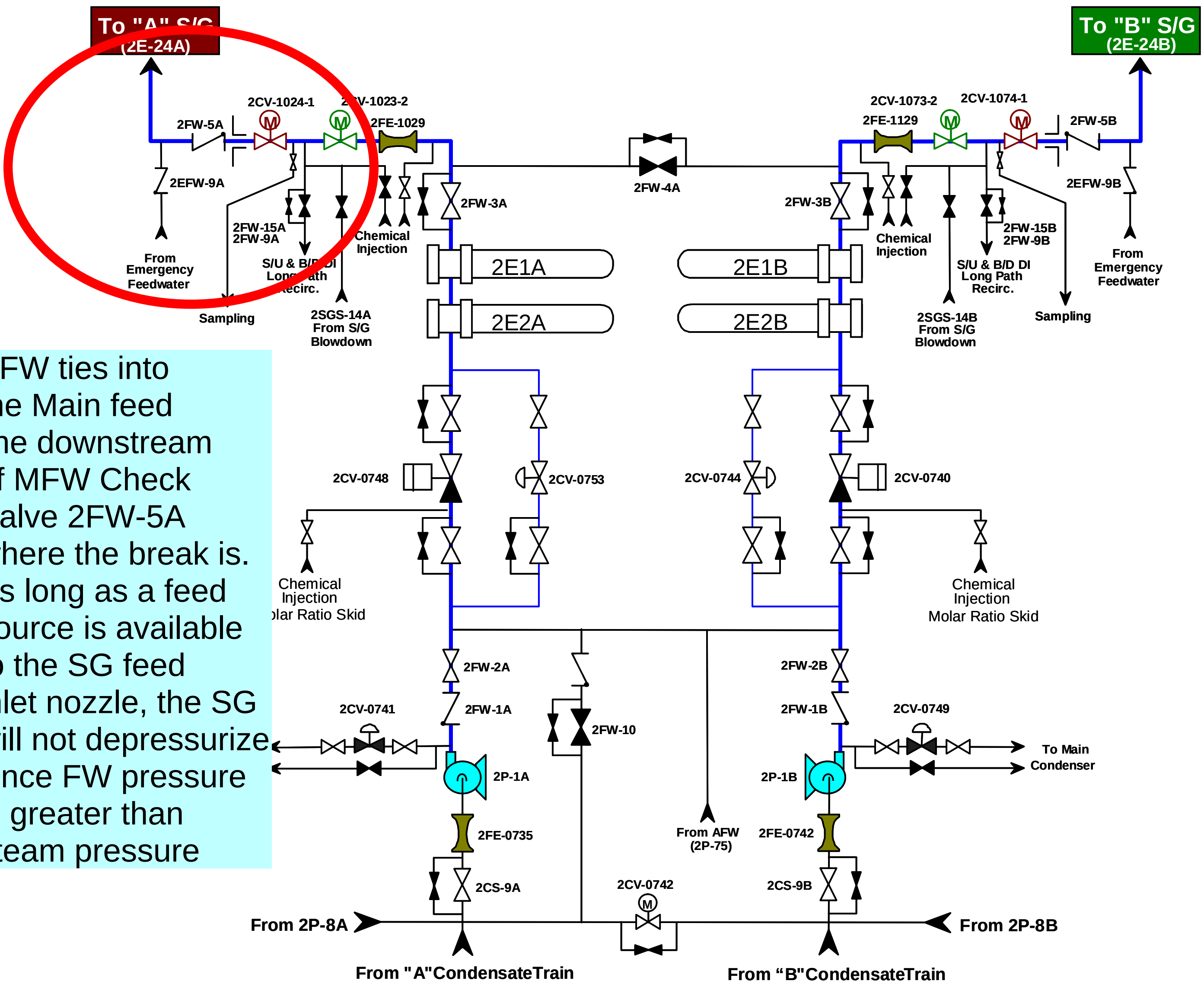
A 12” line is provided between the MFWIVs on each header to allow routing Feedwater flow to the Start-up and blowdown demineralizer for long path clean up recirculation. Another 4” line


WATER LEVEL TAP
481.00 REF.
WIDE RANGE SPAN
WATER LEVEL
WIDE RANGE
NARROW RANGE
100 %
90 %
80 %
70 %
60 %
50 %
40 %
30 %
20 %
10 %
0 %
100 %
90 %
80 %
70 %
60 %
50 %
40 %
30 %
20 %
10 %
0 %
206.57 REF.
NARROW RANGE SPAN
STEAM OUTLET NOZZLE
DRY & SATURATED STEAM
SECONDARY SEPARATOR
PRIMARY SEPARATOR
FULL POWER WATER LEVEL
ZERO POWER WATER LEVEL
STEAM-WATER MIXTURE
FEED RING
FEEDWATER NOZZLE
U-TUBES
WRAPPER
TUBE SUPPORT PLATES
DOWNCOMER ANNULUS
BLOWDOWN NOZZLE
WIDE RANGE LEVEL TAP
TUBE PLATE
CHANNEL HEAD
PRIMARY NOZZLE
ONE (1) INLET
TWO (2) OUTLETS

connects off of this header to provide return from the Steam Generator Blowdown pumps for Steam Generator recirculation during shutdown operations. Chemicals can be injected into this line during recirculation to maintain Steam Generator and feedwater chemistry.

An additional line ties into the Feedwater header between the MFWIVs and diverts feedwater to the Secondary sampling system for feedwater chemistry analysis.

Boric acid can be injected into the Feedwater system just upstream of the Feedwater flow elements 2FE-1029 and 2FE-1129 on each header.



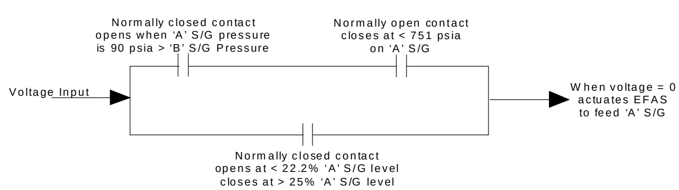
EFW ties into the Main feed line downstream of MFW Check Valve 2FW-5A where the break is. As long as a feed source is available to the SG feed inlet nozzle, the SG will not depressurize since FW pressure is greater than steam pressure

Description	Valve	Power Supply	Control Power
2P7A to ‘A’ S/G	2CV-1026-2	Green DC (2D26-A4)	2D24
2P7A to ‘A’ S/G	2CV-1037-1	Red DC (2D27-B1)	2D23
2P7A to ‘B’ S/G	2CV-1076-2	Green DC (2D26-C1)	2D24
2P7A to ‘B’ S/G	2CV-1039-1	Red DC (2D27-B2)	2D23
2P7B to ‘A’ S/G	2CV-1025-1	Red AC (2B51-N2)	2D23
2P7B to ‘A’ S/G	2CV-1038-2	Green AC (2B63-H3)	2D24
2P7B to ‘B’ S/G	2CV-1075-1	Red AC (2B53-J2)	2D23
2P7B to ‘B’ S/G	2CV-1036-2	Green AC (2B63-H1)	2D24

The valves receive an automatic open signal on low S/G level of 22.2% due to an Emergency Feedwater Actuation Signal (EFAS) from the Plant Protection System (PPS). When S/G level raises to ~ 25%, (~ EFAS reset valve) the automatic open signal is removed and the valve will close. This is different from most Emergency Safety Feature Actuation System automatic signals in that the valves automatically reposition when the trip reset setpoint has been reached. This action prevents overcooling of the Reactor Cooling System due to excessive feeding of the S/G's. It should be noted that the automatic reset of EFAS applies to the EFW Pump Discharge Valves to the S/G's and not any other EFAS actuated component. For more information on the PPS and EFAS, refer to STM 2-63.

2.3.3.1 EFW Pump Discharge to the S/G's Motor Operated Valves Simplified EFAS Actuation Circuit

A Main Steam Isolation Signal (MSIS) also effects the position of the EFW Pump Discharge Valves to the S/G's. When either S/G pressure lowers to 751 psia an MSIS is generated in the PPS and the S/G's steam and feedwater lines are isolated. This helps minimize the overcooling of the RCS. For the EFW to S/G's valves the MSIS removes the EFAS signal and allows the valves to close. Once the PPS has determined the affected S/G, the EFAS to the unaffected S/G is returned to normal and the valves will cycle on S/G level. The unaffected S/G is the S/G that has a 90 psia higher pressure. If the pressure in both S/Gs rises to > 751 psia then the valves will cycle on S/G level only.



Simplifed EFAS Actuation to "A" S/G Circuit

The control valves can be manually controlled at the valves. In the area of the 4 control valves in the Upper South Piping penetration Area is a level indication for the “A” Steam Generator. This provides a means for the operator to control S/G level locally at the valves. The

ORIGINAL BANK QUESTION 1500 WAS USED TO DEVELOP 2017 NRC EXAM QID#01 (Used on the 2008 NRC Exam)

Questions For All QID In Exam Bank

Bank: 1500	**Rev:** 001	**Rev Date:** 1/4/2008 2:35:47 P		**QID #:** 13		**Author:** Hatman	
Lic Level: R	**Difficulty:** 2	**Taxonomy:** H		**Source:** NRC Bank 0069 (1998 NRC Exam)			
Search 00CE06A102	**10CFR55:** 41.7 / 45.5 / 45.6			**Safety Function** 4			
System Title: Loss of Feedwater				**System Number** E06		**K/A** EA1.2	
Tier: 1	**Group:** 1	**RO Imp:** 3.4	**SRO Imp:** 4.0	**L. Plan:** A2LP-RO-ELOSF		**OBJ** 5	

Description: Ability to operate and/or monitor the following as they apply to the (Loss of Feedwater): - Operating behavior characteristics of the facility.

Question:

Given the following:

* The plant is at full power.
* A 200 gpm Main Feedwater line break downstream of Main Feedwater Check valve (2FW-5A) occurs.
* Containment temperature, pressure and humidity start rising.
* The plant is manually tripped.
* EFAS is automatically actuated.

Based on these conditions the affected Steam Generator will depressurize and start an uncontrolled cooldown when:

A. Steam Generator 'A' level drops below 22.3% Narrow Range level.

B. The main feedwater isolation valve is closed to "A" Steam Generator.

C. Main and Emergency Feedwater to "A" Steam Generator is secured.

D. Steam Generator 'A' level drops below 300 inches Wide Range level.

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☑	☑

Audit Exam History

2006	☑
2008	☐

Answer:

C. Main and Emergency Feedwater to "A" Steam Generator is secured.

Notes:

As long as a feed source exists to "A" Steam Generator, the feed source will be at a higher pressure than the Steam Generator, therefore the feed source (MFW/EFW) will be going out the leak. Once all feed is secured by MSIS & 90# delta P then steaming of the generator through the break will occur and cause an uncontrolled cooldown.

References:

STM2-19, Sections 1.0 and 8.2.

Historical Comments:

Used on 1998 NRC Exam; 10/24/2007.

Revised on 01/04/2008 based on validation comments. Had to assume EFW was in service on the previous revision.

Question 02

Bank:	2332	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	2	**Author:**	Coble		
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam				
Search	000077K102	**10CFR55:**	41.5	**Safety Function**	6						
Title:	Generator Voltage and Electric Grid Disturbances	**System Number**	077	**K/A**	AK1.02						
Tier:	1	**Group:**	1	**RO Imp:**	3.3	**SRO Imp:**	3.4	**L. Plan:**	A2LP-RO-MGEN	**OBJ**	7

Description: Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances: - Over-excitation

Question:

(REFERENCE PROVIDED)

Given the following at full power:

* A Loss of the 500 KV Mabelvale line has occurred.
* Main Generator Megawatts are 1050 MWe.
* Main Generator Hydrogen Pressure is 60 psig.
* Main Generator Reactive Load is 400 MVARS Out "+".
* Main Generator Field Exciter Current is 4750 Amps.
* Generator Voltage Regulator is in AUTO.

Based on these conditions, in accordance with OP-2106.009, Turbine Generator Operations, Main Generator _____________ are required to be reduced to within the required limits by bumping the _______________________________ COUNTER-CLOCKWISE (Lower) to prevent overheating of the Main Generator.

A. MVARS; Terminal Voltage Adjust handswitch

B. FIELD AMPS; Terminal Voltage Adjust handswitch

C. MVARS; Generator Field Voltage Adjust handswitch

D. FIELD AMPS; Generator Field Voltage Adjust handswitch

Answer:

A. MVARS; Terminal Voltage Adjust handswitch

Notes:

Answer A is correct because the Main Generator MVARS of 400 out are outside the range of the Generator Capability curve for 1050 Mwe and 60 psig of Hydrogen Pressure IAW NOP 2106.009, Turbine Generator Operations, Attachment C and with the Voltage Regulator in AUTO, the "Terminal Voltage adjust handswitch" must be bumped COUNTER-CLOCKWISE to lower VARs.

Distracter B is incorrect because the Main Generator Field Amp limit is 5448 Amps and actual is 4750 Amps which is much less than the limit but plausible because AMPs are higher than the normal 100% load value of 4400 Amps taken daily on the OPS B-2 Logs but plausible as the overexcited generator will have > 100% load AMPs and plausible as the correct switch is listed to lower MVARs.

Distracter C is incorrect because with the voltage regulator in AUTO, the wrong switch is used to lower MVARs but plausible as this would be the correct switch if the voltage regulator was in "Manual" and the correct parameter that is over the limit is listed.

Distracter D is incorrect because the Main Generator Field Amp limit is 5448 Amps and actual is 4750 Amps which is much less than the limit but plausible because AMPs are higher than the normal 100% load value of 4400 Amps taken daily on the OPS B-2 Logs and incorrect because with the voltage regulator in AUTO, the wrong switch is used to lower MVARs but plausible as this would be the correct switch if the voltage regulator was in "Manual" and plausible as the overexcited generator will have > 100% load AMPs.

This question matches the K&A because the candidate must have knowledge of the implications of overheating the main Generator rotor/field at power due to high excitation of the main generator based on a Grid disturbance and how to prevent the overheating of the field from occurring.

References:

NOP 2106.009 Turbine Generator Operations Rev. 81 Exhibit 2 Main Turbine Generator and Electrical Bus Limits (Verified reference updated 11/10/16); NOP 2106.009 Turbine Generator Operations Rev. 81 Attachment C Reactive Capability Curves (Verified reference updated 11/10/16); NOP OP 2106.009, Turbine Generator Operations, Rev. 81 Step 19.1 (Verified reference updated 11/10/16)

NOP 2106.009 Turbine Generator Operations Rev. 81 Attachment C Reactive Capability Curves should be provided as a Handout.

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Deleted Exhibit 2 as a reference and lowered Field amps to just above the normal 100% value. Deleted extra generator parameters on generator voltage and current. Updated distractor B and D analysis.

UNIT 2 MAIN GENERATOR LIMITS			
PARAMETER	**LOW LIMIT**	**HIGH LIMIT**	**REFERENCES**
KVolts	**21.7 KVolts**	**23.1 KVolts**	2203.012B (Gen Field Overvoltage) OPS-B2
MVARS	**-200 MVARS In**	**MVARS Out Per 2106.009, Att. C**	2203.012B (Gen Field Overvoltage) 2106.009
KAmps		**29 KAmps**	2203.012B (Gen Field Overvoltage) OPS-B2
EXCITER AMPS LIMIT		**5448 AMPS**	2203.012B (Exciter Field Overcurrent)
NEGATIVE SEQUENCE		**6%**	2203.012B (Neg Seq HI)
RECTIFIER TEMPERATURE		**140°C (285 °F)** (ALARM)	2203.012Y (Field Rectifier Overtemp) 2203.012B
GENERATOR OUTLET TEMPERATURE (SWC) T9754, 2TS-9779		**162.5 °F** **OR 72.5 °C** (ALARM)	2203.012B (Gen Prot Circuit Energized)
GENERATOR OUTLET TEMPERATURE SWC) 2TS-9747A/B/C		**171.5 °F** **OR 77.5 °C** (RUNBACK)	2203.012B (Gen Prot Circuit Energized) 2106.004, 2106.009
GENERATOR INLET PRESSURE (SWC) 2PS-9777B/C/D	**47 psig** (RUNBACK)		2203.012B (Gen Prot Circuit Energized) 2106.004, 2106.009
GENERATOR INLET FLOW (SWC) 2FS-9722A/B/C	**498 gpm** (RUNBACK)		2203.012B (Gen Prot Circuit Energized) 2106.004, 2106.009
STATOR WATER COOLING NOT IN-SERVICE		Trip MTG ≤ 60 minutes after Runback complete	2203.024 (Loss Of Turbine Load)
GENERATOR COLD GAS TEMPERATURE 2TIS-9760, T1630	**30 °C OR 86 °F**	**50 °C OR 122 °F**	2203.012Y (Machine Gas Temp High) OPS-B1
MAIN GENERATOR H2 PRESSURE High 2PS-9750 Low 2PS-9751	**58 psig** (ALARM) **50 psig** (With ACW aligned to H2 Cooler) **30 psig** (Log Limit) **5 psig** (Min Ops)	**62 PSIG** (ALARM)	2203.012Y (Machine Gas Pressure H/L) 2106.002 2106.009 ATT C OPS-B1

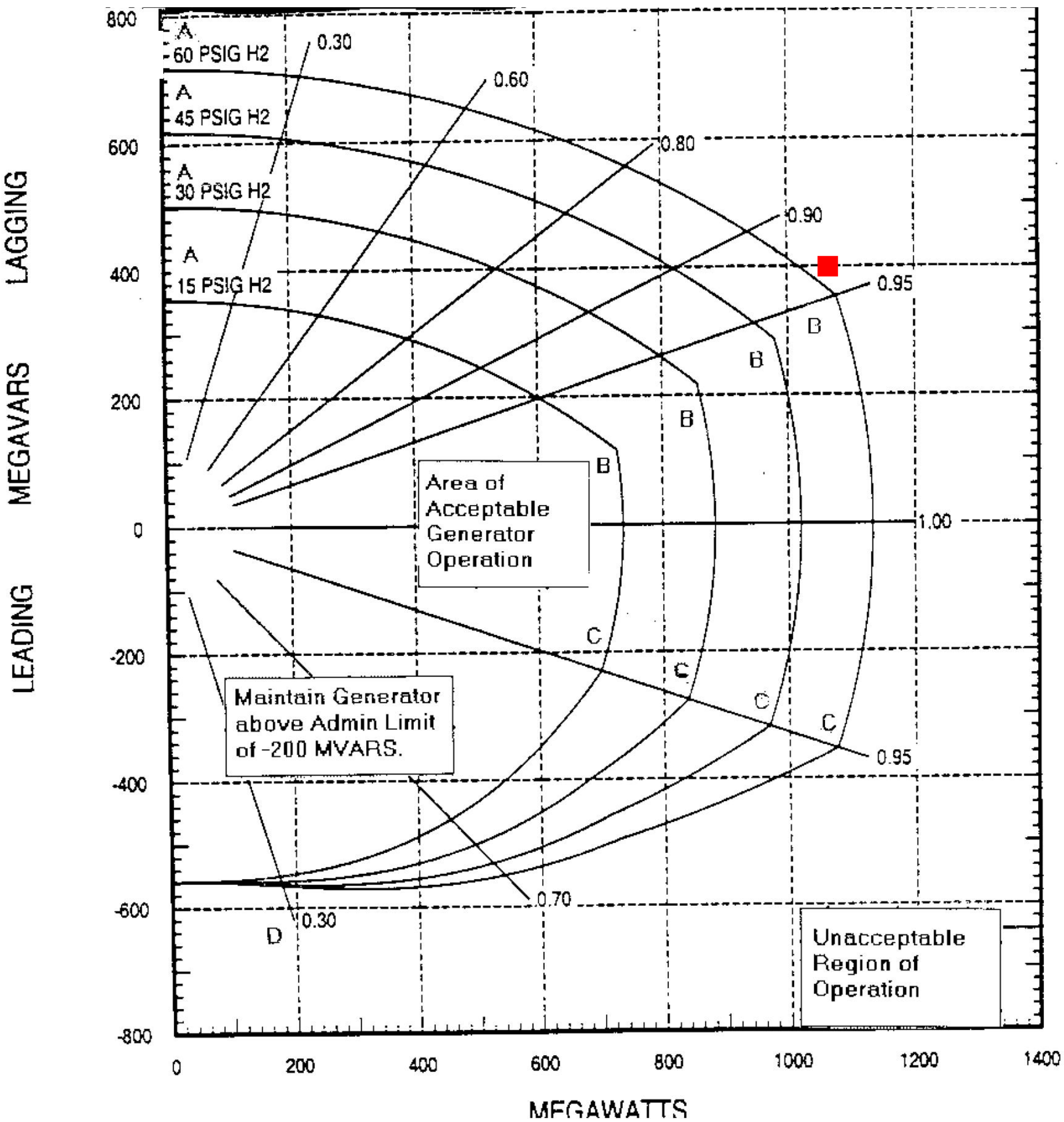
ASSOCIATED BUS LIMITS			
PARAMETER	**LOW LIMIT**	**HIGH LIMIT**	**REFERENCES**
2B5/2B6 VOLTAGE	**445 V** (DEGRADED) **436 V** (SUP 4)	**515 V** **505 V** (Modes 5&6)	2107.001 OPS-B2
161KV SU2 Regulated SPDS EST2R PMS E4013	**158 KV**	**170 KV**	2107.001 OPS-B2
22KV SU3 Regulated SPDS E2ST3R PMS E9664	**22.0 KV**	**23.1 KV**	2107.001 OPS-B2
2A3/2A4 Voltage	**4000 V** (LOG LIMIT) **3640 V** (SUP 4)	**4440 V** **4400 V** (Modes 5&6)	2107.001 OPS-B2
2H1/2H2 VOLTAGE	**6600 V**	**7240 V**	OPS-B2
2A1/2A2 VOLTAGE	**4000 V**	**4440 V**	OPS-B2

ATTACHMENT C

PAGE 1 OF 1

REACTIVE CAPABILITY CURVES

NOTE: VARS IN indicated by negative values, VARS OUT indicated by positive values.



CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
CURVE CD LIMITED BY ARMATURE CORE END HEATING

19.0 ADJUSTING GENERATOR VARS OR TERMINAL VOLTAGE

CAUTION

- Maintain MVARs less than -200 IN.
- VARS should be maintained within the Capability Curve and generating voltage limits (Attachment C), but should not exceed the Exciter Amps limit of 5448 Amps.

19.1 IF desired to adjust VARs, THEN perform the following:

19.1.1 Notify Unit 1 Control Room that generator VARS will be adjusted.

* 19.1.2 IF EITHER Unit 1 or Unit 2 EDG or AACG tied to grid, THEN monitor applicable Diesel Generator VARs while adjusting Generator VARs.

19.1.3 IF Generator VAR loading is excessive VARs "out" or "+" AND Voltage Regulator switch (2HS-9726) in AUTO, THEN lower Generator Excitation by bumping Generator Terminal Voltage Adjust handswitch (2HS-9723) COUNTER-CLOCKWISE until desired VARs loading obtained.

19.1.4 IF Generator VAR loading is excessive VARs "in" or "-" AND Voltage Regulator switch (2HS-9726) in AUTO, THEN raise Generator Excitation by bumping Generator Terminal Voltage Adjust handswitch (2HS-9723) CLOCKWISE until desired VARs loading obtained.

19.1.5 IF Generator VAR loading is excessive VARs "out" or "+" AND Voltage Regulator switch (2HS-9726) in MANUAL, THEN lower Generator Excitation by bumping Generator Field Voltage Adjust handswitch (2HS-9724) COUNTER CLOCKWISE until desired VARs loading obtained.

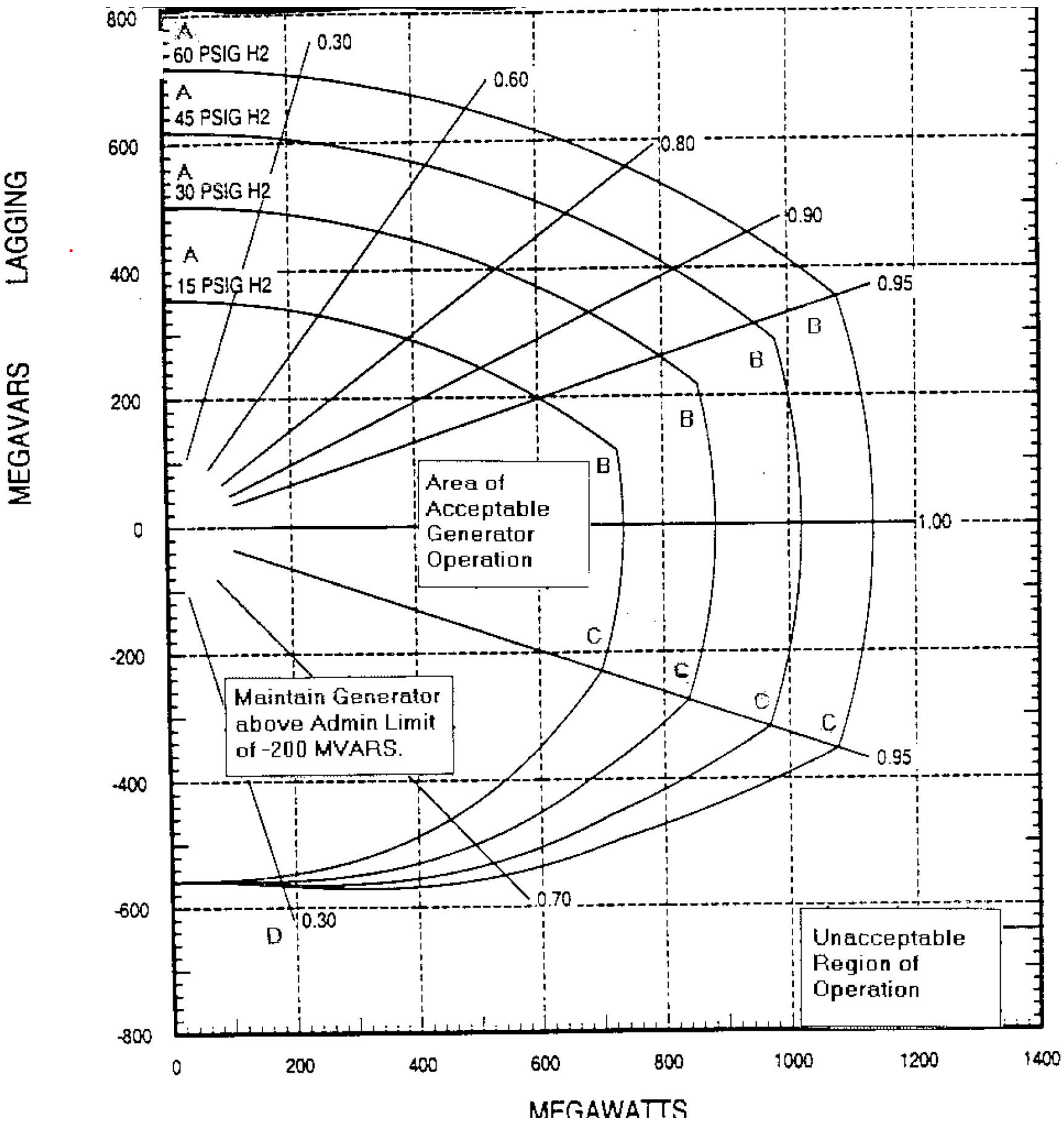
19.1.6 IF Generator VAR loading is excessive VARs "in" or "-" AND Voltage Regulator switch (2HS-9726) in MANUAL, THEN raise Generator Excitation by bumping Generator Field Voltage Adjust handswitch (2HS-9724) CLOCKWISE until desired VARs loading obtained.

19.1.7 WHEN adjustment of generator VARS complete, THEN notify Unit 1 Control Room.

Question 02 Reference Handout

ATTACHMENT C

REACTIVE CAPABILITY CURVES



CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
CURVE CD LIMITED BY ARMATURE CORE END HEATING

Question 03

Bank:	2333	**Rev:**	3	**Rev Date:**	1/12/2017	**2017 TEST QID #:**	3	**Author:**	Foster
Lic Level:	RO	**Difficulty:**	4	**Taxonomy:**	H	**Source:**	NRC Exam Bank 1936		
Search	00CE06K201	**10CFR55:**	41.7	**Safety Function**	4				
Title:	Loss of Feedwater	**System Number**	E06	**K/A**	EK2.1				
Tier:	1	**Group:**	1	**RO Imp:**	3.3	**SRO Imp:**	3.7	**L. Plan:**	A2LP-RO-ELOSF
OBJ	3								

Description: Knowledge of the interrelations between the (Loss of Feedwater) and the following: - Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Question:

Given the following:

* Unit 2 has tripped due to an Excess Steam Demand event.
* Containment Building pressure is 24 psia and trending up.
* Annunciator 2K01-B7 "LO RELAY TRIP" for 2A1 is in alarm.
* Neither Emergency Feedwater (EFW) pump is available.
* "A" Steam Generator (S/G) Level is 200" (WR) and trending down.
* "B" Steam Generator (S/G) Level is 21% (NR) and trending down slowly.
* SPTAs have been completed.
* CRS has entered OP-2202.009, Functional Recovery EOP.
* Standard Attachment 50, Condensate Pump Start, has been completed up to the step to start a Condensate Pump.

Based on these conditions, a Steam Generator feed source will be accomplished by starting Condensate Pump __________ from the Control Room.

A. 2P-2A

B. 2P-2B

C. 2P-2C

D. 2P-2D

Answer:

D. 2P-2D

Notes:

D is correct: For the given conditions, a SIAS, CCAS, EFAS, CSAS, and a MSIS/MFWIS are all present. The CSAS and MSIS signals, along with the MFWIS, will isolate feed and steam to/from the containment to minimize the energy release into containment. OP-2202.009, Functional Recovery EOP, would be entered due to 2 events in progress (ESD and Loss of Feedwater). Normally the safety EFW pumps would supply feedwater for RCS heat removal but they are unavailable. The non-safety Auxiliary FW pump is powered from non-vital 4160 Volt Bus 2A1 so it is unavailable. The Functional would direct over riding the MFWIS signal on both the "B" or "C" condensate pumps by opening the control power breakers. This action makes the "B" and "C" Condensate Pumps unavailable to be started from the Control Room. "A" Condensate Pump is not available due to 2A1 being locked out. So the only remaining option is to start "D" Condensate pump from the Control Room.

A is incorrect: "A" Condensate pump is unavailable because electrical bus 2A1 is locked out but plausible if power were available to 2A1

B is Incorrect: "B" Condensate pump is unavailable to be started form the control room because it has no control power. The action in the EOP to open "B" and "C" Condensate Pump control power breakers made "B" unavailable for start from the Control Room but plausible as the "B" Condensate pump could be started locally because the MFWIS lockout is overridden when control power is removed.

C is Incorrect: "C" Condensate Pump is unavailable because electrical bus 2A1 is locked out but plausible if the candidate does not recall the power supply for the "C" Condensate Pump.

This question matches the K&A because the candidate must have knowledge of the safety actuation signals associated with the given instrument indications and the lockouts/interlocks associated with the condensate pumps based on the safety actuations and the manual actions that will be taken in the EOPs to override the lockouts to allow satisfying the RCS Heat Removal Safety Function.

References:

AOP 2203.012A ANNUNCIATOR 2K01 CORRECTIVE ACTION for 2K01 B-7 LO RELAY TRIP for 2A1 REV.47 (Verified reference updated 11/10/16); EOP 2202.009 FRP REV.18 HR-2 step 38.E page 60 of 95; EOP-2202.009, Functional Recovery Tech Guide, Rev 018, HR-2 step 38 page 260 of 341 (Verified reference updated 11/10/16); EOP-2202.010, Standard Attachments, Rev 23, Attachment 57 step 2 page 180 (Verified reference updated 11/10/16); EOP-2202.010, Standard Attachments, Rev 23, Attachment 50 step 3 page 150 (Verified reference updated 11/10/16); STM 2-70, Engineered Safety Features Actuation System, Rev 19, section 2.2.4.1 pages 5 and 6 (Verified reference updated 11/10/16); STM 2-70, Engineered Safety Features Actuation System, Rev 19 MFWIS table page 55 (Verified reference updated 11/10/16).

Historical Comments:

Used on the 2014 NRC Exam
To be used on the 2017 NRC Exam: Altered distractor A to be more plausible based on OPS Representative feedback. Added last bullet to the stem to inform applicant that Standard Attachment 50 is in progress to allow a more logical flow to the question based on Facility Representative feedback.

REV. 1 based on NRC Chief Examiner Feedback BNC. Reworded last bullet to say "Standard Attachment 50, Condensate Pump Start, has been completed up to the step to start a Condensate pump".

REV. 2 based on NRC Chief Examiner Feedback BNC. Removed the second part of the stem and the second half of the distractors base on Chief Examiner feedback that the 2nd half is not needed to match the K/A.

Rev. 3 based on post submittal validation comments: Added " from the Control Room" to the end of the stem.

ANNUNCIATOR 2K01

B-7

2A1 BUS
L.O. RELAY TRIP

1.0 CAUSES

1.1 2A1 lockout relay tripped due to any of the following:

- Over-current (phase or ground) from SU2
- Over-current (phase or ground) from Unit Aux
- Over-current (phase or ground) from SU3
- Unit Aux breaker Failure Protection (0.2 sec TDC):
  - Generator Lockout exists
  - 2A1 Unit Aux feeder breaker closed
  - Either SU2 or SU3 Transformer 2A1 feeder breaker closed:
    - ♦ SU 2 to 2A1 (2A-111)
    - ♦ SU 3 to 2A1 (2A-113)

2.0 ACTION REQUIRED

2.1 Check status of 2A1 lockout relay located on Unit Aux to 2A1 Bkr (2A-112).

2.2 IF 2A1 lockout relay tripped,
THEN verify all Feeder breakers to 2A1 tripped:

- SU 2 to 2A1 (2A-111)
- Unit Aux to 2A1 (2A-112)
- SU 3 to 2A1 (2A-113)

2.3 IF Reactor tripped,
THEN GO TO Standard Post Trip Actions (2202.001).

2.4 IF Main Feed pump tripped,
THEN GO TO Loss of Main Feedwater Pump (2203.027).

2.5 IF Shutdown Cooling lost,
THEN GO TO Lower Mode Functional Recovery (2202.011).

(B-7 Continued on next page)

INSTRUCTIONS	CONTINGENCY ACTIONS
■**38. Establish a feed source to unisolated SG from at least one of the following (listed in preferred order):** A. EFW Pump 2P7B using 2202.010 Attachment 53, Recovery From Loss of Feed With 2P7B. B. EFW Pump 2P7A using 2202.010 Attachment 54, Recovery From Loss of Feed With 2P7A. C. AFW Pump 2P75 using 2202.010 Attachment 55, Recovery From Loss of Feed With 2P75. D. MFW Pumps using 2202.010 Attachment 56, Recovery From Loss of Feed With Main Feed Pumps. E. Condensate Pumps using 2202.010 Attachment 57, Recovery From Loss of Feed With Condensate Pumps.	Feed Sources 'A' through 'D' are not available. EFW not available due to initial conditions. AFW not available due to loss of electrical bus 2A1. MFW is not available due to CSAS/MSIS. So Condensate pumps are the only option. Condensate Pumps A and C are powered from electrical bus 2A1 so they are not available. Condensate Pump B has no control power so it cannot be operated from the control room.
***39. <u>IF</u> SG feed flow established from MFW pumps or Condensate pumps, <u>THEN</u> maintain Condenser Hotwell level greater than 38%.**	

EOP STEP: HR-2: SG Heat Sink with SIAS

38. **Establish a feed source to unisolated SG from at least one of the following (listed in preferred order):**

EPG STEP:

*32.
*33.

DEVIATION? Yes

BASIS FOR DEVIATION:

Since the EPG step is lacking any specific guidance on restoring Feedwater, this step lists an order of priority. Specific restoration guidance is given for each individual feedwater source in attachments (12, 13, 14, 15, 16).
The attachments for EFW and AFW address the EPG step that discusses actions to minimize the possibility of SG feed ring damage. The design at ANO-2 does not include a method for measuring MFW pump flow or Condensate pump flow of 150 gpm or less in order to minimize the possibility of SG feed ring damage.

SOURCE DOCUMENTS:

1. 2106.006, Emergency Feedwater Systems Operation.
2. M-2204, sheet 4, P&ID, Condensate and Feedwater.
3. ANO-2 EOP Setpoint Document, setpoints G.11, G.12 & F.2;
4. ANO-2 EOP Setpoint Document, setpoint G.13.
5. P5590, Clarify use of EFW pump 2P7A
6. 2203.012E, Annunciator 2K05 Corrective Action.
7. M-2202, sheet 4, P&ID, Main Steam.
8. 2107.001, Electrical System Operation.
9. 2202.010, Standard Attachments, Attachment 29, Startup XFMR #2 Usage.
   2202.010, Standard Attachments, Attachment 43, Startup XFMR #2 (ONE 4160v Non-Vital Bus) Usage.
10. 2203.012C, Annunciator 2K03 Corrective Action.
11. ER975122N202 MSIS and CSAS trips on 2P75.
12. 2202.010, Standard Attachments, Attachment 53, Recovery From Loss of Feed With 2P7B.
13. 2202.010, Standard Attachments, Attachment 54, Recovery From Loss of Feed With 2P7A.
14. 2202.010, Standard Attachments, Attachment 55, Recovery From Loss of Feed With 2P75.
15. 2202.010, Standard Attachments, Attachment 56, Recovery From Loss of Feed With Main Feed Pumps.
16. 2202.010, Standard Attachments, Attachment 57, Recovery From Loss of Feed With Condensate Pumps.

ATTACHMENT 57

RECOVERY FROM LOSS OF FEED WITH CONDENSATE PUMPS

1. IF adequate shutdown margin NOT established,
   THEN perform 2202.010 Attachment 35, Boric Acid Alignment for Cooldown.

2. IF at least one Condensate pump NOT running,
   THEN start a Condensate Pump using 2202.010 Attachment 50, Condensate Pump Start.

3. Verify the following valves closed:

   - MFW Reg Valve for 2P-1A (2CV-0748)
   - MFW Reg Valve Bypass for 2P-1A (2CV-0753)
   - MFW Reg Valve for 2P-1B (2CV-0740)
   - MFW Reg Valve Bypass for 2P-1B (2CV-0744)

*4. IF SG pressure greater than or equal to 600 psia,
   THEN lower SG pressure to less than 600 psia as follows:

   A. Reset Low SG pressure and Low PZR pressure setpoints during cooldown.

   B. Control SG depressurization with SDBCS Bypass valves or ADVs IAW Exhibit 3 of 2105.008, Steam Dump and Bypass Control Operations.

   C. IF possible,
      THEN maintain PZR level 29% to 80% [50% to 70%] using Charging and Letdown.

ATTACHMENT 50
CONDENSATE PUMP START

1. IF MSIS NOT reset
   AND it is desired,
   THEN reset MSIS using 2202.010 Attachment 14, MSIS Reset.

2. IF CSAS NOT reset
   AND it is desired,
   THEN reset CSAS by performing the following:

   A. Verify CNTMT Spray Termination criteria met:

      - CNTMT pressure less than 22.5 psia.
      - CNTMT temperature less than 140°F.
      - ALL available CNTMT Cooling fans running in Emergency Mode using 2202.010 Exhibit 9, ESFAS Actuation.
      - TSC determines CNTMT Spray NOT required for CNTMT Iodine removal.
      - CNTMT Spray NOT required for decay heat removal following RAS actuation.

   B. Reset CSAS using 2202.010 Attachment 45, CSAS Reset.

3. IF MSIS AND CSAS NOT reset,
   THEN locally open “DC CONTROL POWER” breaker in the following breaker cubicles:

   - “CONDENSATE PUMP 2P-2C” 2A106
   - “CONDENSATE PUMP 2P-2B” 2A205

4. Verify Hotwell level greater than 38%.

5. Verify the following Recirc valves closed
   AND Flow Indicating Controllers in MANUAL at 0% demand:

   - Condensate Pump Recirc 2CV-0662 (2FIC-0662)
   - Condensate Pump Recirc 2CV-0663 (2FIC-0663)
   - “A” MFP Recirc 2CV-0741 (2FIC-0735) (R/L then M/A and close)
   - “B” MFP Recirc 2CV-0749 (2FIC-0742) (R/L then M/A and close)

6. Locally verify the following valves closed:

   - “INLET TO 2PCV-4505 ISOL” (2CS-57)
   - “2PCV-4505 BYPASS” (2CS-59)

7. Locally open selected Condensate Pump Discharge valve 10 turns:

   - "2P-2A DISCHARGE" (2CS-2A)
   - "2P-2B DISCHARGE" (2CS-2B)
   - "2P-2C DISCHARGE" (2CS-2C)
   - "2P-2D DISCHARGE" (2CS-2D)

CAUTION

Maintaining Condensate pump discharge pressure greater than 753 psig for three minutes or greater will result in pump trip.

8. Start selected Condensate pump.

9. Locally open selected Condensate Pump Discharge valve:

   - "2P-2A DISCHARGE" (2CS-2A)
   - "2P-2B DISCHARGE" (2CS-2B)
   - "2P-2C DISCHARGE" (2CS-2C)
   - "2P-2D DISCHARGE" (2CS-2D)

Since control power for the 2P-2B Condensate pump was removed in Step 3 of Attachment 50 and there is no power on electrical bus 2A1 (A and C condensate pump power), this leaves only 2P-2D as the only condensate pump that can be started from the control room.

*10. Throttle Condensate Pump Recirc valves <u>OR</u> MFW Pump Recirc valves to maintain discharge pressure less than 700 psig.

11. Locally position the following valves as desired IAW 2106.016, Condensate and Feedwater Operations.

   - “INLET TO 2PCV-4505 ISOL” (2CS-57)
   - “2PCV-4505 BYPASS” (2CS-59)

The SIAS, along with the Containment Cooling Actuation Signal (CCAS), is actuated by a Low Pressurizer Pressure trip and/or High Containment Pressure. CCAS will be discussed shortly.

All components actuated by a SIAS are listed in the ESFAS Actuated Components table starting on page 60.

2.2.2 Containment Isolation Actuation Signal (CIAS)

The Containment Isolation Actuation Signal (CIAS) provides for Containment Isolation during a design LOCA or MSLB event to prevent the release of radioactive material.

The CIAS is actuated by a High Containment pressure. Certain CIAS valves are automatically actuated (closed) by a SIAS. This provides diverse signals for these valves for containment isolation.

All components actuated by a CIAS are listed in the ESFAS Actuated Components table starting on page 46.

2.2.3 Containment Cooling Actuation Signal (CCAS)

A Containment Cooling Actuation Signal (CCAS) provides for Containment Cooling during a design basis LOCA or MSLB event. This actuation limits post-accident containment pressure to design values during and following these events.

The CCAS is actuated by a Low Pressurizer Pressure signal, which actuates SIAS at the same time, and/or a High Containment Pressure.

All components actuated by a CCAS are listed in the ESFAS Actuated Components table starting on page 45.

2.2.4 Containment Spray Actuation Signal (CSAS)

A Containment Spray Actuation Signal (CSAS) provides for Containment Spray during a design basis LOCA or MSLB accident in order to remove heat and iodine from the containment area, and to hold containment temperature and pressure below design values.

The CSAS is initiated by a High-High Containment Pressure, in conjunction with a SIAS.

The operating mode of the CSAS is automatically changed by receipt of a Recirculation Actuation Signal (RAS). The RAS actuation is discussed in the following paragraphs. The operating mode of the CSAS will be discussed later.

All components actuated by a CSAS are listed in the ESFAS Actuated Components table starting on page 49.

2.2.4.1 Main Feedwater Isolation Signal (MFWIS)

Safety analysis performed for the steam generator replacement project, and subsequent power uprate, indicated that Containment Building (CB) pressure with Main Feedwater (MFW) and Main Steam (MS) line breaks in containment could exceed the pressure limits. (Exceeding the pressure limit could result from the higher power level, increased Steam Generator inventory, and addition of MS blowdown orifices.) Use of the Low Steam Generator pressure signal to initiate MSIS mitigation of the pressure rise would not be sufficient, per the safety analysis.

During 2R-14, a design change was installed to provide modifications to actuate equipment necessary to prevent exceeding the CB pressure limits. This was accomplished by using the Hi-Hi

Containment Pressure (CSAS) signal to terminate forced MFW flow, isolate MFW, and terminate MS flow. This termination and isolation is accomplished through generation of a Main Feedwater Isolation Signal (MFWIS).

CSAS and MSIS actuation relay contact combination were applied to actuate the components that isolate MFW and MS. This arrangement will terminate forced flow, such that the MFW isolation and/or backup valves can close, stop the Condensate Pumps, Heater Drain Pumps, and MFW pumps.

Once a MFWIS is actuated, regardless of which train, both condensate and feedwater trains will be secured.

All components actuated by a MFWIS are listed in the ESFAS Actuated Components table starting on page 55.

2.2.5 Main Steam Isolation Signal (MSIS)

The Main Steam Isolation Signal (MSIS) provides for Main Steam Isolation during design basis Steam Generator line breaks or Steam Generator tube ruptures.

The MSIS is actuated by a Low SG Pressure signal.

All components actuated by an MSIS are listed in the ESFAS Actuated Components table starting on page 51.

2.2.5.1 2R-14 MSIS Changes

As documented by CR-2-99-0282, Tech Spec LCO Table 3.3.3 implies MFW and MS isolation by MSIS are redundant. Contrary to the implication, the existing design required both trains of MSIS to isolate MFW. That is, the actuation of MSIS would only secure its respective train of feedwater and condensate. Therefore, both actuations would have to occur in order to completely secure MFW.

In addition, single failures existed that could result in MSIS relay de-energization. And under that design, a single MSIS relay failure could result in the tripping of a MFP. Although the MSIS relays fail safe (tripped), a “Loss of MFW Event” would result.

It was determined that the Condensate, Heater Drain, and MFW pumps must be coasting down before the safety related redundant isolation and backup MFW valves could completely close. The valves were not designed to close against normal forced feedwater flow. (They are only considered redundant for upstream water retention after forced flow differential decrease.)

To provide for Tech Spec compliance, the MSIS trip contacts were included in the generation of a MFWIS. By providing the MSIS input, both trains of feedwater and condensate will be secured upon actuation of either train of MSIS. Additionally, the change installed two separate contacts, from two separate relays, in series. Thus, failure of a single relay or contact can no longer cause a loss of main feedwater event.

2.2.6 Recirculation Actuation Signal (RAS)

The Recirculation Actuation Signal (RAS) provides for recirculation of core cooling water within the reactor building during a design basis LOCA event. This actuation continues long term post-accident core cooling, following a loss of coolant accident.

MFWIS - MAIN FEEDWATER ISOLATION SIGNAL				
Component	**Designation**	**Actuated Position**	**Time Delay**	**Relay Number**
Condensate Pumps	2P-2A	Off		MFWIS1 or MFWIS2
	2P-2B	Off		
	2P-2C	Off		
	2P-2D	Off		
Heater Drain pumps	2P-8A	Off		
	2P-8B	Off		
AFW pump	2P-75	Off		

MFWIS1 relay is located in 2P-2C Condensate Pump breaker cubicle 2A-106.
MFWIS2 relay is located in 2P-2B Condensate Pump breaker cubicle 2A-205.

ORIGINAL BANK QUESTION 1936 WAS USED TO DEVELOP 2017 NRC EXAM QID#03 (Used on the 2014 NRC Exam)

Questions For All QID In Exam Bank

Bank:	1936	**Rev:**	4	**Rev Date:**	2/21/2014 8:01:00	**QID #:**	12	**Author:**	foster
Lic Level:	R	**Difficulty:**	4	**Taxonomy:**	H	**Source:**	NEW		
Search	00CE06K201	**10CFR55:**	41.7 / 45.7	**Safety Function**	4				
System Title:	Loss of Feedwater	**System Number**	E06	**K/A**	EK2.1				
Tier:	1	**Group:**	1	**RO Imp:**	3.3	**SRO Imp:**	3.7	**L. Plan:**	A2LP-RO-ELOSF
OBJ	3								

Description: Knowledge of the interrelations between the (Loss of Feedwater) and the following: - Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Question:

Consider the following:

- Unit 2 has tripped due to high Containment Building pressure
- Containment Building pressure is 24 psia and trending up
- Annunciator 2K01 B-7 "LO RELAY TRIP" for 2A1 is in alarm
- Neither Emergency Feedwater (EFW) pump is available
- "A" Steam Generator (S/G) Level is 200" (WR) and trending down
- "B" Steam Generator (S/G) Level is 21% (NR) and trending down slowly
- SPTAs have been completed
- CRS has entered OP-2202.009, Functional Recovery EOP

RCS heat removal would be accomplished from the Control Room by starting ______ Condensate pump because _______________.

A. 2P-2A; there is no control power for the "B" pump

B. 2P-2B; the MFWIS signal cannot be overridden for the "A" pump

C. 2P-2C; the MFWIS signal cannot be overridden for the "D" pump

D. 2P-2D; there is no control power for the "B" pump

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☒

Answer:

D. Correct

Notes:

D. Correct: For the given conditions, a SIAS, CCAS, EFAS, CSAS, and a MSIS are all present. The CSAS and MSIS signals, along with the MFWIS, will isolate feed and steam to/from the containment to minimize the energ release into containment. OP-2202.009, Functional Recovery EOP, would be entered due to 2 events in progress (ESD and Loss of Feedwater) Normally the EFW pumps would supply feedwater for RCS heat removal but they are unavailable. AFW pump is powered from 2A1 is unavailable. The Functional would direct over riding the MFWIS signal on either the "B" or "C" condensate pumps by opening the control power breakers. This action makes the "B" and "C" Condensate Pumps unavailable to be started from the Control Room. "A" Condensate Pump is not available due to 2A1 being locked out. So the only remaining option is to start "D" Condensate pump from the Control Room.
A. Incorrect: "A" Condensate pump is unavailable because 2A1 is locked out. Second half of the answer is correct.
B. Incorrect: "B" Condensate pump is unavailable because it has not control power. The action in the EOP to open "B" and "C" Condensate Pump control power breakers made "B" unavailable for start from the Control Room.
C. Incorrect: "C" Condensate Pump is unavailable because 2A1 is locked out.

Question 04

Bank:	2334	**Rev:**	3	**Rev Date:**	1/13/2017	**2017 TEST QID #:**	4	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	00CE05K301	**10CFR55:**	41.5	**Safety Function**	4				
Title:	Excess Steam Demand	**System Number**	E05	**K/A**	EK3.1				
Tier:	1	**Group:**	1	**RO Imp:**	3.6	**SRO Imp:**	3.8	**L. Plan:**	A2LP-RO-ESFAS
OBJ	2								

Description: Knowledge of the reasons for the following responses as they apply to the (Excess Steam Demand): - 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.

Question:

From 100% Power, which of the following describes the primary reason for the automatic initiation of a SIAS during an Excess Steam Demand event?

A. To inject a large amount of water inventory into the RCS to overcome the rapid drop in the Pressurizer level.

B. To restore RCS Pressure back to the normal operating pressure band to overcome the rapid drop in RCS Pressure.

C. To ensure continued long term cooling of the Reactor core after the Excess Steam Demand event has terminated due to loss of SG inventory.

D. To inject a large amount of boric acid to the core to overcome the positive reactivity added due to the negative MTC to ensure adequate SDM is maintained.

Answer:

D. To inject a large amount of boric acid to the core to overcome the positive reactivity added due to the negative MTC to ensure adequate SDM is maintained.

Notes:

D is correct due to negative reactivity that will be added to the core during the rapid cooldown due to the large negative MTC required for 0% Hot Standby cooldown at EOL. The accident analysis for the worst case negative MTC for a PWR is at the End of Life conditions when boric acid concentration is the lowest during a rapid cooldown. A highly concentrated boric acid solution will be needed to maintain the required SDM allowed by Technical Specifications.

A is incorrect as no loss of inventory has occurred in the RCS but plausible because this is the primary reason for an SIAS during a LOCA event where there is a loss of inventory.

B is incorrect as pressure should not be allowed to rise significantly after the excess steam demand event has terminated to prevent Pressurized Thermal Shock but plausible because maintaining pressure control during a LOCA event will assist in maintaining forced circulation to allow for a quicker cooldown.

C is incorrect because no inventory has been lost from the RCS therefore adding more inventory may not even go through the core since there is no break but plausible because cooling the core is another major reason for SIAS during a LOCA event.

This question matches the K&A because the candidate must have knowledge of the reasons for changing the chemistry makeup of the RCS during an Excess Steam Demand event due to the reactivity feedback on a lowering RCS temperature affecting the SDM TS limits.

References:

TS 3/4.1.1 Boration Control Basis for SDM (Verified reference updated 11/10/16);
STM 2-70 Rev. 19 ESFAS Section 2.2.1 (Verified reference updated 11/10/16);
ULD-2-SYS-02 ANO-2 HPSI Section 1.0 and 2.0;
EOP-2202.005 Excess Steam Demand Rev.15 Step 37 (Verified reference updated 11/10/16);
EOP-2202.005 Excess Steam Demand TG Rev.15 Step 37 (Verified reference updated 11/10/16);
EOP-2202.005 Excess Steam Demand TG Rev. 15 Step 47 (Verified reference updated 11/10/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Reworded distractor A to say "To inject a large amount of borated water into the RCS to overcome the rapid drop in the Pressurizer level".

REV. 2 based on NRC Chief Examiner Feedback BNC. Removed the second part of the stem that referenced Hot Zero Power EOL core conditions and added " From 100% Power" to the beginning of the stem.

Rev. 3 based on post submittal validation comments: in Answer 'D', changed the words "a lot" to " a large amount" to be consistent with the wording in Distractor 'A'

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN (SDM) ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SDM requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{AVE}. The most restrictive condition occurs at EOL, with T_{AVE} at no load operating temperature, and is associated with a postulated steam line break accident, and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SDM is required to control the reactivity transient. Accordingly, the SDM requirement is based upon this limiting condition and is consistent with SAR safety analysis assumptions. With $T_{AVE} \leq 200$ °F, the reactivity transients resulting from any postulated accident are minimal and the SDM provides adequate protection.

In Modes 1 and 2, SDM verification performed as a result of an immovable CEA that is fully inserted in the core need only account for the most reactive CEA assumed to fail to insert upon a reactor trip, with no additional penalty for the immovable CEA since it has no "withdrawn worth." Likewise, SDM verification when a single CEA is found immovable at 50" withdrawn need only account for the reactivity worth of the most reactive CEA and the reactivity worth of the 50" of the immovable CEA that is assumed to fail to insert upon a reactor trip.

SR 4.1.1.1.1.a and SR 4.1.1.2.a require SDM determination within one hour and once every 12 hours thereafter following identification of one or more inoperable CEAs. Because CEAs are not required to be OPERABLE in Modes 3, 4, and 5 (reference TSs 3.1.3.1, 3.1.3.4, 3.1.3.5, and 3.1.3.6), these SRs would only be applicable in Modes 1 and 2. Such application is consistent with the ITSs (NUREG 1430, Rev. 4), which contain no CEA-related SRs within the individual SDM TSs. However, the ITS Definition of SDM provides a relationship between SDM and CEAs in that the assumption of an additional CEA of highest reactivity worth being fully withdrawn need not be applied in the SDM determination if all CEAs have been verified fully inserted by two independent means. Conversely, the ITS SDM Definition indicates that CEA penalty (the actual non-inserted rod worth) is applicable in Modes 3, 4, and 5 for any CEA which has not been verified to be fully inserted by two independent means. Therefore, the ANO-2 SDM calculation conservatively considers the following for any CEA that has not been verified to be fully inserted by two independent means during operation in Modes 3, 4, and 5:

1. If the CEA(s) position is known (i.e., height verified by two independent means), the reactivity worth of the portion of the CEA not inserted must be accounted for in the SDM determination.
2. For any CEA position that cannot be verified by two independent means, the full reactivity worth of the CEA must be accounted for in the SDM determination.

Conservatively accounting for these potential reactivity penalties in the ANO-2 SDM determination is beyond the SR applicability of the related ANO-2 SDM TSs, but consistent with the ITS application of SDM requirements.

logic matrix relays when, de-energized, cause the associated contacts to open in each of four Trip Paths. The opening of these contacts breaks the 12v DC circuit in the trip paths causing each respective Trip Path Relay to de-energize. The Trip Path Relays control the Solid State Relays (SSRs). The SSRs function to route 28v DC power to the ESFAS Auxiliary Relays. When the Trip Path Relays are de-energized, each associated SSR will deenergize, causing their associated Auxiliary Relays to de-energize.

When the power supply to the Auxiliary Relays is interrupted the ESF Trip Function associated with that Trip Path Relay will actuate, causing the selected ESFAS components (pumps, valves, etc.) to perform their design function.

Manual actuation, or Remote Trip, switches are provided for each ESFAS function. These manual Actuations completely bypass the actuation logic mentioned above.

The Plant Protection System (PPS) cabinet is designated 2C23. The PPS cabinet assembly consists the following components:

- PPS Local Status Panel (LSP)
- Bistable Control Panels (BCP)
- Matrix Test Modules (MTM)
- Relay Card Rack Assemblies
- Cabinet Cooler Assemblies
- Actuation Reset Panels
- AC Vital Power Panels
- Power Supplies
- Trip Path Relay Panels
- PPS Remote Control Modules (RCM)

The remaining portions of the Emergency Safety Features Actuation System is housed in the Auxiliary Relay Cabinets 2C39 and 2C40. These cabinets represent the two isolated trains of ESF system equipment. 2C39 is the designated Red train cabinet and 2C40 is the Green train cabinet.

Each of the ESF Systems (2C39 and 2C40) are composed of two redundant load groups. Load Group I is composed of Red train components while Load Group II are Green train loads. The components and their instrumentation and controls for each Load Group are physically and electrically separate. Each Load Group is capable of providing 100% of the necessary capacity to carry out the designed system function during design basis events.

2.2 Actuations

The following section discusses each of the ESFAS actuations that are produced in the ESFAS Auxiliary Relay cabinets 2C39 & 2C40. A summary of actuation setpoint and purpose is shown in the table on page 8.

2.2.1 Safety Injection Actuation Signal (SIAS)

The Safety Injection Actuation Signal, referred to as SIAS, provides for Safety Injection during design basis Loss of Coolant Accidents (LOCA), Main Steam Line Break (MSLB) events, or Steam Generator Tube Ruptures (SGTR). This actuation provides cooling to limit core damage and assure adequate shutdown margin, regardless of temperature.

1.0 SYSTEM DEFINITION

1.1 PRIMARY SYSTEM FUNCTION

The ANO-2 High Pressure Safety Injection System (HPSI) is part of the Emergency Core Cooling System (ECCS) which provides short term and long term emergency core cooling and core reactivity control following a Loss of Coolant Accident (LOCA). The HPSI System is also capable of injecting borated water following a Main Steam Line Break (MSLB). **(REF. 3, 49, 51)**

1.2 SYSTEM BOUNDARIES

The principal components of the HPSI System are:

A. HPSI Pumps 2P89A, 2P89B and 2P89C

B. HPSI flow restricting orifices 2F0-5101, 2F0-5102

C. HPSI Hot Leg Injection Orifices 2F0-5103, 2F0-5104

D. HPSI Hot Leg Injection Orifice Bypass Valves 2CV-5103-1, 2CV-5104-2

E. HPSI Injection Valves 2CV-5015-1, 2CV-5035-1, 2CV-5055-1, 2CV-5075-1, 2CV-5016-2, 2CV-5036-2, 2CV-5056-2, 2CV-5076-2

F. HPSI Injection Balance Valves 2SI-68, 2SI-69, 2SI-70, 2SI-71, 2SI-72, 2SI-73, 2SI-74, and 2SI-75

G. HPSI Hot Leg Injection Valves 2CV-5101-1 and 2CV-5102-2

H. HPSI Pump Recirculation Isolation Valves 2CV-5126-1, 2CV-5127-1 and 2CV-5128-1

I. HPSI to SIT Drain Header Isolation Valves 2CV-5105-1, 2CV-5106-2

HPSI System piping and interfaces with other systems are shown in system drawings. The HPSI System IEEE designator is "BQ". Attachment 1 shows a simplified sketch of the HPSI System. **(REF. 24)**

2.0 SYSTEM FUNCTIONS

2.1 SAFETY RELATED FUNCTIONS

2.1.1 Provide short term cooling during the initial phase of a LOCA by injecting borated water from the Refueling Water Tank (RWT) into the core to limit cladding metal-water reaction to a negligible amount and prevent fuel and cladding damage which could interfere with continued effective cooling. **(REF. 1, 2, 3, 48, 49, 51)**

2.1.2 Maintain core subcriticality following a LOCA by providing enough boric acid in the injected water as a neutron absorber to maintain the reactor core in a subcritical condition. **(REF. 3, 35, 48, 49, 51)**

2.1.3 Provide long term cooling following a LOCA, in conjunction with containment cooling and spray by recirculating containment sump water through the core for extended periods of time. During recirculation, the core must be cooled by boil-off of water in the reactor vessel. For some break sizes and locations (large cold leg break), this may result in a long term buildup of solids (boron precipitation) in the reactor vessel. Hot Leg Injection must be initiated to flush solids from the core region if RCS Margin to Saturation has remained less than 30°F for more than two hours and the LOCA is not due to the ECCS Vent Isolation Valves being opened. **(REF. 2, 3, 18, 49, 51)**

2.1.4 Provide increased shutdown margin during a Main Steam Line Break (MSLB) by injecting borated water from the RWT into the Reactor Coolant System, minimizing potential core criticality and return to power. **(REF. 3, 35, 46, 48, 49, 51)**

2.2 REGULATORY/SAFETY SIGNIFICANT FUNCTIONS

2.2.1 Providing a means for RCS inventory control during recovery from SIAS. (REF. 18)

2.2.2 Provide a backup coolant inventory supply during a Loss of Decay Heat Removal (DHR) Event. (Loss of Shutdown Cooling.) See Loss of DHR Topical ULD, ULD-0-TOP-09, for additional information. **(REF. 15, 50)**

2.2.3 Provide an alternate boration path in modes 5 and 6. **(REF. 55**)

2.2.4 Provide once through cooling via the ECCS vents. **(REF. 18, 56)**

2.3 NON-SAFETY FUNCTIONS

2.3.1 Providing a means for filling the SITs. **(REF. 16)**

2.3.2 Providing an alternate means for RCS fill operation. **(REF. 17)**

3.0 <u>DESIGN REQUIREMENTS/COMMITMENTS</u>

3.1 PERFORMANCE REQUIREMENTS

General Design Criterion (GDC) 35 "Emergency Core Cooling" requires a system be designed such that fuel and clad damage that could interfere with continued core cooling be prevented and clad metal-water reaction be limited following a loss of coolant accident. **(REF. 1)**

The HPSI System must be designed, in conjunction with the remainder of the ECCS (Safety Injection Tanks (SITs) and Low Pressure Safety Injection System (LPSI)) to provide the flowrate at the RCS pressure required to satisfy the assumptions used in LOCA analyses. These analyses include overall ECCS response and include short term and long term cooling for all sizes of LOCAs. To ensure injection capability for small break LOCAs, HPSI pump developed head must also exceed the setpoint pressure of the secondary (steam generator) safety valves. The analyses, including the Reload Analysis Report and Groundrules, must verify that overall ECCS performance meets the peak cladding temperature, cladding oxidation, hydrogen generation, coolable geometry and long term cooling acceptance criteria of 10 CFR 50.46. **(REF. 1, 2, 3, 48, 49, 51, 57, 70)**

INSTRUCTIONS	CONTINGENCY ACTIONS
***37. IF SIAS actuated, THEN reset SIAS as follows:**	
A. Check adequate shutdown margin established using 2202.010 Attachment 28, Boric Acid Required for Shutdown Margin.	A. Perform 2202.010 Attachment 35, Boric Acid Alignment for Cooldown.
B. Reset SIAS using 2202.010 Attachment 13, SIAS Reset.	B. IF SIAS can NOT be reset, THEN **GO TO** Step 39.
C. Restore SIAS actuated components and system as needed.	
38. WHEN PZR level greater than 29% [50%], THEN establish Letdown using 2104.002, Chemical and Volume Control.	
■ **39. IF RCS water solid conditions indicated by ALL of the following:** • PZR level greater than 80% [70%]. • RVLMS LVL 01 indicates WET. • RCS subcooled with exaggerated or severe pressure response associated with RCS inventory or temperature changes. **THEN perform 2202.010 Attachment 34, RCS Water Solid Operations.**	

EOP STEP:

*37. **IF SIAS actuated,**
THEN reset SIAS as follows:

EPG STEP:

*42.

DEVIATION? Yes

BASIS FOR DEVIATION:

SIAS is required to be reset prior to restoring certain non-vital systems, such as letdown (1), therefore this EPG step has been moved up in the EOP prior to the step to restore letdown. If SIAS cannot be reset, then the operator bypasses this step and the step to restore letdown.

Resetting the SIAS signal will realign several boric acid valves to their pre-SIAS positions. This is not desirable, since shutdown margin must be assured prior to and during cooldown (5). If adequate shutdown margin has not been established, then this step verifies the boric acid alignment to the Charging pumps to be in the SIAS configuration (2) using a standard attachment (4).

If SIAS is to be reset, then the operators use a standard attachment (3) that allows various components to be returned to their normal or standby configurations.

SOURCE DOCUMENTS:

1. M-2417, sheet 1 of 8, Functional Description and Logic Diagram Chemical and Volume Control System.
2. M-2417, sheets 3-6 of 8, Functional Description and Logic Diagram Chemical and Volume Control system.
3. 2202.010, Standard Attachments, Attachment 13, SIAS Reset.
4. 2202.010, Standard Attachments, Attachment 35, Boric Acid Alignment for Cooldown.
5. 2202.010. Standard Attachments, Attachment 28, Boric Acid Required for Shutdown Margin

EOP STEP:

*47. **Maintain shutdown margin using 2202.010 Attachment 28, Boric Acid Required for Shutdown Margin.**

EPG STEP:

*36.

DEVIATION? Yes

BASIS FOR DEVIATION:

This EPG step was moved ahead of the steps for determination of performing an cooldown. Establishing and maintaining shutdown margin is required whether or not the plant is cooled down.

A standard attachment is referenced in the EOP for maintaining shutdown margin (1). This attachment will be utilized to determine the required SDM for either a cooldown, or maintenance of existing conditions using normal procedures (2). Should additional boration be needed, the operators will continue borating (which is or should be in progress due to SIAS) until the necessary boron concentration is reached. This standard attachment provides the operator instruction to secure boration once the required SDM has been achieve.

SOURCE DOCUMENTS:

1. 2202.010, Standard Attachments, Attachment 28, Boric Acid Required for Shutdown Margin.
2. 2103.015, Reactivity Balance Calculation;
   2103.004, Soluble Poison Concentration Control;
   2104.003, Chemical Addition.

Question 05

Bank:	2335	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	5	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC Exam Bank 0456		
Search	000027A218	**10CFR55:**	41.10	**Safety Function**	3				
Title:	Pressurizer Pressure Control (PZR PCS) Malfunction	**System Number**	027	**K/A**	AA2.18				
Tier:	1	**Group:**	1	**RO Imp:**	3.4	**SRO Imp:**	3.5	**L. Plan:**	A2LP-RO-APZRM
OBJ	2								

Description: Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: - Operable control channel

Question:

Given the following conditions:

* The plant is at 100% power.
* PZR PRESS CHANNEL SELECT Switch 2HS-4626 is selected to CH "A".
* PZR Pressure Transmitter 2PT-4626A fails high.
* Annunciator "Control CH 1/2 Pressure HI/LO" comes into alarm.

Which of the following actions is procedurally required to be taken FIRST to restore RCS pressure control in accordance with OP-2203.028, PZR Systems Malfunction?

A. Place PZR PRESSURE CONTROLLER 2PIC-4626A in manual to restore RCS pressure control.

B. Take manual control of PZR proportional heaters and spray valves to restore RCS pressure control.

C. Verify PZR Pressure Transmitter 2PT-4626B NOT failed and transfer 2HS-4626 "PZR PRESS CHANNEL SELECT" to CH "B".

D. Energize all PZR backup heaters to restore RCS pressure that has been lost due to the excess amount of PZR spray flow.

Answer:

C. Verify PZR Pressure Transmitter 2PT-4626B NOT failed and transfer 2HS-4626 "PZR PRESS CHANNEL SELECT" to CH "B".

Notes:

Answer C is correct based on the direction given in OP-2203.028 REV. 13 PZR Systems Malfunction, Contingency Action Step 6. A & C. After a determination of a good operable Pressure input, RCS pressure control can be quickly restored by selecting the unaffected operable control channel.

A is incorrect because it is a slower and more complicated method but is plausible as operators are trained to take manual control of components not working in AUTO. The AOP does not direct this action for a single input channel failure but does direct this action if the controller fails.

B is incorrect for a single channel failure but plausible and directed by the AOP if both Pressurizer control channels inputs fail.

D is incorrect but plausible as this would be an next action to take as directed by the AOP after the unaffected channel was selected if pressure continued to lower.

This question matches the K&A because the candidate must interpret and determine the operable control channel and then

understand the reason for the action in the PZR Systems Malfunction AOP.

References:

AOP-2203.028 REV. 13 PZR Systems Malfunction, Entry Conditions and Step 1 (Verified reference updated 11/10/16); AOP-2203.028 REV. 13 PZR Systems Malfunction, Contingency Action Step 6. A & C (Verified reference updated 11/10/16); AOP-2203.028 REV. 13 PZR Systems Malfunction TG Step 6 (Verified reference updated 11/10/16).

Historical Comments:

NRC Exam Bank 0456 was used on the 2005 NRC Exam
To be used on the 2017 NRC Exam but altered the order of the correct answer from the last time the QID was given. Also altered the stem to ask what is procedurally required FIRST.

REV. 1 based on NRC Chief Examiner Feedback BNC. Added "proportional" to distractor B, removed the word "automatically" from distracter D and made into an action,

PZR SYSTEMS MALFUNCTION

PURPOSE

This procedure provides actions for a PZR System Malfunction.

ENTRY CONDITIONS

ANY of the following conditions exist:

1. PZR level deviating from setpoint during steady state operations.
2. RCS pressure deviating from setpoint during steady state operations.
3. "CNTRL CH 1/2 PRESSURE HI/LO" annunciator (2K10-E6/E7) in alarm.
4. "CNTRL CH 1/2 LEVEL LO" annunciator (2K10-G6/G7) in alarm.
5. "CNTRL CH 1/2 LEVEL HI" annunciator (2K10-J6/J7) in alarm.
6. PZR Spray valve failed open or closed.

EXIT CONDITIONS

ANY of the following conditions exists:

1. Unaffected PZR level control channel has been selected and Letdown restored.
2. Unaffected PZR pressure control channel has been selected and RCS pressure restored to 2025 psia to 2275 psia.

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

Steps marked with (*) are continuous action steps.

1. Check the following criteria:	
A. IF any PZR spray valve failed open, THEN **GO TO** Step 2.	
B. IF any PZR spray valve failed closed, THEN **GO TO** Step 4.	
C. Check "RRS TROUBLE" annunciator (2K10-H2) clear.	C. **GO TO** Step 5.
D. Check "CNTRL CH 1/2 PRESSURE HI/LO" annunciators (2K10-E6/E7) clear.	D. **GO TO** Step 6.
E. Check the following PZR level annunciators clear: • "CNTRL CH 1/2 LEVEL LO" 2K10-G6/G7 • "CNTRL CH 1/2 LEVEL HI" 2K10-J6/J7	E. **GO TO** Step 7.

INSTRUCTIONS	CONTINGENCY ACTIONS
6. Check "CNTRL CH 1/2 PRESSURE HI/LO" annunciators (2K10-E6/E7) clear.	**6. Perform the following:** A. Compare PZR pressure instruments to determine affected channel. B. IF BOTH PZR Pressure Control channels failed, THEN perform the following: 1) Manually control PZR Heaters and Spray Valves to restore RCS pressure 2025 psia to 2275 psia. 2) Place the SDBCS in required contingency alignment using Attachment A, of this procedure. C. IF only the selected control channel affected, THEN perform the following: 1) Place PZR Pressure Channel Select switch (2HS-4626) to the unaffected channel. **NOTE** **Proportional Heater controller is located on Page 3.** 2) Restore PZR Spray valves and Heater control to automatic. D. IF Pressure Control Channel 1 failed, THEN place the SDBCS in required contingency alignment using Attachment A, of this procedure. E. IF Pressure Control Channel 2 failed, THEN place the SDBCS in required contingency alignment using Attachment A, of this procedure.

AOP STEP:

6. **Check "CNTRL CH 1/2 PRESSURE HI/LO" annunciators (2K10-E6/E7) clear.**

BASIS:

Annunciators provide the Operator with a quick, easy method to determine if Pressurizer Pressure Controls are functioning properly. An annunciator in alarm indicates possible malfunction or failure within the system. Contingency actions determine the affected control channel, and stabilize the plant by placing the control system on the unaffected channel. RCS pressure band established in the contingency actions is consistent with Technical Specification 3.2.8.

If both channels failed, directions are provided to control PZR pressure and align the SDBCS using Attachment A (SDBCS Operations) of this procedure.

If the selected control channel fails, the other channel is selected.

Attachment A (SDBCS Operations) provides actions to take in the event of a single channel failure. If both channels fail, then all actions of Attachment A will be performed.

PZR control channel pressure is an input to SDBCS. Channel 1 is an input to the main calculator, and Channel 2 to the permissive calculator. A failed channel will provide an inappropriate bias to the setpoint calculator; therefore, contingencies are provided not only for stabilizing RCS pressure but also for addressing the SDBCS calculator.

The purpose of the permissive calculator is to prevent inadvertent spurious opening of the SDBCS valves. If the permissive setpoint is affected by the PZR pressure control system malfunction, a maximum of one permissive handswitch is allowed in manual. This limits excessive steam flow to 11.5%, due to inadvertent SDBCS valve opening.

SOURCE DOCUMENTS:

1 - ANO-2 Technical Specification 3.2.8.
2 - 2103.005, Pressurizer Operations.
3 - 2105.008, Steam Dump and Bypass Control System Operation.
4 - 2203.012J, Annunciator 2K10 Corrective Actions.
5 - STM 2-03, Reactor Coolant System.
6 - STM 2-23, Steam Dump and Bypass Control System.

ORIGINAL BANK QUESTION 456 WAS USED TO DEVELOP 2017 NRC EXAM QID#5 (Used on the 2005 NRC Exam)

Questions For All QID In Exam Bank

Bank:	0456	**Rev:**	0	**Rev Date:**	10/5/2004	**QID #:**		**Author:**	COBLE
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	A	**Source:**	NEW		
Search		**10CFR55:**		**Safety Function**					
System Title:	PZR PRESSURE CONTROL MALFUNCTION	**System Number**	027	**K/A**	AK3.02				
Tier:	1	**Group:**	1	**RO Imp:**	2.9	**SRO Imp:**	3.0	**L. Plan:**	
OBJ									

Description: Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunctions: Verification of alternate transmitter and/or plant computer prior to shifting flow chart transmitters.

Question:

Given the following conditions:

* The plant is at 100% power.
* PZR Pressure Control Channel Select Switch 2HS-4626 is selected to Channel 4626A
* PZR Pressure Transmitter 2PT-4626A fails high.

Which ONE of the following actions should be taken to restore RCS pressure in accordance with AOP 2203.028, PZR Systems Malfunction?

A. Take manual control of PZR Pressure Controller 2PIC-4626A to restore RCS pressure control.

B. Verify PZR Pressure Transmitter 2PT-4626B NOT failed and transfer 2HS-4626 to Channel 4626B.

C. Take manual control of PZR heaters and spray valves to restore RCS pressure control.

D. Verify all PZR backup heaters energize automatically to restore RCS pressure that has been lost due to the excess amount of PZR spray flow.

QID use History

	RO	SRO
2003	☐	☐
2005	☑	☑
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☑
2008	☐

Answer:

B. Verify PZR Pressure Transmitter 2PT-4626B not failed and transfer 2HS-4626 to Channel 4626B.

Notes:

Although distracter A would work, it is a slower and more complicated method than answer B which is called for in the procedure.
Distracter C is only applicable if both the A and B PZR pressure transmitters have failed.
Distracter D is not a viable method since heater capacity cannot overcome heat losses due to spray flow.

References:

STM 2-03-1, PZR Pressure and Level Control, Sections 2.2.1/2/3.
AOP-2203.028, Contingency Step 5.C
AOP TG 2203.028 Step 5.
A2LP-RO-EAOP OBJ. 21, Discuss the Mitigation strategy, Entry Conditions, Instructions and Exit Conditions (a per AOP and Tech Guide) of OP 2203.028, Pressurizer Systems Malfunction.

Historical Comments:

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

Question 06

Bank:	2336	**Rev:**	2	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	6	**Author:**	Simpson
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NRC Exam Bank 0595		
Search	000056K302	**10CFR55:**	41.5	**Safety Function**	6				
Title:	Loss of Offsite Power	**System Number**	056	**K/A**	AK3.02				
Tier:	1	**Group:**	1	**RO Imp:**	4.4	**SRO Imp:**	4.7	**L. Plan:**	A2LP-RO-ELOOP
OBJ	2								

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: - Actions contained in EOP for loss of offsite power

Question:

Given the following:

* A Loss of Offsite Power has occurred and the LOOP EOP has been implemented.
* The CRS directs the ATC to isolate Letdown per the LOOP EOP.

When isolating Letdown, ______________ is the LEAST preferred isolation to use because ________________________________.

A. Letdown Isolation 2CV-4820-2; it could cause the Regen Heat Exchanger to overpressurize.

B. Regen HX Outlet 2CV-4823-2; it could cause the Regen Heat Exchanger to overpressurize.

C. Letdown Isolation 2CV-4820-2; it is NOT designed to isolate letdown at normal pressure/flow.

D. Regen HX Outlet 2CV-4823-2; it is NOT designed to isolate letdown at normal pressure/flow.

Answer:

D. Regen HX Outlet 2CV-4823-2; it is NOT designed to isolate letdown at normal pressure/flow.

Notes:

D is correct based on the design of the air operated valve 2CV-4823-2 as described in the LOOP EOP TG Step 15. The valve is designed to close on a CIAS and prevent leakage from Containment up to the DBA pressure of 59 psig but NOT designed to isolate letdown at normal pressure/flow thus it is the Least preferred valve to isolate Letdown with.

A is incorrect as 2CV-4820-2 is inside containment upstream of the Regen Heat Exchanger thus will prevent RCS pressure from being felt on the Regen Heat Exchanger but plausible if the applicant does not know the location of the valve relative to the Regen Heat Exchanger because it is designed to withhold RCS pressure.

B. is incorrect due to the design of 2CV-4823-2 as described in the LOOP EOP TG Step 15 (will not isolate against full RCS pressure) but plausible because this is the least preferred valve and downstream of the Regen Heat Exchanger outside of Containment

C is incorrect as 2CV-4820-2 is designed to hold full RCS pressure but plausible if the applicant does not understand the difference in the design of this two valves.

This question matches the K&A as it requires the knowledge of the reason for a step in the LOOP EOP action for isolating Letdown.

References:

EOP-2202.007 LOOP EOP REV 13 Step 15 (Verified reference updated 11/10/16);
EOP-2202.007 LOOP EOP TG REV 13 Step 15 (Verified reference updated 11/10/16);
STM_2-04__31-1 CVCS Drawing (Verified reference updated 11/10/16).

Historical Comments:

Used on the 2006 NRC Exam
To be used on the 2017 NRC Exam but altered for a LOOP instead of Excess RCS Leakage. Also altered distractor C to be more plausible.

REV. 1 based on NRC Chief Examiner Feedback BNC. Change the format to a 2 x 2, eliminating the 2nd half of distractor A and C and updated distractor analysis.

REV. 2 based on NRC Chief Examiner Feedback BNC. Changed the word "will" to "could" in the 2nd half of distractors A and B.

***13. Establish RCS pressure control as follows:**

A. Check BOTH 120v Instrument AC buses 2Y1 and 2Y2 energized.

A. IF only ONE 120v Instrument bus energized, THEN place the following selector switches to energized bus position:

1) PZR Level Channel Selector switch (2HS-4628).

2) Lo-Lo Level Cutout switch (2HS-4642).

3) PZR Pressure Control Channel Selector switch (2HS-4626).

B. WHEN PZR level greater than 29%, THEN verify PZR Proportional heaters restored.

***14. Check RCS pressure less than 2250 psia.**

***14. IF RCS pressure greater than 2250 psia, THEN perform the following:**

A. Verify maximum of ONE Charging pump running.

B. Reduce RCS pressure to less than 2250 psia using 2202.010 Attachment 48, RCS Pressure Control.

15. Verify at least ONE Letdown Isolation valve closed.

- 2CV-4820-2
- 2CV-4821-1
- 2CV-4823-2 (least preferred)

EOP STEP - Section 1 – DG Operations:

15. **Verify at least ONE Letdown Isolation valve closed.**

EPG STEP: N/A

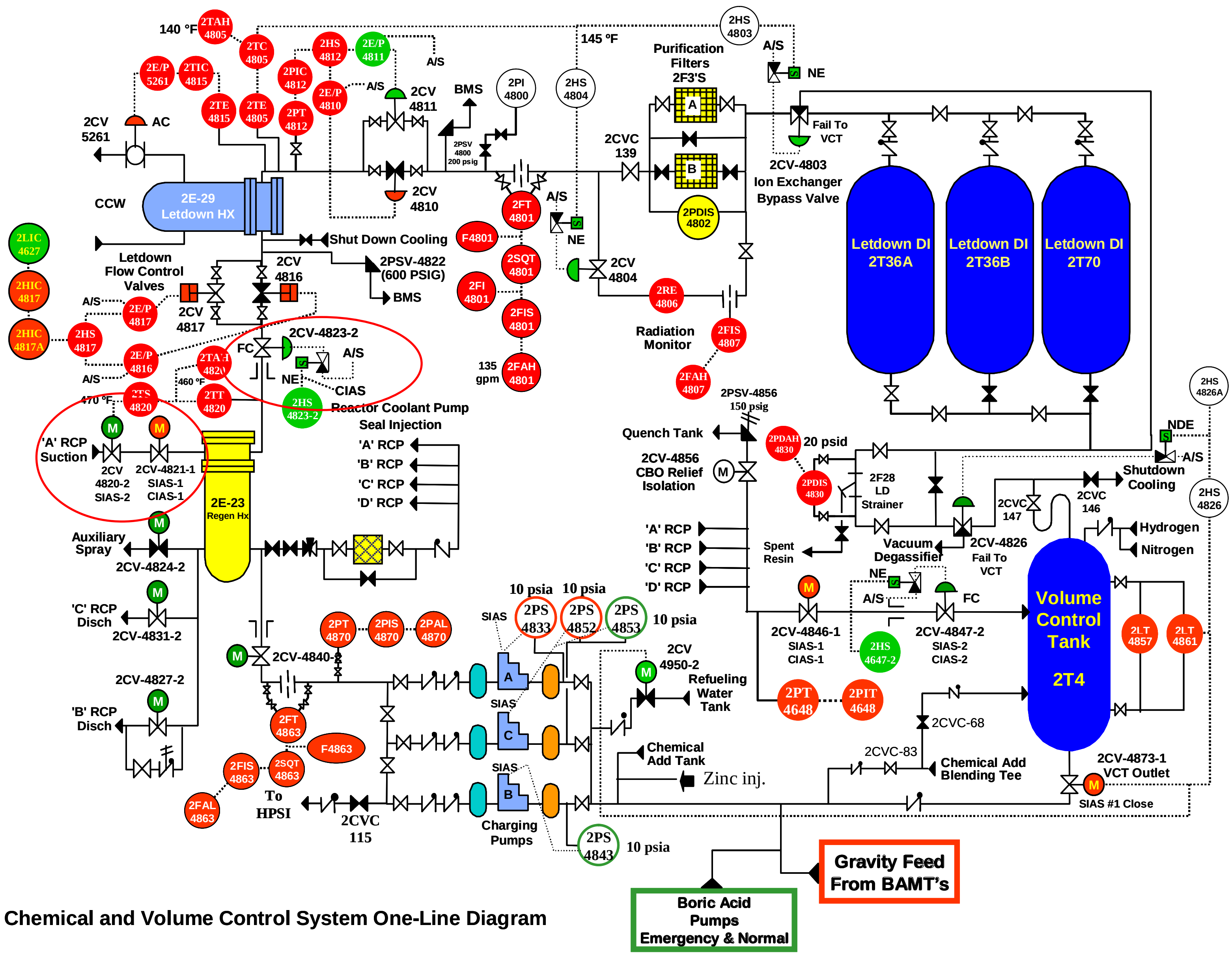
DEVIATION? Yes

BASIS FOR DEVIATION

This step was added to ensure letdown is isolated. Letdown must remain isolated until CCW has been restored. At least one letdown isolation valve must be closed to conserve RCS inventory. Previously, the letdown isolation valve outside containment was used to allow for local operation or inspection in the event of valve malfunction. However, a failure of the backpressure regulator control and subsequent failure of outside isolation valve to fully isolate Letdown led to CR-ANO-2-1999-0743 (2). This CR determined that 2CV-4823-2 is not designed to isolate full RCS letdown at normal letdown pressure/flow. In fact, the valve is designed to isolate a broken pipe against Containment design pressure (about 58 psid) i.e. 2CV-4823-2 is a containment isolation valve vice an RCS isolation valve. Consequently the step has been restructured to allow the operator to isolate Letdown by verifying at least one letdown isolation valve closed, and listing all three isolation valves as options with 2CV-4823-2 being least preferred.

SOURCE DOCUMENTS:

1. M-2231, sheet 1 of 2, P&ID, Chemical & Volume Control System.
2. CR-ANO-2-1999-0743



Chemical and Volume Control System One-Line Diagram

ORIGINAL BANK QUESTION 595 WAS USED TO DEVELOP 2017 NRC EXAM QID#6 (Used on the 2006 NRC Exam)

Questions For All QID In Exam Bank

Bank:	0595	**Rev:**	0	**Rev Date:**	5/11/2006	**QID #:**	27	**Author:**	Simpson
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	1	**Source:**	New 2006		
Search		**10CFR55:**		**Safety Function**					
System Title:	Excess RCS Leakage	**System Number**	A16	**K/A**	AK3.2				
Tier:	1	**Group:**	2	**RO Imp:**	2.8	**SRO Imp:**		**L. Plan:**	
OBJ									

Description: Knowledge of the reasons for normal, abnormal, and emergency operating procedures associated with Excess RCS Leakage

Question:

The Excess RCS Leakage AOP is being implemented for an unidentified leak in containment. The CRS directs the ATCO to isolate Letdown due to pressurizer level deviating from setpoint.

When isolating Letdown, ______________ is the LEAST preferred isolation to use because ________________________________.

A. Letdown Isolation 2CV-4820-2; it is inside containment and may not fully close in this high moisture environment

B. Regen HX Outlet 2CV-4823-2; it will cause the Regen Heat Exchanger to overpressurize

C. Letdown Isolation 2CV-4820-2; it is powered from non-vital MCC 2B81 and will likely trip its breaker

D. Regen HX Outlet 2CV-4823-2; it is NOT designed to isolate letdown at normal pressure and flow

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☑	☑
2008	☐	☐

Audit Exam History

2006	☐
2008	☐

Answer:

D. Regen HX Outlet 2CV-4823-2; it is NOT designed to isolate letdown at normal pressure and flow

Notes:

2CV-4820-2 is prefered for leak determination since it is the first valve in the letdown flowpath.
The regen HX is design for full RCS pressure.

References:

2203.016 and Tech Guide, step 3

Historical Comments:

Question 07

Bank:	2337	**Rev:**	2	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	7	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC Exam Bank 1522		
Search	0000262404	**10CFR55:**	41.10	**Safety Function**	8				
Title:	Loss of Component Cooling Water (CCW)	**System Number**	026	**K/A**	2.4.4				
Tier:	1	**Group:**	1	**RO Imp:**	4.5	**SRO Imp:**	4.7	**L. Plan:**	A2LP-RO-ARCP
OBJ	4								

Description: Emergency Procedures/Plan - Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Question:

Given the following:

* The plant is at 100% power.
* CCW Pump 2P33B is running supplying the CCW System.
* CCW Pump 2P33A is in standby.
* CCW Pump 2P33C is tagged out for motor replacement.

NOW the following alarms come in:

* Annunciator 2K12-E7 "2P-33B DISCH PRESS HI/LO".
* Annunciators 2K11-A1/A3/A5/A7 "RCP CCW DISC FLOW LO" .
* RCP Controlled Bleedoff (CBO) temperatures are 145°F and rising.

Which of the following actions is required to be taken FIRST based on these alarms and conditions?

A. Ensure CCW Pump 2P33A starts automatically then place 2P33B in Pull to Lock.

B. Restore CCW to RCPs using one second modulations on Return Valve 2CV-5255-1.

C. Enter the RCP Emergencies AOP and attempt to restore CCW flow to the RCPs.

D. Trip the Reactor, stop ALL RCPs, isolate RCP CBO flow and then GO TO SPTAs.

Answer:

C. Enter the RCP Emergencies AOP and attempt to restore CCW flow to the RCPs.

Notes:

C is correct as the "CCW DISC FLOW LO" is an entry condition to the RCP Emergencies AOP with the intent of restoring flow to the RCPs within 10 minutes or tripping the reactor if not restored within 10 minutes. This alarm is a RED colored Annunciator (Highest Priority) and require prompt attention and action first to prevent an unnecessary reactor trip.

A is incorrect because the 'A' CCW pump has no automatic starts and would not mitigate the event since the CCW cross connect valves to CCW Loop 2 (RCP Supply Loop) close in this event but is a plausible and an eventual manual RCP Emergency AOP action to restore CCW Low 2 CCW Flow if a pump failure caused the alarm.

B is incorrect as the RCP Emergency AOP allows 10 minutes to restore CCW to the RCP and can be restored with full flow as long as RCP CBO temperatures are less than 180°F but plausible as this action is directed by the RCP emergency AOPAOP Step 7 Contingency Action A if RCP CBO temperatures are greater than 180°F. Standard Attachment 21 Step 4 requires slowly restoring CCW using 1 second modulations on the return valve.

D is incorrect as the RCP Emergency AOP allows 10 minutes to restore CCW to the RCP prior to tripping the Reactor and

RCPs but plausible as this action is directed by the AOP if the prior attempts to restore CCW within 10 minutes fail and the reactor is tripped. This action bottles up the seals and prevents any more hot RCS from cooking the seals causing a possible LOCA..

This question matches the K&A as the given indications must be recognized as entry conditions to an abnormal operating procedure.

References:

AOP 2203.012L, REV 49 ACA for Annunciator 2K12-E7 "2P-33B DISCH PRESS HI/LO" (Verified reference updated 11/10/16); AOP 2203.012K, REV 46 ACA for Annunciator 2K11-A1 "CCW DISC FLOW LO" (Verified reference updated 11/10/16); AOP 2203.025 REV. 18 RCP Emergencies Entry Conditions (Verified reference updated 11/10/16); AOP 2203.025 REV. 18 RCP Emergencies Step 1.A (Verified reference updated 11/10/16); AOP 2203.025 REV. 18 RCP Emergencies Step 3. (Verified reference updated 11/10/16);
AOP 2203.025 TG REV. 17 RCP Emergencies Step 3 (Verified reference updated 11/10/16).
AOP 2203.025 REV. 18 RCP Emergencies Step 7 and 2202.010 Standard Attachment 21 Step 4.

Historical Comments:

NRC Exam Bank 1522 question was used on the 2008 NRC Exam
To be used on the 2017 NRC Exam with alterations to stem bullets and distracter A to prevent overlap of event in scenario #4.

REV. 1 based on NRC Chief Examiner Feedback BNC. Removed the word "immediately from distractor B and reworded distractor D to IAW procedure direction.

REV. 2 based on NRC Chief Examiner Feedback BNC. Replaced Distractor B with another contingency action step 7.A out of the RCP Emergency Procedure which directs performing Standard Attachment 21 Step 4. Also updated the analysis for B

ANNUNCIATOR 2K12

E-7

2P-33B DISCH PRESS HI/LO

1.0 CAUSES

1.1 HIGH - CCW pump 2P-33B discharge pressure (2PS-5228) ≥ 125 psig

1.2 LOW - 2P-33B handswitch (2HS-5228) in NORMAL-AFTER-START for 10 seconds
AND CCW pump 2P-33B discharge pressure (2PS-5228) ≤ 88 psig

2.0 ACTION REQUIRED

2.1 Locally check discharge pressure (2PI-5228).

2.2 IF HIGH pressure,
THEN perform the following:

NOTE
Cross-Over valve handswitches must be held in position for ~ 5 seconds after receiving full open or closed indication to ensure complete valve travel.

2.2.1 Verify applicable Cross-Over valves open:

- 2P-33B & C Cross-Overs:
  - 2CV-5221
  - 2CV-5232
- 2P-33A & B Cross-Overs:
  - 2CV-5220
  - 2CV-5230

2.2.2 Verify applicable Pressure Control valve opening to reduce pressure:

- Loop 1 CCW Pressure Control valve (2PCV-5203)
- Loop 2 CCW Pressure Control valve (2PCV-5212)

2.2.3 Verify normal valve alignment using Component Cooling Water System Operation (2104.028).

(E-7 Continued on next page)

ANNUNCIATOR 2K12

E-7

2P-33B DISCH PRESS HI/LO
(Continued)

NOTE

- Low discharge pressure on 2P-33B with the handswitch in NORMAL-AFTER-START will cause the following:
  - 2P-33B will trip
  - 2P-33C will auto-start
  - All Cross-Over valves will close
- Placing 2P-33B in PULL-TO-LOCK will defeat all the above interlocks.

2.3 IF LOW pressure,
THEN perform the following:

2.3.1 IF CCW flow NOT available to Reactor Coolant Pumps,
THEN GO TO RCP Emergencies (2203.025).

2.3.2 Verify applicable Pressure Control valve closed:

- Loop 1 CCW Pressure Control valve (2PCV-5203)
- Loop 2 CCW Pressure Control valve (2PCV-5212)

2.3.3 Vent CCW system at the following locations as applicable:

- CCW Pump 2P-33C Vent (2CCW-1031)
- CCW Pump 2P-33B Vent (2CCW-1029)
- CCW Pump 2P-33A Vent (2CCW-1001)
- CCW HX 2E-28C Vent (2CCW-1110)
- CCW HX 2E-28B Vent (2CCW-1020)
- CCW HX 2E-28A Vent (2CCW-1017)

2.3.4 IF air present in Pump or Heat Exchanger,
THEN perform the following:

- Vent components supplied by CCW.
- Determine location of air ingress into CCW System.

3.0 TO CLEAR ALARM

3.1 Lower pressure to < 125 psig.

3.2 Restore pressure to > 88 psig or stop 2P-33B.

4.0 REFERENCES

4.1 E-2457-4

ANNUNCIATOR 2K11

A-1

CCW DISCH FLOW LO

1.0 CAUSES

1.1 Component Cooling Water return flow from Reactor Coolant Pump 2P-32A < 240 gpm (2FIS-5240) with 2P-32A handswitch (2HS-4620) in position other than PULL TO LOCK

2.0 ACTION REQUIRED

2.1 Dispatch Operator to check proper operation of Loop 2 CCW. Refer to Component Cooling Water System Operations (2104.028).

2.2 Check CCW Loop 2 Flow (2FIS-5202).

2.3 IF Loop 2 CCW flow low or degraded,
THEN **GO TO** Reactor Coolant Pump Emergencies (2203.025).

2.4 IF Loop 2 CCW flow normal,
THEN perform the following:

2.4.1 Monitor RCP 2P-32A pump and motor temperature trends on PMS/PDS.

2.4.2 IF RCP 2P-32A pump and motor temperatures trending up,
THEN **GO TO** Reactor Coolant Pump Emergencies (2203.025).

3.0 TO CLEAR ALARM

3.1 Restore RCP 2P-32A CCW flow to > 250 gpm.

3.2 Secure RCP 2P-32A and place handswitch (2HS-4620) in PULL TO LOCK.

4.0 REFERENCES

4.1 E-2457-1

RCP EMERGENCIES

PURPOSE

This procedure provides actions for RCP Emergencies.

ENTRY CONDITIONS

ANY of the following conditions exist:

1. RCP individual seal ΔP less than 50 psid.
2. Unexplained "RCP BLEEDOFF FLOW HI/LO" annunciator (2K11-G3) in alarm.
3. "RCP BLEEDOFF TEMP HI" annunciator (2K11-G2) in alarm.
4. "UPPER/LOWER OIL RSVR LEVEL LO" annunciator (2K11-F1/F3/F5/F7) in alarm.
5. "CCW DISCH FLOW LO" annunciator (2K11-A1/A3/A5/A7) in alarm.
6. "UPPER THRUST BRG METAL TEMP HI" annunciator (2K11-B1/B3/B5/B7) in alarm.
7. "LOWER THRUST BRG METAL TEMP HI" annunciator (2K11-B2/B4/B6/B8) in alarm.
8. "STATOR WDG TEMP HI" annunciator (2K11-A2/A4/A6/A8) in alarm.
9. "RCP VIBRATION HI" annunciator (2K11-G6) in alarm.
10. Abnormal RCP amps and D/P (Indications of RCP sheared shaft)

EXIT CONDITIONS

EITHER of the following conditions exists:

1. CCW to RCP restored and ALL RCP annunciators clear.
2. ALL affected RCPs stopped.

INSTRUCTIONS	CONTINGENCY ACTIONS

> **NOTE**
>
> - Attachment D provides RCP trip and plant shutdown criteria.
> - Steps marked with (*) are continuous action steps.

*1. **CHECK** the following criteria for EACH RCP satisfied:

INSTRUCTIONS	CONTINGENCY ACTIONS
A. CCW flow aligned to RCPs.	A. **PERFORM** the following: 1) **IF** confirmed fire in U2 Turbine Bldg, U2 Aux Blg OR U2 Aux Ext. Bldg. coincident with loss of CCW, **THEN GO TO** Step 2. 2) **GO TO** Step 3.
B. In-service CCW Surge Tank level greater than 13% following restoration of CCW to RCPs	B. **GO TO** Step 4.
C. Vapor Seal pressure less than 1500 psia.	C. **GO TO** Step 5.
D. Seal Stage ΔP greater than 50 psid.	D. **GO TO** Step 6.
E. Controlled Bleedoff temperature less than 180°F.	E. **GO TO** Step 7.
F. Controlled Bleedoff flow less than 3.0 gpm.	F. **GO TO** Step 8.
G. Bearing and Motor Temperature annunciators clear.	G. **GO TO** Step 9.
H. Upper and Lower Oil Reservoir annunciators clear.	H. **GO TO** Step 10.
I. High Vibration annunciator clear.	I. **GO TO** Step 11.
J. Normal RCP amps and D/P	J. **GO TO** Step 12.

INSTRUCTIONS	CONTINGENCY ACTIONS

*3. **IF** CCW flow **NOT** aligned to RCPs,
THEN PERFORM the following:

A. **RECORD** time CCW flow to RCPs lost:

- Time: _______________

B. **DISPATCH** operator to CCW Pump Room.

C. **IF** CCW can **NOT** be restored within 10 minutes,
THEN PERFORM the following:

1) **IF** in Mode 1 OR 2,
THEN TRIP Reactor.

2) **STOP** ALL RCPs.

3) **ENSURE** BOTH PZR Spray valves in MANUAL and closed:

- 2CV-4651
- 2CV-4652

CAUTION

Failure to isolate Controlled Bleedoff may result in failure of RCP seals.

4) **ISOLATE** Controlled Bleedoff as follows:

a) **CLOSE** RCP Bleedoff to VCT valve (2CV-4847-2).

b) **CLOSE** RCP Bleedoff Isolation to VCT valve (2CV-4846-1).

c) **CLOSE** RCP Bleedoff Relief to Quench Tank valve (2CV-4856).

5) **IF** Reactor manually tripped,
THEN GO TO 2202.001,
Standard Post Trip Actions.

(Step 3 continued on next page)

INSTRUCTIONS	CONTINGENCY ACTIONS
3. (continued)	
	NOTE If a CIAS relay has actuated, it will be necessary to override CCW Containment Isolation valves. In this case, TS 3.0.3 will be applicable until a dedicated operator can be established.
D. **CHECK** the following valves open: • CCW CNTMT Supply valve (2CV-5236-1). • RCP CCW Return valve (2CV-5254-2). • RCP CCW Return valve (2CV-5255-1).	D. **RESTORE** CCW to RCPs as follows: 1) **OPEN** CCW CNTMT Supply valve (2CV-5236-1). 2) **OPEN** RCP CCW Return valve (2CV-5254-2). 3) **OPEN** RCP CCW Return valve (2CV-5255-1). 4) **IF** ANY CCW pump running, **THEN GO TO** step 4.
E. **ENSURE** in-service CCW Surge Tank (2T-37A/B) level greater than 13% on 2LIS-5210/5214.	
F. **PERFORM** the following for L/D Temp controller (2TIC-4815): 1) **PLACE** controller in MANUAL. 2) **ADJUST** output to 10%.	
G. **PLACE** ALL handswitches for non-running CCW pumps in PTL: • 2P33A • 2P33B • 2P33C	

(Step 3 continued on next page)

INSTRUCTIONS	CONTINGENCY ACTIONS

3. (continued)

NOTE
Swapping CCW pumps with Loops 1 and 2 CCW crosstied will result in surge tank level changes.

H. **IF** available,
THEN START 2P33C.

I. **CHECK** 2P33C running.

2P-33C is not available due to being tagged out in initial conditions.

I. **PERFORM** the following to start 2P33A on Loop 2:

1) **ENSURE** 2P33C in PTL.

NOTE
CCW Cross-Over valve handswitches must be held in position for ~ five seconds after receiving full open or closed indication to ensure complete valve travel.

2) **ENSURE** the following Cross-Over valves open:

- 2P-33B & C Cross-Over valves:
  - 2CV-5221
  - 2CV-5232
- 2P-33A & B Cross-Over valves:
  - 2CV-5220
  - 2CV-5230

3) **START** 2P33A.

(Step 3 continued on next page)

AOP STEP:

*3. **<u>IF</u> CCW flow <u>NOT</u> aligned to RCPs**
<u>THEN</u> perform the following:

BASIS:

The most likely causes of loss of CCW to the RCPs that could be corrected within 10 minutes are inadvertent closure of a containment isolation valve or loss of a CCW pump. Other potential causes such as air entrapment in the pump or a system rupture are not considered in this AOP due to the unlikely ability to correct the condition within 10 minutes.

A. The time that CCW flow was lost is recorded so that a determination can be made as to when the 10 minutes is up.

B. An operator is dispatched to the CCW Pump Room to rapidly respond if 2P33B Discharge Valve 2CCW-29 has to be throttled for pump start.

C. The RCP tech manual requires that after 10 minutes of operation with no CCW flow, the affected pumps must be secured to prevent seal damage. If CCW flow cannot be restored, then the reactor is tripped and all RCPs are secured. The reactor is manually tripped first if the Plant is in Mode 1 or 2 to prevent an automatic reactor trip when the first RCP is secured. The 10 minute limit is also important for RCP motor protection. It is possible that high bearing/stator temperature alarms will annunciate before 10 minutes have elapsed. If a high motor temperature alarm annunciates before the 10 minute limit, then the operator is expected to trip the reactor and trip the affected RCPs in accordance with the procedure. The requirement to secure all RCPs and isolate Controlled Bleedoff flow still applies to the remaining RCPs if CCW flow cannot be restored within the 10 minute time limit.

   Controlled Bleedoff flow is isolated if the RCPs must be secured due to the loss of CCW flow. Isolating CBO minimizes heat input to the seals. A caution is provided to warn the operator that failure to isolate CBO flow could lead to seal failure.

   If the reactor is tripped and all pumps secured due to low flow, then the operator meets the EXIT CONDITION of all RCPs secured and is transitioned to the EOP.

D. The CCW Containment Isolations are checked open and the contingency action provides instructions to open them if they are closed. A note is included to remind the operator that they may have to be overridden if a CIAS relay has actuated. The contingency action bypasses the remaining instructions in this step if a CCW pump is running because either this fixed the problem, or one or more valves can not be opened and there is no need to perform actions for loss of the pumps. If CCW MOVs are overridden on both trains, Tech Spec 3.0.3 should be entered until a dedicated operator can be stationed.

E. Greater than 13% level in the CCW surge tank is verified to ensure adequate NPSH is available for the CCW pumps.

F. A loss of CCW will cause the Letdown HX temperature control valve (2TIC-4815) to go fully open. 2TIC-4815 is placed in manual at 10% open to reduce flow demand on the CCW pump upon restart, and to reduce the potential of thermal shock to the HX.

G. The CCW pump low discharge pressure switch and interlock relay is only enabled if the associated pump handswitch is in Normal After Start. All non-running CCW pumps are placed in Pull to Lock to block actuation of these relays and allow manual operation of the pumps and loop cross-over valves.

H. If 2P-33C is available it is started first since this is the preferred pump for Loop 2.

I. If 2P-33C will not start, then contingency actions direct starting 2P-33A after verifying the crossover valves open. It is the next preferred pump because it does not have a low discharge pressure automatic trip.

J. If neither 2P-33A or 2P-33C is running, contingency actions are provided to place 2P-33B in service. The discharge valve is throttled two turns open prior to pump start, and throttled open to prevent pump trip on low discharge pressure (88 psig).

Notes remind the operator that the cross-over valve handswitches must be held in position for ~ 5 seconds after receiving full open or closed indication to ensure complete valve travel.

K. When the system has stabilized, the CCW to Letdown HX Temp Controller can be placed back in AUTO for a normal configuration.

L. Loss of CCW flow will result in Letdown Radiation Monitor Inlet Valve 2CV-4804 closure. This step reminds the operator to open the valve to restore flow.

M. In the event the Crossover Valves were interlocked closed, this step directs reopening of the valves if desired.

Steps to slowly restore CCW to RCPs following restoration of a CCW pump are not included based on the event documented in CR-ANO-2-2007-0313 in which CBO temperature never exceeded 160°F.

4. IF RCP seal temperatures 180°F or greater,
THEN restore CCW to RCPs as follows:

A. Verify RCP CCW Return valve (2CV-5255-1) closed.

B. Verify RCP CCW Supply valve (2CV-5236-1) open.

C. Verify RCP CCW Return valve (2CV-5254-2) open.

D. Throttle open RCP CCW Return valve (2CV-5255-1) with a single 1 second modulation.

E. Verify ANY CCW pump in service.

F. IF unexplained CCW Surge Tank level changes observed,
THEN perform the following:

1) Verify ALL RCPs stopped.

2) Isolate CCW to RCPs.

3) Verify RCP Bleedoff to VCT valves closed:

- 2CV-4846-1
- 2CV-4847-2

4) Verify RCP Bleedoff Relief Isolation to Quench Tank valve (2CV-4856) closed.

G. IF RCP seal temperatures greater than 300°F,
THEN use PMS to monitor seal cooldown rate:

RCP	CONTROLLED BLEEDOFF	LOWER SEAL CAVITY
A	T6008	T6009
B	T6018	T6019
C	T6028	T6029
D	T6038	T6039

(Step 4 continued on next page)

ORIGINAL BANK QUESTION 1522 WAS USED TO DEVELOP 2017 NRC EXAM QID#7 (Used on the 2008 NRC Exam)

Questions For All QID In Exam Bank

Bank:	1522	**Rev:**	0	**Rev Date:**	10/30/2007 12:42:	**QID #:**	35	**Author:**	Coble
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW		
Search	0080002404	**10CFR55:**	41.10 / 43.2 / 45.6	**Safety Function**	8				
System Title:	Component Cooling Water System (CCWS)	**System Number**	008	**K/A**	2.4.4				
Tier:	2	**Group:**	1	**RO Imp:**	4.0	**SRO Imp:**	4.3	**L. Plan:**		**OBJ**	

Description: Emergency Procedures/Plan - Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Question:

Given the following:

* The plant is at 100% power
* CCW Pump 2P33C is running supplying the CCW System.
* CCW Pumps 2P33A and 2P33B are in Standby.
* Annunciators 2K11-A1/A3/A5/A7 "CCW DISC FLOW LO" come in.
* CCW Containment Supply Valve 2CV-5632-1 on 2C-17 has closed.

Which of the following actions should be taken first based on these alarms and indications?

A. Start CCW Pumps 2P33A and 2P33B to clear alarms then place 2P33C in Pull to Lock (PTL).

B. Trip the Reactor and commence EOP Standard Post Trip Actions (SPTAs).

C. Enter the RCP Emergencies AOP and attempt to restore CCW to the RCPs.

D. Trip the Reactor and isolate Controlled Bleedoff from the RCPs due to loss of CCW cooling.

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☑	☑

Audit Exam History

2006	☐
2008	☐

Answer:

C. Enter the RCP Emergencies AOP and attempt to restore CCW to the RCPs.

Notes:

These alarms are entry conditions for the RCP Emergency AOP and monitor CCW flow to the Containment. These are RED colored Annunciators (Highest Priority) and require prompt action because if CCW cannot be restored within 10 minutes, then the plant should be tripped and the RCPs secured.
Starting the CCW pumps would not mitigate the event since the Containment CCW supply valve has failed close
Isolating controlled bleedoff would be an action after the plant trip if CCW cannot be restored to prevent cooking the RCP seals.

References:

OP 2203.025 Entry Conditions and Step 2.
OP 2203.012K, ACAs for Annunciators 2K11-A1/A3/A5/A7 "CCW DISC FLOW LO"

Historical Comments:

Question 08

Bank:	2338	**Rev:**	2	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	8	**Author:**	Coble	
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	Modified NRC Exam Bank 0642			
Search	000008K305	**10CFR55:**	41.10			**Safety Function**	3			
Title:	Pressurizer (PZR) Vapor Space Accident (Relief Valv					**System Number**	008	**K/A**	AK3.05	
Tier:	1	**Group:**	1	**RO Imp:**	4.0	**SRO Imp:**	4.5	**L. Plan:**	A2LP-RO-ELOCA	**OBJ** 8

Description: Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident: - ECCS termination or throttling criteria

Question:

A plant pressure spike has caused a PZR Code Safety to lift with the following conditions:

* The unit tripped on high RCS pressure.
* Two HPSI Pumps are running due to subsequent low RCS pressure after the safety lifted.

* NOW the stuck open Code Safety valve has partially closed but is still leaking at 60 gpm.
* The CRS has entered the LOCA EOP Procedure OP-2203.003

* RCS press is 1340 psia and rising.
* PZR level is now 80% and rising slowly.
* T-hot is 533°F, T-cold is 520°F.
* Ave CET temp is 540°F and stable.
* RVLMS Level 3 indicates WET.
* A S/G is 23%.
* B S/G is 22%.
* Feed water flow to A S/G is 250 GPM and rising.
* Feed water flow to B S/G is 250 GPM and rising.
* All RCPs have been secured.

Based on the given plant conditions, which of the following is the procedurally required action to take at this time and the reason for the action?

A. Override and throttle HPSI flow as needed using HPSI Injection MOVs to control RCS pressure and inventory.

B. Override and throttle HPSI flow as needed using HPSI Injection MOVs to restore Margin to Saturation (MTS) below the MAXIMUM Limit.

C. Continue full flow injection to the RCS with both HPSI pumps and monitor RCS pressure and inventory.

D. Continue full flow injection to the RCS with both HPSI pumps to restore Margin to Saturation (MTS) above the MINIMUM limit.

Answer:

A. Override and throttle HPSI flow as needed using HPSI Injection MOVs to control RCS pressure and inventory.

Notes:

A is correct as the criteria for throttling HPSI flow is met in the above plant conditions LOCA EOP 2203.003 Section 2 Step 1(MTS is 41.4 degrees F based on Ave CET temp and PZR Pressure, PZR LVL is > 29% and controlled, RVLMS LVL 3 WET, SG levels > 10% with adequate FW flow) and the LOCA procedures directs this action using Exhibit 10

Step 2.2. The reason for throttling is listed in EOP 2202.003 LOCA EOP TG, Section 2 Step 1 is to control RCS pressure and inventory.

B is incorrect because Margin to Saturation (MTS) is already meeting HPSI termination criteria (30 to 45°F) but plausible if MTS is miscalculated as being too high (> 45 degrees F) or does not remember the correct band to maintain.

C is incorrect as HPSI Termination criteria is met IAW EOP 2202.003 REV. 15, LOCA EOP, Section 2 Step 1 but plausible as the correct reason is listed and RCS pressure and inventory must always be monitored in the RCS.

D is incorrect because all termination criteria is met but plausible if the candidate miscalculates the MTS too low for the given conditions (< 30 Degrees F) and determines that HPSI termination criteria is not met.

This QID matches the K&A statement as the candidate must know the reason for the actions taken during a Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)

References:

EOP 2202.003 REV. 15, LOCA EOP, Section 2 Step 1; EOP 2202.003 REV. 15, LOCA EOP TG, Section 2 Step 1 (Verified reference updated 11/10/16); EOP 2202.010 REV. 23 Exhibit 10 (Verified reference updated 11/10/16); EOP 2202.003 REV. 15, LOCA EOP TG, Section 2 Step 23 (Verified reference updated 11/10/16).

Historical Comments:

Original Bank 0642 used on the 2003 NRC Exam
To be used on the 2017 NRC Exam but modified for the 2017 NRC Exam.

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed the leakrate to a specific 60 gpm in third bullet, added a bullet to say ALL RCPs secured, and changed the word "control" to "and monitor" in distractor C.

REV. 2 based on NRC Chief Examiner Feedback BNC. In distractor B, replaced the words" within limit" at the end with "below the MAXIMUM Limit" to be more plausible. Also in distractor D, replaced the words "within limits" at the end to "above the MINIMUM limit" to be more plausible. Also updated the analysis for B

INSTRUCTIONS	CONTINGENCY ACTIONS
■1. **Terminate/throttle HPSI flow as follows:**	
A. Check the following criteria satisfied: 1) RCS MTS 30°F or greater. 2) PZR level greater than 29% [50%] and controlled. 3) RVLMS LVL 03 or higher elevation indicates WET. 4) At least ONE intact SG available for Heat Removal by EITHER of the following: • Level 10% to 90% [20% to 90%] with FW available. • Level being restored with total FW flow of 485 gpm or greater.	A. **GO TO** Step 2. All criteria in Step 1.A Met
B. Terminate/throttle HPSI flow using 2202.010 Exhibit 10, HPSI Termination/Throttling Criteria.	
*2. **Monitor HPSI termination/throttle criteria satisfied for duration of event.**	*2. **Re-establish RCS pressure, inventory, and heat removal with HPSI as follows:** **NOTE** **It is not necessary to re-establish full HPSI flow as long as HPSI termination/throttle criteria can be met by either throttling MOVs or starting pumps as needed.** A. Raise HPSI flow by performing the following as necessary: • Throttle open desired HPSI MOVs • Start additional HPSI pump(s) B. Verify control of MTS, PZR level, RVLMS level, SG level, and FW flow.

EOP STEP - Section 2 – Isolated LOCA

■1. **Terminate/throttle HPSI flow as follows:**

EPG STEP:

*53.

DEVIATION? Yes

BASIS FOR DEVIATION:

The EOP presents the plant specific criteria for HPSI termination/throttling (1). These criteria are consistent with those of the EPG with more specific criteria added to define an "available" SG. The EOP uses the normal plant specific post accident SG level range (1). This range is wider that the EPG's "normal band" to allow for raised flexibility during accident conditions. The prescribed band is sufficient to maintain RCS heat removal using an intact SG.

Once these criteria have been satisfied, the EPG allows HPSI flow to be throttled or stopped as necessary to control RCS pressure and inventory. The EOP directs the operator to an exhibit that provides the required actions to be taken in regard to HPSI flow (3). If HPSI pumps are required to be stopped, then HPSI pumps are placed in PTL since SIAS has not yet been reset (2).

SOURCE DOCUMENTS:

1. ANO-2 EOP Setpoint Document, setpoints G.5, G.6, G.7, G.8, & F.3.
   ANO-2 EOP Setpoint Document, setpoints M.1, D.5, D.6, & R.2.
1. M-2418, sheet 3 of 5, Functional Description and Logic Diagram, Safety Injection System.
2. 2202.010, Standard Attachments, Exhibit 10, HPSI Termination/Throttling Criteria.

EXHIBIT 10

HPSI TERMINATION/THROTTLING CRITERIA

1.0 **IF HPSI flow actuation was due only to an RCS cooldown, THEN perform the following:**

Not Applicable

1.1 IF NOT previously accomplished,
THEN verify the following criteria satisfied:

- RCS MTS 30°F or greater.
- PZR level greater than 29% [50%] and controlled.
- RVLMS LVL 03 or higher elevation indicates WET.
- At least ONE intact SG available for Heat Removal by EITHER of the following:
  - Level 10 to 90% [20 to 90%] with FW available.
  - Level being restored with total FW flow of 485 gpm or greater

1.2 Place all operating HPSI Pumps in Pull To Lock.

1.3 Override and close HPSI Injection MOVs:

- 2CV-5015-1
- 2CV-5035-1
- 2CV-5055-1
- 2CV-5075-1
- 2CV-5016-2
- 2CV-5036-2
- 2CV-5056-2
- 2CV-5076-2

EXHIBIT 10

HPSI TERMINATION/THROTTLING CRITERIA

2.0 **IF HPSI flow actuation was due to a loss of RCS inventory, THEN perform the following:**

2.1 IF NOT previously accomplished, THEN verify the following criteria satisfied:

- RCS MTS 30°F or greater.
- PZR level greater than 29% [50%] and controlled.
- RVLMS LVL 03 or higher elevation indicates WET.
- At least ONE intact SG available for Heat Removal by EITHER of the following:
  - Level 10 to 90% [20 to 90%] with FW available.
  - Level being restored with total FW flow of 485 gpm or greater

2.2 Override and throttle HPSI flow as needed using HPSI Injection MOVs to control RCS pressure, inventory, and heat removal:

- 2CV-5015-1
- 2CV-5035-1
- 2CV-5055-1
- 2CV-5075-1
- 2CV-5016-2
- 2CV-5036-2
- 2CV-5056-2
- 2CV-5076-2

2.3 IF desired to secure any HPSI Pumps, THEN obtain CRS/SM approval prior to securing.

INSTRUCTIONS	CONTINGENCY ACTIONS
■23. Minimize primary break flow as follows:	
A. Check ANY RCP running.	A. IF ALL RCPs secured, THEN perform the following: 1) Maintain RCS MTS 30 to 45°F, refer to 2202.010 Attachment 1, P-T Limits 2) **GO TO** Step 23.C
B. Maintain RCS pressure within 100 psia above minimum RCP NPSH requirements, refer to 2202.010 Attachment 1, P-T Limits.	
C. Use ONE of the following to depressurize RCS: 1) 2202.010 Attachment 48, RCS Pressure Control. 2) IF HPSI throttle criteria met, THEN cycle Charging pumps or throttle HPSI flow to lower PZR pressure.	
■24. WHEN RCS T_C less than 510°F, THEN reduce number of running RCPs as follows:	
A. Verify maximum of ONE RCP running in EACH loop.	
B. IF RCP 2P32A or 2P32B stopped, THEN verify associated PZR Spray valve in MANUAL and closed.	

PARENT QUESTION 642 WAS USED TO MODIFY 2017 NRC EXAM QID#8 (Used on the 2003 NRC Exam)

Data for 2003 NRC RO/SRO Exam

Bank: 0642	**Rev:** 0	**Rev Date:**		**QID #:**		**Author:**	
Lic Level:	**Difficulty:** 0	**Taxonomy:**		**Source:** New 2003			
Search	**10CFR55:**			**Safety Function**			
System Title:				**System Number**		**K/A** EK1.2	
Tier:	**Group:**	**RO Imp:**	**SRO Imp:**	**L. Plan:**		**OBJ**	

Description:

Question:

The plant has suffered a LOCA with the following conditions:

* All RCPs have been secured; All HPSI and LPSI pumps are running.
* RCS press is 1200 psia and rising.
* PZR level is 100 %.
* Thot is 510 F, Tcold is 500 F
* Ave CET temp is 560 F
* A S/G is 20 % NR
* B S/G is 25% NR
* Feed water flow to A S/G is 260 GPM
* Feed water flow to B S/G is 220 GPM.

Based on the given plant conditions, which ONE (1) of the following actions are correct?

A. Throttle HPSI flow to prevent placing the RCS in a solid plant condition and to control RCS pressure.

B. Secure both HPSI and LPSI pumps, maintain plant pressure using charging pumps as necessary.

C. Continue injection to the RCS with both HPSI pumps and verify HPSI flow using HPSI flow curves.

D. Secure the LPSI pumps and 1 HPSI pump. Throttle flow from the running HPSI pump as necessary to control RCS pressure but maintain at least 50 gpm.

QID use History

	RO	SRO
2003	☑	☑
2005	☐	☐
2006	☐	☐
2008	☐	☐
2009	☐	☐
2011	☐	☐

Audit Exam History

2011	☐

Answe C. Continue injection to the RCS with both HPSI pumps and verify HPSI flow using HPSI flow curves.

Notes:

References:

Historical Comments:

Question 09

Bank:	2339	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	9	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	4	**Taxonomy:**	H	**Source:**	NEW for 2017 NRC Exam		
Search	000058A203	**10CFR55:**	41.7	**Safety Function**	6				
Title:	Loss of DC Power	**System Number**	058	**K/A**	AA2.03				
Tier:	1	**Group:**	1	**RO Imp:**	3.5	**SRO Imp:**	3.9	**L. Plan:**	A2LP-RO-ED125 **OBJ** 3

Description: Ability to determine and interpret the following as they apply to the Loss of DC Power: - DC loads lost; impact on to operate and monitor plant systems

Question:

Given the following plant conditions:

* Plant trips from 100% power due to a Loss of Offsite Power (LOOP).
* The Power Supply Breaker from 2D01 to 2D23, 2D-31, has tripped open.
* All attempts to reclose Supply Breaker 2D-31 have failed.
* #1 Emergency Diesel 2DG1 does not automatically start.
* CRS directs crew to place 2DG1 in service on the Vital 4160 VAC 2A3 bus.

Which of the following describes the starting and stopping of 2DG1 with these conditions?

A. 2DG1 can be started from the Control Room but has to be secured locally in the #1 EDG Room.

B. 2DG1 has to be started locally in the #1 EDG Room but can be secured from the Control Room.

C. 2DG1 has to be started and stopped locally in the #1 EDG Room.

D. 2DG1 can be started and stopped remotely from the Control Room.

Answer:

C. 2DG1 has to be started and stopped locally in the #1 EDG Room.

Notes:

C is correct because the LOOP does not allow power to come from offsite through SU XFMR #3 so to start and stop #1 EDG requires DC control power to energize the air start solenoids and the shut down relay to dump hydraulic fluid off the governor to place the fuel racks in the no fuel position. However, the air start solenoids can be manually overridden open locally and a backup source of 12 VDC power can be aligned for field flashing. This will get the EDG started. The manual pushbutton that actuates the mechanical overspeed shutdown device can be pushed locally to shut the engine down by closing the fuel inlet rack with a loss of DC which would normally energize a relay circuit to close the fuel inlet rack remotely.

This explanation makes answers A, B and D wrong.

A is plausible if the candidate incorrectly assumes the air start valves are not powered by DC and only the shutdown relay requires DC to operate.

B is plausible if the candidate incorrectly assumes that DC power to secure the EDG will be supplied from the EDG itself after the start.

D is plausible if the candidate incorrectly assumes that DC is not needed to start and stop the EDG remotely or does not know that 2D23 is the DC power supply to the #1 EDG

This question matches the K&A as the candidate must interpret the Loss of Red Train 125 VDC conditions and determine

its impact on starting 2DG1 as directed by the Loss of DC Power AOP.

References:

AOP-2203.037, Loss of 125 V DC AOP, Rev 13, Section 2, Step 34 (Verified reference updated 11/10/16); AOP-2203.037, Loss of 125 V DC TG, Rev 13, Section 2, Step 34 (Verified reference updated 11/10/16); OP-2104.036 EDG Operations Rev. 91 Exhibit 1 Note and Step 11 (Verified reference updated 11/10/16); STM_2-31_34-1 EDGs Air Start Circuit (Verified reference updated 11/10/16);
STM_2-31_34-1 EDGs Stopping Circuit (Verified reference updated 11/10/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed bullet #3 to a loss of 2D23 which supplies the stating and stopping circuit for the Red train EDG. Removed the words " without Red DC" from the stem.

INSTRUCTIONS		CONTINGENCY ACTIONS
34.	**CHECK** SU XFMR #3 voltage greater than 22 KV.	34. **IF** SU XFMR #3 **NOT** available, **THEN PERFORM** the following: A. **ENERGIZE** 4160v Vital bus 2A3 by performing EITHER of the following: 1) **START** AACG AND align to 4160v Vital bus 2A3 using 2104.037 Attachment E, AAC Generator Emergency Start. 2) Locally **START** 2DG1 and **ALIGN** to 4160v Vital bus 2A3 using 2104.036 Exhibit 1, Starting 2DG1 Without DC Control Power. B. **IF** 4160v Vital bus 2A3 energized, **THEN GO TO** Step 40. C. **IF** 2A3 can **NOT** be energized from DG, **THEN PERFORM** the following as necessary: 1) **SECURE** AACG using 2104.037, Alternate AC Diesel Generator Operations. 2) Locally **STOP** 2DG1 using 2104.036 Exhibit 1, Starting 2DG1 Without DC Control Power.

(Step 34 continued on next page)

AOP STEP - SECTION 2 (RED TRAIN DC):

34. **CHECK** SU XFMR #3 voltage greater than 22 KV.

BASIS:

This step verifies S/U XFMR #3 primary voltage is adequate to ensure electrical reliability for necessary loading. If S/U XFMR #3 voltage is adequate the operator proceeds to the following steps which prep 4160v buses 2A1 and 2A3 for power restoration to 2A1, and then restore power to 2A1/2A3.

If S/U XFMR #3 voltage is not adequate, operator is directed in contingency column to align power to 2A3 from (in order of preference) AACG (3), 2DG1 (2) and S/U XFMR #2 (1). If this step is successful, operator proceeds to step which restarts SW pump on 2A3 which restores Loop 1 Service Water. If this step is unsuccessful, operator is directed to secure DG started and then maximize efforts to restore power to Vital Red Train DC and 2A3.

SOURCE DOCUMENTS:

1. 2107.001, Electrical System Operation.
2. 2104.036, Emergency Diesel Generator Operation
3. 2104.037, Alternate AC Diesel Generator Operations

2104.036 **EXHIBIT 1** **Revised 7/27/16**

STARTING 2DG1 WITHOUT DC CONTROL POWER

NOTE

- This procedure assumes no Red AC or DC power available.
- Alternate Shutdown does not require the use of Electrical Safety equipment and the electrical line up will be performed by the Alternate Shutdown procedure.
- Instructions for manual operation of 4160 VAC Breakers are contained in 2107.001, Electrical System Operations, Exhibits 1 and 2.
- A medium sized ladder will be required to access 2T-30A Inlet Solenoid 2SV-2802-1 Bypass (2ED-45).

CAUTION

Fault condition that caused loss of DC may still be present.

1.0 **PERFORM** the following initial conditions and electrical lineups:

1.1 **IF** directed to perform this exhibit per 2203.014, Alternate Shutdown, **THEN GO TO** Step 2.0.

1.2 **OBTAIN** arc flash PPE (suit, hood, and voltage rated or leather gloves) for Manual Operation of 4160/6900V AC Breaker with Door Open.

1.3 **OPEN** ONE of the following sets of breakers:

1.3.1 Breakers in 2D23:

- 2C107 Power Supply (2D23-2)
- 2E11 Power Supply (2D23-6)
- 2C107 Power Supply (2D23-8)

1.3.2 Breakers in EDG room:

- 2E11 Breaker located inside 2E12 (D2E11NA6)
- Breaker inside 2C107 (D2C107NA8)
- Breaker inside 2C107 (D2C107NA2)

2104.036 **EXHIBIT 1** **Revised 7/27/16**

STARTING 2DG1 WITHOUT DC CONTROL POWER

PAGE 5 OF 6

11.0 **WHEN** desired to secure 2DG1 without DC Control Power,
THEN:

11.1 **ENSURE** loads being supplied by 2DG1 are not required.

11.2 **ENSURE** Service Water Pump breakers open:

- Service Water Pump 2P-4A (2A-302)
- Service Water Pump 2P-4B (2A-303)

11.3 **OPEN** 2DG1 Output breaker (2A-308).

11.4 **TRIP** 2DG1 using Emergency Stop pushbutton (2HS-2800A).

11.5 **UNLOCK** local Engine Control switch (2HS-2815-1).

11.6 **PLACE** Engine Control switch (2HS-2815-1) in LOCKOUT.

12.0 **PERFORM** the following to restore 2DG1 to normal:

12.1 **IF** Backup Flashing Circuit was used,
THEN:

- **PLACE** Flashing Power Select Switch (DG1F/SS) in NORMAL.
- **REMOVE** Backup Flash fuses installed in 2E12.

12.2 **ENSURE** the following breakers closed:

12.2.1 Breakers in 2D23:

- 2C107 Power Supply (2D23-2)
- 2E11 Power Supply (2D23-6)
- 2C107 Power Supply (2D23-8)

12.2.2 Breakers in EDG room:

- 2E11 Breaker located inside 2E12 (D2E11NA6)
- Breaker inside 2C107 (D2C107NA8)
- Breaker inside 2C107 (D2C107NA2)

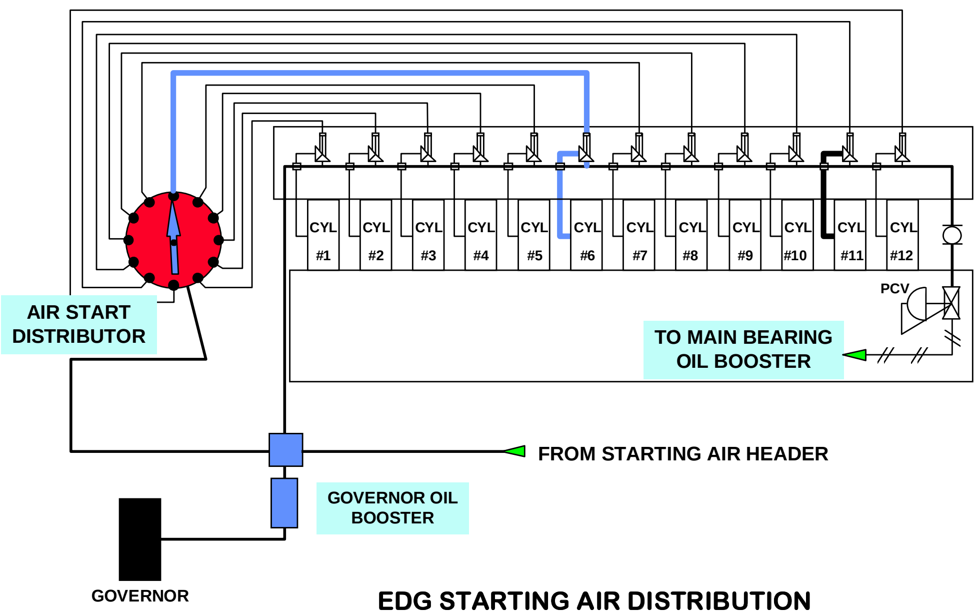
2.2.2 Air Admission To The Engine

To admit starting air to the engine cylinders to cause crankshaft rotation the following components are utilized.

1. Air Start Header Solenoid Valves
2. The Air start Distributors.
3. The Spring Loaded Air Admission Valves

2.2.2.1Air Start Header Isolation and Vent Solenoid Valves

The air start header is pressurized when the Air Start Header Solenoid Valves are energized by the Engine Start Control Scheme. The air start header solenoid operated isolation valves are powered from vital station 125 VDC from the start circuit.



EDG STARTING AIR DISTRIBUTION

The air start header solenoid operated vent valves are powered from vital 125 VDC from the stopping control scheme. When the engine start opens the header isolation solenoid valves it also energized and closes the air start header vent solenoid valve. When the engine is running on fuel as detected by speed relays (>250 rpm or >8 psig jacket cooling water pump discharge pressure) the solenoid operated vent valve is deenergized and opens.

2.2.2.2 Air Start Header Loads

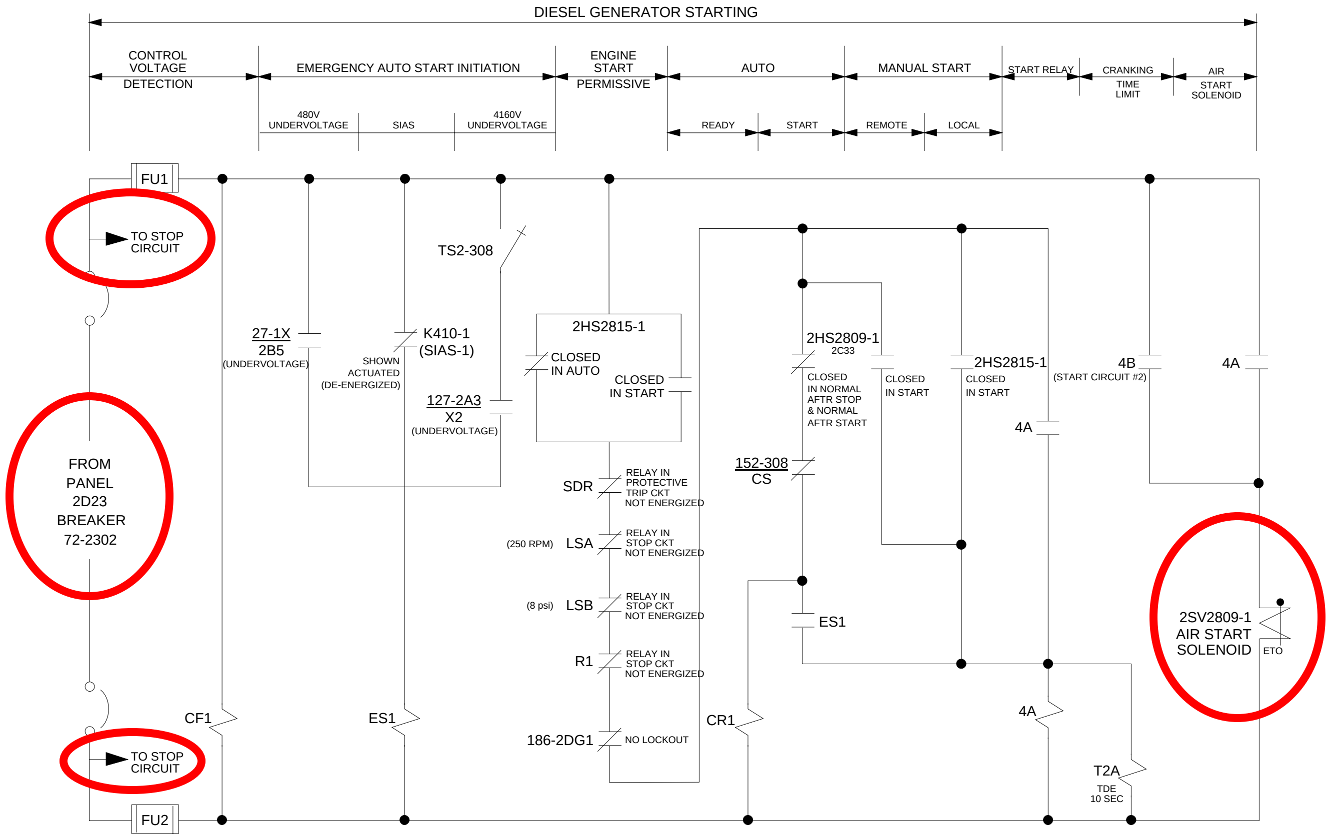
The starting air header is shown above and applies air to the following components when it is pressurized by opening the solenoid operated isolation valves.

1. Hydraulic Boost Relays for the governor
2. Lube Oil Boost to the Generator bearing on the lower engine crank.
3. Air Start Distributor.
4. Engine Air Admission Spring Loaded Check Valves

3.3 Engine Control Schemes

Each relay in each of the following control circuits is considered to be a component.

3.3.1 Engine Start Circuit



EMERGENCY DIESEL GENERATOR 2K4A (2DG1) START CIRCUIT #1 (REDUNDANT)

The following is a brief discussion of the sequence of events that take place in the Start Circuit during an automatic start of the Emergency Diesel Generator. A more detailed discussion of each of the relays shown on the illustration above immediately follows this section.

An emergency start signal energizes the Emergency Start Relays ES1 and ES2. Normally open contacts ES1 & ES2 close to energize the Start Relays 4A & 4B and the Cranking Time Limit Relays T2A & T2B.

Normally open contacts of the Start Relays 4A & 4B close to energize the Air Start Solenoid valves. These solenoids open their respective Starting Air Valves and pressurize the Starting Air header. The following occur when these valves open:

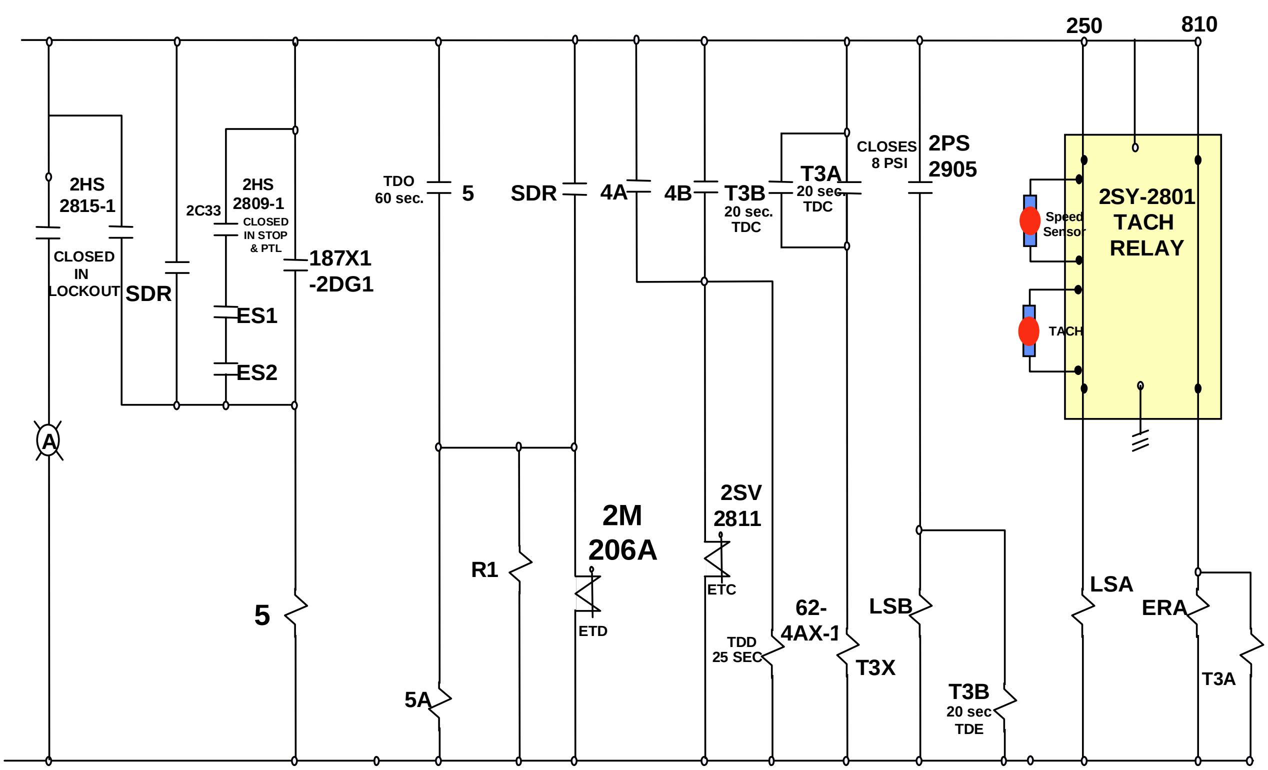
- Air pressure causes the Main Bearing Booster accumulator to dump oil to the last bearing on the lower crankshaft. The purpose of this oil shot is to provide lubrication as soon as the engine starts to turn. This is necessary because this bearing carries more weight than the other bearings on the

3.3.2 Engine Stop Control Circuit

Refer to the Engine Stop circuit shown.

The following is a brief discussion of the sequence of events that take place in the Stop Circuit to stop the Emergency Diesel Generator. A more detailed discussion of each of the relays shown on the following illustration follows later in this section.

The Stop Circuit uses the governor to secure fuel to the engine, thus stopping it. The basic technique is to dump the hydraulic output of the Governor back to its hydraulic reservoir which will allow spring pressure to drive the fuel racks to the NO FUEL position.



Placing the Control Room handswitch for a running Diesel Generator to the STOP position closes a normally open contact to energize the Stopping Relay, designated the 5 Relay.

The **5 Relay** can be energized by any of the following.

- Differential Relay Trip
- Local Handswitch to the "Lockout" position
- Control Handswitch to Stop or Pull-to-Lock. (Either Emergency Start Relay being energized will block this handswitch from energizing the 5 Relay.)
- The Engine Shutdown Relay

When the **5 Relay** is energized, normally open contacts close to energize the **5A Relay,** the **R1 Relay** and the Governor Shutdown

Solenoid (**ETO**) in the Governor. This contact is a 60 second time delay open (TDO) contact. This means that when the Stopping Relay (5) is de-energized, this contact remains closed for an additional 60 seconds. There is a similar contact in the Lube Oil Protection circuit that is a time delay close (TDC) contact. This contact immediately opens when the 5 Relay is energized and will not close until 60 seconds after the 5 Relay is de-energized. The time delay for the two contacts associated with the 5 Relay ensure that the functions they cause to happen are maintained for 60 seconds after the 5 Relay is de-energized. This allows the engine to coast down and come to a stop.

An example of the above discussion would be taking the Engine handswitch in the Control Room to STOP and returning it to NORMAL. This would energize the 5 Relay for 1 or 2 seconds and then it would de-energize when the handswitch returns to NORMAL. The two contacts associated with the 5 Relay would remain in the "shutdown" position for 60 seconds. The engine will coast to a stop in about 45-50 seconds.

3.3.2.1 Stop Circuit Relay Discussion

Several relays make up the Stop Control circuit for the Emergency Diesel Generators. The discussion below addresses the relays in the EDG 2K4A Engine Stop Control circuit. The relays in the 2K4B Stop circuit are similar.

- **5 Relay**

This relay is energized if any of the following conditions are satisfied:

- Placing the local handswitch, 2HS-2815-1, in Lockout, or
- The generator differential current relay (187X1-2DG1) is energized, or
- The Shutdown Relay (in the Protective Trip circuit) is energized, or
- The Control Switch in the Control Room, 2HS-2809-1, is in Stop or pull-to-lock IF neither Emergency Start Relay, ES1 or ES2, is energized.

It should be noted that the 5 Relay is time delay de-energize. Once it is de-energized, some of its contacts do not change state for 60 seconds.. When energized, the 5 Relay changes the state of contacts that to initiate the following functions:

- a contact closes to energize the 5A and R1 relays and the Governor shutdown solenoid, 2M206A. The 5 contact is a TDO contact with a 60 second time delay. This contact will stay closed for 60 seconds after the 5 Relay has de-energized. This prevents spurious re-starts of the engine during an engine coast down.
- a contact closes which causes an Engine / Exciter Shutdown alarm in the Control Room.
- a contact opens to block the Trip Enable permissive for the Low Lube Oil Pressure relays (in the Protective Trip circuit) during an engine coast down. Recall that the 5 Relay is a 60 second time delay de-energize relay. The 5 contact in this circuit will remain open for 60 seconds after the 5 Relay de-energizes.

Question 10

Bank:	2340	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	10	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	4	**Taxonomy:**	H	**Source:**	NRC Exam Bank 1494		
Search	000025A112	**10CFR55:**	41.7	**Safety Function**	4				
Title:	Loss of Residual Heat Removal System (RHRS)	**System Number**	025	**K/A**	AA1.12				
Tier:	1	**Group:**	1	**RO Imp:**	3.6	**SRO Imp:**	3.5	**L. Plan:**	A2LP-RO-ASDC
OBJ	3								

Description: Ability to operate and/or monitor the following as they apply to the Loss of Residual Heat Removal System: - RCS temperature indicators

Question:

Given the following conditions:

* The plant is shutdown to replace a failed RCP seal.
* OP-1015.008, Unit 2 SDC Control, Attachment B, Verification of SDC System Alignment, has been completed.
* LPSI Pump 2P60A has been placed in service through SDC HX 2E-35A with the flows established during completion of OP-1015.008 Attachment B.
* The RCS has been drained to lowered inventory for seal replacement.
* RCS Temperature is 115°F and steady.

* NOW a loss of 125 VDC power to the SDC Temperature Control Valve 2CV-5093 solenoid causes the temperature control valve to go to its failed position.
* All other components in the SDC System remain the same as before the failure.

Based on these conditions, the SDC HX 2E-35 Inlet and Outlet RCS temperatures would ____________ as compared to a loss of power to 2CV-5093 without OP-1015.008 Attachment B having been completed.

A. lower slower

B. rise faster

C. rise slower

D. lower faster

Answer:

C. rise slower.

Notes:

C is correct. The SDC TCV 2CV-5093 will lose IA on a loss of power to its DC solenoid causing the valve to fail closed. However, OP 1015.008 Attachment B Step 7 throttles the SDC Temperature Control Valve 2CV-5093 Bypass Valve 2SI-5093-3 to ensure at least 75% (25% Less Flow than before the failure) of the flow from the SDC HX is available as a mitigation strategy should 2CV-5093 fail Closed in lowered inventory. This will cause the RCS and SDC HX inlet temperature to rise slowly since the temperature was steady before the failure.

A is incorrect because cooling flow is lowered thus temperatures would rise but plausible if the candidate believes cooling flow rises due to the mitigating strategy of using the FCV Bypass or believes the TCV fails open.

B is incorrect because there is still 75% of the flow going through the bypass so the temperature would not go up rapidly but plausible if the candidate does not remember the mitigating steps taken in OP 1015.008 Attachment B Step 7 and 2CV-5093 does fail Closed.

Distracter D is incorrect because the valve fails closed and cooling flow is lowered not raised but plausible if the candidate believes 2CV-5093 Fails "OPEN"

This question matches the K&A because there is a Loss of RHR event going on and the candidate is asked to monitor and determine the correct response of RCS Temperature indications going into the RHR cooler and exit temperature going to the RCS. When on SDC the on Unit 2, the SDC HX 2E-35 Inlet and outlet temperatures are the official RCS temperature indications as most normal RCS temperature indications will bottom out above 500 degrees or do not have flow past their temperature well.

References:

STM_2-14_14-1 SDC System Section 2.6 (Verified reference updated 11/10/16);
STM_2-14_14-1 SDC System SDC Drawing (Verified reference updated 11/10/16);
NOP 1015.008 Unit 2 SDC Control REV 54 Attachment B Step 7 (Verified reference updated 11/10/16); AOP 2203.029 Loss of SDC Rev 19 Step 14 (Verified reference updated 11/15/16)

Historical Comments:

NRC Exam Bank 1494 was used on the 2008 NRC Exam
To be used on the 2017 NRC Exam but altered the order of the answer from the last time the QID was given. Also deleted the reason for the change in temperature based on feedback from the peer review that this would allow answering the question based solely on knowing the fail position of 2CV-5093. Also altered the stem to include repeated words in the answer/distractors based on peer review feedback.

REV. 1 based on NRC Chief Examiner Feedback BNC. Change the stem to compare conditions between Attachment B being complete or not.

2.6 SDC Temperature Control Valve, 2CV-5093

SDC temperature control valve, 2CV-5093, is an 8 inch, air operated, butterfly valve located in the lower south piping penetration room. There is "NO" valve seat in 2CV-5093. It is not a leak tight valve and is not designed to be leak tight. It is basically a disk in a piece of pipe.

The position of this valve is controlled by hand indicating controller, 2HIC-5093, located in the control room on panel 2C04. This controller should be adjusted such that fully closed on the controller represents minimum leakage past the associated valve disk on 2CV-5093. The signal from the HIC going to the valve is manually adjusted by the operator to maintain desired temperature. This valve is used to control the flow of reactor coolant going through the SDC heat exchanger. The line taps into the LPSI discharge header, upstream of flow transmitter 2FT-5091.

Instrument air to the valve is controlled by solenoid valve 2SV-5093. This valve is controlled by a two position (ESF STBY-SDC) maintained key operated switch located on control room panel 2C04. The key for this switch is removable only in the SDC position. Power for this solenoid valve comes from 125V DC in panel 2C04 which is supplied by 125V DC distribution panel 2D22 breaker 12. When the handswitch is taken to ESF STBY, the solenoid valve is de-energized which blocks air to the valve and allows the air stored in a local volume tank to position the valve full closed. When the handswitch is taken to the SDC position, the solenoid valve is energized which aligns air to the valve allowing its position to be controlled by 2HIC-5093.

This valve fails closed on loss of instrument air. This is accomplished by a local trip valve and volume tank. As instrument air pressure goes down, between 58 and 75 psig, the trip valve is set to switch the source of operating air to the valve from instrument air to the volume tank. The air stored in the tank will cause the valve to go to its full closed position. Refer to the figure on page **51** for a schematic and description of the trip valve.

Another place this valve can be controlled is from hand indicating controller 2HIC-5093A at the remote shutdown panel, 2C80. This controller can be operated in either AUTO or MANUAL as selected by a slide bar on the lower portion of the controller. With the slide bar selected to MANUAL, demand output to 2CV-5093 is controlled by adjusting the knob on 2HIC-5093A. With the slide bar selected to AUTO, the demand output from 2HIC-5093, in the control room, is passed through the controller and sent to the valve. Due to this circuit, in order to have control of this valve from the control room, 2HIC-5093A must be in AUTO. The only place that operation of 2CV-5093 from 2C80 is discussed is OP-2203.030, Remote Shutdown.

2.6.1 Manual Operation of 2CV-5093

This valve also has the capability to be manually overridden and operated locally. This is performed in accordance with Exhibit 1 of OP-2104.040, LPSI System Operations. It is done, in part, by rotating the handwheel until the drilled hole in the exposed surface of

the worm gear hub aligns with the semicircular notch in the pointer then inserting a pin to engage the handwheel.

There are two cautions in the procedure for local operations of 2CV-5093.

- 2CV-5093 and 2SI-5093-3 discs do not have a seating surface in the valve body. Attempting to operate either valve after minimal resistance is felt in either direction will damage the valve operating gear teeth.
- Attempting to operate 2CV-5093 remotely with engaging pin installed will damage valve operating stem and gear teeth.

Because there is no hard stop for the disk in the valve seat the valve operator has a hard stop that limits travel of the operating gear. If this gear is operated into its stop with too much force the worm gear will strip the teeth from the operating gear.

2.6.2 Loss of Instrument Air Mitigation

OP-1015.008, Unit 2 SDC Control Attachment B, requires alignment of the manual valves associated with SDC flow control valve 2CV-5091 and temperature control valve 2CV-5093. For 2CV-5091 the inlet (2SI-5091-1) OR outlet (2SI-5091-2) is throttled. For 2CV-5093, the bypass (2CV-5093-3) is throttled.

The purpose of this manual valve alignment is to prevent vortexing due to excessive flow in the LPSI header due to 2CV-5091 failing open on a loss of instrument air. The bypass for 2CV-5093 is also throttled in order to prevent the loss of SDC flow through the SDC heat exchanger due to a loss of instrument air.

Refer to OP-1015.008, Unit 2 SDC Control Attachment B for the procedural requirements for this evolution.

2.7 Over Pressure Protection

This section discusses the over pressure protection provided for the SDC system piping.

2.7.1 Suction Line Relief Valve, 2PSV-5085

Suction line relief valve 2PSV-5085 is installed between the two suction line isolation valves in the containment. It protects against an overpressure condition caused by:

- Suction isolation valve 2CV-5084-1 leaking by (the one closest to the RCS).
- Thermal expansion of the fluid, trapped between the two valves, due to a heatup.

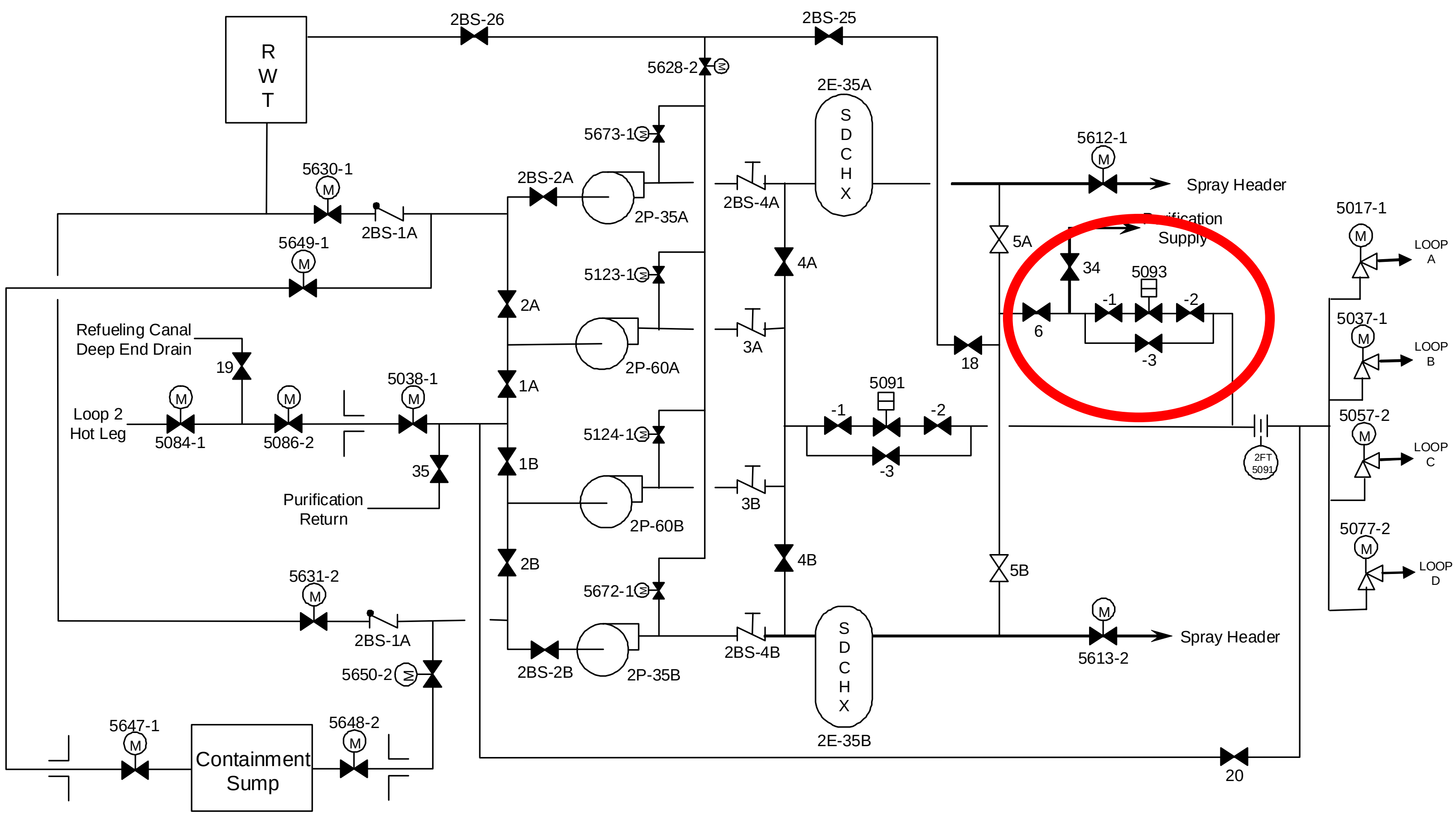
Its setpoint is 2485 psig and it relieves to the reactor drain tank (RDT).

2.7.2 Suction Line Relief Valve, 2PSV-5087

Suction line relief valve 2PSV-5087 is installed between the two suction line isolation valves on either side of the containment penetration. It protects against an overpressure condition caused by:

- Both suction isolation valves in the containment leaking by (2CV-5084-1 and 2CV-5086-2).
- Thermal expansion of the fluid trapped between the two valves due to a heatup.

Figures



Shutdown Cooling System

All valves, unless otherwise noted, start with 2SI-.
All MOVs and AOVs start with 2CV-.

7.0 IF any of the following conditions apply:

- "Commencement of Shutdown Cooling" section of Shutdown Cooling System (2104.004) in progress.
- 1015.008 Attachment D, LOWERED INVENTORY Checklist in progress.
- It is desired to adjust valves due to changing heat load.

THEN throttle manual valves as follows:

7.1 Adjust 2CV-5091 AND 2CV-5093 to desired flow and record:

SDC HX Flow ______________ Total Flow ______________
(F5610/F5616) (F5091)

7.2 Simultaneously close 2CV-5093 AND throttle 2SI-5093-3 until 2CV-5093 is closed and SDC HX flow is 75% of flow in step 7.1.

NOTE
2SI-5091-2 is preferred throttle valve due to flow affects on 2CV-5091.

7.3 Simultaneously open 2CV-5091 and throttle 2SI-5091-1 or 2SI-5091-2 until total SDC flow is within the following limits with 2CV-5091 open and 2CV-5093 closed:

- 2100 gpm minimum flow
- Within acceptable region of "Maintenance Levels and SDC Vortex Curve", Exhibit 1 of Draining the Reactor Coolant System (2103.011)
- 4800 gpm maximum flow with level above LOWERED INVENTORY.

7.4 Restore 2CV-5091 and 2CV-5093 to desired positions and verify flow is within acceptable region of "Maintenance Levels and SDC Vortex Curve", Exhibit 1 of Draining the Reactor Coolant System (2103.011).

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE
SDC HX Return valve (2CV-5093) fails closed on loss of IA.

*14. **CHECK** adequate SDC HX flowrate for RCS temperature control as follows:

- Normal in-service SDC HX CNTMT Spray Header Flowrate by ANY of the following:
  - 2FIS-5610
  - 2FIS-5616
  - SPDS-SFD
- IA pressure greater than 65 psig.

*14. **PERFORM** the following:

A. **<u>IF</u>** low flowrate due to SDC HX RETURN VALVE (2CV-5093) closure, **<u>THEN</u> PERFORM** EITHER of the following to control RCS temperature:

- **THROTTLE** open "SDC HX OUTLET TO SDC FLOW CONTROL BLOCK BYPASS" valve (2SI-5093-3).
- Locally **OPERATE** "SDC HX RETURN VALVE" (2CV-5093) using 2104.040, LPSI System, Exhibit 2.

B. **<u>IF</u>** high flowrate due to SDC HX RETURN VALVE (2CV-5093) failure, **<u>THEN</u> PERFORM** EITHER of the following to control RCS temperature:

- **THROTTLE** "SDC HX OUTLET TO SDC FLOW CONTROL BLOCK INLET" valve (2SI-5093-2).
- Locally **OPERATE** "SDC HX RETURN VALVE" (2CV-5093) using 2104.040, LPSI System, Exhibit 2.

C. **MAINTAIN** SDC flow greater than 2400 gpm.

* D. **CONTROL** RCS cooldown rate within TS limits, **REFER TO** 2202.010 Attachment 8, RCS Cooldown Table.

ORIGINAL BANK QUESTION 1494 WAS USED TO DEVELOP 2017 NRC EXAM QID#10 (Used on the 2008 NRC Exam)

Questions For All QID In Exam Bank

Bank: 1494	**Rev:** 0	**Rev Date:** 11/7/2004	**QID #:** 7	**Author:** COBLE			
Lic Level: R	**Difficulty:** 3	**Taxonomy:** H	**Source:** NEW				
Search 000025A112	**10CFR55:** 41.7 / 45.5 / 45.6	**Safety Function** 4					
System Title: Loss of Residual Heat Removal System (RHRS)	**System Number** 025	**K/A** AA1.12					
Tier: 1	**Group:** 1	**RO Imp:** 3.6	**SRO Imp:** 3.5	**L. Plan:** A2LP-RO-SDCC	**OBJ** 4		

Description: Ability to operate and/or monitor the following as they apply to the Loss of Residual Heat Removal System: - RCS temperature indicators.

Question:

Given the following conditions:

* The plant is shutdown to replace a failed RCP seal.
* OP 1015.008, Unit 2 SDC Control, Attachment B, Verification of SDC System Alignment, has just been completed.
* SDC Pump 2P60A is in service through SDC HX 2E-35A with the same flows established during completion of OP 1015.008 Attachment B.
* The RCS is currently in reduced inventory
* RCS Temperature is 115°F and steady.
* Now a loss of 125 VDC power to the SDC Temperature Control Valve 2CV-5093 solenoid causes the temperature control valve to go to its failed position.
* All other components in the SDC system remain the same as before the failure.

Which of the following would be the effect on RCS Temperature?

A. RCS temperature would rise slowly due to approximately 25% loss of flow through 2E-35A.

B. RCS temperature would rise rapidly with a loss of cooling due to 2CV-5093 failing full closed.

C. RCS temperature would drop slowly due to approximately 25% additional flow through 2E-35A.

D. RCS temperature would drop rapidly with much more cooling due to 2CV-5093 failing full open.

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☑	☑

Audit Exam History

2006	☐
2008	☐

Answer:

A. RCS temperature would rise slowly due to approximately 25% reduction of flow through 2E-35A.

Notes:

2CV-5093 will lose IA on a loss of power to its DC solenoid causing the valve to fail closed. However, OP 1015.008 Attachment B Step 6.2 throttles the SDC Temperature Control Valve 2CV-5093 Bypass Valve 2SI-5093-3 to ensue at least 75% of the flow from the SDC HX is available as a mitigation strategy should 2CV-5093 fail Closed. This makes answer A correct.

Distracter B is incorrect because there is still 75% of the flow going through the bypass so the temperature would not go up rapidly.

Distracter C is incorrect because cooling flow is lowered not raised.

Distracter D is incorrect because cooling flow is lowered not raised.

References:

STM 2-14, SDC System, Section 2.6 and 2.6.2.
OP 1015.008, SDC Control, Attachment B, Steps 6.1. and 6.2
AOP 2203.029, Loss of SDC, Step 9

Question 11

Bank:	2341	**Rev:**	2	**Rev Date:**	10/12/2016	**2017 TEST QID #:**	11	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	0000552119	**10CFR55:**	41.10	**Safety Function**	6				
Title:	Loss of Offsite and Onsite Power (Station Blackout)	**System Number**	055	**K/A**	2.1.19				
Tier:	1	**Group:**	1	**RO Imp:**	3.9	**SRO Imp:**	3.8	**L. Plan:**	A2LP-RO-ESBO
OBJ	1								
Description:	Conduct of Operations - Ability to use plant computers to evaluate system or component status.								

Question:

(1 PAGE OF SPDS SAFETY FUNCTION DISPLAY (SFD) CHANNEL 1 SCREEN ATTACHED)

Given the following:

* The plant has tripped from 100% power due to a transient.
* The current indications in the attached SPDS SFD screen exist after SPTAs are complete.
* SPDS SFD Channel 2 indications are the same as Channel 1.
* The Thot and Tcold temperatures have a lowering trend.

Referring to the attached SPDS indications, the correct recovery procedure would be ______________________ and proper Natural Circulation conditions ___________ been established.

A. Natural Circulation Operations AOP; have not

B. Station Blackout EOP; have

C. Natural Circulation Operations AOP; have

D. Station Blackout EOP; have not

Answer:

D. Station Blackout EOP; have not

Notes:

D is correct as both Vital 4160 Volt Buses 2A3 and 2A4 are deenergized and 6900 KV busses 2H1 and 2H2 are de-energized meaning no RCPs are running based on the given SPDS indications however, Natural Circulations (NC) conditions have not been established per the Station Blackout EOP Step 13 due to THOT and Average CET temperature being > 10 degrees apart (4th bullet).

A is incorrect because no vital 4160 VAC bus are energized at the end of SPTAs but plausible because RCPs are secured (No power on 2H1 and 2H2) and NC conditions are not met. Also plausible if only a Loss of Forced Circulation was in progress because under these condition only, EOP 2202.010 Exhibit 8 (Diagnostic Flow Chart) would direct entry into Natural Circulation AOP.

B is incorrect because NC conditions are not met but plausible because this in the correct recovery procedure based on the plant conditions shown on the attachment.

C is incorrect because NC conditions are not met and no vital 4160 VAC bus are energized at the end of SPTAs which will direct entry into the Station Blaclkout procedure but plausible if only a Loss of Forced Circulation event was in progress. In this case Exhibit 8 (Diagnostic Flow Chart would direct entry into Natural Circulation AOP.

This question matches the K&A as it requires the candidate to access the status of the electrical power to plant busses and establishment of proper Natural Circulation conditions using the plant computer and determine the appropriate procedure

and system status.

References:

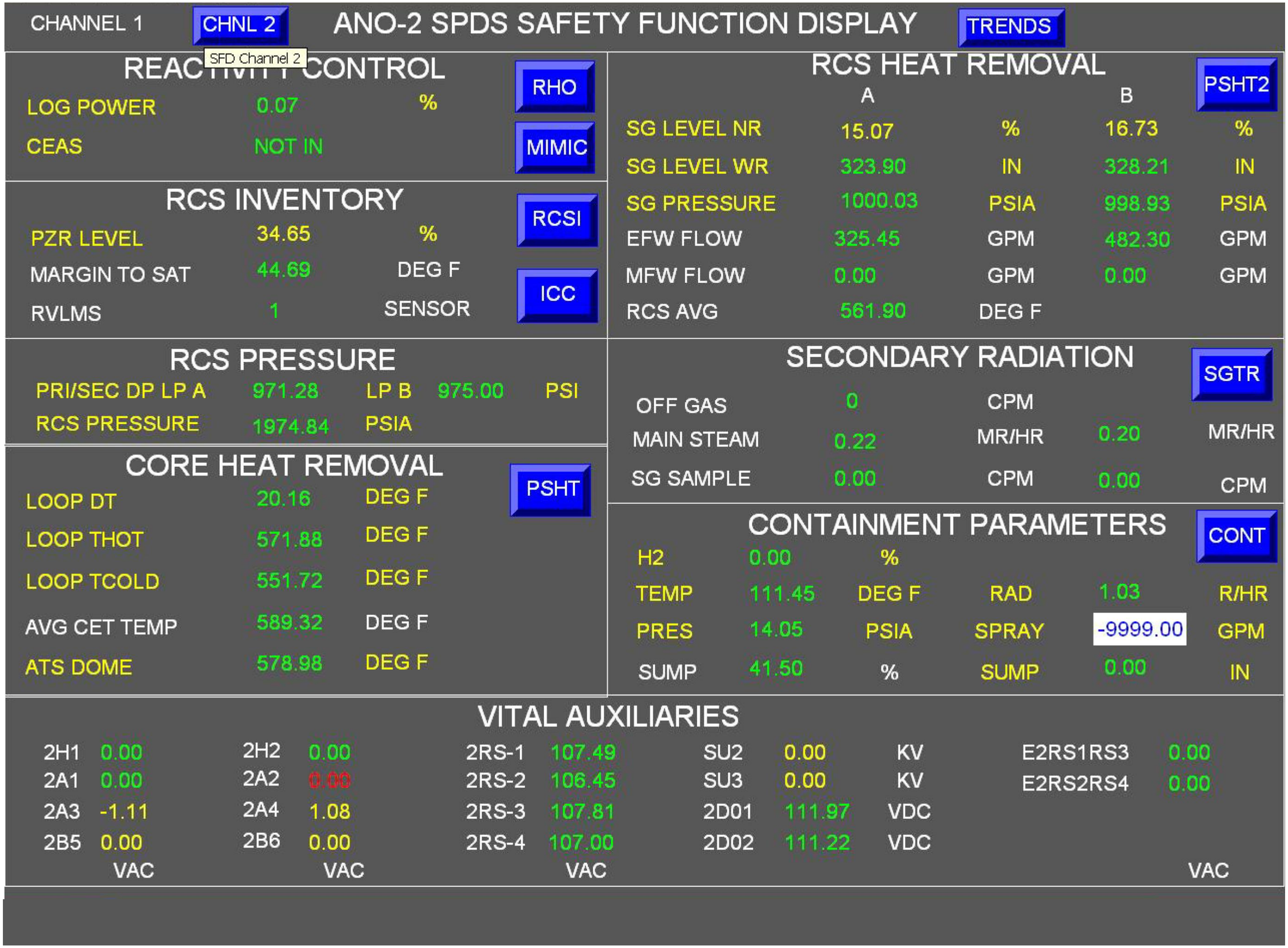
Attachment of SPDS Screen of and SFD (Safety Function Display) Channel 1(Verified reference updated 11/10/16); EOP 2202.010 Standard Attachments Rev. 23 Exhibit 8 Diagnostic Actions (Verified reference updated 11/10/16); EOP 2202.008 Station Blackout REV. 13 Entry Conditions and Step 13 (Verified reference updated 11/10/16); AOP 2203.013, Natural Circulation Operations, Rev. 16 (Verified reference updated 11/10/16).

Historical Comments:

To be used on the 2017 NRC Exam
Rev, 1 Question modified based on feedback from the NRC chief examiner (10 question preliminary review) due to non credible distractors. Modified the question to remove the Loss of Offsite Power non-credible distractors and any reference to starting the AACDG. The questions was written to review the Safety Function Display SFD Channel 1 to assess the plant conditions only prior to making a decision to go to either the Station Blackout Recovery procedure or Natural Circulation AOP at the end of SPTAs. The SFD Channel 1 indications will be incorporated as part of the question. ROs must be able to recognize entry conditions for EOP/AOP and be able to prioritize the EOP over the AOP. Also removed references to vital and no-vital buses.

Rev. 2 revised the second part to "proper Natural Circulations conditions have/have not been established " instead of " and Natural Circulation conditions are/are not met based on feedback from the NRC Chief examiner.

ATTACHMENT FOR QID #11



LOOP THOT and AVE CET TEMP ARE NOT < 10 DEGREES

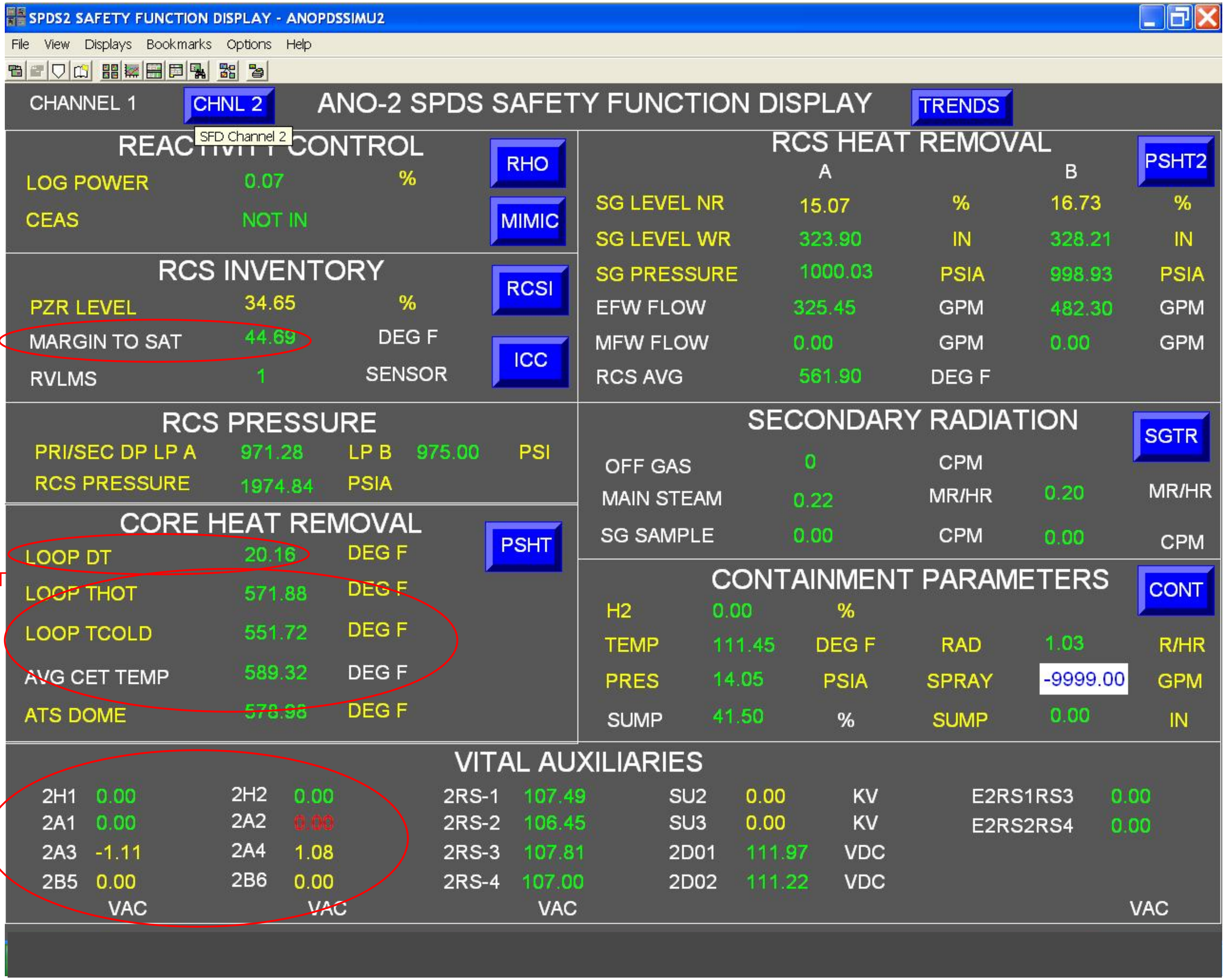
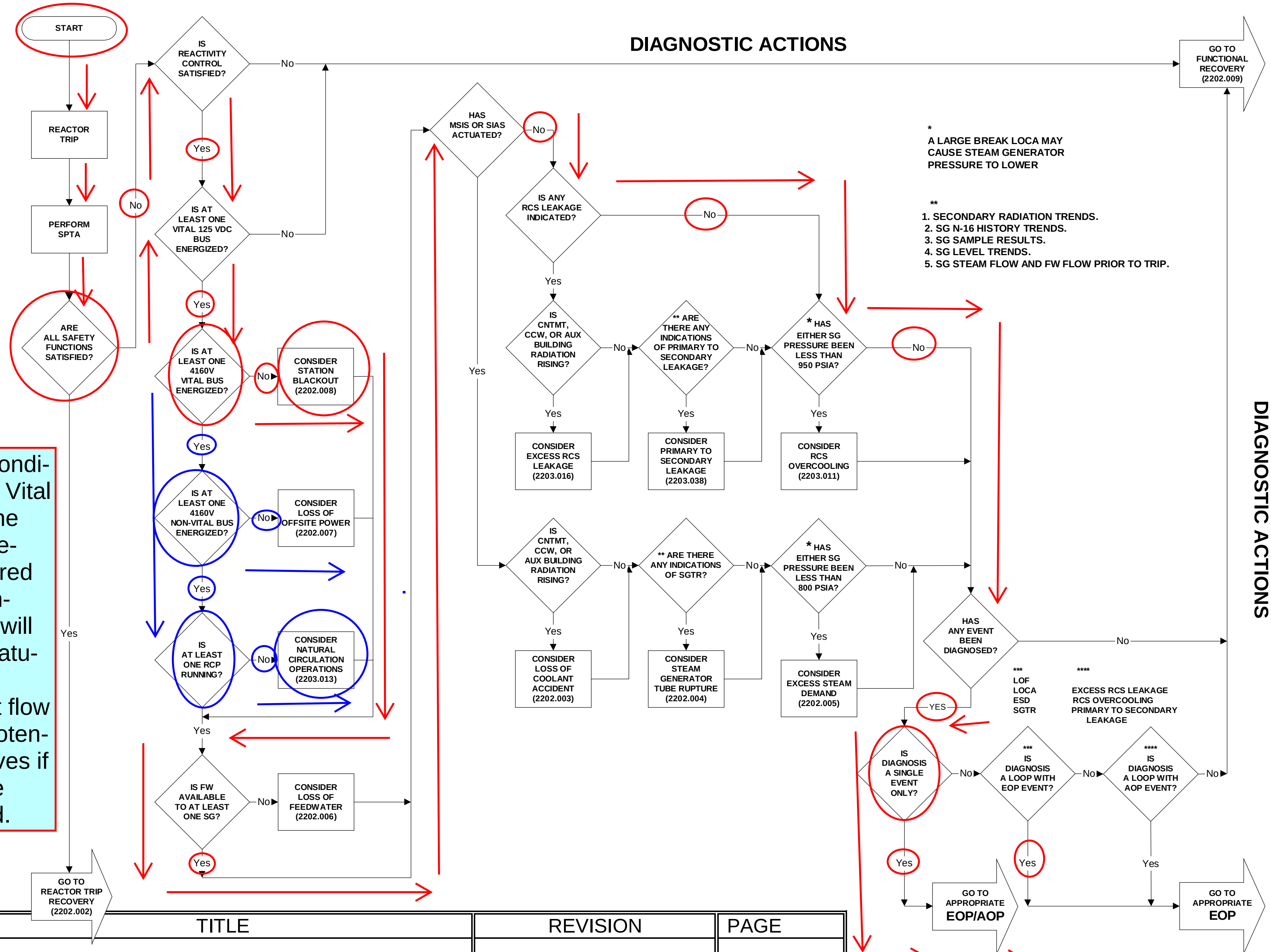


EXHIBIT 8



Natural Circulations conditions exist but with no Vital Bus Energized then the Station Blackout procedure which is considered first in order will be entered. The SBO EOP will check for adequate Natural Circulation.
Red line is the correct flow path. Blue lines are potential plausible alternatives if the vital and/or off site buses were energized.

STATION BLACKOUT

PURPOSE

This procedure provides actions for Station Blackout (SB).

ENTRY CONDITIONS

BOTH 4160v Vital buses 2A3 AND 2A4 de-energized.

EXIT CONDITIONS

ANY of the following conditions exist:

1. ANY SFSC acceptance criteria NOT satisfied.
2. At least ONE 4160v Vital bus energized from a diesel (EDG or AACG).
3. All appropriate actions completed.

INSTRUCTIONS	CONTINGENCY ACTIONS

12. (continued)

D. Check at least ONE DG running and energizing associated 4160v Vital bus.

D. IF BOTH DGs unavailable, THEN perform the following:

1) Start AACG AND align to associated 4160v Vital bus using 2104.037, Alternate AC Diesel Generator Operations, Attachment E.

2) IF AACG started AND EITHER 4160v Vital bus 2A3 OR 2A4 energized, THEN **GO TO** 2202.007, Loss of Offsite Power.

3) IF BOTH 4160v Vital buses de-energized, THEN **GO TO** Step 13.

E. **GO TO** 2202.007, Loss of Offsite Power.

NOTE

Plant response will be delayed during natural circulation due to longer loop cycle times.

*13. **Check natural circulation conditions established in at least ONE loop by ALL of the following:**

- Loop ΔT less than 50°F.
- T_H and T_C constant or lowering.
- RCS MTS 30°F or greater.
- ΔT between T_H and average CETs less than 10°F.

*13. **IF natural circulation can NOT be confirmed, THEN verify the following:**

A. RCS heat removal with SDBCS or MSSVs.

B. SG levels being maintained 10% to 90%.

14. **Perform 2203.002, Spent Fuel Pool Emergencies, in conjunction with this procedure. (IER L1 11-2 Rec 5)**

NATURAL CIRCULATION OPERATIONS

PURPOSE

This procedure provides actions for Natural Circulation Operations.

ENTRY CONDITIONS

RCS forced circulation lost.

EXIT CONDITIONS

WHEN ONE of the following conditions exist,
THEN exit this procedure:

1. At least ONE RCP restarted.

2. Shutdown cooling established.

3. ANY SFSC acceptance criteria NOT satisfied.

Question 12

Bank:	2342	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	12	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	000057K301	**10CFR55:**	41.10	**Safety Function**	6				
Title:	Loss of Vital AC Electrical Instrument Bus	**System Number**	057	**K/A**	AK3.01				
Tier:	1	**Group:**	1	**RO Imp:**	4.1	**SRO Imp:**	4.4	**L. Plan:**	A2LP-RO-ESBO
OBJ	2								

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: - Actions contained in EOP for loss of vital ac electrical instrument bus

Question:

Given the following:

* Unit 2 has experienced a Station Blackout.
* The EOP Station Blackout OP-2202.008 has been implemented.
* Station Blackout conditions are expected to last for 2.6 hours
* The CRS directs Standard Attachment 25, "Load Shedding of Vital Battery Loads," be performed.

Standard Attachment 25 requires that 120 VAC Vital Instrument Buses __________should be secured to ____.

A. 2RS3 & 2RS4; prevent ESF actuations upon restoration of AC power

B. 2RS3 & 2RS4; reduce the loads on the unit Vital 125 VDC batteries

C 2RS1 & 2RS2; prevent ESF actuations upon restoration of AC power

D. 2RS1 & 2RS2; reduce the loads on the unit Vital 125 VDC batteries

Answer:

B. 2RS3 & 2RS4; reduce the loads on the unit Vital 125 VDC batteries

Notes:

B is correct as directed by step 2.C of EOP 2202.010 Standard Attachment 25, Load Shedding of Vital Battery Loads and the reason is correct per the name of the attachment and EOP 2202.008 Station Blackout TG Rev 13 Section 1 Step 20. This decision is made only after the Station Blackout has been going on two hours because there are attempts being made during this time frame to restore power and performing this step will cause a full ESFAS actuation in the logic matrix. Power should typically be restored prior to 2 hours and this would cause unnecessary safety related pump starts if the RS buses were de-energized earlier.

A is incorrect because these channels would cause a Full ESFAS actuation but plausible as the reason could be determined to be correct if the candidate does not correctly understand the ESF actuation logic. The candidate could assume that with two channels out ,the actuation logic could be 1 of 2 vice 2 of 4 similar to PPS if one channel is in bypass and one in trip.

C. Is incorrect due to the wrong RS buses, and reason but plausible as 2RS1 and 2RS2 will also reduce vital battery loads however 2R1 and 2R2 have the majority of the post accident and remote SD instrumentation available so 2RS3 and 2RS4 are the selected buses to secure only after 2 hours into the event.

D. Is incorrect due to the wrong RS buses, but plausible as 2RS1 and 2RS2 will also reduce vital battery loads however 2R1 and 2R2 have the majority of the post accident and remote SD instrumentation available so 2RS3 and 2RS4 are the selected buses to secure only after 2 hours into the event and this is the correct reason for RS Bus removal .

This question matches the K&A because it requires knowledge of the EOP actions associated with removing Vital AC

Instrument busses from service and the reason for the EOP Actions.

References:

EOP 2202.010 Standard Attachment 25, Load Shedding of Vital Battery Loads, REV 23 Step 2.C (Verified reference updated 11/10/16);
EOP 2202.008 Station Blackout TG Rev 13 Section 1 Step 20 (Verified reference updated 11/10/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Removed the number "2" from the stem. Updated distractor analysis for the correct answer.

ATTACHMENT 25
LOAD SHEDDING OF VITAL BATTERY LOADS

1. IF Blackout expected to exceed 60 minutes (one hour),
   THEN de-energize PMS inverter (2Y25) using "De-energizing PMS Inverter 2Y25 During Blackout" Exhibit 17 of 2107.003, Inverter and 120 VAC Electrical System Operation.

2. IF Blackout expected to exceed two hours,
   THEN perform the following:

   A. Locally open RAS Sump and RWT Valve breakers:

      - 2B52-G3 "CNTMT SUMP SUCTION ISOL 2CV-5649-1"
      - 2B52-E4 "RWT 2T3 OUTLET 2CV-5630-1"
      - 2B62-G3 "CNTMT SUMP SUCTION ISOL 2CV-5650-2"
      - 2B61-F3 "RWT 2T3 OUTLET 2CV-5631-2"

   B. Place the following pump handswitches in PTL:

      - HPSI Pump 2P89A (2HS-5078-1)
      - HPSI Pump 2P89B (2HS-5079-2)
      - HPSI Pump 2P89C (2HS-5080-1)
      - HPSI Pump 2P89C (2HS-5080-2)
      - LPSI Pump 2P60A (2HS-5018-1)
      - LPSI Pump 2P60B (2HS-5019-2)
      - CNTMT Spray Pump 2P35A (2HS-5623-1)
      - CNTMT Spray Pump 2P35B (2HS-5626-2)

(Step 2 continued on next page)

CAUTION
Performance of this step will cause full ESFAS actuation.

C. Secure the following inverters:

1) Secure Inverter supplying 120v Vital AC bus 2RS3 (2Y13 or 2Y1113) using "De-energizing 2RS3 (Securing 2Y13/2Y1113) During Blackout" exhibit of 2107.003, Inverter and 120 VAC Electrical System Operation.

2) Secure Inverter supplying 120v Vital AC bus 2RS4 (2Y24 or 2Y2224) using "De-energizing 2RS4 (Securing 2Y24/2Y2224) During Blackout" exhibit of 2107.003, Inverter and 120 VAC Electrical System Operation.

D) Establish control of EFW pump (2P7A) as follows:

1) IF 2P7A tripped on overspeed,
THEN establish local control using 2106.006, Emergency Feedwater Operations.

2) IF 2P7A running,
THEN use Speed controller (2HIC-0336-2) to control SG levels.

3. IF Blackout expected to exceed six hours,
THEN perform the following:

A. Locally de-energize 2P7A "EFW Discharge Isolation" valves by opening the following breakers:

- 2D26-A4 "EFW 2P-7A DISCH to S/G A 2CV-1026-2"
- 2D26-C1 "EFW 2P-7A DISCH to S/G B 2CV-1076-2"
- 2D27-B1 "EFW 2P-7A DISCH to S/G A 2CV-1037-1"
- 2D27-B2 "EFW 2P-7A DISCH to S/G B 2CV-1039-1"

B. Manually control 2P7A using 2106.006, Emergency Feedwater System Operation Exhibit 3, Manual Control of 2P7A.

EOP STEP - Section 1 – DG Operations:

20. **IF FDS-002, Unit 2 Extended Loss of AC Power NOT in progress,**
THEN reduce loads on Vital batteries using 2202.010 Attachment 25, Load Shedding of Vital Battery Loads.

EPG STEP :

10.

DEVIATION? No

BASES FOR DEVIATION:

The EPG does not delineate between vital and non-vital DC loads. This step addresses only vital DC loads using a plant specific methodology allowed by the EPG. Non-vital DC loads are addressed in a subsequent step.

BASIS FOR STEP:

The intent of this step is to strip unnecessary loads off the DC buses to conserve battery capacity. During a station blackout event, the DC loads will be supplied by the vital station batteries. The battery chargers will not be available to maintain the batteries charged to capacity. Therefore, until power is restored, unnecessary loads should be stripped off the DC buses to conserve DC power.

If a SBO is expected to exceed 1 hour (5), then direction is given to secure the plant computer using the appropriate procedure (1). The plant computer is not required to maintain vital instrumentation; therefore, it can be secured within 1 hour of the event. This will not impact the ability to restore power.

If a SBO is expected to exceed 2 hours, then direction is given to perform a Standard Attachment (2). Performing this attachment will actuate all ESF systems; therefore, this should only be performed when the blackout is expected to exceed 2 hours.

If a SBO is expected to exceed 6 hours, then direction is given to locally de-energize and open EFW discharge isolation valves and take manual control of 2P7A using the appropriate procedure (3). A plant-unique EFW control valve power supply and control scheme can, under certain failure combinations, result in isolation of the EFW discharge lines to both steam generators from the only operating EFW pump. This isolation potential exists because the outboard EFW flow control valves work in parallel with the inboard EFW flow control valves, but are powered from the opposite power supply than the EFW pump and the inboard EFW flow control valves.(4) In order to proceed with a cooldown of the plant to cold shutdown conditions, de-energizing and opening the isolation valves along with taking manual control of 2P7A ensures that control of steam generator levels will be maintained if vital DC power is lost after 8 hours of SBO conditions.

Question 13

Bank:	2343	**Rev:**	0	**Rev Date:**	8/30/2016	**2017 TEST QID #:**	13	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	4	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	000015K207	**10CFR55:**	41.7	**Safety Function**	4				
Title:	017 Reactor Coolant Pump (RCP) Malfunctions	**System Number**	015	**K/A**	AK2.07				
Tier:	1	**Group:**	1	**RO Imp:**	2.9	**SRO Imp:**	2.9	**L. Plan:**	A2LP-RO-ARCP
OBJ	3/4								

Description: Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: - RCP seals

Question:

Given the following at 100% Power:

* Annunciator 2K11-G3 "RCP BLEEDOFF FLOW HI/LO" comes into alarm.
* RCP 2P-32B Controlled Bleedoff (CBO) flow is 1.8 gpm on PMS and rising.
* RCP 2P-32B Seal Pressure indications are as follows:
  - Middle Seal Pressure is 1800 psia.
  - Upper Seal Pressure is 1575 psia.
  - Vapor Seal Pressure is 1530 psia.
* All other RCPs are operating with normal indications.
* OP-2203.025, RCP EMERGENCIES has been entered.

Which of the following is correct concerning RCP 2P-32B seals and the correct procedural action to take?

A. ONLY the upper seal has failed; Commence a plant shutdown and secure RCP 2P-32B after the Reactor is tripped.

B. The middle and upper seals have failed; Commence a plant shutdown and secure RCP 2P-32B after the Reactor is tripped.

C. ONLY the upper seal has failed; the reactor should be tripped from 100% power and then secure RCP 2P-32B.

D. The middle and upper seals have failed; the reactor should be tripped from 100% power and then secure RCP 2P-32B.

Answer:

C. ONLY the upper seal has failed; the reactor should be tripped from 100% power and then secure RCP 2P-32B.

Notes:

C is correct based on the only seal that is completely failed is the upper seal at less than 50 PSID (1575 - 1530) = 45 psid (Seal Failure is defined in the RCP Emergency AOP 2203.025 Step 6 and the RCP STM 2-03-2 Section 1.5.1). The lower seal DP is RCS pressure 2200-1800 = 400 psid thus not failed. The middle seal DP is 1800 - 1575 = 225 PSID thus not failed. The upper seal failure has caused the Vapor Seal pressure to be greater than 1500 psia which is reactor trip criteria per RCP Emergency AOP 2203.025 Step 5. The Vapor seal is only designed to withstand full system pressure when coasting down or stopped. Therefore, the Reactor must be tripped first as an automatic trip will occur if the RCP is secured first then the affected RCP is required to be secured..

A is incorrect because the plant needs to trip based on Vapor Seal pressure and EOP Step 5.A (Vapor Seal Greater than 1500 psia and a plant shutdown is not directed in EOP Step 6.A but plausible as only one seal has failed.

B is incorrect as only one seal has failed but plausible as another seal is degraded and the correct AOP Contingency Action Step 6.B is provided for two seal failures. However, the vapor seal high pressure takes precedence over the normal plant shutdown and a reactor trip from 100% power should be taken.

D is incorrect as only one seal has failed but plausible as two other seals are degraded (Lower and Vapor) and the correct contingency action is listed for three seals failed per CA Step 6.C is a reactor trip from 100% power.

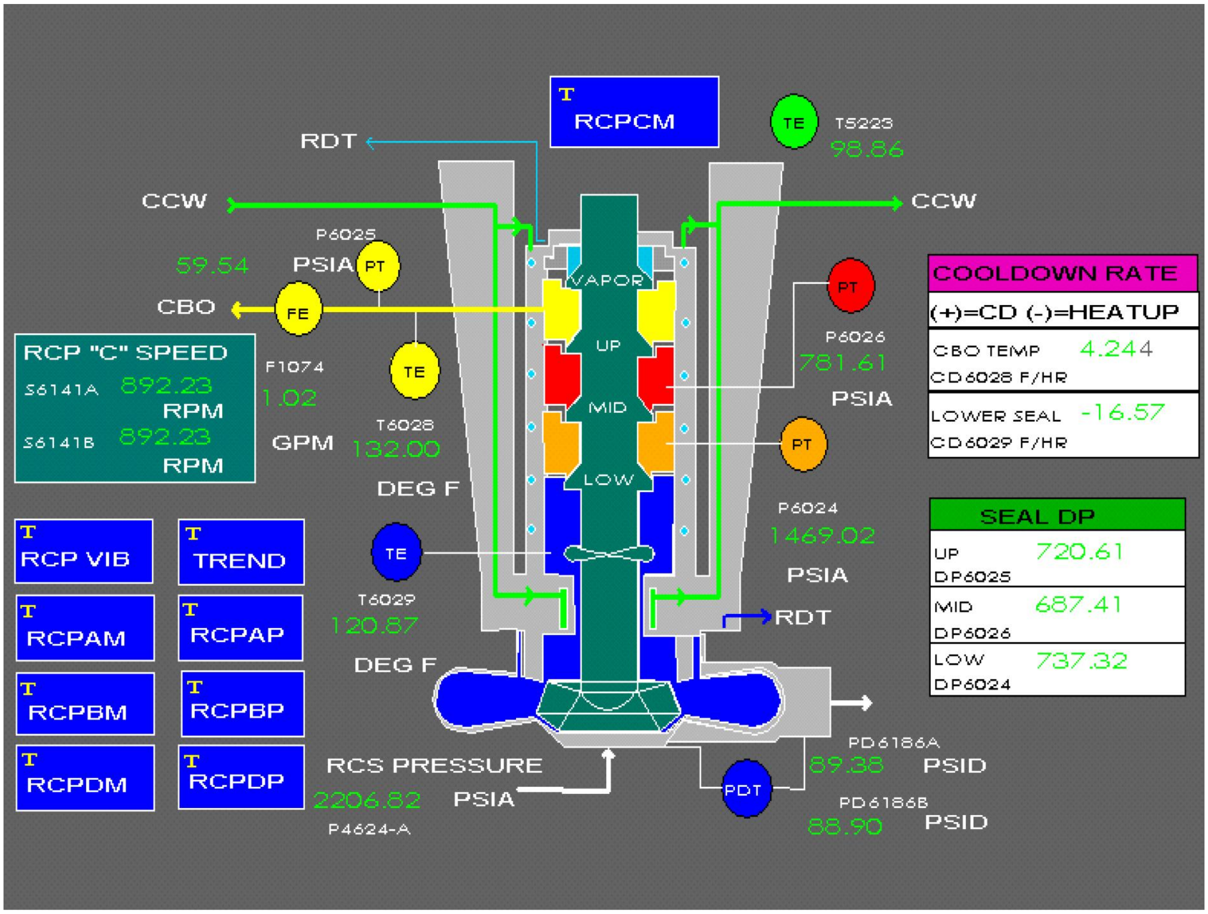
This question matches the K&A as the knowledge of seal flow and seal DP is needed to understand the interrelations between the seals and the RCPs to apply the requirements of the RCP malfunction AOP

References:

STM_2-03-2_ Rev. 18 RCP and RCP Vibration Monitoring Pages 6 and 7 (Verified reference updated 11/10/16); AOP 2203012K Rev. 46 ANNUNCIATOR 2K11 G-3 CORRECTIVE ACTION (Verified reference updated 11/10/16); AOP 2203025 REV 18, RCP Emergencies Steps 5 and 6; AOP-2203025 TG REV 17 Steps 5 and 6 (Verified reference updated 11/15/16); AOP 2203025 REV 18, RCP Emergencies Attachment B RCP Seal Diagram (Verified reference updated 11/15/16)

Historical Comments:

To be used on the 2017 NRC Exam



1.5.1 Seal Instrumentation

Refer to the drawing above or on page 47. RCP seal pressures can be monitored in the control room on 2C14 and on PMS. The lower, middle, and upper seals break down the RCS seal flow pressure in stages at approximately one third of the RCS pressure through each stage. The inlet to the lower seal is assumed to be equal to the current RCS pressure. After the pressure is reduced past the lower seal, the inlet pressure to the middle seal pressure is sent to 2C14 and PMS by 2PT-6024. After the pressure is reduced past the middle seal, the inlet pressure to the upper seal is sent to 2C14 and PMS by 2PT-6026. After the pressure is reduced past the upper seal, the inlet pressure to the VCT is sent to 2C14 and PMS by 2PT-6025. Normal pressure drop across each of the three seals would be approximately 700 psid. Any seal pressure less than 50 psid is considered to be a failed RCP seal. The vapor seal acts as a backup to the first three seals and only allows a small amount of RCS cooling flow past its seal face (~1% of the 1gpm controlled bleed off flow through the seal package).

Controlled Bleed off temperature and flow can also be monitored in the control room on 2C14 and PMS but these parameters also provide alarms to alert the operator of degrading conditions of the seals. Bleedoff temperature will cause an alarm on 2K11 if Controlled Bleedoff temperature

is ≥ 180°F for any RCP. Flow will cause a high or low flow alarm on 2K11 if bleedoff flow is ≥ 1.5 gpm **OR** ≤ 0.6 gpm for any RCP. Controlled Bleedoff pressure can be monitored on 2C09 and on PMS, and will cause a high or high-high pressure alarm on 2K11 if the combined Controlled Bleedoff pressure to the VCT is > 120 psig (for the High alarm) or > 250 psig (for High-High alarm).

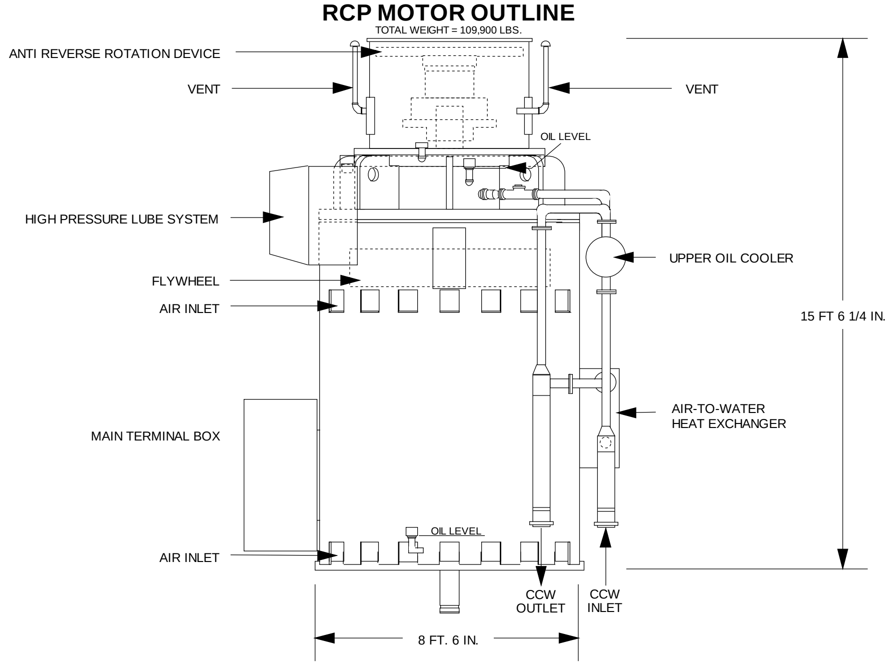
1.6 RCP Motor Description

Each Reactor Coolant Pump is driven by a three-phase, squirrel cage induction motor rated at 6750 horsepower and operates at a speed of 900 rpm. The RCP Motors are powered from 6900 volt buses 2H1 and 2H2. 2P-32A & D are powered from 2H1 and 2P-32B & C powered from 2H2. Refer to the illustration on the following page and also on page 49. Each pump motor is double-end ventilated with ambient air. An air-to-water heat exchanger is provided to cool the discharge air.

The motor also has upper and lower guide bearings, a double acting *Kingsbury* thrust bearing (for up and down thrust) in addition to the bearing in the anti-reverse rotation mechanism. Both the upper and lower bearing assemblies are oil lubricated and water cooled.

A flywheel is attached to the motor shaft between the rotor and the upper bearing. The flywheel and motor-pump rotating assembly will act to improve the flow coastdown characteristics during a loss of power to the pump.

RCP MOTOR OUTLINE



The Reactor Coolant Pump motor is air cooled. Fans mounted on both ends of the rotor take suction on RCP cavity and distributes air to the rotor, stator and flywheel. The air then flows through the motor air cooler, which is cooled by the Component Cooling Water system and discharged to the RCP cavity.

ANNUNCIATOR 2K11

G-3

RCP BLEEDOFF FLOW HI/LO

1.0 CAUSES

1.1 Controlled Bleedoff flow ≥ 1.5 gpm or ≤ 0.6 gpm for any Reactor Coolant Pump:

- 2P-32A (2FS-6008)
- 2P-32B (2FS-6018)
- 2P-32C (2FS-6028)
- 2P-32D (2FS-6038)

2.0 ACTION REQUIRED

NOTE

Annunciator Reflash unit (2K420) is located in back of 2C14 and has the following applicable alarm indications:

- RCP 2P-32A Controlled Bleedoff Flow HI/LO
- RCP 2P-32B Controlled Bleedoff Flow HI/LO
- RCP 2P-32C Controlled Bleedoff Flow HI/LO
- RCP 2P-32D Controlled Bleedoff Flow HI/LO

2.1 Check RCP chart recorders, PMS/PDS trends, and Annunciator Reflash unit (2K420) as necessary to determine affected pump.

2.2 To allow reflash, perform the following:

- Acknowledge alarm by pressing "ACK ALM" on affected RCP chart recorder.
- Acknowledge RIS unit (2K420) in back of 2C-14.

* 2.3 Monitor affected RCP(s) CBO temperature and seal pressures on PMS/PDS.

2.4 Refer to RCP Emergencies (2203.025).

2.5 IF RCP BLEEDOFF FLOW HI/LO alarm becomes nuisance, THEN alarm can be locked in using Bypass switch located under 2K420 in 2C14 IAW Annunciator Removal From Service Or Modification Form 1015.001B, Conduct Of Operations (1015.001).

3.0 TO CLEAR ALARM

3.1 Restore Controlled Bleedoff flow of 0.6 to 1.5 gpm.

4.0 REFERENCES

4.1 E-2457-1

4.2 DCP-91-2012

INSTRUCTIONS	CONTINGENCY ACTIONS
*5. **CHECK** ALL RCP Vapor Seal pressures less than 1500 psia: • RCP A, 2PI-6005 • RCP B, 2PI-6015 • RCP C, 2PI-6025 • RCP D, 2PI-6035	*5. **<u>IF</u>** ANY RCP Vapor Seal pressure greater than 1500 psia, **<u>THEN</u> PERFORM** the following: A. **<u>IF</u>** in Mode 1 OR 2, **<u>THEN</u> TRIP** Reactor. B. **STOP** ANY affected RCP. C. **<u>IF</u>** only ONE RCP affected, **<u>AND</u>** desired to balance reactor coolant loop temperatures, **<u>THEN</u> ENSURE** ONE RCP secured in EACH loop. D. **<u>IF</u>** RCP 2P32A OR 2P32B stopped, **<u>THEN</u> ENSURE** associated PZR Spray valve in MANUAL and closed: • 2CV-4651 • 2CV-4652 E. **<u>IF</u>** Reactor was manually tripped, **<u>THEN</u> GO TO** 2202.001, Standard Post Trip Actions.

INSTRUCTIONS	CONTINGENCY ACTIONS
	NOTE Attachment B, RCP Seal Diagram may be helpful in determining failed RCP seal stages.
*6. **CHECK** ΔP across EACH RCP Seal stage greater than 50 psid.	*6. **PERFORM** the following: A. **IF** only ONE stage failed, **THEN PERFORM** the following: 1) **MONITOR** RCP Controlled Bleedoff flow and temperature. 2) **NOTIFY** Operations Management.

(Step 6 continued on next page)

INSTRUCTIONS | **CONTINGENCY ACTIONS**

6. (continued)

B. **IF** TWO stages failed on ONE pump,
THEN PERFORM the following:

1) **IF** in Mode 1 OR 2,
THEN PERFORM the following:

a) **REFER** to applicable reactivity plan.

b) **COMMENCE** plant shutdown using EITHER of the following:

- 2203.053, Rapid Power Reduction
- 2102.004, Power Operations

c) After reactor tripped, **STOP** ANY affected RCP.

d) **IF** only ONE RCP affected,
AND desired to balance reactor coolant loop temperatures,
THEN ENSURE ONE RCP secured in EACH loop.

e) **IF** RCP 2P32A OR 2P32B stopped,
THEN ENSURE associated PZR Spray valve in MANUAL and closed:

- 2CV-4651
- 2CV-4652

f) **GO TO** 2202.001, Standard Post Trip Actions.

(Step 6 continued on next page)

INSTRUCTIONS	**CONTINGENCY ACTIONS**
6. (continued)	

2) **<u>IF</u>** in Mode 3, 4, OR 5,
<u>THEN</u> PERFORM the following:

a) **STOP** ANY affected RCP.

b) **<u>IF</u>** only ONE RCP affected,
<u>AND</u> desired to balance reactor coolant loop temperatures,
<u>THEN</u> ENSURE ONE RCP secured in EACH loop.

c) **<u>IF</u>** RCP 2P32A OR 2P32B stopped,
<u>THEN</u> ENSURE associated PZR Spray valve in MANUAL and closed:

- 2CV-4651
- 2CV-4652

C. **<u>IF</u>** THREE or more stages failed on ONE pump,
<u>THEN</u> PERFORM the following:

1) **<u>IF</u>** in Mode 1 OR 2,
<u>THEN</u> TRIP Reactor.

2) **STOP** ANY affected RCP.

3) **<u>IF</u>** only ONE RCP affected,
<u>AND</u> desired to balance reactor coolant loop temperatures,
<u>THEN</u> ENSURE ONE RCP secured in EACH loop.

4) **<u>IF</u>** RCP 2P32A OR 2P32B stopped,
<u>THEN</u> ENSURE associated PZR Spray valve in MANUAL and closed:

- 2CV-4651
- 2CV-4652

5) **GO TO** 2202.001,
Standard Post Trip Actions.

AOP STEP:

*4. **<u>IF</u> in-service CCW Surge Tank (2T-37A/B) level lowers to less than 13% following restoration of CCW to RCPs, <u>THEN</u> perform the following:**

BASIS:

Greater than 13% level in the CCW surge tank is needed to ensure adequate NPSH is available for the CCW pumps. If level lowers uncontrollably following restoration of CCW flow to the RCPs, this is indicative of a large CCW leak. Direction is given to trip the reactor and secure all RCPs. The reactor is manually tripped first if the Plant is in Mode 1 or 2 to prevent an automatic reactor trip when the first RCP is secured.
Controlled Bleedoff flow is isolated to prevent possible seal failure. CCW flow is isolated to the RCPs if the leak is inside the Containment Building.

SOURCE DOCUMENTS:

1 - 2202.001, Standard Post Trip Actions.
2 - TDB580 0040, RCP Technical Manual.
3 - ANO-2 SAR Section 5.5.1, Reactor Coolant Pumps.

AOP STEP:

*5. **Check ALL RCP Vapor Seal pressures less than 1500 psia:**

BASIS:

A vapor seal pressure of greater than 1500 psia is an indication of severe degradation of the lower three seals. Contingency actions attempt to minimize the impact on the plant should the vapor seal fail by tripping the reactor, securing affected RCPs, and transitioning the operator to the EOP.

The vapor seal is designed to hold full pressure of 2500 psia in a static condition and during coastdown following failure of the other 3 seals. It is not desired to operate in this abnormal condition due to the raised risk of vapor seal failure.

SOURCE DOCUMENTS:

1 - 2202.001, Standard Post Trip Actions.
2 - TDB580 0040, RCP Technical Manual.
3 - ANO-2 SAR Section 5.5.1, Reactor Coolant Pumps.

AOP STEP:

*6. **Check ΔP across EACH RCP Seal stage greater than 50 psid.**

BASIS:

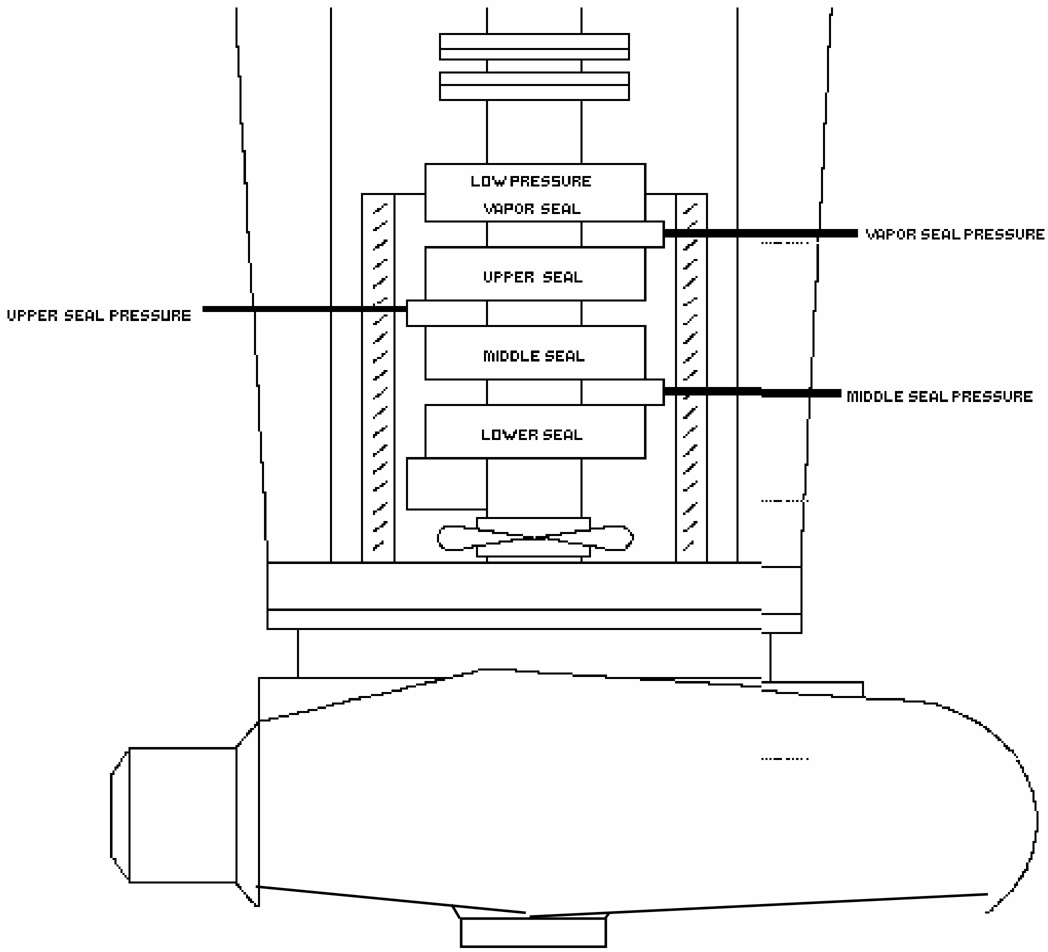
The RCP seal assembly is a series of four seals. At normal operating pressure, RCP seal ΔPs are well in excess of 50 psid. Differential pressure less than 50 psid across any RCP seal indicates seal degradation to the point that the seal is offering almost no resistance to flow.

The contingency actions allow a controlled plant shutdown in Mode 1 or 2 if 2 seals are failed because 2 seals are still available. In Mode 3, 4, or 5, the affected RCP is secured.
If 3 seals are failed, only 1 is left to contain full RCS pressure, so the contingency actions require that the reactor be tripped and the affected pump secured.

SOURCE DOCUMENTS:

1 - 2103.006, Reactor Coolant Pump Operations.
2 - 2202.001, Standard Post Trip Actions.
3 - TDB580 0040, RCP Tech. Manual.

ATTACHMENT B
RCP SEAL DIAGRAM



Question 14

Bank:	2344	**Rev:**	0	**Rev Date:**	10/10/2016	**2017 TEST QID #:**	14	**Author:**	Coble	
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam			
Search	000009K203	**10CFR55:**	41.7	**Safety Function**	3					
Title:	Small Break LOCA	**System Number**	009	**K/A**	EK2.03					
Tier:	1	**Group:**	1	**RO Imp:**	3.0	**SRO Imp:**	3.3	**L. Plan:**	A2LP-RO-ELOCA	**OBJ** 16

Description: Knowledge of the interrelations between the small break LOCA and the following: - S/Gs

Question:

Given the following:

* A small break LOCA has occurred inside Containment.
* The plant has been tripped.
* CIAS, SIAS, CCAS, CSAS, and MSIS have actuated.
* All RCPs have been secured.
* Total EFW flow to the SGs is currently 425 gpm.
* LOCA Recovery OP-2202.003 has been entered.
* Containment Temperature has risen to 250°F.
* Containment Pressure has risen to 30 PSIA.
* RCS Tcold is 550°F and stable.

Which of the following combinations of Steam Generator levels listed would be the LOWEST that would still meet the MINIMUM requirements to satisfy the LOCA EOP RCS Heat Removal Safety Function during this event?

A. SG 'A' 10% and SG 'B' 8%.

B. SG 'A' 15% and SG 'B' 13%.

C. SG 'A' 20% and SG 'B' 18%.

D. SG 'A' 23% and SG 'B' 22%.

Answer:

C. SG 'A' 20% and SG 'B' 18%.

Notes:

C is correct because only one SG with adequate level is needed for minimum required RCS heat removal. Since Containment Temperature is greater than 200°F, Harsh Containment Environment parameters (Brackets 20% to 90%) must be used to gage the RCS Heat Removal Safety Function do 20% in SG A would meet this requirement..

A is incorrect but plausible if the candidate uses the normal non harsh environment parameters and correctly assumes only one SG is needed to meet the minimum required RCS Heat Removal Safety Function.

B is incorrect but plausible if the candidate assumes both SG needs to be above 10% to meet the minimum required RCS Heat Removal Safety Function and does not recognize Containment Harsh environment.

D is incorrect but plausible if the candidate assumes both SGs needs to be above 20% to meet the minimum required RCS Heat Removal Safety Function with Harsh Containment conditions..

This question matches the K&A due to the knowledge of how the SGs levels are monitored during a small break LOCA to ensure the minimum RCS Heat Removal Safety Function is being met.

References:

LOCA EOP 2202003 REV. 15, Safety Function Status Check RCS heat Removal (Verified reference updated 11/10/16); LOCA EOP 2202003 REV. 15 Safety Function Status Check Note for Harsh Environment (Verified reference updated 11/10/16); LOCA EOP-2202003 TG REV. 15 SFSC Step 6 (Verified reference updated 11/10/16).

Historical Comments:

To be used on the 2017 NRC Exam

SAFETY FUNCTION | **ACCEPTANCE CRITERIA**

5. Core Heat Removal

5. A. RCS T_H and average CET temperatures less than superheated.

OR

B. 1) RCS T_H and average CET temperatures less than 10°F superheat.

2) RVLMS LVL 06 or higher elevation indicates WET.

OR

C. 1) BOTH RVLMS channels inoperable.

2) RCS T_H and average CET temperatures less than 10°F superheat.

3) RCS temperatures NOT rising.

4) CET temperatures less than 700°F.

6. RCS Heat Removal

6. A. 1) At least ONE SG level maintained 10% to 90% [20% to 90%] with FW available.

OR

2) SG level being restored by total FW flow greater than 485 gpm.

B. RCS T_C stable or lowering.

NOTE

Parameters in brackets [ ] reflect normal values corrected for harsh CNTMT environment with CNTMT temperature greater than 200°F or CNTMT radiation greater than 10^5 R/hr.

Major Recovery Strategies for LOCA

- Maximize SI flow and attempt to isolate leak.
- Restore RCS pressure and inventory control and maintain RCS Heat Removal after LOCA isolated.
- Perform controlled cooldown to SDC after LOCA isolated.
- Perform cooldown to remove Core heat during unisolated LOCA.
- Initiate Hot and Cold leg injection during unisolated LOCA.
- Maintaining Long Term Core cooling during unisolated LOCA.

TIME: ____ ____ ____ ____

1. Reactivity Control

1. A. 1) Reactor power lowering.

OR

2) Reactor power less than 10^{-1}% AND stable or lowering.

B. 1) Maximum of ONE CEA NOT fully inserted.

OR

2) Emergency Boration in progress or completed.

5. The EOP, as does the EPG, uses T_H and CET temperature less than superheated as the limit to ensure that the recovery strategy is effective in core heat removal. The EOP uses additional criteria which allow temperature to be as much as 10°F superheated as long as RVLMS indicates that the core is covered. This is consistent with the bases discussion presented in the EPG. The EOP contains a final acceptance criteria in the event that temperature is superheated, but by not more than 10°F, and the RVLMS is inoperable. In this case, temperature is required to not be rising and less than 700°F. This ensures that core heat removal is controlled adequately and that core uncovery is not progressing. The plant specific values used in this criteria are consistent with the EPG and are detailed in the EOP setpoint file (4).

6. The EOP, as is the EPG, is designed to check that at least one SG is available for removing heat from the RCS. The EOP is consistent with the EPG in that it requires SG level to be above a certain minimum value or feed flow adequate for heat removal (5) and that RCS temperature be controlled. A harsh CNTMT value for low SG level has been added (5).

7. The EOP Containment Isolation acceptance criteria is consistent with the EPG in that it requires secondary radiation alarms to be clear, and that containment pressure be below the CIAS setpoint or CIAS actuated (6). The EOP also requires that containment high range area radiation monitor be less than 1000 R/hr or that CIAS be actuated to isolate the containment (8, 9). ANO-2 specific radiation monitors were added to EOP condition 7.A (7). An ANO-2 specific setpoint (8), which is less than the alarm setpoint, was added to condition 7.B to provide an "early warning" value for the operators.

8. The ANO-2 CNTMT Cooling Fans are not equivalent to CNTMT Spray pumps as is the case for the EPG design plant. They are used together to ensure environmental qualification criteria are not exceeded. If CNTMT temperature and pressure are not within the acceptance criteria (16), then all available CNTMT Cooling Fans must be operating in the Emergency Mode with at least one CNTMT Spray header delivering minimum flow (16). The term <u>available</u> is used to define the number of CNTMT Cooling Fans needed in order to prevent a transition to the Functional Recovery procedure when only one electrical train is available. In this instance, transition to Functional Recovery to attempt to restore CNTMT temperature and pressure control would hinder mitigation of the LOCA when the Safety Injection System is designed to function with only one train.

Question 15

Bank:	2345	**Rev:**	0	**Rev Date:**	7/6/2016	**2017 TEST QID #:**	15	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	Modified NRC Exam Bank 1865		
Search	000029A205	**10CFR55:**	41.7	**Safety Function**	1				
Title:	Anticipated Transient Without Scram (ATWS)	**System Number**	029	**K/A**	EA2.05				
Tier:	1	**Group:**	1	**RO Imp:**	3.4	**SRO Imp:**	3.4	**L. Plan:**	A2LP-RO-DEFAS
OBJ	4								

Description: Ability to determine and interpret the following as they apply to a ATWS: - System component valve position indications

Question:

Given the following:

* Unit 2 is operating at 100% power.
* "A" Stator Water Cooling pump trips and the standby pump did NOT start.
* Main Turbine Runback is in progress.
* RCS pressure rises to 2450 psia and the Reactor trips.
* "A" and "B" Steam Generator levels are 14% and trending down.
* EFAS 1 and EFAS 2 have failed to actuate.
* NO operator action is taken.

The Reactor trip was caused by ________________and Emergency Feedwater Flow Control Valves (FCVs) will_________________________________.

A. CPC Auxiliary trip; cycle open and closed to maintain SG levels 22.2% to 25%

B. CPC Auxiliary trip; open and feed the SG levels to 80% and then reclose

C. DSS Trip; cycle open and closed to maintain SG levels 22.2% to 25%

D. DSS trip; open and feed the SG levels to 80% and then reclose

Answer:

D. DSS trip; open and feed the SG levels to 80% and then reclose

Notes:

D is correct as an ATWS has occurred due to failure of the normal RPS reactor trip at 2362 psia and the backup aux trip at 2375 psia. Therefore the Diverse Scram Signal DSS setpoint of 2450 psia without a previous RPS trip tripped the Reactor and the DEFAS system will actuate the EFW FCVs if EFAS failed to actuate and SG level drop below 15%. DEFAS is a "One Shot Actuation" and stops feeding the SGs when level reaches 80%.

A is incorrect but plausible as the CPC Aux trip should have tripped the Reactor at 2375 psia but failed to do so in this case. Also the EFAS system if actuated would cycle the FCVs to maintain SG levels 22.2% to 25% but failed to actuate in this question.

B is incorrect as the CPC Aux trip should have tripped the Reactor at 2375 psia but failed to do so in this case but plausible because DEFAS system will actuate the EFW FCVs if EFAS failed to actuate and SG level drop below 15%. DEFAS is a "One Shot Actuation" and stops feeding the SGs when level reaches 80%.

C is plausible because the DSS setpoint is 2450 psia without a previous RPS trip but incorrect because EFAS system if actuated would cycle the FCVs to maintain SG levels 22.2% to 25% but failed to actuate in this question.

This question matches the K&A because it requires knowledge of the ATWS/DSS setpoints and the associated change in

EFW FCV response and indications based on a valid DEFAS which requires a valid DSS input to actuate.

References:

STM_2-63-1_2-1 Diverse Scram SYS Section 2.0 (Verified reference updated 11/10/16);
STM_2-70-1_R6 DEFAS Section 2.4 - 2.7 (Verified reference updated 11/10/16);
STM_2-19-2_39-1 EFW and AFW SYS Section 2.3.3 (Verified reference updated 11/10/16).

Historical Comments:

NRC Exam Bank 1865 was used on the 2012 NRC Exam
To be used on the 2017 NRC Exam but modified for the 2017 NRC Exam

2.0 Detailed System Description

The Diverse Scram System, or DSS, is diverse and independent from the existing Reactor Protection System (RPS) from the process transmitter output to the interruption of power to the Control Element Drive Mechanism Control System (CEDMCS) buses. This reduces the risk of an ATWS. The DSS is designed to be a highly reliable system to avoid spurious Reactor trips. The DSS is an “energized to actuate” system while the RPS is a “de-energize to actuate” system. The DSS is not credited for design basis accident protective action, and will not prevent the Engineered Safety Features Actuation System from performing its mitigating function. Some microprocessor failures however, may result in one Contactor opening. This would be similar to a RPS “half-leg trip”.

The degree of diversity of the DSS is accomplished by employing four Pressurizer pressure loops unique only to DSS. A trip condition is satisfied when any two-out-of-four process channels reaching 2450 psia. The trip function is accomplished by removing CEDM power by opening Contactors which are downstream of the CEDM Motor-Generator Sets, yet upstream of the Reactor Trip Switchgear.

During a plant heatup, DSS in placed in service when the Shutdown bank CEAs are withdrawn to establish cocked rod protection. During a plant shutdown, after the Reactor Trip Breakers are opened the DSS system is placed in bypass if a cooldown is required.

Although the Diverse Scram System serves no safety related functions, ANO has committed to the NRC to maintain DSS operable whenever the plant is in Mode 1. In order to assess system operability, Operations is required to perform daily channel checks on DSS pressurizer pressure instruments, monthly matrix testing, bi-monthly battery checks (performed every other month during matrix testing), and system functional checks prior to criticality.

2.1 DSS Trip Signal Generation

The DSS is a 2/4 logic system designed to trip the Reactor if Pressurizer pressure reaches or exceeds a preset trip setpoint on any 2 of its dedicated pressure channels. Four Pressurizer pressure transmitters, 2PT4600-1 through 4, provide this input to the DSS. The 2/4 logic function is provided by a four separate Foxboro Spec 200 Micro controller systems. These software based micro processors are housed in the DSS Logic cabinet, 2C409; located in the CEDMCS room near the Remote Shutdown Panel. The four micro controllers are used to perform the necessary alarming and the 2-of-4 logic functions for each independent channel.

Each channel of the Diverse Scram System, or DSS, uses a current-to-voltage input converter to convert a 4-10mA input from the pressure transmitters into a 0-10 Vdc signal for the Spec 200 Micro controller input. Four Spec 200 Micro controllers, located on the DSS Logic Cabinet 2C409, are used to perform the necessary alarming and the 2-of-4 logic functions for each independent channel.

- Calibrated Neutron Flux Power
- ΔT Power
- 20% canned value

- AZTILT

The output of the LPD calculation is sent to:

- A trip sequence algorithm and is compared to setpoints. This algorithm will be discussed later.
- An LPD Margin meter on 2C03 (one for each channel) which indicates the difference between calculated LPD and the LPD trip setpoint. When this meter reaches 0, the trip setpoint is met.

2.4 Trip Logic (TRIPSEQ)

The outputs of the DNBR/Quality Margin and LPD calculations plus the outputs of several other calculations are input into the Trip Logic (TRIPSEQ) where they are compared to setpoints. If these setpoints are exceeded, either a Pre-Trip or a Trip signal for DNBR and/or LPD will be issued. A CEA Withdrawal Prohibit (CWP) could also be issued from the trip logic if the conditions are satisfied.

2.4.1 LPD Pre-Trip

The LPD Pre-Trip is PID 067 and its value is inputed by the operator. A Pre-Trip alarm on 2K04 and a CEA Withdrawal Prohibit (CWP) is generated once the variable setpoint is exceeded.

2.4.2 DNBR Pre-Trip

The DNBR Pre-Trip is PID 066 and its value is inputed by the operator. A Pre-Trip alarm on 2K04 and a CEA Withdrawal Prohibit (CWP) is generated once the variable setpoint is exceeded.

2.4.3 CEA Withdrawal Prohibit (CWP)

A CEA Withdrawal Prohibit (CWP) prevents the outward motion of CEAs in all modes except Manual Individual. A CWP is initiated when ANY of the following conditions exist on 2 out of 4 channels:

- DNBR Pre-Trip
- LPD Pre-Trip
- Either CEAC detects an outward or inward CEA deviation of 5 inches.
- CEA Subgroup deviation $\geq$ 4.95 inches
- CEA Regulating Group Out Of Sequence (sequential subgroups get within 7.5 inches of each other)
- High Pressurizer Pressure Pre-Trip (from RPS)

2.4.4 LPD Trip

If conditions continue to degrade following the Pre-Trip setpoint, the following condition will cause an LPD Trip:

- LPD $\geq$ 21 kw/ft

2.4.5 DNBR Trip

If conditions continue to degrade following the Pre-Trip setpoint, the following conditions will cause a DNBR Trip:

- DNBR < 1.25
- Quality Margin $\leq$ 0

2.4.6 Aux Trips

In addition, there are conditions that will cause a DNBR and an LPD Trip without initiating a Pre-Trip. These conditions are referred to as Aux Trips and will cause the above trips if particular

parameters are outside the "design space" for CPCs. These trips do not mean that an actual DNBR or LPD limit have been exceeded, they just indicate the parameters are outside the design limits for the CPC calculations. They are:

2.4.6.1 JTRP

JTRP (PID 189) is the trip flag which is set when the plant is operating outside of the design space for CPCs. The conditions that will set the JTRP flag are:

- **Tcold out of range** (< 495 ^{o}F or ≥ 580 ^{o}F). This sets JTRP to 1.
- **Pressurizer Pressure out of range** (< 1860 psia OR ≥ 2375 psia). This sets JTRP to 1
- **QASI (Hot Pin Axial Shape Index) out of range** (< -0.45 OR ≥ +0.45). This sets JTRP to 1
  - This trip uses a "static" value below 17% power and shifts to a live value at 17% on an up power. On a down power, the value shifts back to a "static" value at 14.5% power.
- **One Pin Peaking factor out of range** (< 1.28 OR ≥ 7.00). This sets JTRP to 1
- **Asymmetric Steam Generator Transient (ASGT).** This sets JTRP to 20.
  - Refer to the figure on page 55. ASGT is a power dependent trip. Its function is to protect against transients that cause power asymmetries between the two SGs (inadvertent closure of one MSIV at power). It uses the maximum of either Calibrated Excore Neutron Flux Power or ΔT Power and it uses both cold leg temperatures (remember each CPC uses two cold leg inputs from diagonally opposite loops). The ASGT uses a constant setpoint of 11^{o}F ΔT between SGs (PID 096) and adds to that constant based upon current power level.
    - From 0 to 20% power, the setpoint is 31^{o}F (PID 096 + 20^{o}F)
    - From 20 to 50% power, the setpoint is ramped from 31^{o}F to 16^{o}F (PID 096 + 20^{o}F to 5^{o}F ramped based on power)
    - From 50 to 80% power, the setpoint is ramped from 16^{o}F to 11^{o}F (PID 096 + 5^{o}F to 0^{o}F ramped based on power)
    - From 80 to 100% power the setpoint is 11^{o}F (PID 096 + 0^{o}F)
- **Variable Overpower Trip (VOPT)**. This sets JTRP to 40
  - Refer to the figure on pages 56 and 57. VOPT provides protection from transients with rapid power increases such as low power CEA withdrawals or large excess load events.
  - VOPT uses the maximum of the following to compare to the trip setpoint:
    - Raw excore power

DEFAS-2 trip signals for EFW initiation to Steam Generator-2 are output from the two 2-out-of-4 logic matrices for Steam Generator-2. Trip signals are transmitted to each half of the EFAS-2 circuitry in 2C39 and also to each half of the EFAS-2 circuitry in 2C40.

2.3 Steam Generator Level Instrumentation

Four narrow range level transmitters per Steam Generator supply signals to the DEFAS logic circuits. These transmitters are designated:

* 2LT-1031-1C through 2LT-1031-4C for the "A" SG, and
* 2LT-1131-1C through 2LT-1131-4C for the "B" SG.

The signals from these transmitters are routed to the DEFAS logic circuit through 2C15(x).

2.4 2-of-4 Logic Trip Paths (2C410)

As stated earlier, DEFAS uses a 2-out-of-4 logic system. This logic circuit is installed in DEFAS cabinet 2C410. It is designed to actuate the Emergency Feedwater System (EFW) if a low Steam Generator level of $\leq$15% is detected on 2-of-4 level instruments from either Steam Generator IF a valid DSS signal is present. Each Trip Channel will reset if Steam Generator level increases to >80% narrow range on at least 3 of the 4 channels.

A DEFAS-1('A' S/G) actuation is developed based on trip signals from Trip Channel 1(2C39) and Trip Channel 2(2C40), if a DSS (Diverse Scram System) permissive is present. This permissive signal is discussed in the next section.

A DEFAS-2('B' S/G) actuation is developed based on trip signals from Trip Channel 3 (2C39) and Trip Channel 4 (2C40), if a DSS permissive is present.

A 2-of-2 trip logic is used in the Aux. Relay Cabinets 2C39/40 to actuate either train of DEFAS.

2.4.1 DSS Enable

As mentioned in the previous section, a DSS permissive signal is required to actuate DEFAS. This DSS permissive is developed in the DSS cabinet 2C409 if DSS channel 1 or 3 AND 2 or 4 are actuated. Refer to the figure on page 20. Both channels present ensure that a valid DSS signal has been sent to the DEFAS circuit.

In order for the DSS permissive signal to be generated, the following conditions must exist:

- DSS Enable Bypass switches on 2C409 must not be in the bypass position, and.
- the power supply handswitches for 2C408 (2HS-9902) and 2C407 (2HS-9901) must be in the "Normal" or "Test" position.

It is important to point out that the manual DSS trip push-buttons on 2C03 are NOT part of this permissive circuit.

The DSS permissive signal logic is essentially a "select 2-of-4" logic. This means that the signals must be supplied from the correct 2-of-4 channels before the permissive is enabled. The DSS contacts in 2C409 are designated R1, R3, R2 and R4. These contacts are arranged in two parallel networks as shown in figures on pages 20

and 21. For the permissive signal to be generated, contacts R1 or R3 AND R2 or R4 must be closed.

A simplified DSS logic diagram is shown on page 19. For more details regarding DSS circuit, refer to STM 2-63.

2.4.2 Manual Trip

In addition to automatic actuations, DEFAS can also be manually actuated from push-buttons on Control Room panel 2C14. Push-button 2HS-9903 is used to actuate DEFAS-1and push-button 2HS-9904 actuates DEFAS-2. These push-buttons are depicted in the figures on pages 21 and 22.

Depressing these push-buttons sends an artificial low Steam Generator level condition (<15% narrow range) to the DEFAS logic in 2C410. A DSS permissive is still required to cause a DEFAS-1 or DEFAS-2 actuation to be sent to the ESFAS cabinets 2C39/40.

2.4.3 DEFAS Block Signals

A DEFAS actuation will be prevented if any of the following conditions exist:

- EFAS is actuated, or
- MSIS is actuated.
- Bypass push-button on the front of cabinets 2C39 and 2C40 (4 total)

2.5 2C39/40

The DEFAS trip signals from the DEFAS cabinet, 2C410, are isolated from the EFAS safety related circuitry in 2C39 and 2C40 by the use of safety related digital (contact) fiber optic receivers mounted in the Auxiliary cabinets.

Refer to the figure on page 24.

When DEFAS trip signals are received by the fiber optic receivers (RCVR), they generate a digital output that energizes the respective DEFAS initiation relay (K527 or K727 for DEFAS-1; K627 or K827 for DEFAS-2). This will result in an EFW actuation only if MSIS relay and EFAS lockout relay contacts are closed (normal, non-accident position). If either the MSIS lockout relay or EFAS lockout relay contacts are open, the DEFAS cannot initiate EFW. The lockout relay contacts being open indicates that the ESFAS has already initiated EFW or isolated a ruptured Steam Generator. The need for ATWS mitigation, if an ATWS has occurred, has been satisfied by the ESFAS; thus DEFAS actuation is blocked.

If the MSIS and the EFAS lockout relay contacts are closed (non-accident state), then the DEFAS can initiate a "one-cycle" EFW signal. The figure on page 24 shows the DEFAS Sub-Panel components. This figure shows the DEFAS relays' interface with the ESFAS/ESF Aux. relays. After the Steam Generator level is restored to 80% level, the DEFAS initiation relay K527, will de-energize and secure EFW flow to the Steam Generator. Since DEFAS actuated the EFAS Lockout relay K521, the associated

contact K521/EFAS in the DEFAS trip path will open and will prevent the DEFAS from initiating a second EFW flow cycle; hence, DEFAS is a “one-shot” actuation. One cycle takes approximately 30 minutes or more for the Steam Generator level to return to normal and an additional 30 minutes for the level to again decrease to the low level setpoint.

The DEFAS initiation signals will actuate the EFW pumps and valves only if there is a requirement for an EFAS signal and this signal has not been generated by the ESFAS.

2.6 Features Local to 2C39 and 2C40

The Trip/Test and Bypass push-buttons for the DEFAS actuation circuits are located in bay 5 and 8 of 2C39 and 2C40. These are shown on the illustrations on pages 29 and 30.

The DEFAS Trip Leg 1, 3 panel is located inside the door of Bay 5. A Plexiglas window is installed so that the buttons can be seen without opening the door.

The DEFAS Trip Leg 2, 4 panel is located outside the door of Bay 8.

2.6.1 Typical Bay

DEFAS relays are installed inside Bays 5, 6, 7 and 8 of 2C39 & 40. The bay sub-panels are the same for each cabinet. The relay designations correspond to the respective bay they are installed (i.e., K526 is bay 5 while K726 is installed in bay 7).

The following DEFAS relays are mounted inside the sub-panels;

DEFAS-1 RELAYS IN BAY 5	DEFAS-2 RELAYS IN BAY 6	DEFAS-1 RELAYS IN BAY 7	DEFAS-2 RELAYS IN BAY 8	FUNCTION
K521	K621	K721	K821	EFAS Aux. Relay
K526	K626	K726	K826	Fault Relay
K527	K627	K727	K827	Interposing Relay (Not the 24v ESF Interposing relay)
K528	K628	K728	K828	Trip Relay
K529	K629	K729	K829	Trouble/Test Relay
K530	K630	K730	K830	Trip Aux. Relay
K531	K631	K731	K831	MSIS Aux. Relay

2.7 Detailed DEFAS Actuation Summary

The above relays are "DEFAS” relays only. The details outlined below explain the function of these relays and how they interface with the ESFAS Auxiliary Relays. The discussion discusses only those relays located in bay 5. The DEFAS trip legs in the other bays are similar, only the relay designations are different. Refer to the figures on pages 23 and 24.

The Trip Relay contact K528 in the EFAS actuation path is normally closed (N/C) with no ESF actuation.

A trip signal from DEFAS energizes relay K527 which closes contact K527. If **both** ESF Aux. Relays contacts, K531-MSIS and K521-EFAS, are closed, which they are when the actuations are reset, the DEFAS Trip relay K528 energizes. Relay K528 will then shut its own seal-in contact K528, that is in parallel with the K521/EFAS. The S52A TRIP lamp will illuminate when Relay K528 energizes.

The Trip Relay contact K528 in the EFAS actuation path will also open when relay K528 energizes. This will de-energize the EFAS Interposing relay K524 located in the ESFAS Aux. Relay Cabinet.

The DEFAS Actuation, with ESF Interposing Relay actuation, in turn opens a contact to de-energize the EFAS Subgroup Relays and the Lockout Relay K521. This will open the K521/EFAS contact that is in parallel with DEFAS Trip seal-in contact K528 mentioned earlier. The K528 Trip contact will keep the DEFAS actuation path energized.

If ESF MSIS lockout relay K519 trips (from an actual PPS/ESFAS actuation), the DEFAS K531/MSIS contact will open. This will open the DEFAS TRIP Relay K528 path. Relay K528 will de-energize disabling the DEFAS actuation. An MSIS will also prevent a DEFAS actuation if the MSIS came in first.

If the fiber optic receiver, RCVR, loses its input signal, a Fault Relay K526 will de-energize, resulting in a TROUBLE/TEST alarm.

The DEFAS Relay K527 will reset if Steam Generator level increases to 80%. If this happens the DEFAS Trip Relay K528 will de-energize. This causes the K528 Trip seal-in contact to open. With the EFAS contact K521 being open due to the EFAS Lockout Relay being energized there is now no actuation path for subsequent DEFAS actuations. Hence, DEFAS is a “one shot” actuation.

2.8 Power Supplies

The DEFAS Logic sub-panel has two Uninterruptable Power Supplies (UPS), consisting of multi-nest power supplies, battery chargers, and a battery backup. A vital 120 VAC power supply is provided to the DEFAS cabinet. This power supply is used internally to power the cabinet’s two Multinest Power supplies (15 VDC), the RIS panel, local indicating lights, cabinet fan, and the two battery chargers. The following breakers are used to energize this 120 V supply:

- 2Y3-breaker 11 and 2Y4-breaker 11

The 15 VDC busses are used to poer all modules in the DEFAS cabinet and the Interlink fiber optics transmitters to 2C-39 and 2C-40.

2.8.1 Battery Backup Module

A Battery Backup module provides a temporary source of power for the Multinest power supplies, to allow uninterrupted operation in the event of an AC line power failure.

- 2K07-G9 (2P7B TO A S/G FLOW HI/LO) from 2FIS-0710-1
  - >400 gpm and < 240 gpm
- 2K07-H9 (2P7B TO B S/G FLOW HI/LO) from 2FIS-0717-1.
  - >325 gpm and < 240 gpm

The minimum recirc flow for 2P-7B is indicated on 2FIS-0710-1.

The flow annunciators are delayed for 10 second after pump start to allow adequate time to develop required flow. Located downstream of the control valves, the flow from the EFW pumps must go through the associated EFW Pump Discharge Stop Check to the S/G's (2EFW-7A/B and 2EFW- 8A/B), then the S/G EFW Inlet Check Valves (2EFW-9A/B) and finally enters the Main Feedwater line to the S/G's.

2.3.2 EFW Pump Discharge Header Back Leakage Identification

During normal conditions the temperature of the Main Feedwater entering the S/G's is approximately 450 °F. If the stop check valves and the check valves should have leakby, the high temperature could cause steam binding of the EFW pumps and render the pumps inoperable. Therefore, it is important to identify back leakage of Main Feedwater into the EFW system. Indications of check valve leakage:

* Increasing EFW discharge header temperature in a secured EFW train that cannot be attributed to changes in the ambient temperature.
* Higher EFW pump discharge pressure with the respective pump secured.
* Rise in EFW pipe surface temperature.

- Steam binding in an EFW pump can be readily identified by oscillating or low pump discharge pressure and flow.

The Waste Control Operator monitors the EFW pump discharge temperature at least once per shift.

2.3.3 EFW Pump Discharge to the S/G's Motor Operated Valves

The flow from the EFW pumps to the S/G's is through 4 headers each with 2 valves in series. As stated previously either EFW pump can supply either S/G. The ‘A’ S/G can be supplied by 2P-7A through 2CV-1026-2 and 2CV-1037-1 or by 2P-7B through 2CV-1025-1 and 2CV-1038-2. The ‘B’ S/G can be supplied by 2P-7B through 2CV-1075-1 and 2CV-1036-2 or from 2P-7A through 2CV-1076-2 and 2CV-1039-2.

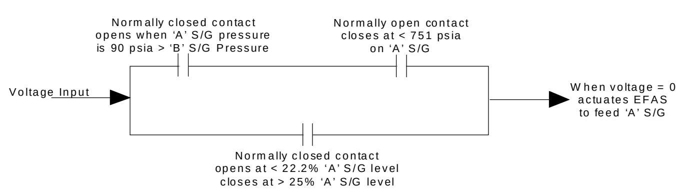
All eight valves provide Control Room annunciators to warn the operator of the power supply breaker being open. Annunciator 2K04-K2, “ESF PROCESS SYSTEM INOP”, is activated when the breaker for 2CV-1026-2, 2CV-1036-2, 2CV-1038-2, or 2CV-1076-2 is opened. Annunciator 2K07-K2, “ESF PROCESS SYSTEM INOP”, is activated when the breaker for 2CV-1025-1, 2CV-1037-1, 2CV-1039-1, or 2CV-1075-1 is opened. The power supplies and location of the 8 valves are:

Description	Valve	Power Supply	Control Power
2P7A to ‘A’ S/G	2CV-1026-2	Green DC (2D26-A4)	2D24
2P7A to ‘A’ S/G	2CV-1037-1	Red DC (2D27-B1)	2D23
2P7A to ‘B’ S/G	2CV-1076-2	Green DC (2D26-C1)	2D24
2P7A to ‘B’ S/G	2CV-1039-1	Red DC (2D27-B2)	2D23
2P7B to ‘A’ S/G	2CV-1025-1	Red AC (2B51-N2)	2D23
2P7B to ‘A’ S/G	2CV-1038-2	Green AC (2B63-H3)	2D24
2P7B to ‘B’ S/G	2CV-1075-1	Red AC (2B53-J2)	2D23
2P7B to ‘B’ S/G	2CV-1036-2	Green AC (2B63-H1)	2D24

The valves receive an automatic open signal on low S/G level of 22.2% due to an Emergency Feedwater Actuation Signal (EFAS) from the Plant Protection System (PPS). When S/G level raises to ~ 25%, (~ EFAS reset valve) the automatic open signal is removed and the valve will close. This is different from most Emergency Safety Feature Actuation System automatic signals in that the valves automatically reposition when the trip reset setpoint has been reached. This action prevents overcooling of the Reactor Cooling System due to excessive feeding of the S/G's. It should be noted that the automatic reset of EFAS applies to the EFW Pump Discharge Valves to the S/G's and not any other EFAS actuated component. For more information on the PPS and EFAS, refer to STM 2-63.

2.3.3.1 EFW Pump Discharge to the S/G's Motor Operated Valves Simplified EFAS Actuation Circuit

A Main Steam Isolation Signal (MSIS) also effects the position of the EFW Pump Discharge Valves to the S/G's. When either S/G pressure lowers to 751 psia an MSIS is generated in the PPS and the S/G's steam and feedwater lines are isolated. This helps minimize the overcooling of the RCS. For the EFW to S/G's valves the MSIS removes the EFAS signal and allows the valves to close. Once the PPS has determined the affected S/G, the EFAS to the unaffected S/G is returned to normal and the valves will cycle on S/G level. The unaffected S/G is the S/G that has a 90 psia higher pressure. If the pressure in both S/Gs rises to > 751 psia then the valves will cycle on S/G level only.



Simplifed EFAS Actuation to "A" S/G Circuit

The control valves can be manually controlled at the valves. In the area of the 4 control valves in the Upper South Piping penetration Area is a level indication for the “A” Steam Generator. This provides a means for the operator to control S/G level locally at the valves. The

PARENT QUESTION 1865 WAS USED TO MODIFY 2017 NRC EXAM QID#15 (Used on the 2012 NRC Exam)

Data for 2012 NRC RO/SRO Exam

Bank:	1865	**Rev:**	0	**Rev Date:**	4/24/2012 4:09:43	**QID #:**	44	**Author:**	Foster
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	Modified NRC bank 1755		
Search	061000K602	**10CFR55:**	41.7			**Safety Function**	4		
System Title:	Auxiliary / Emergency Feedwater (AFW) Syste					**System Number**	061	**K/A**	K6.02
Tier:	2	**Group:**	1	**RO Imp:**	2.6	**SRO Imp:**	2.7	**L. Plan:**	A2LP-RO-DEFAS
OBJ	14								

Description: Knowledge of the effect of a loss or malfunction of the following will have on the AFW System components: - Pumps

Question:

Consider the following:

- Unit 2 is operating at 100% power
- "A" Stator Water Cooling pump trips and the standby pump did not start
- Main Turbine Runback is in progress
- RCS pressure is 2450 psia and the Reactor automatically trips
- "A" and "B" Steam Generator levels are 14% and trending down
- EFAS 1 and EFAS 2 have failed to actuate
- NO operator action is taken

The Automatic Reactor Scram was caused by ________________and Emergency Feedwater Pumps _______ automatically start on a DEFAS signal.

A. CPC Auxiliary trip; will

B. CPC Auxiliary trip; will not

C. DSS trip; will

D. DSS trip; will not

QID use History

	RO	SRO
2005	☐	☐
2006	☐	☐
2008	☐	☐
2009	☐	☐
2011	☐	☐
2012	☑	☑

Audit Exam History

2006	☐
2009	☐
2011	☐

Answer:

C. Correct, DSS trip will occur at a RCS pressure of 2450 psia these transmitter and trip path are separate from the RPS system and protect from a ATWS; EFW will automatically Start on a DEFAS signal (Automatic DSS without EFAS actuation and SG level <15%)

Notes:

A. Incorrect, CPC Auxiliary trip setpoint is 2375 psia
B. Incorrect, CPC Auxiliary trip setpoint is 2375 psia and EFW will automatically start and feed SGs due to a DEFAS signal
D. Incorrect, EFW will automatically start and feed SGs due to a DEFAS signal

References:

STM 2-70-1, Diverse Emergency Feedwater Actuation System, Rev. 6, section 2.2 (DEFAS Actuation Logic) [page 3]
STM 2-63-1, Diverse Scram System, Rev. 2, section 2.0 (describes the trip on high RCS pressure and setpoint of 2450 psia) [page 3]
STM 2-65-1, Core Protection Calculator System, Rev. 16, section 2.4.6 (Aux Trips) [page 15]
A2LP-RO-DEFAS, Rev. 3, Obj. 14, Describe the enabling function of DSS with respect to DEFAS

Historical Comments:

Question 16

Bank:	2346	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	16	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	000065A205	**10CFR55:**	41.10	**Safety Function**	8				
Title:	Loss of Instrument Air	**System Number**	065	**K/A**	AA2.05				
Tier:	1	**Group:**	1	**RO Imp:**	3.4	**SRO Imp:**	4.1	**L. Plan:**	A2LP-RO-ALIA
OBJ	3								

Description: Ability to determine and interpret the following as they apply to the Loss of Instrument Air: - When to commence plant shutdown if instrument air pressure is decreasing

Question:

Given the following:

* The plant is at full power.
* Annunciator 2K12-A8, INSTR AIR PRESS HI/LO, comes in.
* Instrument Air Header pressure has lowered to 55 psig and stabilizing.
* The Loss of Instrument Air AOP OP-2203.021 has been entered.
* CNTMT Chill Water Isolation Valves 2CV-3851-1 and 2CV-3852-1 have failed CLOSED.
* I&C has commenced monitoring CEA CEDM Coil Temperatures.
* Restoration of Instrument Air and CNTNT Chill water is not imminent.

Which of the listed CEA coil temperature would be the LOWEST temperature that would require commencement of a plant shutdown in accordance with the Loss of Instrument Air procedure
OP-2203.021?

A. 411°F

B. 435°F

C. 474°F

D. 504°F

Answer:

C. 474°F

Notes:

C is correct: If coil temperatures are projected to exceed 450°F and restoration of CEDM cooling is not imminent then a plant shutdown should be commenced IAW the Loss of IA AOP.

A is incorrect but plausible as this is a very high CEA coil temperature.

B is incorrect but plausible as this is a very high CEA coil temperature.

D is incorrect but plausible as this number would require a plant trip but is not the lowest temperature listed required to commence a plant shutdown.

This question matches the K&A as it requires an integrated knowledge of how Loss of IA affects Main Chill Water and CEA operations and the required plant shutdown conditions on a lowering IA pressure and CEA coil temperatures.

References:

Loss of IA AOP 2203.021REV 18 Step 18 Contingency Action Step B6 (Verified reference updated 11/10/16); Loss of IA AOP-2203.021 TG REV 18 Step 18 (Verified reference updated 11/10/16).

Historical Comments:

To be used on the 2017 NRC Exam.

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed distractors numbers to allow D to be over 500.

INSTRUCTIONS

*18. **CHECK** ALL CNTMT Chilled Water Isolation valves open:

- 2CV-3850-2
- 2CV-3851-1
- 2CV-3852-1

(Step 18 continued on next page)

CONTINGENCY ACTIONS

*18. **<u>IF</u>** Chilled Water **<u>NOT</u>** aligned to CNTMT, **<u>THEN</u> PERFORM** the following:

A. **REFER TO** TS 3.6.1.4, Internal Pressure and Air Temperature.

B. **<u>IF</u>** Plant in mode 1 or 2, **<u>THEN</u> PERFORM** the following to monitor CEDM Coil temperatures:

1) **MINIMIZE** CEA movement.

2) **<u>IF</u>** ANY CEA dropped or misaligned, **<u>THEN</u> PERFORM** 2203.003, CEA Malfunction in conjunction with this procedure.

3) **NOTIFY** I&C to monitor CEDM coil temperatures.

4) **<u>IF</u>** I&C **<u>NOT</u>** available, **<u>THEN</u>** locally **MONITOR** CEDM Coil temperature using 2105.009, CEDM Control System Operation, Exhibit #2, CEA #01 Upper Gripper Coil Temperature Measurement.

5) **REQUEST** assistance from System Engineer.

INSTRUCTIONS	CONTINGENCY ACTIONS
18. (continued)	

6) **IF** coil temperatures projected to be greater than 450°F for an extended period
AND restoration of CEDM Cooling **NOT** imminent,
THEN PERFORM the following:

a) **REFER TO** applicable reactivity plan.

b) **COMMENCE** Plant Shutdown using EITHER of the following:

- 2102.004, Power Operations
- 2203.053, Rapid Power Reduction

7) **IF** CEDM System Engineer **NOT** available,
AND coil temperatures exceed 500°F,
THEN PERFORM the following:

a) Manually **TRIP** Reactor.

b) **PERFORM** 2202.001, Standard Post Trip Actions, in conjunction with this procedure.

C. **IF** Plant in mode 3 or 4
AND CEDMs energized,
THEN ENSURE TCBs open.

1) **PERFORM** 2202.010 Exhibit 8, Diagnostic Actions in conjunction with this procedure.

AOP STEP:

18. **CHECK** ALL CNTMT Chilled Water Isolation valves open:

BASIS:

CNTMT chilled water valves 2CV-3851-1 and 2CV-3852-1 fail closed on loss of IA. This will cause a loss of CNTMT and CEDM cooling. Contingency actions direct the operator to monitor CNTMT temperature and pressure for Tech Spec limits. This also results in a loss of cooling from the CEDM cooling fans. CEA motion is minimized and if CEA(s) drop, the operator is directed to use 2203.003, CEA Malfunction AOP (7) in conjunction with this procedure.

If the plant is in mode 1 or 2, the Reactor should be tripped if coil temperatures exceed 500°F to prevent degradation of the coil potting compound (4). This is monitored by I&C and if I&C unavailable this is monitored by Operations (8). The TCBs are opened if the plant is in Mode 3, 4, or 5 and the operator is directed to use 2202.010, Standard Attachments, Exhibit 8, Diagnostic Actions to determine further action for the plant(5, 6).

SOURCE DOCUMENTS:

1 ANO-2 Technical Specifications
2 E-2370 Sh. 2, Containment Chilled Water and Purge Air Isolation SV’s.
3 Tech Manual C480.0140, Combustion Engineering CEDM Technical Manual.
4 CR-ANO-2-1999-0433
5 2202.010, Standard Attachments, Exhibit 8, Diagnostic Actions
6 1015.021, EOP/AOP User’s Guide
7 2203.003, CEA Malfunction AOP
8 2105.009 (CEDM Control System Operation) Exhibit #2, CEA #01 Upper Gripper coil Temperature Measurement.

Question 17

Bank:	2347	**Rev:**	2	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	17	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	000040A108	**10CFR55:**	41.5	**Safety Function**	4				
Title:	Steam Line Rupture	**System Number**	040	**K/A**	AA1.08				
Tier:	1	**Group:**	1	**RO Imp:**	3.6	**SRO Imp:**	3.7	**L. Plan:**	A2LP-RO-EESD
OBJ	1								

Description: Ability to operate and/or monitor the following as they apply to the Steam Line Rupture: - Normal operating steam parameters, as a function of power

Question:

Given the following:

* Plant is operating at 100% power at 450 EFPD.
* The Main Turbine Load Limit Pot Light is ON.
* Core Operating Limit Supervisory System (COLSS) is Operable.

Based on the above conditions, which of the following set of parameters would indicate a steam line rupture of the Main Steam line going to the Main Feedwater Pumps?

A. COLSS Indicated Reactor power lowering; Turbine First Stage Pressure remains the same.

B. COLSS Indicated Reactor power rising; Turbine First Stage Pressure lowering.

C. COLSS Indicated Reactor power lowering; Turbine First Stage Pressure lowering.

D. COLSS Indicated Reactor power rising; Turbine First Stage Pressure remains the same.

Answer:

B. COLSS Indicated Reactor power rising; Turbine First Stage Pressure lowering.

Notes:

B is correct because at 450 EFPD the Moderator Temperature Coefficient (MTC) will be negative causing a rise in power due to the rising steam flow causing a lowering of RCS temperature. The steam line going to the Main Feedwater pump is upstream of the Main Turbine so pressure to the Main Turbine First Stage Pressure (TFSP) will lower with the turbine on the Load Limit Pot.

A is incorrect because reactor power will be rising and TFSP will be lowering but plausible since TFSP is an input to the COLSS calculated power logic and is linear based on pressure. Also plausible because if the Turbine is on Load Set instead of Load Limit, then as Steam Header pressure lowers the steam admission valves to the Main turbine will come open to maintain a constant TFSP

C is incorrect because reactor power will be rising. But plausible since TFSP will be lowering and is an input to the COLSS calculated power logic and is linear based on pressure and TFSP is lowering so Turbine power input to COLSS will be lowering.

D is incorrect because TFSP will be lowering but plausible as reactor power will be rising and TFSP is an input to the COLSS calculated power logic and is linear based on pressure. Also plausible because if the Turbine is on Load Set instead of Load Limit, then as Steam Header pressure lowers the steam admission valves to the Main turbine will come open to maintain a constant TFSP

This question matches the K&A because it requires the candidate to monitor TFSP along with primary power during a steam line rupture and to analyze the coupling of the primary to the secondary based on the negative MTC.

References:

EOP 2202.005 Excess Steam Demand REV. 15 Entry Conditions (Verified reference updated 11/10/16); AOP 2203 050 Overcooling Event in Mode 1 or 2 Rev. 3 Entry Conditions (Verified reference updated 11/10/16). STM 2-24-1, Main Turbine Control System, Rev. 22 Section 2.2.3 Load Limit Circuit.

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed distractors A and D to TFSP remaining the same and updated analysis to explain why.

REV. 2 based on NRC Chief Examiner Feedback BNC. Changed up the stem as requested adding a reference to the turbine being on the Load Limit Pot and COLSS operable. Added the word "Indicated" to the beginning of each distractor and answer.

OVERCOOLING EVENT IN MODE 1 OR 2

PURPOSE

This procedure provides actions to be taken for an RCS overcooling/excessive steam demand event during operation in Mode 1 or 2.

ENTRY CONDITIONS

1. Abnormal SG pressure or level
2. Abnormal Main Steam or Main Feedwater flow
3. Unexplained drop in RCS temperature
4. Lowering PZR pressure and level NOT due to RCS leak OR CVCS malfunction
5. COLSS Power Margin Exceeded Alarm (2K10-A2)
6. Reactor TAVE-TREF LO Alarm (2K10-F3)
7. Indication of steam flow to atmosphere (loud noise, high temperature, visual observation)
8. Rise in CNTMT Building parameters (temperature, pressure, dew point, or sump level) with NO corresponding rise in CNTMT Building radiation levels
9. Unexpected change in electrical or thermal power

EXIT CONDITIONS

1. Reactor tripped and SPTAs entered.

OR

2. All appropriate actions of this procedure have been completed.

OVERCOOLING EVENT IN MODE 1 OR 2

PURPOSE

This procedure provides actions to be taken for an RCS overcooling/excessive steam demand event during operation in Mode 1 or 2.

ENTRY CONDITIONS

1. Abnormal SG pressure or level
2. Abnormal Main Steam or Main Feedwater flow
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4. Lowering PZR pressure and level NOT due to RCS leak OR CVCS malfunction
5. COLSS Power Margin Exceeded Alarm (2K10-A2)
6. Reactor TAVE-TREF LO Alarm (2K10-F3)
7. Indication of steam flow to atmosphere (loud noise, high temperature, visual observation)
8. Rise in CNTMT Building parameters (temperature, pressure, dew point, or sump level) with NO corresponding rise in CNTMT Building radiation levels
9. Unexpected change in electrical or thermal power

EXIT CONDITIONS

1. Reactor tripped and SPTAs entered.

   OR

2. All appropriate actions of this procedure have been completed.

Let's assume that Load Set Reference signal is still at -4.0 Vdc and the First Stage Pressure signal is still at +4.0 Vdc. Recall from the discussion above that this would call for a valve position signal of -4.0 Vdc.

Remember that Control valve regulation will apply a gain of 20 to the Speed Error signal. This is applied before the Speed Error signal reaches the Stage Pressure Feedback circuit. If Turbine speed decreased 1 rpm to 1799 rpm this would result in a Speed Error change of .00056% or -.0028 Vdc (.00056% of -5 Vdc).

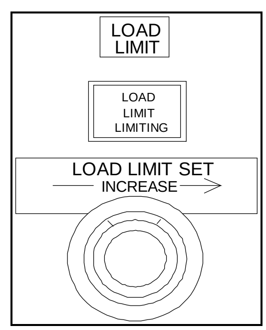
This Speed Error voltage is first applied to the Control Valve regulation gain of 20 resulting in an input signal to SPF of -.056Vdc. When this signal is applied a gain of 16, the resulting input to the SPF circuit would be -.896 Vdc. This is added to the -4.0 signal already present resulting in a valve position signal of -4.896 Vdc. This would cause the Control Valves to move to nearly the wide open position in an attempt to increase Turbine speed. (valves are 100% open at 5 Vdc). So, in this example, a 1 rpm change in speed resulted in a 18% change in the valve position signal.

It can be plainly seen that a small change the Speed Error signal can have a significant effect on the position signal being sent to the Control Valves via the SPF circuit.

The gains that would be applied if SPF were full IN are still applied to the circuit if SPF is OFF when either the #1, #2, or the #3 Control Valve TEST pushbutton is depressed. The potentiometer will not move. It is bypassed internally by contacts which close to apply the gains. The response of the system during a CV test would be identical to the response if the SPF were IN.

2.2.3 Load Limit circuit

The Load Limit Circuit allows the Operator to reduce Turbine load from 0-100% by adjusting a Potentiometer on the Control Panel. When the Potentiometer setting is below the Load Set Reference signal, then the Load Limit circuit is placing a maximum load or "Ceiling" on what Turbine load can be. Whenever the Load Limit circuit is actively holding Load at the desired value, a "Load Limit Limiting" light will illuminate on the Turbine Control panel on 2C01.



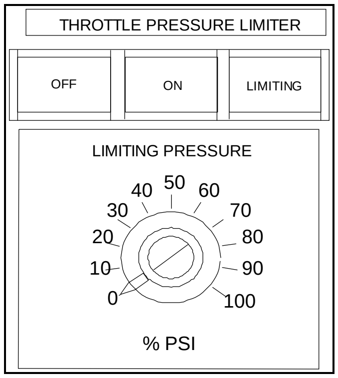
Operators need to be aware that the Load Set motor will not automatically lower in order to maintain just above the Load Limit setpoint. When the Load Limit potentiometer is in control (i.e., Load Limit pot setting is below Load Set Reference value), any load increases using the Load Limit potentiometer will be at the rate the pot. is turned until load reaches the higher Load Set motor position.

The Load Limit potentiometer is used when at steady state power conditions in order to maintain a constant Control Valve position.

The Load Limit Potentiometer is also utilized by the Operators to regulate the loading rate during all controlled power changes. The Load Set Reference is not used because it swings Tave and ASI too much.

In the increase direction the Operator uses Load Limit in order to maintain the Load increase within allowable Reactor Fuel pre-conditioning guideline limits. In the decrease direction the Load Limit potentiometer provides the Operator with a much smoother method of load reduction than is available through the Load Set "DECREASE" push-button.

2.2.4 Throttle Pressure Limiter



The Throttle Pressure Limiter circuit is used by the EHC system to protect the Turbine against an excessive decrease of Throttle Pressure as sensed by 2PT-0251. Normally this circuit is left in OFF since it would cause a major plant perturbation if 2PT-0251 fails. The pressure or "Lowest desired Throttle Pressure" setpoint is adjustable from 0-100% of the full rated throttle pressure (885.3 psig) by means of a knob on the Control panel. If it is desired to place it in service, this knob is procedurally set at 90% at ANO-2. This places the TPL setpoint at a pressure equal to ≈800 psig.

The Valve Regulation in the Throttle Pressure Limit circuit is 10%. This means that the CVs will start stroking closed when Throttle pressure decreases to the limiting pressure (~800 psig for a 90% setting) and will be at the 2% open position should pressure decrease to 80% of rated throttle pressure. The Control Valves will reopen conversely on increasing pressure if TPL is still calling for the lowest CV opening.

2.2.5 Chest/Shell Warming Circuit

Valve Chest warming[3] is necessary prior to startup to match valve metal temperature with Initial steam temperature (Throttle steam) to minimize thermal stresses. Chest Warming heats only the Valve Chest area via a small internal bypass valve around the #2 Main Stop Valve (2CV-0206). All other Main Turbine valves are closed.

Turbine Shell warming is provided for matching HP Turbine shell metal temperature and Main steam temperature. During Turbine

FOOTNOTES:

[3] The Chest area is the area located between the Main Stop Valves and the Control Valves. When referring to pressures, Chest Pressure is measured at the same point as Throttle Pressure. The Shell area means the High Pressure Turbine shell.

COLSS Plant Power Selection

INPUT	QUALITY	PLANT POWER SELECT LARGER OF:
	BDELT < 15%	
BDELT	GOOD	BDELT
BDELT BTFSP	BAD GOOD	BTFSP
BDELT BTFSP	BAD BAD	-FAILED-
	BDELT ≥ 15% & BSCAL < 80%	
CBDELT CBTFSP BSCAL	GOOD GOOD GOOD	CBDELT or CBTFSP
CBDELT CBTFSP BSCAL	GOOD BAD GOOD	CBDELT
CBDELT CBTFSP BSCAL	BAD GOOD GOOD	CBTFSP or BSCAL
CBDELT CBTFSP BSCAL	BAD BAD GOOD	BSCAL
	BSCAL ≥ 80%	
CBTFSP BSCAL	GOOD GOOD	CBTFSP or BSCAL
CBTFSP BSCAL	BAD GOOD	BSCAL

NOTES:

1. From 15 to 100% - IF BSCAL is "BAD", then display is highest of BDELT or BTFSP.

Question 18

Bank:	2348	**Rev:**	2	**Rev Date:**	1/12/2017	**2017 TEST QID #:**	18	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	0000222236	**10CFR55:**	41.10	**Safety Function**	2				
Title:	Loss of Reactor Coolant Makeup	**System Number**	022	**K/A**	2.2.36				
Tier:	1	**Group:**	1	**RO Imp:**	3.1	**SRO Imp:**	4.2	**L. Plan:**	A2LP-RO-CVCS
OBJ	12								

Description: Equipment Control - Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Question:

Consider the following:

* Unit 2 is in Mode 6.
* GREEN TRAIN is protected.
* "B" Shutdown Cooling train is in service.
* Core reload is in progress.
* The Refueling Canal is at 401.2 feet and 2675 ppm boric acid concentration.
* Makeup to the Refuel Canal from the RWT in progress to raise level to 401.5 feet.
* While filling the Refuel Canal, the RWT level was noted to be at 7.2% level.
* Boric Acid MU Tank (BAMT) 2T-6A is tagged and drained for heater replacement.
* Batch additions of boric acid have been made to BAMT 2T-6B.
* BAMT 2T-6B is on recirc using BAM Pump 2P-39B for a sample.
* BAM Pump 2P-39B breaker trips on overcurrent due to a pump seal/bearing failure.
* BAMT 2T-6B level is currently at 34% and slowly trending down.

Which of the following actions is required to be performed FIRST in accordance with TRM 3.1.7 Borated Water Sources - Shutdown?

A. Suspend any operation that could add coolant to the RCS with less than the TS required boron concentration

B. Start BAM Pump 2P-39A and continue to recirc 2T-6B to obtain chemistry operability sample.

C. Add additional batches of boric acid solution to BAM Tank 2T-6B to restore 2T-6B to operable level status.

D. Perform blended MU to the Refueling Water Tank to restore RWT level to operable level status.

Answer:

A. Suspend any operation that could add coolant to the RCS with less than the TS required boron concentration

Notes:

A. is correct, TRM 3.1.7 directs to Suspend any operation that could add coolant to the RCS with less than the TS required boron concentration without a functional boration source which is either one BAMT greater than 36% or the RWT greater than 7.5%.

B is incorrect as this is not the first action needed but plausible as a sample is still needed but will not make the BAMT operable due to the current level.

C is incorrect as this is not the first action needed but plausible as more inventory is needed in BAMT 2T-6B to have an operable level in a BAMT.

D is incorrect as this is not the first action needed but plausible as more inventory is needed in the RWT to have an operable level in the RWT

This question matches the K& A as the failure of the power supply to the BAM Pump 2P-39B along with maintenance activities being performed on the redundant BAMT places the unit in a TRM LCO requirement for an immediate action to suspend loading fuel.

References:

TRM 3.1.7 for Borated Water Sources for Mode 5-6, Rev. 49 (Verified reference updated 11/10/16);

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed the correct answer to the 2nd part of TRM LCO 3.1.7 and updated analysis.

Rev. 2 based on post submittal validation comments: Changed the Refueling Canal level from 401.0 feet to 401.2 feet in the fifth bullet.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES – SHUTDOWN

TECHNICAL REQUIREMENT FOR OPERATION

3.1.7 As a minimum, one of the following borated water sources shall be FUNCTIONAL:

a. One boric acid makeup tank with:

1. A minimum indicated tank level of 36%,

2. A boric acid concentration between 3.0 WT% and 3.5 WT%, and

3. A minimum solution temperature of 55 °F.

b. The refueling water tank with:

1. A minimum indicated tank level of 7.5%,

2. A minimum boron concentration of 2500 ppm, and

3. A minimum solution temperature of 40 °F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources FUNCTIONAL, suspend loading irradiated fuel in the core or operations that could introduce into the RCS, coolant with boron concentration less than required by Technical Specification 3.1.1.2 or 3.9.1 as applicable until at least one borated water source is restored to FUNCTIONAL status.

TEST REQUIREMENTS

4.1.7 The above required borated water source shall be demonstrated FUNCTIONAL:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume of the tank, and

3. Verifying the boric acid makeup tank solution temperature is greater than 55 °F.

b. At least once per 24 hours by verifying the RWT temperature when it is the source of borated water and the outside air temperature is < 40 °F.

Question 19

Bank:	2349	**Rev:**	2	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	19	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NRC Exam Bank 1554		
Search	00CA11K202	**10CFR55:**	41.7	**Safety Function**	4				
Title:	RCS Overcooling	**System Number**	A11	**K/A**	EK2.2				
Tier:	1	**Group:**	2	**RO Imp:**	3.2	**SRO Imp:**	3.4	**L. Plan:**	A2LP-RO-FWCS
OBJ	11								

Description: Knowledge of the interrelations between the (RCS Overcooling) and the following: - Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Question:

Which of the following statements best describes the purpose and function of Main Feedwater Reactor Trip Override (RTO) signal after a Reactor Trip?

A. To rapidly add feedwater to the SGs to ensure an adequate RCS heat sink.

B. To slowly add feedwater to the SGs to prevent overcooling the RCS.

C. To rapidly add feedwater to the SGs to prevent an EFAS actuation.

D. To slowly add feedwater to the SG to limit thermal stresses to the feed rings.

Answer:

B. To slowly add feedwater to the SGs to prevent overcooling the RCS.

Notes:

B is Correct. A design feature of the FWCS is to refill the SGs following a Rx trip due to SG Shrink without causing a primary coolant overcooling event and is accomplished with the RTO signal when the system detects a Rx trip. RTO causes the running MFP to go to minimum speed demand and controls the feed flow based on RCS temperature. The MFRV is closed and the Bypass valve modulates between 19% and 50.7% valve position.

A is incorrect as the valve demand goes way down to slowly refill the SGs but plausible because the shrinkage of the SG levels post trip due to a rapid rise in Steam Pressure make it appear that the Primary Heat sink is lost.

C is incorrect as the feedwater is added fairly slowly to prevent overcooling but plausible as the rapid shrinkage of the SG levels post trip do approach the EFAS actuation setpoints and may even actuate.

D is incorrect as the feedwater temperature post trip is still fairly high due to residual heat in the FW HXs but plausible if the feed water to the SGs were completely lost as slowly refilling of steam generators is directed by EOPs during hot conditions if reinitiating FW after a complete loss.

This question matches the K&A because the candidate must have a Knowledge of the Interrelations of the Main Feedwater heat removal system and the effect it will have on the primary coolant post trip to prevent an RCS overcooling event.

References:

STM_2-69_14-1 FWCS Section 3.3 RTO Response to a Reactor-Turbine Trip (Verified reference updated 11/10/16).

Historical Comments:

NRC Exam Bank 1554 was used on the 2008 NRC Exam
To be used on the 2017 NRC Exam but altered the order of the answer from the last time the QID was given.

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed "cold water" to "feedwater " in distractor A.

REV. 2 based on NRC Chief Examiner Feedback BNC. Changed the word "refill" to "add" in distractors A and B as requested.

The sudden decrease in Turbine load causes an increase in SG pressure which causes a corresponding rapid decrease (shrink) in Steam Generator level. This level decrease is sensed by the compensated Level Error network. The rapid decrease in Steam flow is sensed by the compensated Flow Error network. The rapid initial decrease in level indicates a need to increase the Feedwater flow rate, but the rapid decrease in Steam flow indicates a need to decrease the Feedwater flow rate. The selection of the system gains and time constants is such that the Flow Error dominates initially and results in an overall decrease in the Feedwater flow rate (in spite of the shrink in Steam Generator level), although the Feedwater flow rate will still be higher than the Steam flow rate. The result is a controlled increase in Steam Generator level. Level will overshoot the setpoint slightly. As the transient progresses, the Flow Error decays away (400 second time lag) and the Level Error signal dominates and adjusts the Feedwater flow rate as necessary to restore Steam Generator Narrow range level to its setpoint.

3.3 Response to a Reactor/Turbine Trip (RTO)

Assumptions

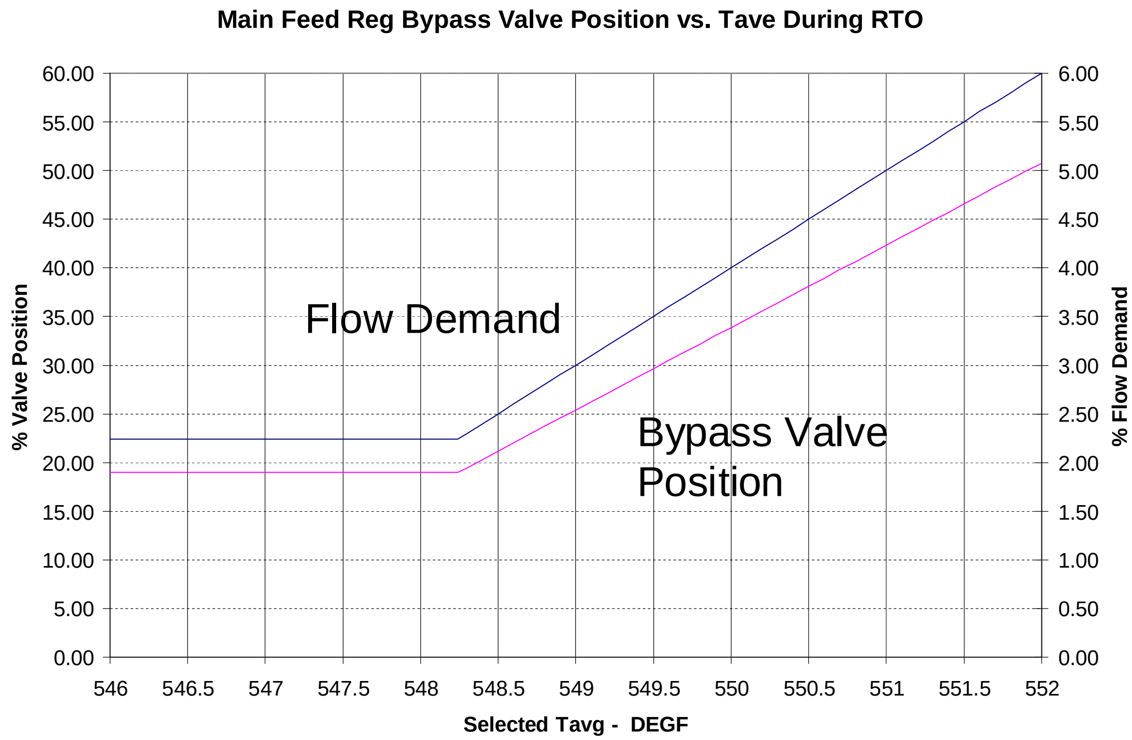
- Plant power is such that the FWCS is operating in the High Power mode,
- the Bypass valve is fully closed,
- all four control stations are in automatic,
- normal plant operating conditions exist,
- Steam Generator level is at or near its setpoint, and
- the plant is at steady state conditions when a Reactor trip occurs.

When the reactor trips the Main Turbine receives a trip signal and the resulting rapid increase in SG pressure results in a rapid decrease (shrink) in Steam Generator level.

The FWCS receives input signals from the Undervoltage Relays on the Control Element Drive Mechanism System (CEDMCS) buses 2C70 and 2C71. When these UV relays detect a low voltage on these buses due to the opening of the Reactor Trip Circuit Breakers, the FWCS is placed into the Reactor Trip Override, (RTO), mode. In addition, the Steam Generator level setpoint is automatically ramped from 70% to 60% due to turbine first stage pressure decreasing to <20% of the full power value.

When in RTO, the Total Flow Demand signal that is sent to the Feedwater valve controllers is set to a Flow Demand that will cause the Feedwater Regulating valve to ramp closed at 1.4% per second. The Main Feedwater Regulating Bypass Valve to go to a preset position based upon the selected RCS average temperature. A linear bias signal corresponding to a minimum of 2.24% (19% valve position) and up to 6% (50.7% valve position) flow demand will be applied to the bypass valve. The minimum bias is applied at an RCS average temperature of 548.24°F, and the maximum bias is applied at an RCS average temperature of 552°F. The maximum bias is clamped so that the 6% demand is not exceeded to prevent excessive RCS cooldown. The minimum bias is clamped so that a minimum of

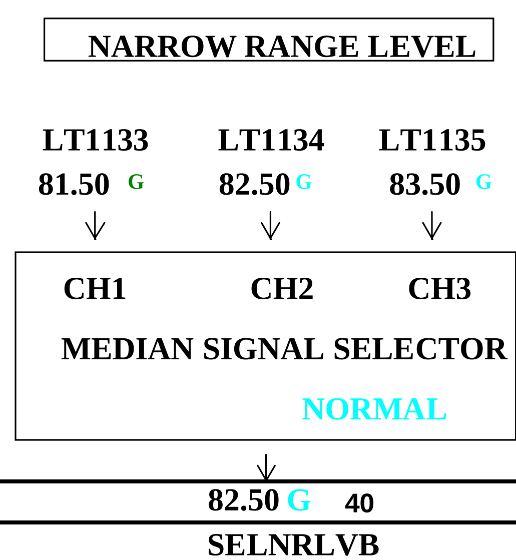
2.24% flow demand will slowly recover the Steam Generator level. In addition, the Main Feedwater pump speed controller receives a 0.0 demand signal and goes to a minimum speed of approximately 3150 rpm. When Narrow range level is restored to a setpoint of 55% ***and*** a minimum of 60 seconds from the time of the trip have transpired, RTO releases to the Low Power Mode control.



If any individual Fisher Porter Controller is in manual that respective component will not receive a RTO signal.

3.4 High Level Override (HLO)

A feature called High Level Override (HLO) will initiate when the selected SG Narrow Range Level increases to 82%. When the selected SG Narrow Range Level decreases to 80% the HLO will clear.



When initiated, the HLO signal sets the Total Flow Demand signal to 0 for the effected channel. This causes both Feedwater control valves to go closed. The Main Feedwater pump speed circuit selects the higher of the two FWCS flow demand signals. If the other Steam Generator is below the HLO setpoint its FWCS Flow Demand will be sent to the speed controller on the affected FWCS. When the Feedwater valves close, Steam

ORIGINAL BANK QUESTION 1554 WAS USED TO DEVELOP 2017 NRC EXAM QID#19 (Used on the 2008 NRC Exam)

Questions For All QID In Exam Bank

Bank:	1554	**Rev:**	0	**Rev Date:**	1/24/2008 2:23:58	**QID #:**	67	**Author:**	Coble
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW		
Search	1940012128	**10CFR55:**	41.7	**Safety Function**					
System Title:	Generic	**System Number**	GENERIC	**K/A**	2.1.28				
Tier:	3	**Group:**	1	**RO Imp:**	3.2	**SRO Imp:**	3.3	**L. Plan:**	A2LP-RO-FWCS
OBJ	11								

Description: Conduct of Operations - Knowledge of the purpose and function of major system components and controls.

Question:

Which of the following statements best describes the purpose and function of Main Feedwater Reactor Trip Override (RTO) signal after a Reactor Trip?

A. To rapidly refill the SGs with cold water to ensure an adequate RCS heat sink.

B. To slowly add feedwater to the SG to limit thermal stresses to the feed rings.

C. To rapidly add feedwater to the SGs to prevent an EFAS actuation.

D. To slowly refill the SGs with feedwater to prevent overcooling the RCS.

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☑	☑

Audit Exam History

2006	☐
2008	☐

Answer:

D. To slowly refill the SGs with feedwater to prevent overcooling the RCS.

Notes:

After a Reactor Trip an RTO signal is sent to the MFW regulating valve to ramp closed quickly and the MFW regulating bypass valves will ramp open slowly based on RCS Tave. The minimum bias to the bypass valves is clamped so that a minimum of 2.24% flow demand will slowly recover the Steam Generator level to prevent overcooling the RCS and loss of Pressurizer level. The feedwater being added to the SG is still relatively warm so there is no concern with shocking the SG feed rings. The feed water is slowly added so distracters A and C are incorrect.

References:

STM 2-69, Feedwater Control System, Section 3.3

Historical Comments:

Question 20

Bank: 2350 **Rev:** 2 **Rev Date:** 12/16/2016 **2017 TEST QID #:** 20 **Author:** Coble

Lic Level: RO **Difficulty:** 3 **Taxonomy:** H **Source:** NEW FOR 2017 NRC Exam

Search 000037A110 **10CFR55:** 41.10 **Safety Function** 3

Title: Steam Generator (S/G) Tube Leak **System Number** 037 **K/A** AA1.10

Tier: 1 **Group:** 2 **RO Imp:** 2.9 **SRO Imp:** 3.1 **L. Plan:** A2LP-RO-APSEC **OBJ** 9

Description: Ability to operate and/or monitor the following as they apply to the Steam Generator Tube Leak:- CVCS makeup tank level indicator

Question:

(REFERENCE PROVIDED)

Given the following:

* The plant is at 100% power steady state.
* Annunciator 2K11-A10 "SEC SYS RADIATION HI" comes in.
* Main Steam Line Rad Monitors start rising.
* SG Tube Leak N-16 Monitors start trending up.
* The CRS enters OP-2203.038, Primary to Secondary Leakage due to a calculated 29 gpm RCS leakrate.

If RCS temperature, and Pressurizer level remain constant during the next 10 minutes, the VCT will drop _________% and the procedural required action to mitigate the Steam Generator tube leak is to ____.

A. 8.6; commence a rapid plant shutdown to Mode 3

B. 5.4; commence a rapid plant shutdown to Mode 3

C. 8.6; trip the Reactor, actuate SIAS/CCAS, and commence SPTAs

D. 5.4; trip the Reactor, actuate SIAS/CCAS, and commence SPTAs

Answer:

A. 8.6; commence a rapid plant shutdown to Mode 3

Notes:

A is correct. VCT volume is = 33.8 gal/%. 10 minutes X 29 gpm = 290 gallons 290 gallons divided by 33.8 gpm = 8.6%. IAW AOP 2203.038 Step 8, this places the plant in Action Level 3. IAW Attachment A of the AOP Step 1.C.2, the plant needs to be shutdown to Mode 3.

B is incorrect as the drop in VCT level is incorrect but plausible as the action for the given conditions is correct and the leak rate is plausible if the candidate uses the PZR volume of 53.5 gal/%. 290 gallons divided by 53.5 gal/% = 5.4%

C is incorrect as the leak is smaller than the 44 gpm value required by the OP-2203.038, Primary to Secondary Leakage AOP step 6 to trip the plant but plausible as the VCT level drop of 8.6% is correct.

D incorrect as the VCT level drop is incorrect but plausible if the candidate uses the PZR volume of 53.5 gal/%. 290 gallons divided by 53.5 gal/% = 5.4% The action to trip the Reactor is the correct action to take if leakage is > 44 gpm.

This question matches the K&A as it requires the Candidate to monitor the CVCS MU Tank level indicator and determine the amount of SG Tube leakage and the correct operational action to take.

References:

AOP 2203.038 Primary to Secondary Leakage Rev 16 Entry Conditions (Verified reference updated 11/10/16);
AOP 2203.038 Primary to Secondary Leakage Rev 16 Step 3 (Verified reference updated 11/10/16);
AOP 2203.038 Primary to Secondary Leakage Rev 16 Step 6 and 8 (Verified reference updated 11/10/16);
AOP 2203.038 Primary to Secondary Leakage Rev 16 Attachment A Step 1.C.2 (Verified reference updated 11/10/16).

SOP 2305.002 RCS Leak Detection Rev 27 Exhibit 1 Component Volume Verses Level should be provided as a reference to the candidates. (Verified reference updated 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Reworded question stem and distractors along with analysis to eliminate any overlap with scenarios.

REV. 2 based on NRC Chief Examiner Feedback BNC. Made the last blank shorter to make the stem 2 lines instead of 3.

PRIMARY TO SECONDARY LEAKAGE

PURPOSE

This procedure provides actions for a Primary to Secondary Leak.

ENTRY CONDITIONS

ANY of the following conditions exist:

1. Rising trends on ANY of the following:
    - Secondary System Trend recorder (2RR-1057)
    - Main steamline monitors (2RITS-1007/2RITS-1057)
    - SG sample line monitors (2RITS-5854/2RITS-5864)
    - Condenser Off Gas monitor (2RITS-0645)
    - SG Tube Leak N-16 monitors (2RITS-0200/2RITS-0201)
2. Unexplained "SEC SYS RADIATION HI" annunciator (2K11-A10) alarm.
3. SG sample results indicate rising activity.
4. Unexplained "TROUBLE/LKRT HI" annunciator (2K11-K8) alarm.
5. Unexplained "RATE OF CHANGE HI" annunciator (2K11-J8) alarm.

EXIT CONDITIONS

ANY of the following conditions exist:

1. RCS leakage greater than 44 gpm, Reactor tripped and 2202.001, Standard Post Trip Actions entered.
2. RCS leakage greater than 44 gpm, all CEAs inserted, and procedure directs user to 2202.010 Exhibit 8, Diagnostic Actions or 2202.011, Lower Mode Functional Recovery.
3. ALL appropriate actions of this procedure have been performed.
4. ANY SFSC acceptance criteria NOT satisfied.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

> **NOTE**
>
> - **Steps marked with (*) are continuous action steps.**
> - **Steps marked with (■) are floating steps.**

1. **Open placekeeping page.**

2. **Notify Control Board Operators to monitor floating steps.**

> **NOTE**
>
> **N-16 monitors only calculate SG leak rates with plant power (CV-9000) greater than 20%.**

*3. **Determine Primary to Secondary leakrate by ANY of the following:**

- Computer RCS LKRT programs.
- Check PZR level stable and use Charging and Letdown mismatch minus Controlled Bleed Off.
- Check Letdown isolated and estimate RCS leak rate by total Charging flow minus Controlled Bleed Off.
- Chemistry leakrate calculation using 1604.013, Measurement of Primary to Secondary Leakage.
- SG Tube Leak N-16 monitors.
- Manual leakrate calculation.

2305.002 EXHIBIT 1 REVISED 06/30/16

COMPONENT VOLUME VS LEVEL

PAGE 1 OF 1

PRESSURIZER	53.5 gal/%
VCT	33.8 gal/%
CNTMT SUMP	39.0 gal/%
QUENCH TANK & RDT	17.6 gal/%
CCW SURGE TANK (CCW Loops Split)	9.3 gal/%
CCW SURGE TANK (CCW Loops Cross-Connected)	18.6 gal/%
SIT	120.6 gal/%
RWT	4787.8 gal/%
ABS	9.48 gal/%
2T-20	57.0 gal/%
2T-12	425 gal/%

INSTRUCTIONS	CONTINGENCY ACTIONS
■6. Check BOTH of the following are true: • **RCS leakage LESS than 44 gpm** • **PZR level maintained within 10% of setpoint**	**■6. IF EITHER of the following conditions exist:** • **RCS leakage greater than or equal to 44 gpm** • **PZR level continues to lower with ALL available Charging pumps running AND Letdown isolated** **THEN perform the following:** A. Verify closed Main Steam Supply valve to 2P7A from leaking SG: • SG "A" TO EMER FW PUMP TURBINE (2CV-1000-1) • SG "B" TO EMER FW PUMP TURBINE 2CV-1050-2 B. Refer to TS 3.7.1.2, Emergency Feedwater System. C. IF in Modes 1 OR 2, THEN perform the following: 1) Trip Reactor. 2) Actuate SIAS. 3) Actuate CCAS. 4) **GO TO** 2202.001, Standard Post Trip Actions.

(Step 6 continued on next page)

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

- **Leakage is confirmed if TWO independent radiation monitors trending upward.**
- **The probability of locating a tube leak after plant shutdown with leakrates less than 50 gpd (.035 gpm) is low.**

8. **WHEN confirmed primary to secondary leakrate determined,**
   THEN perform the applicable action per the table below:

Parameter	Value	Action
ANY SG OR TOTAL (BOTH SGs)	≥ 44 gpm	**RETURN TO** Step 6.
ANY SG	> 100 gpd (> 0.069 gpm)	Perform ACTION LEVEL THREE section of Attachment A while continuing with this procedure.
ANY SG	≥ 75 gpd (0.052 gpm)	Perform ACTION LEVEL TWO section of Attachment A while continuing with this procedure.
ANY SG	≥ 30 gpd (.021 gpm)	Perform ACTION LEVEL ONE section of Attachment A while continuing with this procedure.
TOTAL (both SGs)	≥ 5 gpd (.0035 gpm)	Perform RAISED MONITORING section of Attachment A while continuing with this procedure.
TOTAL (both SGs)	< 5 gpd (.0035 gpm)	Perform ACTION PLAN of Attachment A while continuing with this procedure.

ATTACHMENT A

PRIMARY TO SECONDARY LEAK REQUIRED ACTIONS

1. ACTION LEVEL **THREE** (> 100 gpd)

A. Record current time: ____________

* B. IF ANY SG leakrate rises to ≥44 gpm
THEN **GO TO** Step 6 in the body of this procedure.

C. IF at power,
THEN perform the following:

1) Refer to applicable reactivity plan.

2) Initiate the following using 2102.004, Power Operations OR 2203.053, Rapid Power Reduction as necessary to be ≤ 50% power within one hour of time recorded above, AND in Mode 3 in the following two hours:

*a) IF RCS leakage greater than or equal to 10 gpm,
THEN perform RCS boration using 2104.003, Chemical Addition, Attachment R, RCS Boration From the RWT OR BAMT.

b) IF leakage less than 10 gpm,
THEN perform EITHER of the following:

- RCS boration using 2104.003, Chemical Addition, Attachment R, RCS Boration from the RWT or BAMT.
- RCS boration using 2104.003, Chemical Addition, Exhibit 3, Normal RCS Boration at Power.

D. Ensure any out of service leak monitoring equipment returned to service as soon as practical.

E. IF PMS out of service,
THEN log required radiation monitor readings every 15 minutes locally at N-16 Cabinet (2C433) using Attachment B, Primary to Secondary Leak Rate Log:

- “A” SG (2RITS-0200)
- “B” SG (2RITS-0201)

F. Perform remaining applicable actions in the body of this procedure beginning with step 9.

Question 20 Reference Handout

2305.002 EXHIBIT 1 REVISED 06/30/16

COMPONENT VOLUME VS LEVEL

PAGE 1 OF 1

PRESSURIZER	53.5 gal/%
VCT	33.8 gal/%
CNTMT SUMP	39.0 gal/%
QUENCH TANK & RDT	17.6 gal/%
CCW SURGE TANK (CCW Loops Split)	9.3 gal/%
CCW SURGE TANK (CCW Loops Cross-Connected)	18.6 gal/%
SIT	120.6 gal/%
RWT	4787.8 gal/%
ABS	9.48 gal/%
2T-20	57.0 gal/%
2T-12	425 gal/%

Question 21

Bank:	2351	**Rev:**	1	**Rev Date:**	1/13/2017	**2017 TEST QID #:**	21	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC Exam Bank 1506		
Search	000028K203	**10CFR55:**	41.7	**Safety Function**	2				
Title:	Pressurizer (PZR) Level Control Malfunction	**System Number**	028	**K/A**	AK2.03				
Tier:	1	**Group:**	2	**RO Imp:**	2.6	**SRO Imp:**	2.9	**L. Plan:**	A2LP-RO-PZR
OBJ	9/10								

Description: Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following: - Controllers and positioners

Question:

Given the following plant conditions:

* The plant is at full power.
* Pressurizer Level Control System master controller is in AUTO REMOTE.
* Pressurizer Level Control is selected to "CHANNEL 4627-A".
* Pressurizer Heater Low Level Cutout Switch is selected to Both "A & B".
* Charging Pump Selector Switch, 2HS-4868, is in "A & B".
* Pressurizer Level Indicator 2LT-4627-1 fails high.
* No operator action is taken.

Based on these conditions Backup Charging Pumps 'A' and 'B' will ________________ and all the Pressurizer Heaters will ________________.

A. start; energize.

B. start; de-energize.

C. get a stop signal; energize.

D. get a stop signal, de-energize.

Answer:

C. get a stop signal; energize.

Notes:

C is correct because the high in service indicated level input to the Pressurizer Level controller and associated bistables cause PZR level to indicate above set point by > 4.5%. This will in turn send a stop signal to the backup charging pumps in this case pumps A and B since 2HS-4868 is in the A&B position (the lead pump will continue to run). The High Level signal will also energize all pressurizer heaters to ensure the additional water added to the PZR will quickly rise to saturated conditions.

A is incorrect as the BU charging pumps stop but plausible because the incorrect statements would occur if the in-service PZR level instrument failed low and the heaters will energize.

B is incorrect because the A and B charging pumps get a stop signal and the heaters energize but plausible as incorrect statements would occur if the in-service PZR level instrument failed low.

D is incorrect because the heaters energize but plausible as the A and B charging pumps do get a stop signal and but plausible as the heaters will de-energize if the in-service PZR level instrument failed low.

This question matches the K&A as it requires knowledge of the of the effects of a Pressurizer Level Control Malfunction on the associated interrelated controllers and positioners.

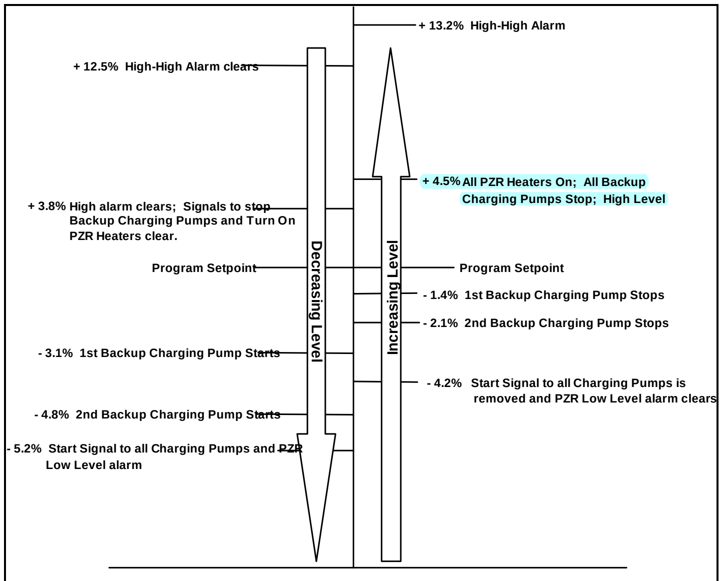
References:

STM_2-03-1_17-1 PZR Pressure and Level Control Section 3.2 and PZR Level Control Program Level Tree.(Verified reference updated 11/10/16)

Historical Comments:

NRC Exam Bank 1506 was used on the 2008 NRC Exam
To be used on the 2017 NRC Exam. Altered stem to failing high and removed references to Letdown flow in the distractors and formatted question stem and 4 choices into a 2x2 format based on feedback from the Facility Rep review.

Rev. 1 based on post submittal validation comments: In the 4th bullet, added the word "Switch" after "Cutout" to clarify noun name.

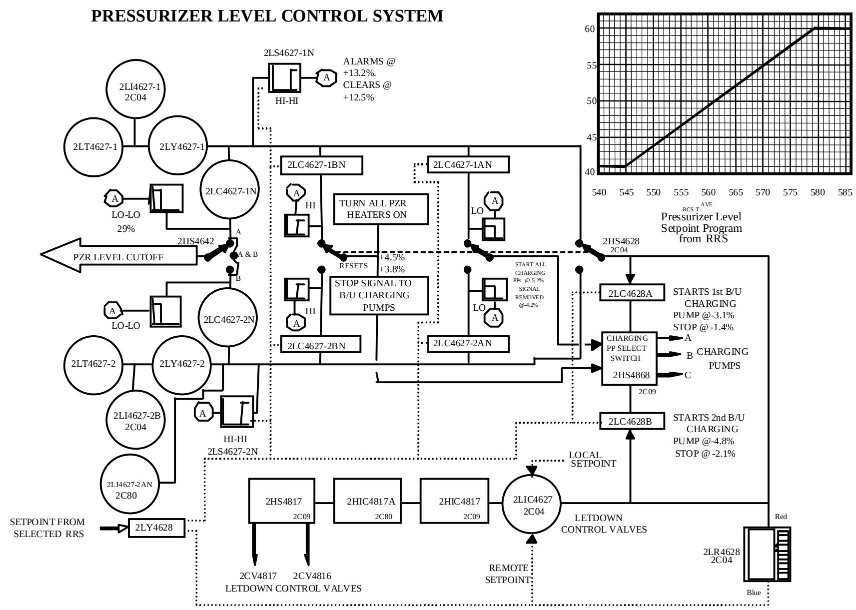


Pressurizer Level Control Program

The level tree diagram above illustrates the control actions performed by the Pressurizer Level Control System as indicated pressurizer level deviates from setpoint. Note that the Letdown Control valves to not operate as a function of deviation from setpoint per se'; they modulate as a function of controller output signal. For example, with the controller characteristics, if level is deviating high above setpoint the valves will open wider the longer the deviation exists.

3.2 Component Description

This section discusses the components associated with the Pressurizer Level Control System.



3.2.1 Level Transmitters 2LT-4627-1 and 4627-2

Two (2) redundant narrow range level transmitters provide two separate pressurizer level control channels. 2LT-4627-1 and 2LT-4627-2 send level signals to pressurizer level control channels "A" and "B" respectively. Each channel has remote level indication on 2C04. Both transmitters are narrow range transmitters and detect pressurizer water level that is between 36.375" and 324.375" above the bottom of the pressurizer[2] (288" span). Channel "A" level in percent is indicated on 2LI-4627-1 while Channel "B" can be read on 2LI-4627-2B.

These level transmitters are calibrated to read accurately at a water temperature of 653°F. During a plant cooldown, as RCS temperature lowers, indicated level as read on these instruments will be higher than actual level. This is due to the rise in density of the water column in the pressurizer, making the DP cell which is the actual measuring device "think" that there is more water in the pressurizer than there really is.

However the signals from the level transmitters are temperature compensated by networks located in 2C336 and 2C384 respectively. This allows accurate pressurizer level to be provided to the control system for the full range of pressurizer temperatures from cold shutdown to normal operating temperature.

3.2.2 PZR Low Level Cutoff Selector Switch 2HS-4642

Handswitch 2HS-4642 is the Pressurizer Low Level Cutoff Select switch and can be used to determine which level channel is used to de-energize the PZR heaters during extremely low level conditions. This handswitch is a 3-position switch. The positions available are "A", "A" &

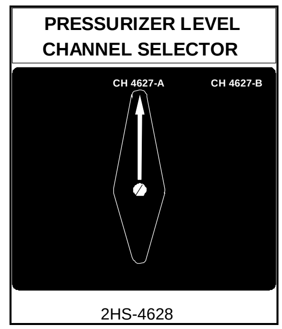
FOOTNOTES:

2 The bottom of the Pressurizer is actually measured from the bottom of the Pressurizer support skirt and not the bottom of the lower head. The upper level tap is shared with the Pressurizer Pressure transmitters.

B", and "B". The position selected determines which pressurizer level transmitter(s) (2LT-4627-A or 2LT-4627-B) will cause the pressurizer heaters to de-energize on low-low level. When in position "A & B" either transmitter less than 29% will cause the cut off to occur. It should be noted that a loss of power to the control relays for this feature will cause the respective breakers to trip.

Whenever indicated pressurizer level lowers to less than 29%, the Pressurizer Level Control system closes contacts in the pressurizer heaters circuitry causing them to de-energize. The purpose of this feature is to prevent damage to the PZR heaters due to overheating should they become uncovered. The tallest PZR heater elevation is actually 25.9% (74.9" above the Datum Line) so the 29% value is conservative. The pressurizer level indicators show collapsed liquid level (comparison of weights of water) to the Operators. In actuality with the heaters in service, considerable voiding exists due to the two-phase operation of the pressurizer. This elevated voiding the two-phase level in the pressurizer above that which is "seen" by the level instruments. This higher level should also provide for adequate heat transfer from the heaters, to prevent burnout, counteracting any instrument uncertainties.

3.2.3 PZR Level Control Selector Switch 2HS-4628



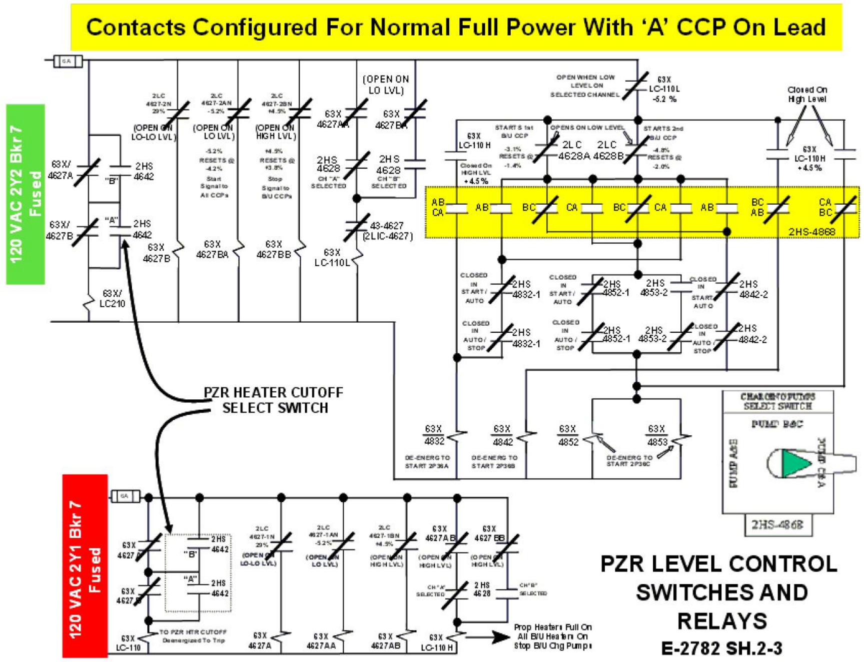
The Pressurizer Level Channel Selector switch 2HS-4628 is used to select which level channel will serve as the controlling channel. This handswitch is located on 2C04 and is a 2-position select switch. The operator can place it in either "CH 4627-A" or "CH 4627-B" as desired. Once selected, that channel is used for all controlling actions performed by the Pressurizer Level Control System. This selected channel is compared to the pressurizer level program setpoint from the Reactor Regulating System (RRS) in several bistable comparators in the system as well as the PZR level controller 2LIC-4627. When indicated level as sensed by the selected channel deviates from the setpoint, various control actions to take place to restore level. As mentioned previously, the program pressurizer level setpoint varies as function of T_{ave}.

3.2.4 Charging Pump Controls

The 3 positive displacement Charging pumps of the Chemical and Volume Control System (CVCS) can be started and stopped as needed by the Pressurizer Level Control System to maintain the desired PZR level. As the "selected" PZR level (see 2HS-4628 discussion above) deviates from the desired PZR level setpoint several comparators send control signals to the charging pump circuitry via the charging pump select switch 2HS-4868.

3.2.5 Charging Pump Select Switch 2HS-4868

The Charging Pump Selector switch 2HS-4868 is located on 2C04 and is used by the Control Board Operator to determine which charging



pumps are to be designated as the "Backup" charging pumps. The available positions for this switch are "A & B", "B & C", and "C & A". The first pump designated on each selection will designate the "1st Backup" pump while the next pump will be the "2nd Backup" charging pump. The pump that is not designated will receive a start signal and normally always be running if it's respective control handswitch is in AUTO.

The above figure shows how these pumps are started and stopped through the use of relays within the Pressurizer Level Control System. A detailed description of this circuit immediately follows.

3.2.6 Backup Pump Start/Stop Circuit

Referring to the Charging pump control circuit figure above, it can be seen that two separately powered control schemes make up the PZR Level Control Charging pump start circuit. These are not entirely redundant circuits. If power is lost to either string, the control of the Backup Charging pumps from deviating pressurizer level will be degraded. Take note of the fact that the relays that actually cause the Charging pumps to start have to be de-energized to start the pumps. If 2Y2 Breaker 7 is opened, a start signal will be sent all of the Charging pumps.

2LC-4628A will send a start signal to the designated 1st backup charging pump (i.e., if "A & B" selected on 2HS-4868, the "A" charging pump is the 1st backup pump while "B" is the 2nd) if indicated level on the controlling channel deviates from the RRS pressurizer level setpoint by -3.1% and decreasing. 2LC-4628A will reset when pressurizer level reaches -1.4% increasing.

2LC-4628B will start the 2nd backup charging pump if pressurizer level continues to lower and reaches -4.8% below setpoint. This start signal to the 2nd backup pump will be removed if level rises to just -2.0% below setpoint.

If level continues to lower on the selected level channel even after both backup pumps should have started, the Pressurizer Level Control

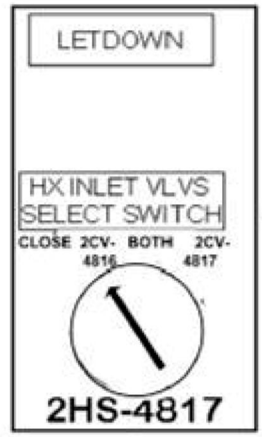
to the letdown valves using the output pushbuttons located just below the A/M pushbutton.

3.2.10 2HIC-4817A (2C80)

The output from 2HIC-4817 passes through another controller, 2HIC-4817A, located at the remote shutdown panel, 2C80.

When 2HIC-4817A on 2C80 is in auto, the signal from 2HIC-4817 on 2C09 will control the letdown flow valves. If 2HIC-4817A is in manual, the output signal to the letdown flow control valves is determined by the output on 2HIC-4817A on 2C80.

3.2.11 2HS-4817 (2C09)



The selector switch used by the operator to select which letdown flow control valve will receive position signals is located on 2C09. This switch is 2HS-4817 and is labeled "HX Inlet Valves Select Switch".

The "Both" position is used when the RCS is at very low pressures during startup or shutdowns when the flow through one valve alone is insufficient.

The close position allows the operator to apply a hard shunted close to both valves in the event of a fire that could threaten loss of control of either 2CV-4816 or 2CV-4817. This handswitch design inhibits spurious valve opening.

The signal from 2HIC-4817/2HIC-4817A is routed through this handswitch to the selected valve(s).

3.2.12 PZR Heater Control from PZR Level Control

As discussed previously, the Pressurizer Heaters receive a cutoff signal if pressurizer level lowers to 29%. The channel that initiates this cutoff signal is determined by the position of the Pressurizer Heater Cutoff Select Switch 2HS-4642. A level comparator on the channel that is selected senses the low-low level condition and this results in contacts closing in the auto off circuits for the pressurizer heaters. These contacts cause the Load Center (2B5, 2B6, 2B9, and 2B10) breakers for all pressurizer heaters to trip. The proportional heater handswitches are "spring return-to-normal" type handswitches, which means that if these handswitches in the On position, the proportional heater breakers would be closed. The Operator would have to deliberately hold the handswitch in the On position. If actual pressurizer level is in deed <29% and any of the heater banks are on then the heaters may burnup due to overheating.

If the selected level channel senses a high level condition, > +4.5% above the level setpoint, other contacts close in the heater Auto On circuits. These contacts cause the pressurizer backup heaters to come on and the proportional banks of heaters to go to maximum firing rate on their SCRs, if the associated handswitches are not in OFF.

Other contacts also open in the heater Auto Off circuits on a high level condition preventing the heaters from going off automatically due to pressurizer pressure. Both sets of contacts return to their normal state should pressurizer level lowers to within +3.8% of setpoint. The purpose of this feature is to minimize the subcooling in the pressurizer water phase due to an insurge of "colder" water from the RCS.

3.2.13 Pressurizer Level Control Tree

ORIGINAL BANK QUESTION 1506 WAS USED TO DEVELOP 2017 NRC EXAM QID#21 (Used on the 2008 NRC Exam)

Questions For All QID In Exam Bank

Bank:	1506	**Rev:**	000	**Rev Date:**	10/10/2001 5:35:5	**QID #:**	19	**Author:**	Coble
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC Bank 0341 (2002 NRC Exam)		
Search	000028K203	**10CFR55:**	41.7 / 45.7	**Safety Function**	2				
System Title:	Pressurizer (PZR) Level Control Malfunction	**System Number**	028	**K/A**	AK2.03				
Tier:	1	**Group:**	2	**RO Imp:**	2.6	**SRO Imp:**	2.9	**L. Plan:**	A2LP-RO-PZR
OBJ	9/10								

Description: Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following: - Controllers and positioners.

Question:

Given the following plant conditions:

* The plant is at full power.
* Pressurizer Level Control System master controller is in AUTO REMOTE.
* Pressurizer Level Control is selected to "CH 4627-A".
* Pressurizer Heater Low Level Cutout is selected to Both "A & B".
* Charging Pump Selector Switch, 2HS-4868, is in "A & B".
* Pressurizer Reference leg 2LT-4627-1 develops a leak.
* No operator action is taken.

WHICH ONE of the following describes the response of the Pressurizer Level Control System?

A. Charging Pumps A and B start, heaters energize, letdown flow rises.

B. Charging Pumps A and B start, heaters cutout, letdown flow lowers.

C. Charging Pumps A and B get a stop signal, heaters energize, letdown flow rises.

D. Charging Pumps A, B, and C get a stop signal, heaters cutout, letdown flow rises.

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☑	☑

Audit Exam History

2006	☐
2008	☐

Answer:

C. Charging Pumps A and B get a stop signal, heaters energize, letdown flow rises.

Notes:

The reference leg leak will cause a high indicated level input to the Pressurizer Level controller and associated bistables to cause level to indicate above set point by > 4.5%. This will in turn send a stop signal to the backup charging pumps in this case pumps A and B (the lead pump will continue to run), a signal to energize all pressurizer heaters and force the Letdown Flow Controller to maximum output.

References:

STM 2-3-1, Pressurizer Pressure and Level Control, Sections 3.2
2103.005, Step 6.7 (Pressurizer Operations)

Historical Comments:

Question 22

Bank:	2352	**Rev:**	2	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	22	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	Modified NRC Exam Bank 1508		
Search	000051A202	**10CFR55:**	41.10	**Safety Function**	4				
Title:	Loss of Condenser Vacuum	**System Number**	051	**K/A**	AA2.02				
Tier:	1	**Group:**	2	**RO Imp:**	3.9	**SRO Imp:**	4.1	**L. Plan:**	A2LP-RO-AVAC
OBJ	2								

Description: Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: - Conditions requiring reactor and/or turbine trip

Question:

Given the following:

* Plant power ascension is in progress following a refueling outage.
* Reactor power is at 64% and slowly rising.
* SDBCS is in its normal line up.
* Condenser vacuum has started degrading.
* Annunciators 2K03-A3/A4 "2E11A/B Pressure HIGH" come in.
* Condenser vacuum is 5.7 inches Hg absolute and rising.

In accordance with OP-2203.019, Loss of Condenser Vacuum, the _______________ is procedurally REQUIRED to be tripped if condenser vacuum exceeds a MAXIMUM of ___________ inches Hg absolute.

A. Turbine; 7.0

B. Reactor; 7.0

C. Turbine; 6.0

D. Reactor; 6.0

Answer:

B. Reactor; 7.0

Notes:

Answer B is correct as the normal system line up capacity for steam bypass valves and atmospheric dump valves (ADVs) is ~51 percent (refer to page 2 of STM 2-23 Rev 17 Section 1.2) because the (2) Upstream ADVs are maintained in the "Off" position and isolated using Motor operated Isolation valves because of valve failure during plant S/U testing. When vacuum rises 5.3 inches Hg absolute, the bypass valves to the condenser become unavailable due to condenser interlock so the total currently available ADV capacity is ~23% therefore, the direction given in AOP 2203.019, Loss of Condenser Vacuum, REV 13 Step 7.0, Contingency Action A is applicable. Even if the upstream ADVs were opened then the available ADV capacity is 46% which is less than 64% power given in the stem.

Answer A is incorrect as this would cause an automatic Reactor trip but plausible if the candidate believes the total Capacity of the SDBCS is ~74% as it would be if the upstream ADVs were not isolated and no condenser interlock existed.

Answer C is incorrect as this would cause an automatic Reactor trip but plausible if the candidate believes he has to trip the Main Turbine to prevent going into the UNACCEPTABLE region of AOP 2203.019, Loss of Condenser Vacuum, Attachment 'A'. The procedure has you reduce power and turbine load to get below the unacceptable region of Attachment 'A' in the AOP but not trip.

Answer D is incorrect based on the wrong vacuum setpoint to trip the reactor but plausible if the candidate believes he has to trip the Reactor to automatically trip the Main Turbine to prevent going into the UNACCEPTABLE region of AOP

2203.019, Loss of Condenser Vacuum, Attachment 'A'. The procedure has you reduce power and turbine load to get below the unacceptable region of Attachment 'A' in the AOP but not trip.

This question matches the K&A because the candidate interpret the degrading vacuum conditions and determine when actions should be taken to trip the reactor which will then trip the turbine.

References:

2203012C ANNUNCIATOR 2K03 CORRECTIVE ACTION Rev 33 Window A-3 (Verified reference updated 11/10/16);
2203012C ANNUNCIATOR 2K03 CORRECTIVE ACTION Rev 33 Window A-4 (Verified reference updated 11/10/16);
STM_2-23_17-1 SDBCS Section 1.2 (Verified reference updated 11/10/16); AOP 2203.019, Loss of Condenser Vacuum, REV 13 Step 5.0, Contingency Action A and Step 6 (Verified reference updated 11/10/16);
AOP 2203.019, Loss of Condenser Vacuum, REV 13 Attachment A (Verified reference updated 11/10/16).

Historical Comments:

NRC Exam Bank 1508 was used on the 2008 NRC Exam
To be used on the 2017 NRC Exam but modified for the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed the numbers to be at 7.1/6.1 in the answer/distractors

REV. 2 based on NRC Chief Examiner Feedback BNC. Changed the stem question to a statement with blanks in a 2 X 2 format as requested and used the word MAXIMUM to eliminate any subset issues.

ANNUNCIATOR 2K03

A-3

2EllA PRESSURE HI

1.0 CAUSES

1.1 Main Condenser 2E-llA pressure (2PIS-0605) > 5.3 inches HgA.

2.0 ACTION REQUIRED

2.1 Validate alarm using PMS displays/trends and Condenser pressure indications:

- 2E11A (2PIS-0605)
- 2E11B (2LIS-0644)

2.2 IF desired,
THEN align Condenser Vacuum pumps to Manual Hogging operation using Condenser Vacuum System (2106.010).

2.3 IF **UNEXPLAINED** rise in Condenser pressure,
THEN **GO TO** Loss Of Condenser Vacuum (2203.019).

NOTE

- Examples of past high ambient temperature conditions include outside air temperature of 105°F and Circ Water inlet temperature of 100°F with outlet temperature of 135°F. (CR-ANO-2-2011-02765)
- An explained condenser pressure rise is usually caused by conditions such as high Circ Water temperature or high outside ambient temperature.

2.4 IF **EXPLAINED** rise in Condenser pressure,
THEN perform the following:

* 2.4.1 Verify plant operation within ACCEPTABLE region of Loss Of Condenser Vacuum (2203.019), Attachment A, Back Pressure Limits.

2.4.2 Perform operating checks on running Vacuum pump(s) 2C-5A and/or 2C-5B. Refer to Condenser Vacuum System (2106.010).

2.4.3 Notify Condenser Vacuum System Engineer.

2.4.4 Set a programmable alarm for P0605 at 5.5 inches HgA.

2.4.5 IF Condenser pressure greater than 5.6 inches HgA (2PIS-0605 OR 2PIS-0644),
THEN initiate Bridge call with Operations Management.

(A-3 Continued on next page)

ANNUNCIATOR 2K03

A-4

2EllB PRESSURE HI

1.0 CAUSES

1.1 Main Condenser 2E-11B pressure (2PIS-0644) > 5.3 inches HgA.

2.0 ACTION REQUIRED

2.1 Validate alarm using PMS displays/trends and Condenser pressure indications:

- 2E11A (2PIS-0605)
- 2E11B (2PIS-0644)

2.2 IF desired,
THEN align Condenser Vacuum pumps to Manual Hogging operation using Condenser Vacuum System (2106.010).

2.3 IF **UNEXPLAINED** rise in Condenser pressure,
THEN GO TO Loss Of Condenser Vacuum (2203.019).

(A-4 Continued on next page)

1.0 Introduction

1.1 System Functions

The functions of the Steam Dump and Bypass Control System (SDBCS) are as follows:

1) The system automatically dissipates a limited amount of excess energy in the Reactor Coolant System by regulating the flow of Main Steam through Turbine Bypass and Atmospheric Dump valves. Steam Header pressure is thereby controlled so that:

   a. Small load rejections can be accommodated without tripping the reactor or lifting either the pressurizer or main steam safety valves.

   b. Hot zero-power, or Hot Standby, conditions can be maintained.

   c. Desired RCS thermal conditions can be achieved during periods when Reactor power is required to be greater than turbine power (for example, during Turbine synchronization).

2) The system in manual allows operator control of Reactor Coolant System temperature during heatup or cooldown when the condenser is available using the Condenser Bypass valves or , when the condenser is unavailable, using the Atmospheric Dump valves.

1.2 General System Description

The Steam Dump and Bypass Control System (SDBCS) is a non-safety related system whose primary purpose is to provide a steam path from the Steam Generators to the Condenser or to the atmosphere. This is necessary in order to limit Main Steam Header pressure and, as a result, Reactor Coolant System temperatures during periods where the Main Turbine is unavailable or a Plant load transient is occurring.

The SDBCS controls seven valves. Four of the valves *Dump* to the atmosphere and have a capacity of 46% of total steam flow. Two of these Atmospheric Dump valves (ADVs) are located upstream of the Main Steam Isolation valves and the remaining two ADVs are located downstream.

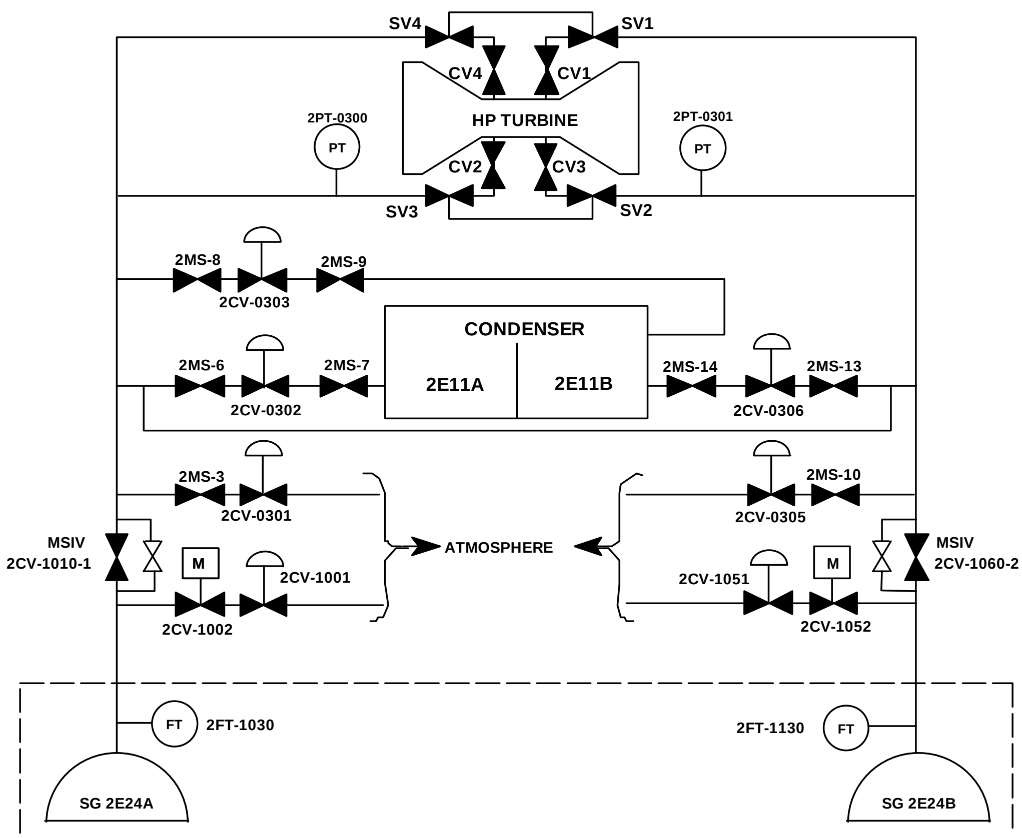
Three Turbine *Bypass* valves are provided to bypass Main Steam around the Turbine directly to the main Condenser. One of these valves has a capacity of 5% and the two other bypass valves have a capacity of ~11.5%. The combined Dump and Bypass capacity is ~74% of total Steam flow. The original design of the 11.5% Bypass and Dump valves was 13 1/3% capacity each, but the valves internals were modified during plant startup testing to reduce valve failure and improve reliability.

The following table lists the capacity of each of the SDBCS valves (900 psia @ turbine stop valves) and whether the valve is a Dump valve or a Bypass valve.

SDBCS Valve Summary Table

VALVE NUMBER	CAPACITY	TYPE
2CV-0303	5%	Bypass
2CV-0302	11.5%	Bypass
2CV-0306	11.5%	Bypass
2CV-0301	11.5%	Dump
2CV-0305	11.5%	Dump
2CV-1001 (Upstream MSIV)	11.5%	Dump
2CV-1051 (Upstream MSIV)	11.5%	Dump

The normal mode of operation is with the (3) Bypass valves and both (2) Downstream Atmospheric Dump valves (ADVs) set up for Automatic operation. The (2) Upstream ADVs are maintained in the "Off" position and isolated using Motor operated Isolation valves because of valve failure during plant S/U testing. This results in a normal line-up system capacity of ~51%.



SDBCS ONE-LINE DIAGRAM

The SDBCS is designed so that no single component failure or Operator error will result in the opening of more than one valve.

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

- **Main Feedwater pumps trip at Condenser vacuum of 13.4 inches HG Abs.**
- **Turbine Generator trips at Condenser vacuum of 7.8 inches HG Abs.**
- **SDBCS Condenser interlock automatically resets at 5.15 inches HG Abs with controllers in automatic.**

■5. **Check Condenser vacuum less than 7 inches HG Abs.**

ADV capacity is 23% with upstream ADVs isolated and 46% with the upstream ADVs open

■5. **IF Condenser vacuum greater than or equal to 7 inches HG Abs, THEN perform the following:**

A. IF Reactor power greater than available ADV capacity, THEN perform the following:

1) Trip Reactor.

2) **GO TO** 2202.001, Standard Post Trip Actions.

B. IF Reactor power less than available ADV capacity, THEN perform the following:

1) Verify ALL available ADVs in AUTO.

2) Open Upstream ADV Isolation valves:

- 2CV-1002
- 2CV-1052

3) Trip Main turbine.

4) **GO TO** 2203.024, Loss Of Turbine Load.

C. IF Reactor trips, THEN **GO TO** 2202.001, Standard Post Trip Actions.

INSTRUCTIONS	CONTINGENCY ACTIONS
■6. **Check Condenser Vacuum stable or improving.**	■6. **Perform the following as needed to maintain Condenser Vacuum less than 5.3 inches HG Abs:** A. Commence Emergency Boration using 2202.010 Exhibit 1, Emergency Boration. B. Perform Attachment A, Backpressure and Temperature Limits. C. IF CEA insertion required to maintain RCS Tc less than 554.7°F, THEN perform the following: 1) Insert Group 6 or Group P CEAs using 2105.009 Exhibit 3. 2) Maintain CEAs in Acceptable region of COLR. D. IF RCS Tc exceeds 554.7°F, THEN refer to TS 3.2.6 Reactor Coolant System Cold Leg temperature. E. Check condenser vacuum maintained within Acceptable region of Attachment A, Backpressure and Temperature Limits. F. WHEN Condenser vacuum stable or improving, THEN secure Emergency Boration as desired. G. Perform the following: 1) Refer to Attachment B, Circ Water Temperature Limiting Rx Power Levels, for a suggested maximum power level for the present Circ Water inlet temperature. 2) Commence a power reduction using 2102.004, Power Operations until condenser pressure is less than 5.15 inches HG Abs.

ATTACHMENT A
BACKPRESSURE AND TEMPERATURE LIMITS

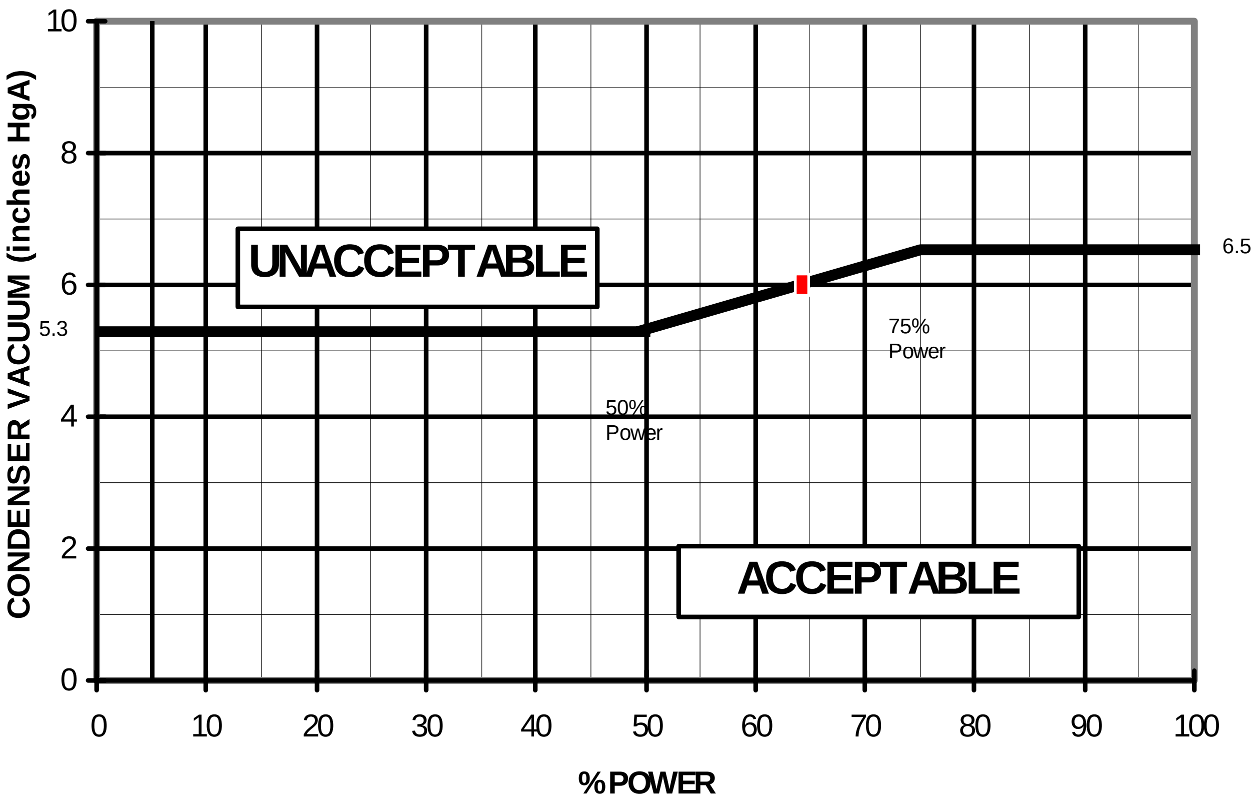
CAUTION

Operation of Turbine Generator at loads less than 285 MW and Condenser vacuum greater than 5.3 inches HG will raise probability of turbine damage.

1. Reduce Turbine Load to maintain the following:

   - Condenser Pressure within acceptable region of Figure 1
   - RCS Tc 542°F to 554.7°F

FIGURE 1



PARENT QUESTION 1508 WAS USED TO MODIFY 2017 NRC EXAM QID#22 (Used on the 2008 NRC Exam)

Data for 2008 NRC SRO Exam

Bank: 1508 **Rev:** 002 **Rev Date:** 10/29/2007 9:39:4 **QID #:** 21 **Author:** Coble

Lic Level: R **Difficulty:** 2 **Taxonomy:** F **Source:** Modified NRC 0028 (1998 NRC Exam)

Search 000051A202 **10CFR55:** 43.5 / 45.13 **Safety Function** 4

System Title: Loss of Condenser Vacuum **System Number** 051 **K/A** AA2.02

Tier: 1 **Group:** 2 **RO Imp:** 3.9 **SRO Imp:** 4.1 **L. Plan:** A2LP-RO-EAOP **OBJ** 14

Description: Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: - Conditions requiring reactor and/or turbine trip.

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☑	☑

Audit Exam History

2008	☐

Question:

Given the following:

* Reactor power is at 15% and steady.
* SDBCS is in its normal line up for this power.
* A main turbine roll to 1800 rpm is in progress.
* Condenser vacuum has begun degrading.
* Annunciators 2K03-A3/A4 "2E11A/B Pressure HIGH are actuated.
* Both condenser Vacuum pumps are running.

In accordance with OP 2203.019, Loss of Condenser Vacuum, which one (1) of the following actions should be taken by the Crew if vacuum continues to degrade?

A. Trip the turbine if vacuum exceeds 5.3 inches Hg absolute.

B. Trip the Reactor and Turbine if vacuum exceeds 5.3 inches Hg absolute.

C. Trip the turbine if vacuum exceeds 7.0 inches Hg absolute.

D. Trip the Reactor and Turbine if vacuum exceeds 7.0 inches Hg absolute.

Answer:

C. Trip the turbine before exceeding 7 inches Hg absolute.

Notes:

Answer "A" is incorrect because although this is in the unacceptable region, the actions of the procedure try to restore vacuum before tripping at 7.0 "HG absolute.
Answer "B" is incorrect because reactor power is within the capacity of SDBCS and the reactor should not be tripped at this time and the vacuum is less than 7.0 " HG absolute.
Answer "D" is incorrect because reactor power is within the capacity of SDBCS and the reactor should not be tripped.

References:

2203.019, Loss of Condenser Vacuum, Step 7.0, contingency action B and Attachment A

Historical Comments:

Rev 001 - 08/11/98 - Revised distracter "B" from "Trip the reactor and go to 2202.001, Standard Post Trip Actions" to "Raise Tave to reduce SDBCS load" due to NRC review comments that "B" was also a correct answer.

Question 23

Bank:	2353	**Rev:**	2	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	23	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW for 2017 NRC Exam		
Search	0000242107	**10CFR55:**	41.7	**Safety Function**	1				
Title:	Emergency Boration	**System Number**	024	**K/A**	2.1.7				
Tier:	1	**Group:**	2	**RO Imp:**	4.4	**SRO Imp:**	4.7	**L. Plan:**	A2LP-RO-EBOR
OBJ	4								

Description: Conduct of Operations - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Question:

Given the following:

* Power to Green Vital 4160 VAC ESF Bus 2A4 has been lost.
* The Green EDG fails to start and load Electrical Bus 2A4.
* The Plant subsequently trips and SPTAs are in progress.
* All but one (1) CEA rod bottom lights are illuminated Green.
* CEACs 1 and 2 agree with the rod bottom light indications.
* Reactor power is lowering.

Based on the above conditions, SPTAs directs ______________________________________.

A. commencing emergency boration using gravity feed valves

B. depressing BOTH "Reactor Trip" pushbuttons on 2C03

C. commencing emergency boration using either BAM Pump

D. opening the breakers for the MG sets 2B712 and 2B812

Answer:

A. commencing emergency boration using gravity feed valves

Notes:

A is correct because SPTAs conservatively directs emergency boration be commenced if any CEAs are stuck out but all other EOP safety functions allow 1 CEA to be stuck out without emergency boration. Also use of gravity feed valves is the only RED Train option to get boric acid flowing to the RCS as the BAM Pumps and the RWT Outlet are powered from Green Train Vital AC which has been lost. The VCT outlet valve is powered from Red Train ESF busses so it can be closed.

B is incorrect because SPTAs will direct emergency boration if one CEA does not fully insert but plausible as all other EOP safety functions allow 1 CEA to be stuck out without emergency boration and the action listed would be a potential correct action if Reactor Power was not lowering.

C is incorrect as the BAM pumps cannot be used due to no power available but plausible as SPTAs will direct emergency boration be commenced with only one CEA stuck out and BAM pumps are the preferred method to use if power is available in case the VCT outlet does not close..

D is incorrect because SPTAs will direct emergency boration if one CEA does not fully insert but plausible as all other EOP safety functions allow 1 CEA to be stuck out without emergency borating and the action listed would be a potential correct action if Reactor Power was not lowering and CEDMCS buses were still energized.

This question matches the K&A because the candidate must be able to evaluate plant performance based on plant electrical

power and available shutdown reactivity for the reactor based on instrument interpretation and make the correct line up decision for emergency boration.

References:

EOP 2202.001 SPTAs Rev. 15 Step 3 (Verified reference updated 11/10/16);
EOP 2202.001 SPTAs TG Rev. 15 Step 3 (Verified reference updated 11/10/16);
EOP 2202.010 Standard Attachments Rev. 23 Exhibit 1 Emergency Boration (Verified reference updated 11/10/16);
STM_2-04__31-1 CVCS Drawing (Verified reference updated 11/10/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Reworded the stem and changed the 2nd part of distractors B and D to more plausible conditions. Updated the distractor analysis.

REV. 2 based on NRC Chief Examiner Feedback BNC. Removed the 1st part of the distractors to determine if emergency boration is or is not required and changed the stem to just ask the second part. Added "and/or any applicable standard attachment/exhibit directs" as the actual guidance for selecting the correct emergency boration flow path is in OP 2202.010, Standard Attachments, Exhibit 1.

INSTRUCTIONS	CONTINGENCY ACTIONS
3. Check Reactivity Control established as follows:	
_____ A. Reactor power lowering.	A. Perform the following: 1) Perform the following as needed to manually trip CEAs: • Depress BOTH Reactor Trip pushbuttons on 2C03. • Depress DSS Emergency Reactor Trip pushbutton on 2C03. • Depress BOTH Manual Reactor Trip pushbuttons on 2C14. 2) IF ANY CEDMCS bus remains ENERGIZED THEN perform the following: a) Open the following breakers on 2C10 to de-energize MG sets: • 2B712 • 2B812 b) WHEN breakers have been open 10 seconds, THEN close 2B712 and 2B812. _____ 3) Check reactor power lowering.
B. Check startup rate is negative.	
_____ C. ALL CEAs fully inserted by observing ANY of the following: 1) CEA Rod bottom lights illuminated. 2) CEAC 1 indicates ALL CEAs fully inserted. 3) CEAC 2 indicates ALL CEAs fully inserted.	_____ C. Verify emergency boration in progress using Exhibit 1, Emergency Boration.

A conservative approach of "checking ALL CEAs fully inserted" is used. The operator can use CEA Rod Bottom lights, CEAC # 1, or CEAC # 2 to perform this check. At least one of these three methods must indicate that all CEAs are fully inserted. If this is not the case, emergency boration must be commenced and continued until a more detailed assessment of CEA status and emergency boration conditions can be performed (when time is less critical).
This prevents the possibility of the operators making an inaccurate determination as to whether one or more than one CEA is not fully inserted. If only one CEA is later determined to be not fully inserted, emergency boration may be secured if appropriate termination conditions exist. However for EOL conditions where low initial boron concentrations are present in the RCS, additional Boron may need to be added to ensure Tech Spec required Shutdown Margin is satisfied (3). It should be noted that the Safety Analysis ensures adequate Shutdown Margin is present during normal operation. The amount of boration which may occur is deemed insignificant when compared to the possibility of not commencing boration if more than one CEA is not fully inserted.

The condition “adequate shutdown margin” is not used in the EOP contingency action. Establishing emergency boration meets the requirements of the technical specifications and a determination of shutdown margin will be performed later (when time is less critical). An exhibit is used to provide the alignment for emergency boration. This allows the control board operator to perform the alignment without the CRS having to provide the direction.

As per 1015.021, ANO-2 EOP/AOP Users Guide (5) both power lowering and ALL CEAs inserted are required to meet the safety function.
The Emergency Boration contingency has also been identified as meeting the safety function.
Power lowering is checked in SPTA and power < 10^{-1} % reactor power is checked throughout the ORPs and Functionals. (6)

SOURCE DOCUMENTS:

1. E-2008, Single Line Meter & Relay Diagram, 480 Volt Load Centers.
2. 2105.009, CEDM Control System Operation.
3. SAR Chapter 9.3.4 (Chemical and Volume Control System), Section 9.3.4.4.1.
4. 2202.010, Standard Attachments, Exhibit 1, Emergency Boration.
5. 1015.021, ANO-2 EOP/AOP Users Guide
6. ANO-2 EOP Setpoint Document, Setpoint U.11.

EXHIBIT 1

EMERGENCY BORATION

1. Select ONE of the following Emergency Boration flowpaths:

FLOWPATH	ACTIONS REQUIRED
A. Gravity Feed	A. Verify at least ONE BAM Tank Gravity Feed valve open: • 2CV-4920-1 • 2CV-4921-1
B. BAM pumps	B. 1) Start at least ONE BAM pump. 2) Open Emergency Borate valve (2CV-4916-2). 3) Verify Boric Acid Makeup Flow Control valve (2CV-4926) closed.

BAM pumps are not available due to the loss of Electrical Bus 2A4

> **CAUTION**
>
> **Aligning Charging pump suction to RWT during RWT purification with ALL Charging pumps running may cause Charging pumps to trip due to low suction pressure.**

FLOWPATH	ACTIONS REQUIRED
C. RWT to Charging pumps	C. Open Charging Pump Suction Source From RWT valve (2CV-4950-2).

2. Close VCT Outlet valve (2CV-4873-1).

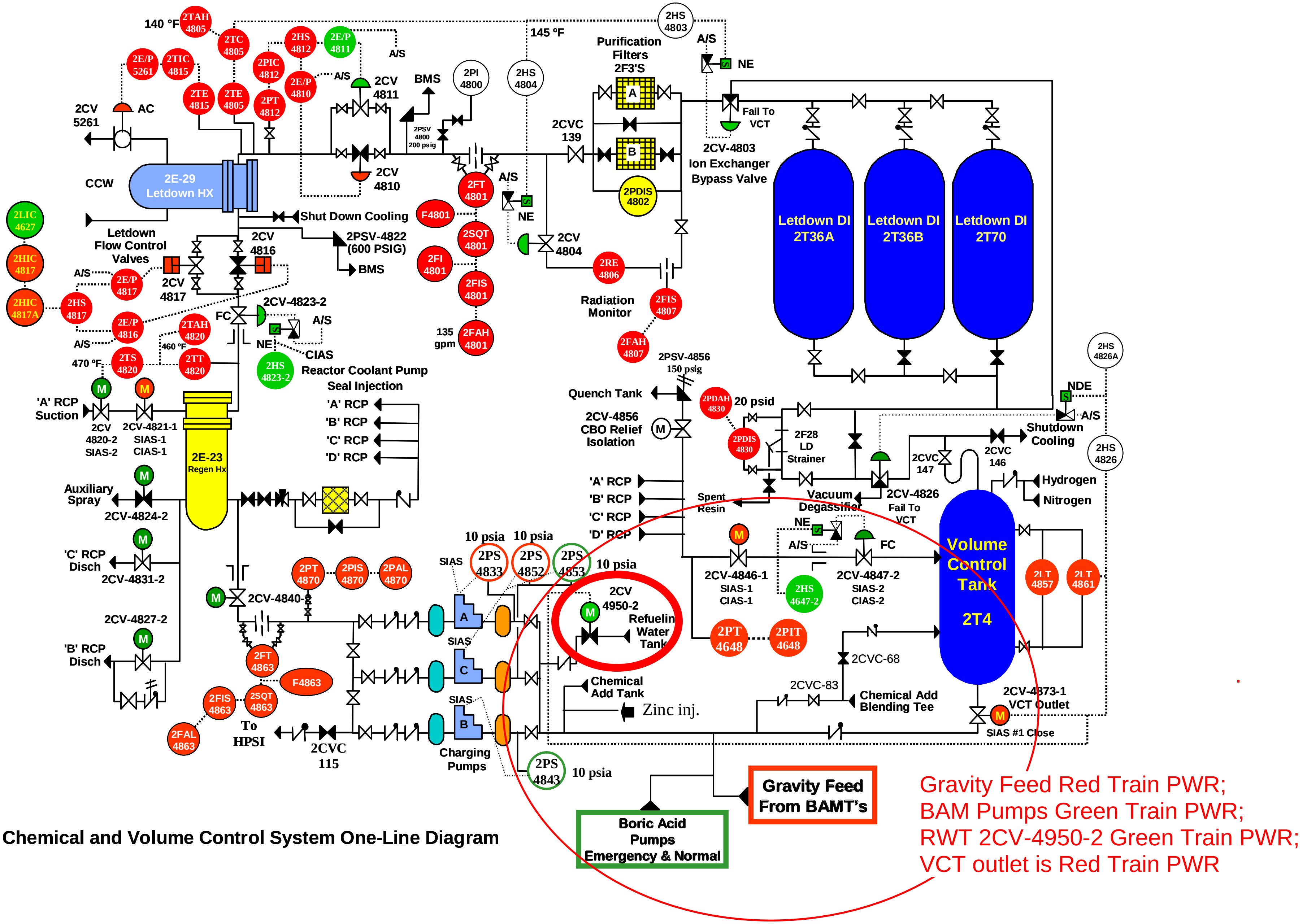
3. IF VCT Outlet valve does NOT close,
   THEN verify BAM Pumps Emergency Boration flowpath selected.

4. Verify Reactor Makeup Water Flow Control valve (2CV-4927) closed.

5. Verify at least ONE Charging pump running.

6. Verify charging header flow greater than 40 gpm by either of the following:

   - 2FIS-4863 Disch Flow (2C09)
   - Computer Point F4863 (PDS, PMS or SPDS)



Chemical and Volume Control System One-Line Diagram

Gravity Feed Red Train PWR;
BAM Pumps Green Train PWR;
RWT 2CV-4950-2 Green Train PWR;
VCT outlet is Red Train PWR

Question 24

Bank:	2354	**Rev:**	0	**Rev Date:**	7/12/2016	**2017 TEST QID #:**	24	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	Modified NRC Exam Bank 1674		
Search	000067K101	**10CFR55:**	41.8	**Safety Function**	9				
Title:	Plant Fire on Site	**System Number**	067	**K/A**	AK1.01				
Tier:	1	**Group:**	2	**RO Imp:**	2.9	**SRO Imp:**	3.9	**L. Plan:**	ASLP-FP-CLASS_B
OBJ	1								

Description: Knowledge of the operational implications of the following concepts as they apply to Plant Fire on Site: - Fire classifications, by type

Question:

Given the following:

* Unit 2 is in a refueling outage.
* A report comes in from a qualified operator that a Class 'B' fire is in progress in the basement of Containment.

What kind of hazards exist to the personnel during this fire and which action can be taken to reduce this hazard?

A. Smoke inhalation from burning trash; evacuate the containment building area.

B. Intense heat from burning liquid fuel; ensure fire water aligned to the containment.

C. Potential electrocution; de-energize the applicable power supply to containment.

D. Flying sparks from burning metal; evacuate the local area on the 335 elevation.

Answer:

B. Intense heat from burning liquid fuel; ensure fire water aligned to the containment.

Notes:

B is correct for a Class B Fire and water should be used in enough volume to absorb the heat produced from the liquid fuel.

A is incorrect as this is a Class A Fire but plausible as this is one of the 4 classes of Fire and the correct hazard and action to take for a Class A Fire.

C is incorrect as this is a Class C Fire but plausible as this is one of the 4 classes of Fire and the correct hazard and action to take for a Class C Fire.

D is incorrect as this is a Class D Fire but plausible as this is one of the 4 classes of Fire and the correct hazard and action to take for a Class D Fire.

This question matches the K&A must have the knowledge of the different classes of fire and the operational implications of each.

References:

ASLP-FP-CLASS_B Flammable Liquid Fires description and mitigating strategies.

Historical Comments:

NRC Exam Bank 1674 was used on the 2009 NRC Exam.
To be used on the 2017 NRC Exam but modified for the 2017 NRC Exam

Characteristics of Class B Materials

- Flammable and Combustible liquids may also be divided into:
  - Hydrocarbons
    - Petroleum based products.
    - These materials do not mix with water.
  - Polar solvents
    - Alcohol based or non petroleum liquids
    - These materials will mix with water.

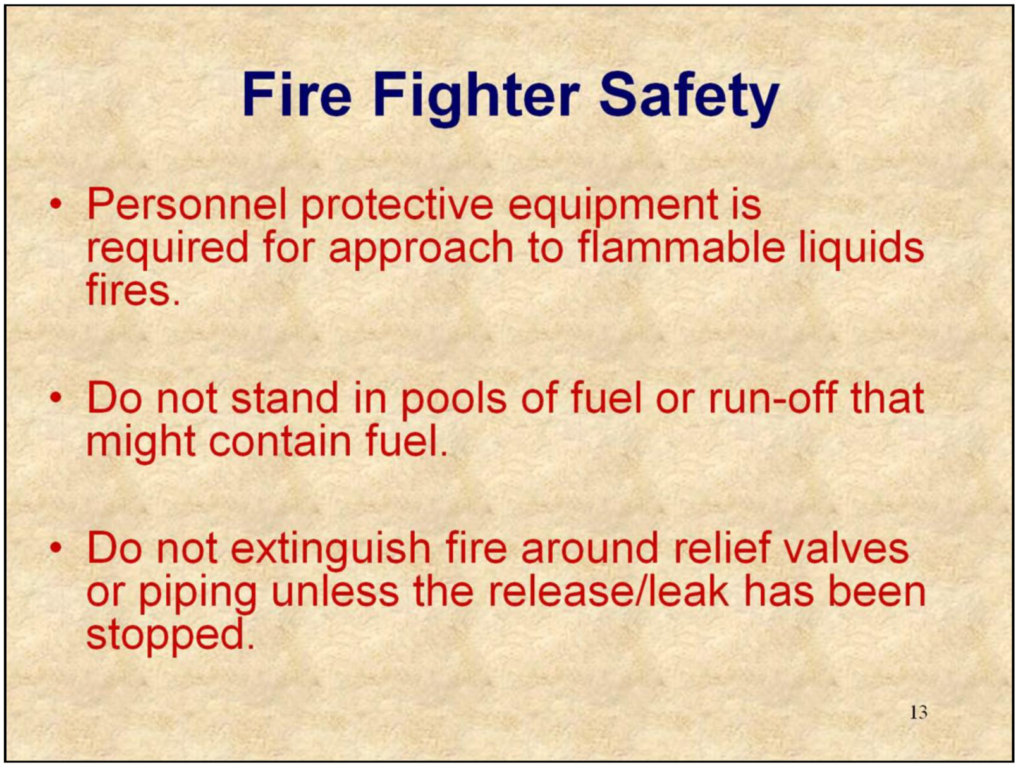
•.Flammable and combustible liquids can be further divided into **hydrocarbons** that do not mix with water and

•**Polar solvents** that do mix with water.

Characteristics of Class B Materials

- Promote rapid flame spread over the entire surface area
- Mixtures too rich to burn will pass through the flammable range as the area is ventilated

12



Give two reasons not to stand in pools of run off water that might contain fuel.

Answer:

1. If the fuel in the water ignites you are standing in fire then.
2. Protective clothing can absorb fuel in a wicking action, which can lead to skin irritation and even to the clothing catching on fire if an ignition source is present.

Fire Fighter Safety

- Do try to control all ignition sources in the area of the spill.

14

Examples of ignition sources

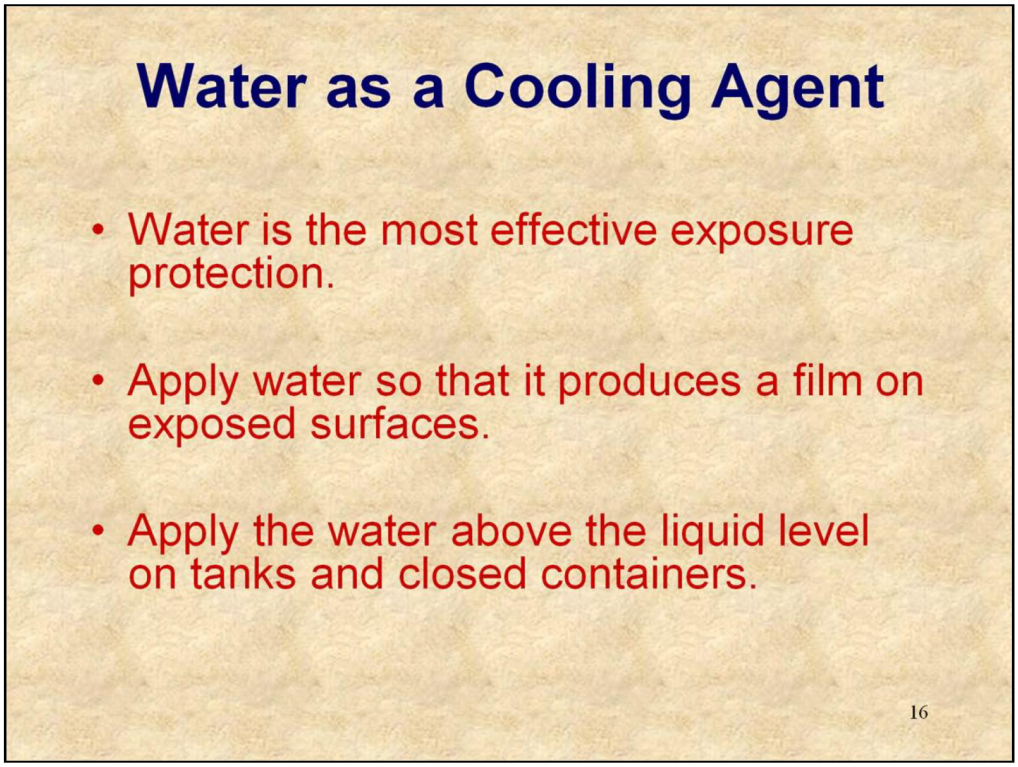
Vehicles, smoking materials, electrical fixtures, and sparks from steel tools can provide an ignition source sufficient to ignite flammable liquid vapors.

Water as a Cooling Agent

- Fires involving heavy oils can be extinguished using water in enough volume to absorb the heat produced.

- Do not allow tanks to overflow and spread spill and fire.

15



Water curtians

Monitors

2 1/2 inch lines

PARENT QUESTION 1674 WAS USED TO MODIFY 2017 NRC EXAM QID#24 (Used on the 2009 NRC Exam)

Data for 2009 NRC RO/SRO Exam

Bank:	1674	**Rev:**	1	**Rev Date:**	7/24/2009 3:48:45	**QID #:**	65	**Author:**	Coble		
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	F	**Source:**	New				
Search	086000K504	**10CFR55:**	41.10	**Safety Function**	8						
System Title:	Fire Protection System (FPS)	**System Number**	086	**K/A**	K5.04						
Tier:	2	**Group:**	2	**RO Imp:**	2.9	**SRO Imp:**	3.5	**L. Plan:**	ASLP-FP-ELEC	**OBJ**	1.2

Description: Knowledge of the operational implications of the following concepts as they apply to the Fire Protection System: - Hazards to personnel as a result of fire type and methods of protection

Question:

Given the following:

* Unit 2 is in a refueling outage
* A report comes in from the fire brigade leader that a Class 'C" fire is in progress on the 372 elevation in containment

What kind of hazards exist to the personnel during this fire and which action can be taken to reduce this hazard?

A. Smoke inhalation from burning trash; evacuate the containment building

B. Intense heat from burning oil; align fire water to the containment building

C. Potential electrocution; de-energize the applicable power supply to containment

D. Flying sparks from burning metal; evacuate the local area on the 372 elevation

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐
2009	☑	☑

Audit Exam History

2009	☐

Answer:

C. Potential electrocution; de-energize the applicable power supply to containment

Notes:

There are 2 480 Volt MCC electrical buses on the 372 elevation in containment (2B71 and 2B81). A class C fire would be in one of those buses as they are the only major electrical components on that elevation. A trash fire is a Class A fire. Oil is a Class B fire and burning metal is a Class D fire.

References:

STM 2-32-3, 480 Volt Distribution System, Rev. 13, Table 2 on page 42
Lesson Plan ASLP-FP-ELEC, Fighting Electrical Fires.Rev.2, Objectives 1.2: Discuss the hazards associated with fires involving energized electrical equipment and cables and 1.4: Describe the actions necessary to control a Class "C" Fire.

Historical Comments:

Question 25

Bank:	2355	**Rev:**	2	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	25	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	H	**Source:**	NRC Exam Bank 1631		
Search	000036A103	**10CFR55:**	41.7	**Safety Function**	8				
Title:	Fuel Handling Incidents	**System Number**	036	**K/A**	AA1.03				
Tier:	1	**Group:**	2	**RO Imp:**	3.5	**SRO Imp:**	3.9	**L. Plan:**	A2LP-RO-FH
OBJ	4								

Description: Ability to operate and/or monitor the following as they apply to the Fuel Handling Incidents: - Reactor building containment evacuation alarm enable switch

Question:

The following plant conditions exist:

* Mode 6 with core re-load in progress
* The Control Room receives a report that as a spent fuel assembly was being inserted into the core, the refueling machine started moving on its own.
* Bubbles are rising around the fuel assemblies in the core.
* Low Range Rad Monitors on the 404 elevation are indicating a large rise in radiation.
* OP-2502.001, Refueling Shuffle, Att. M, Refueling Accident is in progress.

Per the procedure, to inform the required individuals to commence an evacuation, the Evacuation Warning Handswitch, 2HS-EVAC on panel ________ should be taken to ______________.

A. 2C22; "CTMT" ONLY

B. 2C20; "CTMT" ONLY

C. 2C22; "PLANT" and "CTMT"

D. 2C20; "PLANT" and "CTMT"

Answer:

A. 2C22; "CTMT" ONLY

Notes:

A is correct. The Plant/Containment Evacuation alarm switch is located on panel 2C22 (which is next to the Generator Lockout panel 2C20) and has four positions: OFF, CTMT, CTMT AUX, and PLANT which will sound the alarms in the Containment, and the General Plant respectively.

B is incorrect because the wrong panel is listed but plausible because the 2C20 is located right next to 2C22. and the correct switch position is listed.

C is incorrect because the switch is placed in PLANT but plausible as this is the correct panel and the switch could be placed in PLANT if the accident was bad enough to evacuate the plant site but by procedure a plant page would be used for warning the site of the event.

D is incorrect because the switch is placed in PLANT and the listed panel is wrong but plausible as this panel is next top 2C22 and the switch could be placed in PLANT if the accident was bad enough to evacuate the plant site but by procedure a plant page would be used for warning the site of the event.

This question matches the K&A as it requires the candidate to have the ability to operate the Containment Evacuation

Alarm switch.

References:

NOP 2502.001, Refueling Shuffle, Rev. 52 Attachment M, Refueling Accident, Step 2.5.6 (Verified reference updated 11/10/16); NOP 2105.016, Radiation Monitoring and Evacuation System, Rev. 31, Attachment D Step 2.0 (Verified reference updated 11/10/16).

Historical Comments:

NRC Exam Bank 1631 was used on the 2009 NRC Exam
To be used on the 2017 NRC Exam; However, 2C14 was changed to 2C20 to have more plausible distractors and altered the order of the answer from the last time the QID was given.

REV. 1 based on NRC Chief Examiner Feedback BNC. Deleted references to accident being in Containment in the conditions and changed CTMT AUX to PLANT in distractors C and D.

REV. 2 based on NRC Chief Examiner Feedback BNC. Added a bullet to say "OP-2502.001, Refueling Shuffle, Att. M, Refueling Accident is in progress". Reworded the first part of the stem to say "Per the procedure, to inform the required individuals to commence an evacuation". Added the word "only" to A and B. Added the words "and CTMT" to C and D.

ATTACHMENT M

REFUELING ACCIDENT

1.0 SYMPTOMS/ENTRY CONDITIONS

- Bubbles emerging from a submerged Spent Fuel assembly that was dropped or damaged.
- Visual inspection reveals abnormally bent, twisted, or warped Spent Fuel assembly.
- High or abnormal air activity indications on installed or portable air radiation monitor.
- SRO in charge of fuel handling believes a Spent Fuel assembly has been damaged.

2.0 IMMEDIATE ACTIONS

2.1 Persons discovering emergency condition will notify Control Room by most expedient means available.

* 2.2 IF specified fuel handling termination criteria discussed in IPTE/Pre-Job brief is exceeded,
THEN load should be left and job site exited.

{4.3.2} 2.3 **IF fuel handling in progress,**
AND time/dose permits,
THEN the SRO in charge of fuel handling may evaluate placing the fuel assembly in the safest location available prior to bridge evacuation.

2.4 IF accident has occurred at Spent Fuel Pool,
THEN perform the following:

2.4.1 Announce location and nature of emergency

2.4.2 Perform localized evacuation IAW Plant Evacuation (1903.030).

2.5 IF accident has occurred inside Containment,
THEN perform the following:

2.5.1 Verify CNTMT purge secured IAW Containment Atmosphere Control (2104.033).

2.5.2 Obtain SM concurrence for CNTMT evacuation and closure.

2.5.3 Perform “Containment Closure Checklist”, Attachment F of SDC Control (1015.008).

2.5.4 Notify Radiation Protection to evacuate CNTMT of all personnel NOT involved with CNTMT closure.

2.5.5 Make the following announcement using Plant Page system:

"Attention all personnel. Attention all personnel.
A Unit 2 CNTMT evacuation is required.
All personnel except those performing CNTMT closure evacuate CNTMT."

2.5.6 Actuate CNTMT Evacuation alarm on 2C22.

2.5.7 Repeat Steps 2.5.5 and 2.5.6 one time.

3.0 FOLLOW-UP ACTIONS

3.1 Notify Emergency Coordinator.

3.2 Refer to Plant Evacuation (1903.030).

3.3 IF accident occurred inside Containment,
THEN perform the following:

* 3.3.1 Monitor the following for activity.

- Containment Purge Rad Monitor (2RITS-8233) on 2C25.
- U2 CNTMT Purge Exhaust (SPING 5)

3.3.2 Initiate controlled purging of Containment atmosphere IAW Containment Atmosphere Control (2104.033).

* 3.3.3 IF high activity detected,
THEN verify purge immediately secured IAW Containment Atmosphere Control (2104.033).

3.3.4 Verify activity being recorded by at least ONE of the following:

- U2 CNTMT Purge Exhaust (SPING 5)
- Process Gas Radiation Monitor (2RR-0645) on 2C25. (permanent record)

ATTACHMENT D

EVACUATION ALARM AND MEDICAL TEAM PAGER TEST

This test demonstrates response of evacuation alarm to operator actuation from Unit 2 Control Room and tests various Emergency Team pagers. It does not satisfy any Tech Spec or other regulatory requirements. This test is performed to satisfy SAR section 9.5.2.3.

1.0 Perform the following to test Plant Evacuation Alarm:

1.1 Announce on Plant Paging System:
"Attention all personnel. Attention all personnel. The following is a test of the Plant Evacuation Alarm System. Please disregard. The following is a test of the Plant Evacuation Alarm System. Please disregard."

1.2 Cycle Evacuation Warning Circuit handswitch (2HS-EVAC) to PLANT and then back to OFF.

1.3 Check alarm sounds.

2.0 IF EITHER containment building is open for general access, THEN perform the following to test Containment Building Evacuation Alarm system:

2.1 Verify personnel in containment to check alarm sounds.

2.2 Announce on the Plant Paging System:
"Attention personnel in Unit 1 (2) Containment Building. Attention personnel in Unit 1 (2) Containment Building. The following is a test of the Containment Building Evacuation Alarm system. Please disregard. The following is a test of the Containment Building Evacuation Alarm system. Please disregard."

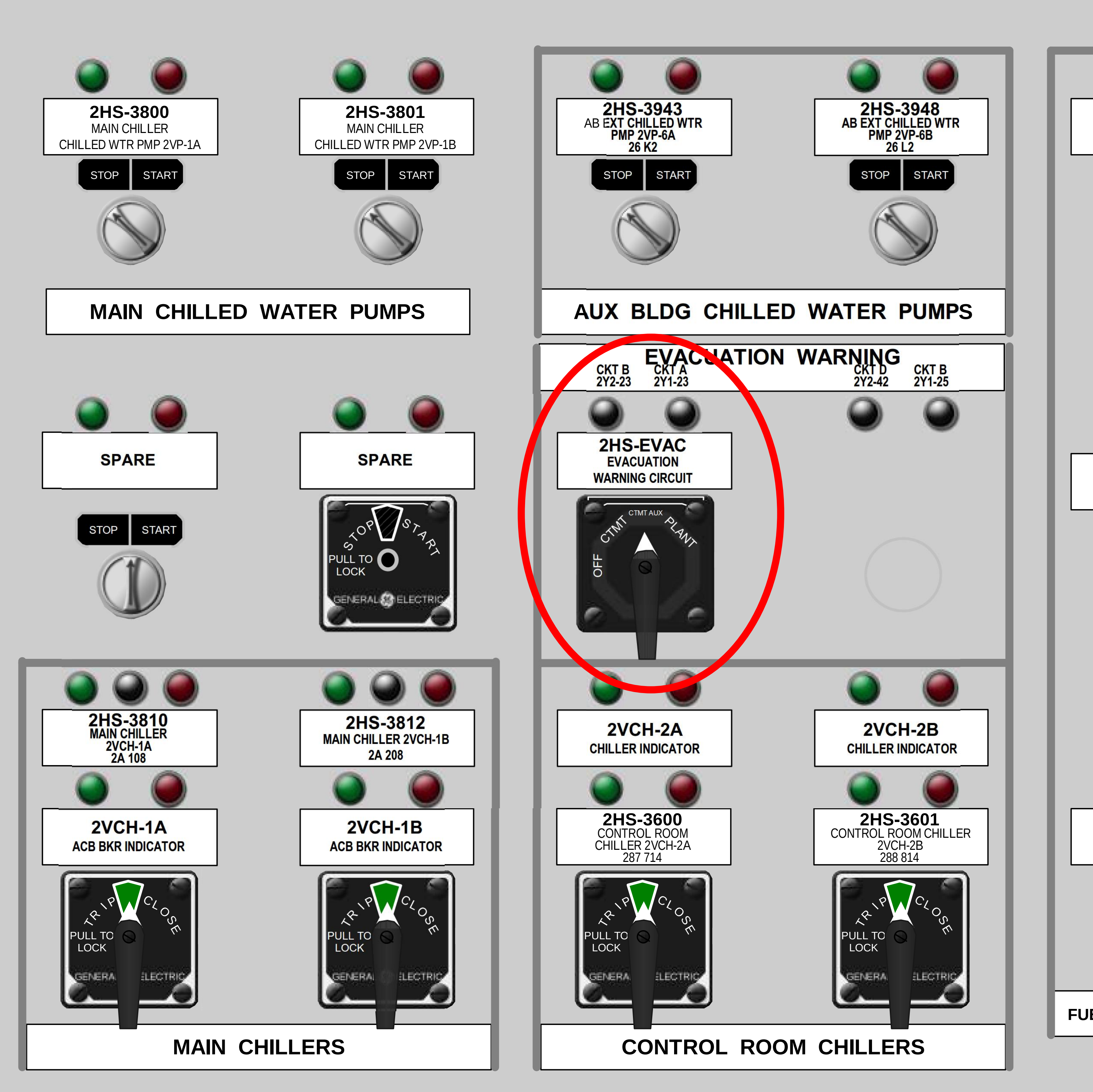
2.3 Place appropriate HS to actuated position for ~5 seconds.

- HS-EVAC (C25) to REAC BLD
- 2HS-EVAC (2C22) to CTMT

2.4 Place HS-EVAC (2HS-EVAC) to OFF.

2.5 Check alarm sounds.

3.0 Announce on Plant Paging System:
"Attention all personnel. Attention all personnel. The test of the Evacuation Alarm system is complete. Regard all further alarms. The test of the Evacuation Alarm system is complete. Regard all further alarms."
"Attention Medical Emergency Team and ANO Rescue Team, the following is a pager test.
Attention Medical Emergency Team, and ANO Rescue Team, the following is a pager test."


2HS-3800
MAIN CHILLER
CHILLED WTR PMP 2VP-1A
STOP START
2HS-3801
MAIN CHILLER
CHILLED WTR PMP 2VP-1B
STOP START
2HS-3943
AB EXT CHILLED WTR
PMP 2VP-6A
26 K2
STOP START
2HS-3948
AB EXT CHILLED WTR
PMP 2VP-6B
26 L2
STOP START
MAIN CHILLED WATER PUMPS
AUX BLDG CHILLED WATER PUMPS
EVACUATION WARNING
CKT B 2Y2-23
CKT A 2Y1-23
CKT D 2Y2-42
CKT B 2Y1-25
SPARE
SPARE
STOP START
STOP START
PULL TO LOCK
GENERAL ELECTRIC
2HS-EVAC
EVACUATION
WARNING CIRCUIT
OFF CTMT CTMT AUX PLANT
2HS-3810
MAIN CHILLER
2VCH-1A
2A 108
2HS-3812
MAIN CHILLER 2VCH-1B
2A 208
2VCH-2A
CHILLER INDICATOR
2VCH-2B
CHILLER INDICATOR
2VCH-1A
ACB BKR INDICATOR
2VCH-1B
ACB BKR INDICATOR
2HS-3600
CONTROL ROOM
CHILLER 2VCH-2A
287 714
2HS-3601
CONTROL ROOM CHILLER
2VCH-2B
288 814
TRIP CLOSE
PULL TO LOCK
MAIN CHILLERS
CONTROL ROOM CHILLERS
FUE

ORIGINAL QUESTION 1631 WAS USED TO DEVELOP 2017 NRC EXAM QID#25 (Used on the 2009 NRC Exam)

Questions For All QID In Exam Bank

Bank:	1631	**Rev:**	1	**Rev Date:**	7/24/2009 12:40:3	**QID #:**	22	**Author:**	Coble
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	H	**Source:**	New		
Search	000036A103	**10CFR55:**	41.9	**Safety Function**	8				
System Title:	Fuel Handling Incidents	**System Number**	036	**K/A**	AA1.03				
Tier:	1	**Group:**	2	**RO Imp:**	3.5	**SRO Imp:**	3.9	**L. Plan:**	A2LP-RO-FH
OBJ	4								

Description: Ability to operate and/or monitor the following as they apply to the Fuel Handling Incidents: - Reactor building containment evacuation alarm enable switch

Question:

The following plant conditions exist:

* Mode 6 with core re-load in progress
* The Control Room receives a report that as a spent fuel assembly was being inserted into the core, the refueling machine started moving on its own
* A lot of bubbles are rising around the fuel assemblies in the core
* Containment Low Range Rad Monitors on the 404 elevation are indicating a large rise in radiation

To warn the individuals in Containment of this accident, the Evacuation Warning Handswitch, 2HS-EVAC on panel ________ should be taken to ______________?

A. 2C22; CNTMT AUX

B. 2C14; CNTMT AUX

C. 2C22; CNTMT

D. 2C14; CNTMT

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☒

Answer:

C. 2C22; CNTMT

Notes:

The Plant/Containment Evacuation alarm switch is located on panel 2C22 (which is next to the radiation monitoring panel 2C25) and has four positions: OFF, CNTMT, CNTMT AUX, and PLANT which will sound the alarms in the Containment, Containment Auxiliary Building, and the General Plant respectively. Distracters A ar B are incorrect because they have the switch placed in CNTMT AUX. Distracters B and D are incorrect because the switch is not located on 2C14.

References:

NOP 2502.001, Refueling Shuffle, Rev. 37, Attachment M, Refueling Accident, Step 2.5.6
NOP 2105.016, Radiation Monitoring and Evacuation System, Rev. 26, Attachment D Step 2.0
Lesson Plan A2LP-RO-FH, Rev. 1, Objective 4: Given a fuel handling evolution or condition, determine the correct response.

Historical Comments:

Question 26

Bank:	2356	**Rev:**	3	**Rev Date:**	1/12/2017	**2017 TEST QID #:**	26	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NRC Exam Bank 1905		
Search	000032K101	**10CFR55:**	41.10	**Safety Function**	7				
Title:	Loss of Source Range Nuclear Instrumentation	**System Number**	032	**K/A**	AK1.01				
Tier:	1	**Group:**	2	**RO Imp:**	2.5	**SRO Imp:**	3.1	**L. Plan:**	A2LP-RO-TS	**OBJ**	4B

Description: Knowledge of the operational implications of the following concepts as they apply to Loss of Source Range Nuclear Instrumentation: - Effects of voltage changes on performance

Question:

Given the following:

* The plant is in Mode 6 with core reload in progress.
* I&C is performing the required 7 day Channel Functional Test on SU Channel 2 NI
* I&C reports that the power supply to SU Channel 2 NI was below the acceptable allowed voltage value during the surveillance.

Based on this report the SU Channel #2 NI would be _________________ and the count rate in the Control Room would read __________ counts than actual.

A. more sensitive to gammas; higher

B. more sensitive to neutrons; higher

C. less sensitive to gammas; lower

D. less sensitive to neutrons; lower

Answer:

D. less sensitive to neutrons; lower

Notes:

D is correct. Refer to the referenced drawing of ANO Unit 2 SU Channels and Section 4.2 of STM 2-67, Excore NI, for the theory of a fission chamber. Startup source range channels on ANO Unit 2 are fission chambers with pulse height discriminators with a high voltage input of ~850 volts. If the HV input lowers the amount of potential across the detector chamber will be lower thus less sensitive to the incoming neutrons to create a pulse or a smaller pulse due to the gas amplification effect of the original fission event. So then the pulse from neutron fission sensed at the discriminator may not be high enough to overcome the pulse height discriminator and provide therefore the output of the discriminator will send fewer counts to the control room indication.

A is incorrect as the energy of the gammas is much smaller than the energy from a neutron fission event and the pulse height discriminator will eliminate any pulse from any gamma induced event and the detector output will be lower but plausible if the applicant does not know that the SU NI are fission Chambers and understand how they work.

B is incorrect because this is the response on a higher voltage supply but plausible if the applicant does not remember or understand how the SU NI channels work.

C is incorrect as the energy of the gammas is much smaller than the energy from a neutron fissionevent and the pulse height discriminator will eliminate any pulse from any gamma induced event but plausible as the output of the detector will be lower.

This question matches the K&A because the candidate must be able to recognize the low voltage input to the SU Channel

#2 being below acceptable limits during a surveillance will on the detector performance.

References:

STM-2-67-1, Excore Nuclear Instruments, Rev. 11 Sections 4.2 and 4.3.1.
OP-2203012J ANNUNCIATOR 2K10 CORRECTIVE ACTION Window K-5 SU Channel 2 Trouble Rev. 43; (Verified reference updated 11/15/16).
STM_2-67-1_11-1 Excore Nis Section 2.3; (Verified reference updated 11/15/16).
STM_2-67-1_11-1 Excore Nis SU Channel Drawing (Verified reference updated 11/15/16).

Historical Comments:

NRC Exam Bank QID #1905 was used on the 2012 NRC Exam
To be used on the 2017 NRC Exam but altered stem to have a failed TS surveillance , alter some of the wording in the distractors to be more plausible and altered the order of the correct answer.

REV. 1 based on NRC Chief Examiner Feedback BNC. Rewrote the question based on feedback that the original did not meet the K/A. The new version better meets the effect of voltage changes on the performance of the detector.

REV. 2 based on NRC Chief Examiner Feedback BNC. Added the words "on SU Channel 2 NI" to the end of the 2nd bullet instead of "required by TS Surveillance 4.9.1".

Rev. 3 based on post submittal validation comments: Removed the words "High Voltage" from the third bullet (1st part) and added "voltage" prior to value in the third bullet (2nd part)

4.2 Fission Chambers

Fission chambers are the sensitive elements in the excore nuclear instrumentation system. They operate as ion chambers, in voltage saturation, which avoids non-linearity. There is no gas amplification, but a fission chamber can be used in the pulse counting mode because of the large amplitude of each current pulse from a neutron interaction (fission).

Each detector assembly houses two (startup channels) or three (safety channels) fission chambers. Each fission chamber consists of two aluminum electrodes coated with ~8 grams of enriched uranium oxide, and a fill gas of 5% nitrogen and 95% argon. For greater sensitivity, the cylindrical electrodes overlap to provide a greater chance that a neutron will be absorbed by the enriched uranium coating (see figure 3 on page 22).

A thermal neutron, if absorbed by the uranium, is likely to fission, resulting in the release of two or more fission fragments. These fragments will result in an energy release of ~ 165 Mev. The high voltage applied to the electrodes will accelerate the ions, causing further ionization of the fill gas, and a resulting “spike” of current flow, or if a large number of neutrons is causing fission, then there will be a flow of current through the detector.

An alpha ionization from the natural decay of uranium will result in a ~4.7 Mev pulse, and a gamma will result in significantly less energy.

The fission of the uranium results in fission fragments that travel through the chamber (detector) and ionize the gas in their path. Because of the electric field at the electrodes the ions migrate to these electrodes, and cause a flow of current. If the amount of fission is low, the individual pulses can be counted. If there is a large amount of fission in the detector, then the current flow can be measured.

In order for the measured countrate to accurately indicate neutron flux, the current pulses caused by other reactions must not be counted.

The other competing reactions are from gamma radiation from external sources, and alpha particles emitted by the natural decay of the Uranium coating on the fission chamber electrode surfaces.

4.3 Detector Modes

The fission chambers used in the startup channels use three modes in monitoring the core;

- Pulse counting mode in the source range and log power range.
- Campbelling mode in log power range.
- Current mode in the linear power range.

4.3.1 Pulse Counting

Pulse counting is useful as long as there is some time interval between two consecutive pulses. At high count rates, the pulses start to overlap and the measured countrate will be in error due to pulse pile-up. Pulse amplitude discrimination is used to eliminate the

unwanted pulses because the energy from the fission fragments are ~165 Mev, while the alpha decay of Uranium is typically 4.7 Mev, and the gamma photons deposit even less energy per interaction.

Most fissions from the detection of neutrons deposit only half the available energy into the fill gas (the other fission fragment crashes into the detector tube wall), however, the pulse amplitude is still much greater than those resulting from either gamma photons or alpha particles.

The start up source range uses both detectors, therefore increasing sensitivity (see figure 10 on page 29). The high voltage is isolated from the input to the preamps by the capacitors. The capacitors allow only the pulses or changes in current to be input to the preamps.

The outputs of preamps A3 and A4 combine to feed the startup (source range) pulse height discriminator. The output of this pulse height discriminator is one of the three signals sent to the signal processor.

The output signal is conditioned to provide a “clean” pulse for each pulse above the discriminator setting. This output pulse is made of a consistent voltage and duration to provide a quality pulse to the signal processor.

4.3.2 Current Mode

Refer to figure 13 on page 32. If there is a tremendous amount of neutrons causing fissions in the detector (typically 2.5 X 10^9 nv), instead of pulses from the detector, there will be a detector current. Also the current from gamma is small and proportional to the neutron flux when the reactor is at power. Therefore, all that is needed is an amplifier that produces an output that is proportionate to the input.

The current in the guarded fission chamber flows through a resistor to common. As the voltage drop across this resistor rises, the input to the linear preamp A5 also rises. The output of A5 goes to the signal processor for linear reactor power. (This mode is not used in the Startup channels. Startup Channel 1 detectors are both unguarded where as Startup Channel 2 has one guarded and one unguarded).

The capacitor bypassing the input resistor shunts any AC signal components to ground, providing a cleaner signal to the preamp. This is the second of the three neutron flux signals sent to the signal processor.

ANNUNCIATOR 2Kl0

K-5

STARTUP CHANNEL 2 TROUBLE

1.0 CAUSES

1.1 Startup Channel 2 Trouble as indicated by NON-OPERATE light on 2JIT-9003-2 (2C336-2) due to any of the following:

- 15-volt power supply voltage low
- Magnitude of High Voltage Power Supply voltage low
- Magnitude of current from guarded fission chamber low, indicating cable or detector integrity lost
- Test switch (S3) on the Test Generator Board (A9) in the signal conditioner is depressed

2.0 ACTION REQUIRED

2.1 IF in Mode 6
AND EITHER of the following occur:

- Either Startup Channel inoperable
- Both Audible Count Rate indications inoperable,

THEN perform the following:

- Suspend all core alterations.
- Suspend all positive reactivity additions.
- Refer to Tech Spec 3.9.2.

2.2 Notify I&C to troubleshoot as required.

3.0 TO CLEAR ALARM

3.1 Place test switches to OPERATE or OFF.

3.2 Replace printed circuit board.

3.3 Adjust power supply voltage.

4.0 REFERENCES

4.1 E-2456-4

4.2 DCP-88-2111

2.2.3 Safety Channel Instrumentation

The Excore Safety Channels provide the following instrumentation for monitoring.

Indication	Range	Type	Location
Linear Power	1-200%	meter	2C23
Linear Power	1-200%	chart	2C03
Log Power	10^{-8} - 200%	meter	2C23
Log Power	10^{-8} - 200%	bar graph	2C03
Log Power	10^{-8} - 200%	bar graph	2C80 (B only)
Log Power	10^{-8} - 200%	visual	PMS
Start-up Rate	-1 to +7 dpm	meter	2C23
Start-up Rate	-1 to +7 dpm	bar graph	2C03
Raw Power	1-200%	point display	CPCs

2.3 Startup Channels

This section will provide a detailed description of the Excore Startup Channels

2.3.1 Detectors

The startup channel detectors construction is the same as the safety channel detectors. Refer to figures 2 and 3 on pages 21 and 22.

The two startup channels have two detectors per channel located approximately 180^o apart, outside the reactor vessel. The startup channel provides indication from the source range of 0.1 cps up to 200% power with the fission chamber detectors, and does not require deenergizing of high voltage as power is raised above the source range.

Both detectors in each channel are used for the source range only, from 0.1 to 10^6 cps. The log range of $10^{-8\%}$ power to full power uses only one fission chamber detector. The source range uses both detectors for more sensitivity.

The output of each detector is connected directly to the amplifiers and signal processors located immediately outside containment. There are no electronic preamps or other circuits located inside containment. The only component other than the detectors, inside containment, is the connection box, which connects the detector cables to the transmission cable. The connection box is environmentally sealed to withstand a harsh containment environment.

Since all amplification is located outside containment and will not be exposed to high temperatures or high radiation, and the detectors are designed to operate in an extremely high gamma environment, Startup Channel 1 meets the requirements of Reg. Guide 1.97.

However, the cabling associated with Startup Channel 2 has exceeded its qualified dose limit of 3E9 rad, it is no longer

considered reliable in a harsh environment. Therefore, Channel 2 does not meet the Reg. Guide 1.97 requirements.

The startup channels are also powered from class 1E power (2RS1 and 2RS2).

2.3.2 Startup Channel Amplifier Assembly and Signal Processor

The startup channel amplifiers contain the detector high voltage and signal conditioning circuits.

The startup channel signals from the detectors are first input to a preamplifier. Channel #1 preamp is located in the upper south electrical penetration room in the vicinity of the containment hatch. The amplified signal then goes to the control room to the Startup Channel unit located on 2C336-1, in the back of the control room. Startup Channel #2 is located in the upper north electrical penetration room. It has additional equipment to provide SPDS input that is isolated from the control room. Channel #2 has an amplifier and signal processor, but then has two outputs with isolation amplifiers, such that, shorting or opening one circuit will not affect the other circuit. One signal goes to the control room panel 2C336-2 (same as channel #1) and the other signal goes to a signal processor/display in the UNEPR. This signal processor only provides source range indication to SPDS, 10^{-1} to 10^{5} cps. The local processor has only the source range display, and at any power above 10^{5} cps, the display will flash, indicating that it is over-ranged. This is the most reliable indication during alternate shutdown due to a control room fire.

The startup channels provide indication for the following;

- Remote Shutdown Panel (Channel #1 only)
- SPDS Count Rate (Channel #2 only) (NISU2N)
- SPDS Log Power (NISU1, NISU2)
- Chart recorder on 2C03, source range 1 to 10^{6} cps

The Excore Instrumentation for both the safety channels and the start-up channels is almost identical in operation. Both systems use fission chambers for detectors, and both use counting and campbelling for the log power and detector current power ranges. The most significant difference is that the start-up channel provides indication below 1 X 10^{-8}% power, by using a source range counting circuit. The theory of operation for both systems is very similar.

Startup channels preamplifiers, high voltage power supplies, and discriminators are located in the containment penetration rooms. This is as close as possible to the detectors, without being exposed to post-accident conditions, such as a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB).

The startup channels can provide an accurate indication of neutron flux over the full range with a gamma background of 10^{4} R/hr, and as low as 10^{-7}% power with a gamma background of 10^{6} R/hr. The startup channels are designed for a design basis seismic event following 40 years of operation under normal conditions.

The red train, channel #1, preamp is located in the upper south electrical penetration room (USEPR), on the wall opposite the containment personnel hatch. The amplified and conditioned signals are then sent to the control room signal processor, located at 2C336-

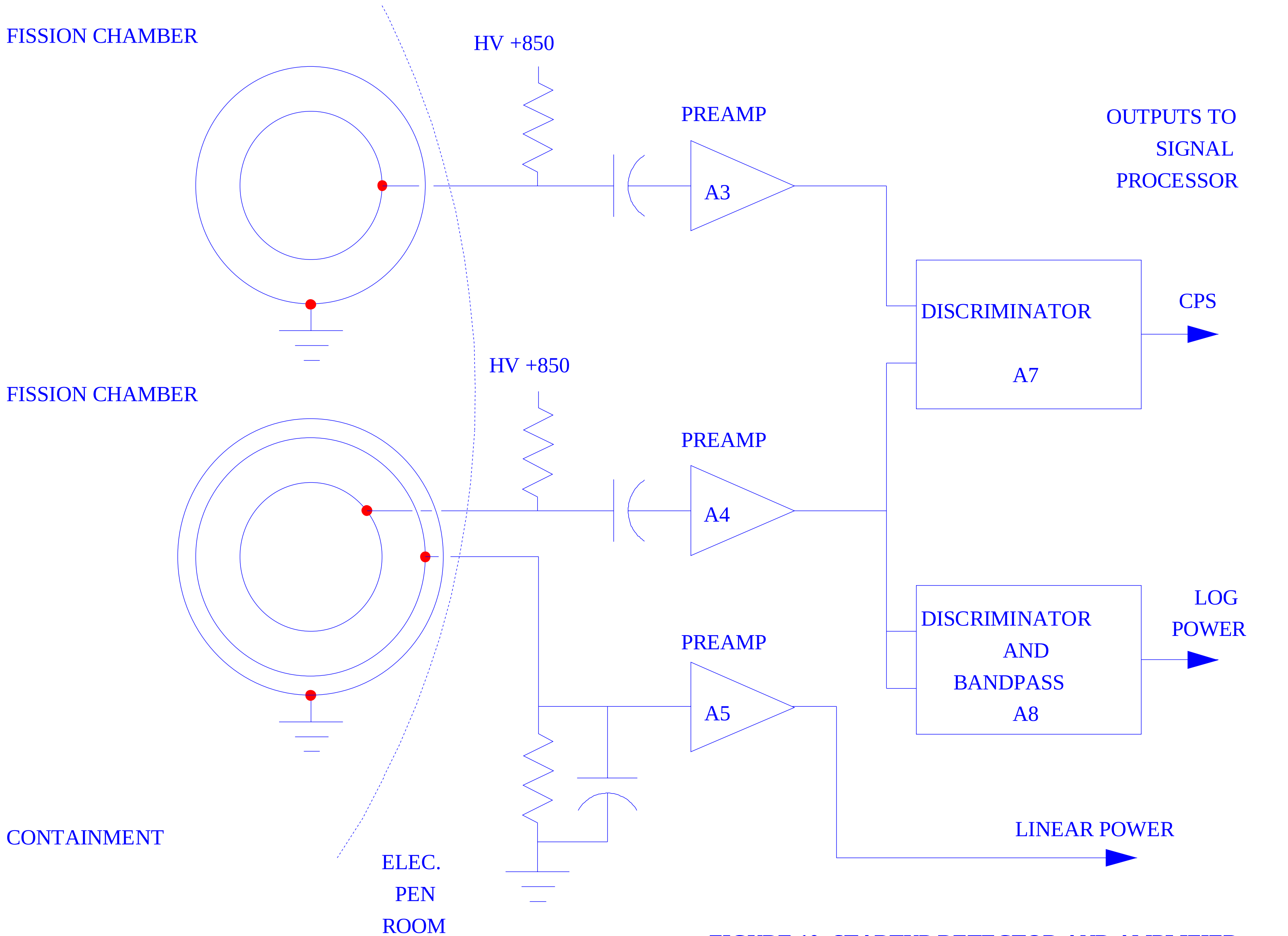


FIGURE 10: STARTUP DETECTOR AND AMPLIFIER

ORIGINAL BANK QUESTION 1905 WAS USED TO DEVELOP 2017 NRC EXAM QID#26 (Used on 2012 NRC Exam)

Questions For All QID In Exam Bank

New Question

Bank: 1905 **Rev:** 0 **Rev Date:** 5/23/2012 9:16:56 **QID #:** 84 **Author:** Foster

Lic Level: S **Difficulty:** 2 **Taxonomy:** F **Source:** Modified NRC bank 1709

Search 0000322411 **10CFR55:** 41.10 / 43.5 / 45.13 **Safety Function** 7

System Title: Loss of Source Range Nuclear Instrumentation **System Number** 032 **K/A** 2.4.11

Tier: 1 **Group:** 2 **RO Imp:** 4.0 **SRO Imp:** 4.2 **L. Plan:** A2LP-RO-FH **OBJ** 5

Description: Emergency Procedures/Plan - Knowledge of abnormal condition procedures.

Question:

Consider the following:

- Unit 2 is in mode 6
- Core Reload is in progress
- Refueling Boron concentration is 2825 ppm
- #2 source range neutron flux monitor has failed low

What action(s) is(are) required (if any) due to the above conditions?

A. No action required, T.S. 3.9.2 requirements are satisfied

B. Immediately suspend all operations involving CORE ALTERATIONS

C. Verify Audible count rate on the refueling bride and continue with fuel movement

D. Restore the source range monitor to OPERABLE status within 1 hour, or suspend CORE ALTERATIONS

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☒

Answer:

B. Correct; T.S. 3.9.2 action a. has you immediately suspend all operations involving CORE ALTERATIONS

Notes:

With the Unit in refuel mode of operations the source range instrumentation requirements and actions if a failure were to occur, will be directed from the governing procedure and T.S. therefore the T.S. will be used to cope with a source range NI failure and actions will be directed accordingly.
A, C, and D are Incorrect based on T.S. 3.9.3 requirements. In Mode 6, a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room or core onload cannot continue.

References:

T.S 3.9.2, Refueling Operations Instrumentation, and the Basis 3/4.9.2, Rev. 48
A2LP-RO-FH, Rev. 1, Obj. 5, given a set of plant conditions associated with fuel handling, evaluate Technical Specification and Technical Requirement entry conditions and describe any LCO actions that may be required.
A2LP-RO-NI Rev. 11, Obj. 8: Using the Unit 2 Tech Specs explain the Limiting Condition for operations and Surveillance requirements associated with the Excore NI system.

Historical Comments:

Modified NRC bank question 1709, this question has not been previously used. It was devloped for the 2009 NRC RO exam but was deleted because it is too close to another question.

Question 27

Bank:	2357	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	27	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	Modified NRC Exam Bank 2209		
Search	0000762446	**10CFR55:**	41.5	**Safety Function**	9				
Title:	High Reactor Coolant Activity	**System Number**	076	**K/A**	2.4.46				
Tier:	1	**Group:**	2	**RO Imp:**	4.2	**SRO Imp:**	4.2	**L. Plan:**	A2LP-RO-ARCSA
OBJ	2								

Description: Emergency Procedures/Plan - Ability to verify that the alarms are consistent with the plant conditions.

Question:

Given the following:

* The plant is at 50% during power ascension coming out of a refueling outage.
* Annunciator "LETDOWN RADIATION HI/LO" (2K12-A1) comes in Alarm.
* Letdown Gross Activity Monitor (2RITS-4806-A) reads 5.5E+5 CPM and rising rapidly.
* Letdown I-131 Activity Monitor (2RITS-4806-B) reads normal at 350 cpm on the 1E3 scale.

Which of the following events occurred for the given plant conditions?

A. Failed fuel

B. RCS crud burst

C. RCS zinc addition

D. Letdown Demineralizer exhausted

Answer:

B. RCS crud burst

Notes:

B is correct as only RCS Gross activity is rising and not I-131. Refer to STM 2-04 Rev31 Section 2.1.13. With normal readings on the I-131 Activity, a fuel failure can be ruled out.

A is incorrect because I-131 activity is not rising but plausible because fuel failure could be expected to show up first just after a refueling outage during power ascension.

C is incorrect as zinc injection is a normal chemical we add at power to coat the passive layer of corrosion on the inside of the RCS piping and does not cause a crud burst but plausible as we do chemically shock the RCS to cause a crud burst going into a outage to allow for cleanup of the RCS prior to maintenance.

D is incorrect as old Letdown demineralizer resin is replaced with new resin during refueling outages but plausible as an exhausted demineralizer that also acts as a filter could release activity into the RCS but since the resin is new coming out of an outage, this would not occur until late in core life. Also exhausted resin would not be as high of an gross activity reading as given in the above conditions nor would the activity be rising rapidly.

This question matches the K&A because it requires the candidate to access a plant alarm and based on the readings of the CVCS PRM and plant conditions, determine that a Crud Burst has occurred in the RCS.

References:

STM_2-04__31-1 CVCS Section 2.1.13 CVCS PRM 2RE-4806; (Verified reference updated 11/15/16)
STM_2-62_23-1 Radiation Monitoring Sys Section 2.2.1; (Verified reference updated 11/15/16)
AOP 2203.020 High RCS Activity REV 12 Entry Conditions; (Verified reference updated 11/15/16)

AOP 2203.020 High RCS Activity REV 12 Step 7 and 9; (Verified reference updated 11/15/16)

Historical Comments:

NRC Exam Bank 2209 was used on the 2015 NRC Exam
To be used on the 2017 NRC Exam but modified for the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Added the word "rapidly" to the end of bullet #3 and changed distractor C to RCS zinc injection

2CV-4804 can be positioned from the control room panel 2C09. Normally the handswitch for 2CV-4804 is in "automatic" and receives a letdown heat exchanger outlet temperature signal from 2TC-4805.

2K12-C1 “LETDOWN HX 2E29 OUTLET TEMP HI” alarms if letdown temperature out of the letdown heat exchanger is > 140°F based on 2TC-4805.

If temperature as sensed by 2TC-4805 increases to 145°F, 2CV-4804 will close isolating letdown flow from the boronometer and process radiation monitor. If the high temperature condition has been corrected, the handswitch for 2CV-4804 must be taken to "close" and then returned to "auto" or "open" to restore flow.

2.1.13 CVCS Process Radiation Monitor 2RE-4806

The purpose of the CVCS process radiation monitor is to alert the operator of an increase in RCS radioactivity. An increase in RCS gross activity and iodine (I_{131}) may be indicative of possible failed fuel cladding. An increase in gross activity alone would be indicative of a crud burst, although a crud burst with some failed fuel would result in a small I_{131} spike also.

I_{131} is a fission product found in relative abundance in the event of a clad failure. It is released from fuel rods having cladding damage with relative ease and does not plate out on system surfaces. During radioactive decay I_{131} emits a 0.3645 MeV gamma particle that can be readily monitored using discrimination techniques.

I_{131} has an 8.04 day half-life that is long relative to sample time but short enough to indicate the current amount of fission product escape from the fuel[2].

The gamma detector used in the process radiation monitor is a gamma scintillation detector based on a sodium iodide crystal[3]. The detector is 1.5" in diameter x 1" long with a photo multiplier tube and internal pre-amplifier. It is designed to detect gamma radiation in the 100 kev - 3 MeV Range.

A logarithmic rate meter (2RITS-4806A) allows the operator to monitor for gross activity between 10 and 1E6 counts per minute. A single-channel linear rate meter / analyzer (2RITS-4806B) monitors specific I_{131} activity of between 0 and 1E6 cpm. These monitors are located on panel 2C22 in the Unit 2 control room. A Yokogawa

FOOTNOTES:

2 Some fission products that would contribute to gross activity readings plate out on pipe walls. Gross gamma activity after a minimum lag time is dominated by gaseous product activity so the gross activity reading is more representative of actual activity conditions of the RCS.

3 It has been known for many years that certain solids and liquids, called phosphors, emit light when exposed to radiation. When ionizing radiation passes through the phosphorus, the radiation causes the molecules of the material to become ionized or excited; these molecules then emit their excess energy in the form of light, each interacting particle resulting in a flash, generally called a *Scintillation*. In a scintillation detector, a photomultiplier tube detects this light, and the resulting pulses of current out of the photomultiplier indicate passage of the ionizing radiation through the scintillator. The resulting charge is proportional to the energy lost by the ionizing radiation to the scintillation crystals.

controller, 2RR-4806, on panel 2C14 provides a trend for the radiation monitors.

2RE-4806 is shielded with 4 ½" of lead in order to reduce any background radiation. The monitor itself is an in-line type with a sample volume 2 ½" in diameter and is 1 ¾" deep. The detector holder collimating plug must be placed between the detector and the sample cell to increase operating range above 1E-1 μCi/cc I_{131}.

As mentioned previously, the CVCS process radiation monitor is a trend monitor whose primary purpose is to indicate the possibility of fuel clad failure. Adjustable alarm setpoints are normally set at a value slightly above the current reading. It is expected that gross activity and perhaps I_{131} activity will periodically increase above the alarm setpoint due to normal plant transients. Consequently, the alarm will periodically activate and the operator must determine the cause of the alarm. If an alarm is received and the iodine-131 activity has increased and remains significantly above the prior steady state level, additional fuel failure can be assumed to have occurred. However, if an alarm is received due to high gross activity only, a crud burst release can be suspected.

In time, the RCS activity should return to the previous, lower, steady state value. If an alarm is received, the setpoint are immediately raised above the higher levels and the alarm circuits are reset. This ensures that any successive increases will be brought to the operator's attention.

Annunciator 2K12-A1 "LETDOWN RADIATION HI/LO" alarms under the following three conditions:

- RCS letdown gross activity is ≥ $4x10^5$ cpm (2RITS-4806A).
- RCS letdown iodine (I-131) activity exceeding selected scale setpoint (2RITS-4806B).
  - 1E2 ≥ 67 cpm
  - 1E3 ≥ 670 cpm
  - 1E4 ≥ 6,700 cpm
  - 1E5 ≥ 67,000 cpm
  - 1E6 ≥ 670,000 cpm
- detector failure, circuit failure, or loss of power to 2RE-4806.

To ensure proper indication, flow through the radiation monitor needs to be maintained between 0.5 - 1.5 gpm. Flow through the monitored can be observed on 2FIS-4807, located behind the door of a shielded vault on the 335' elevation of the auxiliary building. An annunciator 2K12-J1, "RAD MONITOR FLOW LO" is actuated when flow is ≤ 0.5 gpm. In the event of abnormal flow, a manual valve, 2CVC-139, is located just downstream of the tap-off for the radiation monitor.

The chain operator for 2CVC-139 is located in the letdown purification filter, 2F-3A/B, hallway. This valve may be used to throttle letdown flow to provide adequate back pressure to control flow through the process radiation monitor. The valve position, in terms of turns closed, is logged on the control room status board. If

2.2.1 CVCS Process Radiation Monitor

A small portion of the Reactor Coolant system is diverted through the Chemical and Volume Control System (CVCS) via the Letdown header. This Letdown flow can be used to monitor the RCS coolant radioactivity. A rise in the radioactivity of RCS could be caused by crud released in the RCS or failure of the fuel cladding of the Reactor fuel assemblies. Crud is the result of activated corrosion products that settle out in low flow areas of the RCS. When flow is changed or when a chemical shock occurs these corrosion products are released back into the RCS resulting in a rise in the gross gamma activity.

The CVCS Process Radiation monitor, 2RE-4806 is located on the 335’ elevation of the Auxiliary Building in the North - south hallway along the east wall. This process radiation monitor provides input to two separate monitoring circuits. One circuit provides the gross gamma activity while the other circuit provides specific fission product nuclide activity. The Gross gamma indication is read out on 2RITS-4806A while the specific activity level can be read on 2RITS-4806B.

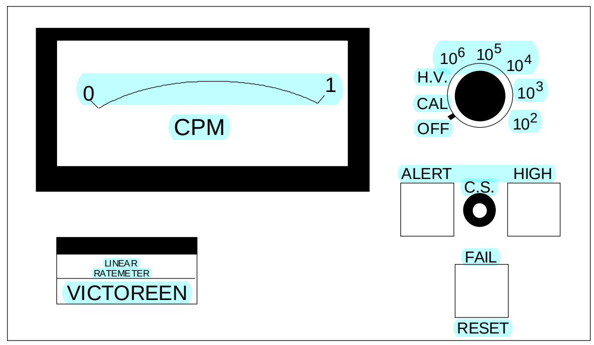
The specific activity monitor 2RITS-4806B monitors the Letdown fluid for the presence of Iodine-131. Iodine-131 is a fission product that is released with relative ease from defective fuel assemblies and does not plate out on the system surfaces. As with all radionuclides, Iodine-131 emits a gamma particle with a specific energy level while undergoing radioactive decay. The 0.364 Mev gamma released by the decay of Iodine -131 can be readily monitored since its half-life of 8 days is longer than the sample lag time but short enough to provide indication of current fission product release into the RCS. A rise in the gross activity only would be an indication of a crud burst, but a rise in both gross and specific would be an indication of fuel failure.

The CVCS Process Radiation Monitor, 2RE-4806, is located in the CVCS Letdown line in parallel with the Purification filter but upstream of the Ion exchangers. This location provides for a continuous sample at relatively low temperature and pressure that can be conveniently obtained and the sample effluent can be returned to the CVCS without difficulty. This location also provides an optimum compromise between a minimum required sample lag time and the delay time required for the sample background radioactivity to decay. The time lag from the RCS to the CVCS Process monitor is sufficient to permit N-16 to decay to a sufficiently low level that will not interfere with monitor readings in any operating condition.

Process radiation Monitors, 2RITS-4806A and 2RITS-4806B, are located in the Unit 2 Control Room on 2C22.

recorder is located in the Unit 2 Control Room on panel 2C14. Computer point R4806A is also fed from 2RITS-4806A.

The controls for the Specific activity monitor 2RITS-4806B, shown below, are very similar to those discussed for the gross activity monitor. The major difference between the two is the Function switch positions and the setpoints for the alarms. Specific activity monitor 2RITS-4806B is a linear ratemeter with an indicating range of 0 to 10^6 counts per minute. This range is broken into 5 scales using the Function switch; 0-100 CPM, 0-1000 CPM, 0-10000 CPM, 0-100000 CPM and 0- 1000000 CPM. The meter reads 0 to 1.0 and that reading must be multiplied by the function switch scale to obtain the correct CPM reading.



2RITS-4608B

The HV and OFF positions perform the same function as they did on 2RITS-4806A.

The CAL switch provides a calibrated signal to the circuitry that corresponds to 60,000 CPM.

The range positions vary the scale of the meter and also change the scale of 2RR-4806B blue pin. The range selected is also indicated on 2C14 below 2RR-4806 by 5 lights.

The alarm lights function the same as previously discussed but with different setpoints. The setpoint for the ALERT light corresponds to a meter reading of 0.4 CPM and the HIGH alarm is set to a meter reading of 0.67 CPM. As an example, if the function switch is in the 10^2 position, the ALERT alarm will come in at 40 CPM and the HIGH alarm will come in at 67 CPM. If the Function switch is in the 10^6 position, the ALERT alarm is at 400,000 CPM and the HIGH alarm is 670,000 CPM. Regardless of when the HIGH alarm comes in, annunciator 2K12-A1 in the Control Room will also alarm.

A low alarm circuit is provided to warn of possible detector malfunction or circuit failure. To prevent nuisance alarms, the low alarm circuit is independent of the selected range. A PIC microcontroller circuit monitors detector counts over a one minute rolling time period and will actuate the low alarm if counts are 10

HIGH ACTIVITY IN RCS

PURPOSE

This procedure provides actions for rising or high specific activity in RCS.

ENTRY CONDITIONS

ANY of the following conditions exist:

1. "LETDOWN RADIATION HI/LO" annunciator (2K12-A1) in alarm.
2. NSSS Process Radiation Recorder (2RR-4806) indicates unexplained rising trends.
3. RCS sample indicates specific activity approaching or exceeding TS 3.4.8 limits.

EXIT CONDITIONS

WHEN ALL actions of this procedure complete, THEN exit procedure.

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

- **A dropped CEA or sudden power rise are examples of transients that could cause fuel damage.**
- **An NUE should not be declared early unless an initiating event (transient) has occurred. If specific activity is rising during steady state operation, no declaration should be made until the limits are reached.**

7. Check BOTH the following conditions exist: • RCS Letdown Iodine 131 monitor (2RITS-4806B) activity stable or lowering. • A transient has NOT occurred that could result in fuel damage.	**7. <u>IF</u> Iodine 131 activity rising <u>AND</u> transient occurred that could cause fuel damage, <u>THEN</u> consider Notification of Unusual Event using 1903.010, Emergency Action Level Classification.**
8. <u>IF</u> Iodine 131 activity rising, <u>THEN</u> contact Reactor Engineering to monitor fuel using EN-NF-102, Corporate Fuel Reliability.	

INSTRUCTIONS	CONTINGENCY ACTIONS

*9. **Check BOTH the following conditions exist:**

- Aux Building Area Radiation monitors less than 0.1 R/hr.
- RCS Letdown Gross monitor (2RITS-4806A) activity stable or lowering.

*9. **At SM discretion perform the following:**

A. Isolate Letdown by verifying at least ONE Letdown Isolation valve closed:

- 2CV-4820-2
- 2CV-4821-1
- 2CV-4823-2 (least preferred)

B. Maintain PZR level within 5% of setpoint by cycling Charging pumps.

C. Record Charging Header data on 2202.010 Attachment 44, Charging Header Data.

D. IF plant shutdown required, THEN perform the following:

1) Refer to applicable reactivity plan.

2) Commence Plant shutdown using 2102.004, Power Operation.

E. Refer to 1903.010, Emergency Action Level Classification.

F. IF Letdown isolated AND Aux Building radiation high, THEN close RCP Bleedoff to VCT valves:

- 2CV-4846-1
- 2CV-4847-2

10. **Notify RP to perform Aux Building area radiation surveys.**

END

PARENT QUESTION 2209 WAS USED TO MODIFY 2017 NRC EXAM QID#27 (Used on 2015 NRC Exam)

Data for 2015 NRC RO/SRO Exam

Bank: 2209 **Rev:** 0 **Rev Date:** 6/9/2015 12:07:51 **QID #:** 23 **Author:** Hatman

Lic Level: R **Difficulty:** 3 **Taxonomy:** F **Source:** NRC Bank #0277

Search 000076A104 **10CFR55:** 41.7 / 45.5 / 45.6 **Safety Function** 9

System Title: High Reactor Coolant Activity **System Number** 076 **K/A** AA1.04

Tier: 1 **Group:** 2 **RO Imp:** 3.2 **SRO Imp:** 3.4 **L. Plan:** A2LP-RO-RWST **OBJ** 4

Description: Ability to operate and/or monitor the following as they apply to the High Reactor Coolant Activity: - Failed fuel-monitoring equipment

Question:

Consider the following:

* Plant is returning to full power after a transient.
* LETDOWN RADIATION HI/LO annunciator (2K12-A1) alarm comes in.
* Letdown Gross Activity Monitor (2RITS-4806-A) reads 2E+5 CPM and rising.
* Letdown I-131 Activity Monitor (2RITS-4806-B) reads 1E+5 CPM and rising.

Which of the following events occurred for the given indications?

A. Failed Fuel

B. RCS chemical shock

C. RCS crud burst

D. Letdown Demineralizer exhausted

QID use History

	RO	SRO
2005	☒	☒
2006	☒	☒
2008	☐	☐
2009	☐	☐
2011	☐	☐
2012	☐	☐
2014	☐	☐

Audit Exam History

2006	☒
2009	☐
2011	☒

Answer:

A. Correct

Notes:

Rising activity on both the Gross and Iodine 131 monitors indicate Fuel Failure.

Distracters B and C are incorrect because only Gross activity would rise in these cases.

Distracter D is incorrect because this case would cause a change in RCS chemistry but not activity.

References:

STM 2-62, Radiation Monitoring System, Section 2.2.1.
2203.020, RCS High Activity AOP, Entry Conditions and Step 6

Historical Comments:

Parent QID 0277 used on 2000 and 2005 NRC Exams.

Question 28

Bank:	2358	**Rev:**	0	**Rev Date:**	7/21/2016	**2017 TEST QID #:**	28	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	Modified NRC Exam Bank 2228		
Search	026000A103	**10CFR55:**	41.7	**Safety Function**	5				
Title:	Containment Spray System (CSS)	**System Number**	026	**K/A**	A1.03				
Tier:	2	**Group:**	1	**RO Imp:**	3.5	**SRO Imp:**	3.5	**L. Plan:**	A2LP-RO-SPRAY
OBJ	8								

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: - Containment sump level

Question:

During a Large Break LOCA Design Basis Accident, the Containment Spray Pumps will provide cooling flow to the Containment from the RWT until the RWT drops below __________. At this point, the level of water in Containment should be no less than ________________________________ to ensure adequate NPSH for the Containment Spray Pumps post RAS.

A. 40%; 86% Containment Sump Level indication on 2C-33

B. 40%; 86 inches Containment Sump Flood Level on 2C-16/17

C. 6%; 86% Containment Sump Level indication on 2C-33

D. 6%; 86 inches Containment Sump Flood Level on 2C-16/17

Answer:

D. 6%; 86 inches Containment Sump Flood Level on 2C-16/17

Notes:

D is correct as the water level in the Containment after RWT water (384,000 gallons) has been introduced and excluding the water in the reactor vessel cavity should be greater then 7.16 feet (86 inches indicated) above the 336’ 6 elevation of the Containment floor. Indication is read out on the Wide Range Containment Flood Level indicators on 2C16/17 and the measurements are in inches. This instrument combines both the level in the containment sump and the containment basement to provide an indication of adequate inventory for NPSH concerns for the ESF pumps.

A is incorrect but plausible as the 40% is when the "RWT Level LO RAS PRETRIP" alarm comes in and the Containment Sump level is read out on 2C-33 and measured in %. This indicator spans the small part of the Containment sump and is used to identify small RCS leaks and to collect normal full power leakage inside containment.

B is incorrect as the 40% is when the "RWT Level LO RAS PRETRIP" alarm comes in but plausible as the correct containment basement level indication is provided for assessment of adequate NPSH.

C is plausible as this is the correct RWT setpoint to transfer the suction of the ESF pumps to the Containment Basement but the incorrect level indication is given for assessment of adequate NPSH.

This question matches the K&A because the candidate must be able to monitor the changes on the correct instrumentation to ensure NPSH design limits have not been exceeded for the Containment Spray Pumps.

References:

RAS Actuated Annunciator Response OP-2203.012G Window 2K07 H1 Rev. 33; (Verified reference updated 11/15/16); RWT Level LO RAS PRETRIP Annunciator Response OP-2203.012F Window 2K06-A9 Rev. 38 (Verified reference updated 11/15/16); STM_2-08_22-1 RCB Spray System Section 3.1.1 Design Basis (Verified reference updated 11/15/16); STM 2-13_17 Sections 5.4.1/5.4.2 Containment Sump/Flood Level. (Verified reference updated 11/15/16);

Historical Comments:

NRC Exam Bank 2228 was used on the 2015 NRC Exam
To be used on the 2017 NRC Exam but modified for the 2017 NRC Exam

ANNUNCIATOR 2K07

H-1

RAS ACT

1.0 CAUSES

1.1 RAS Channel 1 actuated due ONE of the following:

- RWT level ≤ 6% (2 out of 4)
- PPS Testing
- PPS Relay failure

2.0 ACTION REQUIRED

2.1 Determine validity of alarm by checking RWT (2T-3) level.

2.2 IF authorized testing in progress,
THEN no action required.

2.3 IF RAS valid,
THEN GO TO appropriate Emergency Operating Procedure.

2.4 IF RAS invalid,
THEN GO TO Inadvertent RAS (2203.040).

2.5 IF failed relay is the cause,
THEN perform the following:

2.5.1 Refer to Tech Spec 3.3.2.1.

2.5.2 Submit WR.

2.6 WHEN RWT level greater than 40%,
THEN reset RAS trip path as follows:

2.6.1 Place LK/UNLK switch to UNLK (key No. 15).

2.6.2 Depress RAS pushbutton on 2C23.

2.6.3 Verify Trip Path lights reset on local PPS Status panels.

2.6.4 Place LK/UNLK switch to LK and remove key.

2.6.5 Repeat above steps to reset remaining Actuation Trip Paths.

2.6.6 Depress EITHER RAS Lockout Reset pushbutton on 2C40

- Verify BOTH Reset lights ON.

2.6.7 Depress EITHER RAS Lockout Reset pushbutton on 2C39

- Verify BOTH Reset lights ON.

(H-1 Continued on next page)

ANNUNCIATOR 2K06

A-9

RWT LEVEL LO LO RAS PRETRIP

1.0 CAUSES

1.1 Refueling Water Tank (2T-3) level < 40%.

2.0 ACTION REQUIRED

2.1 Check RWT Level.

2.2 IF during ESFAS actuation,
THEN perform actions of Emergency Operating Procedures for RAS Pretrip.

2.3 IF during refueling operations,
THEN maintain RWT level greater than 7.5%.

2.4 Refer to TRM 3.1.7 and TS 3.5.4.

3.0 TO CLEAR ALARM

3.1 Raise RWT level > 40%.

4.0 REFERENCES

4.1 E-2455-2

4.2 LIC-01-017, ANO2 TS Am. 229, Relocation of Boric Acid Makeup Systems from TSs to TRM

Level switch 2LIS-5643A will de-energizes the heaters to prevent damage if RWT level lowers to approximately 7% of indicated level in the control room. Other level instruments, with redundant channels, provide indication and alarms in the Control Room and inputs to the computer, ESFAS and the RAS logic systems. More details on RWT instruments are in the tables at the end of the chapter.

All the level detectors and transmitters located at the tank are housed in heated enclosures to protect the instruments from freezing. Gas pressure temperature indicators provide local indication of temperature inside the housings. The heating element in each housing is thermostatically controlled.

For Modes 1, 2, 3, and 4, Technical Specification 3.5.4 requires:

1. A contained borated water volume of between 384,000 and 503,300 gallons (equivalent to an indicated tank level of between 91.7% and 100%, respectively),
2. Between 2500 and 3000 ppm boron,
3. A minimum solution temperature of 40°F, and
4. A maximum solution temperature of 110°F.

For Modes 5 and 6, Technical Requirements Manual 3.1.2.7 requires:

1. A minimum contained borated water volume of 61,370 gallons (equivalent to 7.5% of indicated tank level),
2. A minimum of 2500 ppm boron, and
3. A minimum solution temperature of 40°F.

With the HPSI Pressurization System (HPS) aligned, RWT level is required to be maintained above 95% to ensure adequate RWT volume is available to meet ECCS design requirements assuming a failure of HPS tubing[1].

3.1.1 Design Bases

RWT capacity is based on the requirement for filling the refueling canal and is in excess of the 384,000 gallons required for Containment Spray and safety injection during any postulated LOCA or MSLB. The safety analysis RWT maximum design temperature is 120°F. The RWT is designed to withstand an internal pressure equal to a column of water 10 feet above the liquid level of the tank when full. An internal vacuum of one (1) psi below atmospheric is also part of its design criteria.

This requirement of 384,000 gallons is based upon all seven ESF pumps operating at their rated flows. This provides sufficient water so that the ESF pumps can take suction from the RWT for a

FOOTNOTES:

[1] Refer to ER-2000-3275-007 for failure assumptions and calculation details.

minimum of 30 minutes after initiation of emergency core cooling and provide adequate water for long term recirculation.

The requirement of 61,370 gallons is based on having the boron capability required below 200°F to provide sufficient shutdown margin after xenon decay and cooldown from 200°F to 140°F.

System design ensures compliance with the pump net positive suction heat (NPSH) requirements of Regulatory Guide 1.1in that no credit is taken for increased containment pressure as a result of accident conditions when determining available NPSH. Minimum Technical Specification allowed containment pressure is assumed as the containment pressure and establishes the minimum sump saturation temperature. Sump temperatures below this are considered subcooled. Sump temperatures above this temperature are considered saturated water conditions with containment pressure set equal to the liquid vapor pressure.

$$NPSHa = h_a - h_{vpa} + h_{st} - h_L$$

h_a = absolute pressure in feet of liquid on the surface of the liquid level.

h_{vpa} = the head in feet corresponding to the vapor pressure of the liquid at the temperature being pumped.

h_{st} = static height in feet that the liquid supply level is above or below the pump centerline or impeller eye.

h_L = all suction line losses in feet including entrance losses and friction losses through pipe, valves and fittings, etc.

The water level in the Containment after RWT water (384,000 gallons) has been introduced and excluding the water in the reactor vessel cavity should be greater then 7.16 feet (86 inches indicated) above the 336’ 6” elevation of the Containment floor. The Safety Injection Tank (SIT) contents of approximately 5600 ft^3 are included in the inventory available for recirculation. This height of water provides adequate NPSH for the ESF pumps even under boiling sump conditions.

3.1.2 RWT Outlet Valves

The RWT outlet valves, 2CV-5630-1 & 2CV-5631-2, are located in the overhead of the northwest corner of the 2T21 tank room. Their valve operators penetrate the ceiling into the waste gas control panel room. Control hand switches (2HS-5630-1 & 2HS-5631-2) and associated indicators are located on Control Room panels 2C17 and 2C16 respectively. Both control switches are spring return to normal with OPEN, center and CLOSE positions. RWT Outlet Valve 2CV-5630-1 may also be operated locally from MCC 2B52-E4. The valves are normally open when the Containment Spray system is lined up for standby operation.

The valves will open if the following conditions are met:

5.3 Containment Pressure

Containment pressure is monitored by two wide range pressure transmitters (0-225 psia) and four narrow range pressure transmitters (0-27 psia).

Of the two wide range pressure transmitters one of them feeds a chart recorder on control room panel 2C-14 and both of them have indication on 2C-33. They can also be monitored on SPDS.

All four narrow range pressure transmitters have indication on 2C-33. One of them feeds a chart recorder on 2C-33. Each of them sends a signal to each of the four channels of the plant protection system (PPS) (for reactor trip, CIAS, SIAS, CCAS at 18.3 psia, and CSAS at 23.3 psia). They can also be monitored on the plant computer. There is a computer point that is the average of the four points.

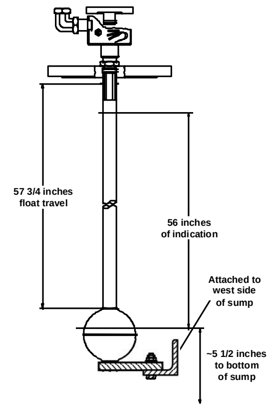
Monitoring containment temperature and pressure is a method of detecting overall gross leakage from all water and steam systems within the containment.

5.4 Containment Level

Changes in the containment sump water level are an indication of overall gross leakage from all water and steam systems within the containment. Containment water level is monitored by a containment sump level detection system and a reactor building flood level monitoring system which provide control room operators with alarm and indication readout of containment water level.

5.4.1 Containment Sump Level

The containment sump level is monitored by one level transmitter as shown in the following figure (see also figure on page 75). It feeds a level indicating switch on control room panel 2C-33. The switch will cause an alarm on 2K10-B7, CNTMT SUMP LEVEL HI, at 85%. This point can also be monitored on SPDS, the plant computer, and on a transmitter in the upper north electrical penetration room. The indications for containment sump level are in percent (0 to 100%) with each percent equal to 39 gallons.



Containment Sump Level Transmitter

5.4.2 Containment Flood Level

Two level transmitters monitor actual level inside the containment. Each transmitter has an input from two separate level elements as shown in the following figure (see also figure on page 76).

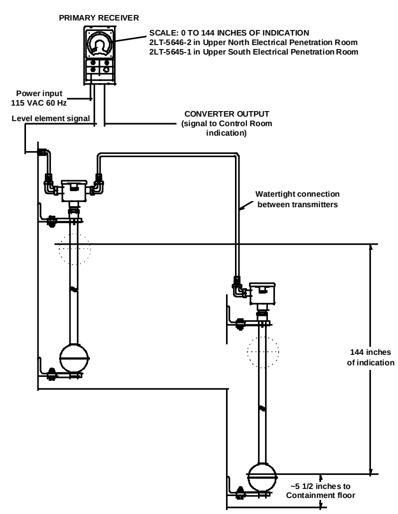
One of the transmitters feeds a chart recorder on 2C-33 and both transmitters go to level indicators, one on control room panel 2C-16 and the other on control room panel 2C-17. Both can be monitored on SPDS, the plant computer and on transmitters in the upper north and south electrical penetration rooms. These indications read out in inches (0 to 144 inches) however, it is important to recognize that there is a 5.5-inch difference between the containment floor and 0 inches indicated level.

Containment flood level indication is used by the operators to ensure that the water level in containment meets or exceeds the minimum established level credited in accident analysis and testing following a recirculation actuation signal (RAS). The minimum water level ensures adequate NPSH and vortex protection for the high-pressure safety injection and containment spray pumps, as well as ensuring the strainer head loss will remain bounded by qualification tests.

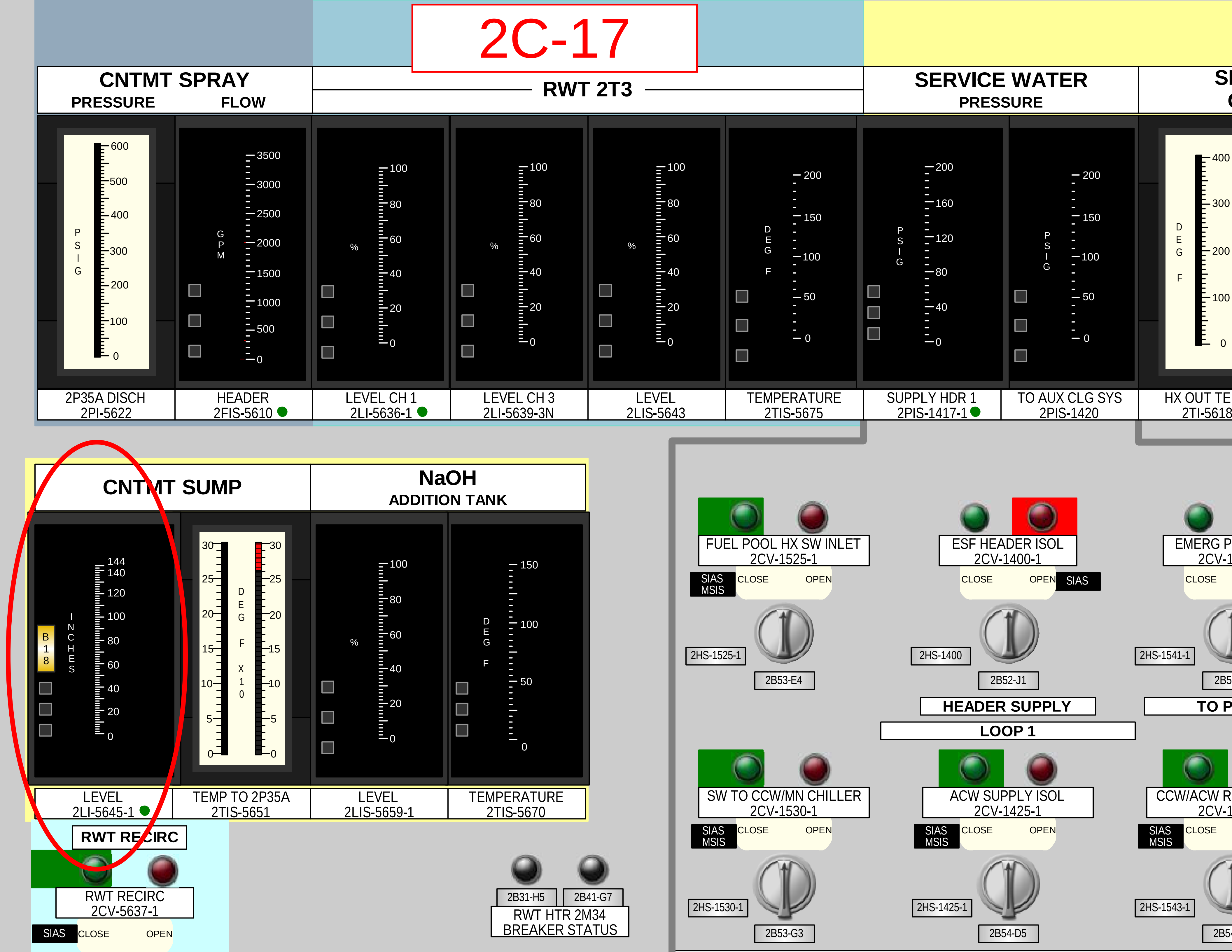
The available NPSH margin after subtracting for strainer head loss should not be credited with a direct reduction of water level in containment, particularly for water levels below the top of the strainer cartridge assemblies. Lower water levels that do not fully submerge the sump strainer cartridges reduce the available strainer surface area, resulting in higher velocities and debris loading on the submerged surface area, which can result in rapid increases in strainer head loss. In addition, the vortex protection grating over the sump suction pipes require a minimum submergence that may not be met if the containment minimum water level is not maintained.

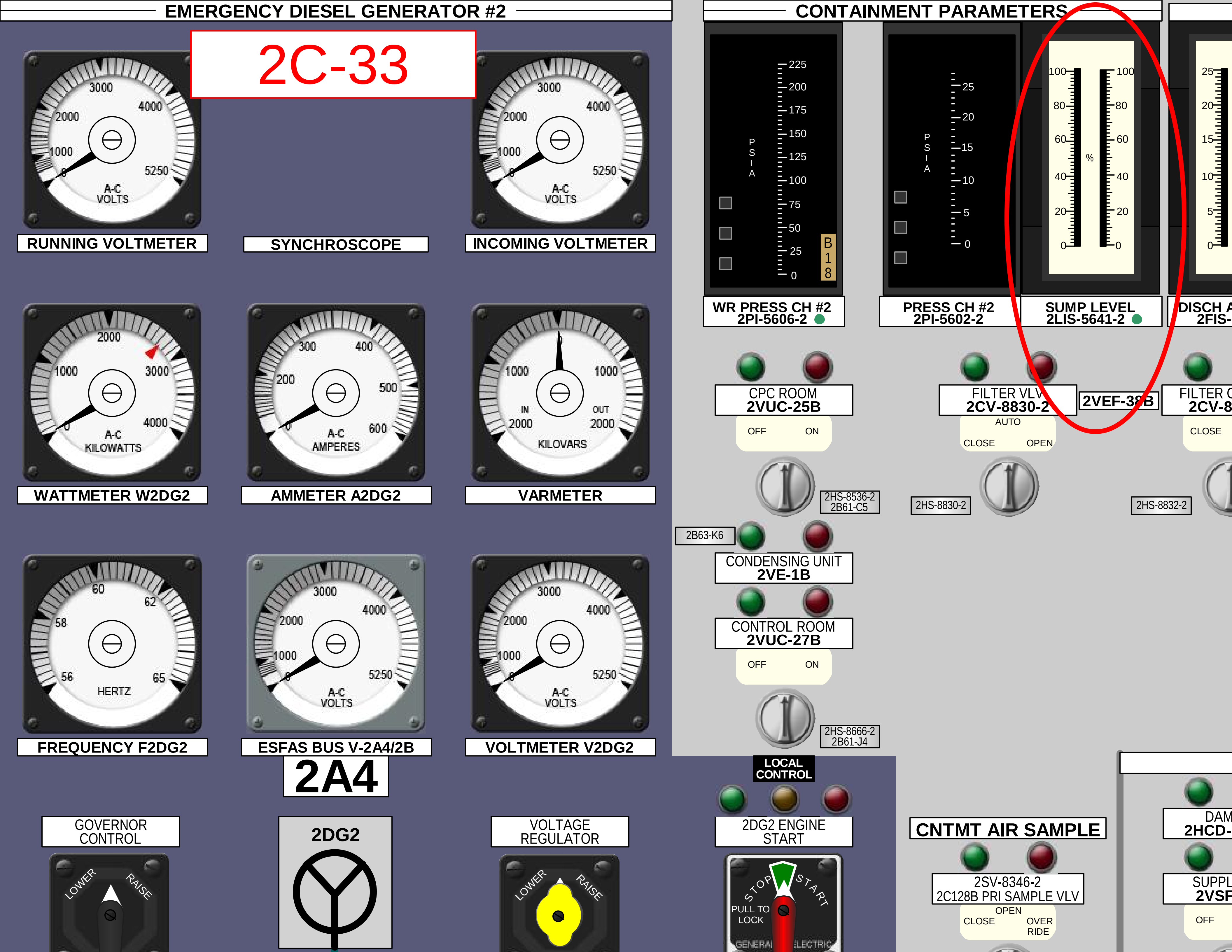
The minimum and maximum containment flood levels are as follows:

- Minimum water level = 7.16 ft (86 inches indicated on 2C16/17)
- Maximum water level = 11.42 ft (137 inches indicated on 2C16/17)



Containment Flood Level Indication


2C-17
CNTMT SPRAY
PRESSURE
FLOW
RWT 2T3
SERVICE WATER
PRESSURE
PSIG
GPM
%
DEG F
2P35A DISCH
2PI-5622
HEADER
2FIS-5610
LEVEL CH 1
2LI-5636-1
LEVEL CH 3
2LI-5639-3N
LEVEL
2LIS-5643
TEMPERATURE
2TIS-5675
SUPPLY HDR 1
2PIS-1417-1
TO AUX CLG SYS
2PIS-1420
HX OUT TE
2TI-5618
CNTMT SUMP
NaOH
ADDITION TANK
INCHES
B18
DEG F X 10
LEVEL
2LI-5645-1
TEMP TO 2P35A
2TIS-5651
LEVEL
2LIS-5659-1
TEMPERATURE
2TIS-5670
RWT RECIRC
RWT RECIRC
2CV-5637-1
SIAS
CLOSE
OPEN
2B31-H5
2B41-G7
RWT HTR 2M34
BREAKER STATUS
FUEL POOL HX SW INLET
2CV-1525-1
SIAS MSIS
2HS-1525-1
2B53-E4
ESF HEADER ISOL
2CV-1400-1
2HS-1400
2B52-J1
EMERG P
2CV-1
2HS-1541-1
HEADER SUPPLY
LOOP 1
TO P
SW TO CCW/MN CHILLER
2CV-1530-1
2HS-1530-1
2B53-G3
ACW SUPPLY ISOL
2CV-1425-1
2HS-1425-1
2B54-D5
CCW/ACW R
2CV-1
2HS-1543-1



EMERGENCY DIESEL GENERATOR #2
2C-33
RUNNING VOLTMETER
SYNCHROSCOPE
INCOMING VOLTMETER
WATTMETER W2DG2
AMMETER A2DG2
VARMETER
FREQUENCY F2DG2
ESFAS BUS V-2A4/2B
VOLTMETER V2DG2
2A4
GOVERNOR CONTROL
2DG2
VOLTAGE REGULATOR
LOWER
RAISE
CONTAINMENT PARAMETERS
WR PRESS CH #2
2PI-5606-2
PRESS CH #2
2PI-5602-2
SUMP LEVEL
2LIS-5641-2
DISCH A
2FIS-
CPC ROOM
2VUC-25B
OFF
ON
2HS-8536-2
2B61-C5
FILTER VLV
2CV-8830-2
AUTO
CLOSE
OPEN
2HS-8830-2
2VEF-38B
FILTER C
2CV-8
2HS-8832-2
2B63-K6
CONDENSING UNIT
2VE-1B
CONTROL ROOM
2VUC-27B
2HS-8666-2
2B61-J4
LOCAL CONTROL
2DG2 ENGINE START
STOP
START
PULL TO LOCK
CNTMT AIR SAMPLE
2SV-8346-2
2C128B PRI SAMPLE VLV
OVER RIDE
DAM
2HCD-
SUPPL
2VSF

PARENT QUESTION 2228 WAS USED TO MODIFY 2017 NRC EXAM QID#28 (Used on 2015 NRC Exam)

Questions For All QID In Exam Bank

Bank: 2228	**Rev:** 0	**Rev Date:** 6/10/2015 9:15:36		**QID #:** 42		**Author:** foster	
Lic Level: R	**Difficulty:** 2	**Taxonomy:** F	**Source:** NEW				
Search 026000K407	**10CFR55:** 41.7		**Safety Function** 5				
System Title: Containment Spray System (CSS)			**System Number** 026		**K/A** K4.07		
Tier: 2	**Group:** 1	**RO Imp:** 3.8	**SRO Imp:** 4.1	**L. Plan:** A2LP-RO-SPRAY		**OBJ** 4	

Description: Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: - Adequate level in containment sump for suction (interlock)

Question:

During a Large Break LOCA, the containment sump isolation valves and RWT outlet valves are interlocked to automatically swap ECCS pump suctions when RWT lowers to ____ to ________________ .

A. 6%; ensure adequate NPSH for ECCS pumps

B. 6%; lower radiation levels in the ECCS pump vaults

C. 7%; ensure adequate NPSH for ECCS pumps

D. 7%; lower radiation levels in the ECCS pump vaults

QID use History

	RO	SRO
2003	☐	☐
2005	☑	☑
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☐

Answer:

A. Correct

Notes:

A. Correct: During a DBA the ECCS pumps will start and inject water from the RWT through the core and into the containment building. As the RWT is depleted ESFAS will trigger a Recirc Actuation Signal (RAS) when RWT level lowers to 6%. This will provide long term cooling to the core and containment building. When RAS is actuated, it will isolate the RWT from the ECCS system by closing the ECCS recirc isolation valves and swap the ECCS suctions to the containment sump by opening the containment sump isolation valves and closing the RWT outlets. The RAS will prevent possible contamination/radiation release outside the Auxiliary building (isolating the RWT from containment sump water) and provide NPSH to the ECCS pumps (where all of the RWT water wa transferred to).
B. Incorrect, a RAS will occur at 6% RWT level but the ECCS suction is automatically swapped to the containment sump to provide NPSH to the ECCS pumps. Plausible due to the RAS signal will isolate the RWT to prevent the possible spread/release of radiation/contamination outside of the auxiliary building. When containment sump water is recalculated through the ECCS rooms radiation levels will rise significantly.
C. Incorrect, a RAS will occur at 6% RWT level not 7% but the ECCS suction do automatically swapped to the containment sump to provide NPSH to the ECCS pumps. Plausible due to the RWT submersion heater low level cutout to prevent heater damage is 7%.
D. Incorrect, a RAS will occur at 6% RWT level not 7% but the ECCS suction do automatically swapped to the containment sump to provide NPSH to the ECCS pumps. Plausible due to the RWT submersion heater low level cutout to prevent heater damage is 7%. Also, plausible due to the RAS signal will isolate the RWT to prevent the possible spread/release of radiation/contamination outside of the auxiliary building. When containment sump water is recalculated through the ECCS rooms radiation levels will rise significantly.

References:

STM 2-08, Containment Spray System, Rev. 22, section 3.1, pages 6 and 7, section3.1.1. pages 7 and 8, section 3.1.2 pages 8 and 9, and table page 39
OP-2203.003, Loss of Coolant Accident, Rev 015, section 3 step 22 page 39 of 74
EOP-2203.003, Loss of Coolant Accident EOP Tech Guide, Rev 015, section 3 step 21 pages 114 and 115 of 141

Historical Comments:

New for 2015 exam
Incorporated validation comments.

Question 29

Bank:	2359	**Rev:**	1	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	29	**Author:**	Coble	
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC Exam Bank 2118			
Search	076000K307	**10CFR55:**	41.8	**Safety Function**	4					
Title:	Service Water System (SWS)	**System Number**	076	**K/A**	K3.07					
Tier:	2	**Group:**	1	**RO Imp:**	3.7	**SRO Imp:**	3.9	**L. Plan:**	A2LP-RO-ESPTA	**OBJ** 11

Description: Knowledge of the effect that a loss or malfunction of the SWS will have on the following: - ESF loads

Question:

Following a Reactor Trip, a check of the Vital Auxiliaries Safety Function status indicates that:

* #1 Emergency Diesel Generator (2DG1) is Running
* 2DG1 Frequency is 60 Hz
* 2DG1 Voltage is 4165 V
* 4160V Vital AC Bus 2A3 is Deenergized
* Annunciator 2K08-B3 "2A3 LO RELAY TRIP" has actuated

2DG1 should be secured to prevent ____ .

A. overheating the diesel engine

B. uneven lower crankshaft bearing wear

C. buildup of unburned fuel in the exhaust manifold

D. running the diesel engine without Jacket Cooling Water flow

Answer:

A. overheating the diesel engine

Notes:

A. Correct: with a lockout on the vital AC bus (2A3) the service water pump would be deenergized and overheating of the engine is a concern.

B. Incorrect: plausible because uneven lower crankshaft bearing wear due to the engine being run unloaded could be a concern but does not required an engine shutdown.

C. Incorrect: plausible because buildup of unburned fuel in the engine exhaust manifold during engine low load operations is a concern but does not required an engine shutdown.

D. Incorrect: plausible because Jacket Cooling Water has a engine driven pump which will maintain Jacket Cooling Water flow to the EDG therefore no requirement to secure EDG.

This question matches the K/A as it requires knowledge of the power supply to the Loop 1 SW pump and the ability to diagnose the loss of the SWS to the #1 EDG ESF load and the knowledge of the procedural action to take.

References:

OP-2202.001, Standard Post Trip Actions, Rev 015, step 4.G contingency action (Verified reference updated 11/15/16); OP-2202.001, Standard Post Trip Actions, TG Rev 015, step 4.G contingency action (Verified reference updated 11/15/16); OP-2104.036 EDG Operations Rev. 91, Step 5.8, 5.9, and 5.10 (Verified reference updated 11/15/16); STM 2-31 EDGs Rev 34 Section 2.6 Engine Lube Oil System.(Verified reference updated 11/15/16).

Historical Comments:

NRC Exam Bank 2118 was used on the 2014-2 NRC Exam
To be used on the 2017 NRC Exam.

REV. 1 based on NRC Chief Examiner Feedback BNC. Replaced the modified version of NRC Exam Bank 2118 with the exact Bank 2118 version as requested.

INSTRUCTIONS	CONTINGENCY ACTIONS
4. Check Maintenance of Vital Auxiliaries satisfied:	
A. Check Main Turbine tripped by BOTH of the following: • ALL Main Stop Valves closed. • Generator megawatts indicate zero.	A. Perform the following: 1) Manually trip Main Turbine at 2C01. 2) IF Main Turbine does NOT trip, THEN close MSIVs: • 2CV-1010-1 • 2CV-1060-2
B. Generator Output breakers open.	B. Open Generator Output breakers: • 5130 • 5134
C. Perform EITHER of the following as required: 1) Check the following valves closed: • MSR 2E-12A Steam Supply From SG A (2CV-0400) • MSR 2E-12B Steam Supply From SG B (2CV-0460) 2) No flow indicated on the following MSR second stage flow instruments: • 2FI-0402 • 2FI-0462	C. Close MSIVs: • 2CV-1010-1 • 2CV-1060-2

(Step 4 continued on next page)

INSTRUCTIONS	CONTINGENCY ACTIONS
4. (continued)	
_____D. At least ONE 6900v AC bus energized.	
_____E. At least ONE 4160v Non-vital AC bus energized.	
_____F. BOTH 4160v Vital AC buses energized.	F. Perform the following: 1) IF de-energized 4160v Vital AC bus available AND associated EDG available, THEN verify associated EDG supplying bus. 2) IF NEITHER DG available, THEN start AACG AND align to associated 4160v Vital bus using 2104.037, Alternate AC Diesel Generator Operations, Attachment E. _____ 3) Check at least ONE 4160v and 480v Vital AC bus energized.
G. BOTH DGs secured.	G. IF ANY DG running AND SW NOT aligned, THEN locally stop DG by unlocking and placing "ENGINE CONTROL" handswitch in LOCKOUT: • 2E11 • 2E21
_____H. At least ONE 125v Vital DC bus energized: • 2D01 - SPDS point E2D01 • 2D02 - SPDS point E2D02	

E. The EOP only requires that one non-vital 4160v be energized vice requiring that all non-vital AC buses have electrical power. The non-vital 4160v and 480v buses provide power to non-vital loads and are not required for safe shutdown. The absence of power to at least one of these buses is used to indicate that a Loss of Offsite Power has occurred, and will require that appropriate operator actions be directed in follow-up EOPs.

F The vital 4160v and 480v AC buses provide equipment essential for the safe shutdown of the plant. These buses are capable of being supplied from either the UAT, Startup Transformer 3, or Startup Transformer 2 via the 4160v non-vital buses, or directly from the Emergency Diesel Generators. These buses supply all the safety-related loads and those non-safety-related loads for which it is desirable to have manual switch access to the diesel generators (2). These buses are redundant, and therefore, only one set of buses is required to be energized to satisfy this safety function. If neither EDG is available, the AACG can energize at least one vital bus.

The RTR, as well as other follow-on procedures, has been designed to be implemented even if not all vital and non-vital AC buses are energized, providing that at least one of each is. The RTR and other follow-on procedures provide the required actions to restore the de-energized buses. If a 4160v Vital AC bus is available and the associated EDG is available, the operator is directed to energize the bus from the EDG. If neither EDG can power up its respective 4160v Vital AC bus, a contingency step is provided to direct starting of the AACG to provide power to at least one Vital AC Train.

G. The EOP does check that both DGs are secured. Should the DG be running with no Service Water aligned, the DG is secured locally to prevent damage. (4)

H. The DC system has been designed to provide a source of reliable, continuous power for control, instrumentation, RPS, ESFAS, and other loads for startup, operation, and shutdown under normal and emergency conditions. The DC system consists of three 125v batteries. The two vital DC battery banks 2D01 and 2D02 each have two full capacity chargers while the non-vital DC battery bank 2D03 has one full capacity charger. Each battery bank has a DC control center and distribution panels. The safety-related DC loads have been grouped into redundant load groups such that the loss of either group will not prevent the minimum safety function from being performed (3). Therefore, only one of these buses is required to satisfy the safety function. Non-vital 125v DC bus 2D03 is not required to satisfy the safety function, since it feeds only non-safety related DC equipment.

5.0 LIMITS AND PRECAUTIONS

5.1 All EDG start attempts are logged in the Station log and recorded using 2104.036A, EDG Start/Load/Run Information Sheet.

5.2 If maintenance performed on Lube Oil System, then Pre-lube pump is run until all equipment filled and vented and oil flow is visually verified at the most remote bearing (cylinder 1 upper crank).

5.3 Due to reactive load sharing concerns, operation with BOTH EDGs tied to the grid is not desirable.

5.4 NEITHER EDG is tied to the grid during periods of offsite grid disturbance.

5.5 During a grid disturbance, the EDG will run loaded with house loads (approximately 700-800 KW). This light load is acceptable as exhaust vapor will reach lube oil vaporization temperature, burn and flow out of the stack. (CR-ANO-C-2000-0024)

5.6 Operating the AAC generator in parallel with an EDG will cause the EDG governor to operate in speed control and may overload or underload attempting to control speed.

5.7 EDG is shutdown during testing if any cylinder exceeds a 300°F differential temperature limit in relation to other cylinders unless installed thermocouples are determined NOT VALID.

5.8 If EDG is run unloaded for greater than ten minutes or multiple starts are performed, then engine is operated at full load for one hour prior to securing. Requirement may be waived by system engineering if engine loaded prior to occurrence.

5.9 EDG running unloaded for greater than 20 minutes OR less than 1400 KW for greater than 20 minutes requires engine to be loaded to 1400 KW for 30 minutes followed by the one hour full load run. However, loading may be as directed by System Engineering if following EDG maintenance.

5.10 EDG loading of less than 1400 KW for long periods of time can cause fuel and oil to build in the exhaust manifold which can cause fires in the exhaust manifold the next time the EDG is started. Therefore, the time required to load an EDG to 1400 KW after starting it up is limited to 10 minutes. However, loading may be as directed by System Engineering if following EDG maintenance.

(Section 5.0 continued on next page)

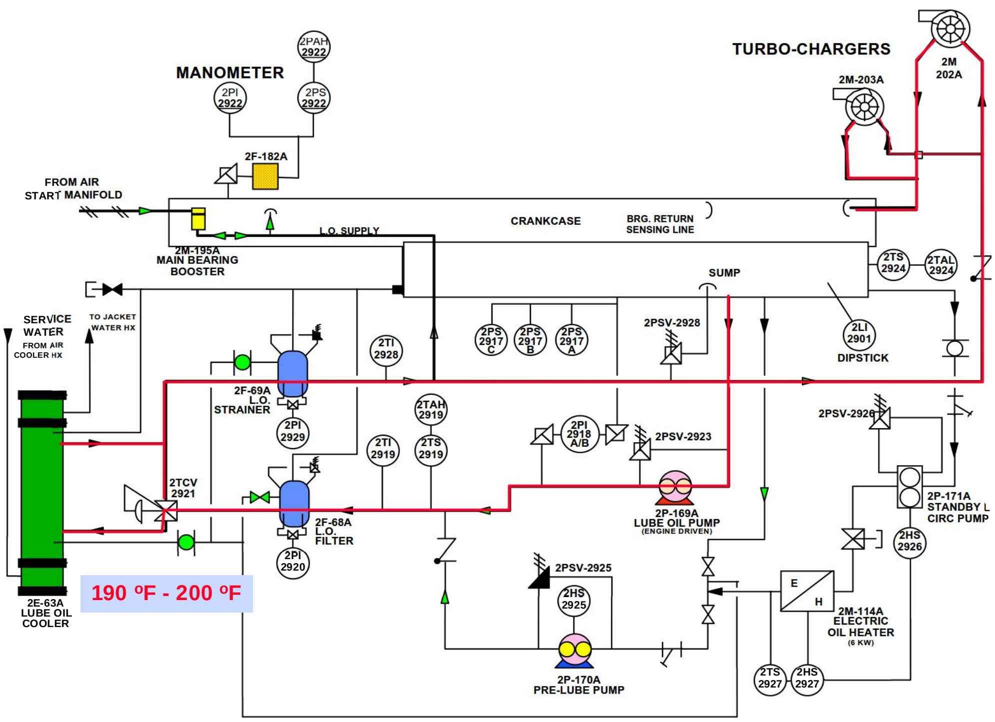
2.6 Engine Lube Oil System

Refer to the following drawings and in the back of this STM.

The Emergency Diesel Engine Lube Oil System consists of the following components.

- Engine Driven Lube Oil Pump (2P-169A/B)
- Motor Driven Pre-Lube Oil Pump (2P-170A/B)
- Motor Driven Sump Recirculation Pump (2P-171A/B)
- Electric Heater for recirculation heating (2M-114A/B)
- Temperature Control Valve (2TCV-2921/2971)
- Full Flow Lube Oil Filter (2F-68A/B)
- Three-Element Full-Flow Strainer (2F-69A/B), and
- Lube Oil Sump/Crankcase

The engine driven Lube Oil Pump is mounted at the control end of the engine, and is driven by the lower crankshaft through gears and a flex drive. The main circulating oil pump is a rotary gear type pump which maintains the required oil flow while the engine is running. Pump discharge pressure is controlled by a relief valve. Normal full power gage board readings are as shown.



The pump takes suction from the engine sump through a suction strainer and discharges through the full flow oil filter to the thermostatic bypass valve, where a portion of the oil flow is directed through the shell side of the lube oil cooler to maintain the required oil operating temperature. After the two flow paths again join at the cooler outlet, the lube oil passes through the lube oil strainer and into the engine main oil headers for distribution to the various engine components. A portion of the lube oil is circulated to the underside of the piston crown through drilled passages in the connecting rods to assist in cooling of the pistons. Normal full power gage board indications are as shown.

Lube Oil cleanliness is maintained by the full flow Lube Oil Filter, and a full flow Lube Oil Strainer.

The full flow Lube Oil Filter (2F-68A/B) is equipped with a relief valve that bypasses the cartridge at 20 psid.

The 3 element Lube Oil Strainer (2F-69A/B) removes particles 81 microns or larger before oil is delivered to engine components.

opening. The annunciator corrective action procedure contains detailed requirements for a low temperature alarm (< 105° F).

The engine Pre-lube pump (2P-170A/B) is a motor driven pump that is used to pre-lube the engine prior to starting. During the pre-lube the upper and lower lube oil headers are filled to the most remote bearing. The length of time this takes is observed and timed during the 18 month PM on the engines. (A typical pre-lube time would be 2½ minutes). Operations runs the pump prior to starting the engine for at least the time frame measured during performance of the last PM.

If the Pre-lube pump is run too long or the engine is not started within 5 minutes of pre-lube, the engine should be rolled to kick lube oil out of the upper pistons. The concern is that excess oil collected in the upper pistons could drain into the cylinder and cause damage to the engine if the piston compresses a liquid.

The Pre-lube oil pump can also be used to fill the strainer and filter after an engine oil change. The pump is run long enough to see a slight indication on the strainer inlet pressure gauge.

There are three alarms directly associated with the Lube Oil System. They are:

ALARM	CAUSE	COMMENT
1/3 Lube Oil Press Lo	Lube Oil pressure ≤ 23 psig on any 1 of 3 pressure switches after **both** of the following: Engine speed reaches 810 rpm **AND** Jacket Water pressure reaches 8 psig	Two of Three pressure switches sensing low oil pressure will trip the EDG. Clear the alarm requires depressing the Engine Shutdown Relay Rest push-button on the local control panel
Lube Oil Temp Hi	Lube Oil temperature from engine ≥ 225°F	Reduce temperature to < 225°F to clear alarm
Lube Oil Temperature Lo	Lube Oil temperature ≤ 105°F	To clear alarm raise Lube Oil temperature > 105°F

OP-2104.036, EDG Operations has multiple Notes, Limits and Precautions related to an engine condition caused by operating the engine at low loads called “souping”. The piston rings in diesel engines are selected and designed to pass an adequate amount of oil for lubrication at rated load. At light loads (<20% load) the substantially lower combustion pressures and temperatures result in excess oil passing the piston rings into the combustion chamber which accumulates in the exhaust system and air box. Indications of souping are a bluish-gray fog like smoke in the exhaust. Extended periods of souping can result in buildup of unburned fuel/lube oil and combustion products on EDG exhaust screens (i.e. carbon fouling of turbo inlet screens) and has the potential for causing an exhaust manifold fire if not periodically cleared out by raising load. Correction of a souping condition can be achieved by loading the affected EDG to at least 40% of full load for at least 30 minutes to clear the exhaust. IER L2-11-46 documented

souping causing frequency oscillations on an EDG which had been lightly loaded (500 KW) for 5 days. After the EDG was loaded in 100 KW increments to full load and operated for 30 minutes frequency stabilized.

ORIGINAL QUESTION 2118 WAS USED TO DEVELOP 2017 NRC EXAM QID#29 (Used on 2014-2 NRC Exam)

Questions For All QID In Exam Bank

Bank:	2118	**Rev:**	1	**Rev Date:**	7/10/2014 2:39:40	**QID #:**	46	**Author:**	foster
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC BANK QID #1161		
Search	076000K307	**10CFR55:**	41.7 / 45.6	**Safety Function**	4				
System Title:	Service Water System (SWS)	**System Number**	076	**K/A**	K3.07				
Tier:	2	**Group:**	1	**RO Imp:**	3.7	**SRO Imp:**	3.9	**L. Plan:**	A2LP-RO-ESPTA
OBJ									

Description: Knowledge of the effect that a loss or malfunction of the SWS will have on the following: - ESF loads

Question:

Following a Reactor Trip, a check of the Vital Auxiliaries Safety Function status indicates that:

- #1 Emergency Diesel Generator (2DG1) is Running
- 2DG1 Frequency is 60 Hz
- 2DG1 Voltage is 4165 V
- 4160V Vital AC Bus 2A3 is Deenergized
- Annunciator 2K08-B3 "2A3 LO RELAY TRIP" has actuated

2DG1 should be secured to prevent ____ .

A. overheating the diesel engine

B. uneven lower crankshaft bearing wear

C. buildup of unburned fuel in the exhaust manifold

D. running the diesel engine without Jacket Cooling Water flow

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☒

Answer:

A. Correct

Notes:

A. Correct: with a lockout on the vital AC bus (2A3) the service water pump would be deenergized and overheating of the engine is a concern.

B. Incorrect: uneven lower crankshaft bearing wear due to the engine being run unloaded could be a concern but does not required an engine shutdown

C. Incorrect: buildup of unburned fuel in the engine exhaust manifold during engine low load operations is a concern but does not required an engine shutdown

D. Incorrect: Jacket Cooling Water has a engine driven pump which will maintain Jacket Cooling Water flow to the EDG therefore no requirement to secure EDG

References:

OP-2202.001, Standard Post Trip Actions, Rev 014, step 5.G contingency action page 5 of 19
EOP-2202.001,Standard Post Trip Actions Tech Guide, Rev 012, step 2 .H page 11 of 42
Lesson Plan A2LP-RO ESPTA, Introduction to EOPs and Standard Post Trip Action, Objective 11: Describe the major actions taken during the performance of SPTA and the basis for each.

Historical Comments:

NRC BANK QID #1161 not previously used on an NRC Exam
Rev 1 truncated distracters to remove "due to" statement and added "diesel" based on NRC review comments. Rearranged distracters for balance after editing. Cms 7-10-14

Question 30

Bank: 2360	**Rev:** 0	**Rev Date:** 7/18/2016	**2017 TEST QID #:** 30	**Author:** Foster					
Lic Level: RO	**Difficulty:** 2	**Taxonomy:** F	**Source:** NRC Exam Bank 2105						
Search 008000K202	**10CFR55:** 41.7	**Safety Function** 8							
Title: Component Cooling Water System (CCWS)	**System Number** 008	**K/A** K2.02							
Tier: 2	**Group:** 1	**RO Imp:** 3.0	**SRO Imp:** 3.2	**L. Plan:** A2LP-RO-CCW	**OBJ** 2				

Description: Knowledge of bus power supplies to the following: - CCW pump, including emergency backup

Question:

Loop 2 Component Cooling Water (CCW) is normally supplied by 2P-33C, "C" Component Cooling Water (CCW) pump, which is powered from ________ and its backup pump, 2P-33B, "B" CCW pump, is powered from ________ .

A. 2B-1; 2B-2

B. 2B-2; 2B-1

C. 2B-2; 2B-7

D. 2B-7; 2B-2

Answer:

D. 2B-7; 2B-2

Notes:

D is Correct:: "C" CCW pump is powered by 480v bus 2B-7; "B" CCW pump is powered 480v bus 2B-2

A is Incorrect: 2B-1 powers 2P-33A and 2B-2 powers 2P-33B but plausible as these are correct CCW Pump Power supplies.

B is Incorrect: 2B-2 powers 2P-33B and 2B-1 powers 2P-33A but plausible as these are correct CCW Pump Power supplies.

C is Incorrect: 2B2 powers 2P-33B and 2B-7 powers 2P-33C but plausible as these are correct CCW Pump Power supplies for the pumps listed.

This question matches the K&A because the knowledge of the normal and backup pump power supplies is needed to correctly answer the question.

References:

STM_2-43_15-1 CCW SYS Section 2.1 (Verified reference updated 11/15/16).

Historical Comments:

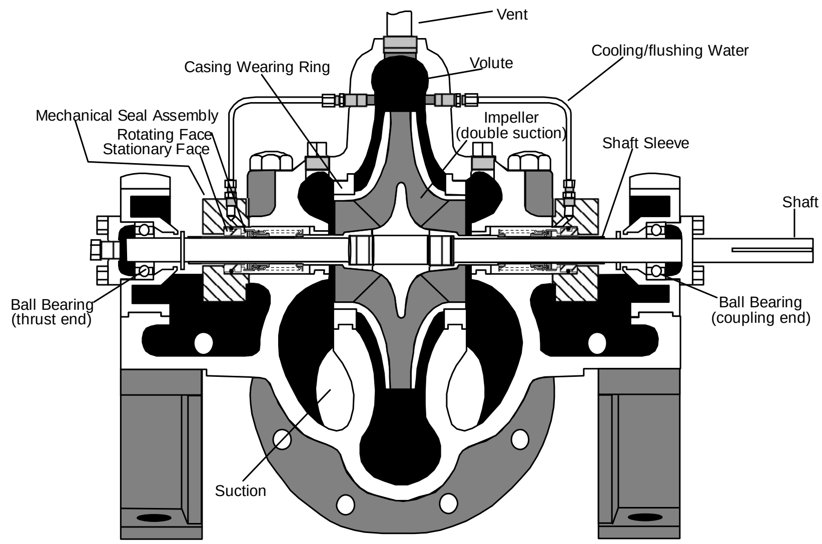
NRC Exam Bank 2105 was used on the 2014-2 NRC Exam.
To be used on the 2017 NRC Exam

2.0 Detailed System Description

2.1 CCW Pumps

There are three Component Cooling Water (CCW) pumps. These pumps are designated 2P-33A, 2P-33B, and 2P-33C. Each is a horizontal, single stage, double suction, centrifugal pump rated at 2900 gpm.

As can be seen in the cutaway view of a CCW pump below, flow into the pump is from the suction piping. At the pump the suction flow splits and enters both sides of the impeller. This is why the pumps have the “double suction” designation. The advantage of the double suction impeller design is that it eliminates a lot of the axial thrust on the pump shaft.



Component Cooling Water Pump
(Double Suction, Single Stage)

The pump shaft is sealed with a mechanical seal vice a conventional packing. The mechanical seal design provides zero leakage along the shaft preventing leakage of potentially contaminated fluid. If any leakage is observed from these seals, they should be replaced.

The CCW pumps are driven by a 480V, 200 HP motor. The power supplies to each of the pumps are as follows:

- 2P-33A is powered from 2B1-23 with dc control power coming from 2D21 breaker 7,
- 2P-33B is powered from 2B2-23 with dc control power coming from 2D2 breaker 7, and
- 2P-33C is powered from 2B7-21 with dc control power coming from 2D21 breaker 10.

All of the pumps are controlled by handswitches located on 2C14 in the Control Room.

Questions For All QID In Exam Bank

ORIGINAL BANK QUESTION 2105 WAS USED TO DEVELOP 2017 NRC EXAM QID#30 (Used on 2014-2 NRC Exam)

Historical Comments:

NRC BANK QID #0476 used on 2005 NRC Exam
Rev 1 for NRC review comments. Fixed formatting for printed view. Fixed inconsistent use of abbreviations for "pressurizer" and "quench tank" in distracters. Removed "slightly" from 'D'. Cms 7-3-14

Bank: 2105 **Rev:** 1 **Rev Date:** 7/3/2014 6:24:09 P **QID #:** 33 **Author:** foster

Lic Level: R **Difficulty:** 2 **Taxonomy:** F **Source:** NEW

Search 008000K202 **10CFR55:** 41.7 **Safety Function** 8

System Title: Component Cooling Water System (CCWS) **System Number** 008 **K/A** K2.02

Tier: 2 **Group:** 1 **RO Imp:** 3.0 **SRO Imp:** 3.2 **L. Plan:** A2LP-RO-CCW **OBJ** 2

Description: Knowledge of bus power supplies to the following: - CCW pump, including emergency backup

Question:

Loop 2 Component Cooling Water (CCW) is normally supplied by 2P-33C, "C" Component Cooling Water (CCW) pump, which is powered from ________ and its backup pump, 2P-33B, "B" CCW pump, is powered from ________ .

A. 2B-1; 2B-2

B. 2B-2; 2B-1

C. 2B-2; 2B-7

D. 2B-7; 2B-2

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☒

Answer:

D. Correct

Notes:

D. Correct: "C" CCW pump is powered by 480v bus 2B-7; "B" CCW pump is powered 480v bus 2B-2
A. Incorrect: 2B-1 powers 2P-33A and 2B-2 powers 2P-33B
B Incorrect: 2B-2 powers 2P-33B and 2B-1 powers 2P-33A
C. Incorrect: 2B2 powers 2P-33B and 2B-7 powers 2P-33C

References:

STM 2- 43, Component Cooling Water System, Rev 14, section 2.1 page 2
Lesson Plan A2LP-RO-CCW objective 2: Describe the CCW pumps to include their power supplies and operation.

Historical Comments:

Rev 1. No NRC comments on question. Fixed typo in Notes for which selection is correct. Cms 7-3-14.

Question 31

Bank:	2361	**Rev:**	0	**Rev Date:**	7/25/2016	**2017 TEST QID #:**	31	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NRC Exam Bank 1756		
Search	062000A305	**10CFR55:**	41.7	**Safety Function**	6				
Title:	A.C. Electrical Distribution System	**System Number**	062	**K/A**	A3.05				
Tier:	2	**Group:**	1	**RO Imp:**	3.5	**SRO Imp:**	3.6	**L. Plan:**	A2LP-RO-ECCS
OBJ	10								

Description: Ability to monitor automatic operation of the A.C. Distribution System, including: - Safety-related indicators and controls

Question:

Given the following at full power:

* Tags have been cleared on the "C" HPSI Pump, 2P-89C, Breaker 2A-407.
* The breaker has been racked up with the following indications on 2C-16:

Green light is ON
White light is OFF
Red Light is OFF
Amber light is ON

Based on these indications, the 2P-89C Green Train Breaker, 2A-407, closing springs are _____________ and the Kirk Key lock is _____________________.

A. charged; locked out/NOT available

B. charged; unlocked/available

C. discharged; locked out/NOT available

D. discharged; unlocked/available

Answer:

C. discharged; locked out/NOT available

Notes:

Four indicating lights are located directly above the handswitch for 2P-89C on 2C-16. The GREEN light On indicates the pump power supply breaker is open. The RED light On indicates the pump power supply breaker is closed. The WHITE light On indicates the closing spring for the pump controller breaker is charged. The AMBER light On indicates that the breakers is LOCKED OUT by the Kirk Key for train separation as 2P-89C is the swing HPSI Pump.

C is correct as the White light is OFF and the Amber light is ON.

A is incorrect because the White light is OFF but plausible as the AMBER Light is ON indicating the Kirk-Key is Locked.

B is incorrect as the White light is OFF and the Amber Light is ON but plausible if the candidate does not know the correct meaning of the light status indications.

D is incorrect as the Amber light is ON but plausible as the White light is OFF indicating charging springs are discharged.

This question matches the K&A as it requires the candidate to have the ability to understand the breaker indication status lights for the 4160 VAC breaker that supplies the Swing HPSI Pump 2P-89C and relate these light indications to the status of the controls for 2P-89C HPSI Pump.

References:

STM_2-05_30-1 ECCS Section 3.6.; (Verified reference updated 11/15/16).
STM_2-05_30-1 ECCS Section 3.6.2. (Verified reference updated 11/15/16).

Historical Comments:

NRC Exam Bank 1756 was used on the 2011 NRC exam
To be used on the 2017 NRC Exam but added available/NOT available to clarify indication.

3.4 HPSI Relief Valves

Relief valves 2PSV-5110 and 2PSV-5111 provide over pressure protection to the high pressure headers which could be caused by a temperature rise in isolated piping. These valves, located between the pump discharge check valves (2SI-10A & B) and the header flow detectors, are set at 1950 psi and relieve to the boron management system (BMS) holdup tanks.

Another relief valve (2PSV-5112) installed in header #1 downstream of check valve 2SI-12 protects against over pressure which could occur when using the high pressure header as a flow path from the charging pumps to the RCS. It is set at 2485 psig and also relieves to the BMS holdup tanks.

3.5 HPSI Header Shutoff Valves

The four HPSI headers are required to provide balanced flow to the RCS to mitigate the effects of a break in one of the injection paths. All of the HPSI header shutoff valves are of the same type. The HPSI header shutoff valves are 2CV-5015-1, 2CV-5016-2, 2CV-5035-1, 2CV-5036-2, 2CV-5055-1, 2CV-5056-2, 2CV-5075-1, and 2CV-5076-2.

The header shutoff valves are 2” Anchor/Darling globe valves. These motor operated injection valves open fully upon actuation, relying on manual throttle valves affixed in their throttled position for system flow balance.

The power supplies for the HPSI header shutoff valves are:

Unit:	Power Supply:
2CV-5015-1	2B51-J3
2CV-5016-2	2B61-K1
2CV-5035-1	2B52-F3
2CV-5036-2	2B62-F3
2CV-5055-1	2B51-J2
2CV-5056-2	2B61-J2
2CV-5075-1	2B52-F4
2CV-5076-2	2B62-F4

3.6 HPSI Controls and Interlocks

3.6.1 HPSI Pumps 2P-89A/B

The HPSI pump remote control switches 2HS-5078-1 and 2HS-5079-2 with their associated indicating lights are located on control room panels 2C17 and 2C16 respectively. The switches are spring return to normal with START, NORMAL AFTER START, NORMAL AFTER STOP, STOP and PULL TO LOCK positions. The mechanical indicator on the switch shows a red flag in START and NORMAL AFTER

START, a green flag in STOP and NORMAL AFTER STOP, and a black flag in the PULL-TO-LOCK position.

Three indicating lights are located directly above the handswitch. The GREEN light indicates the pump power supply breaker is open. The RED light indicates the pump power supply breaker is closed. The WHITE light indicates the closing spring for the pump controller breaker is charged. The white light also shows circuit continuity through the closing portion of the control circuit. For the “C” HPSI pump an amber light is also provided above its handswitch. When lit this amber light indicates its power supply breaker is NOT available.

The handswitch at the 2A-3 and 2A-4 4160 volt switchgear (152/CS) is a three position, return to center switch with CLOSE, CENTER, and TRIP positions.

The handswitch contacts from 2P-89A and 2P-89B are used in the automatic start circuit of 2P-89C to prevent 2P-89C from starting on the same bus when 2P-89A and 2P-89B are available. Therefore, 2P-89C may be in NORMAL AFTER STOP (out of PULL TO LOCK) without entering a technical specification action.

The pumps can be started manually from the handswitch in the control room or from the 4160 volt switchgear. A HPSI pump start logic drawing is provided on page 78 for HPSI pumps “A” and “B”. The pumps will start with a 10 second time delay upon receipt of a SIAS if the following conditions are met:

- the control room handswitch is NOT in PULL-TO-LOCK

  AND
- pump 2P-89C is NOT running

  AND
- power is available from either the normal power (2A-3/2A-4) OR from the emergency diesel generator.

  AND
- no undervoltage exists on 2A-3 / 2A-4

If the pump is started manually or automatically, the following electric interlocks are effective:

- the motor space heaters will be interlocked OFF and
- permissives are supplied for 2VUC-1A and 1B room coolers to start. The cooler fans will start after a 70 second time delay. If air flow is not established within 20 seconds of the cooler fan start, a trouble alarm annunciates in the control room.

If an SIAS occurs and the HPSI pump breaker does not close within 30 seconds, a control room annunciator FAILURE ON ESFAS AUTO START occurs. Another annunciator occurs indicating the pump is INOPERABLE if the control room handswitch is placed in the PULL-TO-LOCK position.

If the handswitch is in the START or the NORMAL AFTER START position and the breaker opens, a "b" contact in the breaker actuates the control room annunciator HPSI PUMP BREAKER TRIP. Another alarm, MOTOR OVERLOAD, is actuated by the 151 relay in

the 4160 volt switch gear for a high current condition in phase “B” of the motor.

A logic drawing for stopping HPSI pumps “A” and “B” is provided on page 77. The HPSI pump will stop if the following conditions are met:

- the control switch at 2A-3 or 2A-4 is placed in TRIP

  OR

- if the handswitch in the control room is placed in either STOP or PULL-TO-LOCK

  OR

- if bus voltage drops below a setpoint of 2300 ± 699 volts

  OR

- an electrical fault occurs which opens the motor breaker.

The power supplies for the “A” and “B” HPSI pumps are:

Unit:	Power Supply:
2P-89A	2A-306
2P-89B	2A-406

3.6.2 HPSI Pump 2P-89C

The difference between the logic and interlocks for this pump and the 2P-89A and 2P-89B pumps is that this pump can be powered from two different sources. Handswitch 2HS-5080-1 on control room panel 2C17 will provide power from 2A-3 bus. Handswitch 2HS-5080-2 on control room panel 2C16 will provide power from 2A-4 bus. The control switches on the 4160 switchgear for 2P-89C have kirk key interlocks to prevent both switches from being closed at the same time. Amber lights on 2C17 and 2C16 indicate if the breakers are LOCKED OUT. The handswitches on 2C17 and 2C16 are normally in the PULL-TO-LOCK position. However, 2P-89C is electrically interlocked so that it can NOT be automatically started unless the HPSI pump in the train to which 2P-89C is aligned is in PULL TO LOCK. Note that this interlock does NOT prevent the manual start of the “C” HPSI pump.

A logic drawing for starting HPSI pump “C” is provided on page 79. The 2P-89C pump will start automatically from 2A-3 (or 2A-4) respectively if the following conditions are met:

- 2P-89A (or B) is in PULL-TO-LOCK

  AND

- the associated 2P-89C handswitch is NOT in PULL-TO-LOCK

  AND

- SIAS #1 or #2 occurs depending on what loop the pump is aligned to

  AND

- power is available through the normal feeder breakers to its normal buses (2A-3 or 2A-4) OR from the diesel generator.

ORIGINAL BANK QUESTION 1756 WAS USED TO DEVELOP 2017 NRC EXAM QID#31 (Used on 2011 NRC Exam)

Questions For All QID In Exam Bank

Bank:	1756	**Rev:**	0	**Rev Date:**	10/15/2010 9:33:5	**QID #:**	47	**Author:**	Coble		
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	IH Exam Bank OPS2-3655				
Search	062000A402	**10CFR55:**	41.8	**Safety Function**	6						
System Title:	A.C. Electrical Distribution System	**System Number**	062	**K/A**	A4.02						
Tier:	2	**Group:**	1	**RO Imp:**	2.5	**SRO Imp:**	2.8	**L. Plan:**	A2LP-RO-ECCS	**OBJ**	10

Description: Ability to manually operate and/or monitor in the control room: - Remote racking in and out of breakers

Question:

Given the following at full power:

* Tags have been cleared on the "C" HPSI Pump, 2P89C, Breaker 2A407.
* The breaker has been racked up with the following indications on 2C-16:

Green light is ON
White light is OFF
Red Light is OFF
Amber light is ON

Based on these indications, the 2P-89C Green Train Breaker, 2A407, closing springs are _____________ and the Kirk Key lock is ____________.

A. charged; locked

B. charged; unlocked

C. discharged; locked

D. discharged; unlocked

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☒

Answer:

C. discharged; locked

Notes:

Four indicating lights are located directly above the handswitch for 2P-89C. The GREEN light indicates the pump power supply breaker is open. The RED light indicates the pump power supply breaker is closed. The WHITE light indicates the closing spring for the pump controller breaker is charged. The AMBER light on 2C16 indicate that the breakers is LOCKED OUT by the Kirk Key for train separation as 2P-89C is the swing HPSI Pump. Distracters A and B are incorrect because the Springs are discharged. Distracters B and D are incorrect because the Kirk Key is locked.

References:

STM 2-05, ECCS, Revision 22, Section 3.6

Historical Comments:

Has never been used on an ANO-Unit 2 NRC Exam.

Question 32

Bank:	2362	**Rev:**	2	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	32	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	103000K108	**10CFR55:**	41.7	**Safety Function**	5				
Title:	Containment System	**System Number**	103	**K/A**	K1.08				
Tier:	2	**Group:**	1	**RO Imp:**	3.6	**SRO Imp:**	3.8	**L. Plan:**	A2LP-RO-ESFAS
OBJ	2								

Description: Knowledge of the physical connections and/or cause-effect relationships between the Containment System and the following systems: - SIS, including action of safety injection reset

Question:

During a LOCA, if Containment pressure rises to a MINIMUM of __________ psia, then a SIAS signal is generated and the SIAS cannot be reset in accordance with Standard Attachment 13, SIAS RESET, until Containment Pressure drops below a MAXIMUM of __________ psia.

A. 23.3; 23.0

B. 23.3; 22.5

C 18.3; 18.0

D. 18.3; 17.5

Answer:

D. 18.3; 17.5

Notes:

D is correct as the SIAS is required to be set at less than or equal to 18.3 and the Standard Attachment 13, SIAS RESET, has a note at the beginning stating that Containment Pressure must be 17.5 psia or less to reset the SIAS. This 0.8 psia margin ensures a good probability that the SIAS will not actuate again.

A is incorrect but plausible as this is the CSAS actuation setpoint but the reset pressure margin is too small.

B is incorrect but plausible as this is the CSAS actuation setpoint and the correct reset criteria pressure. For CSAS

C is incorrect but plausible as this is the correct SIAS setpoint but the reset pressure margin is too small.

This question meets the K&A as the candidate must know the cause-effect relation ship between containment pressure and the SIAS generation along with when to take action to reset the SIAS.

References:

STM_2-70_19-1 ESFAS ACTUATION TABLE (Verified reference updated 11/15/16);
EOP 2202.003 LOCA Rev. 15 Section 2 Step 17 (Verified reference updated 11/15/16);
EOP 2202.010 Standard Attachments Rev. 23 Attachment 13 SIAS RESET (Verified reference updated 11/15/16); EOP 2202.003 LOCA Rev. 15 Section 2 Step 19 (Verified reference updated 11/15/16);

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Added "Minimum/Maximum" to the stem to eliminate potential subset issues.

REV. 2 based on NRC Chief Examiner Feedback BNC. Capitalized the words minimum and maximum prior to the blanks in the stem.

ESFAS ACTUATION DESCRIPTION TABLE			
ACTUATION	**PARAMETER**	**SETPOINT (Note 1)**	**PURPOSE**
SIAS	PZR Pressure	≥1650 psia with Variable setpoint. (Ensures actuation at or before reaching 1650 psia)	Provide cooling to limit core damage and assure adequate shutdown margin regardless of temperature, during a LOCA, MSLB, or SGTR.
	Containment pressure	≤18.3 psia (Ensures actuation at or before reaching 18.3 psia)	
CIAS	Containment pressure	≤ 18.3 psia (Ensures actuation at or before reaching 18.3 psia)	Prevent the release of radioactive material during a LOCA or MSLB.
CSAS	Containment pressure	≤23.3 psia (Ensures actuation at or before reaching 23.3 psia)	Remove heat and iodine from the containment to hold containment temperature and pressure below design values during and following a LOCA or MSLB in the containment.
RAS	RWT level	6.0 % ± .5%	Provide continuous long term post accident core cooling following a LOCA.
CCAS	PZR Pressure	≥1650 PSIA with Variable setpoint (Ensures actuation at or before reaching 1650 psia)	Limit post accident containment pressure below design values during and following a LOCA or MSLB in the containment.
	Containment pressure	≤18.3 psia (Ensures actuation at or before reaching 18.3 psia)	
MSIS	Steam Generator pressure	≥751 psia with Variable setpoint (Ensures actuation at or before reaching 751 psia)	Enable rapid termination of steam and feedwater flow if a steam generator line breaks or a steam generator tube ruptures.
EFAS	Steam Generator level	22.2% Narrow range level	Provide the intact steam generator with sufficient feedwater for cooling during and following a MSLB or Loss of Feedwater accident.
	Steam Generator pressure	≥751 psia with Variable setpoint (Ensures actuation at or before reaching 751 psia)	
	Steam Generator differential pressure	≤90 psid between the two steam generators (Ensures actuation at or before reaching 90 psid)	

***Note 1** – These are the values required by Tech Specs. A setpoint of ≥1650 psia ensures that an actuation will occur upon reaching or prior to reaching 1650 psia. A setpoint of ≤23.3 psia ensures that a CSAS will occur upon reaching or prior to reaching 23.3 psia on rising pressure. The other setpoint requirements follow similar logic.*

INSTRUCTIONS	CONTINGENCY ACTIONS
15. Notify Chemistry to perform the following: A. Coordinate with Radiation Protection to perform surveys during sampling. B. Sample RCS for the following: • Iodine 2 to 6 hours following Reactor Trip from greater than 15% power. • Boron. C. WHEN requested by Chemistry, THEN align RCS sample using 2104.002 Exhibit 3.	
***16. Maintain shutdown margin using 2202.010 Attachment 28, Boric Acid Required for Shutdown Margin.**	
17. Reset SIAS as follows: A. Check adequate shutdown margin established using 2202.010 Attachment 28, Boric Acid Required for Shutdown Margin. B. Reset SIAS using 2202.010 Attachment 13, SIAS Reset. C. Restore SIAS actuated components and systems as needed.	**17. IF SIAS can NOT be reset, THEN GO TO Step 18.** A. Perform 2202.010 Attachment 35, Boric Acid Alignment for Cooldown.
18. Check SM determines Plant cooldown required.	**18. IF Plant cooldown NOT required, THEN perform the following:** • Maintain stable plant conditions. • Consult TSC or Operations Management for further guidance. • Do NOT continue.

ATTACHMENT 13
SIAS RESET

Page 1 of 5

NOTE

Following actuation CNTMT pressure must be 17.5 psia or less to reset CIAS and SIAS.

1. IF CIAS NOT reset,
   THEN perform the following:

   A. Verify CNTMT pressure < 17.5 psia.

   B. Place LK/UNLK switch to UNLK (key No. 15).

   C. Depress CIAS pushbutton on 2C23.

   D. Verify Trip Path lights reset on local PPS Status panels. (See Figure 1 on page 5)

   E. Place LK/UNLK switch to LK and remove key.

   F. Repeat above steps to reset remaining Actuation Trip Paths.

   G. Depress either CIAS Lockout Reset pushbutton on 2C40
   AND verify both Reset lights ON.

   H. Depress either CIAS Lockout Reset pushbutton on 2C39
   AND verify both Reset lights ON.

NOTE

Resetting the SIAS signal will realign CCP Suction valves to their pre-SIAS positions (no flyback protection).

2. Verify CCP Suction source will be available after SIAS reset.

3. IF RCS pressure greater than variable setpoint AND CNTMT pressure less than 17.5 psia,
   THEN **GO TO** Step 5.

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

Continued CNTMT Spray operation may be desirable to reduce offsite doses from airborne activity in CNTMT.

***19. IF CNTMT Spray operating, THEN terminate CNTMT Spray as follows:**

A. Check CNTMT Spray termination criteria satisfied:

- CNTMT pressure less than 22.5 psia.
- CNTMT temperature less than 140°F.
- Verify ALL available CNTMT Cooling fans running in Emergency Mode using 2202.010 Exhibit 9, ESFAS Actuation.
- TSC determines CNTMT Spray NOT required for CNTMT Iodine removal.
- CNTMT Spray NOT required for decay heat removal following RAS actuation.

Contingency: A. IF CNTMT Spray termination criteria NOT satisfied, THEN **GO TO** Step 20.

B. Reset CSAS using 2202.010 Attachment 45, CSAS Reset.

Question 33

Bank:	2363	**Rev:**	2	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	33	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	Modified NRC Exam Bank 1623		
Search	012000A207	**10CFR55:**	41.10	**Safety Function**	7				
Title:	Reactor Protection System	**System Number**	012	**K/A**	A2.07				
Tier:	2	**Group:**	1	**RO Imp:**	3.2	**SRO Imp:**	3.7	**L. Plan:**	A2LP-RO-A125V
OBJ	0								

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the RPS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Loss of dc control power

Question:

Given the following:

* The plant is at full power.
* Annunciator 2K01-D11 "BATTERY 2D12 NOT AVAIL" comes into Alarm.
* Annunciator 2K01-A11 "CONT CENTER 2D02 UNDER VOLT" comes into alarm.
* 2D02 Bus Voltage indicates "0 VDC" on SPDS.

Given these conditions, which of the following list the correct procedure to be used to mitigate the consequences of this event on the PPS system and the correct action to take on the PPS System?

A. Inadvertent SIAS OP-2203.018; Place PPS Channels 1 OR 3 in Bypass.

B. Inadvertent SIAS OP-2203.018; Place PPS Channels 2 OR 4 in Bypass.

C. Loss of 125V DC OP-2203.037; Place PPS Channels 1 OR 3 in Bypass.

D. Loss of 125V DC OP-2203.037; Place PPS Channels 2 OR 4 in Bypass.

Answer:

D. Loss of 125V DC AOP-2203.037; Place PPS Channels 2 OR 4 in Bypass

Notes:

D is correct. The Loss of 125V DC AOP Section 3 Step 4 will direct the operator to bypass one channel of PPS. A loss of 2D02 accompanied by a reactor trip without one of the green PPS channels in Trip Channel Bypass, will result in inadvertent ESFAS actuations of Red Train components. By placing one channel in bypass if a reactor trip occurs then the inadvertent actuations will not occur.

A is incorrect as the AOP directions are not in the Inadvertent SIAS AOP 2203.018 and Channels 1 and 3 of PPS are supplied form the Red train Inverters which are powered from Battery 2D11 but plausible as the goal of placing the PPS channel in bypass is to prevent inadvertent starts of ESF Pumps normally started on an SIAS.

B is incorrect as the AOP directions are not in the Inadvertent SIAS AOP 2203.018 but plausible as the goal of placing the PPS channel in bypass is to prevent inadvertent starts of ESF Pumps normally started on an SIAS and only 1 channel either channel 2 or 4 should be bypassed.

C. is incorrect because and Channels 1 and 3 of PPS are supplied form the Red train Inverters which are powered from Battery 2D11 but plausible as the actions for mitigating the consequences of the event on RPS are found in the Loss of 125V DC AOP.

This question matches the K&A as it requires the ability to understand the potential impact of the event on the RPS system and select the correct procedure to mitigate the consequences of the event on the RPS system.

References:

2203012A Rev. 47 ALARM 2K01 CORRECTIVE ACTION Widow D-11 BATTERY 2D12 NOT AVAIL (Verified reference updated 11/15/16); 2203012A Rev. 47 ALARM 2K01 CORRECTIVE ACTION Widow A-11 CONT CENTER 2D02 UNDERVOLT (Verified reference updated 11/15/16);
Loss of 125 VDC AOP 2203.037 Rev. 13 Section 3 Green Train DC Step 5; (Verified reference updated 11/15/16); Loss of 125 VDC AOP 2203.037 TG Rev. 13 Section 3 Green Train DC Step 5 (Verified reference updated 11/15/16).

Historical Comments:

NRC Exam Bank 1623 was used on the 2009 NRC Exam
To be used on the 2017 NRC Exam but modified for the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Deleted "2 AND 4" in distractors A and C and replaced with "3 OR 4" . Applicant will now need to recall the power supply to the Green Train Inverters and RS panel supplies to the ESFAS cabinets. Added clarifying words to the

REV. 2 based on NRC Chief Examiner Feedback BNC. Changed "3 OR 4" to "1 OR 3" in distractors A and C. Changed RPS to PPS in the stem. Also updated distractor analysis.

ANNUNCIATOR 2K01

D-11

BATTERY 2D12 NOT AVAIL

1.0 CAUSES

1.1 2D-12 Battery Disconnect (2D-52) open.

1.2 Battery fuse in Fuse and Relay Cabinet (2D-42) blown or removed.

2.0 ACTION REQUIRED

2.1 IF Battery NOT connected to bus (i.e. disconnect open or fuse blown), THEN perform the following:

2.1.1 Declare Emergency Diesel Generator (2DG2) inoperable. (CR-ANO-C-2003-00877)

- Refer to Tech Specs 3.8.1.1, 3.8.1.2 and 3.4.4.
- Notify Unit 1 to refer to appropriate Tech Specs.

2.1.2 Enter 24 AOT for 2Y22/24/2224 operation without battery backup. Refer to Supplement 4 of Electrical System Operations (2107.001).

2.2 IF Battery Disconnect open, THEN perform the following:

2.2.1 Determine reason for disconnect switch being open.

2.2.2 WHEN appropriate, THEN close disconnect.

2.3 IF fuse blown or removed, THEN perform the following:

2.3.1 Determine cause of blown fuses.

2.3.2 WHEN cause repaired or isolated, THEN replace fuses in Fuse and Relay Cabinet (2D-42).

2.4 Refer to Tech Specs 3.8.1.1, 3.8.1.2, 3.8.2.3 and 3.8.2.4.

2.5 Refer to Control Room Emergency Air Conditioning and Ventilation, 2104.007, Attachment B (Component/Tech Spec Cross-Reference) for Unit 1 and Unit 2 CREVS/CREACS TS/TRM applicability.

(D-11 Continued on next page)

ANNUNCIATOR 2K01

A-11

CONT CENTER 2D02 UNDERVOLT

1.0 CAUSES

1.1 2D02 bus voltage less than or equal to 110 VDC (relay 27-2D02).

2.0 ACTION REQUIRED

2.1 Check 2D02 voltage on Computer Point (E2D02).

2.2 Check the following for Battery bank (2D-12) on Fuse and Relay Cabinet (2D-42):

- Amperage (nominal -10 to +10 amps)
- Voltage (nominal 125 to 135 volts)

2.3 Check the following for 2D-12 Battery charger (2D-32A or 2D-32B):

- A301 DC Output Current (nominal 100 to 150 amps)
- V301 DC Output Voltage (nominal 120 to 140 volts)

2.4 IF battery charger amps high (current limited to 400 amps)
AND 2D-12 is discharging,
THEN secure unnecessary 2D02 loads.

2.5 Refer to Loss of 125 VDC (2203.037).

2.6 Check for overloads or multiple grounds (both positive and negative) on Fuse and Relay Cabinet (2D-42).

2.7 Refer to Tech Specs 3.8.2.3 and 3.8.2.4.

3.0 TO CLEAR ALARM

3.1 Raise bus 2D02 voltage above setpoint.

4.0 REFERENCES

4.1 E-2451-2A

INSTRUCTIONS | **CONTINGENCY ACTIONS**

5. **PERFORM** EITHER of the following:

   - **PLACE** ALL PPS points in bypass on 2C23-2 AND refer to the following:
     - TS 3.3.1.1
     - TS 3.3.2.1
     - TRM 3.3.1
   - **PLACE** ALL PPS points in bypass on 2C23-4 AND refer to the following:
     - TS 3.3.1.1
     - TS 3.3.2.1
     - TRM 3.3.1

6. **NOTIFY** Electrical Maintenance to determine cause and perform corrective maintenance to restore Vital Green Train DC.

7. Locally **ENSURE** Inverters 2Y22, 2Y24 and 2Y2224 shutdown as follows:

   A. **PLACE** "MANUAL BYPASS SWITCH" to ALTERNATE SOURCE position:
   - 2HS-9201 for 2Y22
   - 2HS-9801 for 2Y24
   - 2HS-9601 for 2Y2224

   B. **OPEN** "120 VAC INVERTER OUTPUT" (B-2) breaker.

   C. **OPEN** "125 VDC INPUT" (B-1) breaker.

AOP STEP - SECTION 3 (GREEN TRAIN DC):

4. **REFER TO** TABLE 3 of Attachment B, Tech Spec/Technical Requirements Manual Cross Reference, for a list of affected Tech Specs and Technical Requirements.

BASIS:

This step directs the operator to refer to an attachment that lists possible Technical Specification Action Statements/TRM statements that may require entry due to the loss of DC power.

SOURCE DOCUMENT:

1. ANO-1 Technical Specifications and Technical Requirements Manual
2. ANO-2 Technical Specifications and Technical Requirements Manual
3. LIC-01-017, ANO2 TS Am. 229, Relocation of Boric Acid Makeup Systems from TSs to TRM

AOP STEP - SECTION 3 (GREEN TRAIN DC):

5. **PERFORM** EITHER of the following:

BASIS:

This directs the operator to bypass one channel of PPS. A loss of 2D02 accompanied by a reactor trip without one of the green PPS channels in Trip Channel Bypass, will result in inadvertent actuations of Red Train components. This happens due to the loss of power to both 2RS2 and 2RS4. By placing one channel in bypass if a reactor trip occurs then the inadvertent actuations will not occur.

SOURCE DOCUMENTS:

1. STM-2-63, PPS/RPS

PARENT QUESTION 1623 WAS USED TO MODIFY 2017 NRC EXAM QID#33 (Used on 2009 NRC Exam)

Data for 2009 NRC RO/SRO Exam

Bank:	1623	**Rev:**	1	**Rev Date:**	7/24/2009 12:40:1	**QID #:**	14	**Author:**	Coble			
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	New					
Search	000058A202	**10CFR55:**	41.7	**Safety Function**	6							
System Title:	Loss of DC Power	**System Number**	058	**K/A**	AA2.02							
Tier:	1	**Group:**	1	**RO Imp:**	3.3	**SRO Imp:**	3.6	**L. Plan:**	A2LP-RO-EAOP	**OBJ**	27	

Description: Ability to determine and interpret the following as they apply to the Loss of DC Power: - 125V dc bus voltage, low/critical low, alarm

Question:

Given the following:

* The plant is at full power
* Annunciator 2K01 A-10 "CONT CENTER 2D01 UNDERVOLT" alarms
* The CRS has entered AOP 2203.037, Loss of 125 VDC, due to a loss of 2D01
* The Reactor has not tripped
* The CRS has given direction to place ALL PPS points on the appropriate PPS channel(s) in bypass

Which ONE of the following would be the appropriate PPS channel(s) to bypass?

A. PPS Channel 1 AND PPS Channel 2

B. PPS Channel 1 OR PPS Channel 3

C. PPS Channel 3 AND PPS Channel 4

D. PPS Channel 2 OR PPS Channel 4

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐
2009	☑	☑

Audit Exam History

2009	☐

Answer:

B. PPS Channel 1 OR PPS Channel 3

Notes:

Battery 2D01 supplies 125 VDC bus power to RPS channels 1 and 3 (Red 'A' Train). Interlocks in the PPS system only allow bypassing points on one RPS channel only to ensure the appropriate PPS trip logic is still valid. Distracter A is incorrect because it attempts to bypass more than one PPS channel and 2D01 does not supply power to Channel 2. Distracter C is incorrect because it attempts to bypass more than one PPS channel and 2D01 does not supply power to Channel 4. Distracter D is incorrect because 2D01 does not supply power to PPS Channel 2 or 4.

References:

AOP 2203.037, Loss of 125 VDC , Rev. 6 Section 2 Step 5
Tech Guide 2203.037, Loss of 125 VDC , Rev. 6 Section 2 Step 5
Annunciator Corrective Action (ACA) 2203.012A Alarm 2K01 A-10
Lesson Plan A2LP-RO-EAOP Rev. 12, Objective 27: Discuss the Mitigation strategy, Entry Conditions, applicable industry events, Instructions and Exit Conditions (as per AOP and Tech Guide) of OP 2203.037, Loss Of 125VDC.

Historical Comments:

Question 34

Bank:	2364	**Rev:**	0	**Rev Date:**	8/23/2016	**2017 TEST QID #:**	34	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	004000A410	**10CFR55:**	41.7			**Safety Function**	1		
Title:	Chemical and Volume Control System (CVCS)					**System Number**	004	**K/A**	A4.10
Tier:	2	**Group:**	1	**RO Imp:**	3.6	**SRO Imp:**	3.2	**L. Plan:**	A2LP-RO-CVCS
OBJ	16								

Description: Ability to manually operate and/or monitor in the control room: - Boric acid pumps

Question:

Given the following conditions:

* The plant is at 100% power.
* The CRS directs adding 10 gallons of boric acid to the RCS.
* BAM Pump Select Switch 2HS-4911-2 is selected to BAM Pump 2P-39A.
* The Boric Acid Flow Controller 2FIC-4926 has been set to a flowrate of 10 gpm.
* The Mode Select Switch 2HS-4928 is taken to "BORATE".
* The Boric Acid Batch Controller 2FQIS-4926 has been set to 10 gallons.

When the Mode Select switch is taken to BORATE, the BAM pump 2P-39A will be started __________ and when the 10 gallons of boric acid has been added to the RCS, 2P-39A will be stopped __________.

A. manually; manually

B. manually; automatically

C. automatically; manually

D. automatically; automatically

Answer:

C. automatically; manually

Notes:

C is correct as described in section 2.3.10.3 and Exhibit 1 of the Chemical Addition procedure.. When the mode selector switch is taken to BORATE, the selected BAM Pump will auto start and add the selected amount of Boric Acid but the pump will not auto stop when the flow batch controller goes to zero.. The make up isolation valve 2CV-4926 will auto close at this time.

A is incorrect because the BAM Pump 2P-39A will auto start but plausible as it has to be manually stopped and manually started for a manual make up of boric acid to the RCS.

B is incorrect because the BAM Pump 2P-39A will auto start but has to be manually stopped but plausible as it has to be manually started for a manual make up of boric acid to the RCS and the isolation valve 2CV-4926 will automatically open and close..

D is incorrect because the BAM Pump 2P-39A will need to be manually stopped but plausible as the pump will automatically start and isolation valve 2CV-4926 will automatically open and close.

This question matches the K&A as the candidate must be able to monitor the auto and manual start/stop features of the Boric Acid Pumps during a normal boration at power.

References:

STM_2-04__31-1 CVCS Section 2.3.10 (Verified reference updated 11/15/16);
NOP 2104.003 Chemical Addition Rev. 57 Exhibit 3 (Verified reference updated 11/15/16);
STM_2-04__31-1 CVCS BORATE MODE Drawing (Verified reference updated 11/15/16).

Historical Comments:

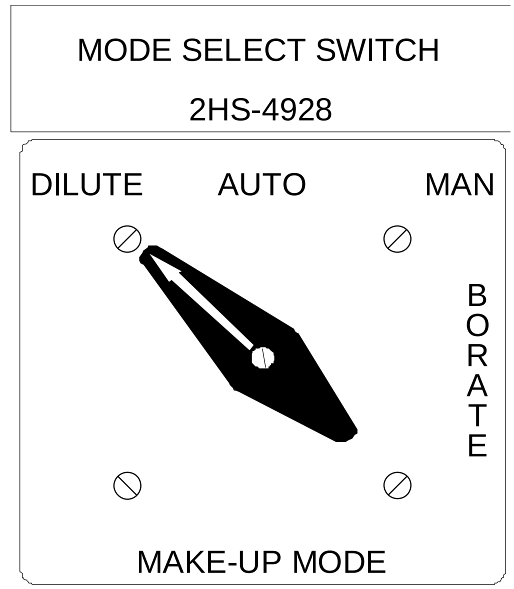
To be used on the 2017 NRC Exam

maximum thermal expansion rate in the event that maximum duplicate heat tracing power were applied to the isolated line.

The heat tracing system has its own local panels for alarm and indicating purposes. These panels are numbered 2C329, 2C330, 2C331, 2C332, 2C333 and 2C334.

2.3.10 Makeup Mode Select Switch 2HS-4928

The makeup mode selector switch 2HS-4928 provides four modes of injecting either soluble boron solutions or reactor makeup water into the reactor coolant system. This switch is located on panel 2C09 next to other controllers used to add water to the RCS. 2HS-4928 is a four-position switch. The positions provided on 2HS-4928 are “DILUTE”, “AUTO”, “MAN”, and “BORATE”.



System operation with 2HS-4928 in MANUAL

The “MANUAL” mode can be used to add boric acid to the VCT. In this mode the desired blended solution may be added to the VCT regardless of its present level. Flow can be directed to either the VCT or to the charging pump suction header.

When using the "MANUAL" mode, the operator must determine the desired flow rates for boric acid and RMW needed to achieve the desired boron concentration and set 2FIC-4926 and 2FIC-4927 accordingly. 2FQIS-4926 and 2FQIS-4927 are used to determine the quantity of boric acid and reactor makeup water, respectively that is to be added. The operator must then manually open the VCT makeup isolation valve 2CV-4941-2. The desired BAM pump and RMW pump are then started manually. The desired amount of boric acid solution is then added to either the VCT or the charging pump suction (depending on the position of manual valves located downstream of 2CV-4941-2).

“MANUAL” mode can also be used to add boric acid to the spent fuel pool or the refueling water tank if desired.

2.3.10.2 System operation with 2HS-4928 in DILUTE

The “DILUTE” mode is used to reduce the RCS boron concentration by the addition of reactor makeup water via the RMW pumps to the volume control tank or to the charging pump suction header. This mode allows a preset amount of RMW to be added at a preset flow rate and to be automatically terminated when that amount has been added.

To perform an RCS dilution using this mode the operator must first use 2FQIS-4927 to set the quantity of RMW to be added. The desired flowrate of the dilution is selected on 2FIC-4927 which will position 2CV-4927 as needed to maintain that flowrate during the dilution operation. The operator will then need to open VCT makeup valve 2CV-4941-2 because this valve will not automatically open in the “DILUTE” mode.

When 2HS-4928 is then positioned to the “dilute” mode the following occurs:

- the selected RMW pump starts,
- RMW flow control valve 2CV-4927 opens to control RMW flow at the desired rate.

When flow totalizer 2FQIS-4927 determines that the total flow added equals that predetermined dilution amount, RMW flow control valve 2CV-4927 closes. The operator manually stops the running RMW pump if desired and closes the VCT makeup valve 2CV-4941. The RMW pump is normally left running to minimize water hammer in the RMW piping.

2.3.10.3 System operation with 2HS-4928 in BORATE

The “BORATE” mode is used to add boric acid from the BAM Tanks, via the selected BAM pump, to the charging pump suction header. In this mode, a preset amount of boric acid at a preset flow rate is directed to the charging pump suction header through control valve 2CV-4930 and is automatically terminated when that preset amount has been added.

To utilize the “BORATE” mode the operator must first pre-select the amount of boric acid to add. This can be done by selecting the desired quantity in gallons on 2FQIS-4926. Boric acid flow controller 2FIC-4926 is then set at the desired flowrate for the boration that is to take place. The operator should also verify that the handswitch for the charging pump suction source from boric acid valve 2CV-4930 is in "AUTO".

With the above conditions met, the following occurs when the mode select switch 2HS-4928 is placed in the "BORATE" position:

- boric acid flow control valve 2CV-4926 opens to control BAM discharge flow at the desired flow rate,
- charging pump suction source from boric acid valve 2CV-4930 opens,
- the selected BAM pump starts.

When the total amount of boric acid, as determined by 2FQIS-4926, has been reached 2CV-4926 will close. The operator will then have to manually stop the running BAM pump and close 2CV-4930.

2.3.10.4 System operation with 2HS-4928 in AUTO

The “AUTO” mode is normally used only to add a makeup solution to the RCS that is of the same boron concentration as the RCS. The flow rates for boric acid and reactor makeup water are preset by the operator to provide a blended solution at the desired boron concentration. RMW, via 2CV-4927, and boric acid, via

NORMAL RCS BORATION AT POWER

R
A
P

CAUTION
The following section has been determined to have a Reactivity Addition Potential (RAP) and this activity is classified as a Risk Level R3.

1.0 IF a Reactivity Management Brief has NOT been conducted,
THEN perform a Reactivity Management Brief per COPD-030 with an SRO.

2.0 IF this is the first Boration of the shift,
THEN verify BAM Flow totalizer (2FQI-4926) reset.

3.0 IF desired,
THEN record initial controller data:

2FIC-4926 Setpoint: ___________ Demand: _____________

4.0 Verify Boric Acid Makeup Flow controller (2FIC-4926) set as follows:

- Setpoint set to desired flow rate.
- IF in MANUAL,
  THEN demand set to desired value.

5.0 Verify desired BAM pump (2P-39A OR 2P-39B) selected for automatic operation using BAM pump Select switch (2HS-4911-2).

6.0 Place Mode Select switch (2HS-4928)to BORATE.

7.0 Verify Charging Pump Suction From Boric Acid (2CV-4930) opens (2HS-4930).

8.0 Verify selected BAM pump running:

- 2P-39A (2HS-4919-2)
- 2P-39B (2HS-4910-2)

* 9.0 Verify BAM Tank Recirc open for running pumps:

- 2T-6A recirc (2HS-4903-2)
- 2T-6B recirc (2HS-4915-2)

(Exhibit-3 Continued on next page)

NORMAL RCS BORATION AT POWER

* 10.0 IF additional boric acid flow required,
THEN manually start additional BAM pump:

- 2P-39A (2HS-4919-2)
- 2P-39B (2HS-4910-2)

CRITICAL STEP

11.0 Operate Boric Acid Makeup Flow Batch controller (2FQIS-4926) as follows:

11.1 Depress AND hold red pushbutton.

11.2 Verify Boric Acid Makeup Flow Batch controller (2FQIS-4926) set for desired quantity.

11.3 Release Red pushbutton.

12.0 Verify Boric Acid Makeup Flow controller (2FIC-4926) indicates desired flow rate.

* 13.0 Perform the following to Start/Stop additional Charging pumps:

13.1 IF desired to raise flow,
THEN perform the following:

C. Start additional charging pumps as necessary.

D. Adjust Boric Acid Makeup Flow controller (2FIC-4926) to desired flow rate.

13.2 IF desired to lower flow,
THEN perform the following:

C. Adjust Boric Acid Makeup Flow controller (2FIC-4926) to desired flow rate.

D. Secure additional Charging Pumps as necessary.

* 14.0 Monitor the following parameters:

- RCS T_{AVE}
- Axial Shape Index
- Reactor power

(Exhibit-3 Continued on next page)

* 15.0 IF ANY of the following occur during RCS Boration,

- Fuel Pool Level High Alarm (2K11-J4)
- Expected VCT level change not observed
- CRS/SM directs securing RCS Boration

THEN perform ANY of the following as necessary:

- Reset Boric Acid Makeup Flow Batch controller (2FQIS-4926) to zero.
- Place Mode Select switch (2HS-4928) in DILUTE.
- Verify Boric Acid Makeup Flow Control 2CV-4926 closed (2FIC-4926).
- Secure running BAM pump:
  - 2P-39A (2HS-4919-2)
  - 2P-39B (2HS-4910-2)

CRITICAL STEP

16.0 WHEN Boric Acid Makeup Flow Batch controller (2FQIS-4926) at zero, THEN verify the following:

- Boric Acid Makeup Flow Control (2CV-4926) closes.
- No flow indicated on Boric Acid Makeup Flow controller (2FIC-4926).

* 17.0 Repeat steps 11.0 through 16.0 as required for boric acid addition.

(Exhibit-3 Continued on next page)

2104.003 EXHIBIT 3 Revised 02/26/16

NORMAL RCS BORATION AT POWER

PAGE 4 OF 4

18.0 WHEN desired to secure boric acid addition lineup, THEN perform the following:

18.1 Place Mode Select switch (2HS-4928) to DILUTE.

18.2 IF BAM pump operation not required, THEN secure running BAM pump:

- 2P-39A (2HS-4919-2)
- 2P-39B (2HS-4910-2)

18.3 Close BAM tank Recirc valve for secured pump:

- 2T-6A recirc (2HS-4903-2)
- 2T-6B recirc (2HS-4915-2)

18.4 Verify Charging Pump Suction From Boric Acid (2CV-4930) closed (2HS-4930).

19.0 IF desired, THEN restore initial controller data recorded in step 3.0.

20.0 Verify Boric Acid Makeup Flow Batch controller (2FQIS-4926) Batch Volume placard updated to current batch volume.

Question 35

Bank:	2365	**Rev:**	1	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	35	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	059000A205	**10CFR55:**	41.10	**Safety Function**	4				
Title:	Main Feedwater (MFW) System	**System Number**	059	**K/A**	A2.05				
Tier:	2	**Group:**	1	**RO Imp:**	3.1	**SRO Imp:**	3.4	**L. Plan:**	A2LP-RO-AMFP
OBJ	3								

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the MFW System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Rupture in MFW suction or discharge line.

Question:

Given the following:

* The plant is at full power.
* Annunciator 2K03-C9 "FW PUMP SUCT PRESS LO" comes in.
* The Main Feed Water Pump suction crossover line has ruptured.
* Steam is filling up the Turbine Building.
* Turbine Building Sumps LEVEL HIGH alarms 2K12-J8/J9 have come in.
* Main FW Pump Suction pressure is 444 psig and lowering rapidly.

Based on the above conditions, the correct action/procedure needed to mitigate the event would be to ____ and if the suction pressure to the MFW Pumps continues to drop, MFW Pumps will trip at a setpoint of _________ psig after a 30 second time delay.

A. commence a rapid Turbine load reduction using OP-2203.027 Loss of MFW Pump; 325

B. trip the Reactor and secure Feed/Condensate using OP-2203.051 Internal Flooding; 325

C. commence a rapid Turbine load reduction using OP-2203.027 Loss of MFW Pump; 425

D. trip the Reactor and secure Feed/Condensate using OP-2203.051 Internal Flooding; 425

Answer:

B. trip the Reactor and secure Feed/Condensate using OP-2203.051 Internal Flooding; 325

Notes:

B is correct as the setpoint is correct and the ACA for the turbine building sump level alarm directs going to and is an entry condition for the Internal Flooding AOP. This procedure directs plant trip and securing of the Feed water, Condensate and Heater Drain Pumps

A is incorrect but plausible as this is the action to take if one MFW pumps trips with out a leak or a smaller leak but both will need to be tripped with the location of the rupture. Also plausible as this is the correct setpoint.

C is incorrect as the setpoint is wrong and the action is incorrect for tripping of both FWPs but plausible as this is the setpoint at which standby Condensate pumps will start after a time delay and plausible as this is the action to take if one MFW pumps trips with out a leak or a smaller leak but both will need to be tripped with the location of the rupture.

D is incorrect as the setpoint is wrong but plausible as this is the setpoint at which standby Condensate pumps will start after a time delay and plausible as this is the correct action to take.

This question matches the K&A as the candidate must predict when the Feedwater Pump will trip based on lowering

suction pressure and have knowledge of which procedure and actions will mitigate the rupture of the Feed Pump Suction crossover line.

References:

2203012C ANNUNCIATOR 2K03 CORRECTIVE ACTION 2K03 Window C-9 Rev 33 (Verified reference updated 11/15/16); 2203012C ANNUNCIATOR 2K03 CORRECTIVE ACTION 2K03 Window C-9 Rev 33 Page 2 (Verified reference updated 11/15/16); AOP 2203.051, Internal Flooding, Rev. 5 Entry Conditions, and steps 6 and Section 3 Step 1 (Verified reference updated 11/15/16); AOP 2203.027 Loss of MFW Pump REV. 17 Step 3 and 4 (Verified reference updated 11/15/16). Drawings of feed and Condensate to show leak location affects both Feedwater Pumps.

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Rearranged the 2nd part to be first and the 1st part to be 2nd in the stem and distractors as requested.

ANNUNCIATOR 2K03

C-9

PUMP SUCT PRESS LO

1.0 CAUSES

1.1 Main Feed Pump 2P-1A Suction pressure (2PS-0735A) < 450 psig.

2.0 ACTION REQUIRED

NOTE

2P-1A trips at < 325 psig suction pressure with a 30 second time delay.

2.1 Validate alarm using trending capability of PMS/PDS – 2P-1A screen.

2.1.1 IF alarm determined to be invalid,
THEN no further action required.

2.2 Verify the following Condensate and Feed Pump Recirc valves closed:

- 2CV-0662 (2FIC-0662)
- 2CV-0663 (2FIC-0663)
- 2CV-0749 (2FIC-0742)
- 2CV-0741 (2FIC-0735)

NOTE

- Condensate pump auto starts at < 425 psig with time delay depending on which condensate pump is in standby. Time delay is 10 seconds for 2P-2A, 8 seconds for 2P-2B, 6 seconds for 2P-2C and 4 seconds for 2P-2D.

Four Condensate Pump Operation Limitations

- Time spent running four condensate pumps and one heater drain pump is limited to 2 to 3 weeks per year (ER-ANO-2-2002-0613).
- Running four condensate pumps is NOT allowed if all ANO2 non-vital 6900V and 4160V buses are energized from SU3 (ER-ANO-2-2002-0613).

2.3 IF necessary to maintain MFP Suction pressure above 450 psig,
THEN verify standby Condensate pump running:

- 2P-2A (2HS-0609)
- 2P-2B (2HS-0620)
- 2P-2C (2HS-0614)
- 2P-2D (2HS-0626)

(C-9 Continued on next page)

ANNUNCIATOR 2K03

C-9

PUMP SUCT PRESS LO
(Continued)

2.4 IF Turbine Building Sump alarm(s) annunciate:

- TURBINE BLDG SUMP STA 1 LEVEL HI (2K12-J8)
- TURBINE BLDG SUMP STA 2 LEVEL HI (2K12-J9)

THEN investigate immediately.

2.5 Check 2P-1A Suction Strainer ΔP (2PDIS-0735).

2.6 IF strainer ΔP greater than 7 psid,
THEN perform the following as necessary:

2.6.1 Refer to Main Feedwater Pump and FWCS Operations (2106.007) for removing 2P-1A from service.

2.6.2 Initiate WR/WO for strainer cleaning.

2.7 IF MFW pump (2P-1A) trips,
THEN **GO TO** Loss Of Main Feedwater Pump (2203.027).

3.0 TO CLEAR ALARM

3.1 Raise Main Feed pump (2P-1A) Suction pressure above 450 psig.

4.0 REFERENCES

4.1 E-2453-2

4.2 ER002344I232, Electrical Input for 4 CNDP OPS in response to FWP Trip

4.3 ER-ANO-2-2002-0613, Electrical Input for 4 CNDP OPS

4.4 ER-ANO-2005-0248, MFP suction strainer installation

INTERNAL FLOODING

2203.051PURPOSE

This procedure provides actions for internal flooding.

ENTRY CONDITIONS

Localized flooding due to gross system leakage or piping/component rupture inside the Turbine Building, Diesel Fuel vault, Aux Building or Aux Extension Building as indicated by any of the following:

1. Local observation.

2. Sump/room high level alarms:

   - ESF ROOM(S) LEVEL HI (2K12-H8)
   - 2T57 A/B ROOM LEVEL HI (2K08-D2)
   - TURBINE BLDG SUMP STA 1 LEVEL HI (2K12-J8)
   - TURBINE BLDG SUMP STA 2 LEVEL HI (2K12-J9)
   - LIQUID RADWASTE SYS TROUBLE (2K11-G9)
     - AUX BLDG SUMP LEVEL HIGH (2K15-A1)
     - WASTE TANK 2T20A LEVEL HIGH (2K15-A2)
     - WASTE TANK 2T20B LEVEL HIGH (2K15-A3)

EXIT CONDITIONS

WHEN ALL appropriate actions performed,
THEN exit this procedure.

6. Check the following system piping intact:

A. Circ Water	A. **GO TO** Section 2, Circ Water.
B. Condensate/Feedwater	B. **GO TO** Section 3, Condensate/Feedwater.
C. Fire Water	C. **GO TO** Section 4, Fire Water.
D. Service Water/ACW	D. **GO TO** Section 5, Service Water/ACW.

7. IF leak isolable,
THEN perform the following:

A. Determine valves necessary for isolation.

B. Close valves as necessary to isolate leak.

C. Verify appropriate configuration control method initiated.

8. Commence Attachment B, Local Flooding Actions.

9. IF Plant power reduction required,
THEN reduce Plant power as necessary using EITHER of the following:

- **2102.004, Power Operation**
- **2203.053, Rapid Power Reduction**

10. Check leak isolated or controlled.

10. Continue efforts to locate, isolate, and control leak.

11. WHEN leak isolated or controlled,
THEN perform Attachment A, Recovery Actions.

12. Continue Operations as directed by Plant Management.

END

*1. **IF Condensate/Feedwater leak rate OR location requires immediate plant shutdown,**
THEN perform the following as necessary:

A. Verify reactor tripped.

B. Verify the following pumps secured:

1) Main Feedwater pumps:
- 2P-1A
- 2P-1B

2) Heater Drain pumps:
- 2P-8A
- 2P-8B

3) Condensate pumps:
- 2P-2A
- 2P-2B
- 2P-2C
- 2P-2D

C. Verify EFAS actuated.

D. Perform 2202.001, Standard Post Trip Actions in conjunction with this procedure.

*2. **IF Hotwell requires makeup,**
THEN raise Hotwell level using 2106.016, Condensate and Feedwater Operations, Exhibit 1, Hotwell Makeup Operations.

3. **Commence Attachment B, Local Flooding Actions.**

INSTRUCTIONS	CONTINGENCY ACTIONS
NOTE • **Steps marked with (*) are continuous action steps.** • **Steps marked with (■) are floating steps.**	
1. Open Placekeeping page.	
2. Notify Control Board operators to monitor Floating Steps.	
3. Check at least ONE MFW pump operating.	**3. Perform the following:** A. Attempt to place standby MFWP in service. B. Attempt to restart tripped MFWP. C. IF Reactor power less than 4%, THEN start Auxiliary Feedwater pump (2P75). Refer to 2202.010 Attachment 52, Establishing AFW Flow. D. IF Reactor power less than 4% AND Auxiliary Feedwater NOT available, THEN start at least ONE Emergency Feedwater pump (2P7A/B). Refer to 2202.010 Attachment 46, Establishing EFW Flow. E. IF Reactor power greater than 4% AND feedwater can NOT be restored, THEN perform the following: 1) Trip Reactor. 2) **GO TO** 2202.001, Standard Post Trip Actions.

INSTRUCTIONS	CONTINGENCY ACTIONS
■4. **Check BOTH of the following conditions exist for each SG:** • Indicated Feed flow greater than or equal to indicated Steam flow. • S/G level restoring to setpoint.	■4. **Perform the following as necessary:**

NOTE

When steam dumps open in response to reducing turbine load, there is no effect on Main Steam flow.

A. Commence **Rapidly** reducing Turbine load until **BOTH** of the following are met:
   - Steam flow less than 6000 KLBM/hr.
   - Feed flow greater than steam flow.

CAUTION

- **At 100% power, large CEA insertions (Group 6 and Group P near but not below their respective Transient Insertion Limit) will be required to sufficiently lower power for single MFP operation.**
- **Due to lower CEA worth, it may not be possible to maintain the plant online at higher boron concentrations.**

B. Commence inserting group 6 (preferred) and/or group P CEAs using 2105.009, Exhibit 3 as necessary to reduce power.

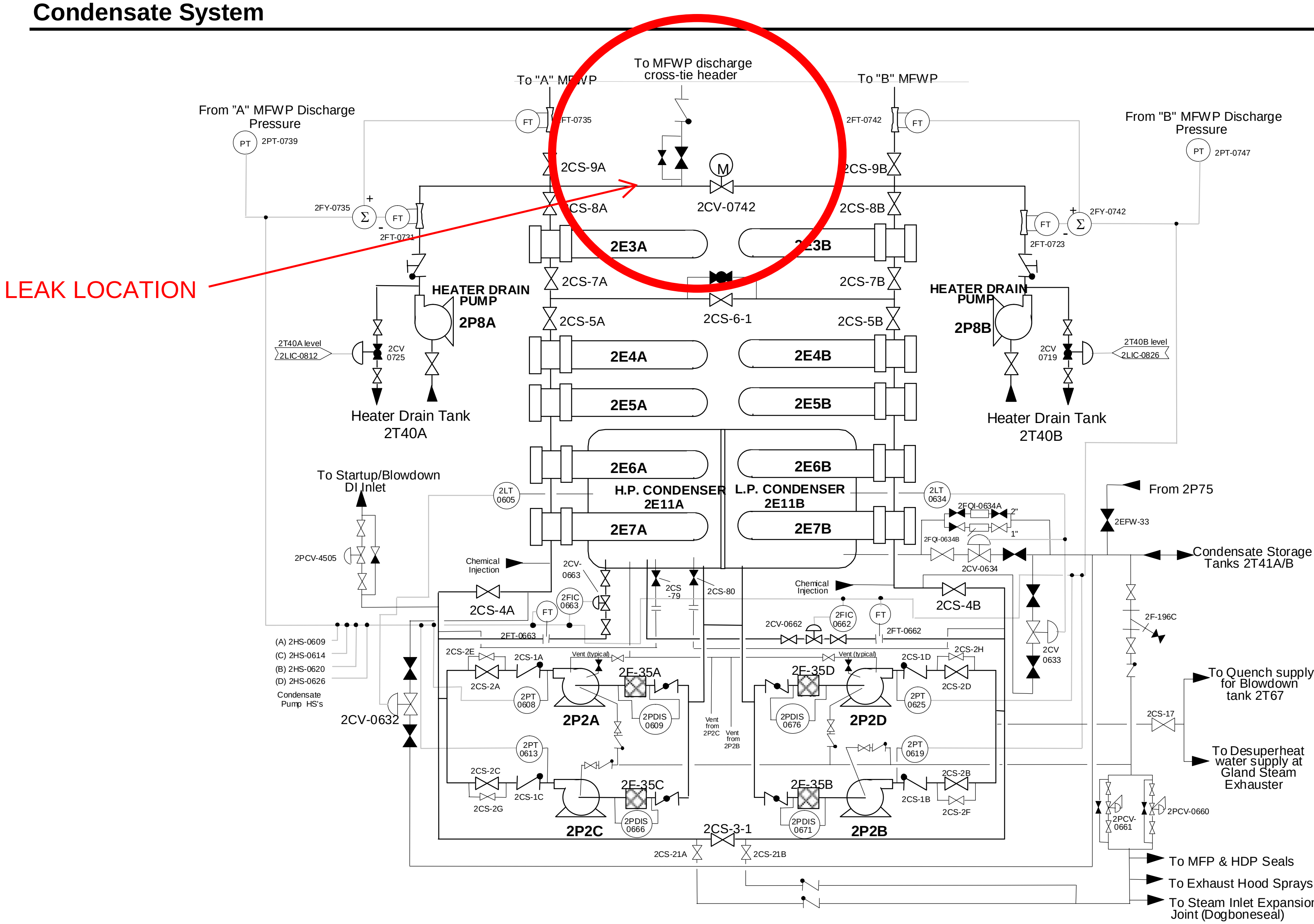
C. Commence boration using 2202.010 Exhibit 1, Emergency Boration, as necessary to reduce power.

D. IF desired,
THEN isolate SG blowdown using 2HIC-1017 and 2HIC-1067.
   - 2HIC-1017 (2CV-1017 Controller)
   - 2HIC-1067 (2CV-1067 Controller)

(Step 4 continued on next page)

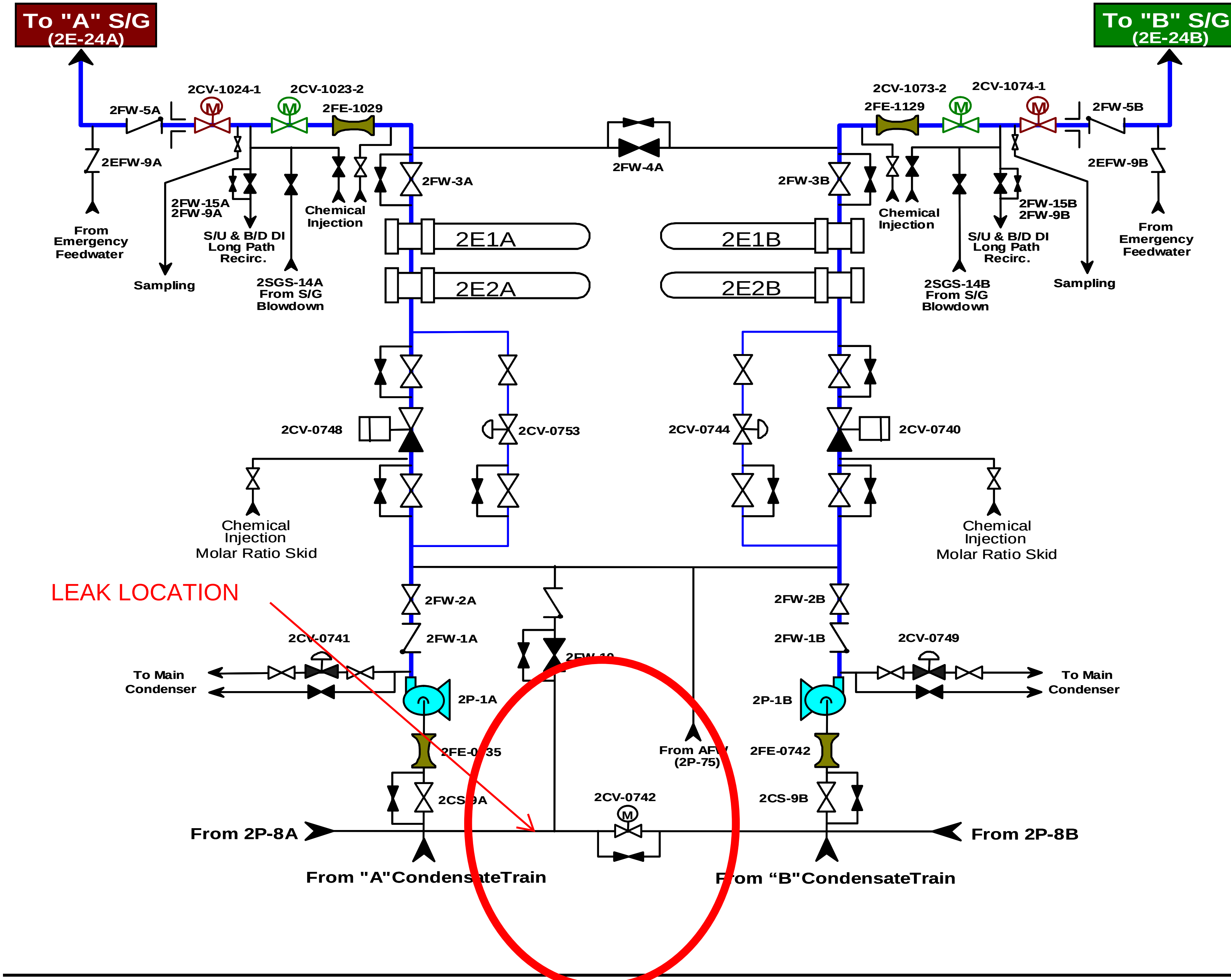
Figures

Condensate System



Figures

Main Feedwater System



Question 36

Bank:	2366	**Rev:**	2	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	36	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	Modified NRC Exam Bank 0526		
Search	022000A101	**10CFR55:**	41.5	**Safety Function**	5				
Title:	Containment Cooling System (CCS)	**System Number**	022	**K/A**	A1.01				
Tier:	2	**Group:**	1	**RO Imp:**	3.6	**SRO Imp:**	3.7	**L. Plan:**	A2LP-RO-CVENT
OBJ	15								

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: - Containment temperature

Question:

(1 PAGE OF PMS SCREEN OF CONTAINMENT PARAMETERS ATTACHED)

Given the following:

* The plant is at full power in the middle of April on dayshift at 0930.
* The running Main Chiller 2VCH-1A trips and will not restart.
* Main Chiller 2VCH-1B was started but trips after every start attempt.
* At 10:30 Annunciator 2K10 A-7 " CNTNT TEMP/HUMIDITY HI" comes in.
* All 4 Containment Fan Coolers are running.
* Current conditions in containment at 10:30 are as shown on the attached PMS screen.

Based on the current containment conditions, which of the following would be correct in accordance with Annunciator 2K10 A-7 " CNTNT TEMP/HUMIDITY HI"?

A. Restore the Containment internal pressure/temperature within the limits of T.S 3.6.1.4, Containment Internal Pressure/Temperature, within one hour.

B. Raise Containment internal pressure to approximately 14.6 psia to provide margin to prevent exceeding TS. 3.6.1.4, Containment Internal Pressure/Temperature, limits.

C. Lower Containment internal temperature/humidity by starting all available CEDM cooling and Containment Recirculation Fans within one hour to prevent exceeding TS 3.6.1.4, Containment Internal Pressure/Temperature, limits.

D. Lower Containment internal temperature/humidity by aligning Service Water to Containment Fan Coolers to prevent exceeding TS. 3.6.1.4, Containment Internal Pressure/Temperature, limits.

Answer:

D. Lower Containment temperature/humidity by aligning Service Water to Containment Fan Coolers to prevent exceeding TS. 3.6.1.4, Containment Internal Pressure/Temperature, limits.

Notes:

D is correct as the current containment pressure and temperature are not exceeding the design limits but are very close and based on loss of the Main Chillers will go higher. The ACA for 2K10-A7 " CNTNT TEMP/HUMIDITY HI" provides guidance to lower containment temperature by aligning SW to the containment fan coolers if the alarm is present.

A is incorrect because the Limits of T.S. 3.6.1.4 have not been exceeded but plausible as the limits will be exceeded if no corrective action is taken and this is the correct action to take if in T.S. 3.6.1.4.

B is plausible as this would provide additional margin to the design limits but the normal procedure has a precaution and limit that prevents exceeding a containment pressure of 14.2 psia. See the reference NOP 2104.033, Containment Atmosphere Control Section 5.0.

C is incorrect because Recirculation fans have no Cooling HX and the Chilled Water has been lost to the CEDM cooling Fans. However this would be a plausible action if Chilled Water were available and recirc fans would mix hot and cold air pockets together but not lower overall average containment temperature.

This question matches the K&A because the candidate must be able to monitor containment temperature, predict changes based on the forecast, and operate the CCS controls to prevent exceeding the design limits of containment temperature.

References:

2203012J, Rev. 43, ANNUNCIATOR 2K10 CORRECTIVE ACTION Window A-7 CNTMT TEMP-HUMIDITY HI (Verified reference updated 11/15/16); ANO-2 TS 303B60 TS 3.6.1.4 Containment Internal Pressure and Temperature Amendment 301 (Verified reference updated 11/15/16);
NOP 2104.033, Rev. 77, Containment Atmosphere Control Section 5.0 (Verified reference updated 11/15/16);

Containment Temp and Pressure Chart Handout is not considered an open reference item as the K&A specifically say to monitor Containment temperature and the handout item is part of the question stem.

Historical Comments:

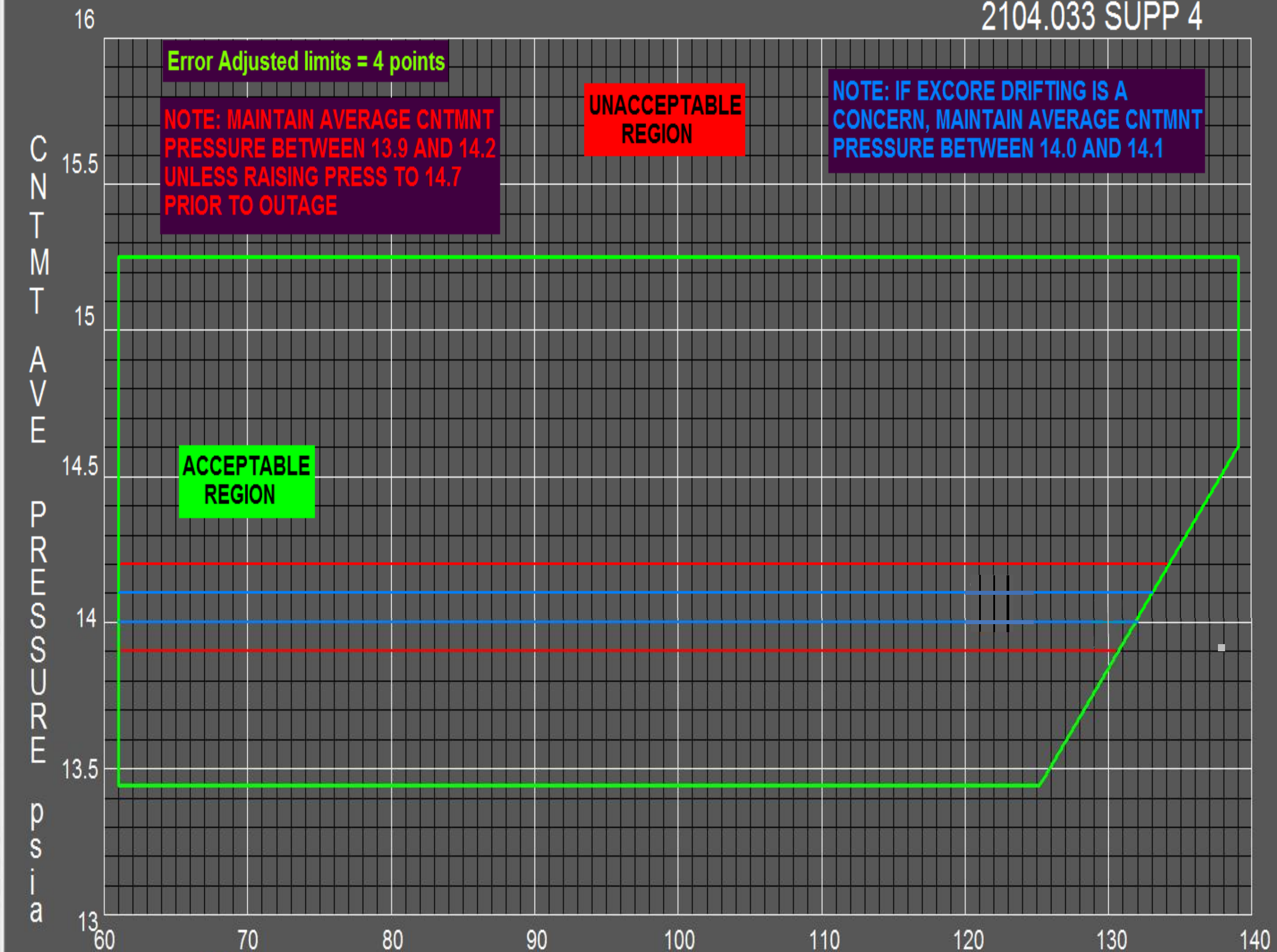
NRC Exam Bank 0526 was used on the 2005 NRC Exam
To be used on the 2017 NRC Exam but modified for the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Added "/humidity" to distractors C and D. Added "exceeding TS. 3.6.1.4, Containment Internal Pressure/Temperature, limits." to distractor C.

REV. 2 based on NRC Chief Examiner Feedback BNC. Added the word "internal" to answer D between Containment and temperature. Also removed the white "X" from the attachment as requested.

ATTACHMENT FOR QUESTION # 36

2104.033 SUPP 4



CONTAINMENT AVG AIR TEMP °F DATE/TIME VERIFIED BY ______ INIT.

	P5601-1	14.11	
	P5602-2	14.08	
	P5603-3	14.08	PSIA
	P5604-4	13.96	
AVG	P5601-A	14.06	
	T5605-5	129.6	
	T5606-6	129.8	°F
AVG	T5600	129.7	

T.S.3.6.1.4 - CIRCLE YES IF ALL 4 CNTMNT PRESS COMPUTER POINTS ARE AVAILABLE AND PLOTTED POINT IS WITHIN THE ACCEPTABLE REGION, OTHERWISE CIRCLE NO. IF NO CIRCLED, THEN PERFORM 2104.033 SUPP 4. YES NO

PERFORMED BY:________________ REVIEWED BY:________________

ANNUNCIATOR 2Kl0

A-7

CNTMT TEMP/HUMIDITY HI

> **NOTE**
> This alarm will reflash.

1.0 CAUSES

1.1 > 92°F dewpoint temperature on any of 5 detectors located in Containment (2MS-5661, 2MS-5662, 2MS-5663, 2MS-5664, or 2MS-5665)

1.2 > 135°F on any of 4 temperature detectors located in Containment (2TS-5661, 2TS-5662, 2TS-5663, 2TS-5664)

1.3 Breaker 43LA-12 manipulation – possible spike of 2MS-5664/5665 on 2MR-5660

2.0 ACTION REQUIRED

2.1 Check Annunciator 2K423 in 2C32 for initiating alarm:

- CNTMT TEMP HI 2TS-5661
- CNTMT TEMP HI 2TS-5662
- CNTMT TEMP HI 2TS-5663
- CNTMT TEMP HI 2TS-5664
- CNTMT HUMIDITY HI 2MS-5661
- CNTMT HUMIDITY HI 2MS-5662
- CNTMT HUMIDITY HI 2MS-5663
- CNTMT HUMIDITY HI 2MS-5664
- CNTMT HUMIDITY HI 2MS-5665

2.2 Check CNTMT Temperature recorder (2TR-5660), Humidity recorder (2MR-5660) and PMS/PDS trends to validate alarm.

- IF alarm invalid
  OR breaker 43LA-12 was manipulated (2MS-5664/5665 affected),
  THEN no further action required.

(A-7 Continued on next page)

ANNUNCIATOR 2Kl0

A-7

CNTMT TEMP/HUMIDITY HI
(Continued)

2.3 Verify proper containment ventilation using Containment Atmosphere Control (2104.033).

2.3.1 IF desired to align Service Water to Loop 1 Containment Coolers (2VCC-2A/B),
THEN perform the following:

A. Verify Service Water Inlet 2CV-1511-1 (2HS-1511-1) open.

B. Verify Service Water Outlet 2CV-1519-1 (2HS-1519-1) open.

C. Adjust Service Water to Loop 2 CCW as necessary per Component Cooling Water System Operations (2104.028).

2.3.2 IF desired to align Service Water to Loop 2 Containment Coolers (2VCC-2C/D),
THEN perform the following:

A. Verify Service Water Inlet 2CV-1510-2 (2HS-1510-2) open.

B. Verify Service Water Outlet 2CV-1513-2 (2HS-1513-2) open.

C. Adjust Service Water to Loop 2 CCW as necessary per Component Cooling Water System Operations (2104.028).

2.3.3 IF desired to secure Service Water to Loop 1 Containment Coolers (2VCC-2A/B),
THEN perform the following:

A. Verify Service Water Outlet 2CV-1519-1 (2HS-1519-1) closed.

B. Verify Service Water Inlet 2CV-1511-1 (2HS-1511-1) closed.

C. Adjust Service Water to Loop 2 CCW as necessary per Component Cooling Water System Operations (2104.028).

(A-7 Continued on next page)

ANNUNCIATOR 2Kl0

A-7

CNTMT TEMP/HUMIDITY HI
(Continued)

2.3.4 IF desired to secure Service Water to Loop 2 Containment Coolers (2VCC-2C/D),
THEN perform the following:

A. Verify Service Water Outlet 2CV-1513-2 (2HS-1513-2) closed.

B. Verify Service Water Inlet 2CV-1510-2 (2HS-1510-2) closed.

C. Adjust Service Water to Loop 2 CCW as necessary per Component Cooling Water System Operations (2104.028).

2.4 Verify Main chiller (2VCH-1A or 2VCH-1B) in service.

2.5 Refer to Tech Spec 3.6.1.4.

2.6 Initiate Primary Leak Rate calculation IAW Reactor Coolant System Leak Detection (2305.002).

2.6.1 IF RCS primary leakage has risen,
THEN GO TO Excess RCS Leakage (2203.016).

2.6.2 IF RCS primary leakage has NOT risen,
THEN check for feedwater or steam leak inside containment.

3.0 TO CLEAR ALARM

3.1 Reduce Containment Dewpoint temperature to < 90.5°F.

3.2 Reduce Containment temperature to < 131.5°F.

4.0 REFERENCES

4.1 E-2456-4

4.2 DCP-88-2111

INTERNAL PRESSURE AND AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 The combination of containment internal pressure and average air temperature shall be maintained within the region of acceptable operation shown on Figure 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

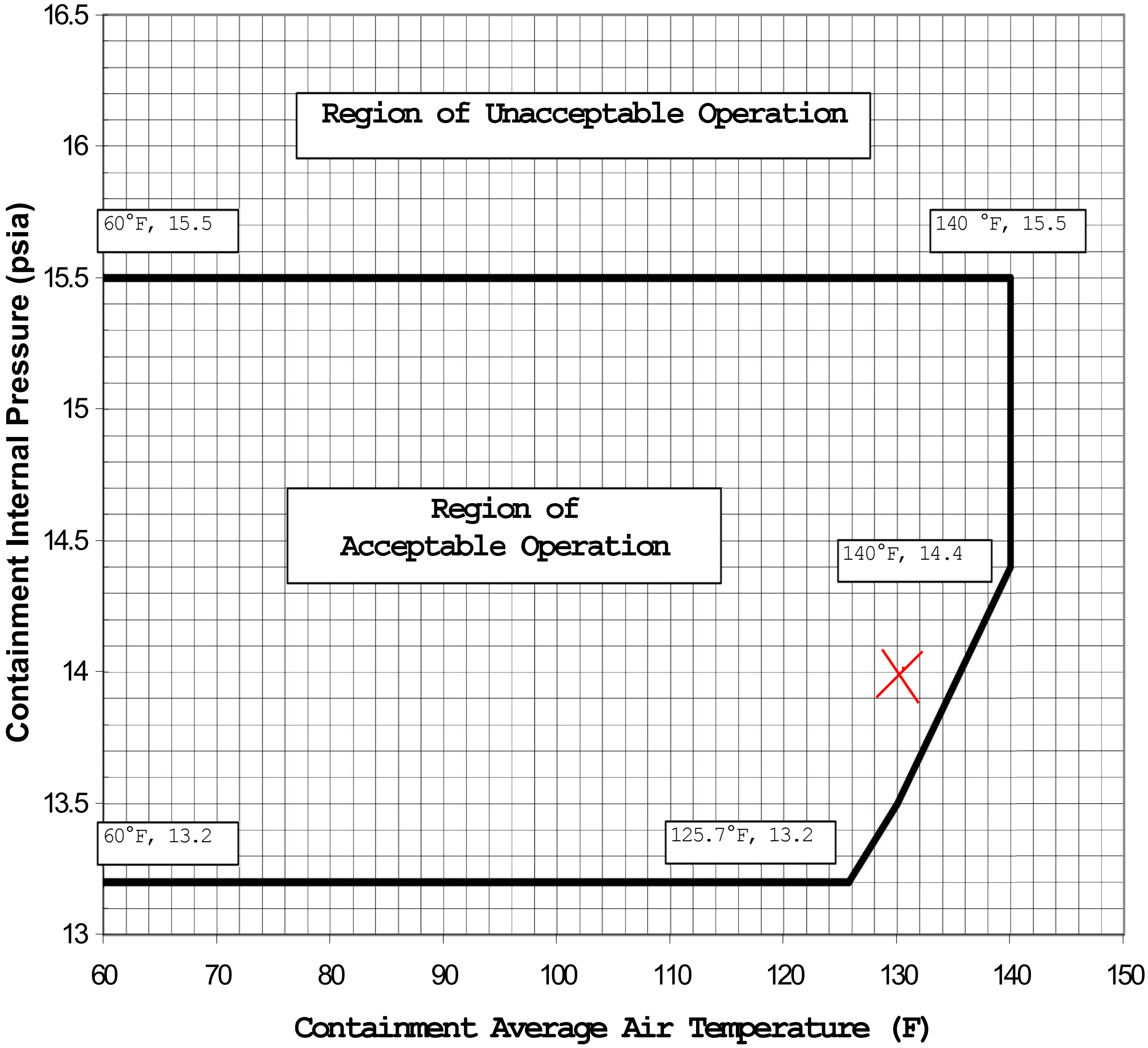
With the point defined by the combination of containment internal pressure and average air temperature outside the region of acceptable operation shown on Figure 3.6-1, restore the combination of containment internal pressure and average air temperature to within the above limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure and average air temperature shall be determined to be within the limits at least once per 12 hours. The containment average air temperature shall be the temperature of the air in the containment HVAC common return air duct upstream of the fan/cooler units.

FIGURE 3.6-1

CONTAINMENT INTERNAL PRESSURE VS. AVERAGE AIR TEMPERATURE



Containment Average Air Temperature (F)

NOTE: Instrument Error is not Included on Curve

5.0 LIMITS AND PRECAUTIONS

5.1 The Ventilation Exhaust Treatment System, consisting of charcoal absorbers and HEPA filters, is required to be in service during ventilation exhaust system operations.

5.2 ODCM L2.2.1 requires grab samples be taken at least once per 12 hours and analyzed for gross activity within 24 hours when CNTMT Exhaust Ventilation in service with SPING 5 (2RX-9820) inoperable.

5.3 Tech Spec 3.3.3.1 and ODCM L2.2.1 require Core Alterations and fuel assembly movement within the Reactor Building be suspended when CNTMT Purge Ventilation in service with either SPING 5 (2RX-9820) or Containment Purge Exhaust Process Monitor (2RITS-8233) inoperable.

5.4 TS 3.3.3.1 Bases requires at least one supply and one exhaust valve to be closed and both fans secured when securing CNTMT Purge to meet TS 3.3.3.1 action requirements.

5.5 Instrument error is not included in Tech Spec 3.6.1.4 graph for Containment Building temperature, relative humidity and pressure.

5.6 Maintaining average CNTMT pressure between 13.9 and 14.2 psia provides a cushion for potential loss of chill water. Maintaining negative pressure enables fresh air to be drawn into the building to maintain oxygen levels for human occupancy. Raising containment pressure to 14.7 psia prior to outage will minimize differential pressure across the personnel hatch.

5.7 An operable Containment Cooling Group requires both cooling units to be operable (service water flow greater than or equal to 1250 gpm with two operable fans). During normal operations, three of four units should maintain building pressure and temperature within region of acceptable operations of Tech Spec TS 4.6.2.3.

5.8 The proper method for securing CNTMT Purge with Containment open to atmosphere is to ensure Exhaust fan (2VEF-15) is in HAND and Supply fan (2VSF-2) secured prior to securing 2VEF-15.

5.9 The operating limit for CNTMT Purge roughing and prefilters is 5.0 inches water ΔP. The operating limit for HEPA filters is 8.0 inches water ΔP.

5.10 Requirements for Containment Purge flow, runtime, and D/P are located in HES-06, Ventilation/Filtration Testing Program (LIC-01-038, TS Amend 230) which can be found on EB Reflib/Engineering Documents/Engineering Standards.

5.11 CNTMT Building Air Makeup Screens (2F-262 and 2F-263) are installed to prevent large debris from entering CAMS return piping. Per ER002239N201, these screens are not required for operability of CNTMT Isol valves in CAMS sample return path. Removal of these screens does not affect operability of CNTMT Isol valves. If 2F-262 or 2F-263 removed, then associated CNTMT Building M/U Isol (2HPA-76 or 2HPA-79) are closed.

PARENT QUESTION 0526 WAS USED TO MODIFY 2017 NRC EXAM QID#36 (Used on 2005 NRC Exam)

Questions For All QID In Exam Bank

Bank:	0526	**Rev:**	0	**Rev Date:**	11/7/2004	**QID #:**		**Author:**	COBLE
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	A	**Source:**	NEW		
Search		**10CFR55:**		**Safety Function**					
System Title:	CONTAINMENT COOLING SYSTEM	**System Number**	022	**K/A**	A2.06				
Tier:	NA	**Group:**	NA	**RO Imp:**	NA	**SRO Imp:**	3.2	**L. Plan:**	
OBJ									

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the CCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of CCS Pump.

Question:

Given the following:

* The plant is at full power.
* Outside ambient air temperature is 92°F.
* Lake temperature is 64°F.
* The running Main Chiller and Main Chilled Water Pump trips.
* All attempts to start the tripped and standby Main Chill Water pumps have failed.
* Containment Temperature and Pressure have gone high outside of the Region of Acceptable Operation per TS 3.6.1.4

Which of the following actions should be completed to mitigate the high temperature/pressure in containment?

A. Start all CEDM Cooling Fans and Containment Recirculation Fans; Commence a plant shutdown and be in Hot Standby in 6 hours due to the high Containment temperature.

B. Start three (3) Containment Cooling Fans and align Service Water to them; Commence a plant shutdown and be in Hot Standby in 6 hours due to the high Containment temperature.

C. Start all Containment Cooling Fans and align Service Water to them; Restore Containment Temperature and Pressure to within the Region of Acceptable Operation within 1 hour.

D. Start all CEDM Cooling Fans and Containment Recirculation Fans; Restore Containment Temperature and Pressure to within the Region of Acceptable Operation within 1 hour.

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☑
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☐

Answer:

C. Start all Containment Fan Coolers and align Service Water to them; Restore Containment Temperature and Pressure to within the Region of Acceptable Operation within 1 hour.

Notes:

Starting and aligning SW to all 4 CFCs will provide adequate cooling at the current lake temperature to restore containment parameters within acceptable limits. This is due to the much larger surface area on the SW cooling HX compared to the Chilled Water HX in the CFCs and a lot more flow.

Distracter A is incorrect because Chill Water also cools the CEDM coolers and Chilled Water is lost and TS allows a 1 hour restoration limit before shutdown.

Distracter B is incorrect because the procedure directs starting all 4 and TS allows 1 hour to restore parameters prior to the shutdown.

Distracter D is incorrect because the CEDM coolers have no cooling medium and would not affect Containment Parameters.

References:

STM 2-09, Containment Cooling and Purge Systems, Sections 2.2 and 2.7.
TS 3.6.1.4 and Figure 3.6-1.
AOP 2203.039, Inadvertent CIAS, Step 10.
A2LP-RO-CVENT, OBJ. 3, Describe the construction and operation of the Containment Cooling Units and Objective 15, Describe the conditions required to satisfy the TS LCOs and TRM TROs associated with the Containment Ventilation System including the basis for each LCO/TRO.

Historical Comments:

This question has not been used on any previous NRC exams. BNC 11/11/2004.

Question 37

Bank:	2367	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	37	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	003000K502	**10CFR55:**	41.5	**Safety Function**	4				
Title:	Reactor Coolant Pump System (RCPS)	**System Number**	003	**K/A**	K5.02				
Tier:	2	**Group:**	1	**RO Imp:**	2.8	**SRO Imp:**	3.2	**L. Plan:**	A2LP-RO-RCP
OBJ	1								

Description: Knowledge of the operational implications of the following concepts as they apply to the RCPS: - Effects of RCP coastdown on RCS parameters

Question:

Which of the following describes the main purpose of the Flywheel on the RCPs?

A. To prevent a pump overspeed transient during a cold leg break in the suction piping.

B. To prevent and/or mitigate vapor seal failures when securing RCPs with degraded seals.

C. To improve RCS flow characteristics through the core during a Loss of Offsite power.

D. To improve the cooling air flow to the upper part of the RCP Motor and thrust bearing.

Answer:

C. To improve RCS flow characteristics through the core during a Loss of Offsite power.

Notes:

C is correct as described in the RCP STM Section 1.6 and the Unit 2 SAR Section 15.1.1.5.1.1 to enhance RCS flow on a LOOP to minimize loss of DNBR

A is incorrect but plausible as this is the reason for the Anti Rotational Device ARD installed at the top of each RCP.

B is incorrect but plausible as this would this would prevent the pump shaft from stopping rapidly with three other RCPs running and jarring the seal package which could cause a sudden failure of the vapor seal with an already degraded seal package.

D is incorrect as fans mounted on each end of the rotor provide cooling air for the motor but plausible as the flywheel rotates and could supply cooling if the flywheel edges were designed like a fan.

This question matches the K&A because it requires knowledge of the design of the RCPs that provides for longer coastdown time providing better flow characteristics on a LOOP.

References:

STM_2-03-2_18-1 RCP Section 1.6 (Verified reference updated 11/15/16);
STM_2-03-2_18-1 RCP Section 1.8.3 (Verified reference updated 11/15/16);
UNIT 2 FSAR A27 Issue 1 Section 15.1.5.1.1 (Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Added clarifying information to distractor B analysis to explain plausibility of distractor B.

is ≥ 180°F for any RCP. Flow will cause a high or low flow alarm on 2K11 if bleedoff flow is ≥ 1.5 gpm **OR** ≤ 0.6 gpm for any RCP. Controlled Bleedoff pressure can be monitored on 2C09 and on PMS, and will cause a high or high-high pressure alarm on 2K11 if the combined Controlled Bleedoff pressure to the VCT is > 120 psig (for the High alarm) or > 250 psig (for High-High alarm).

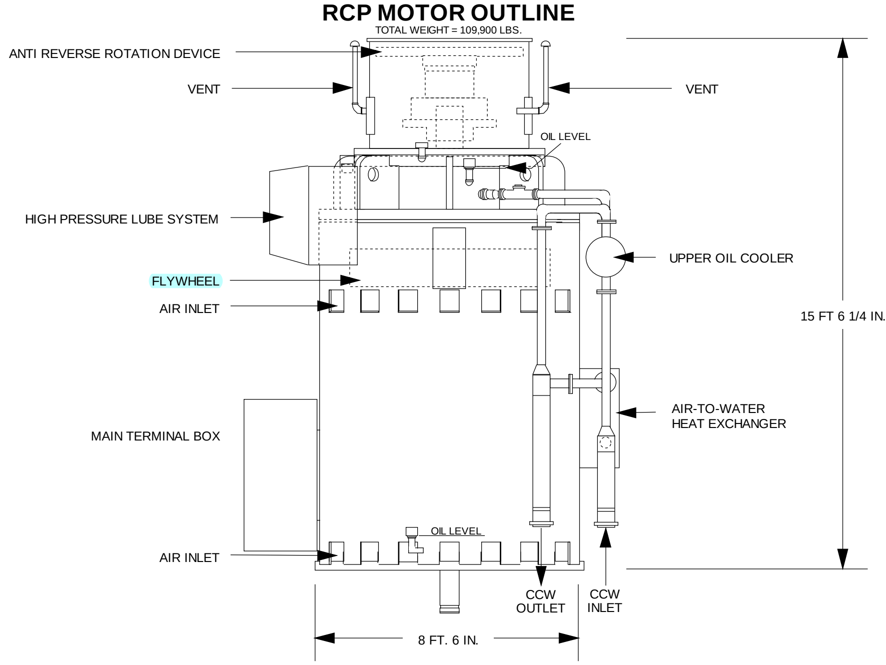
1.6 RCP Motor Description

Each Reactor Coolant Pump is driven by a three-phase, squirrel cage induction motor rated at 6750 horsepower and operates at a speed of 900 rpm. The RCP Motors are powered from 6900 volt buses 2H1 and 2H2. 2P-32A & D are powered from 2H1 and 2P-32B & C powered from 2H2. Refer to the illustration on the following page and also on page 49. Each pump motor is double-end ventilated with ambient air. An air-to-water heat exchanger is provided to cool the discharge air.

The motor also has upper and lower guide bearings, a double acting *Kingsbury* thrust bearing (for up and down thrust) in addition to the bearing in the anti-reverse rotation mechanism. Both the upper and lower bearing assemblies are oil lubricated and water cooled.

A flywheel is attached to the motor shaft between the rotor and the upper bearing. The flywheel and motor-pump rotating assembly will act to improve the flow coastdown characteristics during a loss of power to the pump.

RCP MOTOR OUTLINE



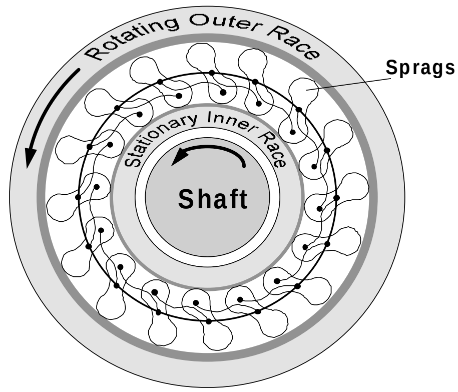
The Reactor Coolant Pump motor is air cooled. Fans mounted on both ends of the rotor take suction on RCP cavity and distributes air to the rotor, stator and flywheel. The air then flows through the motor air cooler, which is cooled by the Component Cooling Water system and discharged to the RCP cavity.

Temperatures are monitored by RTD's in each phase of the stator winding, in all bearings, in the upper oil cooler and in the lower oil reservoir.

If the RCP motor is attempted to be started from an ambient temperature condition, then a second start attempt is allowed after a 30 minute wait. No additional starts should be attempted until the motor returns to rated full load temperature after 200 minutes of idle time or 60 minutes of running time. At that time another start could be attempted.

1.6.1 Anti Reverse Rotation Device

The RCP motors are equipped with an Anti-Reverse Rotation Device, located at the top of the motor. This device is intended to prevent reverse rotation (in the clockwise direction when looking down at the top of the motor) caused by backflow through the impeller. The Anti-Reverse Rotation Device will stop a pump when it decelerates to zero rpm from normal speed (900 rpm nominal) while the remaining pumps continue to operate.



Anti-Reverse Rotation Device

The device consists of concentric inner and outer races, a complement of sprags between those races, and a bearing to maintain the spacing between the races. If reverse rotation of the motor shaft is attempted, the sprags wedge between the inner and outer races, preventing rotation. Rotation in the forward direction frees the sprags, and free forward rotation is permitted. The outer race is secured to the shaft such that it rotates with the shaft and the inner race is held stationary. This device will prevent a pump/flywheel overspeed transient during a cold leg break in the suction piping. One advantage that the anti-reverse rotation device has is that when starting an RCP, the starting current duration is reduced as compared to starting a pump rotating in the reverse direction.

1.6.2 Motor Lubrication System

The Lubrication System associated with the Reactor Coolant Pump motor consists of a high pressure oil lift pump and a low pressure oil lift pump. The low pressure oil lift pump supplies the Anti-Reverse Rotation Device. The high pressure oil lift pump supplies the motor oil lift system. Both pumps are driven by a common 480 VAC induction motor.

NAME	CAUSE(S)
Reverse Rotation	Reverse oil flow which corresponds to a rotational speed of 75 to 200 rpm in the reverse direction for the associated RCP. If this alarm is in when all 4 RCPs are in service the alarm is malfunctioning. If the alarming RCP is not operating, then all running RCPs need to be secured.
HP Oil Lift Press Lo	Associated RCP HP Lift pump discharge pressure < 400 psig AND the associated RCP breaker open.
LP Oil Filter ΔP Hi	> 25 psid across the associated RCP LP Lift Oil Filter
RCP Bleed off Temp Hi	Controlled Bleedoff temperature ≥ 180°F for any Reactor Coolant pump

1.8.3 2203.025 RCP Emergencies

As with every other Abnormal Operating Procedure there is a section at the beginning of this procedure that lists the entry and exit conditions. The next section lists various symptoms that could have caused the entry into this procedure and the appropriate section to refer to in order to mitigate the event.

Depending upon the symptom, either a plant shutdown or an immediate reactor trip will be required. The following is a synopsis of the conditions that would cause either one of these to take place.

- **ANY** of the following conditions will require a reactor **trip** and the affected RCPs stopped:
  - Loss of RCP CCW flow for greater than 10 minutes
  - RCP Vapor Seal pressure greater than 1500 psia
  - **THREE** or more stages failed on **ANY** RCP seal.
  - Valid RCP Motor Stator Winding Temperature alarm with **rising trend**.
  - RCP Upper or Lower Thrust Bearing Temperature greater than 225˚F.
  - Low level alarm on RCP upper or lower oil reservoir with bearing temperature increasing greater than 18 °F/min or bearing temperature exceeding 195 °F.
  - Loop 2 CCW Surge Tank level less than 13 % following restoration of CCW flow to RCPs.
  - RCP Vibration High as determined by any of the following:
    - Motor Vibration > 20 mils
    - Shaft Vibration > 25 mils
    - A rapid rise in vibration with other parameters also indicating a problem.

- **ANY** of the following conditions will require a plant shutdown:
  - **TWO** stages failed on **ANY** RCP seal.

- RCP Controlled Bleedoff temperature can **NOT** be restored to less than 180 °F.
- RCP Controlled Bleedoff flow greater than 3.0 gpm.
- RCP Motor Stator Winding Temperature alarm with stable trend.
- RCP Upper Thrust Bearing Temperature greater than 212˚F (Ops Management discretion)
- RCP Lower Thrust Bearing Temperature greater than 200˚F (Ops Management discretion)

There are some actions required to be taken depending upon the event that are very important but may not necessarily be evident as to why they are important. Below is a list of some of the actions required and the reasoning behind them:

- On a loss of CCW flow to the RCPs for greater than 10 minutes, after the plant has been tripped and the pumps stopped, Controlled Bleedoff from the RCPs is isolated to both the VCT and the Quench Tank. This is done because there is still a large differential pressure across the seal which would cause the flow of HOT Reactor Coolant through the seals without anyway to cool it off. This would essentially "cook" the seals increasing the possibility of the seals failing in a static condition. With Controlled Bleedoff completely isolated, the heat input to the seals is minimized.
- If ANY RCP Vapor Seal pressure is greater than 1500 psia trip the reactor and then trip the pump. This is done because if the Vapor Seal pressure is greater than 1500 psia, that is an indication of severe degradation of the lower three seals. The Vapor Seal is designed to hold full pressure of 2500 psia in a static condition (pump not running) and during coast down following the failure of the other three seals. The longer the pump is operated with this pressure on the vapor seal, the greater risk of the vapor seal failing.
- If Controlled Bleedoff or Lower Seal Cavity temperatures are greater than 300 °F use the PMS points listed in the procedure to monitor seal cooldown. This is done because the RCP recorders on 2C-14 that are normally monitored are only calibrated to 350 °F.
- If either the "A" or the "B" RCP are secured, verify the associated pressurizer spray valves are in MANUAL and closed. This is to prevent "short cycling" pressurizer spray.

15.1.5 TOTAL AND PARTIAL LOSS OF REACTOR COOLANT FORCED FLOW

15.1.5.1 Identification of Causes

A loss of reactor coolant forced flow can result from the occurrence of a mechanical or electrical failure. A partial loss of flow can occur as the result of a mechanical or electrical failure in a reactor coolant pump or from a loss of power to the pump bus. A complete loss of coolant flow results from a simultaneous loss of electrical power to all operating reactor coolant pumps.

During power operation, electrical power to the reactor coolant pumps is supplied through buses from the unit AC power source, i.e., main generator. An alternate supply to the pump buses is also provided from the preferred (off-site) AC power source. Automatic fast transfer is provided to restore the preferred AC power source to the auxiliary power distribution system in the event that the unit AC power source is lost for any reason. If the main generator trips, the pump buses are transferred and there is no significant effect on coolant flow through the core. See Chapter 8 for further discussion of the electrical system.

15.1.5.1.1 Loss of Coolant Flow Resulting From an Electrical Failure

Loss of flow during power operation causes an increase in reactor coolant temperature and results in a reduction in the margin to DNB. The low DNBR trip will prevent the minimum DNBR from falling below allowable limits at any time during the transient.

In the event that this does occur, the flow coastdown energy of the pump motor assembly determines the rate at which the core flow rate drops. The decrease in the reactor coolant flow rate during a free-wheeling coastdown is primarily governed by the rotational inertia of the reactor coolant pump and motor, including a flywheel. However, if the coolant pumps remain connected to their power supply buses, a sufficiently rapid transient frequency reduction could force the coolant pump induction motors to develop negative torque and act as generators. This would cause a coolant flow reduction more rapid than that caused by a loss of power, since some of the kinetic energy of the coolant and flywheel would be converted to electrical energy through the pump and motor.

An analysis demonstrating that an underfrequency condition will not prevent the pumps from performing their coastdown function is contained in Section 15.1.5.2.2.1.

In the event that this does occur, the flow coastdown energy of the pump-motor assembly determines the rate at which the core flow rate drops.

The 4-pump loss of flow is presented because it is the most severe loss of flow resulting from an electrical failure. Although the loss of power to one of the 2-pump buses results in loss of power to one pump in each loop the loss of power to two pumps in the same loop is presented as a limiting case for a 2-pump loss of flow.

15.1.5.1.2 Loss of Coolant Flow Resulting From a Pump Shaft Seizure

The single reactor coolant pump shaft seizure is not an anticipated operational occurrence. This accident is postulated to occur as a consequence of an instantaneous seizure of the pump shaft due to a mechanical failure. The reactor coolant flow is rapidly reduced to the 3-pump value. Since a rapid reduction in coolant flow results in a rapid reduction in the margin to DNB, a low DNBR trip will occur.

Question 38

Bank:	2368	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	38	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC Exam Bank 0416		
Search	078000A401	**10CFR55:**	41.7	**Safety Function**	8				
Title:	Instrument Air System (IAS)	**System Number**	078	**K/A**	A4.01				
Tier:	2	**Group:**	1	**RO Imp:**	3.1	**SRO Imp:**	3.1	**L. Plan:**	A2LP-RO-ALIA
OBJ	3								

Description: Ability to manually operate and/or monitor in the control room: - Pressure gauges

Question:

Given the following with the unit at full power:

* "INSTR AIR PRESS HI/LO" annunciator (2K12 A8) comes in alarm.
* OP-2203.021, Loss of IA AOP, has been implemented.
* Instrument Air pressure is 34 psig on 2PIS-3013 and slowly lowering .
* Unit 1 informs Unit 2 that Unit 1 IA Header Pressure is 70 psig and slowly lowering.
* D/P between IA HDR pressure and IA receiver pressure is 3 psid and steady.

Based on these conditions which of the following actions is REQUIRED to be taken as directed by the Loss of IA AOP?

A. Close the IA to Unit 1 Isolations 2CV-3004 and 2CV-3015.

B. Open the IA Air Dryer Bypass Isolation Valve 2IA-8.

C. Trip the Reactor and perform SPTA procedure 2202.001.

D. Align Service Water Return headers to Lake Dardanelle.

Answer:

C. Trip the Reactor and perform SPTA procedure 2202.001.

Notes:

C is correct IAW step 6 of the Loss of IA AOP.

A is incorrect: The cross connect valves should be closed only if Unit 1 IA header pressure drops below 60 psig. This is a plausible action directed by the AOP if IA pressure on Unit 1 drops to 60 psig. This would be the IA pressure on Unit 1 if the event is on Unit 2 (Step 5 of the Loss of IA AOP)

B is incorrect: The IA Air Dryer Bypass Isolation Valve 2IA-8 should be opened based on IA DP between the header and IA Receiver pressure indicating a block IA dryer or filter. This is a plausible action as directed by the AOP step 8 CA 8.B if the DP between the IA Header Pressure and IA Receiver exceeds 10 PSID.

D is incorrect because SW Returns should be aligned to the ECP not Lake Dardanelle but plausible as it is step 14 of the Loss of IA AOP.

This question matches the K&A because the operator must be able to monitor the IA header pressure indication and operate the controls of the plant to comply with the direction in the AOP.

References:

2203.012L ANNUNCIATOR 2K12 CORRECTIVE ACTION Rev. 49 Window A-8 INSTR AIR PRESS HI-LO (Verified reference updated 11/15/16); AOP 2203.021 Loss of IA Rev. 18 pages 1-6 (Verified reference updated 11/15/16);

Historical Comments:

NRC Exam Bank 0416 was used on the 2002 NRC Exam, however, stem and distractors were altered to be more plausible and order of correct answer was changed. To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Added "on 2PIS-3013" to bullet #3, Added "Unit 1 informs Unit 2 that" to Bullet #4. Added " Based on these conditions" to the stem.

ANNUNCIATOR 2K12

A-8

INSTR AIR PRESS HI/LO

1.0 CAUSES

1.1 HIGH - Instrument Air Filter outlet pressure (2PIS-3013) ≥ 110 psig

1.2 LOW - Instrument Air Filter outlet pressure (2PIS-3013) ≤ 80 psig

2.0 ACTION REQUIRED

NOTE

High header pressure is indicative of unloader failure.

2.1 Check pressure on 2PIS-3013.

2.2 IF HIGH pressure,
THEN shift IA Compressors using Instrument Air System (2104.024).

2.3 IF LOW Pressure,
THEN GO TO Loss of Instrument Air (2203.021).

3.0 TO CLEAR ALARM

3.1 Restore header pressure to between 80 and 110 psig.

4.0 REFERENCES

4.1 E-2457-4

LOSS OF INSTRUMENT AIR

PURPOSE

This procedure provides actions for Degraded or Loss of Instrument Air conditions. Either event could impact the Operators ability to maintain the plant in a stable condition.

ENTRY CONDITIONS

1. IA header pressure less than 80 psig and lowering.

2. "INSTR AIR PRESS HI/LO" annunciator (2K12-A8) in alarm.

3. Unexplained reduction in IA header pressure.

EXIT CONDITIONS

BOTH of the following conditions exist:

1. IA header pressure 80 psig or greater.

2. Air-operated control systems and components restored to operation.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE
• Steps marked with (*) are continuous action steps. • Steps marked with (■) are floating steps.

1. **OPEN** Placekeeping Page.

2. **NOTIFY** Control Board operators to monitor Floating Steps.

3. **ENSURE** IA cross-connected with Unit 1 as follows:

   A. **ENSURE** IA Cross-connect valves open:

   - 2CV-3004
   - 2CV-3015

   B. **INFORM** Unit 1 of IA cross-connect status.

   C. **<u>IF</u>** EC-28743 installed on Breathing Air System for Unit 1 Main Turbine,
   <u>THEN</u> REQUEST Unit 1 locally ensure "BA OUTLET ISOL" (ISOL-1) closed.

*4. **<u>IF</u>** event on Unit 1,
<u>AND</u> Unit 2 IA header pressure drops below 60 psig,
<u>THEN</u> SECURE cross connect as follows:

   A. **CLOSE** IA Cross-connect valves:

   - 2CV-3004
   - 2CV-3015

   B. **INFORM** Unit 1 control room.

   C. **<u>WHEN</u>** Unit 2 IA pressure restored greater than 80 psig,
   <u>THEN</u> EXIT this procedure.

INSTRUCTIONS	CONTINGENCY ACTIONS
*5. **<u>IF</u>** event on Unit 2, **<u>AND</u>** Unit 1 IA header pressure drops below 60 psig, **<u>THEN</u> SECURE** cross-connect with Unit 1 as follows: A. **CLOSE** IA Cross-connect valves: • 2CV-3004 • 2CV-3015 B. Locally **ENSURE** the following valves closed: • MANUAL X-CONNECT TO UNIT 1 (2IA-47) • MANUAL X-CONNECT TO UNIT 1 (2IA-48) • 2F-37 INLET FROM DRY HDR (2IA-192)	
■ 6. **CHECK** IA header pressure greater than 35 psig.	■ 6. **<u>IF</u>** Plant in Mode 1 **<u>THEN</u> PERFORM** the following: A. **MANUALLY** trip Reactor. B. **PERFORM** 2202.001, Standard Post Trip Actions, in conjunction with this procedure.
7. **PERFORM** the following: • **DISPATCH** local operator to investigate. • **INFORM** local operator to refer to local Exhibit 1, Loss of Instrument Air Local Checks.	

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

Symptoms checked by the following TWO steps may be masked with IA cross-connected. Consider closing IA cross-connect valves.

8. Locally **CHECK** BOTH IA receivers pressure greater than 85 psig.
   - "2T88A PRESS IND" 2PI-3033
   - "2T88B PRESS IND" 2PI-3019

   Contingency 8. **IF** IA receivers pressure less than 85 psig, **THEN ENSURE** BOTH IA compressors running (refer to 2104.024, Instrument Air System).

9. Locally **CHECK** IA header pressure and air receivers pressure within 10 psid.
   - "IA MAIN SUPPLY HEADER" (2PIT-3013)
   - "2T88A PRESS IND" 2PI-3033
   - "2T88B PRESS IND" 2PI-3019

   Contingency 9. **IF** IA header and receivers pressure greater than 10 psid, **THEN** locally **PERFORM** the following as necessary:

   A. **OPEN** "COALESCING PREFILTER BYPASS" valve (2IA-186C).

   B. **OPEN** "AIR DRYER BYPASS" valve (2IA-8).

   C. **WHEN** time allows, **THEN PLACE** standby IA Dryer in service (refer to 2104.024, Instrument Air System).

   D. **PLACE** standby IA Filter in service, refer to 2104.024, Instrument Air System.

NOTE

Attachment B aligns critical components to their "fail safe" position to prevent inadvertent repositioning as IA pressure restores.

■ 10. **IF** AOVs have repositioned or are repositioning due to degraded IA pressure, **THEN PERFORM** Attachment B, Valve Switch Safe Positions, as required to prevent inadvertent repositioning.

INSTRUCTIONS	CONTINGENCY ACTIONS
15. **ALIGN** SW Return headers to ECP: A. **OPEN** SW Return header to ECP (2CV-1541-1). B. **ENSURE** SW Return Header to Lake (2CV-1481-1) closes. C. **OPEN** SW Return header to ECP (2CV-1560-2). D. **ENSURE** SW Return Header to Lake (2CV-1480-2) closes.	
16. **CHECK** Plant in Mode 1, 2, 3, or 4.	16. **IF** Plant in Mode 5 or 6, **THEN GO TO** Step 20.
*17. **CHECK** IA header pressure greater than 65 psig.	*17. **PERFORM** the following: A. **IF** Reactor trip occurs, **THEN PERFORM** 2202.001, Standard Post Trip Actions, in conjunction with this procedure. B. **ALIGN** SW to CNTMT Cooling fans as follows: 1) **IF** 2VSF-1A OR 2VSF-1B running, **THEN ENSURE** the following: a) 2VSF-1A/B SW CLG Inlet (2CV-1511-1) open. b) 2VSF-1A/B SW CLG Outlet (2CV-1519-1) open. 2) **IF** 2VSF-1C OR 2VSF-1D running, **THEN ENSURE** the following: a) 2VSF-1C/D SW CLG Inlet (2CV-1510-2) open. b) 2VSF-1C/D SW CLG Outlet (2CV-1513-2) open.

ORIGINAL QUESTION 0416 WAS USED TO DEVELOP 2017 NRC EXAM QID#38 (Used on 2002 NRC Exam)

Questions For All QID In Exam Bank

Bank:	0416	**Rev:**	000	**Rev Date:**	12/8/2001 10:51:2	**QID #:**		**Author:**	Coble
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New		
Search		**10CFR55:**		**Safety Function**					
System Title:	Loss of Instrument Air	**System Number**	065	**K/A**	AA2.06				
Tier:		**Group:**		**RO Imp:**		**SRO Imp:**	4.2	**L. Plan:**	
OBJ									
Description:	Ability to determine and interpret when to trip the reactor if instrument air pressure is decreasing as it applies to Loss of Instrument Air.								

Question:

Given the following:

* The plant is at 100% Power.
* Annunciator 2K12 A-8, INSTR AIR PRESS HI/LO, comes in.
* The CBOT reports that Instrument Air (IA) Header Pressure is 75 psig and dropping.

Which ONE of the following is the correct action to take if IA Header Pressure continues to drop?

A. At 35 psig, trip the Reactor and commence Standard Post Trip Actions.

B. At 40 psig, align Service Water to the Containment fan cooling units.

C. At 60 psig, open the IA cross-connect isolation valves from Unit 1.

D. At 65 psig, start the temporary IA Compressor and enter Loss of IA AOP.

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☒

Answer:

A. At 35 psig, trip the Reactor and commence Standard Post Trip Actions.

Notes:

The Loss of IA AOP entry conditions is 80 psig IA pressure and lowering so the AOP should have already been entered in answer A. Answer B is wrong because the cross-connect isolations should be opened at 80 psig and dropping and then closed at 60 psig and dropping to prevent a loss of IA on Unit 1. A loss of IA will cause an isolation of Main Chilled Water to the Containment Cooling Units so an option would be to line up Service Water to the units; however, this is not a step directed by the Loss of IA AOP thus answer C is wrong. The procedure directs tripping the unit at 35 psig IA header pressure.

References:

ANO-2-LP-RO-EAOP, Revision 5, Objective 16
OP 2203.012L, Annunciator 2K12 Corrective Action, Revision 030-02-0, Window A-8, IA Press Hi/LO
OP 2203.021, Loss of IA AOP, Revision 008-01-0, Entry Conditions, Step 4, and Step 5.

Historical Comments:

Question 39

Bank:	2369	**Rev:**	2	**Rev Date:**	12/9/2016	**2017 TEST QID #:**	39	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC EXAM BANK 1520		
Search	007000K401	**10CFR55:**	41.7	**Safety Function**	5				
Title:	Pressurizer Relief Tank/Quench Tank System (PRTS)	**System Number**	007	**K/A**	K4.01				
Tier:	2	**Group:**	1	**RO Imp:**	2.6	**SRO Imp:**	2.9	**L. Plan:**	A2LP-RO-RCS
OBJ	25								

Description: Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following: - Quench tank cooling

Question:

Given the following:

* The plant is at full power with indications of a Pressurizer Safety Valve leaking.
* The Quench Tank temperature has risen above its alarm limit.
* The CRS directs the crew to cool the Quench Tank using the normal feed and bleed method.

To ensure the sparger in the Quench Tank remains covered during this evolution, Quench Tank level should be maintained greater than a MINIMUM of ________ with makeup water aligned while draining the Quench Tank to the _________________.

A. 75%; Reactor Drain Tank

B. 75%; Containment Sump

C. 87%; Reactor Drain Tank

D. 87%; Containment Sump

Answer:

A. 75%; Reactor Drain Tank

Notes:

A is correct as the Quench Tank(QT) can be aligned to drain to the RDT through 2CV-4692. The minimum allowed level in the Quench Tank is 75% to ensure the sparger remains covered to quench any hot fluid coming into the tank.

B is incorrect as the QT cannot drain to the Containment sump but plausible as this in the minimum level allowed and the QT can be vented or relieved to the containment sump.

C is incorrect as this is the max level to go to in the QT to allow for a open non-liquid space for Nitrogen cover gas and non condensables but plausible as this is the correct location to drain the QT.

D is incorrect but plausible as this is a required level to maintain in the QT and a flow path can be obtained to the containment sump through a relief valve, rupture disc, or vent.

This question matches the K&A as this item requires knowledge of the design features that provide for QT cooling when needed.

References:

2203012J ANNUNCIATOR 2K10 CORRECTIVE ACTION REV 43 Window F-4 QUENCH TANK TEMP HI (Verified reference updated 11/15/16); NOP 2103.007 Quench Tank OPS Rev. 26 Section 7.7 (Verified reference updated 11/15/16);
STM_2-52_19-1 Quench Tank and RDT Integrated Drawing (Verified reference updated 11/15/16);

STM 2-03, 23-1 RCS Section 2.3 Quench Tank (Verified reference updated 11/15/16)

Historical Comments:

NRC EXAM BANK 1520 Used on the 2008 NRC Exam
To be used on the 2017 NRC Exam but updated with more plausible distractor levels.

REV. 1 based on NRC Chief Examiner Feedback BNC. Added "a minimum of" to the stem prior to the first blank.

REV. 2 based on NRC Chief Examiner Feedback BNC. Capitalized the word "minimum" prior to the first blank in the stem.

ANNUNCIATOR 2Kl0

F-4

QUENCH TANK TEMP HI

1.0 CAUSES

1.1 Quench Tank temperature (2TIS-4694) > 125°F

2.0 ACTION REQUIRED

2.1 Check Quench Tank temperature (2TIS-4694) and PMS/PDS trends.

2.2 Attempt to determine leaking Pressurizer relief by monitoring individual downstream temperatures:

- Pressurizer Relief 2PSV-4633 (2TIS-4630)
- Pressurizer Relief 2PSV-4634 (2TIS-463l)

2.3 IF during Heatup or Hot Standby,
THEN locally check Relief valves and ECCS vents for leakage.

2.4 Reduce Quench Tank temperature as necessary using Quench Tank and Reactor Drain Tank Ops (2103.007).

3.0 TO CLEAR ALARM

3.1 Lower Quench Tank temperature to < 125°F.

4.0 REFERENCES

4.1 E-2456-3

7.7 Cooling Quench Tank Using Feed and Bleed (Normal Method)

7.7.1 Verify system aligned IAW System Alignment Verification section of 1015.001, Conduct of Operations.

* 7.7.2 IF NOT vented to atmosphere, THEN maintain greater than 1 psig nitrogen overpressure in Quench Tank IAW applicable "Adding N2 to Quench Tank" step of this procedure. (CR-ANO-2-2000-0208.

* 7.7.3 Monitor the following:

- Quench Tank level (2LIS-4694/L4694)
- Quench Tank temperature (2TIS-4694/T4694)

7.7.4 Close Rx Drain Tank Makeup Water Isol 2CV-4693 (2HS-4693).

7.7.5 Open Containment Makeup Water Supply 2CV-4690-2 (2HS-4690-2).

* 7.7.6 Monitor Reactor Drain Tank level (2LIS-2200/L2200) AND lower level as needed IAW applicable section of this procedure.

* 7.7.7 Throttle open the following valves as needed to maintain Quench Tank level between 75% and 87% (2LIS-4694/L4694):

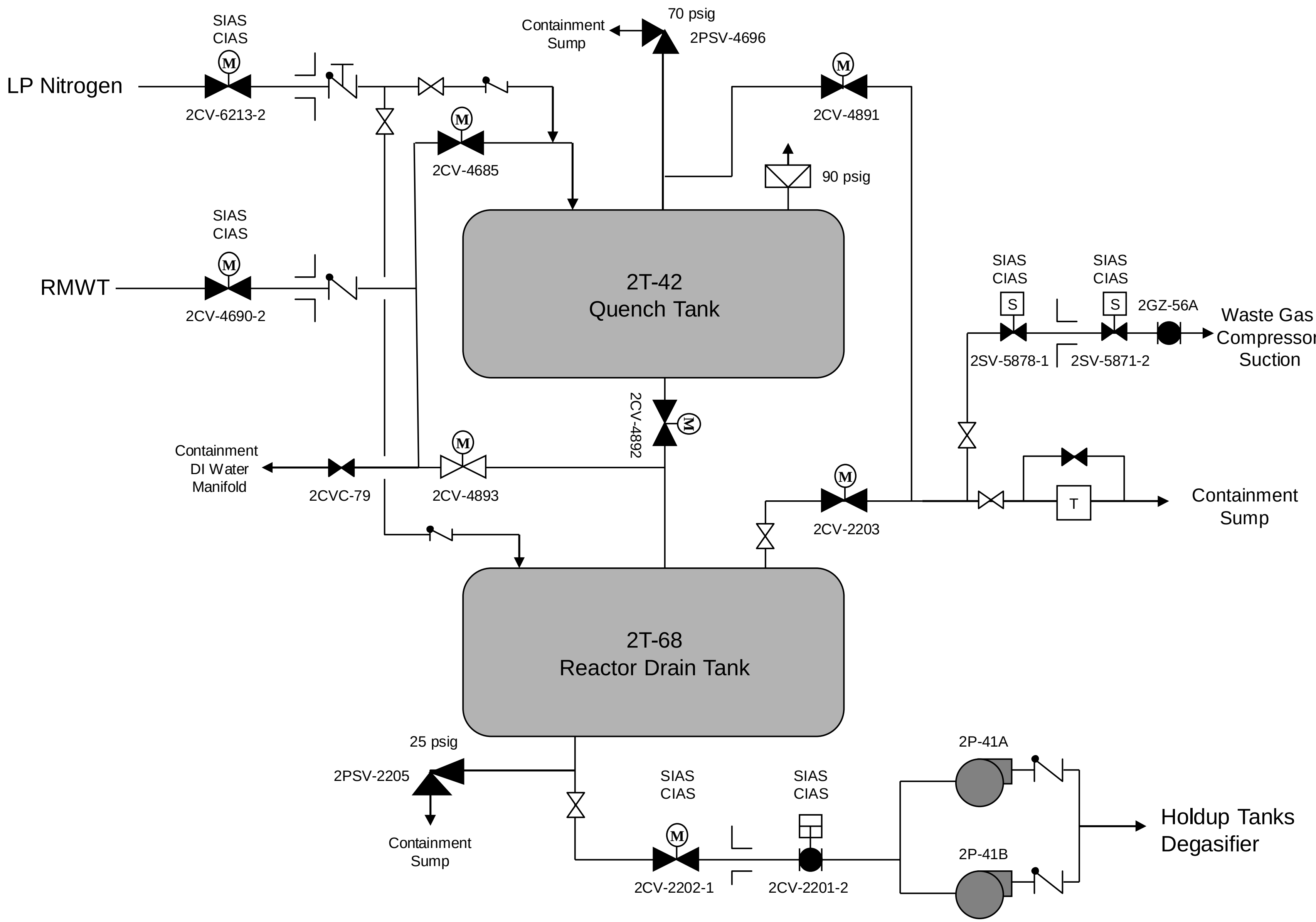
- Quench Tank Supply 2CV-4685 (2HS-4685)
- Quench Tank Outlet 2CV-4692 (2HS-4692)

7.7.8 WHEN Quench Tank reaches desired temperature, THEN close the following valves:

- Containment Makeup Water Supply 2CV-4690-2 (2HS-4690-2).
- Quench Tank Supply 2CV-4685 (2HS-4685)
- Quench Tank Outlet 2CV-4692 (2HS-4692)

{4.3.2} 7.7.9 **Open Rx Drain Tank Makeup Water Isol 2CV-4693 (2HS-4693).**

Quench Tank and Reactor Drain Tank Integrated Drawing



indication of a decrease in the spray valve bypass flow used to minimize thermal shock and ensure adequate chemical mixing. During plant transients at reduced power levels, the low alarm may come in due to an in surge from the RCS.

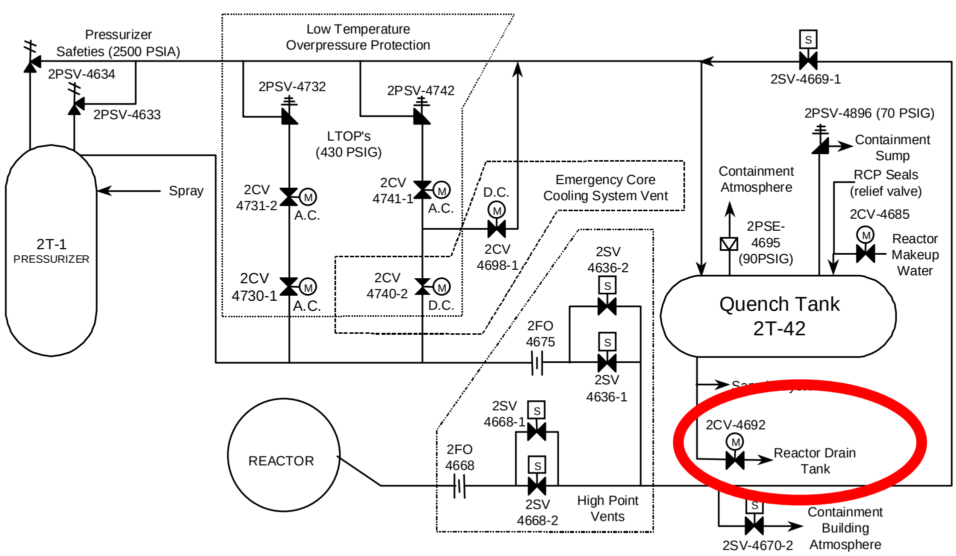
2.2.5.5 Pressurizer Surge Line Temperature Instrument

The pressurizer surge line temperature element, 2TE-4609, provides an input to PMS, and a temperature indicating switch, 2TIS-4609, in the Control Room on 2C-04. An annunciator, 2K10-E5 (“SURGE LINE TEMP HI/LO”), is activated when surge line temperature exceeds 695 °F or drops below 605 °F.

The high temperature alarm indicates a high pressure condition in the RCS; if actual RCS pressure is < 2300 psia then the alarm is considered invalid. The low temperature alarm indicates a decrease in the spray valve bypass flow used to minimize thermal shock and ensure adequate chemical mixing. During plant transients at reduced power levels, the low alarm may come in due to an insurge from the RCS.

2.3 Quench Tank -- 2T-42

The Quench Tank, 2T-42, is designed to receive and condense the normal discharges from the pressurizer safety valves and prevent the discharge from being released to the Containment Building



atmosphere. Tank volume is based on the need to condense steam from the pressurizer safety valves resulting from a loss of load followed immediately by an uncontrolled Control Element Assembly withdrawal. This also assumes no coolant letdown or pressurizer spray.

An additional function of the quench tank is to collect RCP bleedoff, via 2CV-4856 and 2PSV-4836, in the event that normal control bleedoff to the Volume Control Tank is isolated on a Containment Isolation Actuation Signal (CIAS). The Low

The quench tank is a horizontal cylindrical tank located in the Containment Building approximately 20 feet above the 335" elevation in the "A" RCP cavity. The total volume of this tank is 254 ft^3. It is placed at the 353' elevation to ensure that the safety valve header is kept drained.

The quench tank volume is maintained at approximately 210 ft^3, this volume is needed to condense the steam released from the safety valves. The pressurizer safety valve exhaust piping discharges under the water via the sparger in the quench tank. The sparger helps disburse the steam under the water level to enhance condensing of the steam. The tank gas volume (44 ft^3) is based on limiting the maximum tank pressure to 70 psig.

The quench tank is protected from overpressure by a relief valve and a rupture disc. The relief valve, 2PSV-4696, has a setpoint of 70 psig and relieves to the containment sump. The rupture disc, 2PSE-4695, ruptures at 90 psig relieving pressure to the containment atmosphere.

When tank level is high, the quench tank can be drained to the Reactor Drain Tank, located directly underneath the quench tank via the Quench Tank Drain Valve, 2CV-4692. The handswitch for the drain valve is located in the Control Room on 2C-04 and can be throttled to vary the drain rate. To maintain the sparger in the quench tank covered the level should be maintained > 75%.

Water from the Reactor Makeup Water System can be added to the quench tank. The makeup is via the Reactor Makeup to Containment Isolation Valve, 2CV-4690-2, and the Quench Tank Makeup Isolation Valve, 2CV-4685. The handswitch for 2CV-4685 is on 2C-04 and can be throttled to vary the fill rate. The handswitch for 2CV-4690-2 is located on 2C-17 and receives a Containment Isolation Actuation Signal (CIAS) close signal.

The quench tank is required to have a nitrogen overpressure on it to prevent a buildup of an explosive hydrogen mixture should any leak in from the pressurizer. The quench tank over pressure can be maintained with Low Pressure Nitrogen via a manual isolation valve, $2N_2$-2, located on the 335' elevation of the containment outside the biological shield wall.

The quench tank can be vented to the Containment Vent Header via the Quench Tank Vent Valve, 2CV-4691. This valve is controlled from 2C-04.

The quench tank temperature is monitored by 2TE-4694. The temperature element inputs into 2TIS-4694 on 2C-04 and the Plant Computer. Annunciator 2K10-F4, "QUENCH TANK TEMP HI" is activated when 2TIS-4694 indicates > 125 °F. This could indicate a lifted or leaking pressurizer relief valve.

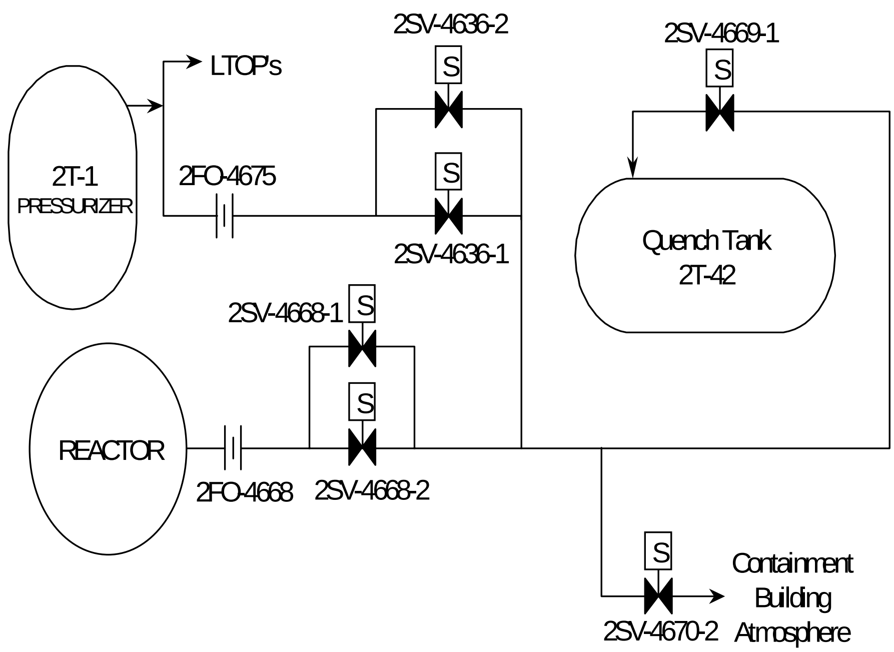
The quench tank level is detected by 2LT-4694. This transmitter provides input into 2LIS-4694 on 2C-04 and the Plant Computer.

Annunciator 2K10-E4, “QUENCH TANK LEVEL HI/LO”, is actuated at > 87% and < 73%.

Pressure transmitter, 2PT-4694, allows the quench tank pressure to be monitored on 2C-04 (2PIS-4694) and the Plant Computer. The “QUENCH TANK PRESSURE HI” annunciator, 2K10-D4, alerts the operator when pressure exceeds 40 PSIG.

2.4 Reactor Coolant System High Point Vents

The RCS High Point Vents are provided to exhaust non-condensable gases and/or steam that could inhibit natural circulation cooling in the RCS. Natural circulation uses the difference in temperature and elevation of the Reactor (heat source) and Steam Generators (heat sink) to provide the driving force for flow. In this mode of cooling, low flow areas such as the Reactor vessel head area may not be sufficiently cooled and have the fluid flash to steam creating a void. If the void grows large enough it can impede the



natural circulation and therefore needs to be removed. The high point vents are one method that can be used to aid in void removal.

In addition to the above function, the RCS High Point Vents are used to remove non-condensable gases from the RCS during RCS fill and provide a vent path during RCS draining activities.

Six solenoid valves are provided on the RCS for a means of venting the RCS. These “energize to open” solenoids provide a vent path from the top of the pressurizer (2SV-4636-1 and 2SV-4636-2) and the Reactor vessel head (2SV-4668-1 and 2SV-4668-2). The two remaining solenoids determine the vent path, either to the containment atmosphere (2SV-4670-2) or the quench tank (2SV-4669-1).

ORIGINAL BANK QUESTION 1520 WAS USED TO DEVELOP 2017 NRC EXAM QID#39 (Used on 2005 NRC Exam)

Questions For All QID In Exam Bank

Bank:	1520	**Rev:**	0	**Rev Date:**	10/30/2007 10:49:	**QID #:**	33	**Author:**	Coble
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW		
Search	007000K401	**10CFR55:**	41.7	**Safety Function**	5				
System Title:	Pressurizer Relief Tank/Quench Tank System (P	**System Number**	007	**K/A**	K4.01				
Tier:	2	**Group:**	1	**RO Imp:**	2.6	**SRO Imp:**	2.9	**L. Plan:**	A2LP-RO-RCS
OBJ									

Description: Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following: - Quench tank cooling.

Question:

Given the following:

* The plant is at full power with indications of a Pressurizer Safety Valve leaking.
* The Quench tank temperature has risen above its alarm limit.
* The CRS directs the crew to cool the Quench Tank using the normal feed and bleed method.

To ensure the sparger in the Quench Tank remains covered during this evolution, tank level should be maintained greater than ________ with makeup water aligned while draining the Quench Tank to the _________________.

A. 75%; Reactor Drain Tank

B. 75%; Containment Sump

C. 55%; Reactor Drain Tank

D. 55%; Containment Sump

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☑	☑

Audit Exam History

2006	☐
2008	☐

Answer:

A. 75%; Reactor Drain Tank

Notes:

The quench tank can be aligned to drain to the RDT through 2CV-4692. It cannot be aligned to drain to the Containment sump unless a tank relief opens or rupture disc ruptures. The minimum allowed level in the Quencl Tank is 75% to ensure the sparger remains covered to quench any hot fluid coming into the tank.

References:

OP 2103.007 Section 7.5
STM 2-03, RCS, Section 2.3 Quench Tank.

Historical Comments:

Question 40

Bank:	2370	**Rev:**	1	**Rev Date:**	7/25/2016	**2017 TEST QID #:**	40	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC Exam Bank 1654		
Search	061000K602	**10CFR55:**	41.7	**Safety Function**	4				
Title:	Auxiliary / Emergency Feedwater (AFW) System	**System Number**	061	**K/A**	K6.02				
Tier:	2	**Group:**	1	**RO Imp:**	2.6	**SRO Imp:**	2.7	**L. Plan:**	A2LP-RO-ESPTA
OBJ	11								

Description: Knowledge of the effect of a loss or malfunction of the following will have on the AFW System components: - Pumps

Question:

Given the following:

* The plant is at full power.
* A Loss of Main Feed Water causes a plant trip.
* An EFAS signal was generated.
* EFW Pump 2P-7A tripped on overspeed during start.
* OP-2202.001 SPTAs has been entered.
* SG A level is 6% NR and slowly rising.
* SG B level is 4% NR and slowly rising.
* EFW Pump 2P-7B started on the EFAS signal.

Which of the following combinations of EFW flow would indicate that EFW Pump 2P-7B is adequately providing the LOWEST amount of flow required to satisfy the RCS Heat Removal Safety Function Requirements during SPTAs? (assume no EFW recirculation flow)

A. A SG 145 gpm; B SG 145 gpm

B. A SG 248 gpm; B SG 242 gpm

C. A SG 234 gpm; B SG 244 gpm

D. A SG 155 gpm; B SG 152 gpm

Answer:

B. A SG 248 gpm; B SG 242 gpm

Notes:

B is correct: Refer to SPTAs Step 8. If the SGs are less than 10% NR, then the minimum total FW flow to remove 100% decay heat load is 485 gpm. Each EFW pump is considered 100% capacity and thus 2P-7B should be able to provide a total flow of >485 gpm. 248 + 242 = 490 gpm which is >485 gpm.

A is incorrect as the total flow is not >485 gpm but plausible because if no feed flow is currently going to a SG during a Loss of Feed Event (Sustained loss of flow), then the Loss of FW EOP Step 12 directs establishment of a feed source to a SG IAW Standard Attachment 53 for 2P-7B. This Attachment has you start the pump and establish FW flow at less than 150 gpm to each SG for 5 minutes or until a SG level rise is noted. This is to prevent damage to the feed ring due to the thermal transient.

C is incorrect as the total flow is not > 485 gpm but plausible as it is very close. 234 + 244 = 478 gpm.

D is incorrect but plausible because if no feed flow is currently going to a SG during a Loss of Feed Event (Sustained loss of flow), then the Loss of FW EOP Step 12 directs establishment of a feed source to a SG IAW Standard Attachment 53 for 2P-7B. This Attachment has you start the pump and establish FW flow at less than 150 gpm to each SG fro 5 minutes

or until a SG level rise is noted. This is to prevent damage to the feed ring due to the thermal transient. Plausible if applicant believes > 150 gpm is the minimum flow required for these conditions.

This question matches the K&A as the candidate must have the knowledge of the minimum amount of FW flow needed to satisfy RCS decay heat removal to be able to evaluate the effect of the loss of 2P-7A will have on the EFW system.

References:

EOP 2202.001 SPTAs Rev. 15 Step 8 and the associated TG Step 8 (Verified reference updated 11/15/16); EOP 2202.006 Loss of MFW REV. 11 Step 12 (Verified reference updated 11/15/16) (Verified reference updated 11/15/16); Standard Attachments 2202.010 REV 23 Attachment 53 Feed with 2P-7B (Verified reference updated 11/15/16).

Historical Comments:

NRC Exam Bank 1654 was used on the 2009 NRC Exam
To be used on the 2017 NRC Exam.

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed this QID from a "Modified Bank" question to just a "BANK" question due to being too similar to the original 1654 Bank question.

INSTRUCTIONS	CONTINGENCY ACTIONS
8. Check RCS Heat Removal:	
A. Check SG available by BOTH of the following:	A. Perform the following:
___ • At least ONE SG level 10% to 90%.	___ 1) IF SG level lowering, THEN verify EFAS actuated.
___ • FW maintaining SG level.	___ 2) IF SG level less than 10%, THEN verify total flow greater than 485 gpm.
	3) IF EITHER SG level greater than 90%, THEN perform the following:
	a) Verify BOTH MFW pumps tripped.
	b) Verify EFW isolated to appropriate SG.
	4) IF FW NOT maintaining SG level, THEN manually control FW flow rate.

(Step 8 continued on next page)

EOP STEP:

8. **Check RCS Heat Removal:**

EPG STEP:

7.

DEVIATION? Yes

BASES FOR DEVIATION:

The EOP, as is the EPG, is designed to check that at least one steam generator is available for removing heat from the RCS. The SG level band specified in the EOP bounds the range of levels expected following a relatively uncomplicated reactor trip, and ensures that under non-harsh containment conditions, level is within the indicating range of the SG narrow range meter (1). Level falling outside this range is an unexpected condition and warrants additional operator attention. Although the level range is not specified as being narrow range, the operators will know to use the narrow range meter since its output is in percent and the wide range instrument's output is in inches. The level band specified by the EOP is consistent with the intent of the EPG in that it ensures a SG with sufficient level to provide for RCS heat removal is available. The EOP checks that feedwater is maintaining SG level instead of the EPG criteria of "available". This ensures that SG level will not drop to unacceptable levels and result in a loss of RCS Heat Removal.

The specified level band does permit SG level to fall below the SG feedring without restricting flow in order to prevent damage to the SG feedring. This is a normal post trip occurrence at ANO-2 and has been determined to be acceptable as long as feed flow is restored in the time frame of the SPTAs, and a sustained loss of feed flow has not occurred. If a sustained loss of flow has occurred, then either the LOF ORP or the FRP will limit flow during the flow re-initiation phase to 150 gpm.

If the SG availability criteria are not met, the EOP provides contingency actions to verify that EFAS actuated if FW not maintaining level and that flow is greater than the minimum needed for removal of the expected decay heat rate following a reactor trip if level less than 10% (3). This ensures that if SG level did not meet the criteria due to being low, that the emergency feedwater system has actuated to help restore level. The EOP also specified that if either SG level is greater than 90%, to verify that both MFW pumps are tripped, and that EFW flow control and block valves are isolated to the affected SG. This substep was added to prevent SG overfill and associated excessive RCS cooldown in the event of an overfeed condition. This also minimizes the possibility of moisture carryover. If the SG availability criteria is not met, action is given to trip one RCP in each loop and to secure blowdown.
This reduces RCS heat input and minimizes SG inventory losses. Two reactor coolant pumps are tripped as a conservative measure to reduce RCS heat input as early in the procedure as possible vice waiting until the event is diagnosed and either the LOF ORP or FRP is implemented as is done

INSTRUCTIONS	CONTINGENCY ACTIONS
11. <u>IF</u> FW established to at least ONE SG, <u>THEN</u> GO TO Step 16.	
■12. Establish a SG feed source from at least one of the following (listed in preferred order): A. EFW Pump 2P7B using 2202.010 Attachment 53, Recovery From Loss of Feed With 2P7B. B. EFW Pump 2P7A using 2202.010 Attachment 54, Recovery From Loss of Feed With 2P7A. C. AFW Pump 2P75 using 2202.010 Attachment 55, Recovery From Loss of Feed With 2P75. D. MFW Pumps using 2202.010 Attachment 56, Recovery From Loss of Feed With Main Feed Pumps. E. Condensate Pumps using 2202.010 Attachment 57, Recovery From Loss of Feed With Condensate Pumps.	
***13. <u>IF</u> SG feed flow established from MFW pumps or Condensate pumps, <u>THEN</u> maintain Condenser Hotwell level greater than 38%.**	

ATTACHMENT 53

RECOVERY FROM LOSS OF FEED WITH 2P7B

1. Verify ONE of the following EFW Pump 2P7B Suction valves open:

   - EFW Pump Suction Source From CST (2CV-0789-1)
   - EFW Pump Suction Source From SW (2CV-0716-1)

2. IF desired to fill SG A,
   THEN perform the following:

   A. IF MSIS or EFAS actuated,
      THEN override 2P7B Feed to SG A (2CV-1025-1) by placing 2HS-1025B-1 to MSIS override AND then to EFAS override.

   B. Verify 2P7B Flow Control Valve To SG A (2CV-1025-1) closed.

   C. IF MSIS or EFAS actuated,
      THEN override 2P7B Discharge to SG A (2CV-1038-2) by placing handswitch to Closed (MSIS) and then to Open (EFAS).

   D. Verify 2P7B Discharge to SG A (2CV-1038-2) open.

3. IF desired to fill SG B,
   THEN perform the following:

   A. IF MSIS or EFAS actuated,
      THEN override 2P7B Feed to SG B (2CV-1075-1) by placing 2HS-1075B-1 to MSIS override AND then to EFAS override.

   B. Verify 2P7B Flow Control Valve To SG B (2CV-1075-1) closed.

   C. IF MSIS or EFAS actuated,
      THEN override 2P7B Discharge to SG B (2CV-1036-2) by placing handswitch to Closed (MSIS) and then to Open (EFAS).

   D. Verify 2P7B Discharge to SG B (2CV-1036-2) open.

4. Start 2P7B.

5. IF 2P7B will NOT start,
   THEN perform the following:

   A. Locally attempt to start 2P7B from breaker 2A-311, "EFWP 2P7B SUPPLY".

   B. IF 2P7B will NOT start,
      THEN **RETURN TO** procedure in effect.

CAUTION
Feed to an impacted SG (less than 49% [60%] level) should be maintained less than 150 gpm until SG level rises or flow has been maintained for greater than 5 minutes.

6. IF SG level(s) less than 49% [60%],
   THEN perform the following:

   A. Throttle open the associated 2P7B Flow Control Valve(s) to establish flow of less than 150 gpm:

      - 2P7B Flow Control Valve to SG A (2CV-1025-1)
      - 2P7B Flow Control Valve to SG B (2CV-1075-1)

   B. Maintain feed flow to SG less than 150 gpm until a SG level rise is noted or feed flow maintained for greater than 5 minutes.

7. Throttle the associated 2P7B Flow Control Valve(s) to Restore SG level(s) to 60% [70%]:

   - 2P7B Flow Control Valve to SG A (2CV-1025-1)
   - 2P7B Flow Control Valve to SG B (2CV-1075-1)

8. IF EFW flow from 2P7B can NOT be established,
   THEN **RETURN TO** procedure in effect.

ORIGINAL QUESTION 1654 WAS USED TO DEVELOP 2017 NRC EXAM QID#40 (Used on 2009 NRC Exam)

Data for 2009 NRC RO/SRO Exam

Bank: 1654	**Rev:** 1	**Rev Date:** 7/24/2009 2:48:47		**QID #:** 45		**Author:** Coble	
Lic Level: R	**Difficulty:** 3	**Taxonomy:** H	**Source:** New				
Search 061000K502	**10CFR55:** 41.10		**Safety Function** 4				
System Title: Auxiliary / Emergency Feedwater (AFW) Syste			**System Number** 061		**K/A** K5.02		
Tier: 2	**Group:** 1	**RO Imp:** 3.2	**SRO Imp:** 3.6	**L. Plan:** A2LP-RO-ESPTA		**OBJ** 11	

Description: Knowledge of the operational implications of the following concepts as they apply to the AFW System: - Decay heat sources and magnitude

Question:

Given the following:

* The plant has tripped due to a Loss of Off Site Power (LOOP)
* The plant had been at 100% power for the last 100 days
* SPTAs are in progress
* SG A level is 8% NR
* SG B level is 7% NR
* All EFW components actuated as designed

Which ONE of the following combinations of EFW flow would be the MINIMUM required to meet the RCS Heat Removal Safety Function just after the trip? (assume no EFW recirculation flow)

A. A SG 150 gpm; B SG 150 gpm

B. A SG 315 gpm; B SG 155 gpm

C. A SG 211 gpm; B SG 275 gpm

D. A SG 250 gpm; B SG 260 gpm

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐
2009	☑	☑

Audit Exam History

2009	☐

Answer:

C. A SG 211 gpm; B SG 275 gpm

Notes:

With SG level less than 10% Narrow Range, total EFW flow post trip from 100% power with at least a 30 day history at 100% power is required to greater than 485 gpm. Answer C is 486 gpm total. Distracter A and B do nc meet the minimum but may have to start out at 150 gpm to each SG in certain EOPs under loss of feed conditions Distracter D is more than the minimum required.

References:

EOP 2202.001, SPTAs, Rev. 9, Contingency Action Step 8 A.2
EOP TG 2202.001, SPTAs, Rev. 9 Step 8, RCS Heat Removal, Step 8
EOP Setpoint Document Setpoint F.3, 485 gpm EFW flow
Lesson Plan A2LP-RO-ESPTA, Rev. 8, Objective 11: Describe the major actions taken during the performance of SPTAs and the basis for each.

Historical Comments:

Question 41

Bank:	2371	**Rev:**	2	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	41	**Author:**	Coble		
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam				
Search	005000K112	**10CFR55:**	41.7	**Safety Function**	4						
Title:	Residual Heat Removal System (RHRS)	**System Number**	005	**K/A**	K1.12						
Tier:	2	**Group:**	1	**RO Imp:**	3.1	**SRO Imp:**	3.4	**L. Plan:**	A2LP-RO-SPRAY	**OBJ**	10

Description: Knowledge of the physical connections and/or cause-effect relationships between the RHRS and the following systems: - Safeguard pumps

Question:

Given the following:

* Plant has just shutdown and is cooling down for a refueling outage.
* Electrical 4160 Vital AC Bus 2A3 maintenance outage in progress.
* RCS Pressure is currently 80 psig.
* The running SDC Pump 2P-60B trips on breaker fault and cannot be restarted.

To restore SDC to service ___________________ should be aligned for SDC and placed in service after RCS pressure has been ____.

A. Spray Pump 2P-35B; reduced to less than 50 psig to prevent RCS inventory loss

B. Spay Pump 2P-35A; reduced to less than 50 psig to prevent RCS inventory loss

C. Spray Pump 2P-35B; raised to greater than 100 psig to ensure adequate pump NPSH

D. Spray Pump 2P-35A; raised to greater than 100 psig to ensure adequate pump NPSH

Answer:

A. Spray Pump 2P-35B; reduced to less than 50 psig to prevent RCS inventory loss

Notes:

A is correct as the containment spray ESF pump can be aligned to provide SDC flow if available but its suction relief valves are set at 80 psig and the procedure requires lowering RCS pressure to less than 50 psig to prevent lifting the Reliefs to prevent over pressurization of the containment sump and RCS inventory loss.

B is incorrect as the 2P-35A spray pump cannot be aligned as a SDC pump due to 2A3 bus outage but plausible as the correct action and reason is listed to reduce RCS pressure.

C is incorrect as RCS pressure should be reduced to less than 50 psig but plausible as the correct pump to align is listed and the concern for adequate NPSH should always be addressed for any Safeguard pump.

D is incorrect as RCS pressure should be reduced to less than 50 psig and the 2P-35A pump has no power but plausible because the concern for adequate NPSH should always be addressed for any ESF pump.

This question matches the K&A as the interrelationship between the RHR pumps and the other Safeguard pumps must be known along with the different restrictions for using the alternate SDC/Spray Pump.

References:

STM_2-14_14-1 SDC System Section 2.2.2 (Verified reference updated 11/15/16); STM_2-14_14-1 SDC System Drawing (Verified reference updated 11/15/16); NOP 2104.004 Rev. 58 SDC Section 11.0 Aligning Spray Pumps for SDC (Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Clarified information in the analysis for distractors B and D as to why HPSI pumps are a plausible choice and added the words " has just" to the first bullet.

REV. 2 based on NRC Chief Examiner Feedback BNC. Removed the words "Red Train" from the beginning of the 2nd bullet. Changed "HPSI Pump 2P-89B" in distractors B and D to "Spray Pump 2P-35A". Updated the distractor analysis.

2.2.2 Containment Spray Pump

The containment spray (CS) pumps 2P-35A and 2P-35B can also be aligned to provide SDC flow. They are single stage, vertical, motor driven, centrifugal pumps with the following power supplies:

- 2P-35A is powered from 2A304
- 2P-35B is powered from 2A404

The pumps are rated at 2200 gpm at ~230 psi and have a maximum flow of 3000 gpm. Due to instrument inaccuracies, the procedure limits flow to <2800 gpm based on maximum continuous motor amps.

The CS pumps are protected from overheating when operating at shutoff head by minimum flow recirculation lines provided with motor operated isolation valves. These lines provide a minimum recirc flow of 200 gpm back to the refueling water tank (RWT). To prevent discharging reactor coolant to the RWT during SDC operations, these recirc valves are verified closed during SDC system lineup using the CS pumps.

The CS pumps are also interlock with their associated containment spray isolation valve:

- 2P-35A with 2CV-5612-1
- 2P-35B with 2CV-5613-2

When the CS pumps breaker closes, a contact in the valve's CLOSE circuit closes and a contact in the valve's OPEN circuit opens. This causes the valve to close if it is open and prevents manually opening the valve if it is closed while the CS pump is running. This is to prevent spraying down the containment when starting the CS pump for SDC. This interlock is defeated if an actual containment spray actuation signal is present.

The SDC heat exchanger outlet to the LPSI header valves (2SI-5A and 2SI-5B) are normally both open. Therefore, opening the containment spray header isolation valve on the train opposite the running LPSI pump will cause the containment to be sprayed down.

For more detailed information on the CS pumps, refer to STM 2-8, Containment Spray System.

2.2.2.1 Using CS for SDC

There are two restrictions for using containment spray pumps for shutdown cooling pumps.

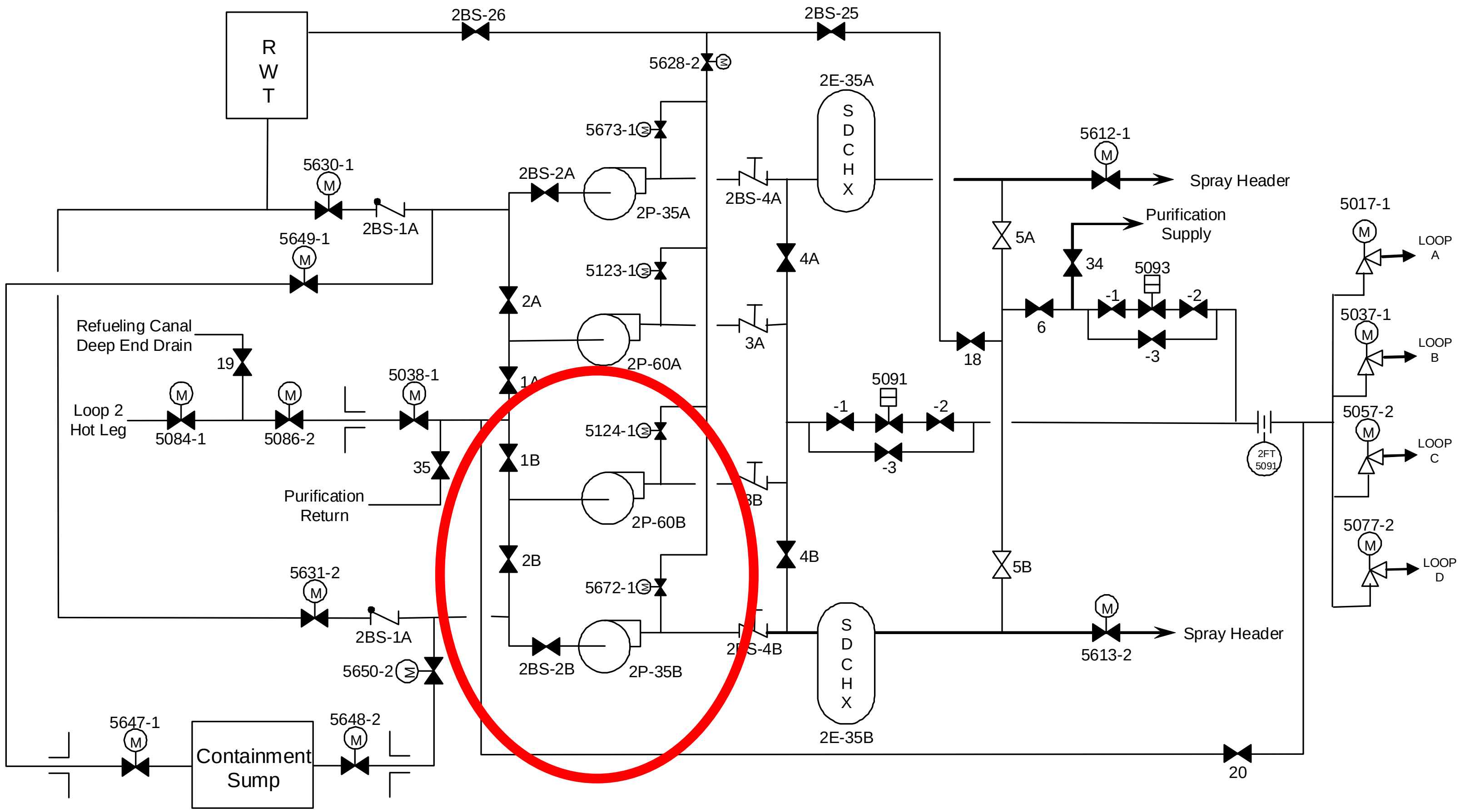
- Maximum design flow rate for containment spray pump motors is 3000 gpm. Limit flow to 2800 gpm to account for instrument errors. This limit is based on continuous motor amperage.
- Verify SDC suction pressure <50 psig on 2PIS-5088 (SDC suction pressure sensed between the two containment isolation valves on the suction line). This limitation is to prevent overpressurization of the containment sump / RWT suction piping and lifting relief valves 2PSV-5653 and 2PSV-5654 (setpoint of 80 psig).

2.2.3 Normal Alignment of SDC Pumps

During normal SDC operating conditions the following is maintained for the SDC pumps and associated valves.

- LPSI pumps have their suction lined up to the RCS.

Figures



Shutdown Cooling System

All valves, unless otherwise noted, start with 2SI-.
All MOVs and AOVs start with 2CV-.

{4.2.3} (All of section 11.0)

11.0 ALIGNING SPRAY PUMPS FOR SDC

11.1 **ENSURE** the following RCP handswitches in PULL-TO-LOCK:

- 2P-32A (2HS-4620)
- 2P-32B (2HS-4621)
- 2P-32C (2HS-4720)
- 2P-32D (2HS-4721)

CRITICAL STEP

11.2 **ENSURE** Suction From RCS (2PIS-5088) less than 50 psig.

11.3 **ADJUST** 2PIS-5088 alarm setpoint to less than 50 psig or as low as practical. **REFER TO** Attachment N, SDC Pump Suction Pressure 2PIS-5088.

11.4 **ENSURE** the following valves closed:

- At least ONE Red Train CNTMT Sump Suction Isol:
  - 2CV-5649-1
  - 2CV-5647-1
- At least ONE Green Train CNTMT Sump Suction Isol:
  - 2CV-5650-2
  - 2CV-5648-2
- CNTMT Spray 2P-35A Recirc Isol (2CV-5673-1)
- CNTMT Spray 2P-35B Recirc Isol (2CV-5672-1)
- HPSI 2P-89A Recirc Isol (2CV-5126-1)
- HPSI 2P-89B Recirc Isol (2CV-5128-1)
- HPSI 2P-89C Recirc Isol (2CV-5127-1)

11.5 **PERFORM** Attachment I, SDC Train Swap Checklist.

Question 42

Bank:	2372	**Rev:**	1	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	42	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	004000K407	**10CFR55:**	41.7	**Safety Function**	1				
Title:	Chemical and Volume Control System (CVCS)	**System Number**	004	**K/A**	K4.07				
Tier:	2	**Group:**	1	**RO Imp:**	3.0	**SRO Imp:**	3.3	**L. Plan:**	A2LP-RO-CVCS
OBJ	4								

Description: Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following: - Water supplies

Question:

During full power operations, if the VCT level drops to the setpoint of __________%, the RWT to Charging Pump Suction 2CV-4950-2 will open to provide a water source to the CVCS system and RCS T-ave will_____________.

A. 5; lower

B. 5; not change

C. 7; lower

D. 7; not change

Answer:

A. 5; lower

Notes:

A is the correct level setpoint and T-ave will start to lower as the higher concentration of boric acid from the RWT is delivered to the RCS. The VCT Outlet also closes at this level.

B is incorrect as T-ave will lower but plausible as 5% is when the swap over to the RWT occurs.

C is incorrect as 7% is the reset when the VCT outlet comes open and the RWT outlet closes but plausible as RCS T-ave will lower

D is incorrect as 7% is the reset when the VCT outlet comes open and the RWT outlet closes and RCS T-ave will lower but plausible as tis is an interlock setpoint on the VCT.

This question matches the K&A as the candidate must have knowledge of the interlock feature of the CVCS that provide for a makeup source of water to the system on low VCT level.

References:

STM_2-04__31-1 CVCS Section 2.2.2 (Verified reference updated 11/15/16);
STM_2-04__31-1 CVCS Drawing (Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Added the words " the setpoint of" before the 1st blank in the stem.

close and the charging pump suction header will realign to the RWT via 2CV-4950-2. If VCT level should rise above 7%, 2CV-4873-1 will re-open and 2CV-4950-2 will reclose (refer to VCT level diagram on page 76 and the electrical scheme for 2CV-4873-1 on page 76).

Located in the VCT room on the 354' elevation of the auxiliary building, 2CV-4873-1 is powered from 2B54-D4. The valve can also be manipulated from the front of breaker 2B54-D4 using 2HS-4873-1A. For the valve to be operated from the handswitch on panel 2C09, 2HS-4873-1A must be in the "remote" position. The valve will close and cannot be re-opened if a SIAS #1 signal is present.

2.2.2 RWT to Charging Pump Suction Isolation Valve 2CV-4950-2

The RWT to charging pump isolation valve, 2CV-4950-2, is provided to align the RWT to charging pump suction. This valve is controlled by a three-position handswitch located on panel 2C09. Whenever this handswitch is in "auto", 2CV-4950-2 will automatically position based on VCT level. 2CV-4950-2 will open on a low-low VCT level signal of 5% as sensed by 2LT-4861. This 5% level signal also closes VCT outlet valve 2CV-4873-1, discussed in the previous section. As mentioned in the discussion of the VCT outlet valve above when VCT level rises to 7%, 2CV-4950-2 will close while 2CV-4873-1 re-opens. (Refer to VCT level diagram on page 76 and electrical scheme for 2CV-4950-2 on page 77)

2CV-4950-2 is powered from 2B62-F2 and is located in the "A" charging pump room along the west wall. An additional handswitch, 2HS-4950-2A, is located on the breaker for local operation of the valve and must be in remote for the valve to be operated from panel 2C09.

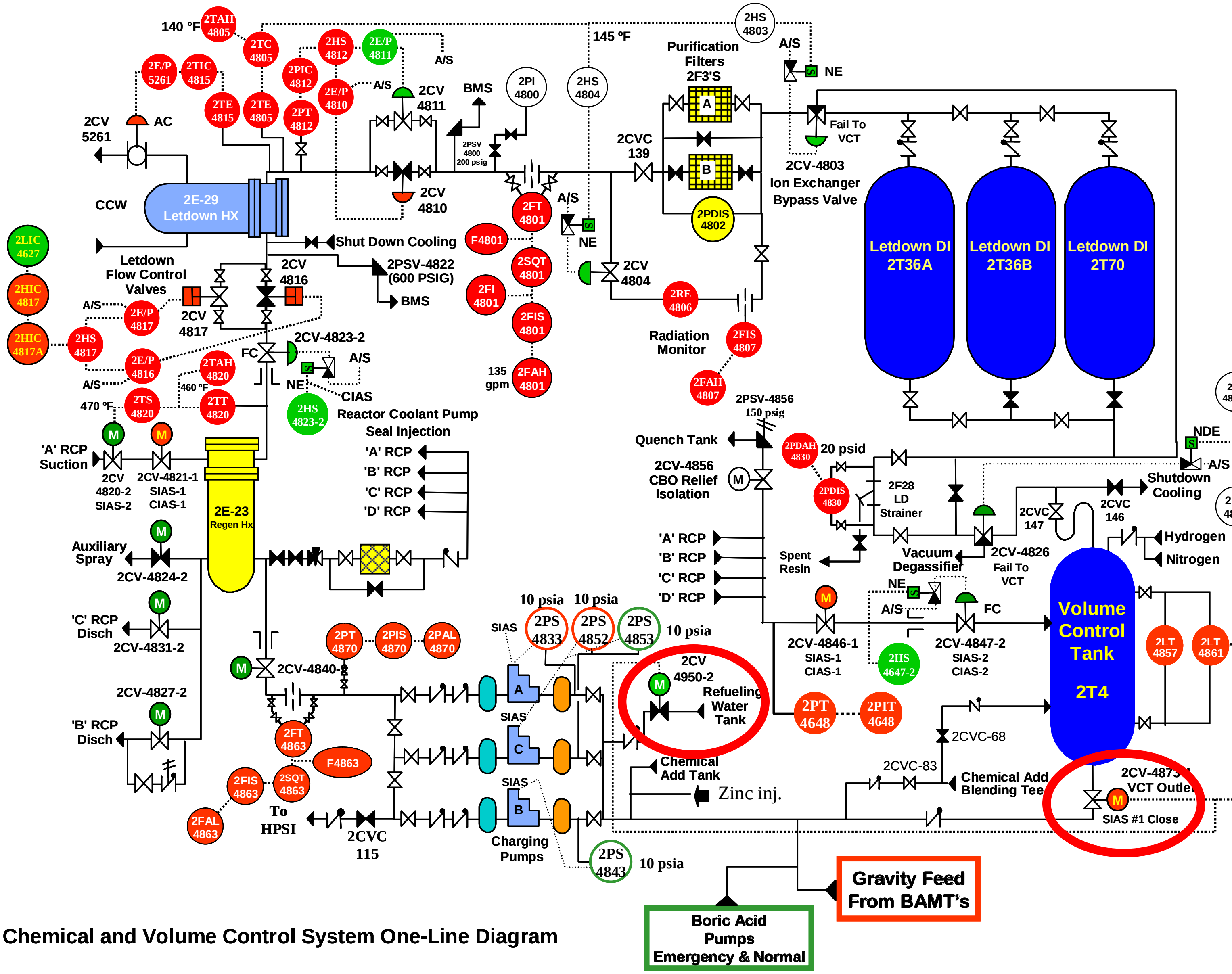
2.2.3 Charging pumps 2P-36A, 2P-36B and 2P-36C

As discussed earlier, RCS water is diverted to the letdown system to provide filtering and purification for the RCS. The charging pumps serve to send the water that was removed back to the RCS in order to maintain RCS inventory at desired levels. During steady state operation, total letdown plus controlled bleed-off flows are equal to total charging flow going back to the RCS[8].

The charging pumps are positive displacement pumps having three plungers (Triplex). Each pump is driven by a 480 vac, 100 horsepower motor. Each pump has its own integral leakage collection system. The internals of the pumps are austenetic stainless steel. Each charging pump has its own seal lubrication provided by a separate pump with its own subsystem.

FOOTNOTES:

[8] During power operations the control board operator can use this concept to identify any leakage that exists in the reactor coolant system. If charging flow is greater than the total letdown plus controlled bleed-off flows then a leak equating to the difference in the flows may exist. Load transients or any perturbation causing a change in RCS temperature will mask any leakage that may be present. Other methods are available to the control room staff to determine actual leakage values.



Chemical and Volume Control System One-Line Diagram

Question 43

Bank:	2373	**Rev:**	1	**Rev Date:**	12/16/2016	**2017 TEST QID #:**	43	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC EXAM BANK 1626		
Search	022000A301	**10CFR55:**	41.7	**Safety Function**	5				
Title:	Containment Cooling System (CCS)	**System Number**	022	**K/A**	A3.01				
Tier:	2	**Group:**	1	**RO Imp:**	4.1	**SRO Imp:**	4.3	**L. Plan:**	A2LP-RO-CVENT
OBJ	18								

Description: Ability to monitor automatic operation of the CCS, including: - Initiation of safeguards mode of operation

Question:

Given the following:

* The plant has tripped from 100% power.
* RCS Pressure is 1800 psia and dropping.
* Steam Generator pressures are 700 psia and dropping.
* Containment pressure is 19.3 psia and rising.

Based on the above conditions, when checking the status of the Containment Cooling Fans, the proper status would have Chilled Water _____________ and Service Water _______________.

A. aligned; aligned

B. isolated; aligned

C. aligned; isolated

D. isolated; isolated

Answer:

B. isolated; aligned

Notes:

B is correct as a Containment Isolation Actuation Signal (CIAS due to >18.3 psia inside Containment) and a Containment Cooling Actuation Signal (CCAS due to > 18.3 psia) have come in due to high Containment Pressure > 18.3 psia. Therefore, Service Water will be aligned and Chill Water will be Isolated.

A is incorrect because Chilled water should be isolated but plausible as this is the configuration that the plant would be in if there were only a CCAS in progress with no CIAS present. CCAS also comes in on low pressure of < 1650 psia.

C is incorrect because Chill Water isolation would have been closed by the CIAS and Service Water would be aligned to the CC Fans by the CCAS but plausible as this is the normal operating configuration of the system without a CCAS or CIAS in progress.

D is incorrect because Service water would be aligned from the CCAS but plausible as this is the configuration that the plant would be in if there were only an inadvertent CIAS was in progress with no CCAS present.

This question matches the K&A because the candidate must have the knowledge and ability to locate the CCS control and verify correct automatic initiation on an ESF signal to the CCS system.

References:

EOP SPTAs 2202.001 REV.15 Contingency Action Step 9.A.1&2 (Verified reference updated 11/15/16); STM_2-09_16 Containment Cooling and Purge System Section 2.11(Verified reference updated 11/15/16); STM_2-09_16 Containment Cooling and Purge System Section 10.2.1(Verified reference updated 11/15/16);

Historical Comments:

NRC EXAM BANK 1626 used on the 2009 NRC Exam
To be used on the 2017 NRC Exam but altered the distractors to make more plausible.

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed the format of this question into a 2 X 2 and deleted references to the CC fans running and the position of the bypass dampers as requested.

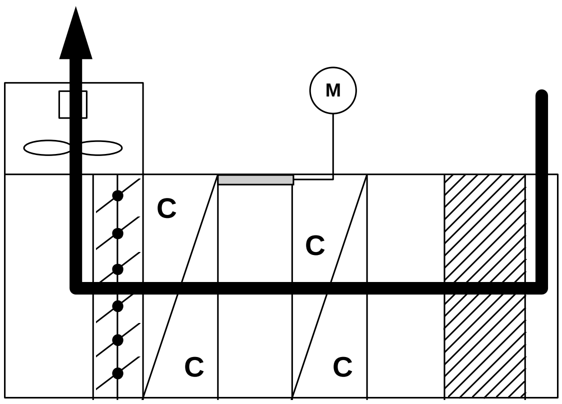
INSTRUCTIONS	CONTINGENCY ACTIONS
9. Check CNTMT parameters:	
A. Temperature and Pressure: ___ • Temperature less than 140°F. ___ • Pressure less than 16 psia.	A. Perform the following: 1) IF CNTMT pressure less than 18.3 psia, THEN verify ALL available CNTMT Cooling fans running with cooling water aligned. 2) IF CNTMT pressure 18.3 psia or greater, THEN verify the following: • CIAS, CCAS, and SIAS actuated on PPS inserts. • At least ONE Emergency Penetration Room Vent Fan running. • CNTMT Cooling fans running in Emergency Mode. 3) IF CNTMT pressure 23.3 psia or greater, THEN verify the following: • CSAS actuated on PPS inserts. • Spray flow greater than 1875 gpm per header. • ALL RCPs stopped AND BOTH PZR Spray valves in MANUAL and closed.
B. Check CNTMT Spray pumps secured.	B. IF CSAS inadvertent, THEN perform the following: 1) Place BOTH CNTMT Spray pumps (2P35A/B) in PTL. 2) Record time: ____________

(Step 9 continued on next page)

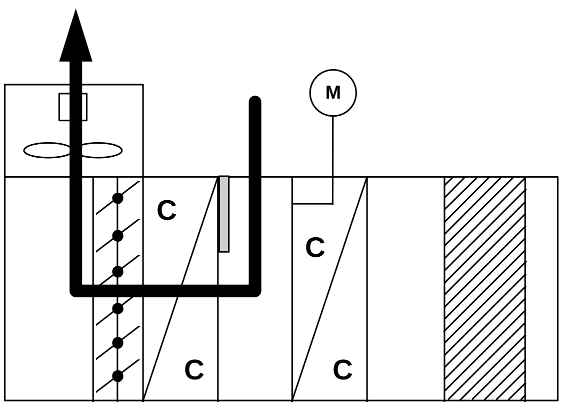
2.11 Review

In review, all of the above components are actuated from the control room with individual handswitches.

Upon receipt of a CCAS, the service water supply header isolation valves open. If there has been a loss of power they open following a 17.1-second time delay. The return header isolation valves open after the supply valves are ~ 10% open. The bypass dampers drop and the fan receives a start signal after an 18.2-second time delay. This is referred to as the “emergency mode”.



Normal flow path: Main Chilled Water lined up through cooling coils Service Water is isolated



Emergency Mode (CCAS): Service Water lined up through cooling coils and Bypass Damper open. Main Chilled Water may be lined up but isolates on CIAS.

Upon receipt of an MSIS, the service water supply header isolation valves open. If there has been a loss of power they open following a 17.1-second time delay. The return header isolation valves open when the supply valves are approximately 10% open.

Upon receipt of a CIAS, the main chill water supply and return isolation valves close.

- Containment activity should lower by approximately one half every 30 minutes.
- If RDACS is inoperable, then SPING 5 can only be read locally.

- Containment recirculation fan monthly test
  - The supplement is performed to satisfy TRM surveillance requirement 4.6.4.3.a
- Containment cooler 14 day test
  - This supplement is performed to satisfy technical specification surveillance requirements 4.6.2.3.a and 4.6.2.3.b.
  - Containment atmospheric temperature may rise while performing this supplement during summer months. This temperature rise may affect primary parameter instrumentation. CR-ANO-2-2001-0607 documents a temperature fluctuation on RCS hot leg temperature indicator T-4610 during a containment cooler 14 day test.
  - When closing containment cooling service water valves, then maintain the handswitch in CLOSE for ~ 2 seconds after the red light goes out to ensure proper torquing of the valve on its shut seat. *There is no "seal-in" feature associated with these valves. Valve motion stops when the handswitch is released.*
  - Technical specification surveillance requirement 4.6.2.3.a.1 requires flow to be greater than or equal to 1250 gpm. The limiting range for operability for this test is adjusted after each service water flow test due to fouling factor adjustment. *This fouling factor adjustment is necessary due to the amount of fouling of the service water tubes that lowers the heat transfer capabilities of the cooling coils.*

10.2 Emergency Operating Procedures

This section discusses the operation of the containment ventilation systems as they are used in each of the emergency operating procedures.

10.2.1 OP 2202.001, Standard Post Trip Actions

In verifying the safety function for containment temperature, pressure and combustible gas control, if containment temperature is greater than 140°F or pressure is greater than 16 psia the following contingency actions are taken:

- If containment pressure is less than 18.3 psia then verify ALL available containment cooling fans are running with cooling water aligned. This is done in order to minimize containment pressure rise and possibly prevent pressure from rising to the CIAS or CSAS setpoints. *18.3 psia is the setpoint for CIAS, CCAS and SIAS. If pressure is less than this, main chill water should still be aligned to the containment.*
- If containment pressure is greater than or equal to 18.3 psia then verify:
  - Containment cooling fans are running in the emergency mode.

Service water is aligned to the containment. Main chill water should be isolated at this point. Bypass dampers open

This is done to ensure maximum cooling to the containment is available to minimize the pressure rise.

10.2.2 OP 2202.003, Loss of Coolant Accident

After confirming a LOCA, the operator has to verify that an SIAS and a CCAS have actuated. If not, they are manually actuated. *This is done because SIAS is expected to have already actuated when entering this EOP. CCAS is actuated by the same signals that actuate an SIAS and therefore should already be in also. CCAS aligns the containment cooling system to respond to the raised containment heat load that is expected during a LOCA inside containment.*

Following this the operator verifies CCAS components actuated by checking the following:

- All available containment cooling fans running
- All containment cooling bypass dampers open
- Both service water outlet valves are open
- Both service water inlet valves are open.

This is done to ensure that the system has actuated to maximize containment heat removal in order to minimize containment temperature and pressure rise that could possibly result in actuation of containment spray.

Once the LOCA is determined to be inside the containment, the operator is required to verify all available miscellaneous containment ventilation systems operating which includes:

- Containment building recirculation fans
- Reactor cavity cooling fans
- Maximum of three CEDM shroud cooling units

This is done in order to maximize the recirculation of the containment atmosphere. The recirculation minimizes the possibility of local accumulations of hydrogen that has evolved due to the accident and allows a representative sample to be taken by the hydrogen analyzers.

10.2.3 OP 2202.004 Steam Generator Tube Rupture

After confirming a SGTR, the operator verifies that an SIAS and a CCAS have actuated. If not, they are manually actuated. *This is done because SIAS is expected to have already actuated when entering this EOP. CCAS is actuated by the same signals that actuate an SIAS and therefore should already be in also.*

Much later in the procedure the operator verifies CCAS components actuated by checking the following:

- All available containment cooling fans running
- All containment cooling bypass dampers open
- Both service water outlet valves are open
- Both service water inlet valves are open.

Although this particular event does not challenge the temperature or pressure limits of the containment, since a CCAS is

ORIGINAL BANK QUESTION 1626 WAS USED TO DEVELOP 2017 NRC EXAM QID#43 (Used on 2005 and 2009 NRC Exam)

Questions For All QID In Exam Bank

Bank:	1626	**Rev:**	1	**Rev Date:**	7/24/2009 12:40:2	**QID #:**	17	**Author:**	Coble
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC Exam Bank 0451		
Search	00CE02K201	**10CFR55:**	41.7	**Safety Function**	1				
System Title:	Reactor Trip Recovery	**System Number**	E02	**K/A**	EK2.1				
Tier:	1	**Group:**	1	**RO Imp:**	3.3	**SRO Imp:**	3.7	**L. Plan:**	A2LP-RO-CVENT

OBJ 18

Description: Knowledge of the interrelations between the (Reactor Trip Recovery) and the following: - Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Question:

Given the following:

* The plant has tripped from 100% power
* RCS Pressure is 1700 psia and dropping
* Steam Generator pressures are 800 psia and dropping
* Containment pressure is 19.3 psia and rising

What equipment status would the Control Room Operators expect to see when checking Containment Cooling (CC) Fan operations?

A. Chilled Water isolated by CCAS signal, 3 CC fans running, Service Water aligned, and bypass dampers Open

B. Chilled Water isolated by CIAS signal, all CC fans running, Service Water aligned, and bypass dampers Open

C. Chilled Water aligned for cooling, 3 CC fans running, Service Water isolated, and bypass dampers Closed

D. Chilled Water aligned for cooling, all CC fans running, Service Water aligned, bypass dampers Closed

QID use History

	RO	SRO
2003	☐	☐
2005	☑	☑
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☐

Answer:

B. Chilled Water isolated by CIAS signal, all CC fans running, Service Water aligned, and bypass dampers Open

Notes:

Distracter A is incorrect because Containment Chill Water isolations are actuated closed by CIAS and the bypass dampers would be Open.
Distracter C is incorrect because Chill Water isolation would have been closed by the CIAS.
Distracter D is incorrect because the bypass dampers should be Open.

References:

EOP 2202.001, Standard Post trip Actions, Rev. 9, Contingency Action 9.A.2
STM 2-09, Containment Cooling and Purge Systems, Rev. 16, Section 10.2.1
Lesson Plan A2LP-RO-CVENT Rev. 4, Objective 18: Describe the emergency operation of the Containment Building Ventilation System.

Historical Comments:

Used on the 2005 NRC Exam

Question 44

Bank:	2374	**Rev:**	2	**Rev Date:**	10/12/2016	**2017 TEST QID #:**	44	**Author:**	Coble		
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam				
Search	0130002408	**10CFR55:**	41.10	**Safety Function**	2						
Title:	Engineered Safety Features Actuation System (ESFA	**System Number**	013	**K/A**	2.4.8						
Tier:	2	**Group:**	1	**RO Imp:**	3.8	**SRO Imp:**	4.5	**L. Plan:**	A2LP-RO-EAOP	**OBJ**	40

Description: Emergency Procedures/Plan - Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Question:

Given the following:

* The Reactor Trips and SPTAs are completed.
* The CRS has entered OP-2202.002 Reactor Trip Recovery (RTR) EOP.

NOW

* While performing RTR actions, an SIAS starts both trains of ESFAS Pumps.
* RCS Pressure is 2157 psia and slowly rising.
* Containment Pressure is 14.4 psia and stable.

Based on these conditions, which of the following is the required direction to take in accordance with OP-1015.021 ANO-2 EOP/AOP Users Guide to address the SIAS?
(LMFRP = Lower Mode Functional Recovery Procedure)

A. Exit RTR EOP Now and GO TO LMFRP and perform the applicable EOP actions.

B. Exit RTR EOP Now and GO TO Inadvertent SIAS and perform applicable AOP actions.

C. Continue RTR actions while REFERENCING LMFRP and perform applicable EOP actions.

D. Continue RTR actions while REFERENCING Inadvertent SIAS and perform applicable AOP actions.

Answer:

D. Continue RTR actions while REFERENCING Inadvertent SIAS and perform applicable AOP actions.

Notes:

D is correct as the entry conditions for Inadvertent SIAS are met even when in the EOPs. RTR EOP can still be conducted while the CRS or a designated Board Operator refers to the inadvertent SIAS AOP and performs the applicable steps of the inadvertent SIAS in conjunction with the other steps in RTR.

A is incorrect as the RTR EOPs should not be exited until completed or a safety function in the EOP is not being met but plausible as the candidate must analyze the given plant indications to determine if the SIAS is valid or not. Also the Inadvertent SIAS procedure provides an option to GO to the LMFRP if the SIAS is valid and the plant is in Mode 3; however the entry conditions for LMFRP are not met in the given conditions.

B is incorrect as the RTR EOPs should not be exited until completed or a safety function in the EOP is not being met but plausible as there are some very important steps to maintain ACW and CCW systems (Loss of RCP cooling that could cause a safety function not to be met and complicate the recovery due to loss of Forced Circulation) in service in the Inadvertent SIAS procedure that need to be performed in addition to securing the ESF pumps. Also there are steps in the AOP to verify the validity of the SIAS before securing the ESF pumps.

C is incorrect because the LMFRP EOP does not need to be referred to once the conditions that cause a SIAS are validated to not be present but plausible as the Inadvertent SIAS procedure provides an option to GO to the LMFRP if the SIAS is valid and the plant is in Mode 3.

This question matches the K&A as it requires knowledge of AOP Usage during EOP Implementation to address actions to take when ESFAS pumps start in while performing an EOP RTR procedure.

References:

OP 2202.002 RTR EOP Procedure Rev. 11 Exit Conditions (Verified reference updated 11/15/16);
Admin Procedure 1015.021, EOP-AOP Users Guide Rev. 15 Step 4.32 and 4.40.17 REFER TO (Verified reference updated 11/15/16); Admin Procedure 1015.021, EOP-AOP Users Guide Rev. 15 4.40.12 GO TO (Verified reference updated 11/15/16); Inadvertent SIAS AOP 2203.018 Rev. 11 Entry Conditions (Verified reference updated 11/15/16);
AOP 2203.018 Inadvertent SIAS Rev. 11 Step 3, 4, 15 and Section 2 Step 1 (Verified reference updated 11/15/16);

Historical Comments:

To be used on the 2017 NRC Exam.
Rev. 1 Updated to be in the Reactor Trip Recovery EOP instead of SPTAs based on feedback from the NRC Chief Examiner during the 10 QID preliminary review due to the SPTAs should be completed quickly and in its entirety before other AOP or EOP is entered. The inadvertent SIAS was removed from the stem to eliminate any cueing. Also changed the stem to ask what is required by the EOP/AOP Users guide.

Rev. 2 Based on feedback from the NRC Chief examiner, changed the A and C distractor to LMFRP from LOCA since the Inadvertent SIAS procedure provides an option to GO to the LMFRP is the SIAS is valid and the plant is in Mode 3.

REACTOR TRIP RECOVERY

PURPOSE

This procedure provides actions for an uncomplicated Reactor trip.

ENTRY CONDITIONS

1. An uncomplicated Reactor trip has occurred.

EXIT CONDITIONS

1. ANY SFSC acceptance criteria NOT satisfied.

OR

2. ALL actions of this procedure completed.

4.31 PRUDENT ACTION (a.k.a. manual action/anticipatory action/action to mitigate the consequences of an accident/skill of the craft)

Operators may take manual control of a component when that component is deviating from its design function state or has the potential to deviate from its design function state. However operators should use judgment when deciding to take manual control of components. The task must be a simple action that usually involves the alignment of a single component such as a valve controller. A more complex task, such as starting a pump that requires proper alignment of valves to prevent equipment damage, would **NOT** be considered a prudent action. This is not a procedure deviation. This can be justified per ANS 3.2 1976 step 5.2.2(3).

Prudent (manual)action is permitted when:

- A system component is deviating from its design function state or has the potential to deviate from its design function state.
- The task is simple and is considered a routine activity based on operational experience or training.
- The task will not interfere with the intent or strategy of procedure in effect.
- The operator obtains concurrence from CRS/SM prior to performing the task.

4.32 REFERENCING

Used to supply additional information from other sources or procedures, which can then be used in conjunction with the current procedure. The operator does not exit the current procedure, but performs the referenced action in parallel with the current one. The words refer to and using are used to indicate referencing (see Defined EOP/AOP Words, Section 4.40). Note that at other plants these steps may be referred to as parallel steps or parallel actions.

4.33 SAFETY FUNCTION

A condition or action that prevents core damage or minimizes radiation release to the public. A complete set of safety functions needs to be fulfilled to ensure proper operator control of the event, and to maximize public safety. A list of safety functions is provided in ATTACHMENT A, Safety Function Hierarchy (see Defined EOP/AOP Words, Section 4.40).

Out-of-limit parameters may be present during the implementation of an optimal or functional procedure. Under very few circumstances should this be allowed for greater than 15 minutes (time for one set of safety function status checks). If longer than 30 minutes is needed to restore a parameter, then the Shift Manager/CRS may take a procedure deviation and remain in the current EOP, only as long as they are convinced that reactor safety and overall safety function are not being jeopardized. This guidance will ensure plant operation remains within the analyzed limits used to ensure that nothing more serious is occurring and bounded conditions are within prescribed limits. For a complete discussion of the bases for this 15/30 minute period, see Attachment D, Control Parameter Restoration Time Limit.

4.40.11 ENSURE

To ensure a specified condition exists, and if it does not to take the physical action(s) required to establish the desired condition. Steps that use the word ENSURE do not generally use contingency actions, since the instruction step includes making the condition so.

4.40.12 GO TO

A branching term used to indicate that the current point in the procedure should be left to move forward to another step or procedure.

GO TO Step X, indicates to proceed to a later step in a procedure.

GO TO Procedure X, Title Y (where X corresponds to the procedure number and Y corresponds to the procedure title), indicates to implement another procedure.

4.40.13 IMPACTED SG

Used to define a steam generator whose level has dropped below the feed ring during a loss of feedwater event, necessitating a slow refill to avoid water hammer/feed ring collapse.

4.40.14 INTACT SG

Used to define a steam generator with no tube leakage or steam line break. It is also used to define the SG that is least affected by leakage, should both SGs indicate primary to secondary leakage. Implicit in this definition is the capability to feed and steam the SG.

4.40.15 JEOPARDIZED SAFETY FUNCTION

Used in the Functional Recovery procedures to indicate a safety function when checked is not meeting acceptance criteria. Jeopardized safety functions are always addressed first by order of hierarchy.

4.40.16 LEAKING SG

Used to define the steam generator with primary to secondary leakage in 2203.038, Primary to Secondary Leakage.

4.40.17 REFER TO

A referencing term used to provide a procedural reference. The operator may perform a task from memory and refer to the applicable document for supplemental information. Note that at other plants these steps may be referred to as parallel steps or parallel actions.

INADVERTENT SIAS

PURPOSE

This procedure provides actions in response to an Inadvertent SIAS.

ENTRY CONDITIONS

ONE or MORE of the following conditions exist which indicate Inadvertent SIAS:

A. ESFAS Channel 1/2 "SIAS ACT" annunciator (2K07-A1/2K04-A1) in alarm.

B. SIAS actuated without Reactor trip.

C. SIAS actuation due to failure to reset SIAS setpoints during controlled RCS depressurization.

D. SIAS components actuated.

EXIT CONDITIONS

WHEN appropriate actions of this procedure completed,
THEN EXIT this procedure.

SECTION 1 ENTRY

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

Steps marked with (*) are continuous action steps.

1. **RECORD** current time: ____________

*2. **CHECK** SIAS actuated without Reactor trip.

*2. **IF** this procedure was entered from Mode 1 or 2,
THEN GO TO 2202.001, Standard Post Trip Actions.

3. **DETERMINE** validity of SIAS actuation as follows:

   - □ **CHECK** RCS pressure greater than SIAS variable setpoint.
   - □ **CHECK** CNTMT pressure less than 18.3 psia.

3. **IF** SIAS valid,
THEN DETERMINE required action as follows:

   A. **IF** in Mode 1 or 2,
   THEN PERFORM the following:

      1) **TRIP** Reactor.
      2) **GO TO** 2202.001, Standard Post Trip Actions.

   B. **IF** SIAS caused by failure to reset SIAS setpoints during controlled cooldown,
   THEN GO TO Step 3.

   C. **IF** Plant in Mode 3, 4 or 5 **AND** Shutdown Cooling **NOT** in service,
   THEN GO TO the following appropriate procedure:

      - □ 2202.001, Standard Post Trip Actions.
      - □ 2202.011, Lower Mode Functional Recovery.

   D. **IF** SDC in service,
   THEN GO TO 2203.029, Loss Of Shutdown Cooling.

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

- Overriding SW MOVs on a single loop of SW will cause the affected SW loop to be inoperable. This will also make the corresponding Emergency Control Room chiller inoperable for Unit 1.
- Overriding BOTH corresponding MOVs on opposite loops of SW will cause BOTH SW loops to be inoperable AND will make BOTH Emergency Control Room chillers inoperable for Unit 1.

4. **RESTORE** SW to non-ESF systems by performing the following:

   A. **OVERRIDE** and **OPEN** affected SW to ACW Supply valves as necessary:

      - ☐ 2CV-1425-1
      - ☐ 2CV-1427-2

   B. **OVERRIDE** and **OPEN** affected CCW/ACW Return valves as necessary:

      - ☐ 2CV-1543-1
      - ☐ 2CV-1542-2

   C. **OVERRIDE** and **OPEN** SW to CCW/Main Chillers Supply valves as necessary:

      - ☐ 2CV-1530-1
      - ☐ 2CV-1531-2

INSTRUCTIONS	CONTINGENCY ACTIONS
14. **NOTIFY** SM to evaluate SW operability using Attachment B, Tech Spec and TRM Evaluation.	
15. **PERFORM** the following:	
A. **IF** SIAS actuation has occurred on BOTH Red **AND** Green Trains, **THEN GO TO** Section 2, Both Trains.	
B. **IF** SIAS actuation has occurred on ONLY the Red Train, **THEN GO TO** Section 3, Red Train.	
C. **IF** SIAS actuation has occurred on ONLY the Green Train, **THEN GO TO** Section 4, Green Train.	

END

SECTION 2 BOTH TRAINS

INSTRUCTIONS	CONTINGENCY ACTIONS
1. **PLACE** ALL the following pumps in PTL:	
A. HPSI:	
☐ 2P89A	
☐ 2P89B	
☐ 2P89C	
B. LPSI:	
☐ 2P60A	
☐ 2P60B	
C. Boric Acid:	
☐ 2P39A	
☐ 2P39B	

Question 45

Bank:	2375	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	45	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	039000A404	**10CFR55:**	41.7	**Safety Function**	4				
Title:	Main and Reheat Steam System (MRSS)	**System Number**	039	**K/A**	A4.04				
Tier:	2	**Group:**	1	**RO Imp:**	3.8	**SRO Imp:**	3.9	**L. Plan:**	A2LP-RO-EFW
OBJ	8								

Description: Ability to manually operate and/or monitor in the control room: - Emergency feedwater pump turbines

Question:

During an EFAS automatic start of the 2P-7A EFW Pump, which of the following is the correct sequence for the Main Steam to 2P-7A EFW Pump Isolations 2CV-0205 and 2CV-0340 and the primary reason for the sequence?

A. 2CV-0205 will open first then 2CV-0340 after a time delay to prevent a turbine overspeed.

B. 2CV-0340 will open first then 2CV-0205 after a time delay to prevent a turbine overspeed.

C. 2CV-0205 will open first then 2CV-0340 after a time delay to reduce DP across 2CV-0340.

D. 2CV-0340 will open first then 2CV-0205 after a time delay to reduce DP across 2CV-0205.

Answer:

A. 2CV-0205 will open first then 2CV-0340 after a time delay to prevent a turbine overspeed.

Notes:

A is correct: 2CV-0205 is a smaller isolation valve to limit initial Main Steam flow to prevent an overspeed on the turbine. The larger steam isolation will open after a 15 second time delay to provide the required amount of steam flow.

B is incorrect as 2CV-0205 opens first but plausible as this is the correct reason for the sequence.

C is incorrect as this is the wrong reason but plausible as this is the correct sequence and the smaller valve opening first does cause a lower DP across the larger valve.

D is incorrect as 2CV-0205 opens first but plausible if the applicant believes that 2CV-0340 is the smaller valve as the smaller valve opening first does cause a lower DP across the larger valve.

References:

STM_2-19-2_39-1 EFW and AFW SYS Section 2.1.1.3 (Verified reference updated 11/15/16);
STM_2-19-2_39-1 EFW and AFW SYS Section 2.1.1.4 (Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Replaced the word "correct" with "primary" in the stem and clarified the plausibility of distractor D.

oil pump. This oil is referred to as gear spray and is returned to the outboard bearing housing.

To ensure proper operation of the turbine the lube oil temperature must be maintained within limits. Since the lube oil is used in the governor control this aspect is even more important. The viscosity[1] of the oil is inversely proportional to the lube oil temperature. The governor response to system changes could be sluggish and in extreme cases the EFW pump may be inoperable.

Lube Oil Temperature Element (2TE-0317) provides input to:

- Local Temperature Indicator (2TI-0317)
- Control Room Temperature Indicating Switch (2TIS-0317)

2TIS-0317 can be monitored on Control Room panel 2C-16 and also inputs into a Control Room annunciator:

- 2K05-D9, 2P7A TURB LUBE OIL TEMP HI/LO
  - Lube oil temperature > 170 ^{o}F
  - Lube oil temperature < 60 ^{o}F

2.1.1.3 Main Steam Isolation Valve to 2K-3, 2CV-0340-2

The steam is normally isolated from 2K-3 by the Main Steam Isolation Valve to 2K-3, 2CV-0340-2, and the Main Steam Isolation Bypass Valve, 2CV-0205-2 . A gate valve, 2CV-0340-2 is located on the 335' elevation of the Auxiliary Building outside the EFW pump rooms, next to the Spent Fuel Pool Cooling Pumps.

Powered from 2D26-B1, 2CV-0340-2 will remain available to the operator in the event of a loss of all AC power. Since it is vital to maintain feedwater to the Steam Generators, being DC powered helps ensure feedwater available when all other feedwater sources are de-energized.

The isolation valve is normally closed but is automatically opened upon an actuation of the Emergency Feed Actuation Signal (EFAS). The EFAS signal is generated when either Steam Generator level drops below 22.2%. Following an EFAS actuation, 2CV-0340-2 may be overridden and closed by taking 2HS-0340, on 2C-16 in the Control Room, to the OPEN position and then to the CLOSE position. This action removes the EFAS signal to open. Annuciator 2K05-C9, 2P7A TURB EFAS OVERRIDDEN" is activated when an EFAS signal is present and the valve has been overridden.

In addition to the automatic actuation, 2CV-0340-2 can be opened from 2C-16 in the Control Room using 2HS-0340-2. The actual opening of 2CV-0340-2 is delayed 15 seconds to allow for 2CV-0205-2 to open providing a smaller steam flow rate while the governor valve gets control of turbine speed. This minimizes the chance of over speeding the turbine.

FOOTNOTES:

[1] Viscosity is the internal fluid resistance to flow. As viscosity rises, the fluid is thicker and therefore does not flow as easy as a fluid with a lower viscosity value.

2.1.1.4 Main Steam Isolation to 2K-3 Bypass Valve, 2CV-0205-2

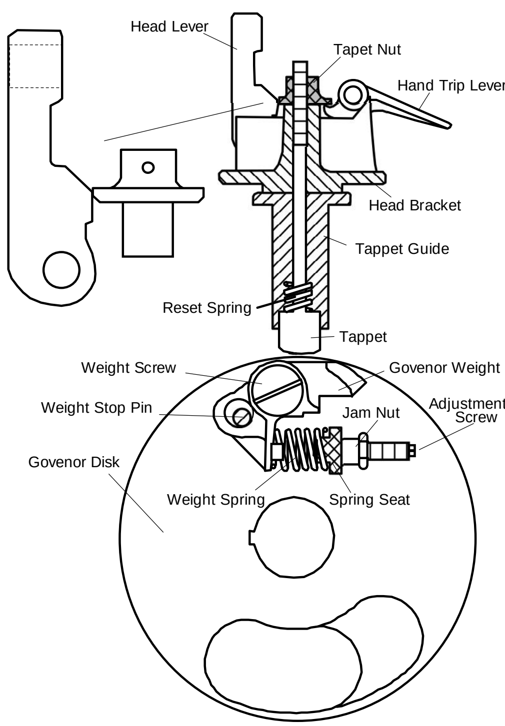
The bypass valve, 2CV-0205-2, is a normally closed gate valve located adjacent to 2CV-0340-2. The bypass valve receives an open signal when an EFAS is generated or when 2HS-0340-2 is placed in the open position. A close valve signal for 2CV-0205-2 is generated when 2CV-0340-2 is full open. The bypass valve is powered from 2D26-C2. A 15/32 inch orifice, 2F-0205, is located downstream of 2CV-0205-2 to limit the steam flow to the turbine.

Upon initial start of the turbine the governor valve, which is used to control turbine speed, is full open. The hydraulic pressure required is provided by a geared oil pump that is driven by the turbine shaft provides the governor control. To prevent an overspeed condition, 2CV-0205-2 opens to provide a small amount of steam to start the turbine rolling. With the shaft turning and the hydraulics are available to the governor valve, the speed of the turbine can be controlled when 2CV-0340-2 opens.

2.1.1.5 2K-3 Trip and Throttle Valve 2CV-0336

The turbine is protected from overspeed by use of a disc type mechanism connected by linkage to the Trip and Throttle Valve, 2CV-0336. The overspeed trip setting is 4470 rpm, when this speed is reached 2CV-0336 trips closed preventing steam from entering the turbine.

The Trip and Throttle valve, 2CV-0336, is a globe valve located in the 2P-7A room. This valve is used to act as an emergency closing valve, tripping shut upon a turbine overspeed. It can also function as a manual throttle for controlling speed of the turbine in the event of a governor or governor valve failure. Normally the valve is full open and set to trip on overspeed.



OVERSPEED TRIP MECHANISM
(Shown in the reset position)

The overspeed trip disc is mounted to the turbine shaft. The governor weight is free to rotate around the weight screw. As the shaft turns, the governor weight tends to rotate outward due to centrifugal force but is held in place by the weight spring. The governor weight remains in the reset position until the speed of the shaft (4470 rpm) provides sufficient centrifugal force to overcome the weight spring. The overspeed setting is adjusted by varying the tension on the weight spring (more tension - higher overspeed setting).

When the overspeed setting has been reached the governor weight rotates outward and strikes the tappet forcing the tappet upward. The upward motion of the tappet releases the head lever which has spring force applied forcing it to the right. Attached to the head lever is a shaft that is connected to the Trip and Throttle Valve,

Question 46

Bank:	2376	**Rev:**	2	**Rev Date:**	12/19/2016	**2017 TEST QID #:**	46	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	003000K604	**10CFR55:**	41.7	**Safety Function**	4				
Title:	Reactor Coolant Pump System (RCPS)	**System Number**	003	**K/A**	K6.04				
Tier:	2	**Group:**	1	**RO Imp:**	2.8	**SRO Imp:**	3.1	**L. Plan:**	A2LP-RO-RCP
OBJ	1								

Description: Knowledge of the effect of a loss or malfunction of the following will have on the RCPS: - Containment isolation valves affecting RCP operation

Question:

Given the following:

* Plant is at 100% Power.
* Annunciator 2K11-G5 "RCP BLEEDOFF PRESSURE HI" comes in.
* RCP Controlled Bleedoff Containment Isolation Valve 2CV-4847-2 has failed CLOSED.

Based on the given conditions, RCP Controlled Bleedoff (CBO) flow will be directed to the ___________.

A. Quench Tank

B. Reactor Drain Tank

C. Containment Sump

D. Volume Control Tank

Answer:

A. Quench Tank

Notes:

A is correct as 2PSV-4836 set at 150 psig will lift and send the RCP bleedoff flow to the Quench Tank.

B is an incorrect flowpath for RCP CBO but plausible as the RCP Vapor seal does go to the RDT along with other RCS Drains and reliefs.

C is incorrect since the RCP CBO is hard piped to the Quench Tank but plausible as a lot of other system reliefs in the containment are directed to the containment sump.

D is incorrect as the CBO Relief Isolation 2CV-4856 is normally open to allow the relief valve 2PSV-4836 to direct RCP CBO to the Quench Tank at 150 psig. However, it is plausible as this is the normal flow path for the Controlled Bleedoff flow so it could be assumed that any CBO relief would go to the VCT.

This question matches the K&A as it requires knowledge of the effect of that a closure of a RCP CBO Containment Isolation Valve will have on the RCPs.

References:

2203012K ANNUNCIATOR 2K11 CORRECTIVE ACTION Rev 46 Window G-5 RCP Bleedoff Pressure H (Verified reference updated 11/15/16)I; STM_2-03-2_18-1 RCPs Section 1.5 (Verified reference updated 11/15/16); STM_2-03-2_18-1 RCP Drawing of RCP controlled bleedoff flow.(Verified reference updated 11/15/16); 2203012K ANNUNCIATOR 2K11 CORRECTIVE ACTION Rev 46 Window G-4 RCP Bleedoff Pressure HI HI (Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed the word "Diverted" to "Directed" in all 4 distractors.

REV. 2 based on NRC Chief Examiner Feedback BNC. Revised the stem to Say "RCP Controlled Bleedoff (CBO) flow will be directed to the ___________."

ANNUNCIATOR 2K11

G-5

RCP BLEEDOFF PRESS HI

1.0 CAUSES

1.1 > 120 psig on Reactor Coolant Pump Combined Bleedoff to VCT (2PIS-4848)

2.0 ACTION REQUIRED

2.1 Check RCP chart recorders and PMS/PDS trends to determine affected pump(s) or alarm validity.

- IF alarm invalid,
  THEN no further action required.

* 2.2 Monitor all RCPs seal staging for indications of further seal deterioration.

* 2.3 Monitor Reactor Drain Tank level:

- RDT 2T-68 Level (2LIS-2200A)
- PMS point (L2200)

2.4 IF indications of failed seals exist,
THEN GO TO RCP Emergencies (2203.025).

2.5 Initiate RCS leak rate calculation using Reactor Coolant System Leak Detection (2305.002).

3.0 TO CLEAR ALARM

3.1 Reduce Combined Bleedoff header pressure to < 120 psig.

4.0 REFERENCES

4.1 E-2457-1

the operator to help verify the status of RCPs following a reactor trip. Normal ΔP for a running RCP is ~85 - 90 psid.

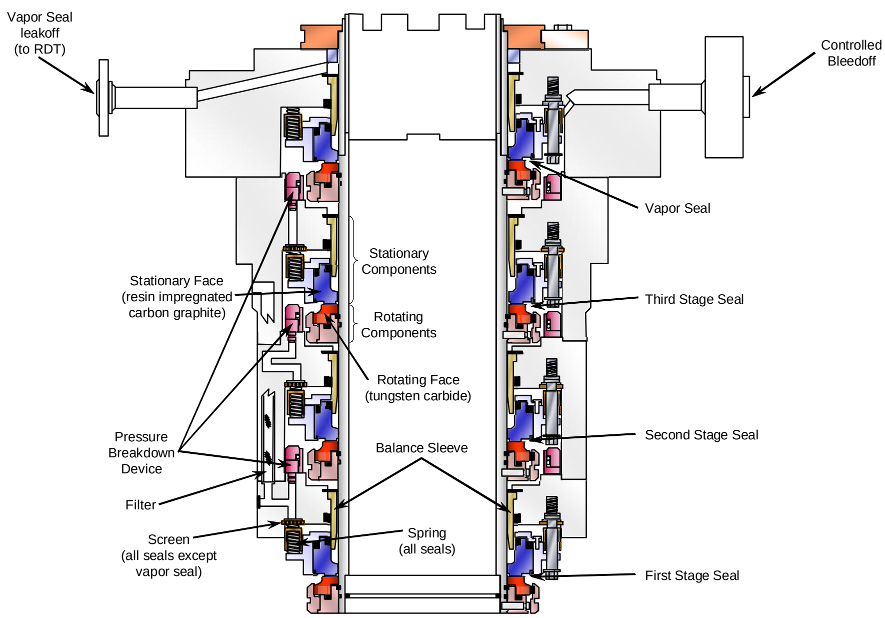
1.4 Pump Casing Inner Gasket Seal Leakoff

Between the pump casing and pump cover there is an inner gasket and outer gasket for sealing. In between the two gaskets is a piping connection that will conduct any leakage past the inner gasket to the Reactor Drain Tank. In this line there is a temperature element that will cause an alarm on 2K11 at ≥ 160°F, which will alert the operator to the fact that the inner gasket is leaking by. If this alarm occurs, and is valid, an RCS leakrate determination is required to be performed.

1.5 RCP Seals

Refer to the figure below or on page 44. The most complex part of the Reactor Coolant Pump is its seal and auxiliaries. The seal package proper consists of four mechanical face seals. Each rotating face is tungsten carbide compound while the stationary face is resin impregnated carbon graphite. Three of the seals are used to contain reactor coolant pressure during normal operation. Each seal is designed to withstand full RCS pressure with the RCP idle and for a limited time with the pump running.

Refer to the figures on pages 45 and 46. The mechanical seals are lubricated and cooled by the 1 gpm controlled Reactor Coolant leakoff. Reactor Coolant enters the seal area at about 1 gpm at the top of the thermal barrier from shaft leakage caused by the ΔP between the RCS and the Volume Control Tank (VCT). The thermal barrier is physically part of the pump cover. The barrier consists of an inner bore with multiple spiral grooves which form a close

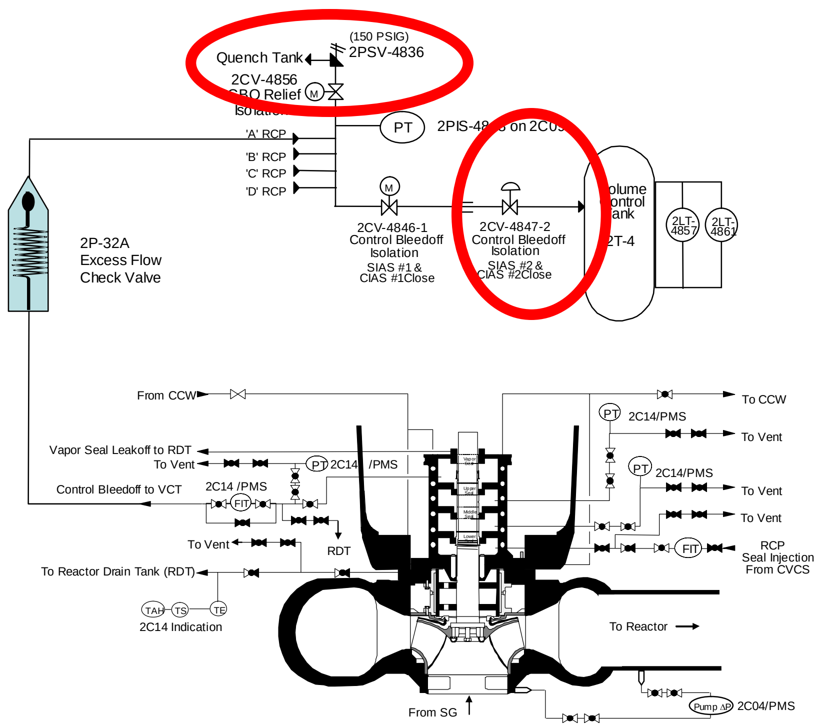


running clearance with the pump shaft, which also has spiral grooves cut into it in the same area, and acts as a flow restrictor.

Attached to the pump shaft above the thermal barrier is a recirculation impeller which circulates the fluid in the seal cavity at 40 gpm into the inner, integral Reactor Coolant Pump cooling coils. The fluid that is circulated through the seal cooler discharges in two places:

- one discharge is back into the seal package where it combines with the shaft leakage and is recirculated back through the seal cooler.
- The second discharge is into the middle of the 1st stage pressure breakdown device, at a rate of 0.9 to 1.0 gpm. This makes its way up through the seal package by sending the majority of the flow up through the pressure breakdown device into the next stage with a very small amount (0.3 - 0.5 gallons per hour) leaking past the seal faces for cooling and lubrication into the next stage. This flow continues through the first 3 stages where the 1 gpm is finally directed to the VCT as Controlled Bleedoff.

Any leakage past the 4th seal (Vapor seal) is directed to the Reactor Drain Tank (RDT). RCS system pressure is broken down approximately the same across each of the three main seals.



REACTOR COOLANT PUMP INSTRUMENTATION

Refer to the drawing above or on page 42. Controlled bleed-off flow from the RCP's passes through excess flow check valves. These valves (2RCP-4A/B/C/D)

work on a forced balance principle. The force developed by controlled bleed-off is balanced by a spring. As controlled bleed-off flow increases the pressure exerted on the spring increases overcoming the force of the spring being used to hold the valve open. If the controlled bleed-off flow increases to 10 gpm the respective RCP's excess flow check valve will start to close. The excess flow check valves are completely closed by 15 gpm. After the pressure drop across the valve has been reduced to 10 - 30 psid the valve reopens.

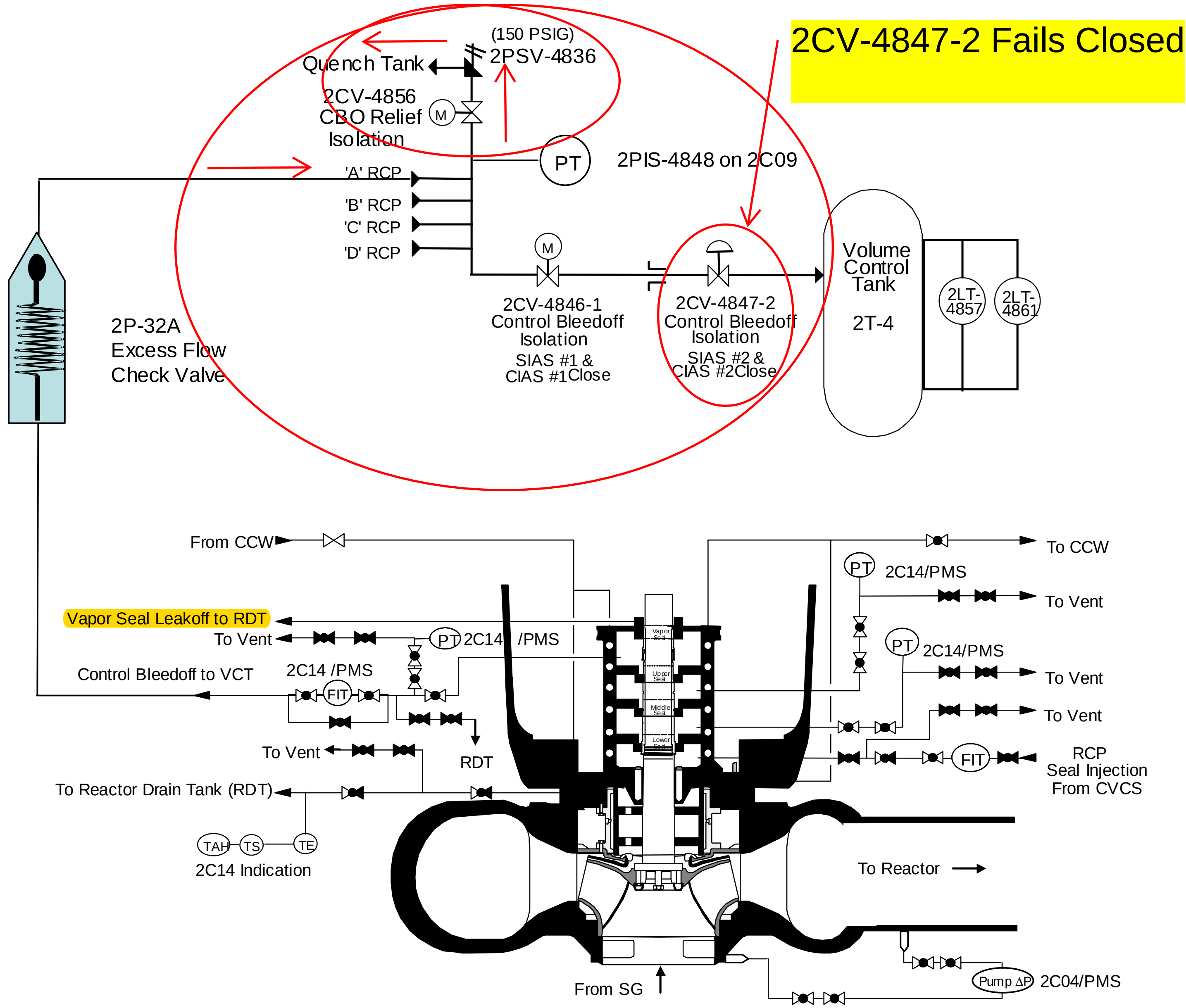
To prevent damage to the RCP seals, the control bleed-off relief, 2PSV-4836, is installed on the controlled bleed-off line upstream of the controlled bleed-off isolation valves to provide a controlled bleed-off flow path should the isolation valves close. The valve is set to lift at 150 psig and discharges to the quench tank. It is sized to pass up to 13 gpm that is equivalent to one RCP having a failed seal and the other three RCP's having normal controlled bleed-off flow.

A motor operated isolation valve, 2CV-4856, is provided to allow isolation of the relief valve if desired. This valve is controlled by a key operated handswitch located on panel 2C09. The valve handswitch is normally in the "open" position. The power supply for this valve is from 2B71-B6.

An annunciator for high pressure is provided on the RCP control bleed-off line. The sensing tap for 2PT-4848 is upstream of 2CV-4846-1. An indication, 2PIS-4848, is provided on panel 2C09. There are two annunciators that are actuated by 2PIS-4848; 2K11-G5 "RCP BLEEDOFF PRESS HI" when indicated pressure is > 120 psig and 2K11-G4 "RCP BLEEDOFF PRESS HI HI" when indicated pressure is > 250 psig.

Cooling for the RCPs is supplied by Component Cooling Water at a rate of 255 gpm per pump. Of this 255 gpm to the pump, 45 gpm of that is directed to the seal area. The seal area cooling water is further subdivided into two streams. One 17 gpm side stream is used to cool the thermal barrier area (the passageway through which the one gpm Controlled Bleedoff is made up.) The remaining 28 gpm side stream passes through the outer annulus of a set of concentric cooling coils. The inner tube of the cooling coils contains the reactor coolant which is recirculated at a rate of 40 gpm by the recirculating impeller.

During RCS fill and vent operations, to prevent the ingress of contaminants that could damage the seal faces, seal injection, from the Charging System is used. Seal injection occurs through the same piping used to vent the lower seal. Injection flows into the area below the first stage seal and down into the RCS taking any contaminants that could enter the seal package with it. This is accomplished by closing the charging header isolation valves and lining up the seal injection valves to maintain 1 to 4 gpm to the seal.



REACTOR COOLANT PUMP Instrumentation

ANNUNCIATOR 2K11

G-4

RCP BLEEDOFF PRESS HI HI

1.0 CAUSES

1.1 > 250 psig on Reactor Coolant Pump Combined Bleedoff to VCT (2PIS-4848)

2.0 ACTION REQUIRED

2.1 Check RCP Bleedoff to VCT Isolations open:

- 2CV-4846-1 (2HS-4846-1)
- 2CV-4847-2 (2HS-4847-2)

2.2 <u>IF</u> (2CV-4846-1 or 2CV-4847-2) closed,
<u>THEN</u> perform the following:

2.2.1 Close RCP Bleedoff Isol (2RCP-5).

2.2.2 Open 2CV-4846-1 (2HS-4846-1).

2.2.3 Open 2CV-4847-2 (2HS-4847-2).

2.2.4 Throttle open RCP Bleedoff Isol (2RCP-5).

2.3 Verify RCP Bleedoff Relief Isol to QT 2CV-4856 (2HS-4856) open.

2.4 Initiate RCS leak rate calculation using Reactor Coolant System Leak Detection (2305.002).

* 2.5 Monitor Reactor Drain Tank level for evidence of vapor seal failure:

- RDT 2T-68 Level (2LIS-2200A)
- PMS point (L2200)

3.0 TO CLEAR ALARM

3.1 Reduce Combined Bleedoff header pressure to < 250 psig.

4.0 REFERENCES

4.1 E-2457-1

Question 47

Bank:	2377	**Rev:**	3	**Rev Date:**	1/12/2017	**2017 TEST QID #:**	47	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	076000A302	**10CFR55:**	41.7	**Safety Function**	4				
Title:	Service Water System (SWS)	**System Number**	076	**K/A**	A3.02				
Tier:	2	**Group:**	1	**RO Imp:**	3.7	**SRO Imp:**	3.7	**L. Plan:**	A2LP-RO-SWACW
OBJ	11								
Description:	Ability to monitor automatic operation of the SWS, including: - Emergency heat loads								

Question:

Consider the following:

* Unit 2 has tripped from 100% power due to a 400 gpm LOCA.
* SIAS has automatically initiated.
* 2 RCPs have been secured.
* Service Water was realigned to CCW HXs using Exhibit 5 during SPTAs.
* The LOCA gets larger over time.
* 45 minutes after the trip, a Recirculation Actuation Signal (RAS) comes in.

After the RAS, Service Water ______________ be aligned to the CCW Heat Exchangers and should be _______________ aligned to the Shutdown Cooling Heat Exchangers.

A. should; automatically

B. should; manually

C. should not; automatically

D. should not; manually

Answer:

C. should not; automatically

Notes:

C is correct: The initial SIAS signal will isolate the CCW HXs. However, on a small break LOCA, it is desired to keep two RCPs running which are cooled by CCW. SPTAs provides direction to override the CCW HX isolation valves to provide this cooling. However, the RAS signal will close these valves again if they have been overridden as a larger break LOCA will require securing all RCPs. The RAS signal will also automatically align Service Water to the SDC HXs which doubles as the Containment Spray Cooling HXs and thus provides cooling for the Containment sump post RAS.

A is incorrect as the CCW HXs will be isolated again but plausible as SW to the SDC HX would be aligned automatically.

B is incorrect as the CCW HXs will be isolated again and the SDC HX would be aligned automatically but plausible if the RAS component positions are confused with SIAS component positions.

D is incorrect as the SDC HX would be aligned automatically but plausible as the CCW HX would not be aligned.

This question matches the K/A as the applicant must have knowledge of the RAS actuated Service water valve to be able to determine the proper response post RAS.

References:

EOP 2202.001 SPTAs Rev. 15 Step 7 E (Verified reference updated 11/15/16);
EOP 2202.001 TG for SPTAs Rev. 15 Step 7 (Verified reference updated 11/15/16);
EOP 2202.010 Standard Attachments Rev. 23 Exhibit 5 CCW-ACW-SW Alignment (Verified reference updated 11/15/16) (Verified reference updated 11/15/16);
STM_2-42_36-1 SW and ACW SYS Table 4 (Verified reference updated 11/15/16);
STM 2-08_22 Containment Spray System Section 4.2 (Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Removed the word "leak" from the 5th bullet and changed the stem/distractors and their associated analysis to SDC HX would be aligned automatically/manually.

REV. 2 based on NRC Chief Examiner Feedback BNC. Changed "would " to "should" prior to the 2nd blank in the stem. In the 1st part of the distractors, changed "would and would not" to "is expected and not expected"

Rev. 3 based on post submittal validation comments: Changed "expected" to "should" in 'A' and 'B' and changed "not expected" to "should not" in 'C' and 'D'.

INSTRUCTIONS	CONTINGENCY ACTIONS
7. Check Core Heat Removal by forced circulation:	
_____A. At least ONE RCP running.	A. IF ALL RCPs stopped, THEN perform the following: 1) Verify BOTH PZR Spray valves in MANUAL and closed. • 2CV-4651 • 2CV-4652 2) **GO TO** Step 8.
B. CCW flow aligned to RCPs.	B. Perform the following: 1) Restore CCW to RCPs using Exhibit 5, CCW/ACW/SW Alignment. 2) IF CCW to RCPs can NOT be re-established, THEN perform the following: a) Secure ALL RCPs. b) Verify BOTH PZR Spray valves in MANUAL and closed: • 2CV-4651 • 2CV-4652 c) IF possible, THEN maintain RCS MTS 50°F or greater. d) Verify RCP Bleedoff to VCT valves closed: • 2CV-4846-1 • 2CV-4847-2 e) Verify RCP Bleedoff to QT (2CV-4856) closed. f) **GO TO** Step 8.

(Step 7 continued on next page)

INSTRUCTIONS	CONTINGENCY ACTIONS
7. (continued)	
____C. Loop delta T less than 10°F.	
____D. RCS MTS 30°F or greater.	
E. Check SW aligned to CCW.	E. IF CCW available, THEN restore SW to CCW, refer to Exhibit 5, CCW/ACW/SW Alignment.
F. IF SIAS or MSIS actuated, THEN maintain SW header pressure greater than 85 psig.	

EOP STEP:

7. **Check Core Heat Removal by forced circulation:**

EPG STEP:

6.

DEVIATION? Yes

BASIS FOR DEVIATION:

The EOP, as is the EPG, is designed to verify single phase forced circulation is in progress in order to satisfy this safety function. If forced circulation is not occurring with at least one RCP operating, then the safety function is not satisfied, and no further actions are necessary to provide the operator with additional useful information. A transition was added therefore to permit operators to bypass unnecessary steps. If at least one RCP is running, then the operator is instructed to check that loop ΔT is less than 10°F (1) and that subcooling is greater than 30°F (2) . This is consistent with the EPG.

Additional steps were added to the EOP which check necessary auxiliaries to maintain the RCPs operating (6).

The operator is directed to check CCW flow aligned to the RCPs (3). Contingency actions are provided to establish the required flow path using a Standard Attachment (7). The operators are directed that if CCW to RCPs cannot be re-established, that the RCPs must be tripped. This is necessary to prevent or mitigate any RCP pump or motor damage as well as minimize seal degradation. If CCW flow cannot be established then the RCPs are secured. Further, as per CR5043, RECOMMENDED CHANGE IN OPERATOR ACTIONS FOLLOWING A LOSS OF COOLING TO ONE OR MORE RCP SEALS, guidance is added to maintain the RCS subcooling above 50F if possible. (8)

Taking these actions within 10 – 15 minutes of event initiation will reduce the potential of the event progressing into an induced RCP seal LOCA by a factor of 2 or more. These actions are consistent with recommendations of RCP seal vendors. The benefit of prompt CBO isolation and reducing the fluid subcooling in the RCP seal area decrease its potential for thermal degradation and hydraulic induced seal stage “pop-open”. Note that if a cooldown is directed from Loss of Off-Site Power 2202.007, the cooldown is conducted using 2203.013, Natural Circulation Operations. This procedure maintains RCS subcooling above 50F.

The operator is directed to check SW aligned to CCW and provided contingency action to establish the required flow path using a Standard Attachment (7).

The operator is directed to ensure that SW header pressure is maintained > 85 psig if SIAS or MSIS actuated (5). Directions for possible ACW restoration are provided in the ORPs.

As per 1015.021, ANO-2 EOP/AOP Users Guide (5) at least on RCP running, MTS and Loop ΔT are required to meet the safety function, and no contingency has been identified as meeting the safety function.

EXHIBIT 5

CCW/ACW/SW ALIGNMENT

1. IF SW suction NOT aligned to lake, THEN **RETURN TO** procedure in effect.

2. IF SW NOT aligned to CCW AND CCW available,
THEN perform the following:

   A. IF RCP seal temperatures less than 180°F,
   THEN restore SW to CCW by performing the following:

      1). Override and open at least ONE SW to CCW/ACW Return valve:

      - 2CV-1543-1
      - 2CV-1542-2

CAUTION

Supplying ACW flow and CCW cooling from a single SW pump may result in low SW header pressure.

      2). Override and throttle open at least ONE SW to CCW /Main Chillers Supply valve:

      - 2CV-1530-1
      - 2CV-1531-2

      3). Maintain SW header pressure greater than 85 psig.

(Step 2 continued on next page)

Table 4

Miscellaneous MOV data							
Component ID	**Description**	**Normal Position**	**SIAS Position**	**MSIS Position**	**RAS Position**	**CCAS Position**	**Other**
2CV-1425-1 2CV-1427-2	ACW Supply MOVs	Open	Closed*	Closed*	Closed (1)	N/A	
2CV-1400-1 2CV-1406-2	ESF Header Isolation MOVs	Open	Open	N/A	N/A	N/A	
2CV-1453-1 2CV-1456-2	SDC Heat Exchanger inlet valves	Closed	N/A	N/A	Open	N/A	
2CV-1511-1	Cnmt Service Water Cooling Coils 2VCC-2A & 2B Supply valve	Open	N/A	Open* (2)	N/A	Open* (2)	
2CV-1519-1	Cnmt Service Water Cooling Coils 2VCC-2A & 2B Return valve	Closed	N/A	Open*	N/A	Open*	
2CV-1510-2	Cnmt Service Water Cooling Coils 2VCC-2C & 2D Supply valve	Open	N/A	Open * (2)	N/A	Open * (2)	
2CV-1513-2	Cnmt Service Water Cooling Coils 2VCC-2C & 2D Return valve	Closed	N/A	Open*	N/A	Open*	
2CV-1530-1 2CV-1531-2	Loops 1 & 2 supplies to CCW HX's 2E28A, B, and C	Open	Closed*	Closed*	Closed (1)	N/A	
2CV-1543-1 2CV-1542-2	Loops 1 & 2 CCW/ACW Return Isolation valves	Open	Closed*	Closed*	Closed (1)	N/A	
2CV-1503-1	EDG 2K-4A Package Outlet valve	Closed					Interlocked to open when the "A" EDG Starts
2CV-1504-2	EDG 2K-4A Package Outlet valve	Closed					Interlocked to open when the "B" EDG Starts

Component ID	Description	Normal Position	SIAS Position	MSIS Position	RAS Position	CCAS Position	Other
2CV-1488-1 2CV-1489-1	Switchgear Rm Coolers 2VUC-2C & 2D Inlet valves						Valves are interlocked with the fans. Fans start on SIAS
2CV-1486-2 2CV-1487-2	Switchgear Rm Coolers 2VUC-2A & 2B Inlet valves						Valves are interlocked with the fans. Fans start on SIAS
2CV-1500-1 2CV-1502-2 2CV-1501-5	Charging pump Rm Coolers 2VUC-7A, 7B, and 7C Inlet valves						Interlocked with respective fan. Fans is interlocked with respective Charging pump which starts on SIAS
2CV-1563-1 2CV-1561-1	Electrical Equipment Unit Coolers 2VUC-19A and B Supply valves						Interlocked with fans which start on SIAS
2CV-1562-2 2CV-1564-2	Electrical Equipment Unit Coolers 2VUC-20A and B Supply valves						Interlocked with fans which start on SIAS
2CV-1529-2	EFW Pump 2P7A Room Cooler 2VUC-6A	Open					Interlocked to open when 2P7A Steam Supply valve opens
2CV-1532-2	EFW Pump 2P7B Room Cooler 2VUC-6B	Open					Interlocked to open when 2P7B starts
2CV-0711-2	Service water to 2P-7A suction	Closed					Interlocked to open when 2P-7A suction header pressure <5 psi with an EFAS signal present
2CV-0716-1	Service water to 2P-7B suction	Closed					Interlocked to open when 2P-7B suction header pressure <5 psi with an EFAS signal present
2CV-1525-1	Service water to SFP Hx	Open	Closed	Closed	N/A	N/A	normally only one SFP Hx MOV will be open
2CV-1526-2	Service water to SFP Hx	Open	Closed	Closed	N/A	N/A	normally only one SFP Hx MOV will be open

* Override capability provided

(1)Valves will close on a RAS only if they have been overridden following an SIAS or MSIS. Inadvertent RAS will not actuate these valves closed.

To protect against severe water hammers, valves will open after time delay for slow refilling of Containment Service Water Cooling Coils and piping via the bypass valve. For details on this time delay, refer to section 3.5.4.

4.0 Automatic Actuation

4.1 CSAS / SIAS

A Safety Injection Actuation Signal (SIAS) is actuated when:

- 2 out of 4 low pressurizer pressure signals (< 1650 psia; variable setpoint)

OR

- 2 out of 4 high Containment pressure signals (> 18.3 psia)

The ESFAS automatically initiates a CSAS to BOTH Containment Spray system trains as follows:

- 2 out of 4 Containment high-high pressure signals (23.3 psia)

AND

- 2 out of 4 Safety Injection Actuation Signals (SIAS)

Manual initiation may be initiated by pushbuttons from the Control Room panel 2C-03 or 2C-14.

4.2 RAS

A RAS is actuated when 2 out of 4 independent and redundant level detectors sense low RWT (2T3) level. The low level setpoint is 6.0 ± 0.5%. A RAS causes the following actions:

- Containment sump isolation valves open:
  - 2CV-5647-1 and 2CV-5648-2 (immediate open command)
  - 2CV-5649-1 and 2CV-5650-2 (37.5 second time delay open)
- RWT outlet valves 2CV-5630-1 and 2CV-5631-2 close.
- Containment Spray, LPSI, HPSI and the Master recirculation isolation valves to close,
- Service Water valves to the SDC heat exchanger(s) open, and
- LPSI pumps stop.
- The following valves will close on a RAS if they had closed on an MSIS or SIAS AND they were overridden back open:
  - CCW heat exchanger Service Water inlet valves 2CV-1530-1 and 2CV-1531-2,
  - CCW heat exchanger Service Water outlet valves 2CV-1543-1 and 2CV-1542-2, and
  - ACW isolation valves 2CV-1425-1 and 2CV-1427-2.

Question 48

Bank:	2378	**Rev:**	1	**Rev Date:**	1/12/2017	**2017 TEST QID #:**	48	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	064000A207	**10CFR55:**	41.5	**Safety Function**	6				
Title:	Emergency Diesel Generator (ED/G) System	**System Number**	064	**K/A**	A2.07				
Tier:	2	**Group:**	1	**RO Imp:**	2.5	**SRO Imp:**	2.7	**L. Plan:**	A2LP-RO-EDG
OBJ	11								

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Consequences of operating under/over-excited

Question:

When an EDG is paralleled with offsite power in an under excited condition, reactive load could be very large as indicated by _________________ and the EDG Normal Operating Procedure 2104.036 can be used to mitigate this condition by ensuring "Incoming Generator" voltage is ______________ than "System Running" voltage on the 2C-33 Indicators prior to closing the EDG output breaker.

A. leading (IN) KVARs; 100 volts higher

B. leading (IN) KVARs; 10 volts higher

C. lagging (OUT) KVARS; 100 volts higher

D. lagging (OUT) KVARS; 10 volts higher

Answer:

A. leading (IN) KVARs; 100 volts higher

Notes:

A is correct in an uderexcited condition, the system voltage will be higher than generator output voltage creating a leading power factor and there have been some occasions when the EDGs were paralleled with offsite and very large leading VARs have been placed on the EDG which create large internal heating currents in the EDG. To prevent this, guidance was place in the EDG normal operating procedure to ensure the EDG is overexcited prior to tying to the grid because the EDG cannot affect the grid as much as the grid can affect the EDG (> 100 Volts Higher).

B is incorrect because the procedure requires the Incoming Generator" voltage to be 100 volts higher on 2C-33 but plausible as the procedure allow Incoming Generator" voltage to be just higher than running voltage if read out on the SPDS computer can be higher by any value.

C is incorrect because KVARs are leading but plausible as the voltage indication on 2C-33 is correct.

D is incorrect as the KVARs are leading but plausible because the guidance in the procedure also allows the use of SPDS indications for this evolution but only requires that EDG voltage be higher than running system voltage.

This question matches the K&A as the candidate must understand the indications associated with the EDG being in an under excited condition and how to mitigate this condition when tying the EDG to the grid.

References:

STM_2-31_34-1 EDGs Section 3.1.3 (Verified reference updated 11/15/16);
NOP 2104.036 EDG Operations Rev.91 Step 9.14.2 (Verified reference updated 11/15/16).

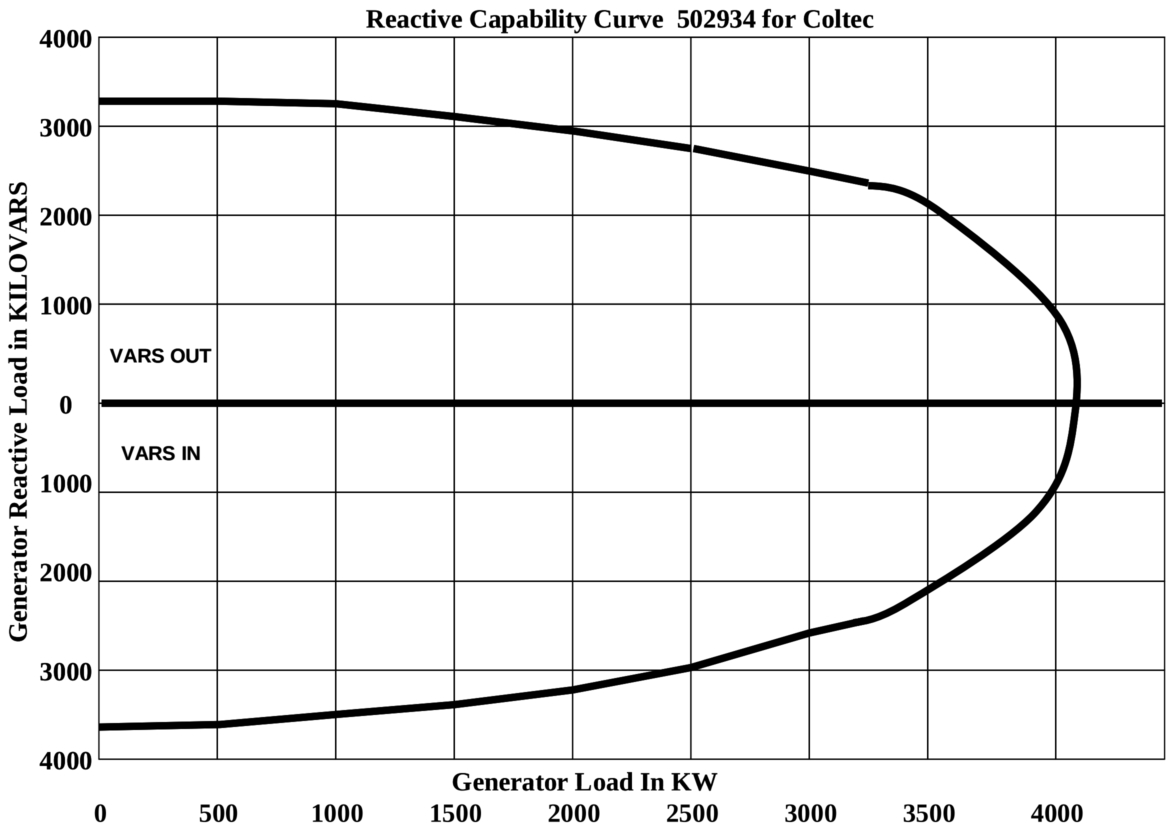
Historical Comments:

To be used on the 2017 NRC Exam

Rev. 1 based on post submittal validation comments: Added (IN) to 'A' and 'B' after leading and added (OUT) to 'C' and 'D' after lagging.

3.1.3 Discussion of Reactive Load

When the EDG is paralleled with off-site power by closing the output breaker reactive load is shared between the off-site power source and the Diesel Generator. The magnitude and direction of the reactive load is determined by the power factors of the two sources of power. There have been some problems with a large leading power factor on the EDG when closing the output breaker due to having machine voltage less than the bus voltage. On a few occasions the reactive load was very large and pegged the VAR meter to >2000 KVARS IN.



To mitigate this problem and guarantee that the EDG power factor is lagging and VARS will be out when the output breaker is closed the operating procedure has been changed as follows.

1. Voltage indication on 2C33 should indicate that incoming voltage (EDG voltage) is ~100V greater than running voltage (bus voltage).
2. If SPDS is available then SPDS should indicate that the EDG voltage is greater than bus voltage by any value. It was determined that the SPDS voltage indication is much more accurate than the volt meters on the operating panels.

Also noteworthy is that the Reactive Load capacity of the EDG is greater than the indicating range of the VAR meter. The attached curve shows the reactive load limits for varying generator loading.

NOTE

Installed watt meters and PMS point J9658 may be used to monitor 2DG1 KW.

9.14 **PERFORM** the following to synchronize and load 2DG1:

9.14.1 **PLACE** 2A-308 Synchronize switch (152-308/SS) to ON.

9.14.2 **ADJUST** Generator voltage (Incoming) using Voltage Regulator switch (CS 3) per BOTH of the following:

- Generator voltage (Incoming) approximately 100 volts higher than System voltage (Running) by 2C33 indication
- **IF** SPDS indication available,
  THEN ENSURE Generator voltage (Incoming) higher than System voltage (Running) by SPDS indication.

9.14.3 **ADJUST** frequency to cause synchroscope to rotate slowly in FAST direction using Governor Control switch (CS 4).

CAUTION

If loading delayed after DG Output breaker is closed, the anti-motoring relay can cause a Generator Lockout.

9.14.4 **WHEN** synchroscope approaches the 12 o'clock position,
THEN:

CRITICAL STEP

A. **CLOSE** 2DG1 Output breaker 2A-308 (152-308 CS).

B. **IF** this is a normal EDG run,
THEN ADJUST load to approximately 1400 KW using Governor Control switch (CS 4).

C. **IF** test being performed following EDG maintenance,
THEN ADJUST load as directed by SYE.

9.14.5 **PLACE** 2A-308 Synchronize switch (152-308/SS) in OFF.

* 9.15 **MAINTAIN** Generator KVARs between 600 KVARs IN and 1800 KVARs OUT using Voltage Regulator switch (CS 3). (Preferred range is 0 to 100 KVARs OUT unless otherwise requested by engineering or to support maintenance/testing)

9.16 **MAINTAIN** load approximately 1400 KW for 10 minutes.

Question 49

Bank:	2379	**Rev:**	1	**Rev Date:**	12/19/2016	**2017 TEST QID #:**	49	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	0100002131	**10CFR55:**	41.7	**Safety Function**	3				
Title:	Pressurizer Pressure Control System (PZR PCS)	**System Number**	010	**K/A**	2.1.31				
Tier:	2	**Group:**	1	**RO Imp:**	4.6	**SRO Imp:**	4.3	**L. Plan:**	A2LP-RO-PZR
OBJ	4								

Description: Conduct of Operations - Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Question:

With the plant at full power the Pressurizer Spray Valves 2CV-4651 and 2CV-4652 can be operated from the Control Room only if the PZR Spray Valve Hand Switches 2HS-4651D and 2HS-4652D on the Remote Shutdown Panel are in the __________ position.

A. 2C03

B. 2C04

C. 2C09

D 2C80

Answer:

B. 2C04

Notes:

B is correct: Refer to the referenced e-print in the STM. The signal from the selected pressurizer pressure controller must first pass through hand switches 2HS-4651D and 4652D located at the Remote Shutdown Panel, 2C80. The position of these hand switches determines if the spray valves will respond automatically from signals from 2C04 or instead, only to open and close signals from 2C80. With these hand switches in the "2C80" position all signals from the Control Room or the Pressurizer Pressure Control System are blocked. The spray valves can be positioned from 2C80 using 2HS-4651C and 2HS-4652C for 2CV- 4651 and 4652 respectively. Control of the spray valves from the Control Room can be regained by repositioning 2HS-4651D and 4652D to their normal "2C04" position.

A is incorrect but plausible as 2C03 is right next to and to the left of 2C04 and other Primary Plant Controls such as CEDM, CPCs and Reactor trip controls.

C is incorrect but plausible as 2C09 is right next to and to the right of 2C04 and other Primary Plant Controls such as Auxiliary PZR Spray, CVCS, and Letdown controls are on this panel.

D is incorrect but plausible as 2C04 is the correct location of the spray valve switches in the control room and 2C80 is a correct position that can be selected at the Remote Shutdown panel but in this position, spray valves can only be controlled from the Remote Shutdown Panel 2C80.

This question matches the K&A as the candidate must have knowledge of the location of the spray valve controls and know the correct normal configuration of hand switches associated with the PZR spray valve controls for full power operations.

References:

STM_2-03-1_17-1 PZR Pressure Control Section 2.2.6 (Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Removed the 1st part of each distractor and revised the stem to just ask for the 2nd part of the stem.

rise. Thus the duty cycle is inversely proportional to the square of the DC input control signal.

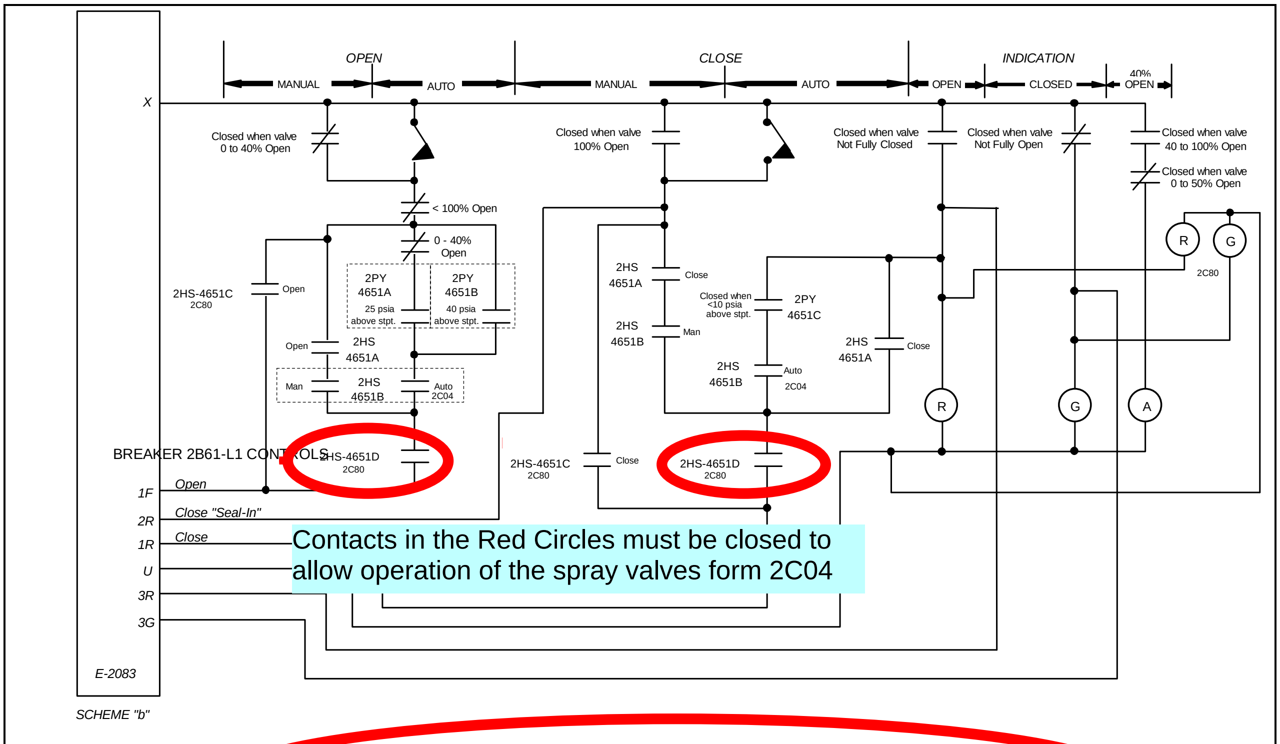
If the selected Pressurizer Level Control System channel senses that indicated pressurizer level has risen to 4.5% above the program setpoint then a relay within the Pressurizer Level Control System de-energizes. This causes a contact within the Pressurizer Pressure Control System to close. Closure of this contact completes a circuit that taps in the pressure controller output before it reaches the SCRs. This causes the controller output signal to the SCRs to go to minimum. The proportional heaters will then go to Full On regardless of what pressurizer pressure is doing. This Full On signal to the proportional heaters will stay in until pressurizer level lowers to 3.8% above the level setpoint.

If the Pressurizer Level Control System senses that pressurizer level has lowered to 29% a contact in the proportional heater circuitry will close causing the proportional heater breaker at 2B5 (Bank #1) and 2B6 (Bank #2) to trip open. If a LO-LO level condition exists caution must be exercised because the breaker will close if the handswitch is taken to the close position. If a heater is energized while it is uncovered during a low level condition, it can explode and cause damage the penetration in the vessel of the pressurizer. This would cause an unisolable leak through a primary strength boundary in the RCS and has happened at ANO.

The proportional heater breakers will also trip on bus undervoltage and can be closed from the handswitch after the undervoltage condition on the bus has cleared. Therefore, after a loss of power condition to the Pressurizer heaters the backup heaters will 'automatically' continue to operate as designed but the proportional heater handswitch must be taken to “On” to close the bus breaker.

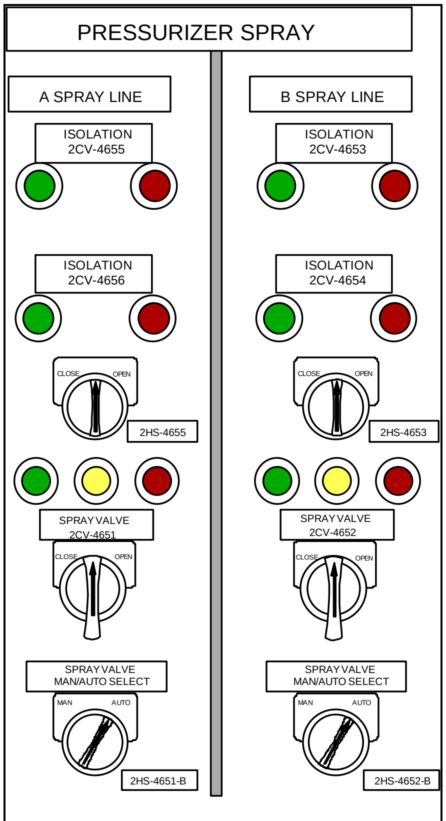
2.2.6 PZR Spray Valve Controls

The Pressurizer Spray system consists of two lines which direct water from the discharge of Reactor Coolant pump 2P-32A and 2P-32B to the pressurizer spray nozzle. Each spray line is equipped with a motor operated spray control valve which is opened as needed by the selected pressurizer pressure controller to reduce pressure back to the desired setpoint. Both spray valves have motor operated isolation valves situated upstream and downstream, which can also be controlled from 2C04. The spray valves together have a maximum design flow rate of ~468 gpm.



SPRAY VALVE 2CV-4651 SCHEMATIC DIAGRAM

(2CV-4652 SIMILAR)



Spray Valve Controls on 2C04

The spray valves are positioned automatically solely by output demand from the selected pressurizer pressure controller 2PIC-4626A or B. (Refer to the schematic diagram above.) As mentioned during the discussion of these controllers, as pressurizer pressure rises above setpoint the output demand of the controller also rises. When pressurizer pressure reaches 25 psia above setpoint the output from the pressure controller causes a contact in calculator relay 2PY-4651A and 2PY-4652A to close which in turn causes the spray valves to move in the open direction. When the spray valves reach the 40% open position a limit switch contact opens stopping valve motion at the 40% position. If pressurizer pressure continues to rise the output on the selected pressure controller rises. When pressure reaches ≥40 psia above setpoint the

output on the controller is sufficient enough to cause another calculator relay contact to close. 2PY-4651B and 2PY-4652B when closed cause their respective spray valve to go full open. As pressure lowers the controller output does the same. Whenever pressure lowers to only 10 psia above setpoint contact 2PY-4651C and 2PY-4652C close, which causes the associated valve to go full, closed.

The signal from the selected pressurizer pressure controller must first pass through handswitches 2HS-4651D and 4652D located at the Remote Shutdown Panel, 2C80. The position of these handswitches determines if the spray valves will respond automatically from signals from 2C04 or instead, only to open and close signals from 2C80. With these handswitches in the "2C80" position all signals from the Control Room or the Pressurizer Pressure Control System are blocked. The spray valves can be positioned from 2C80 using 2HS-4651C and 2HS-4652C for 2CV-4651 and 4652 respectively. Control of the spray valves from the Control Room can be regained by repositioning 2HS-4651D and 4652D to their normal "2C04" position.

The Spray valves will respond to automatic signals from the Pressurizer Pressure Control System only if their respective auto/manual handswitches are in "Auto". These handswitches, 2HS-4651B and 4652B, are located on 2C04.

If the Auto/Manual handswitches are in "Manual", the spray valves can be positioned from 2C04 by using 2HS-4651A and 4652A. These handswitches are "spring-return-to-neutral" handswitches that can be used by the Operator to throttle the spray valves as desired. The spray valves can only be throttled in the Open direction. If the valves are open and their manual handswitches are placed in the Close position, a seal-in contact at the breaker will close as soon as the close contactor closes. This seal-in will drive the valve in the closed direction until the valve is fully closed.

2.2.7 Pressurizer Pressure Control System Annunciators

The Pressurizer Pressure Control System also causes two annunciators to alarm if pressure should reach the alarms' respective setpoints. The following table lists each annunciator associated with the Pressurizer Pressure Control System as well as the parameters that will cause the alarm.

WINDOW NUMBER	DESCRIPTION	CAUSES
2K10-E6	Cntrl CH1 Press Hi/Lo	• High = ≥ 2340 psia 2PS-4626A • Low = ≤ 2100 psia 2PS-4626A
2K10-E7	Cntrl CH 2 Press Hi/Lo	• High = ≥ 2340 psia 2PS-4626B • Low = ≤ 2100 psia 2PS-4626B

Question 50

Bank:	2380	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	50	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	012000K102	**10CFR55:**	41.7	**Safety Function**	7				
Title:	Reactor Protection System	**System Number**	012	**K/A**	K1.02				
Tier:	2	**Group:**	1	**RO Imp:**	3.4	**SRO Imp:**	3.7	**L. Plan:**	A2LP-RO-RPS
OBJ	5								

Description: Knowledge of the physical connections and/or cause-effect relationships between the RPS and the following systems: - 125V dc system

Question:

Reactor Trip Circuit Breakers, TCBs, 4 and 8 have two redundant tripping circuits that ensure the TCBs open when a signal is received from RPS. TCBs 4 and 8 tripping circuits are powered from ________________.

A. 2D36

B. 2D25

C. 2RS2

D. 2Y2

Answer:

A. 2D36

Notes:

A is Correct: TCBs 4 through 8 are tripped automatically by signals from the Reactor Protection System (RPS) portion of the Plant Protection System (PPS). The breakers contain a Shunt Trip Coil and an Under Voltage Coil. When a Manual or Automatic Reactor Trip is generated, the Shunt Trip Coil is energized and the Under voltage Coil is de-energized. The Trip circuits for the TCBs 4 and 8 are fed from vital 125 VDC Battery eliminator 2D36.

B is incorrect but plausible as 2D25 another source of 125V DC but is used for Emergency DC loads in the Turbine Building.

C is incorrect but plausible as the Vital 120 VAC 2RS2 is the power supply to the RPS control cabinet relays that send the signals to the TCB relays to energize the tripping circuits from DC.

D is incorrect but plausible as 2Y2 is a source of 120VAC Instrumentation Bus used for Reactor Control Systems such as PZR Pressure and Level Control Instrumentation and controls

This question matches the K&A because it requires knowledge of the interrelationship between the RPS system and vital 125V DC system.

References:

STM_2-02_24-1 CEDM Section 5.3 TCBs (Verified reference updated 11/15/16);
STM_2-63_12-1 RPS Section 5.0 RPS Trip Paths (Verified reference updated 11/15/16);

Historical Comments:

To be used on the 2017 NRC Exam.

REV. 1 based on NRC Chief Examiner Feedback BNC. Added specific TCBs 4 and 8 to the stem and specific buses to the distractors removing the noun name. Updated analysis.

- When the synchroscope nears 12 o'clock, place the Paralleling Switch to the position corresponding to the MG Set just started.
- When the Output breaker closes, the switch is released and the Synchroscope switch turned off.

The MG Sets have a synchronizing interlock that prevents paralleling them out of phase.

It is also imperative that the operator (self checking) verifies that the voltage buildup light does not come on for at least 30 seconds. A 35 second timer 1TR is activated when the MOTOR ON Pushbutton is depressed on a startup. After 35 seconds has elapsed, the 1TR contact will close and the VOLTAGE BUILDUP light will come on. That is the time it takes for the motor to accelerate to full speed. It is important to wait for full speed operation before establishing exciter field voltage and current.

5.2.2 M-G Set Shutdown

Shutting down the MG Sets is a relatively simple operation with very few steps. If the Reactor and the redundant MG Set are to remain on line, the Operator needs to verify that TCB #9 is closed before performing any other step.

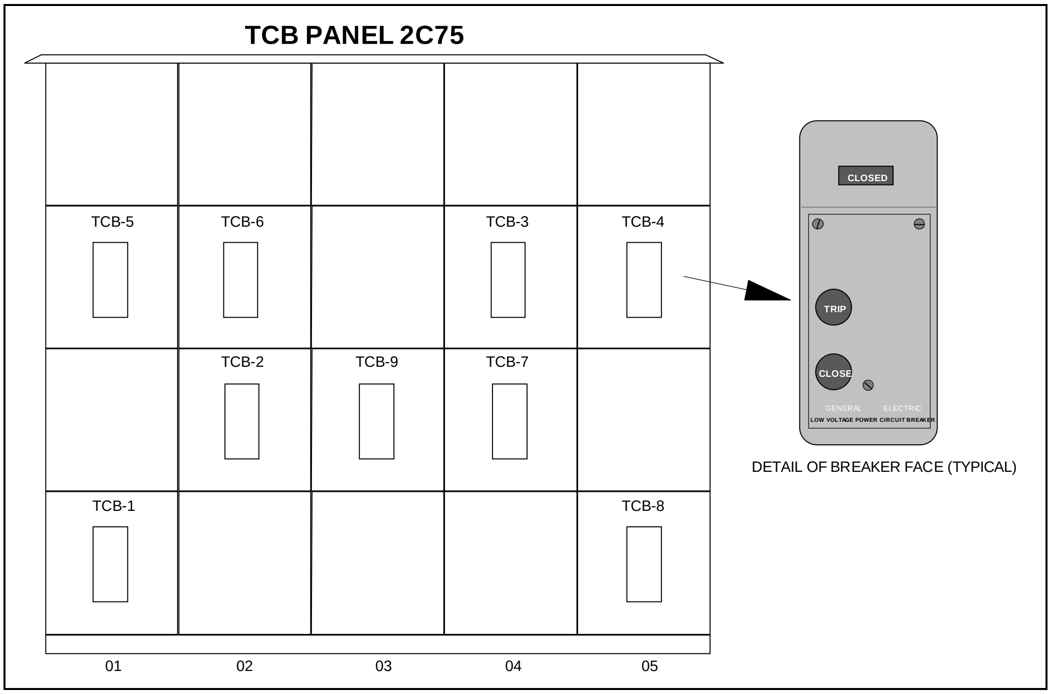
To shutdown a MG Set:

- Open the MG set Output Breaker by depressing the LOAD OFF pushbutton.
- When the Output Breaker opens (Located inside lower right section of cabinet), then depress and hold MOTOR OFF push-button.
- When the MOTOR OFF light is illuminated, then release the MOTOR OFF pushbutton.
- The Motor Input Circuit Breaker is then manually opened.

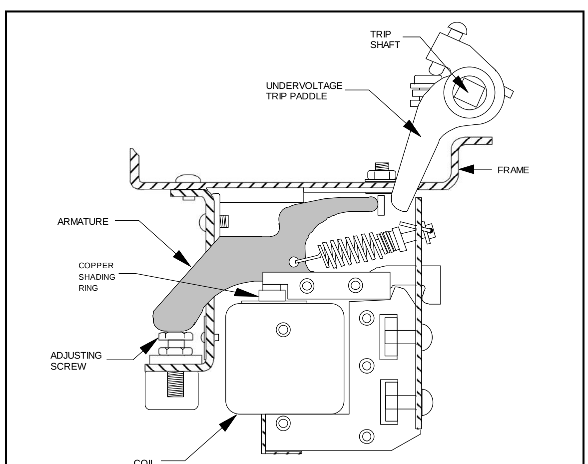
It takes about 2.5 hours for the MG Set to coast to a stop after it is shut down.

5.3 Trip Circuit Breakers

Eight Reactor Trip Circuit Breakers (TCBs) and one Synchronizing breaker are installed to control power being supplied to the CEAs. These TCBs are on panel 2C75 located in the CEDMCS/Remote Shutdown Panel Room. Four of the TCBs are associated with each MG set. Output of MG set #1 is supplied through TCBs 1, 2, 5, and 6. TCBs 1 and 2 supply CEDMCS #1 power supply while 5 and 6 supply CEDMCS #2 power supply. Output of MG set #2 is supplied through TCBs 3, 4, 7, and 8. TCBs 3 and 4 provide a redundant supply to CEDMCS #2 power supply while TCBs 7 and 8 provide a redundant supply to CEDMCS #1 power supply. Breaker #9 is a crosstie breaker between the two MG sets.



TCBs 1 through 8 are tripped automatically by signals from the Reactor Protection System (RPS) portion of the Plant Protection System (PPS). The breakers contain a Shunt Trip Coil and an Undervoltage Coil. When a Manual or Automatic Reactor Trip is generated, the Shunt Trip Coil is energized and the Undervoltage Coil is de-energized. Either of these actions will cause the breaker to open. The circuit breakers are closed from the PPS cabinets. TCB #9 receives no trip signals and is only operated manually. In the figure above, a detail of the TCB face is shown. The Trip pushbutton shown is a mechanical trip pushbutton which at each breaker, is covered by a flat piece of metal stock. This is to prevent inadvertent operation when the cubicle door is closed.



TRIP CIRCUIT BREAKER UNDERVOLTAGE TRIP DEVICE
COIL ENERGIZED POSITION

A Limit and Precaution in 2105.009 requires that the armature on the TCB UV coil be verified to be in contact with the air gap adjustment screw prior to closing the breaker. The reason for this requirement is due to the fact that the UV coil is energized when the

RPS trip path resets. If the armature is not in contact with the adjustment screw it may cause the breaker to trip back open or it might prevent the UV coil from causing a breaker trip when required. If the armature is not down it can be easily pushed down with a finger. If spring pressure is felt the UV coil is not powered up or is defective.

5.3.1 TCB Control Power

The Trip circuits for the TCBs are fed from vital 125 VDC buses while the Closing circuits are powered from separate vital 125 VDC supplies. Refer to the figure on page 82. Two TCBs share the same control power sources. The table below lists the control power sources for each of the TCBs. Note that the same sources power two TCBs. The exception is TCB-9 which shares its control power supplies with TCB-2 and TCB-6.

TCB Control Power Supplies			
TCB	**MG Set**	**Trip Circuit Control Power**	**Closing Circuit Control Power**
TCB-1 (PPS K1)	A	2RA1 (Red) Bkr # 2	2D21 Bkr #9
TCB-5 (PPS K1)	A	2RA1 (Red) Bkr # 2	2D21 Bkr #9
TCB-2 (PPS K2)	A	2RA2 (Grn) Bkr # 2	2D22 Bkr #34
TCB-6 (PPS K2)	A	2RA2 (Grn) Bkr # 2	2D22 Bkr #34
TCB-3 (PPS K3)	B	2D35 (Yellow)	2D21 Bkr #12
TCB-7 (PPS K3)	B	2D35 (Yellow)	2D21 Bkr #12
TCB-4 (PPS K4)	B	2D36 (Blue)	2D22 Bkr #17
TCB-8 (PPS K4)	B	2D36 (Blue)	2D22 Bkr #17
TCB-9	A/B	2RA2 (Grn) Bkr # 2	2D22 Bkr #34

5.0 RPS Trip Paths

The RPS Trip Paths are used to provide automatic opening of the Reactor Trip Circuit Breakers (TCBs). This action interrupts power to the CEDMs allowing the CEAs to drop into the core.

There are four Trip Paths designated TP1, TP2, TP3, and TP4. Each RPS Trip Path controls a Trip Path Relay (which is a solid state relay, sometimes designated as SSR) set, which in turn controls respective TCB control relays (“K” relays). The table below shows the component relationships between the Trip Paths and the TCBs.

RELAY	TRIP PATH	LOCATION	ACTUATES
K1	Trip Path 1	Channel A	TCBs 1 & 5
K2	Trip Path 2	Channel B	TCBs 2 & 6
K3	Trip Path 3	Channel C	TCBs 3 & 7
K4	Trip Path 4	Channel D	TCBs 4 & 8

Again, refer to the figure on page 56. Each Trip Path contains six series contacts. Each one of the six series contacts in a given trip path is operated by a Logic Matrix Relay in each of the six different Logic Matrices Ladders. For example Trip Path 1 contains contacts from Logic Matrix Relays AB1, BC1, BD1, AC1, CD1 and AD1. In all there are 4 Trip Paths with 6 Matrix Contacts in each. With this contact arrangement, you readily see how any "2-out-of-4" channel actuation will trip the TCBs.

Each Trip Path controls three solid state Trip Path relays (SSRs). The three SSRs control the TCB control relays K1 through K4, and also provides for indication and annunciators (see page 64).

Each TCB Control Relay, K(X), operates two pair of contacts;

- One pair associated with two parallel Under Voltage (UV) trip coils
- The other pair, associated with two parallel Shunt Trip (ST) coils.

The figures on pages 70 and 71 illustrate RPS Trip Path 1 in Channel A. The first figure shows the trip path de-energized or with a trip signal present. The second figure illustrates the Trip Path when no trip is initiated.

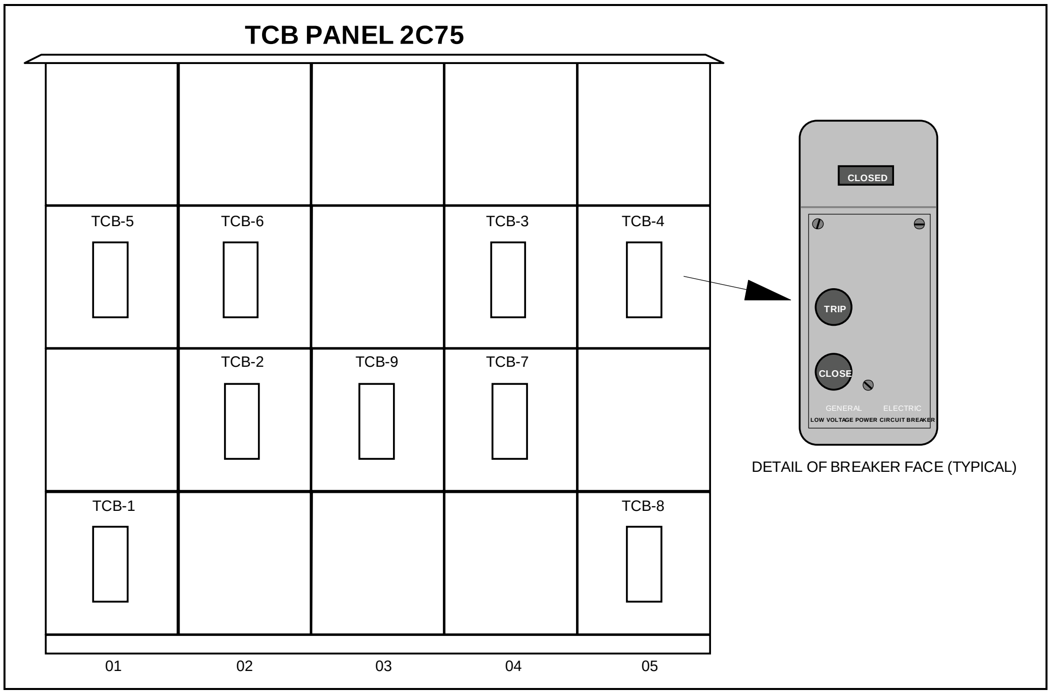
Each Trip Path and TCB control relay is associated with two parallel TCBs. Each TCB can be tripped by either de-energizing its associated UV coils or by energizing its associated ST coils. The UV and ST coils for a given TCB are located in the Reactor Trip Switchgear cabinet 2C75. These coils are connected in parallel across a common 125 vDC power supply (2RA-1, 2RA-4, 2D-35, and 2D-36). The TCB control relays, in 2C23(X), are powered by 120 vAC vital power supplies 2RS-1, 2RS-2, 2RS-3, and 2RS-4.

De-energizing any one TCB control relay causes its associated contacts to operate in such a fashion as to de-energize the UV coils and energize the ST coils for the parallel TCBs controlled by that TCB control relay. This action causes the two TCBs controlled by that TCB control relay to trip open. This results in one of the two parallel AC power paths to the CEDMs (Reactor Trip Switch Gear) to be de-energized. In order to completely remove power to the CEDMs (CEAs), at least four TCBs must be opened, two in each parallel AC power path to the Reactor Trip Switch Gear. CEDMCS busses 2C70 and 2C71 each house 10 sub-group power switches that control CEA coil power. For more details refer to STM 2-02, Control Element Drive Mechanism System.

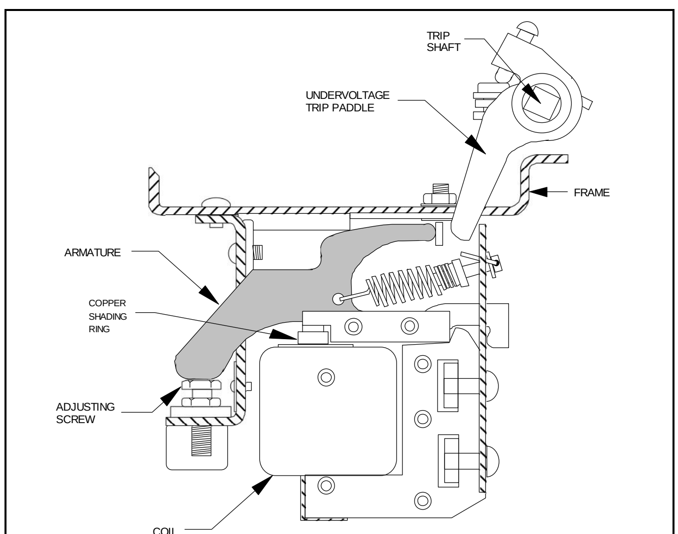
2C70 and 71 each contain two separate UV relays. When MG power to the CEAs is de-energized, these UV relays send trip information to the Turbine and Feedwater control cabinets for trip status and to the PPS local and remote trip status panels for indication.

5.1 Trip Circuit Breakers

Eight Reactor Trip Circuit Breakers (TCBs) and one Synchronizing breaker are installed to control power being supplied to the CEAs. These TCBs are on panel 2C75 located in the CEDMCS/Remote Shutdown Panel Room. Four of the TCBs are associated with each MG set. Output of MG set #1 is supplied through TCBs 1, 2, 5, and 6. TCBs 1 and 2 supply CEDMCS #1 power supply while 5 and 6 supply CEDMCS #2 power supply. Output of MG set #2 is supplied through TCBs 3, 4, 7, and 8. TCBs 3 and 4 provide a redundant supply to CEDMCS #2 power supply while TCBs 7 and 8 provide a redundant supply to CEDMCS #1 power supply. Breaker #9 is a crosstie breaker between the two MG sets.



TCBs 1 through 8 are tripped automatically by signals from the Reactor Protection System (RPS). The breakers contain a Shunt Trip Coil and an Undervoltage Coil. When a Manual or Automatic Reactor Trip is generated, the Shunt Trip Coil is energized and the Undervoltage Coil is de-energized. Either of these actions will cause the breaker to open. The circuit breakers are closed from the PPS cabinets. TCB #9 receives no trip signals and is only operated manually. In the figure above, a detail of the TCB face is shown. The Trip push-button shown is a mechanical trip push-button which at each breaker, is covered by a flat piece of metal stock. This is to prevent inadvertent operation when the cubicle door is closed.



TRIP CIRCUIT BREAKER UNDERVOLTAGE TRIP DEVICE

COIL ENERGIZED POSITION

A Limit and Precaution in procedure 2105.009, requires that the armature on the TCB UV coil is verified to be in contact with the air

Question 51

Bank:	2381	**Rev:**	0	**Rev Date:**	7/21/2016	**2017 TEST QID #:**	51	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	061000K202	**10CFR55:**	41.7	**Safety Function**	4				
Title:	Auxiliary / Emergency Feedwater (AFW) System	**System Number**	061	**K/A**	K2.02				
Tier:	2	**Group:**	1	**RO Imp:**	3.7	**SRO Imp:**	3.7	**L. Plan:**	A2LP-RO-EFW	**OBJ**	10

Description: Knowledge of bus power supplies to the following: - AFW electric driven pumps

Question:

The bus power supply to the 2P-7B Electric Emergency Feedwater Pump is:

A. 2A1 4160 VAC Bus

B. 2A2 4160 VAC Bus

C. 2A3 4160 VAC Bus

D. 2A4 4160 VAC Bus

Answer:

C. 2A3 4160 VAC Bus

Notes:

C is correct as the 2P-7B is the Red Train (#1 Train) EFW pump at ANO Unit 2 (2A3 Vital 4160 VAC Bus). 2P-7A is the Green Train (#2 Train) at ANO Unit 2.

A is incorrect as this is the Red Train Non-Vital 4160 VAC Bus but plausible as this is a 4160 VAC Bus and feeds Bus 2A3.

B is incorrect as this is the Green Train Non-Vital 4160 VAC Bus but plausible as this is a 4160 VAC Bus and feeds Bus 2A4.

D is incorrect as this is the Green Train Vital 4160 VAC Bus but plausible as this is a 4160 VAC Bus that powers the 2P-7A EFW pump.

This question matches the K&A as it requires knowledge of the bus power supply to the electric driven EFW Pumps. (AFW at ANO is a non-vital BU feedwater pump) EFW are the safety related pumps.

References:

STM 2-19-2_39 EFW and AFW Systems Section 2,1,2 (Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam

* At T = 17 seconds (2CV-0340-2 is 20% open), ramp circuit contacts close, energizing and ramping speed upward. The turbine is on ramp control vice idle speed control.
* At T= 34 seconds, turbine speed at 3790 rpm control swaps to 2HIC-0336-2 demand. By now 2CV-0340-2 is full open causing 2CV-0205-2 to close.

Unit 2 strives to keep the steam admission valve seat leakage at a minimum. The heat and water from steam that leaksby is not good for the turbine glands. So, when the seat leakage is small, the turbine starts look different from a normal “hot” start profile. There is a large amount of energy in the steam that is lost to the cold piping. This lost of energy flattens the start profile which causes the profile to look smooth or flat apon a pump start. Also, the turbine speed profile will be slightly different depending upon oil temperature. If the oil is cold, the governor will react slowly, and the speed may not go to idle prior to the ramp signal increasing the turbine speed. When the turbine is warm, the speed profile will be as shown, with some stabilization at idle speed prior to ramping up.

2.1.2 Motor Driven EFW Pump

The 600 horsepower, motor driven EFW pump, 2P-7B, is powered from 2A-311. The pump is controlled from 2C-17 by handswitch 2HS-0710A-1. The Emergency Feedwater Actuation Signal, EFAS, provides an automatic start signal to 2P-7B on a low level in either S/G of 22.2%.

The following interlocks must be met for 2P-7B to automatically start:

* 2C-17 Handswitch (2HS-0710A-1) <u>NOT</u> in Pull-To-Lock
* S/G level less than 22.2%
* EITHER: The 2A3 normal feeder breaker, 2A-309, closed

<u>AND</u>

No Undervoltage condition on 2A3 (Relays 127-2A3/XA and 127-2A3/XB deenergized)

<u>OR</u>

The EDG feeder breaker to 2A3, 2A-308, closed.

The automatic start of 2P-7B is delayed for 85 seconds after EFAS actuation to allow for sequential loading on the Emergency Diesel Generator. This start delay also ensures that 2P-7A has started is up to speed and controlled by governor and thereby reduces the risk of reducing suction pressure too low. When the EFW pump suction pressure lowers to 5 psig and an EFAS actuation is present, the suction automatically swaps to the Service Water system.

Several annunciators provide the operator with indication of 2P-7B problems. Annuciator 2K07-A9, “2P7B FAILURE ON EFAS”, is activated when 2P-7B breaker, 2A-311 fails to close within 100 seconds after an EFAS signal has occurred. Annuciator 2K07-B9, “2P7B

Question 52

Bank:	2382	**Rev:**	1	**Rev Date:**	12/19/2016	**2017 TEST QID #:**	52	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	006000K408	**10CFR55:**	41.7	**Safety Function**	2				
Title:	Emergency Core Cooling System (ECCS)	**System Number**	006	**K/A**	K4.08				
Tier:	2	**Group:**	1	**RO Imp:**	3.4	**SRO Imp:**	3.6	**L. Plan:**	A2LP-RO-ECCS
OBJ	12								

Description: Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following: - Recirculation flowpath of reactor building sump

Question:

Given the following:

* Unit 2 has tripped from full power due to a Large break LOCA.
* RCS Pressure is 150 psia and slowly lowering.
* Containment pressure is 28 psia and lowering.
* Both Trains of ECCS Equipment are in operation as designed.

NOW

* A Recirculation Actuation Signal (RAS) is generated.

With no operator action, which of the following is the correct component positions to verify after the RAS actuations have completed?

A. LPSI pumps trip, HPSI and Containment Spray pumps remain running,
RWT outlet isolations remain Open.

B. Containment Spray pumps trip, HPSI and LPSI pumps remain running,
RWT outlet isolations Close.

C. LPSI pumps trip, HPSI and Containment Spray pumps remain running,
RWT outlet isolations Close.

D. Containment Spray pumps trip, HPSI and LPSI pumps remain running,
RWT outlet isolations remain Open.

Answer:

C. LPSI pumps trip, HPSI and Containment Spray pumps remain running,
RWT outlet isolations Close.

Notes:

C is correct : After an RAS, the LPSI pumps trip, HPSI and Containment Spray pumps continue running, and the RWT outlet isolations Close. The LPSI pumps trip to allow more NPSH for the HPSI pumps which are providing continued core cooling during the LOCA and the Spray pumps which are providing continued containment building cooling.

A is incorrect because the RWT outlets close to prevent the rest of the RWT water from being transferred to the Containment Sump without going through the core due to no check valves in the sump outlet flowpath and the Containment sump valves opening. Question is plausible as all of the other component positions are correct.

B is incorrect because the spray pumps continue to run and LPSI pumps trip but plausible as all the other component positions are correct.

D is incorrect because the spray pumps continue to run and LPSI pumps trip and the RWT outlets close but plausible as the HPSI pump continues to run.

This question matches the K&A as the candidate has to have knowledge of the interlocks associated with a RAS to understand which pumps are providing the flow path to the core and Containment after RAS and where the flow comes from.

References:

EOP 2202.003 LOCA REV 15 Step 22 (Verified reference updated 11/15/16);
EOP 2202.010 Standard Attachment Rev. 23 Attachment 16 RAS Verification (Verified reference updated 11/15/16);
STM_2-05_REV. 31 ECCS Section 3.8 (Verified reference updated 11/15/16);
STM_2-05_REV 31 ECCS Simplified Drawing (Verified reference updated 11/15/16);
STM 2-08 Containment Spray System Section 4.2 (Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Added "With no operator action" to the beginning of the stem. Added the word "remain" after pumps in each distractor and answer. Added the word "remain" prior to Open in distractors A and D.

INSTRUCTIONS	CONTINGENCY ACTIONS
■ **22. <u>WHEN</u> RWT level less than 6%, <u>THEN</u> perform the following:**	
A. Check CNTMT sump level greater than 86 inches.	A. <u>IF</u> CNTMT sump level less than 86 inches, <u>THEN</u> perform the following: 1) Verify ALL HPSI pumps in PTL. 2) Notify TSC.
B. Check RAS actuated on PPS inserts.	B. Manually actuate the following: • Train A RAS on 2C39 • Train B RAS on 2C40
C. Verify CNTMT Spray header flow greater than 2000 gpm.	
D. Verify RAS components actuated using 2202.010 Attachment 16, RAS Verification.	
E. Check EACH running HPSI pump flow greater than 240 gpm.	E. <u>IF</u> ANY running HPSI pump flow less than 240 gpm, <u>THEN</u> perform the following: 1) Stop Charging pumps one at a time as needed to raise HPSI flow to greater than 240 gpm per pump. **<u>NOTE</u>** **Core Cooling is provided with ONE HPSI pump in service.** 2) <u>IF</u> EACH HPSI header flow still less than 240 gpm with ALL Charging pumps stopped, <u>THEN</u> perform the following: a) Transfer ALL HPSI flow to ONE HPSI pump. b) Place HPSI pump <u>NOT</u> aligned to RCS in PTL.

(Step 22 continued on next page)

ATTACHMENT 16
RAS VERIFICATION

Components found in a position other than specified are to be reported to the CRS.

COMPONENT DESCRIPTION	**NUMBER**	**LOCATION**	**POSITION**	√
LPSI PUMP	2P60A	2C17	OFF	
RWT 2T3 OUTLET	2CV-5630-1	2C17	CLOSED	
MINI RECIRC LPSI 2P60A ISOL	2CV-5123-1	2C17	CLOSED	
MINI RECIRC LPSI 2P60B ISOL	2CV-5124-1	2C17	CLOSED	
MINI RECIRC CNTMT SPRAY 2P35A ISOL	2CV-5673-1	2C17	CLOSED	
MINI RECIRC CNTMT SPRAY 2P35B ISOL	2CV-5672-1	2C17	CLOSED	
MINI RECIRC HPSI 2P89A ISOL	2CV-5126-1	2C17	CLOSED	
MINI RECIRC HPSI 2P89B ISOL	2CV-5128-1	2C17	CLOSED	
MINI RECIRC HPSI 2P89C ISOL	2CV-5127-1	2C17	CLOSED	
SW INLET SDN CLG HX 2E35A	2CV-1453-1*	2C17	OPEN	
CNTMT SUMP SUCTION ISOL	2CV-5647-1*	2C17	OPEN	
CNTMT SUMP SUCTION ISOL	2CV-5649-1*	2C17	OPEN	
LPSI PUMP	2P60B	2C16	OFF	
ESF RECIRC HEADER ISOL	2CV-5628-2	2C16	CLOSED	
RWT 2T3 OUTLET	2CV-5631-2	2C16	CLOSED	
SW INLET SDN CLG HX 2E35B	2CV-1456-2*	2C16	OPEN	
CNTMT SUMP SUCTION ISOL	2CV-5648-2*	2C16	OPEN	
CNTMT SUMP SUCTION ISOL	2CV-5650-2*	2C16	OPEN	

* Denotes override capability

steel metal components with appropriately selected elastomers. The Haskel pump has two hydraulic cylinders coupled to a single pneumatic cylinder, referred to as a double acting pump. One cycle results in a compression stroke on each of the two hydraulic cylinders.a Haskel pump that uses Instrument Air (IA) for motive force and has a variable output capacity depending on discharge pressure, air supply pressure, throttle valve setting and HPSI Header leak rate. At approximately 45 psig air pressure and 650 psig discharge pressure, the Haskel pump will produce anywhere from zero to 2.75 GPM flow depending on the leak rate of the HPSI header. If the header is leak tight, the pump will just maintain pressure with no flow. If the header has a leak up to the capacity of the Haskel, the pump will maintain header pressure at that HPSI header leak rate. The HPS system can maintain HPSI header pressure between 635 and 685 psig with a leak out of the HPSI header as high as 2.75 GPM and not exceed the manufacturer’s recommended 40 cycles per minute (CPM) for maximum pump life

Filtered Instrument air (2F-811A/B/C) is provided to the HPS pump air regulators. These HPS pump air regulators (2PCV-5200A, B and C) and air supply throttle valves (2HPS-14, 21 and 27) are used to set the discharge pressure control band for maintaining the HPSI train slightly above SIT pressure. As HPSI pressure slowly decays, the HPS pump automatically strokes as necessary to keep pressure in the desired band. The HPS pumps are located in the east end of the 2T-21 Tank Room.

EC-704 (HPSI Pressurization System Creation), P&IDs M-2232-1/2 (ECCS) and OP 2104.039 (HPSI System Operation) contains details of the HPS system installation and operation and is shown on page 90.

3.8 HPSI Operational Summary

Normal operation of the ECCS system includes hot standby operation and power generation with the reactor coolant system at normal operating pressure and temperature.

During normal operations, the HPSI system is lined up for standby. There are no components in active operation in this condition. The system is lined up to take suction from the refueling water tank (RWT).

A safety injection actuation signal is initiated from either a two-out-or-four LOW-LOW pressurizer pressure (1650 psia) signals or two-out-or-four HIGH CONTAINMENT PRESSURE (18.3 psia) signals. LOW-LOW pressurizer pressure and HIGH containment pressure are indicative of a loss-of-coolant incident or an overcooling event such as a main steam line break. Upon a loss of coolant incident or overcooling event, the reactor is tripped with the SIAS actuation. The turbine is tripped by the reactor trip.

The operation of the HPSI pumps, along with the charging pumps, is effective in maintaining the core covered for small ruptures.

If standby power is available (STARTUP TRANSFORMER) following generation of a SIAS and turbine-generator and reactor trips, a fast transfer of 2A-1 and 2A-2 buses (which supplies power to the

safeguards buses 2A-3 and 2A-4) occurs. The following equipment is then actuated:

- two HPSI pumps (2P-89C is normally in PULL-TO-LOCK)
- two LPSI pumps
- eight HPSI and four LPSI injection valves

All other components of the HPSI system are in the correct position for emergency operation. It should be noted that time delays in the starting circuits of the pumps are effective even with offsite power available.

If standby power is NOT available, the fast transfer of 2A-1 and 2A-2 (and subsequently the power supply to the safeguards buses) is blocked. The plant emergency diesel generators are automatically started and all loads on the safeguards buses shed. Once the diesel generator attains normal voltage and its corresponding safeguards bus is unloaded, the diesel output breaker closes. The HPSI pumps will then start with a 10 second time delay. This time delay is part of the load start sequencing that prevents overloading the diesel generator.

When the RWT level lowers to a preset level (6%), the recirculation actuation signal (RAS) automatically causes the following actions:

- opens the containment sump isolation valves
- closes the RWT outlet valves
- stops the LPSI pumps
- closes all minimum recirculation valves to the RWT
- opens service water valves to the SDC heat exchangers
- closes ACW cooling water header isolation valves (2CV-1427-2 and 2CV-1425-1).

The HPSI pumps continue to run providing sufficient flow to keep the reactor core cooled and prevent core damage.

When at least three hours have elapsed since the start of a LOCA and if SDC cannot be established within 4 hours of the start of the LOCA, then HPSI is aligned for hot leg injection. This requires the orifice bypass valves be shut. The HPSI hot leg injection valves are then adjusted to establish a balance of flow such that half of the total flow is through the cold leg injection piping and the other half through the hot leg injection path. This helps preclude boron precipitation in the core region that could hinder core heat removal.

3.8.1 HPSI Use in RCS Fill

During an initial RCS fill, the normal process uses the charging system to supply water. If it is desired to raise the rate of fill above the capacity of the charging system, a HPSI pump can be lined up for injection. If this is done, the flow from the HPSI pump should be maintained at approximately 200 gpm. This limits flow into the RCS at less than the capacity of one LTOP relief valve.

The HPSI system is also used to fill the safety injection tanks (SIT). The HPSI pump takes suction from the RWT and fills the SIT through its drain line through a solenoid operated bypass valve around the SIT outlet check valve.

4.0 Automatic Actuation

4.1 CSAS / SIAS

A Safety Injection Actuation Signal (SIAS) is actuated when:

- 2 out of 4 low pressurizer pressure signals (< 1650 psia; variable setpoint)

OR

- 2 out of 4 high Containment pressure signals (> 18.3 psia)

The ESFAS automatically initiates a CSAS to BOTH Containment Spray system trains as follows:

- 2 out of 4 Containment high-high pressure signals (23.3 psia)

AND

- 2 out of 4 Safety Injection Actuation Signals (SIAS)

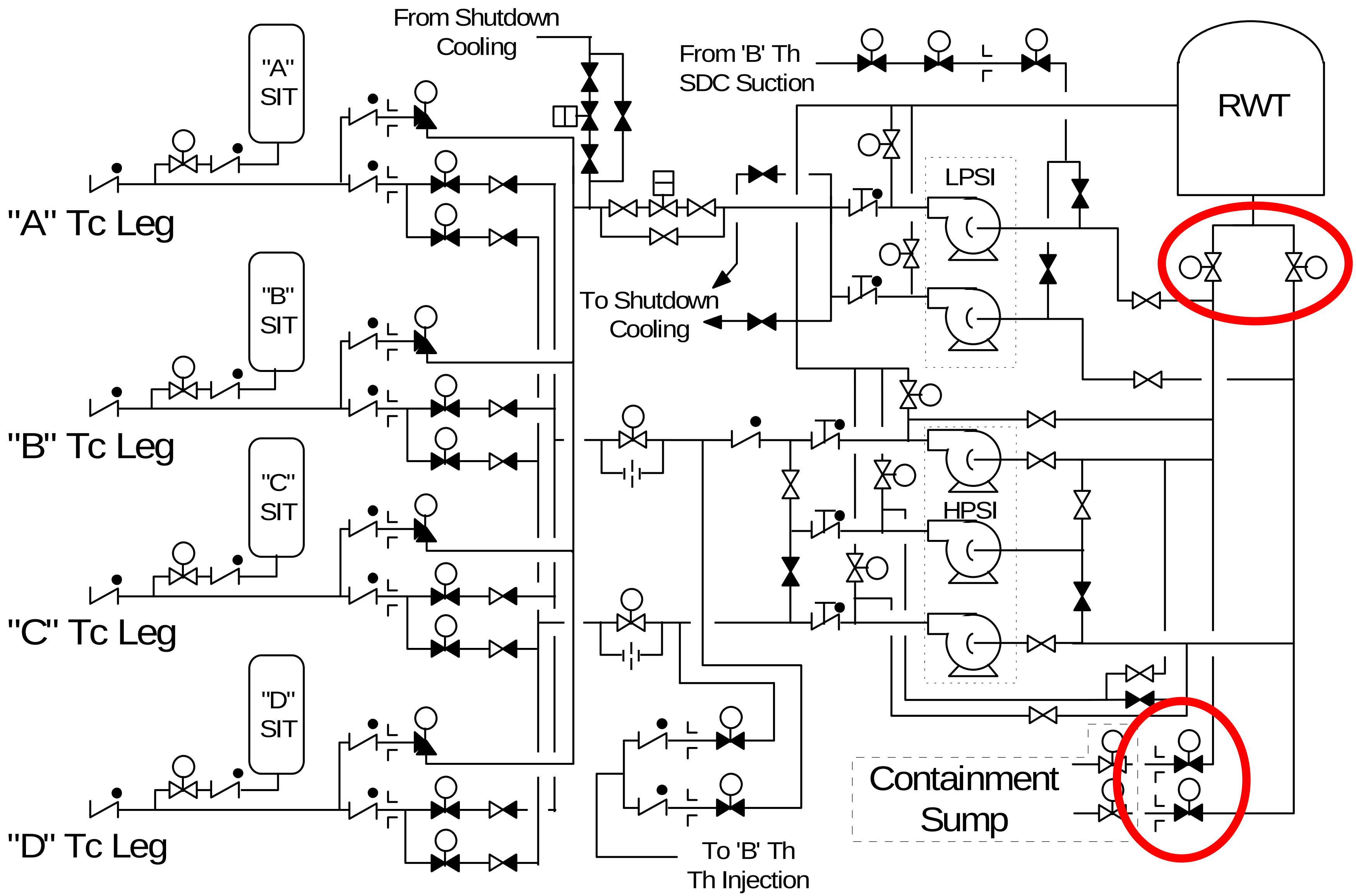
Manual initiation may be initiated by pushbuttons from the Control Room panel 2C-03 or 2C-14.

4.2 RAS

A RAS is actuated when 2 out of 4 independent and redundant level detectors sense low RWT (2T3) level. The low level setpoint is 6.0 ± 0.5%. A RAS causes the following actions:

- Containment sump isolation valves open:
  - 2CV-5647-1 and 2CV-5648-2 (immediate open command)
  - 2CV-5649-1 and 2CV-5650-2 (37.5 second time delay open)
- RWT outlet valves 2CV-5630-1 and 2CV-5631-2 close.
- Containment Spray, LPSI, HPSI and the Master recirculation isolation valves to close,
- Service Water valves to the SDC heat exchanger(s) open, and
- LPSI pumps stop.
- The following valves will close on a RAS if they had closed on an MSIS or SIAS AND they were overridden back open:
  - CCW heat exchanger Service Water inlet valves 2CV-1530-1 and 2CV-1531-2,
  - CCW heat exchanger Service Water outlet valves 2CV-1543-1 and 2CV-1542-2, and
  - ACW isolation valves 2CV-1425-1 and 2CV-1427-2.

Figures



ECCS System Simplified Drawing

Question 53

Bank:	2383	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	53	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	008000K303	**10CFR55:**	41.7	**Safety Function**	8				
Title:	Component Cooling Water System (CCWS)	**System Number**	008	**K/A**	K3.03				
Tier:	2	**Group:**	1	**RO Imp:**	4.1	**SRO Imp:**	4.2	**L. Plan:**	A2LP-RO-RCP
OBJ	5								

Description: Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: - RCP

Question:

In accordance with AOP-2203.025, RCP Emergencies, which of the following is the primary reason for the time limit for operating RCPs during a Loss of CCW event?

A. Overheating of the RCP Motor.

B. Overheating of the RCP Seals.

C. Overheating of the RCP Hydrostatic BRG.

D. Overheating of the RCP Lube Oil.

Answer:

B. Overheating of the RCP Seals

Notes:

B is correct as the 10 minute limit is primarily based on preventing overheating of the RCP Seals to prevent a LOCA if all the seals fail including the vapor seal.

A is incorrect as this is not the main safety reason for restoring cooling but plausible as this is a big concern from an operational and monetary standpoint and maintaining forced circulation after the trip.

C is incorrect as this is not a reason for restoring CCW cooling but plausible as this is a big concern from maintaining forced circulation after the trip. The Hydrostatic BRG is a Pump BRG to maintain pump alignment and is cooled by the RCS flow.

D is incorrect as this is not the main safety reason for restoring cooling but is a big concern from an operational and monetary standpoint.

This question matches the K&A because knowledge of the components cooled on the RCPs by CCW must be known along with the basis for the 10 minute limit for loss of CCW to the RCPs.

References:

AOP-2203.025 RCP Emergency TG Rev. 17 Step 3.C (Verified reference updated 11/15/16);
AOP-2203.025 RCP Emergency Rev. 18 Attachment D(Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Reworded the stem to discuss the primary reason for the time limit for operating RCPs during a loss of CCW event. Changed distractor C to Hydrostatic BRG which is cooled by RCS flow.

AOP STEP:

*3. **<u>IF</u> CCW flow <u>NOT</u> aligned to RCPs**
<u>THEN</u> perform the following:

BASIS:

The most likely causes of loss of CCW to the RCPs that could be corrected within 10 minutes are inadvertent closure of a containment isolation valve or loss of a CCW pump. Other potential causes such as air entrapment in the pump or a system rupture are not considered in this AOP due to the unlikely ability to correct the condition within 10 minutes.

A. The time that CCW flow was lost is recorded so that a determination can be made as to when the 10 minutes is up.

B. An operator is dispatched to the CCW Pump Room to rapidly respond if 2P33B Discharge Valve 2CCW-29 has to be throttled for pump start.

C. The RCP tech manual requires that after 10 minutes of operation with no CCW flow, the affected pumps must be secured to prevent seal damage. If CCW flow cannot be restored, then the reactor is tripped and all RCPs are secured. The reactor is manually tripped first if the Plant is in Mode 1 or 2 to prevent an automatic reactor trip when the first RCP is secured. The 10 minute limit is also important for RCP motor protection. It is possible that high bearing/stator temperature alarms will annunciate before 10 minutes have elapsed. If a high motor temperature alarm annunciates before the 10 minute limit, then the operator is expected to trip the reactor and trip the affected RCPs in accordance with the procedure. The requirement to secure all RCPs and isolate Controlled Bleedoff flow still applies to the remaining RCPs if CCW flow cannot be restored within the 10 minute time limit.

   Controlled Bleedoff flow is isolated if the RCPs must be secured due to the loss of CCW flow. Isolating CBO minimizes heat input to the seals. A caution is provided to warn the operator that failure to isolate CBO flow could lead to seal failure.

   If the reactor is tripped and all pumps secured due to low flow, then the operator meets the EXIT CONDITION of all RCPs secured and is transitioned to the EOP.

D. The CCW Containment Isolations are checked open and the contingency action provides instructions to open them if they are closed. A note is included to remind the operator that they may have to be overridden if a CIAS relay has actuated. The contingency action bypasses the remaining instructions in this step if a CCW pump is running because either this fixed the problem, or one or more valves can not be opened and there is no need to perform actions for loss of the pumps. If CCW MOVs are overridden on both trains, Tech Spec 3.0.3 should be entered until a dedicated operator can be stationed.

E. Greater than 13% level in the CCW surge tank is verified to ensure adequate NPSH is available for the CCW pumps.

ATTACHMENT D
RCP TRIP AND SHUTDOWN CRITERIA

- ANY of the following conditions will require a Reactor trip and affected RCPs stopped:
  - Loss of RCP CCW coincident with confirmed fire in the Unit 2 Turbine Bldg, Unit 2 Aux Bldg or Unit 2 Aux Extension Bldg requires RCPs be secured within five minutes.
  - Loss of RCP CCW flow for greater than 10 minutes.
  - RCP Vapor Seal pressure greater than 1500 psia.
  - THREE or more stages failed on ANY RCP seal.
  - RCP Motor Stator Winding Temperature alarm with rising trend.
  - RCP Upper or Lower Thrust Bearing Temperature greater than 225ºF.
  - Low level alarm on RCP Upper or Lower Oil Reservoir with Bearing temperature rising greater than 18°F/min or bearing temperature exceeding 195°F.
  - CCW Surge Tank level less than 13% following restoration of CCW flow to RCPs.
  - RCP Vibration high as determined by ANY of the following:
    - ♦ Motor vibes > 20 mils
    - ♦ Shaft vibes > 25 mils
    - ♦ A rapid rise in vibration with other parameters also indicating a problem
- ANY of the following conditions will require plant shutdown if in Mode 1 OR 2:
  - TWO stages failed on ANY RCP seal.
  - RCP Controlled Bleedoff temperature can NOT be restored to less than 180°F with ONE or more failed seals OR can NOT be restored to less than 200°F.
  - RCP Controlled Bleedoff flow greater than 3.0 gpm.
  - RCP Motor Stator Winding Temperature alarm with stable trend.
  - RCP Upper Thrust Bearing Temperature greater than 212ºF (Ops Management discretion).
  - RCP Lower Thrust Bearing Temperature greater than 200ºF (Ops Management discretion).

- ANY of the following conditions will require securing of affected RCP if in Mode 3, 4, OR 5:
  - TWO stages failed on ANY RCP seal.
  - RCP Controlled Bleedoff temperature can NOT be restored to less than 180°F with ONE or more failed seals OR can NOT be restored to less than 200°F.
  - RCP Controlled Bleedoff flow greater than 3.0 gpm.
  - RCP Motor Stator Winding Temperature alarm with stable trend.
  - RCP Upper Thrust Bearing Temperature greater than 212ºF (Ops Management discretion).
  - RCP Lower Thrust Bearing Temperature greater than 200ºF (Ops Management discretion).

Question 54

Bank:	2384	**Rev:**	1	**Rev Date:**	12/19/2016	**2017 TEST QID #:**	54	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	073000K503	**10CFR55:**	41.11	**Safety Function**	7				
Title:	Process Radiation Monitoring (PRM) System	**System Number**	073	**K/A**	K5.03				
Tier:	2	**Group:**	1	**RO Imp:**	2.9	**SRO Imp:**	3.4	**L. Plan:**	ASLP-RO-RXT04
OBJ	4.10								

Description: Knowledge of the operational implications of the following concepts as they apply to the PRM System: - Relationship between radiation intensity and exposure limits

Question:

Given the following:

* Unit 2 is at full power operation.
* The WCO is tasked with quantifying a 2P-89B HPSI pump seal leak in the 'B' ESF room.
* The WCO has signed onto 2017-2002 Operations Activities RWP Task 1 Radiation Areas.
* The EAD setpoint limits for RWP 2017-2002 are 4 mrem dose and 40 mrem/hr dose rate.
* The WCO has just entered the 'B' ESF room and the calculated stay time is 30 minutes.

* NOW the CVCS Process Radiation Monitor 2RE-4806 indicates rising activity in the RCS.
* Based on Chemistry input, OPS has started additional Charging Pumps to clean up the RCS.

Based on these conditions, the stay time for the WCO will be ___________ before the RWP allowed dose limit is reached due to the change in radiation dose rates caused by ____________.

A. shorter; RCS N-16 gammas spend less time in the Letdown decay chamber

B. longer; RCS N-16 gammas spend more time in the Letdown decay chamber

C. shorter; more RCS radioactivity buildup in the Letdown Filter raising 'B' ESF room dose

D. longer, RCS radioactivity will be removed faster by the in-service Letdown Demineralizer

Answer:

A. shorter; RCS N-16 gammas spend less time in the Letdown decay chamber

Notes:

A is correct. The Letdown line of CVCS has a decay chamber to allow N-16 short half life gammas time to decay before exiting the containment building. With letdown elevated, letdown spends less time in the decay chamber allowing more N-16 to reach the Aux building, which will raise general area dose rates.
The Letdown Line coming from Containment runs through the 'B' ESF room near the 2P-89B HPSI pump. Before the RCS is cleaned up which could take days, the radiation dose rates will rise near the WCO doing the seal leak quantification causing the stay time to be shorter.

B is incorrect as the higher Letdown flowrate will reduced N-16 decay time in the Letdown decay chamber which will raise dose rates so the stay time will be shorter but plausible if the applicant confuses more decay time with Letdown flowrates with reduced dose rates.

C is incorrect as Letdown flow will rise in the Auxiliary Building carrying the high RCS activity through the line that passes through the 'B' ESF room near the 'B' HPSI pump before the fluid passes through the filters and Ion Exchangers (These are located outside the ESF Room) which will remove the activity but plausible as the stay time will be shorter and the candidate may assume that the activity will clean up during his stay in the room based on higher flowrates .

D is incorrect as Letdown flow will rise in the Auxiliary Building carrying the high RCS activity through the line that passes through the 'B' ESF room near the 'B' HPSI pump before the fluid passes through the filters and Ion Exchanger that will remove the activity but plausible if the applicant does not know the flow path of the CVCS system and believes the fluid does go through the filters and Ion Exchangers prior to the 'B' ESF room.

This question matches the K&A because the applicant must have knowledge of how changing charging flow will affect Letdown flow and the associated radiation intensity rise will that will cause the RWP limits to be exceeded sooner based on high activity in the RCS as indicated on the CVCS process radiation monitor.

References:

OP 2203.020 High RCS Activity Rev. 12 Step 4 Note
OP 2203.020 High RCS Activity Rev. 12 Step 4 TG
STM 2-04 Rev. 31, Chemical And Volume Control System Section 2.1.1

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed the word "intensity" to "dose rates" in the stem. Removed the word "more" from distractor D prior to "RCS"

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

Xe-133 (Noble Gas Activity) sampling will be precluded due to the high levels of radiation exposure if use of 2607.001 Attachment 17 is required.

3. **IF projected RCS activity greater than or equal to 10 μCi/g I-131, THEN perform the following:**

   A. Contact Chemistry to perform the following:

      1) Sample RCS for Dose Equivalent I-131.

      2) Provide recommendation for RCS Letdown flow.

   B. Enter Action b. of TS 3.4.8 for Dose Equivalent Xe-133 NOT within limits.

CAUTION

Raised Letdown flow may cause Aux Building radiation levels to rise significantly.

*4. **IF chemist request raised Letdown flow, THEN establish requested flow, refer to 2104.002, Chemical and Volume Control.**

AOP STEP:

***4. <u>IF</u> chemist request raised Letdown flow, <u>THEN</u> establish requested flow, refer to 2104.002, Chemical and Volume Control.**

BASIS:

Raising letdown flow will raise the rate of removal of fission products. Chemistry will determine if the raised removal will offset the raised dose rates seen by personnel due to the raised flow rate. A recommendation will then be made to operations for Letdown flow adjustments during the clean-up process.

The method used to establish raised letdown flow will be to start additional charging pumps. The letdown flow control valves will respond to the rising pressurizer level in automatic and letdown flow will rise. A caution is provided to warn the operator that raising letdown flow will cause Auxiliary Building radiation levels to rise due to the reduced decay time of N-16 and the rise in RCS activity.

SOURCE DOCUMENTS:

1 - 2104.002, Chemical and Volume Control Operation.

2.0 Detailed System Description

2.1 Letdown

The following discussion describes the CVCS components, in the order that they are used, which perform the various functions of the letdown, charging, and chemical addition portions of the CVCS system (refer to page 66). (When referenced, diagrams of the controls located on control room panel 2C09 are located on pages 72 and 72).

Letdown taps off the suction line to reactor coolant pump 2P32A.

2.1.1 N-16 Decay Chamber

To minimize radiation levels inside the auxiliary building, an N-16 decay chamber is provided. This is an 8" enlargement in the 2" letdown line just prior to the regenerative heat exchanger. Its purpose is to slow the letdown fluid to allow sufficient time for the decay of N-16 gammas. These gamma's have a half-life of ~7.13 seconds. The following is the decay scheme for these gammas.

$$_{8}O^{16} + {}_{0}N^{1} \rightarrow \left({}_{8}O^{17}\right)^{*} \rightarrow {}_{1}P^{1} + {}_{7}N^{16}$$

$$\rightarrow {}_{8}O^{16} + \beta + 2\lambda\,(6.13\ and\ 7.12\ Mev)$$

2.1.2 Letdown Line Stop Valve 2CV-4820-2

The first component downstream of the RCS is letdown line stop valve (2CV-4820-2). This valve is located in the containment near the regenerative heat exchanger. 2CV-4820-2 is a motor operated valve controlled by a handswitch on control room panel 2C09. Valve open and closed position indication is also provided on panel 2C09.

2CV-4820-2 will automatically close and cannot be re-opened if either of the following conditions is present:

- SIAS #2 (Safety Injection Actuation Signal)
- regenerative heat exchanger letdown outlet temperature as sensed by 2TE-4820 is > 470°F.

The reason for having letdown isolate if the regenerative heat exchanger outlet temperature gets too high is to protect the heat exchanger from excessive thermal stress. However, running a charging pump with no letdown causes the charging header fluid temperature to decrease. This temperature decrease presents a thermal concern for the charging nozzles in the “B” and “C” RCS T_c loops.

The SIAS actuation is caused by low RCS pressure and is indicative of either a loss of coolant accident (LOCA) or an over-cooling event such as a main steam line rupture. In these events 2CV-4820-2 closes to provide possible leak isolation and to minimize the amount coolant leaving the RCS.

2CV-4820-2 is powered from 2B61-L3 via series breaker 2B61-K4. Upon loss of power it will remain in the position it was in at time of the power loss.

Question 55

Bank:	2385	**Rev:**	3	**Rev Date:**	12/21/2016	**2017 TEST QID #:**	55	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	062000A103	**10CFR55:**	41.5	**Safety Function**	6				
Title:	A.C. Electrical Distribution System	**System Number**	062	**K/A**	A1.03				
Tier:	2	**Group:**	1	**RO Imp:**	2.5	**SRO Imp:**	2.8	**L. Plan:**	A2LP-RO-ED480
OBJ	7c								

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the A.C. Distribution System controls including: - Effect on instrumentation and controls of switching power supplies

Question:

Given the following:

* Unit 2 is at full power.
* The Main Turbine trips.
* Electrical loads transfer to offsite as designed.
* Offsite Grid Voltage starts to degrade.

Without any operator action, which of the following combination of voltages would be the MAXIMUM voltages on Electrical 480 Volt Buses 2B5 and 2B6 that would cause BOTH Emergency Diesel Generators to START and LOAD their respective Safety Bus. (assume all of the listed voltages have dropped to only the values listed and have been at these values for greater than 15 seconds)

A. 440 VAC on 2B5; 441 VAC an 2B6

B. 435 VAC on 2B5; 438 VAC an 2B6

C. 426 VAC on 2B5; 428 VAC an 2B6

D. 419 VAC on 2B5; 422 VAC an 2B6

Answer:

C. 426 VAC on 2B5; 428 VAC an 2B6

Notes:

C is correct. The Red Train (2B5) millstone relays will actuate at 429.6 plus or minus 6.4 volts AC and is designed to protect the vital AC buses from low offsite voltage. The Millstone relays are designed to strip the vital buses from offsite and then the respective diesel will start on degraded voltage on the bus to re-energize the vital buses from their respective diesel. 429.6 plus 6.4 = 436 VAC therefore C will start both EDGs and load their respective buses and the voltages are higher than D.

A is incorrect as no EDG will be running and the no EDG would have its output breaker closed but plausible as the voltage on the Green train is degraded but not low enough to actuate the Red or Green Train millstone relay.

B is incorrect as only one EDG, the Red train 2DG1 will be running and tied to 2B5 but plausible as the voltage on the Green train is degraded but not low enough to actuate the Green Train millstone relay.

D is incorrect because the voltage combination is lower than C which is the Maximum voltage listed in the 4 choices that will cause both to start but plausible as these voltages will actuate the millstone relays.

This question matches the K&A as it requires the candidate to monitor changes in Vital 480 VAC bus voltage instrumentation and EDG/BRKR control switches and predict the correct Vital bus electrical status to prevent exceeding

any voltage operating design limits on vital bus loads.

References:

STM 2-32-3, 480 VAC Electrical Distribution, Rev. 21 Section 2.2.2

Historical Comments:

To be used on the 2017 NRC Exam

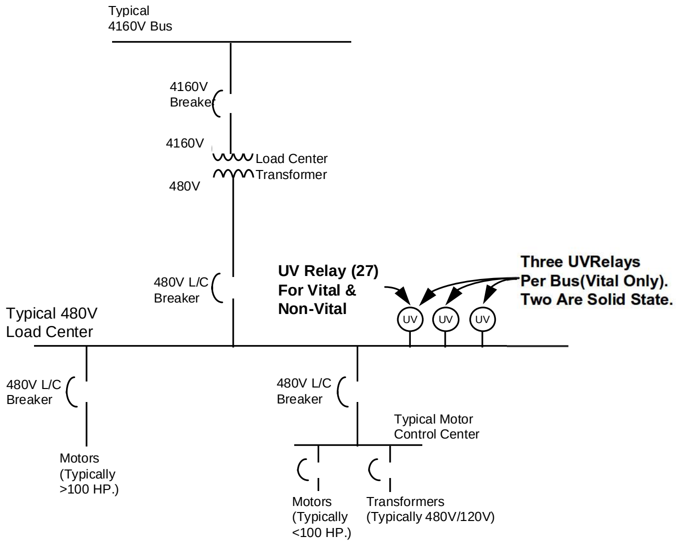
REV. 1 based on NRC Chief Examiner Feedback BNC. Formatted the question to a two by two and added a second part to the stem and made the distractors more plausible and more closely matched the K/A.

REV. 2 based on NRC Chief Examiner feedback BNC. Rejected the original DC Electrical Distribution Question K/A 063 A1.01 with a new randomly selected AC Electrical Distribution K/A 062 A1.03 and BANK Question 1655 previously used on the 2009 NRC Exam.

REV.3 based on NRC Chief Examiner feedback BNC. Generated a new question as suggested by the NRC Chief Examiner by giving the final conditions in the stem and asking for the voltages in the choices.

2.2.2 Bus Protection Relays

The 480V Load Center Buses are monitored for Over Current and Undervoltage. The Overcurrent relays will cause their respective supply breakers to trip. When an overcurrent relay trips drop flags are mechanically released to indicate which relay has tripped. They are for indication only.

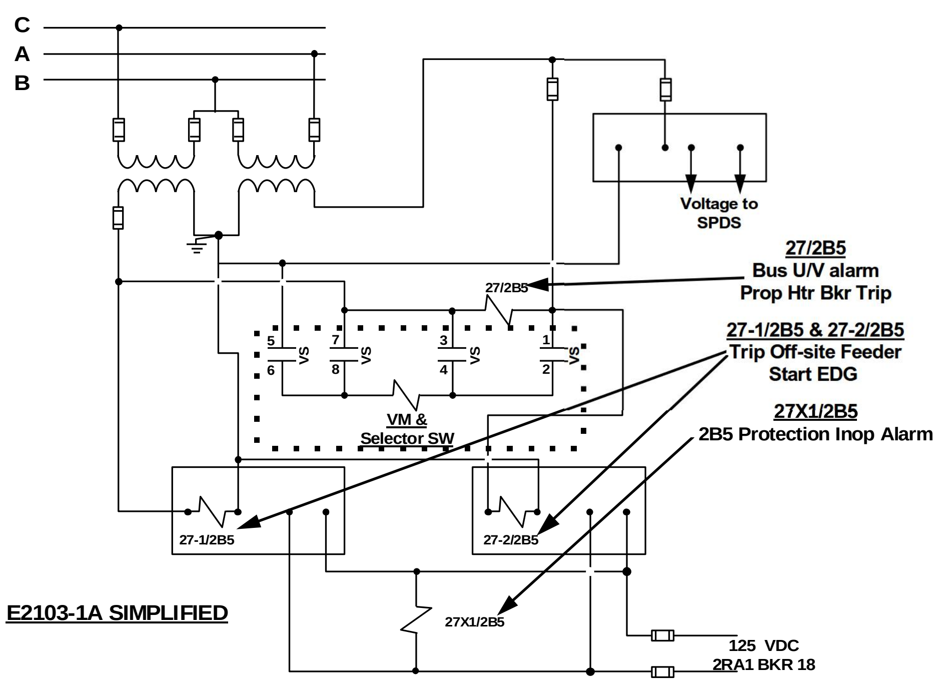


The Undervoltage Relay (27/B) provides for an undervoltage alarm.

2B5 and 2B6 are the vital Load Centers and have three undervoltage relays each (see figure to the left). The 27 relay provides the following.

- Bus Undervoltage alarm
- Trip the Pressurizer Proportional Heaters breaker for the respective vital bus

The other two UV relays on the vital buses are solid state relays (27-1/2B & 27-2/2B), and are powered from Vital DC buses 2RA1 and 2RA2 (i.e. Both relays on 2B5 are powered from 2RA1 and both relays on 2B6 are powered from 2RA2.) These are the Millstone monitoring relays and provide the following interlocks.



- If both relays sense voltage < 429V (429.6 +/-6.4 volts) for > 8 seconds, they will cause the supply breakers supplying 2A3 and 2A4 (2A309/2A409) to trip.
- If both relays sense voltage < 429V (429.6 +/-6.4 volts) for >8 seconds, they will cause the respective Emergency Diesel Generator to start.
- Relay actuation is blocked for 25 seconds when Reactor Coolant Pumps are started due to the possible effect on bus voltage when the pump motors initially draw large amounts of starting current.

If DC power is lost to the solid state UV relays they are inoperable and cannot provide any interlock functions. In other words they fail as is and will not cause any automatic interlocks due to a loss of power. DC Voltage Monitoring Relay (27X1/2B) will cause the 2B5/6 PROTECTION INOPERATIVE alarm to enunciate in the Control Room

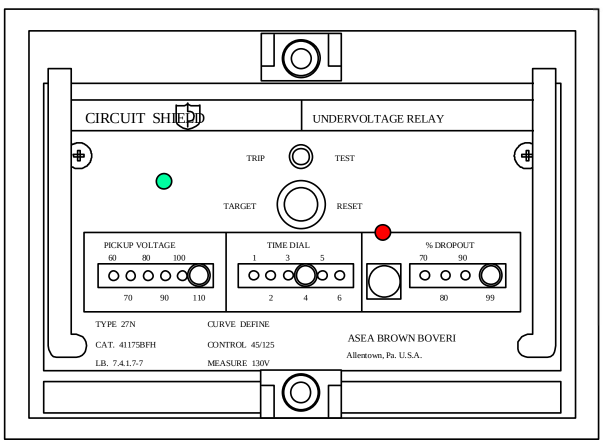
If the supply from the respective Potential Transformer is lost due to a blown fuse or etc., the respective Millstone Relay and the Bus Undervoltage Relay would actuate. However, with respect to the Millstone Relay, it would cause no actuations because both are required to actuate to initiate the divorce from off-site power described below.

The reason that these relays are located on the 480V Load Centers is that these buses would have the most severely degraded voltage during a Millstone event. This is because of the Load Center Transformer impedance losses. Every inductor between the source of power and the bus represents an impedance loss which drops voltage. Quite simply the 480V buses have more impedance losses between them and the source of power and therefore would be most severely effected.

The solid state relays actuate at < 429V (429.6 +/-6.4 volts) **and** have an 8 second time delay to actuate their associated contacts. The purpose of the 8 second time delay is to allow bus voltage time to recover during a temporary degradation. This time delay will also allow the 2A3/2A4 bus undervoltage relays to initiate the start of the EDG during a complete LOOP Loss Of Off-Site Power (LOOP). A Millstone event scenario would be as follows.

If bus voltage on the respective bus is degraded below the Millstone Relay setpoints for >8 seconds the off-site supply breaker would receive a trip signal and the EDG for the respective bus would automatically start and energize the 4160V bus. This automatically divorces the Class 1E Engineering Safeguards Bus from off-site power during degraded bus voltages to protect the following components.

- Motors
- Motor Control Center Control Power Components such as fuses and transformers.



Solid State Undervoltage Relay

Question 56

Bank:	2386	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	56	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	Modified NRC Exam Bank 1673		
Search	075000K101	**10CFR55:**	41.7	**Safety Function**	8				
Title:	Circulating Water System	**System Number**	075	**K/A**	K1.01				
Tier:	2	**Group:**	2	**RO Imp:**	2.5	**SRO Imp:**	2.5	**L. Plan:**	A2LP-RO-CWS
OBJ	2								

Description: Knowledge of the physical connections and/or cause-effect relationships between the Circulating Water System and the following systems: - SWS

Question:

Given the following:

* The plant is at 100% Power.
* The Cooling Tower Basin Level Control Valve 2CV-1460 is in Automatic.

If the Circulating Water cooling tower basin lowers below its current setpoint of 80%, then the Cooling Tower Basin Level Control Valve 2CV-1460 will __________ to provide __________ pressure on the Service Water return header.

A. open; less

B. close; more

C. open; more

D. close; less

Answer:

B. close; more

Notes:

B is correct: The position of 2CV 1460 determines the back pressure on the service water and auxiliary cooling water return headers, and thus the flow of service water and auxiliary cooling water to the circulating water cooling tower basin. When 2CV 1460 is open minimum makeup flow is directed to the cooling tower basin. When the squeeze valve is throttled closed, the back pressure on the service water return header rises. This causes water from the return header to divert to the cooling tower basin if service water to cooling tower MU isolation valve 2CV-1540 is open and it is normally open at 100% power.

A is incorrect but plausible as this would be the response to a level higher than setpoint.

C is incorrect as the squeeze valve will close but plausible as more service water is needed in the cooling tower basin due to low level.

D is incorrect because more water will flow to the cooling tower but plausible as the squeeze valve will close more.

This question matches the K&A as the candidate must understand the cause effect relationship between the Service Water squeeze valve and controller to the interconnect make up to the Circulating water cooling tower basin.

References:

STM_2-40-1_31-1 Circulating Water System Section 2.1.6 (Verified reference updated 11/15/16);
STM_2-42_36-1 SW and ACW SYS Drawing (Verified reference updated 11/15/16).

Historical Comments:

NRC Exam Bank 1673 was used on the 2009 NRC Exam
To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed the name of 2CV-1460 to "Cooling Tower Basin Level Control Valve" as this is the component description in the SW Operating procedure 2104.029. Also changed the last part of the stem to say "pressure on the Service Water Return Header"

Controller output demand is indicated on the demand output meter located directly above the auto/manual pushbuttons.

An operator adjusted setpoint dial is located in the center of the controller face with a deviation from setpoint indicator located directly above the setpoint dial. A deflection to the left of center on the deviation meter indicates that the process is below setpoint and a deflection to the right of center indicates that the process is above setpoint.

Conductivity controller 2CIC-1228 supplies a demand signal to electro-pneumatic positioner 2E/P-1228. 2E/P-1228 provides an air signal to open 2CV-1228 based on the demand signal provided by 2CIC-1228 to control conductivity by raising or lowering blowdown flow.

Circulating water conductivity indicator (2CIS-1228) provides a circulating water conductivity high/low alarm signal to 2K427-7, “COOLING TOWER 2M30 BASIN CONDUCT HI / LO” on panel 2C125. The high and low conductivity setpoints are variable based on chemistry input.

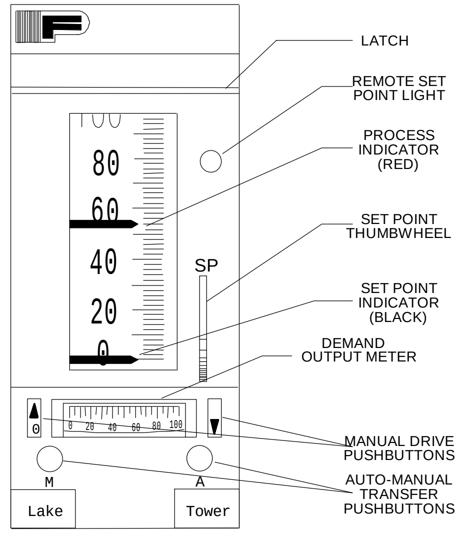
Circulating water blowdown flow is monitored using circulating water blowdown flow element (2FE-1279) to provide a signal to circulating water blowdown flow transmitter (2FT-1279). Circulating water blowdown flow is recorded on the dual pen circulating water flow and conductivity recorder (2FR-1279) on panel 2C125. Blowdown flow is also indicated on circulating water blowdown flow totalizer (2FI-1279) on panel 2C125. Total blowdown flow is indicated on 2FI-1279 by a six digit counter. The indicated total blowdown flow must be multiplied by 100 to get total actual blowdown flow.

2.1.6 Cooling Tower Makeup

The water losses from the cooling tower basin due to blowdown (variable), drift (40 gpm), and evaporation (11,716 gpm) are made up from the service water and auxiliary cooling water system through a 24-inch line to the cooling tower basin. The makeup line enters the tower just north of the 132 inch warm water return line. The normal operating level of the basin is 4.25 feet (80%).

Cooling tower basin level is monitored using an air bubbler type level detector to supply circulating water cooling tower basin level transmitter (2LT-1207) with a basin level signal. 2LT-1207 supplies a basin level signal to circulating water cooling tower basin level recorder (2LR-1207) on panel 2C14 and to circulating water cooling tower basin level controller (2LIC-1207A) on panel 2C14.

2LR-1207 provides a trend recorded level indication on rolled paper and an alarm signal to 2K12-C9, COOLING TOWER 2M30 BASIN LEVEL HIGH LOW on panel 2C14. The high alarm signal is set at ≥ 90% (348' 6"). The low alarm signal is set at ≤ 60% (347').



BASIN LEVEL CONTROLLER (2LIC-1207A)

Basin level controller 2LIC-1207A is an automatic bumpless - balanceless controller with a variable setpoint adjusted by the operator using the setpoint (SP) thumbwheel on the controller front panel. A 0 - 100% level meter containing a red process indicator and a black setpoint indicator is located in the center of the controller face beside the setpoint thumbwheel. A remote setpoint light is also located beside the level meter. Directly below the level meter are the manual drive pushbuttons located on either side of the demand output meter. The auto-manual transfer pushbuttons are located just below the manual drive pushbuttons.

The controller is designed to accept a remote setpoint during maintenance on the controller. A "local-remote" switch is accessed by lifting the latch on the controller face and pulling the controller out of the panel on its slides. The switch is located on the left hand side of the controller by the mode control adjustment potentiometers. When the "local-remote" toggle switch is selected to "R" for remote, the white remote setpoint light on the controller face is illuminated and the I&C technician can substitute a different setpoint controller to control basin level during maintenance on the installed controller. The "local-remote" toggle switch is normally operated in "L" or local position with the white remote setpoint light on the controller face OFF and the installed controller supplying the automatic setpoint signal as adjusted by the setpoint thumbwheel and indicated by the black setpoint indicator on the meter face.

At the bottom of the controller face are the "Auto-Manual" transfer pushbuttons. When the "Auto" transfer pushbutton is depressed, the controller will adjust basin level based on the setpoint established by the operator using the variable setpoint (SP) thumbwheel. When the "Manual" transfer pushbutton is depressed, basin level will be controlled using the yellow manual drive pushbuttons to raise or lower controller demand as indicated on the output meter.

The basin level controller provides a "demand" control signal to a "squeeze" valve 2CV-1460 mounted in service water/auxiliary cooling water return header to Lake Dardanelle. Service water squeeze valve 2CV-1460 is fully CLOSED at a basin level of 25% (345' 3") when level controller output is at 100%. It is full OPEN at a basin level of 100% (349' 0") when controller output demand is at 0%. The position of 2CV-1460 determines the back pressure on the service water and auxiliary cooling water return headers, and thus the flow of service water and auxiliary cooling water to the circulating

water cooling tower basin. When 2CV-1460 is open minimum makeup flow is directed to the cooling tower basin. When the squeeze valve is throttled closed, the back pressure on the service water return header rises. This causes water from the return header to divert to the cooling tower basin if service water valve 2CV-1540 is open. For more details regarding the operation of the squeeze valve refer to STM 2-42, Service Water System.

2.1.7 Cooling Tower Lightning Protection / Aviation Warning Lights

The cooling tower is equipped with a lightning protection subsystem and an aviation warning light subsystem.

The circulating water cooling tower lightning protection subsystem consists of eight (8) copper ground cables running vertically up the sides of the cooling tower. Connected to these ground cables is a cable encircling the top of the tower and a buried cable a minimum of 24 inches below the ground encircling the base of the tower. The handrails at the top of the tower, the access ladder, and the reinforcing bars inside the concrete of the tower structure all connect to the ground cables.

Four (4) additional ground cables are also routed up the cooling tower sides. The ground cables are connected only to the top and bottom grounding cables.

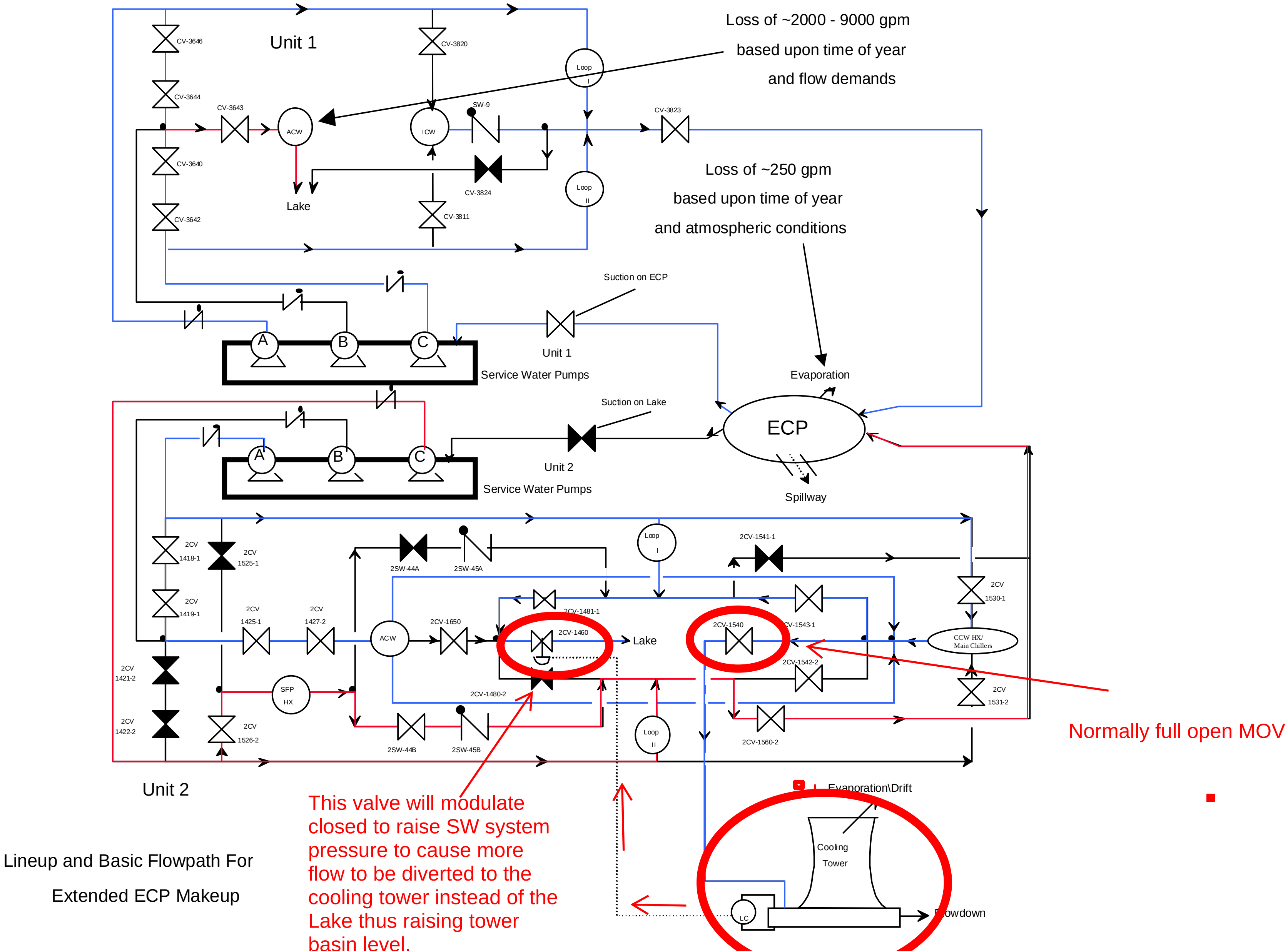
Each of the ground cables are terminated in a standard ground connection.

The aviation warning light system is controlled from aviation warning light control panels, 2C413A and 2C413B. These panels are located at the base of the cooling tower. A two-position control switch is located on the side of aviation warning light control panel 2C413A. It may be placed in either MANUAL (lights on) or AUTO (lights activated by a photo-cell). Normally this switch is in AUTO. Two main disconnect switches are also located on the same mounting rack for disconnecting power from the aviation warning lights. This rack is located at the base of the cooling tower between the blowdown valve and the stairway for access to the cooling tower door.

Four strings of four aviation warning light fixtures are located on the tower veil structure with one string on each of the three platforms and one around the top of the tower.

The second string from the bottom and the top string are *Crouse - Hinds Co.* type FCB-12 300 mm hazard flashing beacons with each at 620 or 700 watts. The 300 mm hazard beacons are watertight and non-ventilated. Each beacon has two sections. Each beacon contains four clear Fresnel lenses and two internal aviation red color screens to emit a 360° concentrated beam. Top and bottom sections each have a light receptacle and are hinged at their midpoint for re-lamping. The beacon light fixture is approximately 31 11/16 inches tall and has a base diameter of approximately 15 inches. Lamp wattage is either 620 or 700 watts.

The first and third strings are *Crouse - Hinds Co.* type EOL obstruction marker lights and are non-flashing lamps. Each lamp in these strings is 116 watts. Each obstruction marker light contains a



Lineup and Basic Flowpath For Extended ECP Makeup

PARENT QUESTION 1673 WAS USED TO MODIFY 2017 NRC EXAM QID#56 (Used on 2009 NRC Exam)

Data for 2009 NRC RO/SRO Exam

Bank:	1673	**Rev:**	1	**Rev Date:**	7/24/2009 3:48:08	**QID #:**	64	**Author:**	Coble
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	New		
Search	0750002128	**10CFR55:**	41.7	**Safety Function**	8				
System Title:	Circulating Water System	**System Number**	075	**K/A**	2.1.28				
Tier:	2	**Group:**	2	**RO Imp:**	4.1	**SRO Imp:**	4.1	**L. Plan:**	A2LP-RO-CWS
OBJ	2								

Description: Conduct of Operations - Knowledge of the purpose and function of major system components and controls.

Question:

Given the following:

* The Service Water Squeeze Valve, (2CV-1460) is in automatic

If the Circulating Water cooling tower basin rises above its current setpoint of 80%, then the Service Water Squeeze Valve, 2CV-1460, will __________ to provide __________ Service Water makeup flow to the Circulating Water cooling tower.

A. open; less

B. close; more

C. open; more

D. close; less

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐
2009	☑	☑

Audit Exam History

2009	☐

Answer:

A. open; less

Notes:

The position of 2CV 1460 determines the back pressure on the service water and auxiliary cooling water return headers, and thus the flow of service water and auxiliary cooling water to the circulating water cooling tower basin. When 2CV 1460 is open minimum makeup flow is directed to the cooling tower basin. When the squeeze valve is throttled closed, the back pressure on the service water return header rises. This causes water from the return header to divert to the cooling tower basin if service water valve 2CV-1540 is open.

References:

STM 2-40-1, CWS, Rev. 24, Section 2.1.6
STM 2-42, Service Water System, Rev. 28, Figure on page 61
Lesson Plan A2LP-RO-CWS, Rev. 8, Objective 2: Describe the operation of the CWS cooling tower including de-icing, bypassing, and makeup.

Historical Comments:

Question 57

Bank:	2387	**Rev:**	2	**Rev Date:**	12/19/2016	**2017 TEST QID #:**	57	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	035000K601	**10CFR55:**	41.7	**Safety Function**	4				
Title:	Steam Generator System (S/GS)	**System Number**	035	**K/A**	K6.01				
Tier:	2	**Group:**	2	**RO Imp:**	3.2	**SRO Imp:**	3.6	**L. Plan:**	A2LP-RO-CPC
OBJ	19								

Description: Knowledge of the effect of a loss or malfunction of the following will have on the S/GS: - MSIVs

Question:

Given the following:

* Plant is at 100% Power.
* The #1 MSIV fails closed.

Right after the MSIV closure and prior to the plant trip , the level in the ________ Steam Generator will lower, and the pressure in the ________ Steam Generator will lower.

A. 'A'; 'B'

B. 'B'; 'A'

C. 'A'; 'A'

D. 'B'; 'B'

Answer:

A. 'A'; 'B'

Notes:

A is correct as level will lower in "A" due to the shrink effect of the of the bubbles (higher density) in the downcomer region of the SG will be compressed due to the rapid rise in SG pressure in SG "A" due to the loss of steam flow. SG "B" pressure will lower based on a larger steam demand/flow coming from the SG to compensate for the loss of steam plow on 'A"SG. The SGs are cross connected on the downstream side.

B is incorrect as the level in SG "B" will rise due to the larger amount of steam flow and lower pressure due to swell and "A" SG Pressure will rise but plausible if the applicant does not understand the concept of shrink and swell on the Steam Generators..

C is incorrect because the pressure in SG "A" will rise due to the loss of steam flow but plausible as the level in SG "A" will lower.

D is incorrect as the level will rise in SG "B" will rise but plausible as the pressure in SG "B" will lower

This question matches the K&A as it requires knowledge of the effect of a MSIV malfunction will have on the SG.

References:

STM_215-1_19-1 Steam Generators and Main Steam Section 3.1.5.1/2 Shrink and Swell (Verified reference updated 11/15/16);

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Clarified distractors A and C as to why the level rising in the SGs is plausible. Added " prior to the plant trip" to the stem.

REV. 2 based on NRC Chief Examiner Feedback. Removed the 2nd half of the Rev. 1 question on all 4 distractors/answer. Revised the stem and question to a 2 X 2 format using the parameters on the "A" and "B" Steam Generators.

Bundle region. This influence in counteracted by the greater friction in the riser and Steam Separators. The net result is that while the mass flow rate of the steam into the Turbine rises, the flow rate of the saturated mixture in the Tube Bundle region remains relatively constant.

3.1.5.1 Swell Effect

Several factors work together to produce the Downcomer level rise effect known as “swell”. Swell occurs when steam flow from the Steam Generators rises rapidly. This phenomenon occurs because raising steam flow causes a step change reduction in steam pressure (P_{STM}) within the Steam Generator. This reduction in P_{STM} causes a very significant increase in the specific volume in the tube bundle area of the Steam Generator. This expansion forces more water back into the Downcomer area therefore raising the water level in this region.

Swell is also the result of a raising reactor power during a startup. The resulting heat increase in the Steam Generators cause the specific volume of fluid in the Steam Generator tube bundle region to increase. This expansion, or rise in specific volume, results the same effect on the downcomer level as described above only it is slower.

At low power, boiling begins high in the Tube Bundle region. As more steam is drawn off, pressure in the Steam Generator lowers causing two major effects:

1. The specific volume of the steam rises (density lowers), and
2. Saturation temperature is reduced which allows boiling to occur lower on the Tube Bundle.

Each of these factors tends to raise the void fraction in the Tube Bundle region. As the steam to liquid ratio rises, the density of the mixture lowers. Since the initial mass of water in the Steam Generator remains constant, the volume occupied by the steam-liquid mixture has risen.

When the void fraction in the Tube Bundle region rises (density lowers), the velocity must also rise to maintain or raise the flow rate. As velocity rises, the headless or $\Delta P_{\text{driving head}}$ required to cause the flow through the Steam Generator also rises.

$$\Delta \mathbf{P}_{\textbf{DRIVING HEAD}} \propto \mathbf{v}^2$$

Where v = velocity of the fluid

The resistance to flow is also greater by the additional turbulence (greater boiling) around the U-tubes. This resistance is felt as a rise in the $\Delta P_{\text{DRIVING HEAD}}$ between the Downcomer region and the Tube Bundle region which causes indicated water level in the Downcomer to rise.

Since the level instrumentation on the Steam Generators actually sense Downcomer region level, swell causes indicated level to rise. The magnitude of the swell is dependent upon the magnitude and rate of change in steam flow and also at the starting power level.

3.1.5.2 Shrink Effect

Steam Generator “shrink” occurs when steam flow from the Steam Generators is rapidly lowered. Lowering steam flow from the Steam Generators results in raising the pressure or P_{STM} inside the Steam Generators. This rise in pressure causes the steam vapor mixture in tube bundle area of the Steam Generator to be compressed, and significantly reduces the specific volume. This causes water in Downcomer area to flow into tube bundle area; therefore, indicated Downcomer water level lowers.

Also when steam flow is lowered and Steam Generator pressure rises accordingly the saturation temperature also goes up, reducing boiling in the Tube Bundle region. The void fraction and the mixture volume in the Tube Bundle region lower. With less turbulence from boiling and lower steam velocity, a lower water level is required in the Downcomer to produce the head necessary for flow.

"Shrink" can also be attributed to the insurge of colder water relative to the temperature of the water in the Steam Generator tube bundle region. This causes a change in the density of the total water volume reducing its specific volume. This results in an indicated Downcomer level lower.

3.1.6 Operation of the Steam Generators During Outages

For draining the Steam Generators, the Blowdown system and the Steam Generator Blowdown tank pumps 2P-139A and B are used. If the Steam Generators are drained to the Condenser Hotwell, then the excess Hotwell water can be drained to the Regenerative Waste Tank 2T-92A or B, or to the Turbine Building Sump. If the water meets fill grade chemistry specifications, the excess Hotwell inventory can be rejected to the Condensate Storage Tanks (CST), aligned for Auxiliary Feedwater (AFW) pump 2P-75 or Emergency Feedwater (EFW) pump 2P-7B suction. Another method for draining the Steam Generators to the Hotwell is allowing the Steam Generator Blowdown tank 2T-67 to overflow to the Condenser via the 2T-67 Vent valve 2SGS-5, to Condenser 2E-11A.

For filling the Steam Generators, the AFW pump, 2P-75 is the preferred method with EFW pump 2P-7B being an alternate method. Either pump is aligned to take suction from the Startup & Blowdown Demineralizer effluent or the CST, 2T-41A or B. Chemicals are added during Steam Generator fill to maintain water quality.

For Steam Generator recirculation, both normal and alternate modes are available. Pumps 2P-139 A or B take suction from the Steam Generators and inject the water back between the Main Feedwater Block Valves in the normal mode. In the alternate mode, water is returned to the Steam Generators through the Emergency Feedwater header. The normal mode is preferred to prevent Steam Generator water from leaking back to the Hotwell.

Partial drain is used when performing maintenance on the Steam Generators. Level is lowered and maintained below the point required for the maintenance. When possible, a vacuum or a nitrogen blanket is established to exclude oxygen from exposed Steam Generator

Question 58

Bank:	2388	**Rev:**	1	**Rev Date:**	12/19/2016	**2017 TEST QID #:**	58	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	4	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	086000K301	**10CFR55:**	41.10	**Safety Function**	8				
Title:	Fire Protection System (FPS)	**System Number**	086	**K/A**	K3.01				
Tier:	2	**Group:**	2	**RO Imp:**	2.7	**SRO Imp:**	3.2	**L. Plan:**	A2LP-RO-AFIRE
OBJ	9								

Description: Knowledge of the effect that a loss or malfunction of the Fire Protection System will have on the following: - Shutdown capability with redundant equipment

Question:

Given the following at full power.

* A severe fire has developed and completely engulfed the #1 EDG Room.
* The #1 EDG Fire Protection Valve 2UAV-3231 fails and CANNOT be opened manually.
* The Shift Manager has directed a Rapid Power Shutdown.

Based on the above conditions, boration should be commenced from the ______________________
and the procedure that is required to be entered for these conditions would be ____________________.

A. RWT directly to the Charging Pumps; OP-2203.014, Alternate Shutdown

B. BAMTs using the BAM Pumps; OP-2203.014, Alternate Shutdown

C. RWT directly to the Charging Pumps; OP-2203.049, Fires in Areas Affecting Safe Shutdown

D. BAMTs using the BAM Pumps; OP-2203.049, Fires in Areas Affecting Safe Shutdown

Answer:

C. RWT directly to the Charging pumps; OP-2203.049, Fires in Areas Affecting Safe Shutdown

Notes:

C is correct as the BAM pumps are located in the South #1 EDG room on the east end and not separated by any fire boundary and thus would be exposed to the fire and not reliable/available. These pumps are Green (B Train Powered). The gravity feed valves are located in the BAM Tank rooms which are above the South EDG room and in the same fire zone so they could potentially be damaged and not available. Therefore the RWT outlet directly to the suction of the charging pumps would be used in this case. The AOP 2203.049 Fires in Ares Affecting Safe Shutdown for Fire directs actions for Fire Area KK, South #1 EDG Room and should be implemented.

A is incorrect as the Alternate Shutdown procedure is used for a select set of safe shutdown zones such as the Control Room and Cable Spreading Room but not the zone in this question. However, this is plausible as the Alternate Shutdown procedure does have actions that deal with a fire in the selected areas and the correct boration source is listed.

B is incorrect because the BAM pumps will not be available but plausible as the Alternate Shutdown procedure is used for a select set of fire zone such as the Control Room and Cable Spreading Room but not the zone in this question however, this procedure does have actions that deal with a fire in the selected areas.

D is incorrect because the BAM pumps will not be available but plausible as the correct implementing procedure is listed.

This question matches the K&A as the malfunction of the 2UAV-2331 failing causes a severe fire that affect the normal capability of the BAM pumps to provide boration for a plant shutdown; therefore redundant equipment must be used to provide the boration source for shutdown capability.

References:

STM_2-04__31-1 CVCS Section 2.3.3 and 2.3.4 (Verified reference updated 11/15/16);
AOP 2203.034 Fire or Explosion Rev. 19 Section 2 Step 14; (Verified reference updated 11/15/16)
AOP 2203.049 Fires in Areas Affecting Safe Shutdown Rev 14 Entry Conditions (Verified reference updated 11/15/16);
AOP 2203.049 Fires in Areas Affecting Safe Shutdown Rev 14 Step 2.0 (Verified reference updated 11/15/16); AOP 2203.049 Fires in Areas Affecting Safe Shutdown Rev 14 Fire Area KK Actions Page 1 and 2 (Verified reference updated 11/15/16); AOP 2203.014 Alternate Shutdown Rev 32 Entry Conditions (Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam.

REV. 1 based on NRC Chief Examiner Feedback BNC. Deleted the "Pre-Fire Plan" Reference. Deleted any words associated with emergency boration in the 3rd bullet and the stem. Revised the first part of the stem to say "Based on the above conditions, boration should be commenced from the" as suggested. Changed 1st part of A and C to " RWT directly to the Charging pumps" and B and D to "BAMTs using the BAM pumps"

2.3.3 Gravity Feed Valves 2CV-4920-1 & 2CV-4921-1

These motor operated gate valves provide a boric acid gravity feed flowpath from BAM tanks to the suction of the charging pumps. This path would be utilized if the BAM pumps were incapable of functioning and emergency boration of the RCS was required. The valves are located in the boric acid makeup tank room on the 386' elevation of the auxiliary building adjacent to the associated tank at floor level.

Both valves are controlled by handswitches located on panel 2C09, 2HS-4920-1 and 2HS-4921-1, and automatically open on a SIAS #1 signal. The power supplies for 2CV-4920-1 and 2CV-4921-1 are 2B52-K4 and 2B52-F1 respectively.

2.3.4 Boric Acid Makeup Pumps 2P39A and 2P39B

Two boric acid makeup (BAM) pumps are provided to send a boric acid solution from the BAMT to various locations. These pumps receive an automatic start signal from SIAS. The discharge of the BAM pumps can be routed through the motor-operated emergency borate valve 2CV-4916-2 (also SIAS actuated) directly to the suction of the charging pumps. The pump discharge can also be routed through flow element 2FE-4926 and boric acid control valve 2CV-4926 to the following locations via the mixing tee:

- directly to the spent fuel pool,
- directly to the refueling water tank (RWT),
- through the CVCS blend control valve 2CV-4941-2 to the volume control tank (VCT), or
- through 2CV-4941-2 to the charging pumps suction line.

A third flow path for concentrated boric acid is directly to the charging pump suction header through a line that taps off just downstream of boric acid flow element 2FE-4926 (normally via 2CV-4926 and 2CV-4930).

The BAM pumps are located on elevation 369' in the #1 EDG room. The pumps have a design head of 100 psig and require a minimum net positive suction head (NPSH) of 2.6 psig. Annunciator 2K12-D4,"BA MU PUMP DISCH PRESS LO", is activated when either 2P-39A or 2P-39B discharge pressure falls to ≤ 85 psig 10 seconds after pump start.

The capacity of each pump is greater than the combined capacity of all three charging pumps. They are powered from vital 480 vac motor control centers. 2P39A receives power from 2B63-A1 while 2P39B is powered from 2B62-B6.

Although both BAM pumps are powered from the same electrical train, redundancy is provided for the system by gravity feed valves that are both powered from the other train (2B52-F1 and 2B52-K4).

Handswitches 2HS-4919-2 and 2HS-4910-2, located on control room panel 2C09, are used to control BAM pumps 2P39A and 2P39B respectively. BAM pump select switch, 2HS-4911-2 at panel 2C09, is used to select which pump will start automatically when

NOTE

A severe fire exists in a zone when ANY of the following conditions are present:

- Smoke in zone prevents assessing status of the fire
- Door/room is hot preventing access to the zone
- Fire in cable tray affecting several cables
- Fire in 4160v bus or 480v load center

*14. **IF** fire is or becomes severe on Unit 2
AND in ANY of the following safety-related areas,

LOCATION	FIRE AREA
B ESF ROOM and Gallery	AA
2P-7A EFW Pump Room	CC
2T-69A/B, 2T-13 RM, El. 335' Aux Bldg general area, Hot Machine Shop, 354'	DD
LSPPR, USPPR and Waste Gas Equipment Room	EE LOWER
LSEP Room	EE UPPER
2P-7B EFW Pump Room	FF
Electrical Equipment Room, UNPPR & LNPPR, MG Set Room	GG
Sample RM 354', VCT RM, Tank & Pump RM 354', General area 354' Resin Addition & 2B62, 2B63 372', Vacuum Degasifier Pump Room 372', Corridor North of Stairway 2001 372'	HH
North Switchgear Room (2A-3/ 2B5/ 2B54)	II
Corridor behind door 340, EL. 372'	JJ
#1 EDG & BAM Tank Room	KK
West D.C. Equipment Room (2D-01), West Battery Room (Red)	MM
Containment Building, South Side	NN SOUTH

LOCATION	FIRE AREA
Containment Building, North Side	NN NORTH
Unit 2 Intake Structure	OO
North EDG Room (#2 EDG)	QQ
East D.C. Equipment Room (2D-02), South Switchgear Room (2A4/ 2B6/ 2B64), East Battery Room (Green)	SS
Electrical Equipment Room (2B9/2B10)	TT
Turbine building including MSIV RM	B-2
2B 53 Room, LNEPR & UNEPR	B-3
CEDMCS Equipment Room	B-4
317 General Access, 2P-89C Room, Tendon Gallery Access, "A" ESF Room	B-6
Concrete Manhole East, NE of intake	2MH01E
Concrete Manhole East, NE of intake	2MH02E
Concrete Manhole East of Turbine building next to train bay	2MH03E

THEN:

- **PERFORM** 2203.049, Fires in Areas Affecting Safe Shutdown.
- **IF** Unit Shutdown required by 2203.049,
  THEN CONSIDER turning over this procedure to Unit 1.

FIRES IN AREAS AFFECTING SAFE SHUTDOWN

ENTRY CONDITIONS

As directed by Fire or Explosion (2203.034) for severe fire in any of the following areas:

LOCATION	FIRE AREA
B ESF ROOM and Gallery	AA
Emergency Feedwater Pump Room 2P-7A (Turbine Driven)	CC
2T-69A/B, 2T-13 RM, El. 335' Aux Bldg general area, Hot Machine Shop, 354'	DD
LSPPR, USPPR and Waste Gas Equipment Room	EE LOWER
Lower South Electrical Penetration Room	EE UPPER
Emergency Feedwater Pump Room 2P-7B (Motor Driven)	FF
Electrical Equipment Room, UNPPR & LNPPR	GG
Sample RM 354', Upper/ Lower VCT RM 372', Tank & Pump RM 354', General area 354' Resin Addition & 2B62, 2B63 372', Vacuum Degasifier Pump Room 372', Corridor North of Stairway 2001 372'	HH
North Switchgear Room (2A-3/ 2B5/ 2B54)	II
Corridor behind door 340, EL. 372'	JJ
South EDG Room (#1 EDG), Boric Acid Makeup Tank Room	KK
West D.C. Equipment Room (2D-01), West Battery Room (Red)	MM
Containment Building, South Side	NN SOUTH

LOCATION	FIRE AREA
Containment Building, North Side	NN NORTH
Unit 2 Intake Structure	OO
North EDG Room (#2 EDG)	QQ
East D.C. Equipment Room (2D-02), South Switchgear Room (2A4/ 2B6/ 2B64), East Battery Room (Green)	SS
Electrical Equipment Room (2B9/2B10)	TT
Turbine building including MSIV RM	B-2
North Electrical Equipment Room, LNEPR & UNEPR	B-3
CEDM Equipment Room	B-4
General Access Room, "C" HPSI Pump Room, Tendon Gallery Access, "A" HPSI, LPSI & Containment Spray Pump Room	B-6
Concrete Manhole East, NE of intake (Actions include a Manhole Map)	2MH01E
Concrete Manhole East, NE of intake (Actions include a Manhole Map)	2MH02E
Concrete Manhole East of Turbine building next to train bay (Actions include a Manhole Map)	2MH03E

EXIT CONDITIONS

As directed by the Shift Manager.

* **1.0 Monitor Spent Fuel Pool level and temperature (INPO IER L1-11-2 Rec 4).**

A. IF it is determined that a potential or actual threat exists that would affect Spent Fuel Pool level (inventory) or cooling capability, THEN perform 2203.002, Spent Fuel Pool Emergencies, in conjunction with this procedure.

2.0 IF severe fire confirmed in ANY of the fire areas listed in Table 1, THEN perform appropriate actions for that area.

TABLE 1

Fire Area	Fire Zone	Description	Page NO.	TAB
AA	2007-LL	B ESF ROOM and Gallery	5	1
CC	2024-JJ	Emergency Feedwater Pump Room 2P-7A (Turbine Driven)	10	2
DD	2019-JJ 2032-JJ 2040-JJ 2068-DD	Boric Acid Condensate Tank Room (2T-69A/B) Spent Resin Storage Tank Room (2T-13) Elevation 335' Aux Bldg general area Hot Machine Shop, 354'	15	3
EE LOWER	2055-JJ 2084-DD	Lower South Piping Penetration Room Upper South Piping Penetration Room and Waste Gas Equipment Room	22	4
EE – UPPER	2111-T	Lower South Electrical Penetration Room	30	5
FF	2025-JJ	Emergency Feedwater Pump Room (2P-7B, Motor Driven)	37	6
GG	2076-HH 2081-HH	Electrical Equipment Room (MG Set Room) Upper North and Lower North Piping Penetration Room	42	7
HH	2063-DD 2072-R 2073-DD 2096-M 2106-R 2107-N	Sample Room, 354' Upper/ Lower Volume Control Tank Room (EL. 372'), Tank & Pump Room (EL. 354') General area elevation 354' Resin Addition, 2B62 Motor Control Center (2B63), EL. 372' Vacuum Degasifier Pump Room, EL. 372' Corridor North of Stairway 2001, EL. 372'	49	8
II	2101-AA	North Switchgear Room (2A-3/ 2B5/ 2B54)	60	9
JJ	2109-U	Corridor behind door 340, EL. 372'	68	10
KK	2093-P 2115-I 2114-I	South Emergency Diesel Generator Room (#1 EDG) Boric Acid Makeup Tank Room EDG Air Intake Room	80	11
MM	2099-W 2103-V	West D.C. Equipment Room (2D-01) West Battery Room (Red)	86	12
NN – SOUTH	2032-K	Containment Building, South Side	96	13
NN – NORTH	2033-K	Containment Building, North Side	105	14
OO	Intake 2	Unit 2 Intake Structure	113	15

CRS Instruction

NOTE
Fire in this area will disable Red Train of EDG System. Charging Pumps can be tripped but may not start from the Control Room. Manual action is required to start a charging pump at the MCC.

1. **IF it is determined to take the Unit off line, THEN consider the following:**
   - Refer to Tech Specs
   - Normal Plant Shutdown (refer to applicable reactivity plan)
   - Rapid Plant Shutdown (refer to applicable reactivity plan)

2. **IF Reactor trip required as a direct result of the fire (i.e. not Tech Spec required shutdown), THEN perform the following:**

   A. Trip the reactor AND verify the following:

      1) All CEAs fully inserted.

      2) Reactor power lowering.

   B. Direct CBOT to perform “CBOT Required Actions”.

   C. Direct IAO to report to Control Room AND perform “IAO Required Actions”.

   D. Supply ATC OPERATOR with “Available Fire Regulatory Instrumentation” table.

   E. Supply Shift Manager with Fire Area KK General Guidance page.

   F. Perform the following procedures in conjunction with this procedure:

      - 2202.001, Standard Post Trip Actions.
      - Appropriate EOP/ AOP.

(Step 2 continued on next page)

CRS Instructions

2. (Continued)

CAUTIONS

- **Establish Charging within 30 minutes of reactor trip to satisfy RCS Inventory Control Safe Shutdown Function.**
- **Establish EFW flow within 70 minutes of reactor trip to satisfy Decay Heat Removal Safe Shutdown Function.**

*G. Establish Hot Standby conditions:

- Establish charging to control PZR level by directing **IAO** to start a charging pump per "**IAO Required Actions**".
- Feed SGs using EFW pumps.
- Verify decay heat removal via "A" SG Safeties or ADV.
- Refer to Natural Circulation (2203.013).
- Refer to Power Operation (2102.004).

H. Proceed to cold shutdown using the following:

- Plant Cooldown (2102.010)
- Applicable EOP/AOP

***3. Manual actions listed below may be required to be performed to transition the Unit to cold shutdown due to fire damage in this Area.**

EQUIPMENT	ACTION
2CV-1001, "SG A ADV"	Manually isolate and vent air operator. Verify valve OPEN.
2CV-1051, "SG B ADV"	Manually isolate and vent air operator. Verify valve OPEN.
2CV-5091, "SDC FLOW CONTROL"	Align 2CV-5091 for local control OR manually THROTTLE 2SI-5091-1 OR 2SI-5091-2.
2CV-5093, "SDC TEMP CONTROL"	Align 2CV-5093 for local control OR manually THROTTLE 2SI-5093-3.
2T-6A, 2T-6B, "BAM TANK"	If charging pumps are aligned to BAMT, suction alignment should be switched to RWT in approximately 2 hours OR if local monitoring of pump suction pressure decreases to 22 psig to ensure sufficient RCS makeup.
2P-36A, 2P-36B, 2P-36C, "Charging Pumps"	Operate locally at breaker.

ALTERNATE SHUTDOWN

PURPOSE

This procedure provides alternate shutdown capability to comply with 10 CFR 50.48, Fire Protection, and to mitigate the consequences of a significant fire in any one of the fire zones listed below.

ENTRY CONDITIONS

ANY of the following conditions exist:

1. Fire in the Control Room that renders the Control Room uninhabitable.

2. Fire in the Control Room that threatens damage to a major portion of vital controls.

NOTE

A severe fire exists when ANY of the following conditions are present:

- **Smoke in zone prevents assessing fire status.**
- **Door/room is hot preventing access to zone.**
- **Fire in cable tray affecting several cables.**
- **Fire in 4160v bus or 480v load center.**

3. Confirmed fire in ANY of the fire zones listed below that is, or becomes severe.

2098-L	Cable Spreading Room
2199-G	Unit 2 Control Room
2150-C	Core Protection Calculator Room (elevation 404)
2136-I	Radiation Protection (RP) Office (elevation 386)
2137-I	Upper South Electrical Penetration Room (includes Hot Tool Room and Hot I&C Shop)
2119-H	Control Room Printer Room
2098-C	Core Protection Calculator Room (elevation 372)

EXIT CONDITIONS

1. Plant control is re-established from the Control Room.

OR

2. An Alternate Shutdown Cooldown has been successfully completed.

1.0 OCCUPANCIES:

Normally unmanned.

2.0 FIRE BRIGADE ACCESS:

Access via Unit 2 controlled access to Stairway 2001 to door 262 to this zone (Primary)

Access from Turbine Building thru Corridor Zone 2109-U to the north Diesel Generator Room then thru doors room #251, 259 (Secondary).

3.0 PLANT PERSONNEL EGRESS:

Egress routes same as above.

4.0 LIGHTING/COMMUNICATIONS:

DC powered and battery pack emergency lighting. Gai-tronics call station G-18 is located on the south wall of the east end of Corridor Zone 2109-U. Call station G-27 is located on the north wall of the Tank Room (Zone 2115-I) which is north of Stairway 2001 at el. 386'.
Gai-tronics call station G-017 is located on east wall of zone near door #262

NOTE: The only assured means of communication is portable radio. If phones or gai-tronics are operable, they can be used as secondary means.

5.0 BARRIERS SEPARATING REDUNDANT SAFE SHUTDOWN EQUIPMENT:

North wall of this zone is a tech spec barrier between this diesel and the redundant diesel in zone 2094-Q.

6.0 HAZARDS:

		MATERIAL	CLASS
6.1	Fire:	Cable Insulation	A/C
		Fuel Oil	B
		Lube Oil	B
6.2	Radiation:	None	

6.3 Hazardous Substances:

Fuel oil (in Day Tank, 2T30A surrounded by block wall on north side of this zone)

- Combustible - cool tank if exposed
- Stable

6.4 Physical Hazards:

- Large amount of piping and equipment in this zone could lead to personnel disorientation.

6.5 Electrical Hazards:

Electrical Hazards	Electrical Equipment	Circuit Breakers
Boric Acid Pump Area Unit Cooler	2VUC24A	52-61C4 (2B61)
Boric Acid Pump Area Unit Cooler	2VUC24B	52-61C3 (2B61)
Boric Acid Pump	2P39A	52-63A1 (2B63)
Boric Acid Pump	2P39B	52-62B6 (2B62)
EDG Starting Air Compressor	2C4A	52-52D1 (2B52)
EDG Starting Air Compressor	2C4B	52-52D2 (2B52)
2K4A Standby Cooling Recirc. Pump	2P167A	52-52B1 (2B52)
2K4A Prelube Pump	2P170A	52-52B1 (2B52)
2K4A Exiter Cabinets	2E11 2E12 2E13 2E14 2E15	
EDG Room Inlet Air Dampers	2TCDM8637-1/8689-1	52-52B5 (2B52)

6.6 Compressed Gases

A) Cylinders (Self-Contained): None
B) Non-Self-Contained: 2 large air storage tanks one on north wall and one n south wall, maximum psi-250.

7.0 FIXED FIRE SYSTEMS:

Suppression:

Preaction, closed-head sprinkler system actuated by smoke and flame detectors. Oil tank spill retainer. Door #251 is watertight. Isolation valves 2FS-63 and 2FS-80 are located in 2149-B stairway area.

Detection: Smoke and flame detectors.

8.0 MANUAL SUPPRESSION:

Portable Extinguishers:

CO2-432 on south wall of access corridor 2109-U near Cable Spreading Room.
CO2-435 and WM-011 on south wall of access corridor 2109-U at west end.
CO2-433 in northwest corner of electrical equipment room (Zone 2108-S) thru door 267 off corridor zone 2109-U.
ABC-087 on east wall of Stairway 2001.
WM-013 in electrical penetration room (Zone 2111-T) thru door 278 off of corridor Zone 2109-U.
ABC-319 on east wall of adjacent North Diesel Generator Room Zone 2094-Q).

In Zone:

- ABC-320 on east wall.

Hose Stations:

- 2HR-29 on east wall of Aux. Bldg. stairwall # 2001 @ elev. 372.

Question 59

Bank:	2389	**Rev:**	3	**Rev Date:**	1/12/2017	**2017 TEST QID #:**	59	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	F	**Source:**	NRC Exam Bank 1509		
Search	029000A102	**10CFR55:**	41.7	**Safety Function**	8				
Title:	Containment Purge System (CPS)	**System Number**	029	**K/A**	A1.02				
Tier:	2	**Group:**	2	**RO Imp:**	3.4	**SRO Imp:**	3.4	**L. Plan:**	A2LP-RO-CVENT
OBJ	13								

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Containment Purge System controls including: - Radiation levels

Question:

Given the following plant conditions:

* Plant is in Mode 5 making preparations to refuel the Reactor.
* RCS is in lowered inventory preparing to install SG nozzle dams.
* Containment Purge System is in service.
* When the 1st set of SG Manways are removed, the Control Room receives Annunciator 2K11-D10 " Process Gas Radiation HI/LO".
* On 2C-25, the Gas Monitor for the Containment Purge, 2RITS-8233, is reading above setpoint.

Which of the following valves will automatically actuate to secure Containment Purge?

A. All three (3) Containment Purge Exhaust isolation valves will go closed ONLY.

B. All six (6) Containment Purge Supply AND Exhaust Isolation valves will go closed.

C. ONLY the Inside-Inside Containment Purge Supply AND Exhaust isolations go closed.

D. ONLY the Outside-Outside Containment Purge Supply AND Exhaust Isolations go closed.

Answer:

D. ONLY the Outside-Outside Containment Purge supply and exhaust Isolations go closed.

Notes:

D is correct: The only valves associated with the Containment Purge System that get a closure signal on a high process radiation alarm is the Outside-Outside supply and exhaust valves. These valves are considered containment isolations and verified closed from the ESF control panels 2C-16 and 17. The valves are labeled OUTSIDE-OUTSIDE on the control panels and only those valve have labels on the control panel showing closure on a High Rad Signal.

A is incorrect as only the valves associated with the Containment Purge System that get a closure signal on a high process radiation alarm is the Outside-Outside supply and exhaust valves but plausible as the exhaust isolation valves closing will prevent a release from containment and will automatically secure the supply fan.

B is incorrect as only the valves associated with the Containment Purge System that get a closure signal on a high process radiation alarm is the Outside-Outside supply and exhaust valves but plausible as all purge isolation valves get an auto closure on SIAS or CIAS.

C is incorrect as only the valves associated with the Containment Purge System that get a closure signal on a high process radiation alarm is the Outside-Outside supply and exhaust valves but plausible as the Inside-Inside purge isolation valves get an auto closure on SIAS or CIAS.

This question matches the K/A must have the knowledge of the containment purge valve interlocks and actuation to have

the ability to monitor and predict the proper response of the Containment Purge System controls during a high radiation signal.

References:

2203012K ANNUNCIATOR 2K11 CORRECTIVE ACTION REV. 46 Window D-10 PROCESS GAS RADIATION HI-LO(Verified reference updated 11/15/16); STM_2-09_16 Containment Cooling and Purge System Drawing (Verified reference updated 11/15/16); STM_2-09_16 Containment Cooling and Purge System Section 7.3 (Verified reference updated 11/15/16); STM_2-09_16 Containment Cooling and Purge System Section 7.5(Verified reference updated 11/15/16); STM_2-09_16 Containment Cooling and Purge System Table 1 (Verified reference updated 11/15/16).

Historical Comments:

NRC Exam Bank 1509 was used on the 2008 NRC Exam
To be used on the 2017 NRC Exam but altered the order of the distractors and answer.

REV. 1 based on NRC Chief Examiner Feedback BNC. Corrected the reference to show QID used on the 2008 NRC exam. Added all "six (6)" to distractor B. Updated the plausibility wording of Distractor A.

REV. 2 based on NRC Chief Examiner Feedback BNC. Added the word "only" to the end of distractor A.

Rev. 3 based on post submittal validation comments: in the 2nd bullet, changed the word "reduced' to lowered.

ANNUNCIATOR 2K11

D-10

PROCESS GAS RADIATION HI/LO

NOTE

- This alarm will reflash.
- The various monitor setpoints are located in Radiation Monitoring and Evacuation Alarm System (2105.016).

1.0 CAUSES

1.1 ANY of the following Gas Monitors in alarm (High or Low):

- Fuel Handling Area Disch 2VEF-14A/B (2RITS-8540)
- Containment Purge Disch 2VEF-15 (2RITS-8233)
- Waste Gas Sys Disch 2T-18A/B/C (2RITS-2429)
- Control Room Supply Air 2VSF-8A/B (2RITS-8750-1A or 2RITS-8750-1B)
- Penetration Room Exh Disch 2VEF-38A (2RITS-8845-1)
- Penetration Room Exh Disch 2VEF-38B (2RITS-8846-2)
- Radwaste Area Disch 2VEF-8A/B (2RITS-8542)

1.2 Loss of power to any of above monitors

1.3 Condenser Offgas Rad Monitor (2RITS-0645) failed low

2.0 ACTION REQUIRED

2.1 Check the following on 2C25 to determine affected monitor:

- Fuel Handling Area Disch 2VEF-14A/B (2RITS-8540)
- Containment Purge Disch 2VEF-15 (2RITS-8233)
- Waste Gas Sys Disch 2T-18A/B/C (2RITS-2429)
- Control Room Supply Air 2VSF-8A/B (2RITS-8750-1A or 2RITS-8750-1B)
- Penetration Room Exh Disch 2VEF-38A (2RITS-8845-1)
- Penetration Room Exh Disch 2VEF-38B (2RITS-8846-2)
- Radwaste Area Disch 2VEF-8A/B (2RITS-8542)

2.2 IF RDACS EAL REPORT GENERATED alarm comes in,
THEN contact Chemistry to run an Emergency Class Report.

2.3 IF release in progress
AND associated ventilation flowpath in HI alarm,
THEN notify chemistry to obtain grab sample and analyze to aid in determining emergency action level.

(D-10 Continued on next page)

ANNUNCIATOR 2K11

D-10

PROCESS GAS RADIATION HI/LO
(Continued)

2.4 IF Radwaste Area (2RITS-8542)
OR Fuel Handling Area monitor (2RITS-8540) in HI Alarm,
THEN notify Radiation Protection to sample to determine source.

2.5 IF Control Room Supply Air 2VSF-8A/B monitor (2RITS-8750-1A or 2RITS-8750-1B) in High Alarm,
THEN perform the following:

2.5.1 IF Control Room already on recirc IAW Control Room Emergency Air Conditioning and Ventilation (2104.007)
AND alarm expected,
THEN no further action required.

2.5.2 Verify Control Room isolated using Control Room Emergency Air Conditioning and Ventilation (2104.007).

2.6 IF Penetration Room Exh Disch 2VEF-38A monitor (2RITS-8845-1) or 2VEF-38B (2RITS-8846-2) in High Alarm
AND CIAS NOT actuated,
THEN secure 2VEF-38A and 2VEF-38B.

2.7 IF alarm on Waste Gas Sys Disch 2T-18A/B/C monitor (2RITS-2429),
THEN perform the following:

2.7.1 Verify WG Decay Tanks to Vent Plenum 2CV-2428 (2HS-2428) closed.

2.7.2 Check WG Decay Tanks to Vent Plenum flow approximately 0 scfm (2FR-2431).

2.7.3 Check 2RE-2429 for proper setting and operation. Refer to Gaseous Radwaste System (2104.022).

2.7.4 IF alarm due to improper setting or spike,
THEN attempt to re-establish release.

2.7.5 IF alarm due to rising trend,
THEN tag out affected tank and submit new release permit.

(D-10 Continued on next page)

ANNUNCIATOR 2K11

D-10

PROCESS GAS RADIATION HI/LO
(Continued)

2.8 IF CNTMT Purge Disch 2VEF-15 monitor (2RITS-8233) in High Alarm, THEN perform the following:

2.8.1 Verify CNTMT purge/ventilation secured:

- 2VEF-15 stack flow (2FR-8315) or RDACS (SPING 5) approximately 0 scfm
- At least ONE CNTMT Purge Supply Isolation valve closed:
  - 2CV-8283-1 (2HS-8283-1)
  - 2CV-8289-1 (2HS-8289-1)
  - 2CV-8284-2 (2HS-8284-2)
- At least ONE CNTMT Purge Exhaust Isolation valve closed:
  - 2CV-8291-1 (2HS-8291-1)
  - 2CV-8285-1 (2HS-8285-1)
  - 2CV-8286-2 (2HS-8286-2)

2.8.2 IF Equipment Hatch open,
THEN verify EITHER of the following:

A. Containment ventilating to Aux Building:

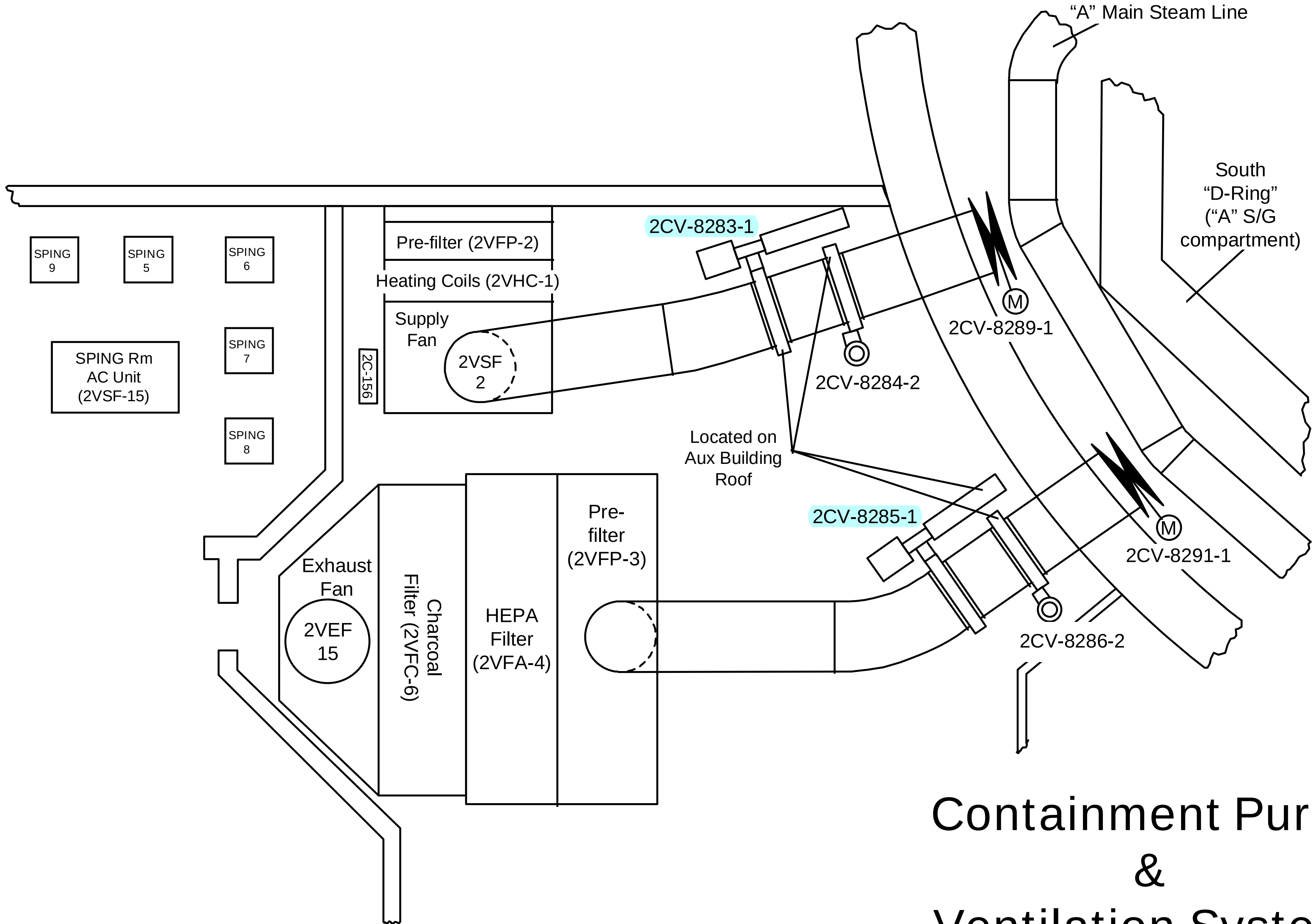
- Personnel Air Lock Doors open.
- Aux Building Exhaust ventilation in service [refer to Ventilation System Operation (2104.035)].
- Radwaste personnel stationed at Equipment Hatch monitoring for air flow into Containment.

B. Equipment Hatch closed.

2.8.3 Refer to Containment Atmosphere Control (2104.033).

2.8.4 IF Outage Control Center (OCC) manned,
THEN notify OCC that CNTMT Purge Exhaust fan (2VEF-15) is secured.

(D-10 Continued on next page)



Containment Purge
&
Ventilation System

7.3 Purge Air Supply Isolation Valves

This section provides a discussion on the construction and operation of the containment isolation valves associated with the containment purge supply fan.

7.3.1 Outside-Outside Containment Isolation Valve, 2CV-8283-1

The outside-outside containment isolation valve 2CV-8283-1 is an air cylinder operated, 54-inch butterfly valve. It is called outside-outside because, of the two outside containment isolation valves, it is the farthest away from the containment.

This valve is operated from the control room on panel 2C-17 with a key operated, two position (OPEN-CLOSE), spring return to center handswitch with the key removable ONLY from the center position.

When the handswitch is taken to OPEN solenoid valve 2SV-8283-1 is energized and aligns air to the air cylinder to cause the valve to open provided no automatic closure signals are present.

When the handswitch is taken to CLOSE the solenoid valve is de-energized and vents air from the air cylinder allowing spring force to close the valve.

This valve automatically closes under any one of the following conditions:

- Containment isolation actuation signal (CIAS)
- Safety injection actuation signal (SIAS)
- High effluent radiation level as sensed by process radiation monitor 2RE-8233.
- Loss of instrument air to the air cylinder
- Loss of DC power to the solenoid valve 2SV-8283-1.

Solenoid valve 2SV-8283-1 is powered from the 125 VDC in panel 2C-17 that comes from 2D23 breaker #01.

This valve is equipped with a “seal-in” feature. When the handswitch is taken to OPEN and released, the solenoid valve stays energized which maintains air pressure to the cylinder until a close signal de-energizes the solenoid.

7.3.2 Inside-Outside Containment Isolation Valve 2CV-8284-2

The inside-outside containment isolation valve 2CV-8284-2 is an air cylinder operated, 54-inch butterfly valve. It is called inside-outside because, of the two outside containment isolation valves, it is the closest to the containment.

This valve is operated from the control room on panel 2C-16 with a key operated, two position (OPEN-CLOSE), spring return to center handswitch with the key removable ONLY from the center position.

When the handswitch is taken to OPEN, solenoid valve 2SV-8284-2 is energized and lines up air to the air cylinder to cause the valve to open provided no automatic closure signals are present.

When the handswitch is taken to CLOSE, the solenoid valve is de-energized and vents air from the air cylinder allowing spring force to close the valve.

This valve automatically closes under any one of the following conditions:

- Containment isolation actuation signal (CIAS)
- Safety injection actuation signal (SIAS)
- Loss of instrument air to the air cylinder
- Loss of DC power to the solenoid valve 2SV-8284-2.

Solenoid valve 2SV-8284-2 is powered from the 125 VDC in panel 2C-16 that comes from 2D24 breaker #01.

This valve is also equipped with a "seal-in" feature. When the handswitch is taken to OPEN and released, the solenoid valve stays energized which maintains air pressure to the cylinder until a close signal de-energizes the solenoid.

7.3.3 Inside Containment Isolation Valve, 2CV-8289-1

Inside containment isolation valve 2CV-8289-1 is a motor operated 54-inch butterfly valve. It is powered from vital 480v MCC 2B51 breaker K3.

This valve is operated from the control room on panel 2C-17 with a key operated, two position (OPEN-CLOSE), spring return to center handswitch with the key removable ONLY from the center position. When the handswitch is taken to OPEN, the valve opens provided no automatic closure signals are present. When it is taken to CLOSE, the valve closes.

The valve automatically closes upon receipt of either of the following:

- Containment isolation actuation signal (CIAS)
- Safety injection actuation signal (SIAS)

This valve has a "seal-in" feature such that when the handswitch is taken to either OPEN or CLOSE and then released, the valve continues to travel to the appropriate position until a torque switch opens to stop valve travel.

The motor for this valve is equipped with a thermal overload bypass that bypasses the thermal overload trip whenever an SIAS or a CIAS is present. This causes the valve to continue to travel in the closed direction even if a condition exists that would normally stop valve travel by opening the thermal overloads.

Since this valve is in containment and its power supply is outside of containment, it is required by technical specifications to have backup overcurrent protection device. That device is provided by vital 480V MCC 2B51 breaker L5.

This valve does not have override capability under accident conditions.

7.4 Containment Purge Exhaust Unit

The containment purge exhaust unit consists of the following components:

- Inlet damper 2HCD-8257
- Pre-filter 2VFP-3
- HEPA filter 2VFA-4
- Charcoal filter 2VFC-6
- Vaneaxial fan 2VEF-15

2SV-8256 is powered from the 480/120 volt control power transformer in non-vital 480V MCC 2B85 breaker C7. On a loss of power to the solenoid valve, the damper fails closed. On a loss of instrument air to the solenoid valve, the damper fails "as-is".

7.5 Purge Air Exhaust Isolation Valves

This section provides a discussion on the construction and operation of the containment isolation valves associated with the containment purge exhaust fan.

7.5.1 Outside-Outside Containment Isolation Valve, 2CV-8285-1

The outside-outside containment isolation valve 2CV-8285-1 is an air cylinder operated 54-inch butterfly valve. It is called outside-outside because it is the farthest away from the containment of the two outside containment isolation valves.

2CV-8285-1 is operated from the control room on panel 2C-17 with a key operated, two position (OPEN-CLOSE), spring return to center handswitch with the key removable ONLY from the center position.

When the handswitch is taken to OPEN, solenoid valve 2SV-8285-1 is energized and aligns air to the air cylinder to cause the valve to open provided no automatic closure signals are present.

When the handswitch is taken to CLOSE, the solenoid valve is de-energized and vents air from the air cylinder allowing spring force to close the valve.

This valve automatically closes under any one of the following conditions:

- Containment isolation actuation signal (CIAS)
- Safety injection actuation signal (SIAS)
- High effluent radiation level as sensed by process radiation monitor 2RE-8233
- Loss of instrument air to the air cylinder
- Loss of DC power to the solenoid valve 2SV-8283-1.

Solenoid valve 2SV-8285-1 is powered from the 125 VDC in panel 2C-17 that comes from 2D23 breaker #01.

This valve is equipped with a "seal-in" feature. When the handswitch is taken to OPEN and released, the solenoid valve stays energized which maintains air pressure to the cylinder until a close signal de-energizes the solenoid.

7.5.2 Inside-Outside Containment Isolation Valve 2CV-8286-2

The inside-outside containment isolation valve 2CV-8286-2 is an air cylinder operated 54-inch butterfly valve. It is called inside-outside because it is the closest to the containment of the two outside containment isolation valves.

This valve is operated from the control room on panel 2C-16 with a key operated, two position (OPEN-CLOSE), spring return to center handswitch with the key removable ONLY from the center position.

When the handswitch is taken to OPEN, solenoid valve 2SV-8286-2 is energized and aligns air to the air cylinder to cause the valve to open provided no automatic closure signals are present.

When the handswitch is taken to CLOSE, the solenoid valve is de-energized and vents air from the air cylinder allowing spring force to close the valve.

This valve automatically closes under any one of the following conditions:

- Containment isolation actuation signal (CIAS)
- Safety injection actuation signal (SIAS)
- Loss of instrument air to the air cylinder
- Loss of DC power to the solenoid valve 2SV-8286-2.

Solenoid valve 2SV-8286-2 is powered from the 125 VDC in panel 2C-16 that comes from 2D24 breaker #01.

This valve is also equipped with a “seal-in” feature. When the handswitch is taken to OPEN and released, the solenoid valve stays energized which maintains air pressure to the cylinder until a close signal de-energizes the solenoid.

7.5.3 Inside Containment Isolation Valve, 2CV-8291-1

Inside containment isolation valve 2CV-8291-1 is a motor operated 54-inch butterfly valve. It is powered from vital 480V MCC 2B51 breaker K4.

This valve is operated from the control room on panel 2C-17 with a key operated, two position (OPEN-CLOSE), spring return to center handswitch with the key removable ONLY from the center position.

When the handswitch is taken to OPEN, the valve opens provided no automatic closure signals are present. When it is taken to CLOSE the valve closes.

The valve automatically closes upon receipt of either of the following:

- Containment isolation actuation signal (CIAS)
- Safety injection actuation signal (SIAS)

This valve **DOES NOT** have a “seal-in” feature. The only time the valve travels is when the handswitch is held in the appropriate position. Because of the absence of the “seal-in”, this valve has the capability of being throttled.

The motor for this valve is equipped with a thermal overload bypass that bypasses the thermal overload trip whenever an SIAS or a CIAS is present. This causes the valve to continue to travel in the closed direction even if a condition exists that would normally stop valve travel by opening the thermal overloads.

Since this valve is in containment and its power supply is outside of the containment, it is required by technical specifications to have back up overcurrent protection device. This overcurrent protection is provided by vital 480V MCC 2B51 breaker L6.

This valve does not have override capability under accident conditions.

Table 1 - Containment Ventilation System Alarms				
Description	Sensor Id	Setpoint	Annunciator Window Location	Remarks
CTMT CLG COILS C/D SW FLOW LO	2FIS-1514-2	< 1450 gpm with CCAS or MSIS present	2K05-H7	60 second time delay
CTMT BLDG CLG FANS C/D TROUBLE	2FS-8220-2A 2FS-8220-2B	0.25 inches of water	2K05-J7 (multiple inputs - no reflash)	20 second time delay on fan start
CTMT CLG COILS A/B SW FLOW LO	2FIS-1521-1	< 1450 gpm with CCAS or MSIS present	2K06-H7	60 second time delay
CTMT BLDG CLG FANS A/B TROUBLE	2FS-8207-1A 2FS-8207-1B	0.25 inches of water	2K06-J7 (multiple inputs - no reflash)	20 second time delay on fan start
CTMT RECIRC FANS A/C TROUBLE	2FS-8320-1A 2FS-8320-1B	0.2 inches of water	2K08-E4 (multiple inputs - no reflash)	30 second time delay on fan start
RX CAVITY FAN A TROUBLE	2FS-8246-1	0.6 inches of water	2K08-E5	30 second time delay on fan start
CTMT RECIRC FANS B/D TROUBLE	2FS-8321-2A 2FS-8321-2B	0.2 inches of water	2K09-E4 (multiple inputs - no reflash)	30 second time delay on fan start
RX CAVITY FAN B TROUBLE	2FS-8246-2	0.6 inches of water	2K09-E5	30 second time delay on fan start
PROC GAS RADIATION HI/LO	2RE-8233	Hi - ≤ 2 x background OR as required on release permit Lo - 10 cpm	2K11-D10	Background is recorded on the radiation monitor setpoint data log. High alarm isolates containment purge supply and exhaust dampers 2CV-8283-1 and 2CV-8285-1

ORIGINAL BANK QUESTION 1509 WAS USED TO DEVELOP 2017 NRC EXAM QID#59 (Used on 2008 NRC Exam)

Questions For All QID In Exam Bank

Bank:	1509	**Rev:**	000	**Rev Date:**	11/29/2001 3:54:2	**QID #:**	22	**Author:**	Coble
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	F	**Source:**	NRC Bank 0338 (2002 NRC Exam)		
Search	000060A102	**10CFR55:**	41.7 / 45.5 / 45.6	**Safety Function**	9				
System Title:	Accidental Gaseous Radwaste Release	**System Number**	060	**K/A**	AA1.02				
Tier:	1	**Group:**	2	**RO Imp:**	2.9	**SRO Imp:**	3.1	**L. Plan:**	A2LP-RO-CVENT
OBJ	13								

Description: Ability to operate and/or monitor the following as they apply to the Accidental Gaseous Radwaste Release: - Ventilation system.

Question:

Given the following plant conditions:

* Plant is in Mode 5 making preparations to refuel the reactor.
* RCS is in reduced inventory preparing to install SG nozzle dams.
* Containment Purge System is in service.
* When the 1st set of SG Manways are removed, the Control Room receives Annunciator 2K11 D-10 " Process Gas Radiation HI/LO".
* On 2C-25, the Gas Monitor for the Containment Purge System, 2RITS-8233, reading is above setpoint.
* Annunciator Corrective Action directs verification of Containment Purge secured.

The automatic actions that should have secured Containment Purge would be:

A. All Containment Purge supply and exhaust Isolation valves go closed.

B. Only the Outside-Outside Containment Purge supply and exhaust Isolations go closed.

C. Only the Inside-Inside Containment Purge supply and exhaust isolations go closed.

D. All three (3) Containment Purge exhaust isolation valves go closed.

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☑	☑

Audit Exam History

2006	☐
2008	☐

Answer:

B. Only the Outside-Outside Containment Purge supply and exhaust Isolations go closed.

Notes:

The only valves associated with the Containment Purge System that get a closure signal on a high process radiation alarm is the Outside-Outside supply and exhaust valves. These valves are considered containment isolations and verified closed from the ESF control panels 2C-16 and 17. The closing of these valves will trip the exhaust fan on low suction pressure and the supply fan is interlocked to trip if the exhaust fan is not running.

References:

OP 2203.012K, ACA for Process Gas Radiation High, Window 2K11 D-10
OP 2104.033, Containment Atmospheric Control, Supplement 1 Step 5.20
STM 2-9, Containment Cooling and Purge Systems, Sections 7.6 and Purge one line figure.

Historical Comments:

Question 60

Bank:	2390	**Rev:**	2	**Rev Date:**	12/19/2016	**2017 TEST QID #:**	60	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	011000A201	**10CFR55:**	41.7	**Safety Function**	2				
Title:	Pressurizer Level Control System (PZR LCS)	**System Number**	011	**K/A**	A2.01				
Tier:	2	**Group:**	2	**RO Imp:**	3.2	**SRO Imp:**	3.1	**L. Plan:**	A2LP-RO-APZRM
OBJ	2								

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Excessive letdown

Question:

Given the following at full power:

* One Charging Pump is running.
* Annunciator "RRS TROUBLE" comes in.
* PZR level setpoint input to PZR Level Control System is reading 55%.
* Annunciator 2K12 C-1 "LETDOWN HX 2E29 OUTLET TEMP HI" comes in.
* Letdown HX Outlet temperature on 2TIC-4815 is reading 150°F and slowly lowering.

Based on these conditions, the Letdown Rad Monitor Valve 2CV-4804 would ______________________ Letdown Rad Monitor and after restoring Letdown temperature back to normal, the isolation/bypass valve will be restored ______________________________________.

A. isolate Letdown to the; manually using PZR Malfunction OP 2203.028

B. bypass Letdown around the; manually using CVCS Operations OP 2104.002

C. isolate Letdown to the; automatically and verified with CVCS Operations OP 2104.002

D. bypass Letdown around the; automatically and verified with PZR Malfunction OP 2203.028

Answer:

A. isolate Letdown to the; manually using PZR Malfunction OP 2203..028

Notes:

A is correct as 2CV-4804 will automatically isolate flow to the rad monitor at > 145°F and after the high temperature actuation has been cleared <140°F, then the isolation valve will be restored to open IAW AOP 2203.008 Step 5 Contingency Action G. To do this the hand switch for 2CV-4804 must be taken to close then back to AUTO or OPEN.

B is incorrect as the rad monitor will not be bypassed and the PZR Malfunction AOP has the guidance to restore the Letdown Rad Monitor but plausible as the isolation valve to the rad monitor has to be manually opened to restore flow to the rad monitor.

C is incorrect as the isolation valve will not automatically come back open but plausible as letdown will be isolated and the Letdown Ion Exchanger Bypass Valve 2CV-4803 will bypass the CVCS ion exchangers at > 145°F and automatically align Letdown flow back to the ion exchangers when Letdown temperature drops below 140°F.

D. is incorrect as the rad monitor will not be bypassed nor automatically restored and the PZR Malfunction AOP has the guidance to restore the Letdown Rad Monitor but plausible because the Letdown Ion Exchanger Bypass Valve 2CV-4803 will bypass the CVCS ion exchangers at > 145°F and automatically align Letdown flow back to the ion exchangers when Letdown temperature drops below 140°F.

This question matches the K&A as the candidate must know the interlock actuation of the CVCS Rad monitor isolation due to the high Letdown Temperature due to a malfunction of the PZR Level Control System and the process to use to restore flow to the CVCS Rad Monitor.

References:

2203012L ANNUNCIATOR 2K12 CORRECTIVE ACTION REV. 49 Window C-1 LETDOWN HX 2E29 OUTLET TEMP HI (Verified reference updated 11/15/16);
AOP 2203.028 PZR Malfunction Rev. 13 Step 5 (Verified reference updated 11/15/16);
STM_2-04__31-1 CVCS Section 2.1.12 (Verified reference updated 11/15/16);
STM_2-04__31-1 CVCS Section 2.1.17 (Verified reference updated 11/15/16);
STM_2-04__31-1 CVCS Section 2.1.9 (Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Added "at full power' to the first bullet. Replaced "has lowered to; with "is reading" in bullet #3. Removed the word "Isolation" from the noun description of 2CV-4804 in the stem.

REV. 2 based on NRC Chief Examiner Feedback BNC. Removed the 4th bullet from Rev. 1.

ANNUNCIATOR 2K12

C-1

LETDOWN HX 2E29 OUTLET TEMP HI

1.0 CAUSES

1.1 Letdown temperature out of Letdown Heat Exchanger (2TC-4805) > 140°F

2.0 ACTION REQUIRED

2.1 Check the following indications:

- LD Temp CNTRL (2TIC-4815)
- Computer Point C&VCS HIGH LETDOWN TEMP (T4805)

2.2 Verify letdown flow (2FIS-4801) within 10 gpm of charging flow (2FIS-4863). Refer to Chemical and Volume Control (2104.002).

2.3 Locally verify CCW flow through Letdown Heat Exchanger (2FIS-526l).

2.4 IF Loop 2 CCW temperature high,
THEN reduce temperature using Component Cooling Water System Operation (2104.028).

2.5 IF Letdown HX Temperature controller (2TIC-48l5) NOT controlling in AUTOMATIC,
THEN perform the following:

2.5.1 Place Letdown HX Temperature controller (2TIC-4815) in MANUAL.

2.5.2 Raise CCW flow.

2.6 IF 2TIC-4815 NOT controlling in MANUAL,
THEN locally control temperature by throttling 2CV-5261 Bypass valve (2CCW-526l-3).

2.7 IF letdown temperature greater than or equal to 145°F,
THEN verify the following:

- Letdown to Ion Exchanger (2CV-4803) in BYPASS.
- L/D to Radmonitor valve (2CV-4804) closed.

2.8 IF letdown temperature greater than or equal to 150°F,
THEN verify Letdown isolated to prevent exceeding pipe design temperature. Refer to Chemical and Volume Control (2104.002).

3.0 TO CLEAR ALARM

3.1 Reduce letdown temperature out of Letdown HX to < 140°F.

4.0 REFERENCES

4.1 E-2457-3

INSTRUCTIONS	CONTINGENCY ACTIONS
5. Check "RRS TROUBLE" annunciator (2K10-H2) clear.	**5. IF malfunction caused PZR level setpoint to change, THEN perform the following:** A. Perform the following for Letdown Flow controller (2HIC-4817): 1) Place controller in MANUAL. 2) Adjust output to control PZR level within 5% of setpoint. B. Manually control Charging pumps. C. Manually operate PZR heaters. D. IF Remote Auto PZR Level setpoint incorrect, THEN place PZR Level controller in LOCAL AUTO and adjust setpoint based on T_{AVE} refer to 2102.004 Attachment E, Pressurizer Level Program. E. WHEN Letdown Flow controller (2HIC-4817) automatic and manual signals matched, THEN restore controller to AUTO using 2104.002, Chemical and Volume Control. F. IF failure prevents backup Charging pump operation, AND backup Charging pump required, THEN defeat stop interlock using 2103.005, Pressurizer Operations. G. IF Letdown Radiation monitor isolated due to high temperature AND Letdown HX Outlet temperature lowered to less than 140°F, THEN restore Letdown Radiation Monitor flow by opening Letdown Rad Monitor Isolation, 2CV-4804 (2HS-4804).

high/low pressure alarm set at > 540 psig or < 200 psig. This alarm is 2K12-E1 “LETDOWN HX DISCH PRESS HI/LO”. In order for 2PIC-4812 to function properly in automatic mode, 2HIC-4812A located at the remote shutdown panel 2C80 must also be in automatic. Normally both of the backpressure control valves are in service at the same time. This provides slightly better pressure control and prevents/minimizes the pressure transient should a valve fail closed.

The selected backpressure control valve(s) can be manually controlled from either 2PIC-4812 on panel 2C09 or 2HIC-4812A on the remote shutdown panel 2C80. 2HIC-4812A is the "master" controller and if it is in manual any automatic or manual signals from panel 2C09 controller 2PIC-4812 will be ignored.

During shutdown cooling purification, 2CV-4810/4811 is used to control the amount of purification flow. The flow is controlled with 2PIC-4812 in the manual mode of operation.

2.1.11 Letdown Relief Valves 2PSV-4800 and 2PSV-4822

Letdown relief valve 2PSV-4800 is designed to protect the low pressure piping, filters, ion exchangers and strainers from being over-pressurized. This pressure safety valve relieves at 200 psig and discharges to the 2T12 holdup tanks. The valve is located in the "B" charging pump room. Both relief's 2PSV-4800 and 2PSV-4822 discharge lines pass through the charging pump rooms. If either pressure safety valve has lifted or is leaking, its position can be verified by checking radiation levels and temperature at the discharge piping for abnormal trends.

Problems have been experienced in the past with 2PSV-4800 and 2PSV-4822 lifting and causing an unexplained loss of inventory from the RCS. Some of the most common causes have been high differential pressure across the letdown filters or letdown strainer, as well as 2CVC-139 being throttled during a plant transient causing cycling of letdown flow.

A permanent gauge (2PI-4800) is installed to monitor shutdown cooling purification pressure when operating shutdown cooling purification at elevated RCS pressures. When this indicator is required to be placed in service, pressure is limited to 150 psig. In addition, when in this condition this reading is added to the appropriate WCO log so that this pressure indicator is checked at a 2 hour intervals.

2.1.12 Boronometer and Process Radiation Monitor Isolation Valve 2CV-4804

The process radiation monitor isolation valve, 2CV-4804, serves to isolate the boronometer (which has been abandoned in place) and process radiation monitor automatically if high temperature conditions exist at the outlet of the letdown heat exchanger. The valve is air operated with a solenoid pilot valve and fails closed on loss of instrument air or on a loss of DC power supply 2D22-2. This valve is physically located in the letdown valve gallery, on the 335’ elevation of the auxiliary building.

2CV-4804 can be positioned from the control room panel 2C09. Normally the handswitch for 2CV-4804 is in "automatic" and receives a letdown heat exchanger outlet temperature signal from 2TC-4805.

2K12-C1 “LETDOWN HX 2E29 OUTLET TEMP HI” alarms if letdown temperature out of the letdown heat exchanger is > 140°F based on 2TC-4805.

If temperature as sensed by 2TC-4805 increases to 145°F, 2CV-4804 will close isolating letdown flow from the boronometer and process radiation monitor. If the high temperature condition has been corrected, the handswitch for 2CV-4804 must be taken to "close" and then returned to "auto" or "open" to restore flow.

2.1.13 CVCS Process Radiation Monitor 2RE-4806

The purpose of the CVCS process radiation monitor is to alert the operator of an increase in RCS radioactivity. An increase in RCS gross activity and iodine (I_{131}) may be indicative of possible failed fuel cladding. An increase in gross activity alone would be indicative of a crud burst, although a crud burst with some failed fuel would result in a small I_{131} spike also.

I_{131} is a fission product found in relative abundance in the event of a clad failure. It is released from fuel rods having cladding damage with relative ease and does not plate out on system surfaces. During radioactive decay I_{131} emits a 0.3645 MeV gamma particle that can be readily monitored using discrimination techniques.

I_{131} has an 8.04 day half-life that is long relative to sample time but short enough to indicate the current amount of fission product escape from the fuel[2].

The gamma detector used in the process radiation monitor is a gamma scintillation detector based on a sodium iodide crystal[3]. The detector is 1.5" in diameter x 1" long with a photo multiplier tube and internal pre-amplifier. It is designed to detect gamma radiation in the 100 kev - 3 MeV Range.

A logarithmic rate meter (2RITS-4806A) allows the operator to monitor for gross activity between 10 and 1E6 counts per minute. A single-channel linear rate meter / analyzer (2RITS-4806B) monitors specific I_{131} activity of between 0 and 1E6 cpm. These monitors are located on panel 2C22 in the Unit 2 control room. A Yokogawa

FOOTNOTES:

2 Some fission products that would contribute to gross activity readings plate out on pipe walls. Gross gamma activity after a minimum lag time is dominated by gaseous product activity so the gross activity reading is more representative of actual activity conditions of the RCS.

3 It has been known for many years that certain solids and liquids, called phosphors, emit light when exposed to radiation. When ionizing radiation passes through the phosphorus, the radiation causes the molecules of the material to become ionized or excited; these molecules then emit their excess energy in the form of light, each interacting particle resulting in a flash, generally called a *Scintillation*. In a scintillation detector, a photomultiplier tube detects this light, and the resulting pulses of current out of the photomultiplier indicate passage of the ionizing radiation through the scintillator. The resulting charge is proportional to the energy lost by the ionizing radiation to the scintillation crystals.

exchanger was designed to reduce end-of-life boron concentration by removing boron molecules from the letdown fluid minimizing the amount of reactor makeup water that would be needed to lower boron concentration from low ppm values. The ion exchanger was not borated and contained only anion resin.

The construction and operation of 2T70 is identical to the purification ion exchangers.

2.1.17 Ion Exchanger Bypass Valve 2CV-4803

The purpose of 2CV-4803 is to divert letdown flow around the purification ion exchangers if the outlet temperature of the letdown heat exchanger, as sensed by 2TC-4805, increases to 145°F. This serves to protect the ion exchanger resins from damage. The valve will automatically reposition to direct flow back through the ion exchangers if letdown temperature decreases below 140°F.

Additionally, in the event of dilution accident, 2CV-4803 allows the control room staff to bypass the ion exchangers. As previously mentioned, the ion exchangers may remove the boron in the letdown system. When that diluted borated water is returned to the RCS, a positive reactivity addition can occur to the reactor.

An air operated valve, 2CV-4803 is located in the letdown ion exchanger valve gallery on the 335 elevation of the auxiliary building. 2CV-4803 is controlled by a 3-position handswitch, 2HS-4803, located on control room panel 2C09. When this handswitch is in "auto" the valve will automatically position to bypass the ion exchangers. The valve will fail to the "bypass" position on a loss of instrument air or on a loss of 125 vdc power supply 2D22-02.

2.1.18 Letdown Strainer 2F-28

The letdown strainer is located in the letdown demineralizer valve gallery on elevation 335' and is provided as a back-up to the ion exchanger retention devices to prevent resin from entering the VCT. If the differential pressure across the strainer is ≥ 20 psid an alarm will sound in the control room. Annunciator 2K12-K1,"LETDOWN STRAINER ΔP HI", is actuated by 2PDIS-4830. A hard piped line is installed and procedures are available to flush the strainer to the spent resin tank (2T-13).

2.1.19 VCT Inlet Valve 2CV-4826

2CV-4826 is a 3-way air operated valve provided to direct letdown flow to the volume control tank (VCT) or the vacuum degassifier in the boron management system (BMS). The valve control circuit is powered from 2D22-2 and will fail to the VCT position on loss of instrument air or on a loss of 125 vdc power.

The VCT inlet valve can be controlled from the 3-position "Byp to BMS" handswitch 2HS-4826 located on panel 2C09 or from the remote shutdown panel 2C80 using 2HS-4826A. For 2CV-4826 to position automatically 2HS-4826A (panel 2C80) must be in "remote" and 2HS-4826 must be in "auto". When in "auto" 2CV-4826 will reposition to BMS if VCT level increases to 78%. The valve will shift back to the VCT position when VCT level decreases to 76%.

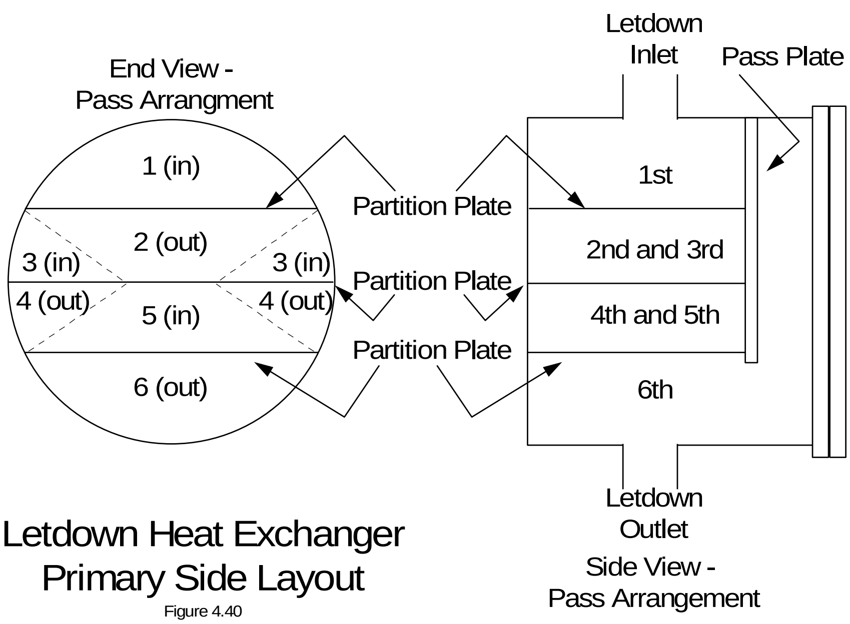
During 2R-11, a permanent gauge, 2PI-4800, was installed on the discharge line of the letdown heat exchanger. As previously mentioned, the backpressure control valves are used for flow control on shutdown cooling purification. To assist in maintaining pressure below the 200 psig setting on 2PSV-4800, 2PI-4800 is aligned in and the reading is recorded every 2 hours by operations. This gauge can also be aligned in during normal operations to monitor the pressure at the relief valve when a plant transient is expected (i.e.; control valve stroke test).

2.1.8 Letdown Relief Valve 2PSV-4822

Located downstream of the letdown flow control valves is relief valve 2PSV-4822. This pressure safety valve lifts at 600 psig and discharges to the 2T12 holdup tanks. 2PSV-4822 is designed to protect the letdown heat exchanger and system piping from being pressurized above the design pressure of 650 psig. The valve is located in the letdown heat exchanger room.

2.1.9 Letdown Heat Exchanger 2E-29

The letdown heat exchanger is provided to reduce the temperature of the letdown fluid from the outlet of regenerative heat exchanger to a temperature compatible with purification ion exchanger resin. The temperature of the letdown entering the 2E-29 is not expected to exceed 450°F and the temperature on the outlet should not exceed 140°F. A letdown heat exchanger outlet temperature of > 140°F, as measured by 2TE-4805, will actuate annunciator 2K12-C1," LETDOWN HX 2E29 OUTLET TEMP HI". In addition to causing an alarm, a letdown heat exchanger outlet temperature of 145°F, as measured by 2TE-4805, will isolate the letdown radiation monitor and bypass the letdown demineralizers.



Letdown Heat Exchanger
Primary Side Layout
Figure 4.40

Letdown flows through the letdown heat exchanger tubes in a "six-pass" flow arrangement. A "pass" is considered to be fluid flowing through one straight leg of a "U" tube. Flow separation between each "pass" is provided by tube arrangement on the tube sheet and by three pass partition plates. The pass partition plates are sealed by the pass plate cover, with a gasket, and bolted in place.

The first pass and second pass are actually one set of "U" tubes joined to the tube sheet as are the third and fourth passes, and the fifth and sixth passes. Flow passes from the letdown inlet into the first pass. Flow returns to the partition plate area via the second pass tubes. It is then directed into the third pass tubes and returns to the partition plate area via the fourth pass tubes. From the fourth pass tubes flow then enters

Question 61

Bank:	2391	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	61	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	072000K501	**10CFR55:**	41.11	**Safety Function**	7				
Title:	Area Radiation Monitoring (ARM) System	**System Number**	072	**K/A**	K5.01				
Tier:	2	**Group:**	2	**RO Imp:**	2.7	**SRO Imp:**	3.0	**L. Plan:**	A2LP-RO-RMON
OBJ	3								

Description: Knowledge of the operational implications of the following concepts as they apply to the ARM system: - Radiation theory, including sources, types, units, and effects

Question:

The High Range Area Radiation Monitor 2RITS-8912, located on the 404 CNMNT South West End Refuel Deck has a detector that is operated in the ______________ region of the Gas Amplification Curve and is designed to detect ______________ radiation events.

A. Geiger- Mueller; gamma only

B. Geiger- Mueller; gamma and neutron

C. Ion Chamber; gamma only

D. Ion Chamber; gamma and neutron

Answer:

A. Geiger- Mueller; gamma only

Notes:

A is Correct : Refer to the NOP 2105.016 Radiation Monitoring first paragraph in Section 3.0: The Area Radiation Monitoring System consists of permanently located radiation detectors, which provide continuous local and remote indication and alarm of direct radiation dose rate. Each monitor consists of a gamma sensitive Geiger-Mueller Tube, remote and local alarms, remote and local indicators and a remote power supply.

B is incorrect as Geiger-Mueller tubes are not designed to detect neutron radiation but plausible as most ARMs are GM detectors and they do detect gammas.

C is incorrect as ARMs do not operate in the ION Chamber region but plausible as this is a similar looking region on the Gas Amplification curve and gamma radiation is correct.

D is incorrect as ARMs do not operate in the ION Chamber region and ARMs do not detect Neutron Radiation but plausible as this is a similar looking region on the Gas Amplification curve and Ion Chambers are designed to detect neutron..

This question matches the K&A as the candidate must apply his knowledge of the ARMs in the plant to the theory of operation including the source and type of radiation and the effect of the radiation event.

References:

NOP 2105.016 Radiation monitoring Rev. 31 Step 3.0; (Verified reference updated 11/15/16)
STM_2-62_23-1 Radiation Monitoring Sys Page 4 - 9 (Verified reference updated 11/15/16);

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed stem to a specific ARM inside Containment that makes

neutron detection more plausible. Changed the distractors to be more plausible.

1.0 PURPOSE

This procedure describes plant area radiation monitors, process radiation monitors, Control Room Radiation monitors, SPING monitors, channel check of radiation monitoring instrumentation, and the Steam Generator Tube Leak N-16 Monitor System. Attachments are included for radiation monitor setpoints, an evacuation alarm and pager test, and a verification of N-16 monitor normal parameters and alarm setpoints.

2.0 SCOPE

The scope of this procedure includes Unit 2 Area Radiation Monitors setpoints, Process Radiation monitors setpoints, Control Room Radiation monitors, SPINGs, Evacuation System alarm checks, and Steam Generator Tube Leak N-16 Monitors checks.

3.0 DESCRIPTION

The Area Radiation Monitoring System consists of permanently located radiation detectors, which provide continuous local and remote indication and alarm of direct radiation dose rate. Each monitor consists of a gamma sensitive Geiger-Mueller Tube, remote and local alarms, remote and local indicators and a remote power supply.

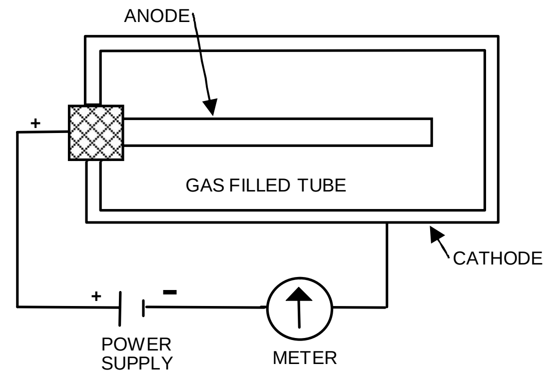
Two high range area radiation monitors are located in the Containment Building. Continuous remote indication is provided on panels 2C336-1 and 2C336-2. These post-LOCA qualified monitors are capable of monitoring from 1 to 10^8 R/hr.

The Process and Effluent Radiological Monitoring Systems consist of remote mounted and shielded detectors. Each gaseous and liquid monitoring system contains an in-line or off-line sample chamber, radiation detector, check source, associated process filters and applicable recording equipment. Gaseous effluents are monitored by the SPING system. The following detectors are utilized:

- Beta and gamma sensitive GM Tubes for gaseous monitoring.
- Gamma scintillation crystal with GM Tube for liquid monitoring.
- The Containment Atmosphere Monitor System (CAMS) uses both direct and scintillation detector systems.

1.5 Gas Amplification Curve

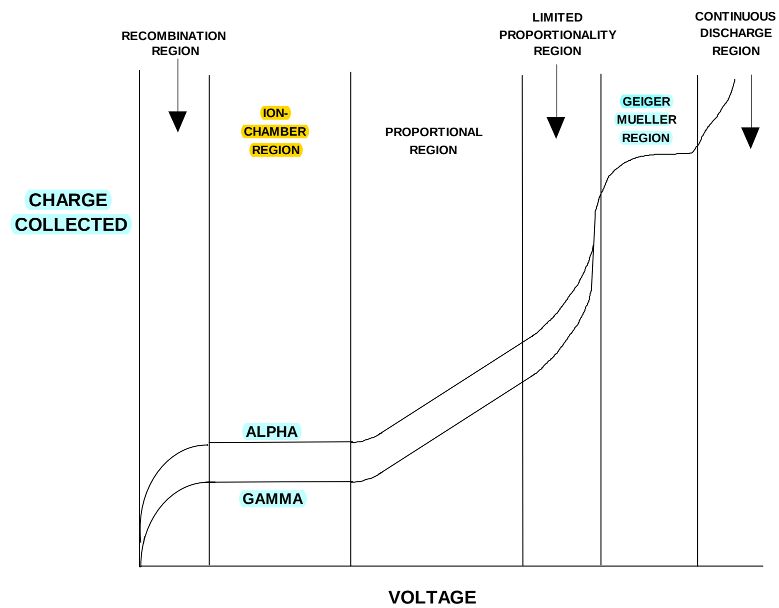
The Gas Amplification Curve is a graphic presentation of the expected output of a gas filled detector for a given voltage. Gases in general are poor conductors of electrons unless they become ionized. Ionization will occur when a gas is subjected to a radiation field. The higher the voltage that is applied to the detector, the higher the speed at which the electrons move toward the positively charged electrode . As the speed of the electrons goes up the chances of the ions recombining (thus reducing the output) will lower. When voltage is raised to a high value, the electrons can be stripped off the negatively charged electrode with no radiation field present.



BASIC GAS FILLED DETECTOR

For the desired application the voltage supplied to the detector (refer to basic detector above) is varied so the detector is operating in the specified region.

The Gas Amplification Curve is depicted below.



GAS AMPLIFICATION CURVE

The Recombination region is the area identified as Region I on the curve. This region is characterized by low voltage. The majority of the ion pairs created recombine prior to reaching the electrode so little or no output is seen for the ionizing event.

The Ion Chamber region is Region II on the curve. The voltage in this region is sufficiently high that all electrons produced by the ionizing events are collected on the anode. The output pulses are equal to the number of the ionizing event. Since more energetic radiation will cause more ions to be formed, the size of the pulse will be larger for events with more initial energy. Due to this the region can be used for dose rates.

Region III is referred to as the Proportional region and has a voltage potential large enough to cause secondary ionization. When an ion pair is produced, the ions move quickly to the electrodes. This added kinetic energy allows the ions to create more ion pairs. This is known as *gas multiplication*. The number of pulses out of the detector is proportional to the number of ionizing events. The size of the pulse is also proportional to the energy of the incoming radiation therefore this region can be used for dose rates.

The Limited Proportional region, or Region IV, is an unstable region. Secondary ionization occurs and can cause complete ionization of the gas. Since it is unknown if the output is due to partial or complete ionization, this region is not used for detection.

Region V is known as the Geiger-Mueller region. This region has a voltage high enough to cause complete ionization of the gas for every radiation event. The complete ionization by secondary ionization is called *avalanche of ionization*. The typical gas amplification ratio for this region is 10^6 / 1. The pulses out of a detector operating in this range are the same size regardless of initial radiation energy. Since this region causes complete ionization to occur, a certain amount of time is required for the gas to stabilize before the next ionization event can be detected.

Region VI of the curve is the Continuous Discharge region. This region is extremely unstable. The applied voltage is so high that gas ionization can occur without any incoming radiation.

1.6 Geiger-Mueller Detector Operation

A Geiger-Mueller tube is a gas filled device that reacts to ionizing radiation events. Basically, it consists of a positive anode wire surrounded by a cylindrical negative cathode. Beta particles or gamma rays enter the tube through a foil window or directly through the wall of the tube. The tube is normally filled with a rare or noble gas and a quenching agent. High voltage is applied to the detector so a single radiation event will cause complete ionization of the tube. Reference basic detector on page 4.

When radiation enters the tube it will interact with the noble gas (normally argon, neon or helium) and cause electrons to be ejected from the gas molecules. This is known as *primary ionization*. Due to the high voltage potential these electrons are accelerated towards the anode. The highly accelerated electrons will knock other electrons out of the gas molecule orbits thereby causing *secondary ionization* events. This process continues until most of the gas is ionized. This whole event is also known as gas amplification.

2.0 DETAILED SYSTEM DESCRIPTION

2.1 Area Radiation Monitoring System

The Area Radiation Monitoring System consists of thirty area monitors. Twenty-four of these monitors are installed for general area normal operation; twenty are low range units and four units are for high range applications. Two of the 30 Area Monitors are high energy gamma detectors and are used in the Containment Building for post accident monitoring. The remaining four Area Monitors are supplied by Eberline. Two of these Eberline units are used in the Post Accident Sampling System Building while the other two Eberlines are used on the Main Steam piping for primary to second tube leakage indication.

The power supplies for the Area Radiation Monitoring System are shown on page 52 and 53.

2.1.1 General Area Normal Operation Area Monitors

The criterion for selecting detector locations is to provide a minimum of one radiation monitor on each elevation of the Auxiliary Building where equipment handling normally radioactive material is located. On each elevation the monitors are placed in those areas having 40 hr/week occupancy designations in such a manner that the monitors, when alarming, can be seen and/or heard from all ingress points as well as the areas having the highest expected occupancy. Three monitors in the Spent Fuel Pool area serve as criticality alarms in the new fuel and spent fuel handling and storage areas.

The Area Radiation Monitors are provided with audible and visual alarms locally on indication of high radiation levels. The remote location for the affected monitor is provided with both high and low level radiation alarms. The associated Control module has alarm lights for both high and low radiation level alarms. A Control Room annunciator, 2K11-B10, will alarm if either the high or low setpoints are exceeded.

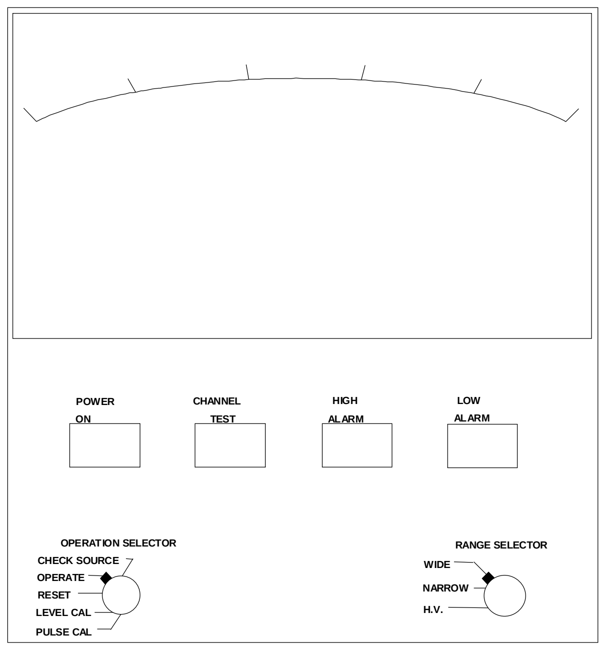
Ambient temperatures in excess of 150° F can cause unreliable radiation level indication on the Area Monitor due to thermally induced “noise”. This condition will not, however, damage the equipment. When ambient temperature lowers to less than 150° F the radiation level indication should be accurate once again.

The alarm setpoints are set in accordance with procedure OP-2105.016, Radiation Monitoring and Evacuation Alarm System. The low alarm setpoints for each Area Monitor is set below the normal background level and basically acts as a circuit failure alarm. The high range area monitors for Containment (2RITS-8912, 2RITS-8909 and 2RITS-8905) have a keep live circuit to prevent spurious low alarms due to low background levels. The keep live circuit provides a “heart- beat pulse” that prevents low alarms as long as it sees a real pulse within a set time period. The time period can be adjusted between 30 seconds and 10 minutes. The actual value is determined by Engineering (EC 13393). EC-52490
installed a keep alive souce in the detector housing of Containment High Range Area Radiation Monitors. This source

provides a consistent inpulse to the detector in normally low background radiation areas. This prevents circuit failure alarms (Low Alarm) due to low background radiation readings.The high alarm setpoint for each monitor is adjusted according to the monitors' location in the plant. For Area Monitors located outside the Controlled Access Area, the high level alarm setpoint is adjusted to 2.5 millirem per hour. The Spent Fuel Pool Area radiation monitors are set to less than or equal to the Tech Spec 3.3.3.1 limit of 1.5 x 10^{-2} R/Hr. All other Area Monitor high radiation level alarm setpoints are set at 3 times normal background radiation level provided that this value is not higher than the Safety Analysis Report design zone dose rate limit.

2.1.2 Control Modules

Each of the Area Radiation Monitors has a Control Module located on 2C25 in the back of the Control Room. There are two switches available on each of these modules. The switches are, *Operation Selection and Range Selection*.



The Operation Selection switch has five positions;

- OPERATE - in this position the meter reads input from the detector
- RESET - in this position the meter goes to low scale and resets the high alarm
- LEVEL CAL - Inserts a test signal which corresponds to a meter indication of approximately 1 X 10^{-2} R/Hr
- PULSE CAL - Inserts a signal from a pulse generator to provide a method of checking the counting circuit.
- CHECK SOURCE - Energizes a solenoid in the detector that pushes a radioactive source out of its lead shielding up beside the Geiger-Mueller tube.

The Range Selection switch has three positions;

- WIDE - selects the range of 1 X 10^{-4} to 1 X 10^{1} R/Hr for the indicator circuits
- NARROW - selects the range of 1 X 10^{-4} to 1 X 10^{-1} R/Hr for the indicator circuits
- H.V. - Used to indicate the high voltage supplied to the detector. Scale of 0 -2.5 kilovolts

The Control Modules on 2C25 also have a POWER ON light as well as a CHANNEL TEST light. The Channel Test light will be on whenever the Operation Selection switch is not in the OPERATE position.

During normal operation of the radiation monitor circuit the Range Selection switch should be in the WIDE position, the Operation Selection switch in the OPERATE position and the POWER ON light illuminated.

Each of the Area Radiation Monitor Control modules also supply a multi-point recorder located on 2C25. This provides immediate trend capability and permanent record retention of the radiation levels in each area.

As stated earlier there are 20 low range Area monitors designed for normal plant operations. These low range Area monitors use a standard gamma sensitive Geiger-Mueller tube that is designed to measure gamma activity over the range of 0.1 milliroentgens to 10 Roentgens. The detector units are wall mounted and provide a local read out. The check source is a 0.9 microcurie Strontium-90 source. Activation of the Check source should cause an up-scale reading of one-third the scale range.

There are 3 high range Area Radiation Monitors designed for normal plant operations. Control Room cabinet 2C25 has an additional module for a high range monitor that is currently a spare. The detectors in the high range monitors are gamma sensitive GM tubes designed to measure the gamma activity of the surrounding air. Each monitor has a range of 10 milliroentgens per hour to 1000 Roentgens per hour. The high ranges Area Radiation Monitors do not have a Check source but they do have a keep alive souce installed per EC52490.

SAR 12.1.4.2 states that on occasion other alarm points may be selected for area radmonitors depending upon work in progress in the area or operations that will vary the normal measured radiation levels in the area. Some monitors are expected to have different alarm points when the reactor is critical than they will have when the reactor is shut down. (CR-ANO-2-2006-01782). NON-TECH SPEC AREA MONITOR ALARM SETPOINT CHANGE (OP-2105.016, Attachment I) has been provided to adjust alarm setpoint of Non-Tech Spec and Non-Containment Area Monitors as needed.

Question 62

Bank:	2392	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	62	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	002000K405	**10CFR55:**	41.7	**Safety Function**	2				
Title:	Reactor Coolant System (RCS)	**System Number**	002	**K/A**	K4.05				
Tier:	2	**Group:**	2	**RO Imp:**	3.8	**SRO Imp:**	4.2	**L. Plan:**	A2LP-RO-RMON
OBJ	18								

Description: Knowledge of RCS design feature(s) and/or interlock(s) which provide for the following: - Detection of RCS leakage

Question:

Which of the following Radiation monitors would detect a small leak upstream of Letdown Isolation valve 2CV-4820-2 at 30% Reactor Power?

A. Letdown Line Radiation Monitors 2RE-4806.

B. Containment Atmosphere Monitor 2RITS-8231.

C. Containment Purge Radiation Monitor 2RITS-8233.

D. Loop 2 CCW Return Radiation Monitor 2RITS-5202.

Answer:

B. Containment Atmosphere Monitor 2RITS-8231.

Notes:

B is correct. The Containment Atmosphere Monitoring System, CAMS, provides redundant channels to continuously monitor gaseous and particulate activity levels in the Containment Building. This system is part of the Reactor Coolant System Leakage Detection System per TS. 3.4.6.1. The Containment Atmosphere Monitoring System (CAMS) is designed to detect small changes in RCS leakage inside Containment. A small rise in RCS leakage between 0.5 gpm and 1.0 gpm should be detectable by CAMS and provides indication to the control room staff. 2CV-4820-2 is located inside Containment and comes directly from the RCS

A is incorrect because the Letdown Line Radiation Monitors will not detect leakage into the Containment Building from Letdown but plausible because they are designed to detect rising RCS activity into the Letdown/ CVCS from a fuel leak or crud burst.

C is incorrect because the Containment Purge system is not used during power operations but plausible as the Containment Purge Radiation Monitors 2RITS-8233 are designed to detect activity from the RCS in the Containment building when shutdown but applicant must understand that Containment Purge cannot be used in Mode 1.

D is incorrect because the Loop 2 CCW Return Radiation Monitor will not detect leakage into the Containment Building but plausible because these monitors can detect an RCS leak from the RCP heat exchangers into the CCW system and the Letdown Regen HX should the leak be located there.

This question matches the K&A a s the candidate must have knowledge of the design features that provide for RCS leakage detection into the Containment Building.

References:

ANO-2 TS3.4.6.1 Amendment 281, LCO and Basis (Verified reference updated 11/15/16);
STM-2-62 Rev.23 Radiation Monitoring Section 2.2.12 CAMs (Verified reference updated 11/15/16); STM-2-09 Rev.16 Containment Cooling and Purge Section 1.2.6 CAMs (Verified reference updated 11/15/16); STM-2-62 Rev.23 Radiation Monitoring Section 2.2.1 CVCS Process Rad Monitors (Verified reference updated 11/15/16); STM-2-62

Rev.23 Radiation Monitoring Section 2.3.7.3 Containment Purge Rad Monitors (Verified reference updated 11/15/16); STM-2-62 Rev.23 Radiation Monitoring Section 2.2.2 CCW Process Rad Monitors (Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed the stem to a specific leak location and removed any reference to inside Containment. Changed Main Steam Rad Monitor to Letdown Line Rad Monitor.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection instrumentation shall be OPERABLE:

a. One containment sump level monitor

b. One containment atmosphere particulate radioactivity monitor, and

c. One containment atmosphere gaseous radioactivity monitor.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one or more containment atmosphere radioactivity monitor(s) inoperable, operation may continue for up to 30 days for each inoperable monitor provided:

   1. grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours, or

   2. a Reactor Coolant System water inventory balance is performed at least once per 24 hours in accordance with Surveillance Requirement 4.4.6.2.1.a;*

   otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the containment sump level monitor inoperable, operation may continue for up to 30 days provided a Reactor Coolant System water inventory balance is performed at least once per 24 hours in accordance with Surveillance Requirement 4.4.6.2.1.a;* otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the containment sump level monitor inoperable and one containment atmosphere radioactivity monitor inoperable, operation may continue for up to 30 days for each inoperable monitor provided a Reactor Coolant System water inventory balance is performed at least once per 24 hours in accordance with Surveillance Requirement 4.4.6.2.1.a;* otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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* Not required until 12 hours after establishment of steady state conditions.

REFERENCES

1. NEI 97-06, *Steam Generator Program Guidelines*.

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 50.67.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, *Basis for Plugging Degraded Steam Generator Tubes*, August 1976.

6. EPRI, *Pressurized Water Reactor Steam Generator Examination Guidelines*.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

GDC 30 of Appendix A to 10 CFR 50 requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems" May 1973. Likewise, the actions implemented upon inoperability of a required leak detection instrument are sufficient in maintaining the diversity and accuracy needed to effectively detect RCS leaks.

Industry practice has shown that water flow changes of 0.5 gpm to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. In addition, the reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Instrument sensitivities of $10 - 10^6$ cpm for particulate and gaseous monitoring are practical for these leakage detection systems.

12 hours is provided by a footnote to allow for plant stabilization before performance of the required reactor coolant inventory balance. This provision is necessary to ensure an accurate measurement is obtained.

The Containment Atmosphere Monitoring System (CAMS) has been modified to permit adjustments in containment pressure and oxygen level. The modification does not interfere with the leak detection capability of the CAMS and is isolated automatically, along with the CAMS, upon receipt of a Containment Isolation Signal. The modification involves insignificant flow rates and is monitored through building exhaust paths; therefore, the modification does not constitute a PURGE.

11. Starts 2VSF-9 (Unit 2 Control Room Emergency Supply Fan)

2.2.12 Containment Atmosphere Monitoring System (CAMS)

The Containment Atmosphere Monitoring System, CAMS, provides redundant channels to continuously monitor gaseous and particulate activity levels in the Containment Building. This system is part of the Reactor Coolant System Leakage Detection System. Changes in Reactor Coolant System leakage rate will cause changes in the Control Room indication of the containment atmosphere activity levels. Each of the channels has an air particulate activity scintillation detector (2RE-8231-1A and 2RE-8271-2A) and a gaseous activity Geiger-Mueller detector (2RE-8231-1B and 2RE-8271-2B).

CAMS 2RE-8231-1A/B is located on the 354 elevation of the Auxiliary Building at the west end of the hallway. The sample pump associated with 2RE-8231 is 2C-47 (2RE-8231-1A/B Sample pump). 2RE-8271-2A/B is located on the 354 elevation of the Auxiliary Building in the Upper North Piping Penetration Room. Its associated sample pump is 2C-48 (2RE-8271-2A/B Sample pump).

The indications and controls for 2RE-8231-1A/B and 2RE-8271-2A/B are on 2RITS-8231-1A/B and 2RITS-8271-2A/B respectively. Both of these modules are located on 2C25 in the back of Unit 2 Control Room. Each module has a read out for Particulate activity and gaseous activity. From these modules the individual pumps, the filter paper for the particulate detector and detector operation for each detector may be operated. The following provides the function of each switch and lamp.

“OPERATION SELECTOR” Switch-

1. CHECK SOURCE - Actuates the check source assembly and should cause an upscale reading of approximately one-third of full scale on the associated meter.
2. OPERATE - This is the normal position for this switch
3. RESET - Reset the HIGH ALARM bistable, neither the local lamp nor Control Room annunciator will return to normal until bistable has been reset.
4. LEVEL CAL - Used setting the high and low setpoints
5. PULSE CAL - Inserts an electronic signal to verify circuitry operation. Should have a reading of approximately 10^5 CPM on associated meter.

“RANGE SELECTOR” Switch-

1. WIDE - Selects the 10^1 to 10^6 CPM scale to the meter. This is the normal switch position.
2. NARROW - Selects the 10^1 to 10^4 scale for the meter
3. H.V. - The meter measures the High Voltage to the detector. Should read approximately 1400 VDC.

summer months is ~110°F.) Although not required, the CEDM shroud cooling system can be used post-accident to assist the containment heat removal system in reducing post-accident containment temperature and pressure.

1.2.5 Containment Purge and Exhaust System

The containment purge and exhaust system is designed to maintain the containment atmosphere activity below 10CFR20 limits for personnel access. This system serves no safety function. It can however, be used for post-accident containment atmosphere cleanup.

1.2.6 Containment Atmosphere Monitoring System

The Containment Atmosphere Monitoring System (CAMS) is designed to detect small changes in RCS leakage inside Containment. A small rise in RCS leakage between 0.5 gpm and 1.0 gpm should be detectable by CAMS and provides indication to the control room staff.

2.2.1 CVCS Process Radiation Monitor

A small portion of the Reactor Coolant system is diverted through the Chemical and Volume Control System (CVCS) via the Letdown header. This Letdown flow can be used to monitor the RCS coolant radioactivity. A rise in the radioactivity of RCS could be caused by crud released in the RCS or failure of the fuel cladding of the Reactor fuel assemblies. Crud is the result of activated corrosion products that settle out in low flow areas of the RCS. When flow is changed or when a chemical shock occurs these corrosion products are released back into the RCS resulting in a rise in the gross gamma activity.

The CVCS Process Radiation monitor, 2RE-4806 is located on the 335' elevation of the Auxiliary Building in the North - south hallway along the east wall. This process radiation monitor provides input to two separate monitoring circuits. One circuit provides the gross gamma activity while the other circuit provides specific fission product nuclide activity. The Gross gamma indication is read out on 2RITS-4806A while the specific activity level can be read on 2RITS-4806B.

The specific activity monitor 2RITS-4806B monitors the Letdown fluid for the presence of Iodine-131. Iodine-131 is a fission product that is released with relative ease from defective fuel assemblies and does not plate out on the system surfaces. As with all radionuclides, Iodine-131 emits a gamma particle with a specific energy level while undergoing radioactive decay. The 0.364 Mev gamma released by the decay of Iodine -131 can be readily monitored since its half-life of 8 days is longer than the sample lag time but short enough to provide indication of current fission product release into the RCS. A rise in the gross activity only would be an indication of a crud burst, but a rise in both gross and specific would be an indication of fuel failure.

The CVCS Process Radiation Monitor, 2RE-4806, is located in the CVCS Letdown line in parallel with the Purification filter but upstream of the Ion exchangers. This location provides for a continuous sample at relatively low temperature and pressure that can be conveniently obtained and the sample effluent can be returned to the CVCS without difficulty. This location also provides an optimum compromise between a minimum required sample lag time and the delay time required for the sample background radioactivity to decay. The time lag from the RCS to the CVCS Process monitor is sufficient to permit N-16 to decay to a sufficiently low level that will not interfere with monitor readings in any operating condition.

Process radiation Monitors, 2RITS-4806A and 2RITS-4806B, are located in the Unit 2 Control Room on 2C22.

activity detection has occurred the sample flow is returned to the exhaust duct.

The indication for 2RE-8542 is 2RITS-8542 located on 2C25 in the back of Unit 2 Control Room. The radiation recorder, 2RR-0645, on 2C25 provides a permanent record of the activity level detected. 2RITS-8542 has a high setpoint of 625 CPM. Upon reaching this setpoint, the module High light will light and annunciator 2K11-D10 will actuate.

2.2.7.3 Containment Purge Ventilation System Radiation Monitoring

The Containment Purge Radiation Monitor provides continuous monitoring of the gaseous activity levels released to the environment during normal Containment Building Purge operations via 2VEF-15 (Containment Purge Exhaust Fan). The Containment Building Purge System is normally out of service except during Refueling or Maintenance outages.

The Containment Purge Radiation Monitor 2RE-8233 monitors the exhaust of 2VEF-15 by pulling a sample of the exhaust flow using 2C-49 (2RE-8233 Sample pump). The monitor, pump and associated equipment are housed on the Unit 2 Auxiliary Building roof. The sample pump is controlled by a local handswitch. After activity detection has occurred the sample flow is returned to the exhaust duct.

The radiation monitor is provided to detect and indicate activity levels in the Containment Purge Exhaust Duct. The indication is on 2RITS-8233 on 2C25 in the back of Unit 2 Control Room, chart recorder 2RR-0645 provides a permanent record of the activity levels. The high setpoint for 2RITS-8233 will be set according to the setpoint listed on the Containment Purge Gaseous Release Permit (OP-2104.033 Supp 1) during initial Containment purging. Once initial purge of the Containment Building is complete the setpoint will be based on the background reading indicated on 2RITS-8233. This setpoint shall not exceed 2 times the normal background. Care should be taken not to adjust potentiometer below 0.00 or above 10.0 or damage could occur to potentiometer causing actual setpoint setting to be unknown (CR-ANO-2-92-0017). The potentiometer is located on top of the 2RITS-8233 module on 2C25 and is accessed by withdrawing module from the cabinet. Upon reaching this setpoint, the module High light will light and annunciator 2K11-D10 will actuate.

ADJUSTMENT OF CONTAINMENT PURGE RAD MONITOR SETPOINT (OP-2104.033, Attachment B) is provided to adjust 2RITS-8233 setpoint ≤ 2 X background following Containment Purge Operations or during Containment Continuous Ventilation Operations. It may also be used to adjust 2RITS-8233 setpoint to prevent a nuisance alarm if in modes 1-4 (CR-ANO-2-2004-00966) or for adjusting 2RITS-8233 setpoint > 2X background during Containment Continuous Ventilation Operations.

The Containment Purge Radiation monitor, 2RITS-8233, is interlocked with 2CV-8283-1 (Containment Purge Supply Isolation

counts or less per minute or a loss of power to the monitor. The low alarm circuit will cause annunciator 2K12-A1, Letdown Radiation Hi/Lo to actuate.

2.2.2 CCW Process Radiation Monitors

The Component Cooling Water system is directly monitored for activity levels currently present by two separate process radiation monitors. Monitor 2RE-5200 is provided for the Loop 1 CCW Return header while and 2RE-5202 monitors the Loop 2 CCW Return header. These monitors will detect leakage into the two CCW loops from components that may contain radioactive particles such as the Letdown Heat Exchangers or the Reactor Coolant pump coolers.

The Loop 1 CCW process monitor, 2RE-5200, is located on the 364’ elevation of the Turbine Building just east of Feed Water Heater 2E-3A. The Loop 2 CCW process monitor, 2RE-5202, is located on the 354’ elevation of the Turbine Building in the CCW room east of the “C” CCW Heat Exchanger 2E-28C.

Any radiation detected in the Component Cooling Water can be read on modules 2RITS-5200 or 2RITS-5202 located on 2C25 in the Unit 2 Control Room. The normal monitor readings are the result of background radiation only since contaminated leakage into the CCW system is not expected. The readings are continuously recorded on chart recorder 2RR-2330C that is also located on 2C25.

The monitors’ circuits have High radiation level alarms that illuminate a local alarm light on the affected module and activates Control Room annunciator 2K11-C10, PROC LIQUID RADIATION HI/LO. The High alarm setpoint for 2RITS-5200 is 160 CPM and the alarm setpoint for 2RITS-5202 is 200 CPM per procedure OP-2105.016, Radiation Monitoring and Plant Evacuation Systems. The annunciator will remain in alarm until the indicated level is below setpoint and the module local alarm is reset.

The modules are also provided with a circuit failure or Low radiation level alarm. This alarm is set at 10 CPM and will actuate annunciator 2K11-C10, PROC LIQUID RADIATION HI/LO. The low level alarm will reset automatically when indicated level is greater than 10 CPM.

2.2.3 Service Water Radiation Monitoring System

The Service Water System is continuously monitored for activity to provide leakage detection for any of the potentially contaminated heat exchangers that are cooled by the system. Leakage from any of the heat exchangers, such as the Shutdown Cooling Heat Exchangers, which provide cooling to radioactive systems could potentially release radioactive material to the Dardanelle Reservoir should a tube leak develop.

There are 5 process radiation monitors associated with the Service Water System. These monitors have indication modules on Control Room panel 2C25 and continuous recording on recorders 2RR-2330A/B that is also located on panel 2C25.

Question 63

Bank:	2393	**Rev:**	0	**Rev Date:**	7/27/2016	**2017 TEST QID #:**	63	**Author:**	Foster		
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NRC Exam Bank 2136				
Search	045000A406	**10CFR55:**	41.7	**Safety Function**	4						
Title:	Main Turbine Generator (MT/G) System	**System Number**	045	**K/A**	A4.06						
Tier:	2	**Group:**	2	**RO Imp:**	2.8	**SRO Imp:**	2.7	**L. Plan:**	A2LP-RO-TURBC	**OBJ**	21

Description: Ability to manually operate and/or monitor in the control room: - Turbine stop valves

Question:

When performing OP-2106.009, Turbine Generator Operations, section 14.0 Turbine Roll, when 100 RPM is selected on the Speed Set controller, the proper Main Turbine Stop Valves opening sequence is:

A. #1 Stop Valve bypass valve opens to equalize pressure then #1 Stop Valve opens followed by #2, #3 and #4 Stop Valves simultaneously.

B. #2 Stop Valve bypass valve opens to equalize pressure then #2 Stop Valve opens followed by #1, #3 and #4 Stop Valves simultaneously.

C #1 Stop Valve bypass valve opens to equalize pressure then #1 Stop Valve opens followed by the opening of #2, #3 and #4 Stop Valves in sequence.

D. #2 Stop Valve bypass valve opens to equalize pressure then #2 Stop Valve opens followed by the opening of #1, #3 and #4 Stop Valves in sequence.

Answer:

B. #2 Stop Valve bypass valve opens to equalize pressure then #2 Stop Valve opens followed by #1, #3 and #4 Stop Valves simultaneously

Notes:

B is Correct: #2 Stop valve is the "Master" and #1, #3 and #4 are the slaves. When #2 fully opens the other Main Turbine Stop Valves receive an open signal and simultaneously open.

A is Incorrect: #1 Stop Valve does not have a bypass valve opens (only #2 Stop Valve has the internal bypass to equalize pressure) but plausible as the "Slave" valves (#1, #3 and #4 Stop Valves) would open simultaneously and typically #1 would the first in most sequences.

C. Incorrect: #1 Stop Valve does not have a bypass valve opens (only #2 Stop Valve has the internal bypass to equalize pressure) and the "Slave" valves (#1, #3 and #4 Stop Valves) would open simultaneously not in sequence but plausible as typically #1 would the first in most sequences.

D is Incorrect but plausible as #2 Stop Valve bypass valve will open and equalize pressure across the Stop valve then #2 Stop Valve will fully open then send a open signal to the "Slave" valves (#1, #3 and #4 Stop Valves); however, the "Slave" valves would open simultaneously not in sequence.

This question matches the K&A as it requires the knowledge and ability to monitor the correct operation of the Main Turbine Stop Valves.

References:

NOP-2106.009, Turbine Generator Operations, Rev 81, Section 14.0 Step 14.10 (Verified reference updated 11/15/16); STM 2-24-1, Main Turbine Control System, Rev 22 Section 2.3.2 (Verified reference updated 11/15/16).

Historical Comments:

NRC Exam Bank 2136 was used on the 2014-2 NRC Exam
To be used on the 2017 NRC Exam but the correct answer/distractors were rearranged to balance out A, B, C and Ds.

NOTE
A FAST starting rate is used to minimize the time delay between selecting a Speed Setpoint and having Control Valves start to open.

14.7 At 2C01 select FAST starting rate.

14.8 Determine the fastest desired starting rate to be used in Step 14.10 or 14.18 as follows:

- IF First Stage Shell Inner Temp (T0218) is greater than 350°F,
  THEN use FAST or MEDIUM or SLOW starting rate.

- IF First Stage Shell Inner Temp (T0218) is between 250°F and 350°F
  OR has NOT been less than or equal to 225°F for more than four hours,
  THEN use MEDIUM or SLOW starting rate.

- IF System Engineering recommends FAST starting rate
  AND SYE monitors Turbine startup,
  THEN FAST starting rate may be used regardless of temperature and any RPM hold point may be skipped at SYE discretion.

NOTE
If Generator Seal Oil pressure has been lowered, rolling the Main Turbine could auto start the ESOP (2P-21).

14.9 IF maintenance has NOT been performed on Main Turbine,
THEN GO TO step 14.18 if desired.

14.10 At 2C01 select 100 on Speed Set-RPM
AND verify the following:

- WHEN Control valves (CV #1, CV #2 and CV#3) crack open,
  THEN select starting rate as determined in step 14.8.

- Main Stop valve (MSV #2) begins to OPEN.

- WHEN MSV #2 fully open,
  THEN MSVs 1, 3 and 4 slowly open.

- Intercept valves (IV #1 and IV #2) slowly open.

- Turbine rolls off turning gear.

- Speed Monitoring INCREASING SPEED light illuminates.

- WHEN set speed is reached,
  THEN Speed Monitoring AT SET SPEED light illuminates.

14.11 Place Turning Gear Motor 2K-6 (2HS-9630) to NORMAL AFTER STOP.

2.3.2 MSV Positioning Units

Of the 4 Main Stop Valves (MSV) only #2 MSV (2CV-0206) is continuously positioned. The #2 MSV functions to perform the following:

- Closed when CLOSE VALVES is selected
- Provide position control during CHEST/SHELL WARMING
- Be positioned wide open when any speed is selected by the Speed Set push buttons.

MSV #2 acts as a MASTER which controls the SLAVE valves, MSV 1,3, and 4. MSV 1,3, and 4 will open when MSV #2 is wide open and will close when MSV #2 is 90% closed.

As discussed earlier, MSV #2 has an internal bypass valve which is positioned by a position reference potentiometer when selected to Chest/Shell warming.

2.3.3 Valve Test Logic

The purpose of the Valve Test logic is to allow regular checking of the operation of all Turbine steam valves during normal on-line Operation. Procedurally Valve testing is performed every 31 days in accordance with Technical Requirements Manual, Section 3/4.3.4.1.

2.3.3.1 Control Valve Testing

CV Testing should not be performed when either Load Limit or Throttle Pressure Limiter is limiting. For any steam valve test the following Operator good practices should be observed.

- Test only one valve at a time,
- Allow the valve being tested to completely stroke back open before going to the next valve.
- Plant parameters should be allowed to recover after each valve test before continuing to the next valve.

The summed available capacity of all four Control Valves is required to be ≥ 110. The Operator can reduce Turbine load using either of the following methods before testing the CV's:

The Operator can either:

- Reduce Turbine load until the summed available capacity of all four Control Valves is ≥110% and Plant parameters are stable. These CV positions give Stage Pressure Feedback room to work with to control Turbine load. Reduce Turbine load while maintaining Reactor Power constant using SDBCS to pick up the extra steam demand. Turbine Load is reduced using the Load Limit Pot (preferred; the Load Set push buttons, Alternate method) while alternately opening a SDBCS valve in manual. Or,
- Using the Power Operations procedure, 2102.004, reduce plant power until the summed available capacity of all four Control Valves is ≥ 110%, then verify Plant parameters are stable.

After verifying that Stage Pressure Feedback is in service, the CVs are tested one at a time.

ORIGINAL BANK QUESTION 2136 WAS USED TO DEVELOP 2017 NRC EXAM QID#53 (Used on 2014-2 NRC Exam)

Questions For All QID In Exam Bank

Bank:	2136	**Rev:**	1	**Rev Date:**	7/11/2014 2:46:48	**QID #:**	64	**Author:**	foster
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW		
Search	045000A406	**10CFR55:**	41.7 / 45.5 to 45.8	**Safety Function**	4				
System Title:	Main Turbine Generator (MT/G) System	**System Number**	045	**K/A**	A4.06				
Tier:	2	**Group:**	2	**RO Imp:**	2.8	**SRO Imp:**	2.7	**L. Plan:**	A2LP-RO-TURBC
OBJ	21								

Description: Ability to manually operate and/or monitor in the control room: - Turbine stop valves

Question:

When performing OP-2106.009, Turbine Generator Operations, section 14.0 Turbine Roll, when 100 RPM is selected on the Speed Set controller, the proper Main Turbine Stop Valves opening sequence is:

A. #1 Stop Valve bypass valve opens to equalize pressure then #1 Stop Valve opens followed by the opening of #2, #3 and #4 Stop Valves in sequence

B. #2 Stop Valve bypass valve opens to equalize pressure then #2 Stop Valve opens followed by the opening of #1, #3 and #4 Stop Valves in sequence

C. #1 Stop Valve bypass valve opens to equalize pressure then #1 Stop Valve opens followed by #2, #3 and #4 Stop Valves simultaneously

D. #2 Stop Valve bypass valve opens to equalize pressure then #2 Stop Valve opens followed by #1, #3 and #4 Stop Valves simultaneously

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☒

Answer:

D. Correct

Notes:

D. Correct: #2 Stop valve is the "Master" and #1, #3 and #4 are the slaves. When #2 fully opens the other Main Turbine Stop Valves receive an open signal and simultaneously open

A. Incorrect: #1 Stop Valve does not have a bypass valve opens (only #2 Stop Valve has the internal bypass to equalize pressure) and the "Slave" valves (#1, #3 and #4 Stop Valves) would open simultaneously not in sequence

B. Incorrect: #2 Stop Valve bypass valve will open and equalize pressure across the Stop valve then #2 Stop Valve will fully open then send a open signal to the "Slave" valves (#1, #3 and #4 Stop Valves) but the "Slave" valves would open simultaneously not in sequence

C. Incorrect: #1 Stop Valve does not have a bypass valve opens (only #2 Stop Valve has the internal bypass to equalize pressure). The "Slave" valves (#1, #3 and #4 Stop Valves) would open simultaneously not in sequence

References:

OP-2106.009, Turbine Generator Operations, Rev 074, section 14.0 step 14.10 page 48 of 143
STM 2-24, Main Turbine, Rev 7 section 2.1 pages 6 and 7
STM 2-24-1, Main Turbine Control System, Rev 20 section 2.3.2 page 18
Lesson Plan A2LP-RO-TURBC objective 21: Describe the opening and closing sequence between the Master and Slave Intercept valves on the ANO-2 Low Pressure Turbines and the Stop Valves on the High Pressure Turbine.

Historical Comments:

Rev 1 corrected typos in Notes section. Cms 7-11-14

Question 64

Bank:	2394	**Rev:**	1	**Rev Date:**	1/12/2017	**2017 TEST QID #:**	64	**Author:**	Coble		
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam				
Search	0410002123	**10CFR55:**	41.5	**Safety Function**	4						
Title:	Steam Dump System (SDS) and Turbine Bypass Cont	**System Number**	041	**K/A**	2.1.23						
Tier:	2	**Group:**	2	**RO Imp:**	4.3	**SRO Imp:**	4.4	**L. Plan:**	A2LP-RO-OPROC	**OBJ**	7

Description: Conduct of Operations - Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question:

Given the following:

* The plant is stabilized at 12% Reactor power following a forced outage at 250 EFPD.
* The Main Turbine has been tied to the grid currently at 50 MW electrical output.
* SDBCS 2CV-303 is in Automatic at approximately 50% open.
* RCS T-ave is 548°F and steady.
* The CRS directs raising turbine load and stabilize Reactor power at 18% in accordance with OP-2104.004 Power Operations Step 8.25.

If no additional operator action is taken in the Control Room EXCEPT for raising turbine load, which of the following is the correct response of Reactor power and T-ave as Turbine Load is raised?

A. Reactor Power will start rising and T-ave lowering as soon as Turbine Load starts rising.

B. Reactor Power will remain the same and T-ave will rise as soon as turbine Load starts rising.

C. Reactor Power and T-ave will remain the same until all SDBCS valves have gone closed.

D. Reactor Power and T-ave will start rising after all the SDBCS valves have gone closed.

Answer:

C. Reactor Power and T-ave will remain the same until all SDBCS valves have gone closed.

Notes:

C is correct: The SDBCS will be in its normal automatic lineup for these plant conditions and the master controller will be getting inputs from the RCS T-ave via the Reactor Regulating System RRS. The master will regulate the SDBCS bypass valves to the condenser to maintain a constant T-ave for the given Reactor Power. These three bypass valve have a capacity of up to 27% Reactor power so they will be ~ 50% open for the given conditions. As turbine load rises the SDBCS will modulate the bypass valve closed to maintain T-ave and Reactor power constant until the bypass valves are fully closed. At this point T-ave will start to lower and the added positive reactivity and steam flow will cause Reactor power to rise.

A is incorrect as reactor power and T-ave will stay the same until all SDBCS go closed but plausible if the integrated plant relationship between Reactor Power, T-ave, SDBCS and Turbine Ops is not fully understood - in this case if SDBCS valves were all closed at the beginning of the turbine load increase, the response of Reactor Power and T-ave would be true.

B is incorrect because T-ave would remain the same but plausible if the integrated plant relationship between Reactor Power, T-ave, SDBCS and Turbine Ops is not fully understood and Reactor power will remain the same.

D is incorrect as T-ave will start to lower after the SDBCS valves have gone closed with no additional actions taken EXCEPT raising turbine load. Normally the ATC operator would anticipate integrated plant relationship between Reactor Power, T-ave, SDBCS and Turbine Ops and dilute the RCS to keep T-ave rising along with Reactor power. D is plausible

as this would be the normal power and temperature response if dilution was occurring.

This question matches the K&A because the candidate must fully understand the integrated plant relationship between Reactor Power, T-ave, SDBCS and Turbine Ops to adequately implement the integrated plant operating procedure 2102.004 - Power Operations.

References:

NOP 2102.004 Power Operation Rev. 61 Step 8.25 (Verified reference updated 11/15/16);
NOP 2102.004 Power Operation Rev. 61 Attachment C (Verified reference updated 11/15/16);
NOP 2105.008 SDBCS OPS Rev. 30 Step 9.0 (Verified reference updated 11/15/16);
STM_2-23_17-1 SDBCS Section 1.2 (Verified reference updated 11/15/16).

Historical Comments:

To be used on the 2017 NRC Exam

Rev. 1 based on post submittal validation comments: Changed the third bullet to state: SDBCS 2CV-303 is in Automatic at approximately 50% open.

8.24 WHEN at ~ 17% Reactor power,
THEN perform the following:

8.24.1 Check QASI Aux Trip (PID 187) enabled by checking EITHER of the following: A /B /C /D __/__/__/__

- Average of PID 10, 11, and 12 greater than or equal to 17% (displayed on PQASI screen for each channel).
- ASI (PID 268) is calculating (vice reading a "canned" value of -1.0000E-2) for each channel.

8.24.2 Verify the following at FWCS Engineering Work Station (EWS):

- ALL Signal RESETs on ALL Signal Validation Screens are Reset OR reason known and desired to continue. ______
- NO unexplained alarms present. ______
- KEY in OPERATE AND removed. ______

8.25 Stabilize Reactor power at 18% to 20% power (PID 177) while performing the following in any order:

8.25.1 Check Excore linear, CPC neutron, and CPC ΔT powers between - 0.5% and + 10% of COLSS power. ______

- IF any power indication out of spec, THEN record in Station log and initiate calibration within 24 hours.
- Refer to Unit Two Operations Linear Power Adjustment (2305.051).

8.25.2 Complete CEA Position surveillance using CPC/CEAC Operations (2105.001). ______

8.25.3 Using Attachment D of this procedure, verify programmed CEA withdrawal for ALL fully withdrawn CEAs. ______

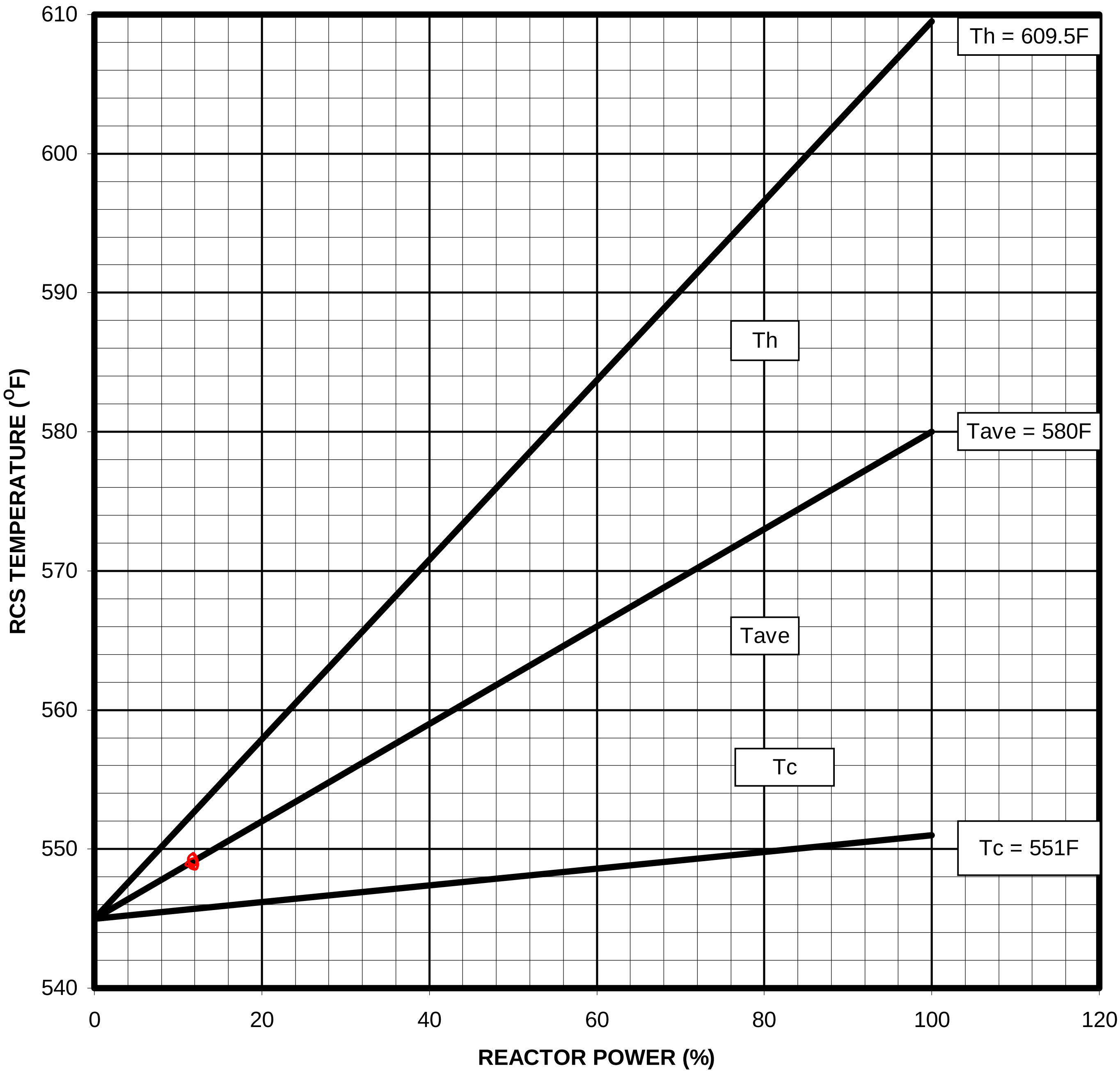
8.25.4 Verify COLSS Monthly Operability Test (2105.013, Supplement 1) completed within last 31 days (TS 4.2.3.c). ______

ATTACHMENT C

PAGE 1 OF 1

RCS TEMPERATURE VS REACTOR POWER

This temperature profile represents desired trend for RCS temperature vs. Reactor power levels. The actual values at near full power may vary.



9.0 OPERATION OF SDBCS DURING PLANT HEATUP

9.1 Review Limits and Precaution 5.4.

9.2 Verify the following initial conditions:

- Condenser #1 and #2 Unavailable lights on SDBCS panel (2C29) extinguished.
- ALL available Turbine Bypass and Downstream Atmospheric Dump valve Permissive switches in AUTO:
  - 2CV-0301 Permissive (2HS-0301)
  - 2CV-0302 Permissive (2HS-0302)
  - 2CV-0303 Permissive (2HS-0303)
  - 2CV-0306 Permissive (2HS-0306)
  - 2CV-0305 Permissive (2HS-0305)
- BOTH Upstream Atmospheric Dump valve Permissive switches in OFF:
  - 2CV-1001 Permissive (2HS-1001)
  - 2CV-1051 Permissive (2HS-1051)
- BOTH Upstream Atmospheric Dump valve Hand Indicating Controllers in MANUAL with less than or equal to zero output:
  - Hdr #1 UPSTM ADV 2CV-1001 (2HIC-1001)
  - Hdr #2 UPSTM ADV 2CV-1051 (2HIC-1051)
- SDBCS Master controller (2PIC-0300) in AUTO.

9.3 Momentarily depress ONE of the following pushbuttons:

- Vac/Emerg Off Reset pushbutton (2C02)
- EO/CI Reset pushbutton on SDBCS Panel (2C29)

* 9.4 WHEN RCS reaches hot zero power condition (545°F),
THEN verify SDBCS valves open to maintain S/G pressure at setpoint.

1.0 Introduction

1.1 System Functions

The functions of the Steam Dump and Bypass Control System (SDBCS) are as follows:

1) The system automatically dissipates a limited amount of excess energy in the Reactor Coolant System by regulating the flow of Main Steam through Turbine Bypass and Atmospheric Dump valves. Steam Header pressure is thereby controlled so that:

   a. Small load rejections can be accommodated without tripping the reactor or lifting either the pressurizer or main steam safety valves.

   b. Hot zero-power, or Hot Standby, conditions can be maintained.

   c. Desired RCS thermal conditions can be achieved during periods when Reactor power is required to be greater than turbine power (for example, during Turbine synchronization).

2) The system in manual allows operator control of Reactor Coolant System temperature during heatup or cooldown when the condenser is available using the Condenser Bypass valves or , when the condenser is unavailable, using the Atmospheric Dump valves.

1.2 General System Description

The Steam Dump and Bypass Control System (SDBCS) is a non-safety related system whose primary purpose is to provide a steam path from the Steam Generators to the Condenser or to the atmosphere. This is necessary in order to limit Main Steam Header pressure and, as a result, Reactor Coolant System temperatures during periods where the Main Turbine is unavailable or a Plant load transient is occurring.

The SDBCS controls seven valves. Four of the valves *Dump* to the atmosphere and have a capacity of 46% of total steam flow. Two of these Atmospheric Dump valves (ADVs) are located upstream of the Main Steam Isolation valves and the remaining two ADVs are located downstream.

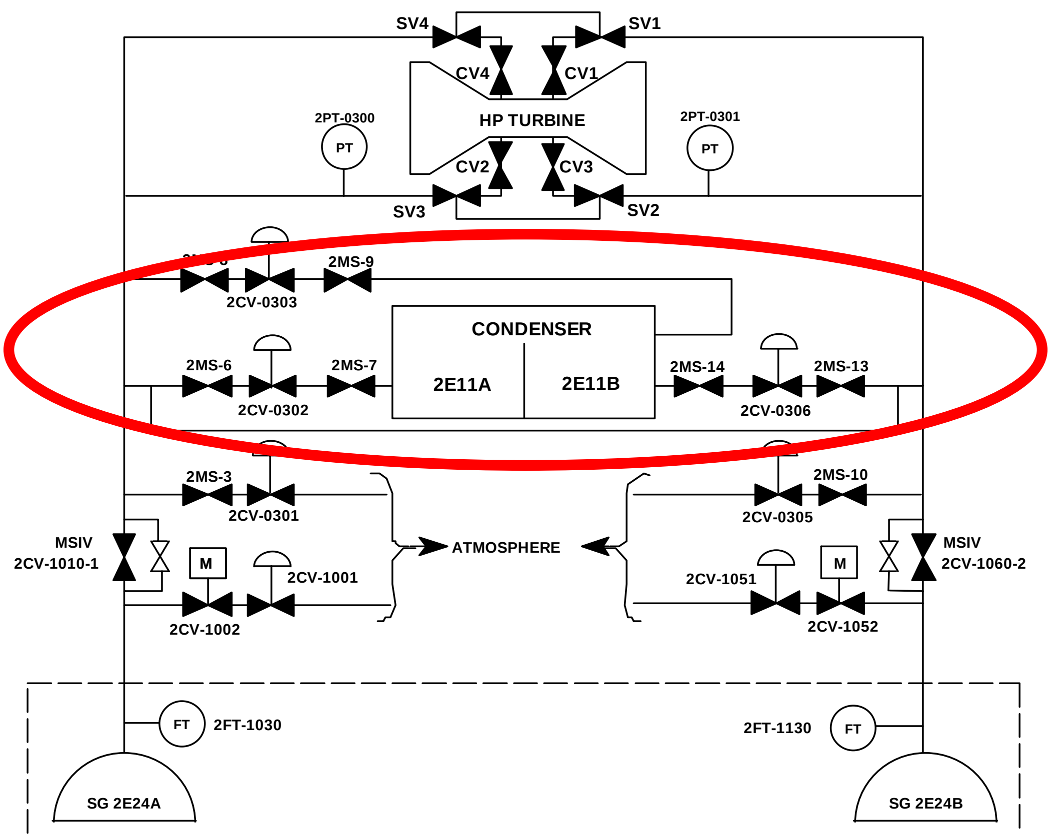
Three Turbine *Bypass* valves are provided to bypass Main Steam around the Turbine directly to the main Condenser. One of these valves has a capacity of 5% and the two other bypass valves have a capacity of ~11.5%. The combined Dump and Bypass capacity is ~74% of total Steam flow. The original design of the 11.5% Bypass and Dump valves was 13 1/3% capacity each, but the valves internals were modified during plant startup testing to reduce valve failure and improve reliability.

The following table lists the capacity of each of the SDBCS valves (900 psia @ turbine stop valves) and whether the valve is a Dump valve or a Bypass valve.

SDBCS Valve Summary Table

VALVE NUMBER	CAPACITY	TYPE
2CV-0303	5%	Bypass
2CV-0302	11.5%	Bypass
2CV-0306	11.5%	Bypass
2CV-0301	11.5%	Dump
2CV-0305	11.5%	Dump
2CV-1001 (Upstream MSIV)	11.5%	Dump
2CV-1051 (Upstream MSIV)	11.5%	Dump

The normal mode of operation is with the (3) Bypass valves and both (2) Downstream Atmospheric Dump valves (ADVs) set up for Automatic operation. The (2) Upstream ADVs are maintained in the "Off" position and isolated using Motor operated Isolation valves because of valve failure during plant S/U testing. This results in a normal line-up system capacity of ~51%.



SDBCS ONE-LINE DIAGRAM

The SDBCS is designed so that no single component failure or Operator error will result in the opening of more than one valve.

Accidental opening of a Steam Dump or Bypass valve could cause an excess heat removal accident. Excess heat removal from the RCS causes a lowering of the Reactor Coolant temperature, a rise in Reactor power due to the negative *Moderator Temperature Coefficient*, and a lowering of the RCS and Steam Generator pressures.

1.2.1 SDBCS Inputs

The SDBCS uses several input signals to perform it's calculations and develop its various output signals. The input signals to SDBCS are: (Refer to Figure on Page 37.)

1. Main Steam Header Pressure
2. Main Steam Header ΔP (Flow)
3. Pressurizer pressure (from Pressurizer Pressure Control System)
4. RCS T_{AVE} (from RRS)
5. Condenser pressure
6. Reactor Trip Signal (from FWCS)
7. Condenser Available signal
8. M/A control station status
9. SDBCS Master controller status
10. Permissive Switch status
11. Emergency Off
12. Emergency Off/Condenser Interlock Reset
13. Master Controller Output

Before we discuss the SDBCS circuitry it is necessary to understand how the mechanical components of the SDBCS valves function to position a SDBCS valve.

Question 65

Bank:	2395	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	65	**Author:**	Coble		
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC Exam Bank 0362				
Search	068000A302	**10CFR55:**	41.13			**Safety Function**	9				
Title:	Liquid Radwaste System (LRS)					**System Number**	068	**K/A**	A3.02		
Tier:	2	**Group:**	2	**RO Imp:**	3.6	**SRO Imp:**	3.6	**L. Plan:**	A2LP-RO-RWST	**OBJ**	4

Description: Ability to monitor automatic operation of the Liquid Radwaste System, including: - Automatic isolation

Question:

Given the following plant conditions:

* Unit 2 is at full power operation.
* Waste Condensate Tank (2T-21A) is on recirc using Pump 2P-53A
* A liquid release is in progress from Boric Acid Condensate Tank 2T-69A.
* Release Isolation 2CV-2330A is Open.
* Release Isolation 2CV-2330B is Open.
* Handswitch 2HS-2330 on 2C112 is in Position '3'.
* Handswitch 2HS-2331 on 2C112 is in Position "BOTH".

* NOW Waste Condensate Tank (2T-21A) Pump 2P-53A Discharge Valve 2CV-2122 is inadvertently opened.

At this point, which of the following is the correct status of the 2T-69A release and the valve positions?

A. 2CV-2330A Open and 2CV-2330B Open; 2P-53A Trips

B. 2CV-2330A Open and 2CV-2330B Closed; 2P-53A Running

C. 2CV-2330A Closed and 2CV-2330B Open; 2P-53A Trips

D. 2CV-2330A Closed and 2CV-2330B Closed; 2P-53A Running

Answer:

D. 2CV-2330A Closed and 2CV-2230B Closed; 2P-53A Running

Notes:

D is correct: The interlock between a Boron Management System tank outlet valve and a Liquid Radwaste tank outlet valve will close release isolations 2CV-2330A AND 2CV-2330B with the switch positions of 2HS-2330/2331 listed in the initial conditions. This is to prevent inadvertent release of a non permitted tank. These valve are in parallel. See attached references. The pump

A is incorrect because the release will terminate because of the interlock between a Boron Management System tank outlet valve and a Liquid Radwaste tank outlet valve will close both isolation valves and the pump will remain running but plausible if the candidate does not understand the integrated interlock between the LRW and the BMS system to prevent an unpermitted tank from being released.

B is incorrect because the release will terminate because both valves will close but plausible if the candidate does not understand the integrated interlock between the LRW and the BMS system to prevent an unpermitted tank from being released and the applicant may think that only the 'B' valve 2CV-2330A is tied to the 'A' Waste Condensate Pump 2P-53A outlet valve 2CV-2122 based on the position of 2HS-2330. This distractor is plausible as the valves are in parallel and the pump will remain running.

C is incorrect because the pump will remain running and the release will terminate by closing both release isolations but plausible if the candidate does not understand the interlock. Plausible because the applicant may think that only the 'A' valve 2CV-2330A is tied to the 'A' Waste Condensate Pump 2P-53A outlet valve 2CV-2122. and the 2CV-2330A/B are in series in the discharge line.

This question matches the K&A as it requires knowledge of the interlock to monitor the release and understand the reason for the automatic isolation to prevent an inadvertent release of an unpermitted tank.

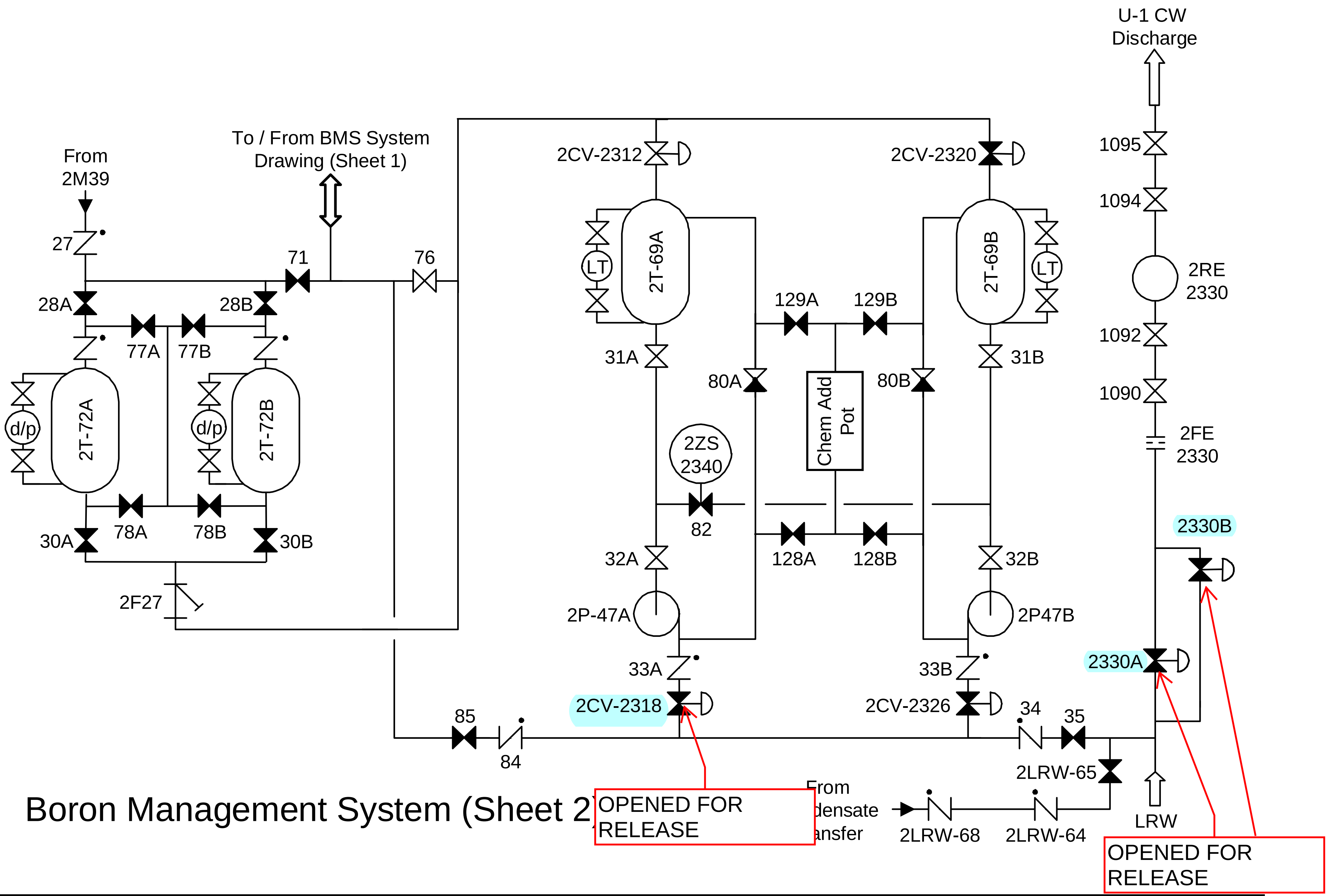
References:

STM_2-52_19-1 Boron Management Drawing (Verified reference updated 11/15/16);
STM_2-52_19-1 LRW Management Drawing (Verified reference updated 11/15/16);
STM_2-52_19-1 Liquid RW and Boron RW Management Section 2.3.2.1(Verified reference updated 11/15/16); STM_2-52_19-1 Liquid RW and Boron RW Management Section 3.5 (Verified reference updated 11/15/16).

Historical Comments:

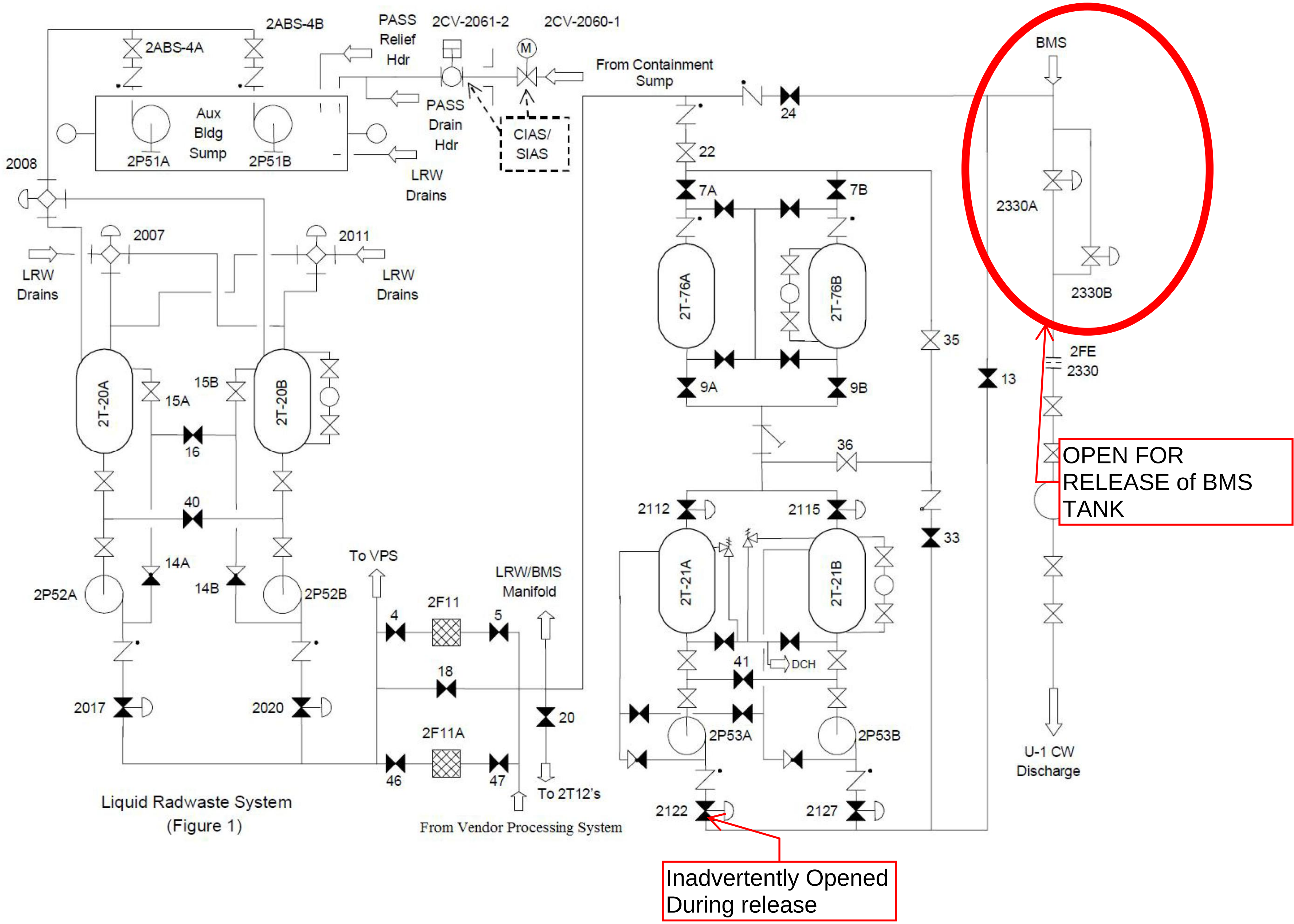
NRC Exam Bank 0362 used on the 2002 NRC Exam
To be used on the 2017 NRC Exam but altered the order to balance A, B, C, & Ds and swapped to 2CV-2230B instead of 2CV-2122.

REV. 1 based on NRC Chief Examiner Feedback BNC. Added the 2nd bullet to the question, removed the status of the release in the distractors, and added the 2P-53A pump status to the distractors to have an action in each one.



Boron Management System (Sheet 2)

Figures



Liquid Radwaste System (Figure 1)

2T21 tank low level interlocks are met. Placing the associated handswitch to STOP trips the associated pump.

The waste condensate pumps transfer processed water from the waste condensate tanks (2T21A/B) through the boron management system for discharge to the Unit-1 circulating water discharge flume.

The waste condensate pump recirculation lines are piped so that waste condensate tank recirculation can be accomplished with the associated discharge valve (2CV-2122 or 2CV-2127) closed.

The waste condensate pump suctions can be cross-connected via 2LRW-41 to transfer the contents of one waste condensate tank to the other or to allow either pump to dump the contents of either tank. 2LRW-41 has position indicating switch 2ZS-2114 associated with it to provide input into the control schemes for each pump and to provide valve position indication on panel 2C113.

A low level of ≤ 9% in the tank being pumped shuts off the pump aligned to it. This protects against a loss of suction even when a pump is cross-connected. The associated pump may be restarted when level is restored to ≥ 10%.

Power supplies for the waste condensate pumps are:

Unit:	Power Supply:
2P53A	2B31-B6
2P53B	2B41-B6

2.3.2.1 Waste Condensate Pump Discharge Valves (2CV-2122, 2CV-2127)

The waste condensate pump discharge valves (2CV-2122, 2CV-2127) are air operated valves designed to "fail closed" on a loss of instrument air or on loss of power.

Taking the handswitch on panel 2C113 to OPEN opens the associated discharge valve.

If either of the boric acid condensate pump discharge valves (2CV-2318 or 2CV-2326) and either of the liquid radwaste condensate pump discharge valves (2CV-2122 or 2CV-2127) are open, then the boron management system circulating water discharge control valves (2CV-2330A and 2CV-2330B) are interlocked closed. This ensures that only one discharge flowpath to the environment is available at any given time.

The power supply for the air solenoid for the waste condensate pump discharge valves is:

Unit:	Power Supply:
2SV-2122 / 2SV-2127	2Y2 breaker 22

3.5 BMS/LRW Circulating Water Discharge Control Valves (2CV-2330A and 2CV-2330B)

The BMS/LRW circulating water discharge control valves (2CV-2330A and 2CV-2330B) control discharge of liquid radwaste to the Unit-1 circulating water discharge canal using a three-position switch (2HS-2330) on panel 2C112. A drawing of the boron management system is shown on pages 46 and 47. The functions of each position of this handswitch are:

- position 1 - opens 2CV-2330A, closes 2CV-2330B.
- position 2 - closes both 2CV-2330A and 2CV-2330B
- position 3 - opens 2CV-2330B, closes 2CV-2330A.

2HS-2331 is installed on panel 2C112 under PC-92-8027. This is a two position handswitch with "BOTH" and ONE" positions. The functions of each of these positions are:

- "ONE" - valves operate as selected by 2HS-2330 (one at a time).
- "BOTH" - both discharge valves open simultaneously in 2HS-2330 positions 1 and 3.

Indicating lights are installed over 2HS-2331 to indicate 2HS-2331 switch position.

A radiation element (2RE-2330) is installed in the discharge line to automatically shut the discharge valves upon receipt of a high activity signal. If 2HS-2330 is in either position 1 or 3, a high discharge radiation signal will close the open valve(s) dependent on the position of 2HS-2333. The setpoint of the high discharge radiation signal is variable and is determined based on tank radionuclide concentration and Unit-1 circulating water flow. Control switch 2HS-2333 is located on panel 2C14 in the control room. It is a two position switch that is interlocked as follows:

- "OVERRIDE" permits the radiation high signal to be overridden in 2HS-2330 positions 1 & 3.
- "RESET" - any open valve(s) (2CV-2330A and/or 2CV-2330B) will shut when 2HS-2330 is in positions 1 or 3 <u>AND</u> any of the following occur:
  - radiation levels are NOT less than setpoint, or
  - a LRW discharge valve (from a 2T-21 tank) is OPEN along with a BMS discharge valve (from a 2T-69 tank).

2HS-2333 is normally maintained in the "RESET" position.

Flow transmitter 2FT-2330 is a 0 - 75 gpm flow transmitter that feeds 2FIC-2330 on panel 2C112. This flow indicator control may be used as a controller to position 2CV-2330A and/or 2CV-2330B based on discharge flow rate. If used, this would permit the operator to control discharge flow based on the available dilution flow obtainable in the Unit-1 circulating water discharge canal. However, this controller is normally maintained in "MANUAL" and not used for this function.

When a 2T-69 has been recirculated, sampled, and found acceptable for discharge; a discharge is performed per the BACT Release Form (OP 2104.014 Supplement 3).

ORIGINAL BANK QUESTION 362 WAS USED TO DEVELOP 2017 NRC EXAM QID#65 (Used on 2002 NRC Exam)

Questions For All QID In Exam Bank

Bank:	0362	**Rev:**	000	**Rev Date:**	1/10/2002 4:12:03	**QID #:**		**Author:**	Coble
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New		
Search		**10CFR55:**		**Safety Function**					
System Title:	Liquid Radwaste System	**System Number**	068	**K/A**	A3.02				
Tier:	2	**Group:**	1	**RO Imp:**	3.6	**SRO Imp:**		**L. Plan:**	
OBJ									

Description: Ability to monitor automatic operation of the Liquid Radwaste System including automatic isolation.

Question:

Given the following plant conditions:

* Unit 2 is at full power operation.
* A liquid release is in progress from Boric Acid Condensate Tank 2T-69A.
* Release Isolation 2CV-2330A is open, Release Isolation 2CV-2330B is closed.
* Control Room Release Handswitch 2HS-2333 on 2C14 is in position "RESET".
* Waste Condensate Tank (2T-21A) Pump 2P-53A Disch Valve (2CV-2122) is opened by WCO.

At this point, what would be the status of the 2T-69A release and the valve positions:

A. Release terminated, 2CV-2330A closed, 2CV-2122 closed.

B. Release terminated, 2CV-2330A closed, 2CV-2122 open.

C. Release in progress, 2CV-2330A open, 2CV-2122 closed.

D. Release in progress, 2CV 2330A open, 2CV-2122 open.

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☒

Answer:

B. Release terminated, 2CV-2330A closed, 2CV-2122 open.

Notes:

The interlock between a Boron Management System tank outlet valve and a Liquid Radwaste tank outlet valve will close release isolation 2CV-2330A to prevent inadvertent release of a non permitted tank. This makes answers C and D wrong. A is wrong because for the given conditions because the Waste Condensate Pump Discharge Valve will remain open.

References:

ANO-2-LP-WCO-BMS, Rev 07, Boron Management System, Objective 15
STM 2-52, Rev 05, LRW/BMS System Description, Sections 2.3.2.1, 3.5 and the LRW/BMS one line figures
E-2401, 2CV-2330 A and B electrical print.

Historical Comments:

Question 66

Bank:	2396	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	66	**Author:**	Cork		
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	Unit 1 NRC Exam Bank 0231				
Search	1940012213	**10CFR55:**	41.10	**Safety Function**							
Title:	Generic	**System Number**	GENERIC	**K/A**	2.2.13						
Tier:	3	**Group:**	1	**RO Imp:**	4.1	**SRO Imp:**	4.3	**L. Plan:**	ASLP-RO-OPSPR	**OBJ**	4

Description: Equipment Control - Knowledge of tagging and clearance procedures.

Question:

Which of the following conditions is correct in accordance with EN-OP-102 with regard to preparation and/or installation authorization of a common unit tagout?

A. Installation must be authorized by either the Unit 1 or the Unit 2 Operations Supervisor.

B. Preparer and reviewer from both units must be licensed operators.

C. Preparer and reviewer may be non-licensed if authorized by both Unit Operations Supervisors.

D. Preparer may be non-licensed as long as the opposite unit reviewer is licensed.

Answer:

B. Preparer and reviewer from both units must be licensed operators.

Notes:

B is correct in accordance with the criteria listed in EN-OP-102, Protective and Caution Tagging, Attachment 9.5 Tag Standards for the common unit tagout process. " The preparer shall be Licensed Operators on one Unit and the reviewer shall be Licensed Operator on the other Unit."

A is incorrect as both unit Operations Supervisors must authorize installation but plausible as only 1 supervisor is normally needed to authorize a tagout.

C is incorrect because the preparation & review must be done by licensed operators but plausible as both Unit Ops Supervisors must approve a common unit tagout.

D is incorrect because the preparation & review must be done by licensed operators on their respective units but plausible as non-licensed normally prepare tag outs.

This question matches the K&A as it requires the knowledge of the requirements for preparing and reviewing a common unit tagout IAW the tagging and clearance procedure.

References:

EN-OP-102 R018 OPS Protective and Caution Tagging Attachment 9.5 Tagout Standards
(Verified references updated 11/15/16)

Historical Comments:

Unit 1 NRC Exam Bank 0231 was used on the Unit 1 2007 NRC Exam
To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Added "/or" to the stem along with the procedure that requires the answer. Changed "may" to "must" in distractor A.

ATTACHMENT 9.5 **SITE SPECIFIC TAG STANDARDS**

SHEET 1 OF 10

1.0 Arkansas Nuclear One

1.1 Common Tagouts (ANO 1 and ANO 2)

NOTE

The determination of whether a TAGOUT should be considered a COMMON TAGOUT is based upon whether the SSC may normally be operated by either unit (both units train and qualify on the system). Examples are not all inclusive.

- The following examples should be considered a COMMON TAGOUT based upon common operator qualification:
  - Primary Hydrogen System
  - Generator Hydrogen System
  - Liquid Nitrogen System
  - T-41B
  - Cardox System (components tagged on both units)
  - Vendor Supplied Demineralized Water Trailers
- The following examples by the shared nature of the systems should be considered a COMMON TAGOUT:
  - Instrument Air cross-ties
  - Turbine Building Crane
  - Fuel Handling Crane (L-3 and 2L-35)
  - MCC B81 (power to both units condensate vacuum degasifiers

1.1.1 IF a Tagout is determined to be Common,
THEN Respond Yes to the COMMON TAGOUT Attribute.

1.1.2 IF a Tagout is determined to be Common,
THEN a LICENSED OPERATOR from each unit shall review it. (For example: The preparer shall be Licensed Operators on one Unit and the reviewer shall be Licensed Operator on the other Unit.)

AND

Both unit OPERATIONS SUPERVISORS must authorize installation. Opposite Unit Supervisors Should sign into the eSOMS Clearance module and select their name from list in the Opposite Unit Supervisor Attribute.

QID: 0231 **Rev:** 1 **Rev Date:** 11/20/00 **Source:** Direct **Originator:** J.Cork

TUOI: ASLP-RO-OPSPR **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of tagging and clearance procedures.

K/A Number: 2.2.13 **CFR Reference:** 41.10 / 45.13

ORIGINAL U1 NRC BANK QUESTION 0231 WAS USED TO DEVELOP 2017 NRC EXAM QID#66 (Used on the U1 2007 initial NRC Exam)

Tier:	3	**RO Imp:**	4.1	**RO Select:**	No	**Difficulty:**	2
Group:	G	**SRO Imp:**	4.3	**SRO Select:**	No	**Taxonomy:**	K

Question: **RO:** **SRO:**

Which of the following conditions is correct with regard to preparation and installation authorization of a common unit tagout?

A. Installation may be authorized by either the Unit 1 or the Unit 2 Operations Supervisor.

B. Preparers and reviewers from both units must be licensed operators.

C. Preparer and reviewer may be non-licensed if authorized by both Unit Operations Supervisors.

D. Preparer may be non-licensed as long as the opposite unit reviewer is licensed.

Answer:

B. Preparers and reviewers from both units must be licensed operators.

Notes:

Answer [b] is correct, procedure requires both the preparer and the reviewer on the unit preparing the tagout have to be licensed.
Answer [a] is incorrect, a common unit tagout requires both Unit's Operations Supervisors to approve it.
Answer (c) is incorrect, both Unit Ops Supervisors must approve but the preparation & review must be done by licensed operators.
Answer [d] is incorrect, the preparation & review must be done by licensed operators on their respective units.

References:

EN-OP-102, Rev. 13

History:

Developed for use in 98 RO Re-exam
Modified for use in 2001 RO/SRO Exam.
Selected for use on 2007 RO Exam.

Question 67

Bank:	2397	**Rev:**	1	**Rev Date:**	12/19/2016	**2017 TEST QID #:**	67	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	1940012144	**10CFR55:**	41.10	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.1.44				
Tier:	3	**Group:**	1	**RO Imp:**	3.9	**SRO Imp:**	3.8	**L. Plan:**	A2LP-RO-FH
OBJ	4/5								

Description: Conduct of Operations - Knowledge of RO duties in the control room during fuel handling, such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.

Question:

Given the following:

* The plant is in Mode 6 with refueling core alterations in progress.

During the core alterations, the neutron count rate should be observed each time a fuel assembly is ______________ the core and the status of the neutron count rate should be reported to the Refueling Bridge. If the neutron count rate cannot be communicated to the refueling bridge, then
____.

A. inserted into; suspend core alterations immediately until communications have been restored

B. inserted into; suspend core alterations after 1 hour if communications have not been restored

C. removed from; suspend core alterations immediately until communications have been restored

D. removed from; suspend core alterations after 1 hour if communications have not been restored

Answer:

A. inserted into; suspend core alterations immediately until communications have been restored

Notes:

A is correct as positive reactivity is added to the core when inserting a fuel assembly and counts must be verified stable at this point prior to release of the fuel assembly. TS 3.9.5 requires immediate suspension of core alterations if the direct communications between the control room and the refueling station is lost.

B is incorrect because the TS action does not allow any time to restore communications but plausible as the counts are recorded when the assembly is inserted and communications have to be verified within one hour prior to the start of core alterations IAW TS Surveillance 4.9.5.

C is incorrect as the Control room does not have to note the neutron count rate when removing positive reactivity from the core but plausible because this is the reverse action of inserting a fuel assembly and the TS required action for loss of communications is correct.

D is incorrect as the Control room does not have to note the neutron count rate when removing positive reactivity from the core but plausible as communications have to be verified within one hour prior to the start of core alterations IAW TS Surveillance 4.9.5.

This question matches the K&A because the RO must understand their duties to watch neutron count rate instrumentation during positive reactivity additions to the core and the immediate TS action to take if the RO cannot communicate this information to the refueling bridge.

References:

RF 2502.001 Refueling Shuffle Steps 10.8.7 and 10.8.8; (Verified references updated 11/15/16)
ANO-2 TS 3.9.5; (Verified references updated 11/15/16)
RF 2502.001 Refueling Shuffle Steps 10.7.7;
RF 2502.001 Refueling Shuffle step 8.6.4.

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Added the words "until communications have been restored" to the 2nd part of A and C. Changed the 2nd part of B and D to "suspend core alterations after 1 hour if communications have not been restored".

NOTE

Unexpected rise in neutron count rate is indicated by sustained doubling or continuously increasing count rate following completion of a change in core geometry.

10.8.7 WHEN inserting Fuel Assembly into the reactor core, THEN note count rate after insertion.

A. Log count rate on Item Control Area (ICA) Transfer Form (EN-NF-200, Att. 9.1).

B. IF unexpected rise in neutron count rate on in-service Startup detector or Auxiliary Incore detector occurs, THEN perform the following:

1. SRO in charge should consider removing the Fuel Assembly.

2. Obtain RCS boron sample and confirm refueling boron concentration requirements remain satisfied.

3. Do NOT proceed with Core Alterations until the following performed:

- Cause of unexpected rise in count rate is determined.

- Operations Manager and Reactor Engineering approval obtained to resume Core Alterations.

10.8.8 Bridge Operator perform the following:

A. Insert Fuel assembly and/or CEA IAW Operation Of Fuel Handling Equipment (2503.003)

B. Verify component is properly seated.

C. Notify Control Room communicator.

D. IF inserted in reactor core, THEN perform the following:

1. Report Hoist elevation to Control Room Communicator.

2. WHEN stable count rate confirmed, THEN ungrapple Fuel Assembly and/or CEA.

E. IF inserted in Spent Fuel Pool, THEN ungrapple Fuel Assembly and/or CEA.

10.8.9 Control Room communicator perform the following:

- Update Tag Board OR Computer Monitoring system.
- Initial and date Item Control Area (ICA) Transfer Form (EN-NF-200, Att. 9.1).
- IF inserted in reactor core, THEN record Reactor Core Hoist elevation reading.
- IF an overload/underload occurs while handling fuel in the reactor core, THEN log pertinent information on “Overload/Underload Log”, Attachment K of this procedure.

10.9 Repeat steps 10.7 and 10.8 IAW Item Control Area (ICA) Transfer Form (EN-NF-200, Att. 9.1) until all fuel assembly and control element assembly movements complete.

REFUELING OPERATIONS

COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

10.7.7 IF removing a Fuel Assembly,
THEN perform the following:

A. Bridge Operator remove fuel assembly IAW Operation of Fuel Handling Equipment (2503.003).

B. Notify Control Room Communicator.

C. IF Fuel Assembly in the reactor core,
THEN report Hoist elevation to Control Room Communicator.

10.7.8 Control Room Communicator perform the following:

- Update Tag board OR Computer Monitoring system.
- Initial and date the Item Control Area (ICA) Transfer Form (EN-NF-200, Att. 9.1).
- IF Fuel Assembly in the reactor core,
  THEN record Reactor Core Hoist elevation reading.
- IF overload/underload occurs while handling fuel in the reactor core,
  THEN log pertinent information on “Overload/Underload Log”, Attachment K of this procedure.

8.6.3 Within 2 Hours of start of fuel assembly or CEA movement:

- Water level over reactor vessel and/or spent fuel storage racks at normal operating level IAW the following:
  - Tech Spec 3.9.9, greater than 23 ft above Fuel seated in reactor vessel (389'-0.5")
  - T.S. 3.9.10, greater than 23 ft above Irradiated Fuel seated in storage rack (400'-1".) **[Satisfies 4.9.9]**

8.6.4 Within 1 Hour prior to start of CORE ALTERATIONS:

- Voice communications verified operational between Control Room, Reactor Refueling Area and Fuel Storage Area (if used) IAW Preparation For Refueling (2502.003).
- Document on Shift Turnover Checklist (Unit 2) OPS-B46. **[Satisfies 4.9.5]**

Performed by:____ Date:____________

Performed by:____ Date:____________

8.7 As a minimum, verify following qualified personnel IAW Refueling Transfer Process are on station and ready to commence fuel movement:

- One Control Room communicator.
- One Spent Fuel Bridge Operator qualified on Spent Fuel Handling Machine.
- One Reactor Fuel Bridge Operator qualified on Main Refueling Bridge.
- One active SRO in Charge of Fuel Handling.
- One Fuel Handling Supervisor stationed at Spent Fuel area.
- One Spotter for SFP area.
- One Spotter for Refueling Bridge.
- {4.3.1} **IF MANUALLY transferring fuel to/from Aux. Bldg THEN one fuel handling area Upender Operator.**
- {4.3.1} **IF MANUALLY transferring fuel to/from Aux. Bldg THEN One reactor building Upender Operator.**

Performed by: ___________________________________ Date______________

Question 68

Bank:	2398	**Rev:**	1	**Rev Date:**	12/15/2016	**2017 TEST QID #:**	68	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NRC Exam Bank1583		
Search	1940012215	**10CFR55:**	41.10	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.2.15				
Tier:	3	**Group:**	1	**RO Imp:**	3.9	**SRO Imp:**	4.3	**L. Plan:**	A2LP-RO-OPROC
OBJ	5								

Description: Equipment Control - Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tagouts, etc.

Question:

Which of the following describes the method of maintaining component configuration control when responding to an abnormal event using an AOP procedure?

A. The CRS keeps a handwritten list of components placed out of position and enters them in the component deviation log as time allows during the event and ensures the components are returned to normal prior to exiting the AOP.

B. Complete valve lineups for the affected systems are required to be performed after the event prior to exiting the AOP.

C. The AOP is reviewed by the CRS after the event to ensure that any equipment that was operated by the AOP procedure is returned to its required position or documented that it is out of its normal position.

D. The AOP procedures have restoration steps in them that will return all manipulated components to a normal configuration prior to exiting.

Answer:

C. The AOP is reviewed by the CRS after the event to ensure that any equipment that was operated by the AOP procedure is returned to its required position or documented that it is out of its normal position.

Notes:

C is correct. During normal plant evolutions, configuration control is maintained by the normal methods of component deviation log, Tagging sheets, etc. However, during emergency situations, due to the importance of timely EOP/AOP execution, it is NOT OPS management’s expectation that every component manipulation directed by EOP/AOP be documented in component deviation log, Station log, etc. However, to ensure that configuration control is regained at conclusion of an event, the EOP/AOP is reviewed step by step by the CRS to ensure that any equipment that was operated by procedure is returned to its required position or documented in its out of normal position. The normal configuration controls and emergency configuration controls are normally updated and reviewed by the CRS.

A is incorrect as the CRS is not required to keep a handwritten list of components placed out of position during the AOP event, but plausible that the CRS would try to annotate the components in the procedure to review at a later time after the event prior to exiting the procedure.

B is incorrect as a complete lineup of the affected systems is not required but plausible as this would ensure the components in the system have been returned to normal.

D is incorrect as AOP and EOP procedures do not have restoration steps in them for manipulated components but plausible as the components would be returned to normal if they di have restoration steps.

This question matches the K&A as the applicant must have the knowledge of how configuration control is maintained

using the AOP/EOP procedure documents after an abnormal event occurs.

References:

OP 1015.021, ANO-2 EOP/AOP Users Guide, Rev. 15 Step 9.1.6

Historical Comments:

NRC Exam Bank 1583 was used on the 2008 NRC Exam
To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Original K/A 2.2.41 was rejected based on feedback from the NRC Chief Examiner that this K/A was more applicable to an ADMIN JPM and that the question was GFE Knowledge. Therefore K/A 2.2 15 was randomly selected from the 2.2 set of generic K/As.

9.1.6 QUESTION 6

How is configuration control maintained during EOP/AOP events?

ANSWER:

Due to the importance of timely EOP/AOP execution, it is not OPS management's expectation that every component manipulation directed by EOP/AOP be documented in COOP log, Station log, etc. However, to ensure that configuration control is regained at conclusion of an event, the following items should be reviewed to ensure status of equipment is identified and controlled.

A. Annunciator Corrective Actions (ACAs) are reviewed and actions taken or component out of normal status is recorded for any alarm(s) still locked in at end of event (governing procedure meets exit conditions). This will ensure that the component(s) returned to normal configuration or the abnormal status is recorded.

B. Review EOP/AOP step by step to ensure that any equipment that was operated by procedure is returned to its required position or documented in its' out of normal position.

C. At direction of SM/CRS all or portions of applicable shift turnover checklist should be performed and the status of any component(s) found out of position is resolved.

D. Any procedure deviations incurred during procedure implementation should be documented in station log and dispositioned via condition reporting system at this time.

EXAMPLE:

During performance of Natural Emergencies AOP due to a thunderstorm warning in Johnson County, exit conditions were met when thunderstorm warning was lifted. No actions were taken during the course of execution of AOP and SM should then determine that no portions of shift turnover checklist need be completed, but Natural Emergencies AOP should be reviewed step by step and a review of annunciators should also be completed.

During performance of Loss of Instrument Air AOP, certain manual valves were manipulated to isolate and repair ruptured Instrument Air tubing. Prior to exiting AOP, the AOP should be reviewed step by step to identify component manipulation(s) and either document abnormal position or restore component(s) to normal configuration. Annunciator(s) should be reviewed and actions taken to clear alarm(s) or document component(s) abnormal condition and SM should direct all or portions of applicable shift turnover checklist be performed on control room panels were component manipulation(s) performed.

11.0 INTERIM CONFIGURATION CONTROL

11.1 IF components are repositioned prior to completing a Configuration Control Record due to an emergency or prudent actions, THEN perform the following:

11.1.1 Use Attachment 3 of this procedure to track the repositioned components.

11.1.2 Transfer configuration control to an approved method in a timely manner following stabilization of the situation.

- IF the Attachment 3 entry is for equipment or personnel protection, THEN a Tagout should be hung.

11.1.3 Initiate a Condition Report documenting the incident.

ATTACHMENT 3

COMPONENT DEVIATION LOG

REMOVAL				RESTORATION OR MOVEMENT TO CONFIGURATION CONTROL METHOD					
COMPONENT	REASON	SRO APPROVAL	DATE/TIME PERFORMED	SRO APPROVAL	DATE/TIME PERFORMED	REQUIRED POSITION (Note 1)	CONFIG METHOD (IF NOT RESTORED) (Note 2)	RESTORED BY (IF RESOTRED)	VERIFIED BY (IF RESTORED)

Note 1: Required position is obtained from valve or breaker lineup or other approved document.
Note 2: If the component entry is for equipment or personnel protection, then a Tagout should be hung.

ORIGINAL BANK QUESTION 1583 WAS USED TO DEVELOP 2017 NRC EXAM QID#68 (Used on the 2008 NRC Exam)

Data for 2008 NRC SRO Exam

Bank: 1583	**Rev:** 0	**Rev Date:** 11/21/2007 2:51:0	**QID #:** 69	**Author:** Coble			
Lic Level: S	**Difficulty:** 2	**Taxonomy:** F	**Source:** IH Bank ANO-OpsUnit2-09444				
Search 1940012215	**10CFR55:** 43.3 / 45.13	**Safety Function**					
System Title: Generic	**System Number** GENERIC	**K/A** 2.2.15					
Tier: 3	**Group:** 1	**RO Imp:** 2.2	**SRO Imp:** 2.9	**L. Plan:** ASLP-RO-OPSPR	**OBJ** 4.i.3.d		

Description: Equipment Control - Ability to identify and utilize as-built design and configuration change documentation to ascertain expected current plant configuration and operate the plant.

Question:

Which of the following describes the method of maintaining component configuration control when responding to a SG tube leak event?

A. The CRS keeps a handwritten list of components placed out of position and enters them in the COOP Log as time allows during the event.

B. Complete valve lineups for the affected systems are required to be performed after the event.

C. The Primary to Secondary Leakage AOP, 2203.038, is reviewed by the CRS after the event to ensure any equipment operated is returned to normal or documented in the proper log.

D. The Primary to Secondary Leakage AOP, 2203.038, has proper restoration steps in it to return all manipulated components to a normal configuration.

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☑	☑

Audit Exam History

2008	☐

Answer:

C. Correct

Notes:

During normal plant evolutions, configuration control is maintained by the normal methods of COOP log, Tagging sheets, etc. However, during emergency situations, due to the importance of timely EOP/AOP execution, it is NOT OPS management's expectation that every component manipulation directed by EOP/AOP be documented in COOP log, Station log, etc.
However, to ensure that configuration control is regained at conclusion of an event, the EOP/AOP is reviewed step by step by the CRS to ensure that any equipment that was operated by procedure is returned to its required position or documented in its' out of normal position. The normal configuration controls and emergency configuration controls are normally updated and reviewed by the CRS.

References:

OP 1015.021, ANO-2 EOP/AOP Users Guide, Step 9.1.6

Historical Comments:

Question 69

Bank:	2399	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	69	**Author:**	Foster
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	1940012103	**10CFR55:**	41.10	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.1.3				
Tier:	3	**Group:**	1	**RO Imp:**	3.7	**SRO Imp:**	3.9	**L. Plan:**	ASLP-RO-COPD
OBJ	15								

Description: Conduct of Operations - Knowledge of shift or short-term relief turnover practices.

Question:

Consider the following:

* Unit 2 is operating at 100% power.
* The Unit 2 Control Room Operating crew consists of the following:
  - A Shift Manager who holds a SRO License
  - A CRS who holds a SRO License
  - Three (3) Control Board Operators licensed as ROs
  - A STA who does not have a license

* The on watch At The Controls (ATC) operator has to leave due to an Emergency at home.

With the watchstander leaving due to an emergency, the Unit Staff ______ meet(s) the Shift Manning Requirements of EN-OP-115, Conduct of Operations, and IF the minimum Shift Manning is NOT met at any time the crew has a MAXIMUM requirement of _____ hours to reestablish minimum Shift Manning.

A. does not; 1.5

B. still ; 1.5

C. does not; 2

D. still; 2

Answer:

D. still; 2

Notes:

D is correct. EN-OP-115, Conduct of Operations, Rev. 17, Attachment 9.1 requires that only 2 RO licensed operators are require to meet the minimum shift manning and three are on watch.

A is incorrect because the maximum time allowed is 2 hours and minimum shift manning is still met but plausible as the requirement is within the 2 hour maximum and The 1.5 hour time is the time required to respond to a page to staff the emergency response organizations during an Alert or Higher declared E-plan event.

B is incorrect because the maximum time allowed is 2 hours but plausible as the requirement is within 2 maximum and immediate action is taken to restore the crew shift compliment thus the restoration could occur prior to the 2 hour limit. Also plausible as the 1.5 hour time is the time required to respond to a page to staff the emergency response organizations (ERO) during an Alert or Higher declared E-plan event.

C is incorrect because minimum shift manning is still met but plausible as this is the correct time frame to fill an unexpected absence in an emergency if needed.

This question matches the K&A because it requires knowledge of short-term relief turnover practices.

References:

EN-OP-115-03, Shift Turnover and Relief, Rev. 2, section 5.2 [1] and 5.2[2] ;
(Verified references updated 11/15/16)
T.S. 6.2.2.b and c., Unit Staff, Rev. 296; (Verified references updated 11/15/16)
COPD001, (rev 070) Operations Expectations and Standards, Section 5.16 Shift Turnover, Step 5.16.1. D.(Verified references updated 11/15/16)

Historical Comments:

NRC Exam Bank 1915 was used on the Unit 2 2012 NRC Exam.
To be used on the 2017 NRC Exam. Altered the time in A and B to be more plausible and changed the order of the answer from A to C.

REV. 1 based on NRC Chief Examiner Feedback BNC. Revised the question to delete the requirement to get permission to leave during an emergency.

[12] At the conclusion of their respective shift turnovers, the CRS, ATC and BOP shall inform the rest of the CCR staff that they have assumed their assigned watch stations. When turnover occurs during a period of high activity, this may be communicated by an update brief.

5.2 WATCH STATION RELIEF

[1] In the case of illness or unexpected absence, SM should hold a shift member over and arrange for replacement personnel to restore the shift complement within two hours.

[2] In the case where a control room operator needs to be relieved during their shift, the following must be performed:

(a) Permission granted by the SM or CRS as applicable

(b) A verbal turnover conducted with a qualified individual as follows:

(1) If the on-coming operator attended the beginning of shift briefing or not discuss applicable items from the list below:

- Current Plant Status (including any anticipated changes in reactivity status)
- Any significant changes that occurred since the beginning of the shift
- Procedures or evolutions in progress including continuous action steps.
- Parameters being monitored and the frequency of monitoring by the person exiting the Control Room.
- Abnormal system lineups/ equipment out of normal alignment
- Defeated interlocks
- Existing or potential plant or equipment problems
- Any issues or concerns outside of normal conditions
- Equipment problems the relief operator needs to be aware of
- Compensatory actions in effect that may need to be performed

(c) An update brief performed informing the shift crew of the relief change

6.2.2 UNIT STAFF

a. A non-licensed operator shall be on site when fuel is in the reactor and two additional non-licensed operators shall be on site when the reactor is in MODES 1, 2, 3, or 4.

b. The minimum shift crew composition for licensed operators shall meet the minimum staffing requirements of 10 CFR 50.54(m)(2)(i) for one unit, one control room.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) for one unit, one control room, and 6.2.2.a and 6.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. The operations manager or the assistant operations manager shall hold a SRO license.

f. When in MODES 1, 2, 3, or 4 an individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operations of the unit. This individual shall meet the qualifications specified by ANSI/ANS 3.1-1993 as endorsed by RG 1.8, Rev. 3, 2000.

5.15.2 Discussion

Dropped chart recorders have resulted in a Reactor Scram at River Bend and partial ESFAS actuation at ANO2. Anticipate the worst when removing any component from a panel and manage defenses appropriately.

5.16 Shift Turnover

5.16.1 Additional Standards and Expectations

A. An Operator shall not assume the watch unless the following conditions are met:

- Any new medical condition or medication has been reviewed by the MRO or designee. (Contact MRO/designee prior to assuming the watch) Shift Manager observations (aberrant or abnormal behavior, or appearance of drowsiness) shall also be considered.
- Operator is prepared to promptly don a respirator by ensuring the following:
  - Respirator Quals are current
  - Face is clean shaven
  - If individual has a corrective eyewear restriction in their license, then appropriate SCBA eyewear (spectacles or contact lenses) readily available
- Maintain DLR qualifications and have possession of a DLR during watch standing duties.
- If assigned to the fire brigade, qualified to enter the Auxiliary Building AND qualified Fire Brigade.

B. The normal method for verification of watch qualification requirement is by the use of eSOMS Narrative Log.

- Oncoming watch relieves the off-going watch in Narrative Log prior to off-going watch leaving.
- <u>IF</u> the eSOMS Narrative Log is not available, <u>THEN</u> alternate checks will be required to ensure qualifications are current prior to watch relief and can be accomplished by reviewing the following as appropriate:
  - Supervisor Review of Crew Training Requirements reports
  - Watch Station Proficiency report
  - Approved qualification database

C. If it is obvious a person is not capable of assuming watch duties, then the Shift Manager shall be apprised of relief personnel reporting for work under the apparent influence of alcohol or drugs, obviously physically ill, or in such a psychological or emotional state that their effectiveness may be impaired.

D. The minimum shift complement requirements delineated in EN-OP-115 and Tech Specs shall NOT be violated except as stated below:

- When unexpected incapacitation or other loss of on duty crew members occurs, then shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours, provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

E. Prior to normal relief time, On-Duty crew members shall complete the applicable portions of the Shift Relief Sheet.

F. Prior to turning over duties to the oncoming shift personnel, the on-duty personnel should apprise their relief of the following:

- Overall plant status
- Procedures or evolutions in progress including Continuous Action Steps, parameters being monitored and frequency, and individual responsible
- Existing or potential problems
- Pertinent activities conducted during the past shift
- Schedule requirements or plans
- Abnormal system lineups
- Key lock functions or interlocks defeated
- Pertinent procedure changes impacting plant operations or activities in progress
- New or revised Night Orders and applicable special instructions issued since their last shift
- Any existing LCOs in effect
- Out of Spec log readings

ORIGIONAL NRC BANK QUESTION 1915 WAS USED TO DEVELOP 2017 NRC EXAM QID#69 (Used on the 2012 initial NRC Exam)

Questions For All QID In Exam Bank

Bank: 1915 **Rev:** 0 **Rev Date:** 5/7/2012 1:36:06 P **QID #:** 94 **Author:** Foster

Lic Level: S **Difficulty:** 3 **Taxonomy:** F **Source:** NEW

Search 1940012103 **10CFR55:** 41.10 / 43.1 **Safety Function**

System Title: Generic **System Number** GENERIC **K/A** 2.1.3

Tier: 3 **Group:** 1 **RO Imp:** 3.7 **SRO Imp:** 3.9 **L. Plan:** ASLP-RO-COPD **OBJ** 15

Description: Conduct of Operations - Knowledge of shift or short-term relief turnover practices.

Question:

Consider the following:

- Unit 2 is operating at 100% power
- The on watch At The Controls (ATC) operator had to leave due to an Emergency at home

With the watchstander leaving due to an emergency, he/she ______ required to ask permission from the Shift Manager prior to leaving and the Shift Manager has _____ hours to replace the ATC.

A. is; 2

B. is not; 2

C. is; 6

D. is not; 6

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☒

Answer:

A. Correct

Notes:

B, C, and D are Incorrect, per EN-OP-115-03, section 5.2[1] states the 2 hour time frame (along with T.S. 6.2.2) section 5.2[2] and 5.2 [3] requires permission [page 6] (as state in [3] Shift Turnover and Relief, personnel who are required as part of shift complement should notify the SM before leaving the site and request permission to leave)

References:

EN-OP-115-03, Shift Turnover and Relief, Rev. 0, section 5.2 [1] and 5.2[2] and5.2[3] (pages 6 and 7)
T.S. 6.2.2.b and c., Unit Staff, Rev. 48
ASLPRO-COPD, Rev. 9, Obj. 15: For EN-OP-115, Conduct of Operation and COPD-001, Conduct of Operations: describe the management expectations and responsibilities of Operational Supervisors (SRO)

Historical Comments:

New Question

Question 70

Bank:	2400	**Rev:**	2	**Rev Date:**	12/19/2016	**2017 TEST QID #:**	70	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NRC Exam Bank 0405		
Search	1940012445	**10CFR55:**	41.10	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.4.45				
Tier:	3	**Group:**	1	**RO Imp:**	4.1	**SRO Imp:**	4.3	**L. Plan:**	A2LP-RO-PANN
OBJ	2								

Description: Emergency Procedures/Plan - Ability to prioritize and interpret the significance of each annunciator or alarm.

Question:

Given the following:

* A Plant transient has occurred causing several annunciators to come in.

In accordance with the ANO-2 EOP/AOP Users Guide, 1015.021, the correct order to address and prioritize plant annunciators based on their color coding from the highest priority to the lowest priority would be:

A. Red, Green and White

B. Red, Amber and Green

C. Red, White and Green

D. Red, Amber and White

Answer:

A. Red, Green and White

Notes:

A is correct: ANO has three colors to code different annunciators based on their significance. Red (High Awareness) would be the highest priority to due the safety significance of the alarm. Green (Medium Awareness) would be next and White (General Awareness) would be last.

B is the incorrect order but plausible as the distractor contains two annunciator colors and one amber color.

C is the incorrect order but plausible as the distractor contains the three correct colors.

D is the incorrect order but plausible as the distractor contains two annunciator colors and one amber color.

This question matches the K&A as it requires the candidate to have the knowledge of the EOP/AOP Users Guide for annunciator usage to have the ability to interpret the significance of annunciators based on their color and prioritize which ones to address first. Next, etc.

References:

Admin 1015.021 ANO-2 EOP-AOP Users Guide Rev. 15 Step 7.1
(Verified references updated 11/15/16)

Historical Comments:

NRC Exam Bank 0405 was used on the 2002 NRC Exam
To be used on the 2017 NRC Exam but altered the answers from the last time the QID was given.

REV. 1 based on NRC Chief Examiner Feedback BNC. Revised the distractor to always have Red as the highest priority and other colors that typically imply warning or caution and are colors used to access plant risk.

REV. 2 based on NRC Chief Examiner Feedback BNC. Changed "Yellow" to "Amber" in distractor B and "Orange" to "Amber" in distractor D.

7.0 ANNUNCIATOR USAGE

7.1 The Unit 2 Control Room annunciators are prioritize by Color Coding for operator awareness. The color-coding is used to assist in operator recognition and prioritization, but does not dictate that actions must immediately follow. Operators are expected to be aware of the alarm, evaluate the alarm, and make a conscious decision on when and if actions are necessary. Actions to be taken in response to alarms will vary with equipment status, type of event in progress, etc. The choice of the appropriate action to take in response to a color-coded annunciator is left to the discretion of the CRS or S/M. Three colors are used to designate annunciator awareness. The following colors will designate awareness groupings:

- RED - HIGH AWARENESS

  Annunciators where the operator should immediately evaluate the need for response. RED annunciators have the potential to impact unit safety, unit availability, or safety system operation.

- GREEN - MEDIUM AWARENESS

  Annunciators where the operator should evaluate the need for prudent action. These annunciators have a potential of developing into a RED annunciator, involve major process equipment trouble, or have the potential for a radiation release.

- WHITE - GENERAL AWARENESS

  Annunciators where the operator should evaluate the need for timely action. These alarms would be addressed as time permits during an event as directed by the CRS or S/M.

7.2 Annunciator Corrective Actions are a lower tiered procedure type (below abnormal/emergency operating procedures) per CEN-152 [Combustion Engineering Owners Group (CEOG) Emergency Procedure Guidelines] FIGURE 1-2, Sequence of Decisions For Off-Normal Operations.

When implementing an abnormal or emergency operating procedure, the annunciator corrective action for alarms received during event should be reviewed under these criteria:

- As time allows.
- At Shift Turnover.
- Before exiting EOP/AOP.

ORIGINAL BANK QUESTION 405 WAS USED TO DEVELOP 2017 NRC EXAM QID#70 (Used on the 2002 initial NRC Exam)

Questions For All QID In Exam Bank

Bank:	0405	**Rev:**	001	**Rev Date:**	1/17/2002 8:10:43	**QID #:**		**Author:**	Coble
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New		
Search		**10CFR55:**		**Safety Function**					
System Title:	Emergency Procedures/Plan	**System Number**		**K/A**	2.4.31				
Tier:	3	**Group:**	2.4	**RO Imp:**	3.3	**SRO Imp:**		**L. Plan:**	
OBJ									

Description: Knowledge of annunciator alarms and indications, and use of the response instructions.

Question:

Given the following:

* A Plant transient has occurred causing several annunciators to come in.

In accordance with the EOP/AOP Users Guide, 1015.021, the correct order to address and prioritize the annunciators based on their color coding would be:

A. Green, White, and Red

B. Red, Green, and White

C. Red, White and Green

D. Green, Red, and White

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☒

Answer:

B. Red, Green, and White

Notes:

ANO has three colors to code different annunciators based on their significance. Red (High Awareness) would be the highest priority to due the safety significance of the alarm. Green (Medium Awareness) would be next and White (General Awareness) would be last.

References:

ANO-2-LP-RO-EAOP, Revision 5, Objective 1
OP 1015.021, ANO-2 EOP/AOP User Guide, Revision 004-02-0, Step 7.0 Annunciator Usage

Historical Comments:

1/10/2002. Removed the color yellow from distracter A. Added "In accordance with the EOP/AOP Users Guide, 1015.021," to the stem based on NRC feedback. 1/17/2002 - Removed the color yellow from all distracters and made the distracters a combination of red white and green to be more credible based on NRC feedback.

Question 71

Bank:	2401	**Rev:**	1	**Rev Date:**	1/12/2017	**2017 TEST QID #:**	71	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	1940012314	**10CFR55:**	41.12	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.3.14				
Tier:	3	**Group:**	1	**RO Imp:**	3.4	**SRO Imp:**	3.8	**L. Plan:**	A2LP-RO-ESGTR
OBJ	1								

Description: Radiological Controls - Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Question:

Given the following:

* The plant has been tripped due to a Steam Generator Tube Rupture.
* Actions in Standard Attachment 19, Control of Secondary Contamination, have been completed.

After Standard Attachment 19 actions are taken, the ______________________________ would still be a contamination hazard and the __________________________ would continue to be a rising radiation source hazard.

A. Running Condensate pump coffer dams; SU/BD demineralizers

B. Running Condensate pump coffer dams; Condensate Inlet Filter 2F-807

C. EFW Pump 2P-7A bearing oil cooling water leak off; SU/BD Demineralizers

D. EFW Pump 2P-7A bearing oil cooling water leak off; Condensate Inlet Filter 2F-807

Answer:

A. Running condensate pump coffer dams; SU/BD demineralizers

Notes:

A is correct as the coffer dams on the condensate pumps are open to atmosphere and tend to spray/splash water around the running pumps and the condensate water will be full of radioactive particulate. The Start up/Blowdown Demineralizers will be in service filtering the condensate/blowdown and will become a radiation hotspot.

B is incorrect as the Condensate filter 2F-807 is bypassed and isolated in Standard Attachment 19, Control of Secondary Contamination, but plausible as it is a filter and would become a hot spot if not bypassed and isolated.

C is incorrect as the bearing oil cooling water leak off comes from the suction of the 2P-7A pump which is taking a suction off of the in-service Condensate Storage Tank CST which will be clean but plausible as any steam leaks that comes from the steam flowing through the turbine will be a potential contamination hazard.

D is incorrect as the bearing oil cooling water leak off comes from the suction of the 2P-7A pump which is taking a suction off of the in-service Condensate Storage Tank CST which will be clean but plausible as any steam leaks that comes from the steam flowing through the turbine will be a potential contamination hazard. Also the Condensate filter 2F-807 is bypassed and isolated in Standard Attachment 19, Control of Secondary Contamination, but plausible as it is a filter and would become a hot spot if not bypassed and isolated.

This question matches the K&A as it requires knowledge of where contamination hazards may arise during a SGTR EOP event.

References:

EOP 2202.004 SGTR Rev. 14 Step 16 B; (Verified references updated 11/15/16)
Standard Attachments 2202.010 Rev 23. Attachment 19 Control of Secondary Contamination; (Verified references updated 11/15/16)
STM_2-20_17-1 Condensate SYS Section 2.3.1; (Verified references updated 11/15/16)
STM_2-19-2_39-1 EFW and AFW SYS Section 2.1.1.2. (Verified references updated 11/15/16)

Historical Comments:

To be used on the 2017 NRC Exam

Rev. 1 based on post submittal validation comments: Added the word "rising" prior to radiation source hazard.

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

SG with highest leakage or activity is considered the ruptured SG.

15. Determine ruptured SG by comparing ANY of the following:

A. Secondary Systems Radiation Trend recorder (2RR-1057).

B. Main Steam Line Radiation Monitors:

- 2RI-1007
- 2RI-1057

C. SG Sample Radiation Monitors:

- 2RITS-5854
- 2RITS-5864

D. SG Tube Leak N-16 monitor history trends.

E. SG levels.

1) Level rising faster in ONE SG with similar FW flow rates and steaming rates in BOTH SGs.

2) Rising SG level with ALL FW isolated.

F. Steam flow and FW flow prior to Reactor trip.

G. SG water sample results.

■ **16. Minimize secondary contamination by performing BOTH of the following:**

A. Commence isolation of ruptured SG by performing local actions ONLY of 2202.010 Attachment 10, SG Isolation.

B. 2202.010 Attachment 19, Control of Secondary Contamination.

ATTACHMENT 19
CONTROL OF SECONDARY CONTAMINATION

1. Initial Actions

   A. Notify non Control Room watchstanders to obtain dosimetry for secondary plant elevated radiological conditions.

   B. Make the following plant announcement:

   "Attention all personnel, attention all personnel, Unit 2 has indications of primary tube leakage on the "___" Steam Generator. The potential for changing radiological conditions exists in the Unit 2 Turbine Building and adjacent areas. Non-essential personnel should exit and remain clear of these areas until further notice."

   C. Repeat the announcement.

NOTE

Actions for steps D through K may be performed in any order.

   D. Contact Chemistry to commence sampling Turbine Building Sump for activity.

   E. Notify Radiation Protection to perform "Primary (RCS) To Secondary Leak" attachment of Unit 2 RP Off-Normal Operations (1601.308).

   * F. Monitor Turbine Building Sump for other causes of rising level.

CAUTION

Regen Waste Tank Dumps, Turbine Building Sump Dumps or hotwell rejects to the Flume are considered radioactive liquid releases in this situation.

   G. Isolate uncontrolled Turbine Building Sump release by verifying the following Turbine Building Sump Pump Control switches in OFF:

   - 2P42A (2HS-4300)
   - 2P42B (2HS-4301)
   - 2P42C (2HS-4302)
   - 2P42D (2HS-4303)
   - 2P96 (2HS-4304)

(Step 1 continued on next page)

1. (continued)

H. At SG Secondary Sample panel (2C377), place the following handswitches in OFF:

- "2HS-5933"
- "2HS-5935"

I. Isolate Corrosion Product Sampling (CPS) system drains to Sump Station No. 1 by closing the following valves: (valves located in 2T-21 Tank Room)

•2E-24A Blowdown Line Drain/Corrosion Product Sample (2SGS-1010)

•2E-24B Blowdown Line Drain/Corrosion Product Sample (2SGS-1038A)

J. Locally check the following systems for radioactive release paths and report ANY leaking components to SM:

- Main Steam
- Auxiliary Steam
- Condensate
- Feedwater
- Startup and Blowdown Demin
- Emergency Feedwater
- Auxiliary Feedwater

K. Submit WR/WO to align 2C377 (SG Sample Panel) drain piping spectacle flange to Neutralizing tank (2T87).

2. Long Term Actions

   A. Isolate Generator Hydrogen Cooler Continuous Vents using 2104.030, Auxiliary Cooling Water System Operations.

   B. Complete 2104.023 Attachment B, Turbine Building Sump Release Permit for any Turbine Building sump releases.

   C. Verify the following filters bypassed and isolated using 2106.024, SU & Blowdown Demineralizer Ops:

      - Condensate Inlet filter (2F-807)
      - SG BD filter (2F-808)

NOTE

Normal Flow Service Alignment is condensate flow at approximately 2200 gpm and Steam Generator blowdown flow aligned to 2T-94A.

   D. Coordinate with Chemistry to determine need to place SU/BD DI 2T94A in Normal Flow Service Alignment using 2106.024, SU & Blowdown Demineralizer Ops.

   E. Maintain Hotwell level by performing the following as necessary:

      - Isolate Hotwell makeup.
      - Reject to 2T92 Tanks via "Service Rinse of Demin Tanks (2T-94A/B)" section of 2106.024, SU and Blowdown Demineralizer Ops.
      - Pumping Hotwell to the flume using 2106.016, Condensate and Feedwater Operations.

   F. IF releasing steam from ADVs, MSSVs, or EFW Pump Turbine (2K3),
   THEN perform the following:

      - Notify Radiation Protection of potential for contaminating ventilation intake filters.
      - Notify Radiation Protection to monitor for airborne contamination.
      - Notify Chemistry to monitor or estimate offsite dose release.

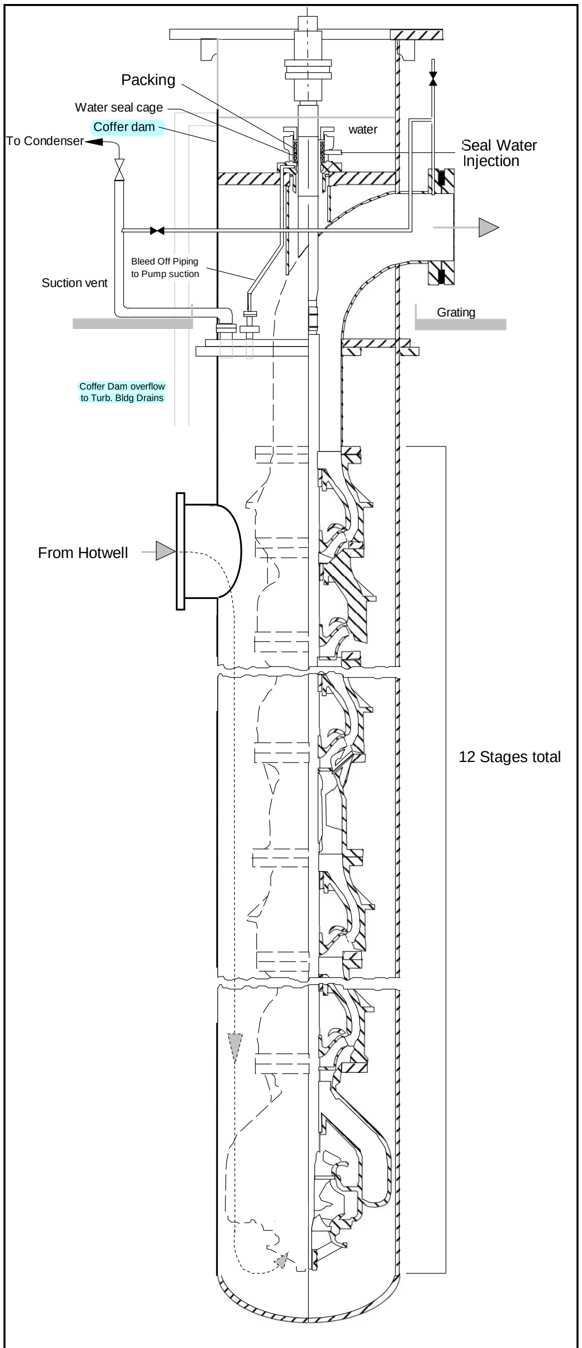
   G. Check SG sample panel drains aligned to Neutralizing tank (2T87).

at 1475 ft. total discharge head (TDH) with an admin limit of 8000 gpm. Pump operation above 8000 gpm is acceptable for short periods of time, but will shorten the life of the motor. (Maintain pump discharge pressure greater than 630 psig to ensure this limit is not exceeded.).

The limiting pump flow is 9000 gpm due to NPSH considerations. The pumps are rated for 8250 gpm at 1475 ft. total discharge head (TDH). The design discharge pressure for each of the pumps is 650 psi.

2.3.1 Construction

The Condensate pumps are twelve stage enclosed impeller pumps. The impellers in each stage are enclosed type, one piece construction, and are dynamically balanced. (For larger drawing of a typical pump packing area, refer to page 43).



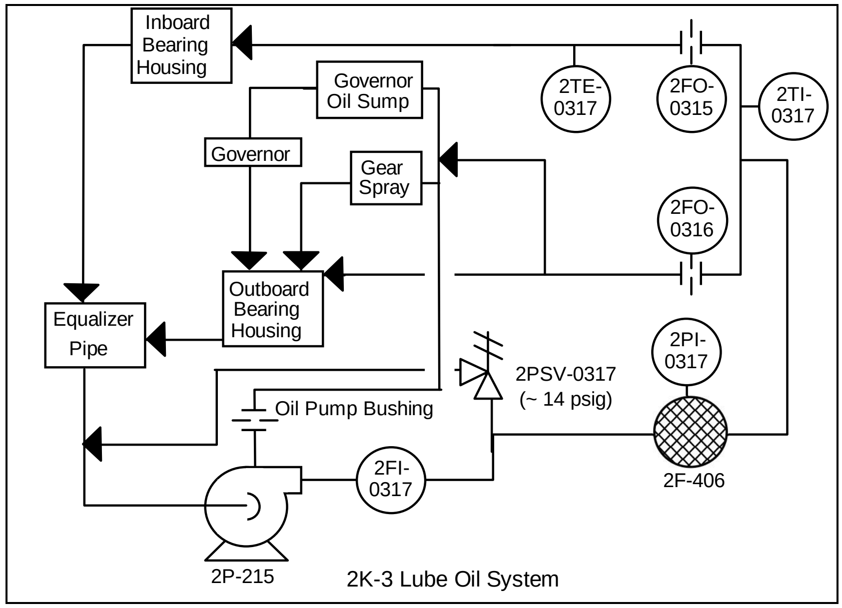
CONDENSATE PUMP

The pump bearings are lubricated by the condensate flowing within the pump. Gland seal (packing) lubrication is provided during startup by the Condensate Storage tank and by the discharge of the Condensate pump, via the Seal header, during normal operations. The packing Bleed-off line is routed back to the suction area of the pump casing. This is to vent the packing area to ensure that the packing will not be exposed to pump discharge pressure. Higher than normal seal pressures with the pump running could be indicative of excessive packing leakoff. Packing leakoff is routed to the Turbine Building Sump at the base of the pump and is monitored by the Auxilliary Operator during rounds to prevent excessive condensate makeup requirements.

The Condensate pumps' upper motor bearings are oil lubricated thrust/guide bearings. The oil is contained in the housing around the bearing. A local sight glass is provided to monitor the oil level. The thrust bearing located on the bottom of the housing acts as a centrifugal pump and provides circulation of oil. The oil flows over the cooling coils immersed in the oil reservoir. The cooling coils are supplied by Component Cooling Water (CCW). The upper guide bearing is located above the thrust bearing. The lower bearing is a radial guide bearing which is lubricated by grease.

The Condensate pumps take suction on the condenser hotwells via two 34" lines. Each 34" suction header supplies two Condensate pumps; 2P2A and 2P2C share one 34" suction header while 2P2B and 2P2D share the other suction header. The two 34" suction headers are cross connected between the condenser hotwells and the Condensate pumps via an unisolable 24" line.

Each pump is equipped with a suction strainer designated 2F-35A, 2F-35B, 2F-35C, and 2F-35D respectively. Each strainer is equipped with a differential



2.1.1.2 Turbine Driver Lube Oil System

The 2K-3 Lube Oil System is comprised of:

- a Turbine Bearing Oil Pump (2P-215)
- an Equalizer Pipe
- Inboard and Outboard Bearing Housings
- an Oil Supply Filter (2F-406)
- an Oil Pump Discharge Relief (2PSV-0317) and
- associated instrumentation

The lube oil is supplied to the inboard and outboard turbine bearings, the speed governor system and the oil pump bushing.

The oil pump, 2P-215, is a positive displacement pump that is driven from the turbine shaft.

The lube oil pressure through the system is controlled by 2PSV-0317, which has a setting of 14 psig.

Each turbine bearing has a reservoir that holds approximately one gallon of oil. Level indication is provided by 2LI-0315, on the inboard bearing reservoir.

The cooling of the oil is accomplished by a U-tube, finned heat exchanger located in the reservoir for each bearing. The cooling water flowing through the tube side is supplied from the suction of the EFW pump via solenoid operated globe valve 2SV-0317-2. When the Emergency Feedwater pump is started, solenoid valve 2SV-0317-2 is de-energized, causing it to open. The water discharging from the tube side of the heat exchanger is routed to the floor drain next to door into the room. 2SV-0317-2 is powered from 2D24-01.

Lube oil is also supplied to the turbine speed governor. The lube oil provides the hydraulics for the operation of the Governor Valve, 2CV-0332. For quicker governor response on startup, a Governor Oil Sump (2M-153) maintains oil available for immediate use. The oil is returned to the outboard bearing housing.

The lube oil is also used to lubricate the turbine shaft gears that drive the Electro-Hydraulic Valve Actuator (EGR) and the Tuthill lube

Question 72

Bank:	2402	**Rev:**	2	**Rev Date:**	12/19/2016	**2017 TEST QID #:**	72	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	1940012414	**10CFR55:**	41.10	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.4.14				
Tier:	3	**Group:**	1	**RO Imp:**	3.8	**SRO Imp:**	4.5	**L. Plan:**	A2LP-RO-EAOP
OBJ	40								

Description: Emergency Procedures/Plan - Knowledge of general guidelines for EOP usage.

Question:

During EOPs, the implementing procedure directs a step that the CBOT states is no longer appropriate for the given plant conditions. Which of the following is the correct process to handle the EOP procedure step in accordance with the ANO-2 EOP/AOP User Guide OP-1015.021?

A. The step MUST be performed as directed without reversal to mitigate the consequences of the accident.

B. The step MUST be performed but may be reversed if both the Shift Manager (SM) and CRS agree with the reversal.

C. The step MAY be skipped if the CRS and STA concur that the step is no longer appropriate for the given conditions.

D. The step MAY be skipped if authorized by the SM and the CRS concurs that the step is no longer appropriate.

Answer:

D. The step MAY be skipped if authorized by the SM and the CRS concurs that the step is no longer appropriate.

Notes:

D is the correct answer: If, due to existing plant conditions, a procedure step is considered by the crew to be unnecessary or inappropriate, the Shift Manager has the authority to authorize skipping the step. Also the CRS must concur because two SROs must be involved in the decision to not perform an AOP/EOP step.

A is incorrect because the users guide allows skipping of steps but plausible as skipping steps in EOP/AOP should be the exception rather than the rule if any doubt exist.

B is incorrect because the users guide allows skipping of steps but plausible as performing the step may be a conservative action if the overall effect of not performing the step is not understood.

C is incorrect because two SROs must be involved in the decision to skip a step but plausible as the STA may be a licensed SRO and may even be qualified as a Shift Manager.

This question matches the K&A as it requires knowledge of the general guidelines for EOP usage.

References:

Admin 1015.021 ANO-2 EOP-AOP Users Guide Rev. 15 Step 9.1.2
(Verified references updated 11/15/16)

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Removed "ONE (1) " from the stem and added a question mark at the end. Changed "can be undone" to "may be reversed" in distractor B. Changed "can' to "may" in Distractors C and D and ATC to CRS in C.

REV. 2 based on NRC Chief Examiner Feedback BNC. Added the words "without reversal" to distractor A. Added clarifying words to B,C, and D to balance all 4 choices. Capitalized the words "must" and "may" in the 4 choices.

9.1.2 **QUESTION 2**

Under what conditions or circumstances would it be permissible to skip a step in a procedure? Are the operators expected to perform each step of the procedure regardless of the situation?

ANSWER

Operators are charged with the responsibility of taking appropriate actions to mitigate the consequences of an accident. The EOPs and AOPs are provided as tools to ensure that the minimum steps are taken to protect both equipment and personnel. In order for the procedures to be an effective tool, the user must understand the overall intent (recovery strategy) and the reason behind each step.

If, due to existing plant conditions, a procedure step is considered by the crew to be unnecessary or inappropriate, the Shift Manager has the authority to authorize skipping the step. Keep in mind, however, that by doing so the Shift Manager assumes full responsibility for any adverse consequences that may result from this decision. In any case, two SROs must be involved in the decision to not perform an AOP/EOP step.

Exercising this option shall be considered the exception rather than the rule. If there is any doubt that skipping the step is necessary, it is advised that the step be performed. If performance of the step does not yield the benefit originally intended by its placement in the procedure, the Shift Manager may consider undoing the step. This is a procedure deviation and procedure deviation annotation/strategy applies.

EXAMPLE:

During a SGTR with a concurrent overcooling problem, the Functional Recovery Procedure directs you to close the SG blowdown valves in an effort to stop the excessive cooldown. You may need the blowdown system to control level in the ruptured SG and have already identified the overcooling to be due to a steam leak in the MSIV room. For this situation, isolating blowdown may not be required.

The conservative approach would be to go ahead and close the valves and observe the effect on the overcooling event. If blowdown was not the source of the problem, unisolate blowdown on the ruptured SG to aid you in level control.

Question 73

Bank:	2403	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	73	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	1940012431	**10CFR55:**	41.10	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.4.31				
Tier:	3	**Group:**	1	**RO Imp:**	4.2	**SRO Imp:**	4.1	**L. Plan:**	A2LP-RO-PANN
OBJ	2								

Description: Emergency Procedures/Plan - Knowledge of annunciator alarms, indications, or response procedures.

Question:

During assessment of annunciators at the end of SPTAs, what does it mean when a Control Room annunciator is in the "Slow Flash " mode?

A. An alarm condition has occurred and is still present and the Operator has NOT acknowledged the alarm.

B. An alarm condition has occurred and the Operator has acknowledged the alarm; however, the alarm condition is still present.

C. An alarm condition has occurred and then cleared and has NOT been acknowledged by the Operator.

D. An alarm condition has occurred and then cleared and then was acknowledged by the Operator.

Answer:

C. An alarm condition has occurred and then cleared and has NOT been acknowledged by the Operator.

Notes:

There are 4 modes that an annunciator can be in: "Out/Clear", "Fast Flash", "Solid/Locked In", and "Slow Flash" and an assessment must be performed a the end of SPTAs to determine the significance of alarms to assist the CRS in diagnosing the event.

C is correct as a slow flashing alarm window means an alarm condition has occurred and then cleared and has NOT been acknowledged by the Operator. This question is based on the SGTR event at Palo Verde in 1992 when a radiation monitoring alarm came in at power but went into a slow flashing mode that was acknowledged/cleared after the plant trip without understanding the meaning of the alarm and thus prevented an accurate diagnosis of the event.

A is incorrect but plausible as this defines a "Fast Flash" annunciator

B is incorrect but plausible as this defines a "Solid/Locked In" annunciator

D is incorrect but plausible as this defines a "Out/Clear" annunciator.

This question matches the K&A because it requires knowledge of the different modes of alarm indications and their meaning.

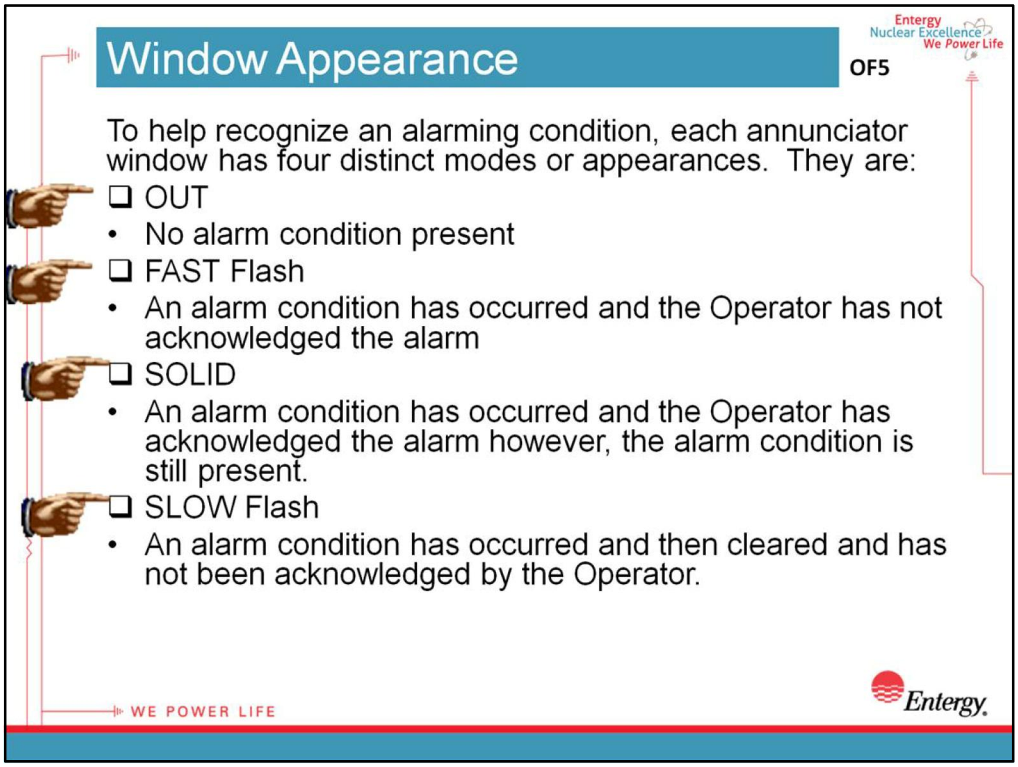
References:

Lesson Plan A2LP-RO-PANN_9-1 Plant Annunciators Rev 9 Objective 2;
Admin COPD.001 OPS Expectations and Standards Rev. 70 Step 5.10.1.C
(Verified all references updated 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Added " During assessment of annunciators at the end of SPTAs" to the stem to tie question to EOPs and the OE at Palo Verde where an operator acknowledged a slow flashing SG high activity alarm and did not realize the significance of the alarm.



Discuss in general the types of annunciator windows used. (Obj. 2)
OF5 – Knowledge of Plant Design, Engineering Principles, and Sciences

EN-OP-115-08
COPD-001, section 5.10
Q. Flashing status's?
A. To help recognize an alarming condition, each annunciator window has four distinct modes or appearances. They are:
❑OUT
No alarm condition present
❑FAST Flash
An alarm condition has occurred and the Operator has not acknowledged the alarm
❑SOLID
An alarm condition has occurred and the Operator has acknowledged the alarm however, the alarm condition is still present.
❑SLOW Flash
An alarm condition has occurred and then cleared and has not been acknowledged by the Operator.

5.10 ANNUNCIATOR RESPONSE

5.10.1 Additional Standards and Expectations

A. In most cases, diverse methods of sensing parameters are available in the Control Room. Reliance should not be placed on a single sensing device or readout when taking corrective action. An evaluation of the conditions should be made based on all information sources and instruments readily available.

B. Multi-point recorders or remote panels where the presence of one alarm blocks the Control Room annunciation of subsequent abnormal conditions should be monitored more frequently.

C. Multiple input alarms with reflash capability are identified as follows:

- A black dot is located on the Control Room alarm window
- The window is identified in the Annunciator Corrective Action
- Acknowledged alarms with reflash capability will go to fast flash on subsequent alarms

D. When responding to confirmed fires or possible high activity leakage, operators should respond with full respiratory protection.

E. When responding to confirmed fires or possible high activity leakage in controlled access, the Shift Manager should notify RP.

F. When responding to or assessing sources of significant high activity leakage in the Auxiliary Building, the operators should proceed as follows:

- As a minimum, notify RP and don SCBAs prior to entry
- If time allows, full protective clothing in addition to SCBAs should be employed and an RP escort requested

G. When implementing an AOP or EOP, the Annunciator Corrective Action for alarms received during the event should be reviewed, and appropriate actions taken, under these criteria:

- As time allows
- At Shift Turnover
- Before exiting EOP/AOP

Question 74

Bank:	2404	**Rev:**	0	**Rev Date:**	8/2/2016	**2017 TEST QID #:**	74	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	1940012312	**10CFR55:**	41.12	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.3.12				
Tier:	3	**Group:**	1	**RO Imp:**	3.2	**SRO Imp:**	3.7	**L. Plan:** A2LP-RO-TS	**OBJ** 3

Description: Radiological Controls - Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Question:

To minimize the radiological dose to refueling personnel in Mode 6 with core refueling in progress, the water level in the refueling canal must be a MINIMUM of 23 feet over the top of the _______________ and to protect the fuel assemblies from overheating and releasing radiation, _______ loop(s) of Shutdown Cooling shall be in operation.

A. reactor pressure vessel flange; 1

B. fuel assemblies seated in the vessel; 1

C. reactor pressure vessel flange; 2

D. fuel assemblies seated in the vessel; 2

Answer:

B. fuel assemblies seated in the vessel; 1

Notes:

B is correct per TS 3.9.9 and 3.9.8.1 applicability.

A is incorrect as the water level must be 23 feet above the top of the seated fuel assemblies but plausible as the candidate must realize that the top of the fuel assemblies are much lower than the reactor vessel flange and only 1 train of SDC is required to be in operation.

C. is incorrect as the water level must be 23 feet above the top of the seated fuel assemblies and only 1 train of SDC is required to be in operation but plausible as the candidate must realize that the top of the fuel assemblies are much lower than the reactor vessel flange and if water level is less than 23 feet above the top of the seated fuel assemblies, then 2 loops of SDC are required to be operable.

D is incorrect because only 1 train of SDC is required to be in operation but plausible because if water level is less than 23 feet above the top of the seated fuel assemblies, then 2 loops of SDC are required to be operable. And the 1st part is correct as to the water level over the top of the fuel assemblies seated in the core.

This question matches the K&A because the RO has the responsibility during fuel handling to ensure adequate water level and core decay heat removal is maintained to limit radiological dose from spent or dropped or melted fuel assemblies.

References:

ANO-2 TS 3.9.8.1 Change 104, 3.9.8.2 Change 233, and 3.9.9 Change 167
(Verified references updated 11/10/16);

Historical Comments:

To be used on the 2017 NRC Exam

SHUTDOWN COOLING AND COOLANT CIRCULATION

SHUTDOWN COOLING – ONE LOOP

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one shutdown cooling loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The shutdown cooling loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 A shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of ≥ 2000 gpm at least once per 24 hours.

SHUTDOWN COOLING – TWO LOOPS

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent shutdown cooling loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling loops OPERABLE, immediately initiate corrective action to return the loops to OPERABLE status as soon as possible.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required shutdown cooling loops shall be determined OPERABLE per the Inservice Testing Program.

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* The normal or emergency power source may be inoperable for each shutdown cooling loop.

REFUELING OPERATIONS

WATER LEVEL – REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of fuel assemblies or CEAs within the reactor pressure vessel while in MODE 6, except during latching and unlatching of CEAs.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or CEAs within the pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or CEAs.

Question 75

Bank:	2405	**Rev:**	4	**Rev Date:**	12/22/2016	**2017 TEST QID #:**	75	**Author:**	Coble
Lic Level:	RO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NRC Exam Bank 1887		
Search	1940012120	**10CFR55:**	41.10	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.1.20				
Tier:	3	**Group:**	1	**RO Imp:**	4.6	**SRO Imp:**	4.6	**L. Plan:**	ASLP-RO-PRCON
OBJ	6								

Description: Conduct of Operations - Ability to interpret and execute procedure steps.

Question:

Per EN-HU-106, Procedure and Work Instructions Use and Adherence, steps identified as bulleted steps rather than numbered or lettered steps ____.

A. will be performed on a continuing basis after the step is presented in the procedure

B. are required to be performed in the sequence as written in the implementing procedure

C. can be performed in any sequence or in parallel when the step is presented in the procedure

D. can be performed at anytime the specified condition in the step exist when the procedure is in use

Answer:

C. can be performed in any sequence or in parallel when the step is presented in the procedure

Notes:

C is correct: From EN-HU-106 Step 5.2.2 [5] Technical Procedure users shall perform the Procedure or Work Instruction steps in the sequence written unless specific allowance to skip sections/steps is permitted in the Procedure or Work Instruction. Steps identified as bulleted steps rather than numbered or lettered steps can be performed in any sequence or in parallel.

A is incorrect as bullets are sub steps of a master step number and this statement describes a "Continuous Action Step" which can be continuously performed in a procedure after the step is 1st presented in the procedure but plausible as this is knowledge of how to execute a procedure step when the condition in the step is met.

B is incorrect because bulleted step are not required to be performed in the sequence as written but plausible as step numbers are required to be performed in the sequence as written.

D is incorrect as bulleted step are only performed in any sequence or in parallel when the master step sequence comes up in the procedure but plausible as this distractor describes a floating step which is a conditional step that can be performed any time the condition presents itself during implementation of an AOP or EOP procedure..

This question matches the K&A as it requires the applicant to have the knowledge of what a bulleted step means to be able to execute a procedure step with bullets correctly.

References:

EN-HU-106, Procedure and Work Instruction Use and Adherence, Rev. 3, Step 5.2.2 [5] and [3]
OP-1015.021, EOP/AOP Users Guide, Rev. 015, section 4.0 step 4.7 (conditional steps), and step 4.20 (floating steps)

Historical Comments:

NRC Exam Bank 1887was used on the 2012 NRC Exam
To be used on the 2017 NRC Exam

REV. 1 based on NRC Chief Examiner Feedback BNC. Changed distractor B to "start the EFW Pump" and made the

question into a statement that includes the 2nd part of the QID of the original.

REV. 2 based on NRC Chief Examiner Feedback BNC. Rejected the 2.1.17 K/A as requested by the Chief NRC Examiner as this question topic and K&A 2.1.17 are already being tested thoroughly during the scenarios. Therefore K&A 2.1.20 was randomly selected and another bank question was selected.

REV. 3 based on NRC Chief Examiner Feedback BNC. Changed distractor A to a continuous action step description due to feedback that the Rev. 2 distractor A could be a potential correct answer.

REV. 4 based on NRC Chief Examiner Feedback BNC. Reworded Distractor D to the verbiage in step 4.20 of OP-1015.021 ANO-2 EOP/AOP USER GUIDE. Added clarifying words to choices A, B, and C to balance the question and clarify the choices.

5.2.2 Cont.

[5] Technical Procedure users shall perform the Procedure or Work Instruction steps in the sequence written unless specific allowance to skip sections/steps is permitted in the Procedure or Work Instruction. Steps identified as bulleted steps rather than numbered or lettered steps can be performed in any sequence or in parallel.

[6] In the event of an emergency situation not covered by a Procedure or Work Instruction, the personnel involved shall take action to minimize personnel injury and damage to the facility.

[7] Before performing a procedural step, the Procedure or Work Instruction user should ensure that:

(a) Action to be performed is on the correct component.

(b) Expected response has been anticipated.

(c) The appropriate monitoring method for the response has been identified.

[8] When indicating a step is complete:

(a) Perform placekeeping in accordance with this procedure (Sec 5.6.3).

(b) When work conditions prevent the performer from physically signing or initialing a step as it is performed (e.g., in a contaminated area), the step shall be signed off as soon as work conditions make signing possible. In no case shall this be later than the end of the shift in which the step was completed.

(c) If an incorrect entry is made to a Procedure or Work Instruction (i.e., data entry, initial, etc.), draw a single line through the entry and initial and date the line through. Locate the initial and date to minimize the likelihood that they are misconstrued as the intended data entry. Do not use White-Out, correction tape, or Liquid Paper type products to make corrections.

[9] If a Procedure or Work Instruction is stopped and cannot be resumed during the same shift (other than shift turnover), prior to restarting, ensure that all Prerequisites and any Initial Conditions necessary for performance are satisfied.

5.2.3 Use of Not Applicable (***N/A***) During Procedure or Work Instruction use:

[1] Procedure or Work Instruction steps that contain conditional statements or provide specific conditions for being marked as not applicable may be marked N/A without additional written justification or supervisory approval.

4.4 BRANCHING

Directs the user to exit the current procedure or step and enter another procedure or step. The words "GO TO" and "RETURN TO" are used to indicate a branch (See Defined EOP/AOP Words, Section 4.40).

4.5 CAUTION

Calls attention to a condition that could result in damage to equipment, personnel hazard, or otherwise adversely affect plant operations.

4.6 COMMUNICATIONS STRATEGIES

Discussions, Transient Briefs and Update Announcements are communications tools employed during implementation of EOPs/AOPs. Definitions and implementation strategy for these tools are found in Operations Expectations and Standards, COPD-001.

4.7 CONDITIONAL STATEMENTS AND LOGIC SEQUENCES

Used in EOPs/AOPs to describe a set of conditions or express complex combinations of conditions. Logic terms include **IF**, **WHEN**, **THEN**, **AND**, **OR** and **NOT**.

4.8 CONTINGENCY ACTION STEPS

Action steps performed if the instruction or expected response is not achieved.

4.9 CONTINUOUS ACTION STEPS

Procedure steps performed on a continuing basis AFTER they are presented. Actions are taken by the operator when needed.
Continuous action steps are prefaced by an asterisk (*) in front of the step number.

4.10 CONTINUING ACTIONS

Actions which are implemented after all success paths for a safety function have been attempted and all safety function acceptance criteria are **NOT** met for a particular safety function (safety function is **NOT** satisfied). The operator is required to continue to work on this safety function and to pursue other jeopardized or challenged safety functions simultaneously.

4.11 DECISION TREES

Flowcharts which provide the operator with a quick method to determine the appropriate success path to use in the functional recovery procedures.

4.19 EXIT CONDITIONS

Conditions that are written to explicitly identify the conditions that must exist before the operators leave the EOP/AOP. In general, the EOP/AOP should be exited if an inappropriate procedure has been implemented, or the EOP/AOP has met its goal.

4.20 FLOATING STEPS

Procedure steps that allow performance of that step at any time the specified conditions exist once the operator has entered that procedure. FLOATING STEPS are prefaced by a solid rectangle (■) in front of the step number.

4.21 FLOWCHARTS

A printed format in which procedures or processes are depicted in graphic form emphasizing decision points and links between decision points.

4.22 FUNCTIONAL RECOVERY PROCEDURE (FRP) and LOWER MODE FUNCTIONAL RECOVERY PROCEDURE (LMFRP)

The ANO-2 plant specific Functional Recovery procedures are used by the operator to ensure the satisfactory control or restoration of all safety functions, and to provide actions to restore and maintain those safety functions when degraded. 2202.009, Functional Recovery Procedure or 2202.011, Lower Mode Functional Recovery Procedure is written such that the user need not diagnose an event in order to establish and maintain the plant in a safe, stable condition.

4.23 IMMEDIATE ACTION STEPS

Steps taken without delay when there are indications of an emergency. These actions are to stop further degradation of existing conditions, to mitigate the consequences and to allow the operators to assess the situation. These actions may be performed simultaneously by different operators and are not sequence dependent unless specifically stated. All Immediate Action Steps are completed even if a previously listed Immediate Action Step results in a branching instruction.

IMMEDIATE ACTION STEPS are designated by a circle around the step designator. All sub-steps under the circled step are also Immediate Action Steps.

4.24 INSTRUCTION STEPS

Action statements describing either the action to be performed or an expected plant response.

4.25 LONG-TERM ACTIONS

Actions which are implemented once all safety function acceptance criteria have been met for a particular safety function (safety function is satisfied). These long term actions shall help ensure that the operator continues to periodically check adequate maintenance of safety functions, assess the status of the plant, implement the appropriate Optimal Recovery Procedure or ORP if conditions warrant, and determine whether it is possible to cooldown to shutdown cooling conditions.

ORIGINAL BANK QUESTION 1887 WAS USED TO DEVELOP 2017 NRC EXAM QID#75 (Used on the 2012 NRC Exam)

Data for 2012 NRC RO/SRO Exam

Bank:	1887	**Rev:**	0	**Rev Date:**	4/27/2012 3:21:51	**QID #:**	66	**Author:**	Foster
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW		
Search	1940012120	**10CFR55:**	41.10	**Safety Function**					
System Title:	Generic	**System Number**	GENERIC	**K/A**	2.1.20				
Tier:	3	**Group:**	1	**RO Imp:**	4.6	**SRO Imp:**	4.6	**L. Plan:**	ASLP-RO-PRCON
OBJ	6								

Description: Conduct of Operations - Ability to interpret and execute procedure steps.

Question:

Bulleted procedural steps ___________________________ .

A. will only be performed if conditions are met

B. are required to be performed in order as written

C. can be performed in any sequence or in parallel

D. can be performed at anytime the procedure is in use

QID use History

	RO	SRO
2005	☐	☐
2006	☐	☐
2008	☐	☐
2009	☐	☐
2011	☐	☐
2012	☑	☑

Audit Exam History

2006	☐
2009	☐
2011	☐

Answer:

C. Corrected, as per EN-AD-102 step 5.2.2.[3](a)

Notes:

A. Incorrect, defines a conditional (or IF/THEN) step
B. Incorrect, defines normal (numbered) steps
D. Incorrect, defines a floating (EOP/AOP) step

References:

EN-HU-106, Procedure and Work Instruction Use and Adherence, Rev. 0, Step 5.2.2 [2] and [3] [pages 8 and 9]
OP-1015.021, EOP/AOP Users Guide, rev. 011, section 4.0 step 4.7 (conditional steps), and step 4.20 (floating steps) [pages 3 and 5]
ASLP-RO-PRCON, Rev. 7, Obj. 6, Describe the expectationd for procedural adherence

Historical Comments:

New Question

Question 76

Bank:	2406	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	76	**Author:**	Burton		
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam				
Search	00CE052409	**10CFR55:**	43.5	**Safety Function**	4						
Title:	Excess Steam Demand	**System Number**	E05	**K/A**	2.4.9						
Tier:	1	**Group:**	1	**RO Imp:**	3.8	**SRO Imp:**	4.2	**L. Plan:**	A2LP-RO-EESD	**OBJ**	8

Description: Emergency Procedures/Plan - Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Question:

Given the following:

* Unit 2 was operating at 4% reactor power.
* Main Steam line break on 'A' SG has occurred in Containment.
* Reactor trip and MSIS automatically initiated.
* 'A' SG level (WR) is off-scale low.
* Containment pressure is 20.5 psia and lowering.
* T-cold is 400°F and stabilizing.
* CET temperature is 420°F and stabilizing.
* Pressurizer pressure is 1950 psia and slowly rising.

Which procedure and required actions should be used?

A. Implement OP-2203.011, RCS Overcooling and stabilize RCS pressure.

B. Implement OP-2202.005, Excess Steam Demand and stabilize RCS pressure.

C. Implement OP-2203.011, RCS Overcooling and lower RCS pressure to maintain RCS MTS less than 200°F.

D. Implement OP-2202.005, Excess Steam Demand and lower RCS pressure to maintain RCS MTS less than 200°F.

Answer:

D. Implement OP-2202.005, Excess Steam Demand and lower RCS pressure to maintain RCS MTS less than 200°F.

Notes:

For the given conditions the correct answer is to implement ESD and lower RCS pressure to within the 200°F limits IAW the P-T curve. RCS Overcooling AOP is plausible because this would be the appropriate procedure to enter if a smaller steam leak occurred at power without an MSIS.

D is correct. ESD is the correct procedure and the MTS is > 200°F so the direction is to stop the cooldown, de-pressurize to maintain the RCS < 200 degrees subcooled.

A is incorrect. Overcooling AOP is wrong but plausible because it would be used for a smaller steam leak. Stabilize RCS pressure is wrong but plausible because this is the expected actions for an ESD. Step 21.C "Maintain RCS pressure within P-T using attachment 48" requires the a depressurization therefore
stabilizing RCS pressure is wrong.

B is incorrect. ESD is the correct procedure but stabilizing RCS pressure is wrong since the RCS is over-subcooled. Step 21.C "Maintain RCS pressure within P-T using attachment 48" requires a depressurization therefore stabilizing RCS

pressure is wrong.

C is incorrect. Overcooling AOP is wrong but plausible because it would be used for a smaller steam leak. Lowering RCS pressure is correct as stated in "D".

KA Match - The event is an ESD as stated in the stem. It matches the low power criteria as the event occurs in Mode 2 additionally the question requires knowledge of ESD mitigation strategies.

References:

EOP-2202.005 (015), Excess Steam Demand
AOP-2203.011 (006), Overcooling Event in Mode 1 or 2
EOP-2202.010 (023), Standard Attachment 1 (P-T Limits)
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Rev 1 - no change to question but added more explanation to Notes clarifying why "B" is wrong.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

If possible during the cooldown, maintain SG to RCS Δp less than 1600 psid.

■ **21. Maintain RCS post-cooldown conditions as follows:**

A. Maintain RCS temperature by steaming intact SG using EITHER of the following:

- Upstream ADV
- Upstream ADV Isolation MOV

B. Control feedwater flow to intact SG using 2202.010 Attachment 46, Establishing EFW Flow.

C. Maintain RCS pressure within P-T limits using 2202.010 Attachment 48, RCS Pressure Control.

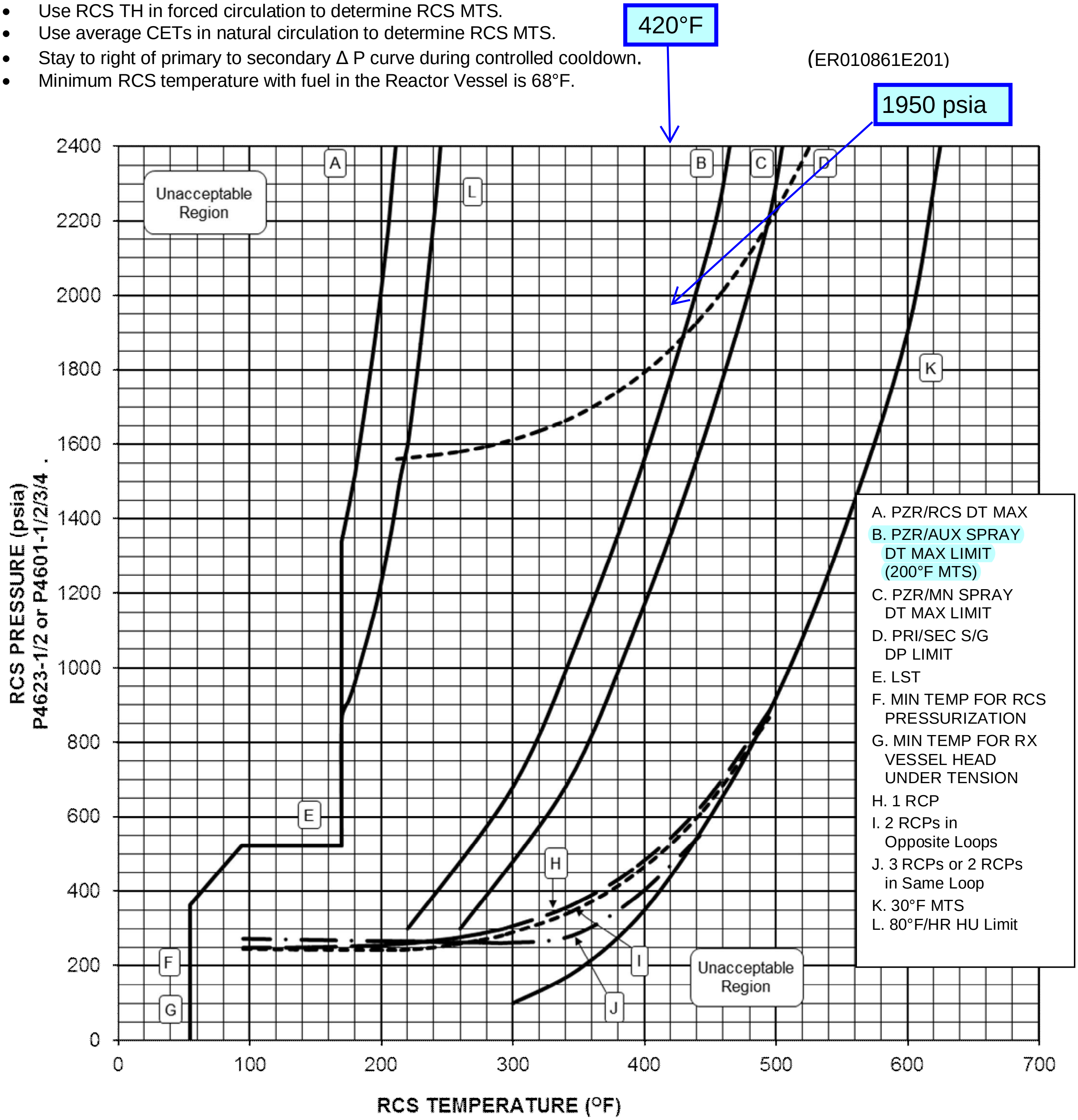
Stabilizing RCS pressure and temperature is the expected actions during an ESD event
Distractors A and B

INSTRUCTIONS	CONTINGENCY ACTIONS
***29. Maintain RCS P-T limits and RCP NPSH requirements, refer to 2202.010 Attachment 1, P-T Limits.**	***29. Perform the following:** A. Restore RCS P-T limits, refer to 2202.010 Attachment 1, P-T Limits. B. IF NPSH requirements violated OR RCS MTS less than 30°F, THEN perform the following: 1) Stop ALL RCPs. 2) Verify BOTH PZR Spray valves in MANUAL and closed.
***30. IF uncontrolled RCS cooldown below 500°F T_C has occurred, THEN perform the following:** A. Stop RCS cooldown. B. Depressurize RCS using 2202.010 Attachment 48, RCS Pressure Control. C. IF HPSI termination criteria satisfied, THEN control HPSI and Charging flow. D. Maintain RCS MTS less than 200°F, refer to 2202.010 Attachment 1, P-T Limits.	Per the stem - RCS cool-down to 400°F so this step should be performed. RCS is greater than 200°F sub-cooled.

ATTACHMENT 1
P-T LIMITS

- Use T4615 (2TR-4615) or T4715 (2TR-4715) if plotting RCS cooldown.
- If plotting cooldown with no RCPs in operation and SDC in service, Tcold should be SDC return temp T5095 (2TR-5097).
- Use Control Board indication for RCP Operating Curves - Includes 12°F and 19 psi instrument errors.
- Use 200°F MTS line for PTS limit for uncontrolled RCS cooldown below 500°F T_C.
- Use RCS TH in forced circulation to determine RCS MTS.
- Use average CETs in natural circulation to determine RCS MTS.
- Stay to right of primary to secondary Δ P curve during controlled cooldown.
- Minimum RCS temperature with fuel in the Reactor Vessel is 68°F.

(ER010861E201)



RCS OVERCOOLING

PURPOSE

This procedure provides actions for RCS overcooling without MSIS.

ENTRY CONDITIONS

Procedure will not be entered because an MSIS was automatically initiated

Reactor shutdown AND EITHER of the following exist:

1. SG pressure less than 950 psia and lowering.
2. Unexplained drop in RCS temperature and pressure.

EXIT CONDITIONS

1. ANY SFSC acceptance criteria NOT satisfied.

OR

2. All actions of this procedure have been completed.

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

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Diagnosing plant/RCS conditions and applying the logic of an IF - THEN step is an SRO function

No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Question 77

Bank:	2407	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	77	**Author:**	Burton
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC Exam Bank #2263		
Search	000025A204	**10CFR55:**	43.5	**Safety Function**	4				
Title:	Loss of Residual Heat Removal System (RHRS)	**System Number**	025	**K/A**	AA2.04				
Tier:	1	**Group:**	1	**RO Imp:**	3.3	**SRO Imp:**	3.6	**L. Plan:**	A2LP-RO-LMFR	**OBJ**	5

Description: Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: - Location and isolability of leaks

Question:

Given the following:

* Unit 2 is in Mode 6 with Core offload in progress.
* "B" train LPSI Pump (2P-60B) is in service through the "B" SDC Heat Exchanger.
* RCS temperature is 130°F.
* 2RITS-1456, 2E-35B Outlet Radiation monitor is in alarm.
* PROC LIQUID RADIATION HI/LO, 2K11-C10 is in alarm.
* RCS level is slowly lowering.

At this point the CRS _________________ and direct isolating ______________ to the "B" SDC Heat Exchanger (2E-35B).

A. should enter OP-2203.029, Loss of Shutdown Cooling; the Reactor Coolant System (RCS) ONLY

B. must enter OP-2202.011, Lower Mode Functional Recovery; the Reactor Coolant System (RCS) ONLY

C. should enter OP-2203.029, Loss of Shutdown Cooling; both the Reactor Coolant System (RCS) and Service Water (SW)

D. must enter OP-2202.011, Lower Mode Functional Recovery; both the Reactor Coolant System (RCS) and Service Water (SW)

Answer:

C. should enter OP-2203.029, Loss of Shutdown Cooling; both Reactor Coolant System (RCS) and Service Water (SW)

Notes:

C. Is correct - RCS level trending down slowly coupled with the "B" SDC Hx rad monitor in alarm gives an indication of an RCS leak. The candidate should determine that Loss of SDC AOP entry conditions have been met and once entered should direct placing the "A" SDC HX in service and isolate both SW and RCS to the leaking HX as directed by the contingency actions and Attachment B.

A. Loss of SDC is correct since entry conditions have been met. Isolating only the RCS is plausible because isolating RCS is the correct and expected action if the Leak was not specifically into the SW system.

B. Is incorrect. LMFRP is plausible because it is found within the Loss of SDC AOP but wrong because none of the transition triggers within the LOSS of SDC procedure have been met. Leaking SDC Heat Exchanger. Isolating only the RCS is plausible because isolating RCS is the correct and expected action if the Leak was not specifically into the SW system.

D. Is incorrect. LMFRP is plausible because it is found within the Loss of SDC AOP but wrong because none of the transition triggers within the LOSS of SDC procedure have been met. Second part is correct, isolate both RCS and SW.

KA Match - Question requires knowledge of location and actions to isolate a leak from the RHR to Service Water system.

References:

AOP-2203.029 (020), Loss of Shutdown Cooling
EOP-2202.011 (013), Lower Mode Functional Recover
ACA-2203.012K (046), Annunciator 2K11 Corrective Action
(All references verified current 11/10/16)

Historical Comments:

Bank Question #2263 from the 2015 SRO exam.
Minor alterations include changing from leaking SDC HX from 'A' to 'B' train.
Changed stem to core offload from reload.
Changed distractor rotation making the correct answer C instead of B.
To be used on the 2017 NRC Exam
Rev 1 - changed would to should in "A and C". Added 'the' to "C and D"

INSTRUCTIONS	CONTINGENCY ACTIONS
2. **CHECK** EITHER 4160V Non-Vital AC bus 2A1 or 2A2 energized.	2. **CHECK** EITHER 4160V Vital AC bus energized.
	A. **PERFORM** the following:
	1) **CLOSE** ALL LPSI Injection MOVs previously open for SDC flow.
	2) **THROTTLE** open ONE LPSI Injection MOV for approximately 2 seconds.
	3) **ENSURE** RCS level adequate, **REFER TO** 2202.010 Attachment 33, RCS Level.
	* 4) **CONTROL** RCS cooldown rate within TS limits, **REFER TO** 2202.010 Attachment 8 RCS Cooldown Table.
	5) **START** SDC pump, **REFER TO** 2104.004, Shutdown Cooling System.
	6) **RAISE** flow to greater than 2400 gpm by opening LPSI Injection MOVs.
One of the LMFRP transtition triggers	7) **GO TO** 2202.011, Lower Mode Functional Recovery.
	B. **IF** SDC cannot be restored, **THEN GO TO** 2202.011, Lower Mode Functional Recovery.
3. **OPEN** Placekeeping Page and **RECORD** the following:	
A. Event start time _______.	
B. RCS temperature ________°F.	
4. **NOTIFY** Control Board Operators to monitor floating steps.	

INSTRUCTIONS	CONTINGENCY ACTIONS
5. **INITIATE** 2202.010 Attachment 32, CNTMT Evacuation Checklist.	
6. **COMMENCE** plotting heatup rate every 15 minutes using Form 1015.016I, RCS/PZR Temperature VS Time.	
*7. **CHECK** RCS pressure less than 300 psia.	*7. **PERFORM** the following: A. **SECURE** the operating SDC pump. B. **CLOSE** at least ONE SDC RCS Isolation MOV: • 2CV-5038-1 • 2CV-5084-1 • 2CV-5086-2 C. **GO TO** Step 9.

INSTRUCTIONS	CONTINGENCY ACTIONS
8. **CHECK** ALL SDC RCS Isolation MOVs open: • 2CV-5038-1 • 2CV-5084-1 • 2CV-5086-2	8. **STOP** running SDC pump(s).

NOTE

During pressurization events affecting indicated level, RCS inventory changes can be validated using alternate indications (i.e., LPSI Pump motor amps, LPSI Pump suction pressure, etc.).

INSTRUCTIONS	CONTINGENCY ACTIONS
*9. **CHECK** RCS level stable or rising. RCS only Isolation steps Plausibility of A and B	*9. **ATTEMPT** to stop inventory loss as follows: A. **STOP** ALL RCS draining. B. **IF** Letdown in service **AND** location of leak unknown, **THEN CLOSE** at least ONE Letdown Isolation valve: • 2CV-4820-2 • 2CV-4821-1 • 2CV-4823-2 (least preferred) C. **WHEN** Letdown isolated, **THEN OPERATE** Charging as needed to maintain RCS level.

(Step 9 continued on next page)

INSTRUCTIONS	CONTINGENCY ACTIONS
9. (continued)	D. **IF** RCS level lowering rapidly, **THEN PERFORM** the following: 1) **ENSURE** running SDC pump(s) stopped. 2) **CLOSE** at least ONE SDC RCS Isolation MOV: 3) **CLOSE** ALL LPSI Injection MOVs: 4) **GO TO** 2202.011, Lower Mode Functional Recovery.
LMFRP transition is not met - NOT rapidly lowering	
	E. **IF** EITHER 2E-35A SDC HX Outlet Radiation monitor indicates RCS leak to Service Water: • 2RITS-1453 for 2E-35A • 2RITS-1456 for 2E-35B
alarm provided in stem	**THEN PERFORM** Attachment B, SDC HX Leakage Actions.
	F. **IF** SDC Purification in service **AND** leak NOT isolated, **THEN ISOLATE** SDC Purification and Alternate Purification System using 2104.004, Shutdown Cooling Operations.
	G. **IF** RCS leak **NOT** isolated, **THEN GO TO** 2202.011, Lower Mode Functional Recovery.
■10. **CHECK** RCS level adequate per EITHER of the following: • PZR level 29 to 82%. • Adequate using Attachment A, RCS Level.	■10. **IF** RCS makeup required, **THEN RESTORE** RCS level using Charging or HPSI.

ATTACHMENT B
SDC HX LEAKAGE ACTIONS

1. **IF** 2E-35A Outlet Radiation monitor (2RITS-1453) indicates RCS leak to Service Water, **THEN PERFORM** the following to align SDC HX 2E-35B:

   A. **ENSURE** SW available to SDC HX 2E-35B.

   B. **OPEN** SW to 2E-35B (2CV-1456-2).

   C. Locally **OPEN** "SHUTDOWN COOLING TO 2E-35B" (2SI-4B).

   D. Locally **CLOSE** "SHUTDOWN COOLING TO 2E-35A" (2SI-4A).

   E. Locally **CLOSE** "SDC COOLER TO LPSI DISCH HEADER" (2SI-5A).

   F. **CLOSE** SW to 2E-35A (2CV-1453-1).

   G. Locally **ENSURE** "SDC HX 2E35A SW OUTLET" (2SW-11A) closed.

2. **IF** 2E-35B Outlet Radiation monitor (2RITS-1456) indicates RCS leakto Service Water, **THEN PERFORM** the following to align SDC HX 2E-35A:

   Isolate BOTH RCS and Service Water

   A. **ENSURE** SW available to SDC HX 2E-35A.

   B. **OPEN** SW to 2E-35A (2CV-1453-1).

   C. Locally **OPEN** "SHUTDOWN COOLING TO 2E-35A" (2SI-4A).

   D. Locally **CLOSE** "SHUTDOWN COOLING TO 2E-35B" (2SI-4B).

   E. Locally **CLOSE** "SDC COOLER TO LPSI DISCH HEADER" (2SI-5B).

   F. **CLOSE** SW to 2E-35B (2CV-1456-2).

   G. Locally **ENSURE** "SDC HX 2E35B SW OUTLET" (2SW-11B) closed.

3. **NOTIFY** Chemistry to perform the following:

   - **SAMPLE** RCS for chlorides.
   - **CALCULATE** ANY offsite releases.

INSTRUCTIONS	CONTINGENCY ACTIONS
K. Check for rising trends or alarm on EITHER SDC HX Outlet Radiation Monitor (2RITS-1453/1456).	K. Perform the following: 1) Verify SW aligned to unaffected SDC HX. 2) Locally open "SHUTDOWN COOLING TO 2E-35A/B" (2SI-4A) or (2SI-4B) from unaffected SDC HX. 3) Locally close "SHUTDOWN COOLING TO 2E-35A/B" (2SI-4A) or (2SI-4B) from affected SDC HX. 4) Locally close "SDC Cooler to LPSI Disch Header" (2SI-5A) or (2SI-5B) from affected SDC HX. 5) Isolate SW Inlet to affected SDC HX 2E-35A/B: • 2CV-1453-1 • 2CV-1456-2 6) Locally verify affected "SDC HX 2E-35A/B SW OUTLET" closed:: • 2SW-11A • 2SW-11B
L. IF RCS leak NOT isolated, THEN perform the following as necessary: • 2202.010 Attachment 17, LOCA Isolation. • 2202.010 Attachment 22, LOCA Outside CNTMT Isolation.	

Same steps found in both the AOP and LMFRP
Plausibility of B and D

QID-77
7 of 10

ANNUNCIATOR 2K11

C-10

Alarming window

PROC LIQUID RADIATION HI/LO

NOTE

- This alarm will reflash.
- Setpoints are in Radiation Monitoring and Evacuation Alarm System (2105.016).

1.0 CAUSES

1.1 ANY of the following Liquid Process Rad Monitors in HIGH or LOW alarm:

- CCW Sys Return Loop 1 (2RITS-5200)
- CCW Sys Return Loop 2 (2RITS-5202)
- SW from CNTMT Cooling Coils 2VCC-2C/D (2RITS-1513-2)
- SW from CNTMT Cooling Coils 2VCC-2A/B (2RITS-1519-1)
- SW from SDN Cooling HX 2E-35A (2RITS-1453)
- SW from SDN Cooling HX 2E-35B (2RITS-1456)
- SW from Fuel Pool HX (2RITS-1525)
- Liquid Rad Waste Disch to Flume (2RITS-2330)
- Turb Bldg Sump Effluents (2RITS-4301)

1.2 EITHER of the following Liquid Process Radiation Monitors failed LOW:

- SG Blowdown Sample from 2E-24A (2RITS-5854)
- SG Blowdown Sample from 2E-24B (2RITS-5864)

1.3 Loss of power to any of monitors listed in steps 1.1 and 1.2

(C-10 Continued on next page)

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8 of 10

ANNUNCIATOR 2K11

C-10

PROC LIQUID RADIATION HI/LO
(Continued)

NOTE

- 2RITS-1453 alarmed due to high background in 2R21 and 2R22 when SDC Purification was placed in service. Background remained elevated until the refueling canal was flooded.
- PMRQ 50010391-01 (MWO 50234500) can be used by I&C to adjust setpoints as directed by Operations

2.5 IF SW from SDN Cooling HX 2E-35A (2RITS-1453)
OR SW from SDN Cooling HX 2E-35B (2RITS-1456) in High Alarm,
THEN perform the following:

2.5.1 Request chemist obtain and analyze samples.

2.5.2 IF POST LOCA,
THEN inform chemist that sample location is in LNPP at the following locations:

- Loop II SW Return To Emerg Pond Drain (2SW-1540)
- Loop I SW Return to Emerg Pond Drain (2SW-1541).

Alarm sends you to Loss of Shutdown Cooling

2.5.3 IF Shutdown Cooling in service with RCS level lowering,
THEN GO TO Loss of Shutdown Cooling (2203.029).

2.5.4 IF both monitors trending up due to high background,
THEN perform the following:

- Establish periodic trending.
- Based on sample results, consider shifting SDC Trains.

2.5.5 IF Shutdown Cooling in progress with RCS level stable
AND cause of alarm NOT due to high background,
THEN perform the following:

- Shift SDC Trains per Shutdown Cooling System (2104.004).
- Perform leak rate on affected heat exchanger.

2.5.6 IF Spray pump NOT required to reduce CNTMT pressure,
THEN secure affected CNTMT Spray header.

2.5.7 Initiate Condition Report to document SW contamination.

2.6 IF SW from Fuel Pool Heat Exchanger (2RITS-1525) in High Alarm,
THEN GO TO Spent Fuel Pool Emergencies (2203.002).

(C-10 Continued on next page)

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

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

SRO level because this question requires implementation of contingency steps.

No

Does the question require one or more of the foll

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

2015 NRC RO/SRO Exam

11-Aug-16

Bank:	2263	**Rev:**	2	**Rev Date:**	8/4/2015 2:17:55	**QID #:**	77	**Author:**	foster
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC BANK QID #1872		
Search	0000252120	**10CFR55:**	41.10 / 43.5 / 45.12	**Safety Function**	4				
System Title:	Loss of Residual Heat Removal System (RHRS)	**System Number**	025	**K/A**	2.1.20				
Tier:	1	**Group:**	1	**RO Imp:**	4.6	**SRO Imp:**	4.6	**L. Plan:**	A2LP-RO-ASDC
OBJ	3								

Description: Conduct of Operations - Ability to interpret and execute procedure steps.

Question:

ORIGINAL BANK QUESTION 2263 (2015 SRO Exam) was altered to create Q #77 on the 2017 NRC Exam.

Consider the following:

- Unit 2 is in mode 6
- "A" LPSI pump (2P-60A) is in service through the "A" SDC Heat Exchanger (2E35A)
- RCS temperature is being maintained at 135°F
- Refueling is in progress
- Radmonitor 2RITS-1453, 2E35A Outlet Radiation monitor, is in alarm
- RCS level is trending down slowly

The CRS would enter ________________________ and direct isolating ____________________ to the "A" SDC Heat Exchanger (2E-35A).

A. OP-2203.029, Loss of Shutdown Cooling; the Reactor Coolant System (RCS) only

B OP-2203.029, Loss of Shutdown Cooling; both Reactor Coolant System (RCS) and Service Water (SW)

C. OP-2202.011, Lower Mode Functional Recovery; the Reactor Coolant System (RCS) only

D. OP-2202.011, Lower Mode Functional Recovery; both Reactor Coolant System (RCS) and Service Water (SW)

QID use History

	RO	SRO
2005	☐	☐
2006	☐	☐
2008	☐	☐
2009	☐	☐
2011	☐	☐
2012	☑	☑
2014	☐	☐

Audit Exam History

2006	☐
2009	☐
2011	☐

Answer:

B. Correct

Notes:

B. Correct; with RCS level trending down slowly coupled with the "A" SDC Hx rad monitor in alarm gives an indication there is a RCS leak. The candidate should determine the Loss of SDC AOP entry conditions have been met and once entered should direct placing the "B" SDC Hx on line and isolating both SW and RCS to the leaking Hx as directed by the contingency actions in the AOP.
A. Incorrect; The Loss of Shutdown Cooling entry conditions have been met but only isolating SW or RCS is not directed by the AOP. Isolation of SW and RCS [shutdown cooling flow] is required
C and D are Incorrect but plausible due to the ; entry conditions are not met for Lower Mode Functional Recovery EOP. The AOP would direct its entry based on RCS rapidly lowering (given in stem that RCS level is slowly trending down)

References:

OP-2203.012 K, Annunciator 2K11 Corrective Action, Rev 044, window C-10 step 2.5.3 pages 107 and 110
OP-2203.029, Loss of Shutdown Cooling, Rev 016, entry section, page 1, contingency step 8.E pages 5 and 6

Historical Comments:

QID #1872 used on 2012 NRC Exam
Rev 1 - Changed to Bank Q per NRC review comments mwf 7/6/15

Question 78

Bank:	2408	**Rev:**	2	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	78	**Author:**	Burton
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	Modified NRC Exam Bank #2003		
Search	000058A203	**10CFR55:**	43.5	**Safety Function**	6				
Title:	Loss of DC Power	**System Number**	058	**K/A**	AA2.03				
Tier:	1	**Group:**	1	**RO Imp:**	3.5	**SRO Imp:**	3.9	**L. Plan:**	A2LP-RO-ED125
OBJ	11								

Description: Ability to determine and interpret the following as they apply to the Loss of DC Power: - DC loads lost; impact on ability to operate and monitor plant systems

Question:

Consider the following:

* Unit 2 is operating at 100% power.
* 2D02 Green Battery disconnect has been opened for maintenance.
* Battery charger, 2D-32B, is in service.
* Now 2D-32B output breaker trips open.

The CRS would direct entry into ___________ and crew will be required to ___________.

A. OP-2203.037, Loss of 125V DC AOP; cycle Charging pumps to maintain PZR level

B. OP-2203.037, Loss of 125V DC AOP; place all of the HPSI, LPSI and CS Pump handswitches in PTL to prevent spurious starts

C. OP-2202.001, Standard Post Trip Action EOP; cycle Charging pumps to maintain PZR level

D. OP-2202.001, Standard Post Trip Action EOP; place all of the HPSI, LPSI and CS Pump handswitches in PTL to prevent spurious starts

Answer:

A. OP-2203.037, Loss of 125V DC AOP; cycle Charging pumps to maintain PZR level

Notes:

With the Green Battery disconnected from its bus [2D02] the battery charger is supplying all green DC loads. When the charger breaker trips open, a loss of green DC will occur. The unit will stay on line and the Loss of DC AOP would be entered. The outside containment letdown isolation valve [2CV-4823-2] is air operated and the air supply is controlled by a green DC solenoid. When green DC is lost, the valve will fail closed isolating letdown and require Pzr level to be maintained by cycling CCPs. The plausibility of B and D is based on the fact that placing the ESF pumps in PTL would be directed if a reactor trip were to occur during this event.

A. Correct: With the Green Battery disconnected from its bus [2D02] the battery charger is supplying all green DC loads. When the charger breaker trips open, a loss of green DC will occur.The unit will stay on line and the Loss of DC AOP would be entered. When green DC is lost, the valve will fail closed isolating letdown and cause Pzr level to be maintained by cycling CCPs.

B. Incorrect: The Loss of DC AOP is entered as stated above. Placing the SI and CS in PTL is plausible because this action is directed in section 3 of 2203.037 if a reactor trip occurs.

C. Incorrect. The Unit does not trip automatically nor is a manual reactor trip required per the Loss of 125V DC AOP. Plausible because a loss of Green DC does automatically open 2 trip circuit breakers but this does not cause a Reactor trip. Cycling charging pumps is correct

D. Incorrect. The Unit does not trip automatically nor is a manual reactor trip required per the Loss of 125V DC AOP. Plausible because a loss of Green DC does automatically open 2 trip circuit breakers but this does not cause a Reactor trip. Placing the SI and CS in PTL is plausible because this action is directed in section 3 of 2203.037 if a reactor trip occurs.

KA Match - Matches the KA because it asks knowledge of loads lost and impact.(Charging pumps must be cycled).

References:

AOP-2203.037 (013), Loss of 125V DC
AOP-2203.037 (013), Loss of 125V DC Tech Guide
STM 2-04 (031), Chemical and Volume Control System - simplified drawing on page 66
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Modified from NRC Exam Bank question 2003 which was used on the 2014 NRC Exam.
"Loss of 125vdc AOP" for part 1 is still correct but changed the condition from which component failed to what are the mitigating AOP actions.
REV. 1 This is a replacement question following the NRC 10 question review.
Changed order of distractors making "A" correct instead of "B".
Rev 2 - changed 'diagnose' to 'direct' in the stem.

LOSS OF 125V DC

PURPOSE

This procedure provides actions for a total Loss of Red Train or Green Train 125v Vital DC under all conditions.

ENTRY CONDITIONS

EITHER of the following:

- Vital Red Train DC buses 2D01, 2D21 and 2D23 de-energized.
- Vital Green Train DC buses 2D02, 2D22 and 2D24 de-energized.

Entry conditions are met

EXIT CONDITIONS

1. Reactor tripped and 2202.001, Standard Post Trip Actions entered.
2. ALL required actions of this procedure complete.

SECTION 3
GREEN TRAIN DC

INSTRUCTIONS | **CONTINGENCY ACTIONS**

*3. **CONTROL** RCS inventory as follows:

A. **CYCLE** Red Train Charging pump to maintain PZR level within 5% of setpoint.

Cycle charging pumps to maintain PZR level

B. **RECORD** Charging Header data on 2202.010 Attachment 44, Charging Header Data.

4. **REFER TO** TABLE 3 of Attachment B, Tech Spec/Technical Requirements Manual Cross-Reference, for a list of affected Technical Specifications and Technical Requirements.

1 gpm from each of the four reactor coolant pumps. The expected pressure drop across the letdown side (tube side) of the regenerative heat exchanger is 70 psi with letdown flow at 132 gpm.

2.1.5 Letdown Containment Isolation Valve 2CV-4823-2

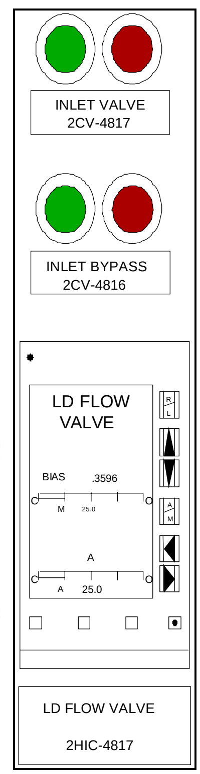
Fails closed on a Loss of DC power

The valve 2CV-4823-2 provides outside containment isolation capability on the letdown header. This valve is an air operated valve and is physically located in the upper south piping penetration room.

2CV-4823-2 is controlled from a handswitch on control room panel 2C09 with open and closed position indication also provided. It will automatically close and cannot be re-opened if a CIAS #2 signal is present.

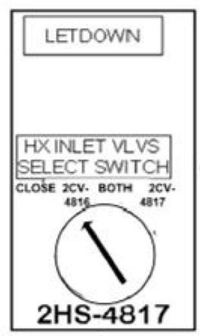
The valve will fail closed on a loss of instrument air or on a loss of DC power from 2D24-3.

2.1.6 Letdown Flow Control Valves 2CV-4816 / 2CV-4817



The letdown flow control valves are provided to control the letdown flow from the RCS in order to maintain RCS pressurizer level at the desired value. These valves are physically located in the upper south piping penetration room.

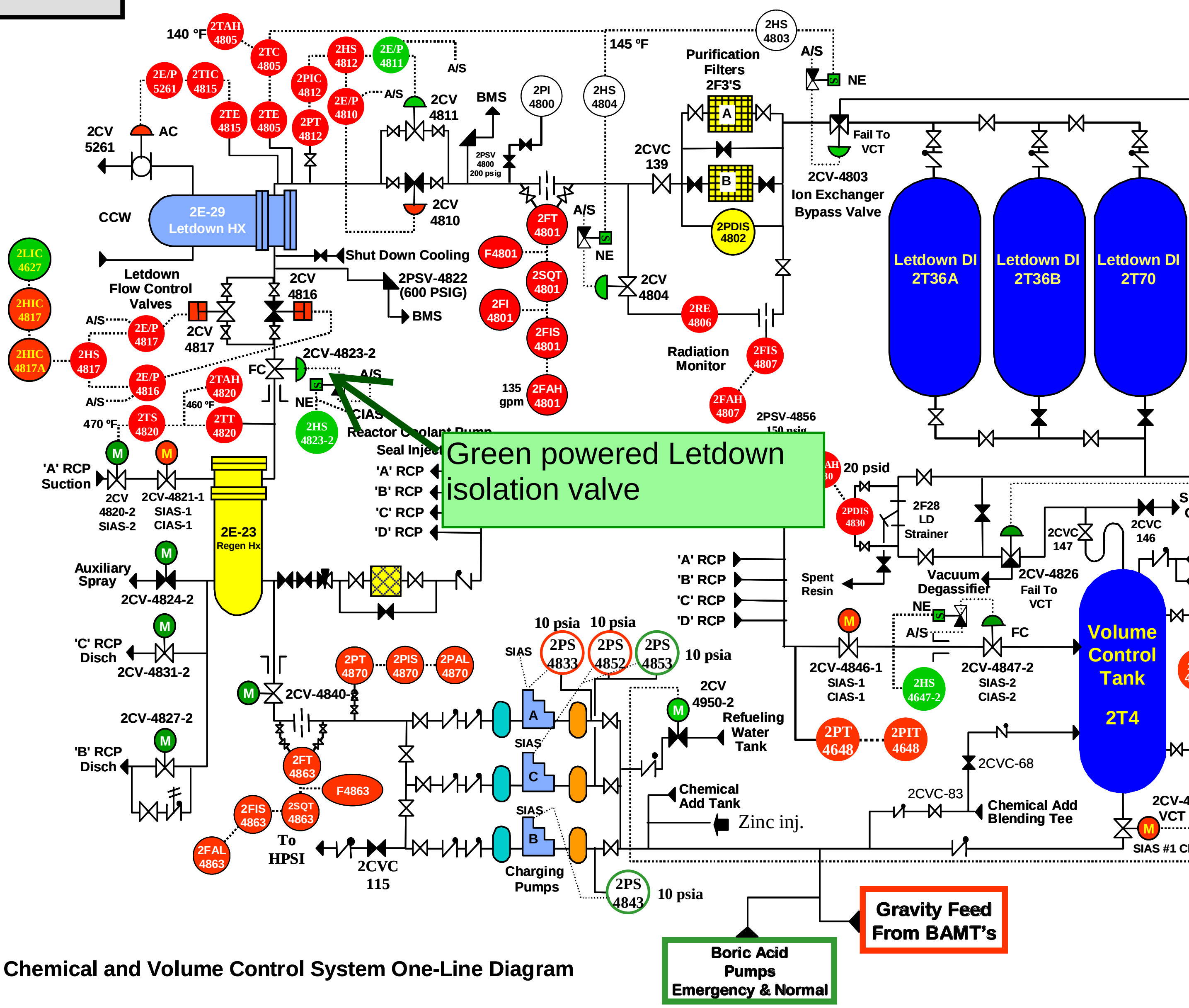
During system operation the control room operator has the capability for selecting one or both of these valves to control flow using 2HS-4817 on panel 2C09. Normally only one of these valves is in service and is controlled automatically by the letdown flow valve controller 2HIC-4817, located on panel 2C09. The close position allows the operator to apply a hard shunted close to both valves in the event of a fire that could threaten loss of control of either 2CV-4816 or 2CV-4817. This handswitch design inhibits spurious valve opening.



This Fisher Porter Controller is powered from 2Y1. If power is lost to the Fisher Porter Controller its output goes to zero. When power is restored to the Fisher Porter it will return in manual with the output that existed prior to the loss of power.

The controller will control the position of the selected valve(s) based on input from the pressurizer level control system. Automatic control of these valves is possible as long as the remote shutdown panel 2C80 controller 2HIC-4817A and 2HIC-4817 on control room panel 2C09 are *both* in automatic.

Chemical and Volume Control System



Chemical and Volume Control System One-Line Diagram

AOP STEP - SECTION 3 (GREEN TRAIN DC):

*3. **CONTROL** RCS inventory as follows:

BASIS:

Letdown isolation valve 2CV-4823-2 fails closed on loss of power to its air supply solenoid valve. This step directs the operator to manually control Charging pumps to maintain RCS inventory.

SOURCE DOCUMENTS:

1. E-2252, Letdown Containment Isolation Valve.

QID-78
6 of 11

Some reactor trip breakers will OPEN lending plausibility to C and D

INTRODUCTION:

This procedure provides actions for a loss of 125V DC event. It first determines which DC bus is de-energized. Actions are provided to restore DC power with the plant on-line. Actions are also provided if a reactor trip occurs while DC power is still lost. Under these circumstances power is restored to vital buses (if lost) from offsite power or the associated DG. It then restores the 125V DC battery chargers and buses.

Loss of either 125V DC bus will not generate a reactor trip in itself, but will open two Reactor Trip Switchgear Breakers. A loss of 2D01 will cause all Red train Charging pumps to stop due to loss of power to the low suction pressure trip relay. A loss of 2D02 will cause all Green train Charging pumps to stop due to loss of power to the low suction pressure trip relay.

2A1 and 2H1/2H2 have a backup source of DC control power (Refer to EC-48973 and EC-48975). 2A1 backup DC source is supplied from 2B32 or B43 (Unit 1) via Battery Eliminator 2D-50A1. 2H1 and 2H2 backup DC source is supplied from 2B32 or B43 (Unit 1) via Battery Eliminator 2D-50H1 and 2D-50H2 respectively. 2H1 and 2H2 backup DC is only supplied to the load breakers; supply breakers are supplied from 2D25 and have no backup DC.

Upon a loss of normal control power (Red or Green DC), the backup sources will provide DC for the affected buses. If these backup DC sources are available, then many of the actions previously performed on a loss of Red or Green DC do not need to be performed.

Note that 2A2 does NOT have backup DC control power.

If a reactor trip were to occur following the loss of DC and prior to placing the appropriate PPS Channel in Trip Channel Bypass, the following would occur if 2A1 does NOT have DC control power.

- A generator lockout will initiate contact closure to initiate a fast transfer of 4160V and 6900V buses. The 4160V buses will not transfer due to the absence of DC Control Voltage. The 6900V bus feeder breakers have black DC Control Voltage and will fast transfer to #3 SU Transformer.
- A reactor trip will not generate the normal generator anti-motoring protection because there is no turbine lockout due to the loss of Red DC. The generator output breakers will not trip until the 30 second time delay for the backup anti-motoring relay has timed out. A transfer to offsite power is not initiated for 2A1 due to the absence of DC control voltage to trip or close the switchgear feeder breakers to the bus.
- As a result of 2A1 not transferring, 2Y11, 2Y13 and 2Y1113 have no alternate 480V AC power supplies, nor do they have a DC power supply. Consequently 2RS1 and 2RS3 lose power. All PPS instrument loops fed from these two power supplies (except RWT Level Transmitters) fail low resulting in SIAS and MSIS actuations. EFAS does not actuate.
  The ESFAS actuations that actuate on a high signal should not actuate due to the reactor trip. The Channel 2 & 3 RWT Level Transmitters have auctioneered power supplies and will not lose power on an independent loss of Red AC or Green AC

SECTION 3
GREEN TRAIN DC

INSTRUCTIONS | **CONTINGENCY ACTIONS**

17. **PLACE** ALL the following pump handswitches in PTL:

- HPSI 2P89A (2HS-5078-1)
- HPSI 2P89B (2HS-5079-2)
- HPSI 2P89C (2HS-5080-1)
- HPSI 2P89C (2HS-5080-2)
- LPSI 2P60A (2HS-5018-1)
- LPSI 2P60B (2HS-5019-2)
- CNTMT Spray 2P35A (2HS-5623-1)
- CNTMT Spray 2P35B (2HS-5626-2)

Required actions if reactor trip does occur

18. **RECORD** time ESFAS components placed in PTL and **REFER TO** the following:

- Time ___________
- TS 3.5.2, 3.5.3 for HPSI/LPSI
- TS 3.0.3 for Spray

AOP STEP - SECTION 3 (GREEN TRAIN DC):

17. **PLACE** ALL the following pump handswitches in PTL:

BASIS:

This step directs the operator to place all ESF pumps in PTL. This action ensures that inadvertent pump starts do not occur when the 120 VAC Vital buses are energized.

SOURCE DOCUMENTS:

1. ANO-2 STM 2-05, Emergency Core Cooling System.
2. ANO-2 STM 2-08, Containment Spray System.
3. ANO-2 STM 2-53, Boron Management.

Plausibility of distractors B and D

AOP STEP - SECTION 3 (GREEN TRAIN DC):

18. **RECORD** time ESFAS components placed in PTL and **REFER TO** the following:

BASIS:

This step provides the operator a place to log entry into TS 3.5.2 (one hour action statement in this case), 3.5.3 and 3.0.3, (one hour action statement). It also reminds the operator to review the appropriate Technical Specifications.

SOURCE DOCUMENT:

1. ANO-2 Technical Specifications
2. CR-ANO-2-2013-0112 CA 6

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

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

This question requires assessment of plant conditions and determining the required procedural actions to mitigate those conditions

No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Bank: 2003 **Rev:** 3 **Rev Date:** 2/25/2014 2:07:11 **QID #:** 79 **Author:** foster

Lic Level: S **Difficulty:** 4 **Taxonomy:** H **Source:** NEW

Search 000058A203 **10CFR55:** 41.7 / 41.10 / 43.5 / 45. **Safety Function** 6

System Title: Loss of DC Power **System Number** 058 **K/A** AA2.03

Tier: 1 **Group:** 1 **RO Imp:** 3.5 **SRO Imp:** 3.9 **L. Plan:** A2LP-RO-ED125 **OBJ** 11

Description: Ability to determine and interpret the following as they apply to the Loss of DC Power: - DC loads lost; impact on ability to operate and monitor plant systems

Question:

Consider the following:

- Unit 2 is operating at 100% power
- 2D02 Green Battery disconnect has been opened for maintenance
- Battery charger, 2D-32B, is in service
- Now 2D-32B output breaker trips open

The CRS would diagnose entry into ______________________________ and Pressurizer level would be maintained by cycling a Coolant Charging Pump (CCP) due to ______________ letdown isolation valve failing closed.

A. OP-2203.037, Loss of 125V DC AOP; 2CV-4820-2, inside containment

B. OP-2203.037, Loss of 125V DC AOP; 2CV-4823-2, outside containment

C. OP-2202.001, Standard Post Trip Action EOP; 2CV-4820-2, inside containment

D. OP-2202.001, Standard Post Trip Action EOP; 2CV-4823-2 outside containment

Answer:

B. Correct

This is the parent question that has been modified to create Q #78 on the 2017 SRO exam.
"Loss of 125vdc AOP" for part 1 is still correct but changed the condition from which component failed to what are the mitigating AOP actions.

Notes:

With the Gree

breaker trips c

outside containment letdown isolation valve [2CV-4823-2] is air operated and the air supply is controlled by a green DC solenoid. When green DC is lost, the valve will fail closed isolating letdown and cause Pzr level to be maintained by cycling CCPs.

B. Correct: With the Green Battery disconnected from its bus [2D02] the battery charger is supplying all green DC loads. When the chargers breaker trips open, a loss of green DC will occur. The unit will stay on line and the Loss of DC AOP would be entered. The outside containment letdown isolation valve [2CV-4823-2] is air operated and the air supply is controlled by a green DC solenoid. When green DC is lost, the valve will fail closed isolating letdown and cause Pzr level to be maintained by cycling CCPs.

A. Incorrect: The Loss of DC AOP is entered but the Letdown Isolation valve inside containment does not automatically close upon a loss of Green DC. Plausible because letdown does isolate on a loss of Green DC, but it is the Letdown Isolation valve outside containment that closes not the valve inside containment.

C. Incorrect. The Unit does not trip automatically nor is it an action required by the AOP to trip the Reactor and go to SPTAs upon a loss of Green DC. Plausible because a loss of Green DC does automatically open 2 trip circuit breakers but this does not cause a Reactor trip.

D. Incorrect. The Unit does not trip automatically nor is it an action required by the AOP to trip the Reactor and go to SPTAs upon a loss of Green DC. Plausible because a loss of Green DC does automatically open 2 trip circuit breakers but this does not cause a Reactor trip.

References:

OP-2203.037, Loss of 125V DC AOP, Rev 009, entry page 1 of 57, section 3 step 2.A. page 33 of 57
OP-2203.037, Loss of 125V DC Tech Guide, Rev 009, section 3 step 2entry page 45 of 75
STM 2-04, Chemical and Volume Control System, Rev 29, section 2.1.5 page 6 and simplified drawing on page 62
STM 2-32-5, 125Vdc Electrical Distribution, Rev 19, section 2.7.2 page 15

Historical Comments:

Rev. 1; changed wording of stem. Added clarification in note section. Added STM 2-04 to the references for 2CV-4823-2 and STM 2-32-5 for modifications to the DC bus. Updated notes. [wmf 1/27/14]
Rev. 2: minor editorial change per NRC comment. (mwf 1/31/14)
Rev 3: changed correct answer from 'D' to 'B' and arranged answers IAW NRC directions [mwf 2/25/2014]

Question 79

Bank:	2409	**Rev:**	2	**Rev Date:**	11/1/2016	**2017 TEST QID #:**	79	**Author:**	Burton
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	0000562421	**10CFR55:**	43.5	**Safety Function**	6				
Title:	Loss of Offsite Power	**System Number**	056	**K/A**	2.4.21				
Tier:	1	**Group:**	1	**RO Imp:**	4.0	**SRO Imp:**	4.6	**L. Plan:**	A2LP-RO-ELOOP
OBJ	4								

Description: Emergency Procedures/Plan - Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc

Question:

Given the following:

* SPTAs have been completed.
* OP-2202.007, Loss of Offsite Power has been entered.
* 3 CEAs stuck fully out on the reactor trip.
* ATC has commenced Emergency Boration per Exhibit 1.
* Charging Pump, 2P-36A is running with a discharge flow of 43 gpm.
* 4160v Vital AC bus 2A4 is de-energized and all attempts to start 2EDG2 have failed.
* T-cold is 557°F and stable.
* T-hot is 586°F and stable.
* CET temperature is 587°F and stable.
* RCS pressure is 1900 psia and slowly rising.
* Pressurizer level 45% and slowly lowering.
* SG levels are 20% NR and rising with EFW flow at 500 gpm per SG.
* SG pressures are 1102 psia and slowly lowering.
* Containment pressure is 16.2 psia and stable.
* Containment temperature is 145°F and slowly rising.

Which of the following is TRUE based on an assessment of the Safety Function Status Checks listed in the Loss of Offsite Power EOP for the above conditions?

A. Reactivity Control is NOT met.

B. Maintenance of Vital Auxiliaries is NOT met.

C. Core Heat Removal is NOT met.

D. CNTMT Temperature and Pressure Control is NOT met.

Answer:

D. CNTMT Temperature and Pressure Control is not met.

Notes:

This question is based on simulator response which is (as required) modeled after the Unit. On a LOOP the normal chillers trip because they are powered from non-vital buses. If there is no ESFAS signals then the Containment Coolers run but with no cooling water causing containment pressure and temperature to rise. Step 22 of the LOOP procedure addresses the alignment of Service Water in order to stabilize containment parameters.

D is correct. Containment Temperature and Pressure Control is not met.
Containment temp and pressure are higher than standard post reactor trip due to the loss of chill water on the LOOP. CTPC Safety Function. Is not met due to Cntmt temp > 140 degrees and Cntmt press > 16 psia.

A. Incorrect
Reactivity Control is met. Plausible because Emergency boration is required due the Stuck CEAs and not all of the paths are available due to the loss of the 4160 vital AC bus, 2A4 the BAM pumps are not available but the gravity path using a charging pump is aligned and providing Emergency Boration.

B. Incorrect
Maintenance of Vital Auxiliaries are met.
Plausible because the electric plant is not in the standard post LOOP alignment with 2A4 and its associated buses de-energized. This requires the examine to determine that even with some vital busses de-energized acceptance criteria is still met.

C. Incorrect.
Core Heat Removal is met.
Plausible because the Examinee must evaluate core temperatures and MTS to determine from memory that Condition 2 criteria is not met, Condition 1 is therefore the Safety Function is satisfied. MTS is met but low because as Natural Circulation is being established T-hot is elevated.

KA Match - The examinee is asked to assess parameters and logic to determine the status of the Safety Functions for the LOOP procedure

References:

EOP-2207.007 (013), Loss of Offsite Power
EOP-2202.010 (023), Standard Attachments - Exhibit 1
OP-2107-002 (037), ESF Electrical System Operation
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Rev. 1 - Changed the wording of the stem as suggested. QID-11 has been modified so there is no cueing.
Rev. 2 - Changed Cntmt pressure to stable after discussion with NRC.

SAFETY FUNCTION STATUS CHECK

SAFETY FUNCTION **ACCEPTANCE CRITERIA**

Major Recovery Strategies for LOOP

- Restore Offsite power.
- Verify natural circulation.
- Restore forced circulation.
- IF RCS cooldown required, THEN perform RCS cooldown to SDC.

TIME: ____ ____ ____ ____

1. Reactivity Control

1. A. 1) Reactor power lowering.

OR

2) Reactor power less than 10^{-1}% AND stable or lowering.

B. 1) Maximum of ONE CEA NOT fully inserted.

OR

2) Emergency Boration in progress or completed.

EXHIBIT 1

EMERGENCY BORATION

1. Select ONE of the following Emergency Boration flowpaths:

FLOWPATH	ACTIONS REQUIRED
A. Gravity Feed	A. Verify at least ONE BAM Tank Gravity Feed valve open: • 2CV-4920-1 • 2CV-4921-1
B. BAM pumps	B. 1) Start at least ONE BAM pump. 2) Open Emergency Borate valve (2CV-4916-2). 3) Verify Boric Acid Makeup Flow Control valve (2CV-4926) closed.

With a LOOP and the failure of the 2EDG2 then 2A4 is de-energized. The BAM pumps are powered from the "B" train and not available.

CAUTION

Aligning Charging pump suction to RWT during RWT purification with ALL Charging pumps running may cause Charging pumps to trip due to low suction pressure.

C. RWT to Charging pumps	C. Open Charging Pump Suction Source From RWT valve (2CV-4950-2).

2. Close VCT Outlet valve (2CV-4873-1).

3. IF VCT Outlet valve does NOT close, THEN verify BAM Pumps Emergency Boration flowpath selected.

4. Verify Reactor Makeup Water Flow Control valve (2CV-4927) closed.

5. Verify at least ONE Charging pump running.

This path is available but not the normal source

6. Verify charging header flow greater than 40 gpm by either of the following:

- 2FIS-4863 Disch Flow (2C09)
- Computer Point F4863 (PDS, PMS or SPDS)

SAFETY FUNCTION STATUS CHECK

SAFETY FUNCTION		ACCEPTANCE CRITERIA	

2.	Maintenance of Vital Auxiliaries (AC and DC Power)	2. A. At least ONE 4160v Vital bus (2A3/2A4) energized.	☐☐☐☐
		B. At least ONE 125v Vital DC bus energized: • 2D01 – SPDS point E2D01 • 2D02 – SPDS point E2D02	☐☐☐☐
		C. At least ONE 120v Vital AC bus energized: • 2RS1 – SPDS point E2RS1 or E2RS1RS3 • 2RS2 – SPDS point E2RS2 or E2RS2RS4 • 2RS3 – SPDS point E2RS3 or E2RS1RS3 • 2RS4 – SPDS point E2RS4 or E2RS2RS4	☐☐☐☐
3.	RCS Inventory Control	3. A. PZR level maintained OR being restored to 10% to 80%.	☐☐☐☐
		B. RCS MTS 30°F or greater based on average CET temperature.	☐☐☐☐
		C. RVLMS LVL 03 or higher elevation indicates WET.	☐☐☐☐
4.	RCS Pressure Control	4. A. PZR heaters and spray maintaining RCS pressure within P-T limits, refer to 2202.010 Attachment 1, P-T Limits.	☐☐☐☐

Examine must determine the status and verify that ONE 4160v Vital as well as Vital AC and DC buses are energized. Electric plant conditions are not in an expected post LOOP alignment for plausibility.

SAFETY FUNCTION STATUS CHECK

SAFETY FUNCTION | **ACCEPTANCE CRITERIA**

NOTE

Meeting the provisions of Condition 1 or Condition 2 will satisfy this Safety Function.

5. Core Heat Removal

CONDITION 1

A. RCS loop ΔT less than 50°F in natural circulation (no RCPs running).

B. RCS MTS 30°F or greater based on average CET temperature.

This Safety Function is met for Natl Circulation but outside the band for Condition 2.

CONDITION 2

A. RCS loop ΔT less than 10°F with forced circulation (RCPs running).

B. RCS MTS 30°F or greater based on average CET temperature.

6. RCS Heat Removal

6. A. 1) At least ONE SG level maintained 10% to 90% with FW available.

OR

2) SG level being restored by total FW flow greater than 485 gpm.

B. Average CET temperature stable or lowering.

SAFETY FUNCTION STATUS CHECK

SAFETY FUNCTION | **ACCEPTANCE CRITERIA**

7. CNTMT Isolation

7. A. CNTMT pressure less than 16 psia.

B. CNTMT Area High Range Radiation monitors less than 1000 R/hr AND NO unexplained rise: (2C336-1, 2)

- 2RITS-8925-1
- 2RITS-8925-2

CTPC is NOT met -- Containment pressure and temperature limits are both exceeded. Temperature and pressure are higher than normal because Chill water was de-energized by the LOOP and Service Water is not aligned until a CCAS is initiated or manually aligned.

C. Secondary Systems Radiation alarms clear AND NO unexplained rise:

1) SG Sample (2C25)

- 2RITS-5854
- 2RITS-5864

2) Condenser Off Gas (2C25)

- 2RITS-0645

3) Steamline Monitors (2C336-2)

- 2RI-1007
- 2RI-1057

8. CNTMT Temperature and Pressure Control

8. A. CNTMT pressure less than 16 psia.

B. CNTMT temperature less than 140°F.

LOSS OF OFFSITE POWER

22. Align CNTMT cooling as follows:

A. Start ALL available CNTMT Cooling fans.

B. IF 2VSF-1A OR 2VSF-1B running,
THEN perform the following:

1) Open “2VSF-1A/B SW CLG Inlet” (2CV-1511-1).

2) Open “2VSF-1A/B SW CLG Outlet” (2CV-1519-1).

C. IF 2VSF-1C OR 2VSF-1D running,
THEN perform the following:

1) Open “2VSF-1C/D SW CLG Inlet” (2CV-1510-2).

2) Open “2VSF-1C/D SW CLG Outlet” (2CV-1513-2).

Steps performed in the LOOP procedure, due to the lack of Containment Cooling

***23. Check SG levels greater than 22.2%.**

***23. IF SG levels less than 22.2%,
THEN perform the following:**

A. Verify EFAS actuated on PPS inserts.

B. IF MSIS prevents feeding SGs,
THEN override and establish EFW to SGs using 2202.010 Attachment 46, Establishing EFW Flow.

C. IF total EFW flow less than 485 gpm,
THEN commence EFAS verification using 2202.010 Attachment 7, EFAS Verification.

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

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?
- Yes → RO question
- No ↓

Can the question be answered *solely* by knowing immediate operator actions?
- Yes → RO question
- No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?
- Yes → RO question
- No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?
- Yes → RO question
- No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

- Yes → SRO-only question
- No → Question might not be linked 10 CFR 55.43(b)(5) for SRO-

Verification of Safety Function status is the responsibility of an SRO. Given these conditions the SRO will have to determine if transitioning to another procedure is required.

Question 80

Bank:	2410	**Rev:**	2	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	80	**Author:**	Foster	
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NRC Exam Bank 2266			
Search	000057A218	**10CFR55:**	43.2	**Safety Function**	6					
Title:	Loss of Vital AC Electrical Instrument Bus	**System Number**	057	**K/A**	AA2.18					
Tier:	1	**Group:**	1	**RO Imp:**	3.1	**SRO Imp:**	3.1	**L. Plan:**	A2LP-RO-RPS	**OBJ** 11

Description: Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: - The indicator, valve, breaker, or damper position which will occur on a loss of power

Question:

Given the following:

* Unit 2 is operating at 100% power.
* PPS Channel 'C' Low SG 2 Level has been declared inoperable due to spurious trips and has been placed in trip channel bypass.
* Currently there is no low SG 2 Level trip in on Channel 'C' PPS.
* AC power is now lost to Low SG 2 level on Channel 'B' PPS.
* I&C has been contacted to initiate troubleshooting and repair on PPS Channel 'B' bistable.

Due to the loss of AC power and to comply with Tech Specs and allow I&C technicians to perform maintenance, the CRS should direct the CBOT to _______________, and if the inoperable channel is bypassed for greater than a MAXIMUM of ______ hours , the desirability of maintaining this channel in the bypassed condition shall be reviewed as soon as possible but no later than the next regularly scheduled OSRC meeting in accordance with the Quality Assurance Program Manual (QAPM).

A. place Channel B in bypass, place Channel C in trip, then remove Channel C from bypass; 48

B. remove Channel C from bypass, place Channel B in bypass, then place Channel C in trip; 48

C. place Channel B in bypass, place Channel C in trip, then remove Channel C from bypass; 72

D. remove Channel C from bypass, place Channel B in bypass, then place Channel C in trip; 72

Answer:

B. remove Channel C from bypass, place Channel B in bypass, then place Channel C in trip; 48

Notes:

B. Correct based on actions 2 and 3 of T.S. 3.3.1.1, the Tech Spec requires actions be performed within 1 hour to align the PPS system in a safe configuration (not cause or prevent a trip). To allow I&C to perform maintenance the effected channel has to be placed in bypass. The CRS should direct this sequence to prevent a plant trip, The PPS system is designed such that 2 channels cannot be placed in bypass at the same time, therefore to allow I&C to troubleshoot the B channel it has to be bypassed. To comply with T.S. 3.3.1.1 the inoperable channels either have to be bypassed or placed in trip. With C channel in bypass it would have to be taken out of bypass then the B channel could be placed in bypass, then C channel would be tripped. The T.S. also directs with the number of channels OPERABLE one less than the Total Number of Channels, operation in the applicable MODES may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour.
Per the BASES, if the inoperable channel is bypassed for greater than 48 hours, the desirability of aintaining this channel in the bypassed condition shall be reviewed as soon as possible but no later than the next regularly scheduled OSRC meeting in accordance with the Quality Assurance Program Manual (QAPM).

A. Incorrect but plausible based on the applicants ability to recall and apply the requirements of T.S. 3.3.1.1 and time limits and the sequence required to prevent both channels being in a tripped condition at the same time without one

channel being bypasses. If the RO depressed the B channel bypass switch first both channels (B and C) would come out of bypass (interlock to prevent more than one channel being in bypass at a time) When C channel is placed in trip the plant would trip due to the B channel is already tripped (2 channels in trip at the same time without one being in bypass)

C. Incorrect based on the time required to comply with T.S. 3.3.1.1 which requires a review if conditions are to be maintained >48 hours not 72 ours. Plausible as the 72 hour time line is in several T.S. statements have a 72 hour time limit to complete actions. Also this is the correct sequence for bypassing/tripping the channels to prevent a plant trip.

D. Incorrect based on the time required to comply with T.S. 3.3.1.1 which requires a review if conditions are to be maintained >48 hours not 72 ours. Plausible as the 72 hour time line is in several T.S. statements have a 72 hour time limit to complete actions. Also plausible bases on the applicants ability to recall and apply the requirements of T.S. 3.3.1.1 and the sequence required to prevent both channels being in a tripped condition at the same time without one channel being bypasses. If B channel was bypassed first then both channels would come out of bypass then when C channel was placed in trip the plant would trip (2 channels in trip at the same time without one being in bypass)

KA Match - The examinee will be required to know that a Loss of Vital AC power to RPS will cause ndicators and instruments to fail low resulting in a trip of RPS parameters such as SG level which have a "low" bistable/setpoint. This question requires the examine to determine bistable/parameter status, then interpret status of all the PPS channels and apply LCO guidance.

References:

STM 2-63 (012), Reactor Protection System
LCO 3.3.1.1 Action 2 and Action 3
(All references verified current 11/10/16)

Historical Comments:

NRC Exam Bank 2266 was used on the 2015 SRO Exam
Minor alterations include changing the affected channels from B and D to 'B' and 'C'.
Rotated distractors changing correct answer from A to B
To be used on the 2017 NRC Exam
Rev 1 - added 'maximum' to stem
Rev 2 - changed difficulty from 4 to 3 and capitalized MAXIMUM

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Neutron detectors are exempt from response time testing. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

4.3.1.1.4 The Core Protection Calculator System shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours.

4.3.1.1.5 The affected Core Protection Calculator Channel shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a valid CPC Cabinet High Temperature alarm.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2 2 sets of 2	1 set of 2 1 set of 2	2 sets of 2 2 sets of 2	1,2 3*,4*,5*	5 8
2. Linear Power Level – High	4	2	3	1,2	2,3
3. Logarithmic Power Level – High					
a. Startup	4	2(a)(d)	3	2,3*,4*,5*	2,3
b. Shutdown	4	0	2	3*,4*,5*	4
4. Pressurizer Pressure – High	4	2	3	1,2	2,3
5. Pressurizer Pressure – Low	4	2(b)	3		
6. Containment Pressure – High	4	2	3		
7. Steam Generator Pressure – Low	4/SG	2/SG	3/SG	1,2	2,3
8. Steam Generator Level – Low	4/SG	2/SG	3/SG	1,2	2,3
9. Local Power Density – High	4	2(c)(d)	3	1,2	2,3

Must comply with actions 2 and 3

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 2 – With the number of channels OPERABLE one less than the Total Number of Channels, operation in the applicable MODES may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed for greater than 48 hours, the desirability of maintaining this channel in the bypassed condition shall be reviewed as soon as possible but no later than the next regularly scheduled OSRC meeting in accordance with the Quality Assurance Program Manual (QAPM). The channel shall be returned to OPERABLE status prior to startup following the next COLD SHUTDOWN.

If the channel is bypassed greater than 48 hours

[illegible] a channel process measurement circuit that affects multiple functional units [illegible]rable or in test, bypass or trip all associated functional units as listed below.

[illegible]ss Measurement Circuit	Functional Unit Bypassed
1. Linear Power (Subchannel or Linear)	Linear Power Level – High Local Power Density – High DNBR – Low Log Power Level – High*
2. Pressurizer Pressure – NR	Pressurizer Pressure – High Local Power Density – High DNBR – Low
3. Containment Pressure – NR	Containment Pressure – High (RPS) Containment Pressure – High (ESFAS) Containment Pressure – High-High (ESFAS)
4. Steam Generator 1 Pressure	Steam Generator 1 Pressure – Low Steam Generator 1 ΔP (EFAS 1) Steam Generator 2 ΔP (EFAS 2)
5. Steam Generator 2 Pressure	Steam Generator 2 Pressure – Low Steam Generator 1 ΔP (EFAS 1) Steam Generator 2 ΔP (EFAS 2)
6. Steam Generator 1 Level	Steam Generator 1 Level – Low Steam Generator 1 ΔP (EFAS 1)
7. Steam Generator 2 Level	Steam Generator 2 Level – Low Steam Generator 2 ΔP (EFAS 2)
8. Core Protection Calculator	Local Power Density – High DNBR – Low

* Only for failure common to both linear power and log power.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 3 – With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, operation in the applicable MODES may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Linear Power Level – High Local Power Density – High DNBR – Low Log Power Level – High**
2. Pressurizer Pressure – NR	Pressurizer Pressure – High Local Power Density – High DNBR – Low
3. Containment Pressure – NR	Containment Pressure – High (RPS) Containment Pressure – High (ESFAS) Containment Pressure – High-High (ESFAS)
4. Steam Generator 1 Pressure	Steam Generator 1 Pressure – Low Steam Generator 1 ΔP (EFAS 1) Steam Generator 2 ΔP (EFAS 2)
5. Steam Generator 2 Pressure	Steam Generator 2 Pressure – Low Steam Generator 1 ΔP (EFAS 1) Steam Generator 2 ΔP (EFAS 2)
6. Steam Generator 1 Level	Steam Generator 1 Level – Low Steam Generator 1 ΔP (EFAS 1)
7. Steam Generator 2 Level	Steam Generator 2 Level – Low Steam Generator 2 ΔP (EFAS 2)
8. Core Protection Calculator	Local Power Density – High DNBR – Low

Operation in the applicable MODES may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent operation in the applicable MODES may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

** Only for failure or activities common to both linear power and log power.

Initiation Logic consists of the trip path power source, matrix relays and their associated contacts, all interconnecting wiring, and the initiation relays (including contacts).

ESFAS Actuation Logic consists of all circuitry housed within the Auxiliary Relay Cabinets (ARCs) used to house the ESF Function; excluding the subgroup relays, and interconnecting wiring to the initiation relay contacts mounted in the PPS cabinet.

For the purposes of this LCO, de-energization of up to three matrix power supplies due to a single failure, such as loss of a vital instrument bus, is to be treated as a single matrix channel failure, providing the affected matrix relays de-energize as designed to produce a half-trip. Although each of the six matrices within an ESFAS Function (e.g., SIAS, MSIS, CSAS, etc.) uses separate power supplies, the matrices for the different ESFAS Functions share power supplies. Thus, failure of a matrix power supply may force entry into the Condition specified for each of the associated ESFAS Functional Units.

Bases requires 48 hour OSRC review

Table 3.3-1 Action 2 and Table 3.3-3 Action 10 allow for continued o[illegible] MODES with the number of channels OPERABLE one less than the Total Number of Channels provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. Channels bypassed for greater than 48 hours require review by the OSRC no later than the next regularly scheduled OSRC meeting. A special OSRC meeting to accommodate review prior to the next regularly scheduled OSRC meeting is not required, but permitted if desired.

If an inoperable Steam Generator ΔP or Refueling Water Tank (RWT) Level - Low channel is placed in the tripped condition, it must be removed from the tripped condition within 48 hours or the plant must be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 30 hours. This condition is limited to 48 hours because of the single failure vulnerability that exists with one of the Steam Generator ΔP or RWT Level - Low channels in the trip condition.

Operation with a channel of Steam Generator ΔP (EFAS 1 or EFAS 2) in the tripped condition renders EFAS susceptible to single failure scenarios. With a channel of Steam Generator ΔP in trip, certain single failures concurrent with a MSIS actuation can result in initiation of EFW to a faulted Steam Generator. Other single failures can result in failure of automatic control of Steam Generator level and could allow Steam Generator overfill. Placing a channel of Steam Generator ΔP in the tripped condition is acceptable for up to 48 hours because operating experience has demonstrated the low probability of the above single failure scenarios to occur.

Operation with a channel of RWT Level - Low in the tripped condition renders the RAS susceptible to a single failure scenario. With a channel of RWT Level - Low in trip, concurrent with the injection phase of a valid SIAS actuation, and a single failure of another RWT Level - Low channel would result in a RAS Actuation. These sequence of events would cause the ECCS suction to be shifted from the RWT to the containment sump prematurely. Placing a channel of RWT Level - Low in the tripped condition is acceptable for up to 48 hours because operating experience has demonstrated the low probability of the above single failure scenario to occur.

If 2 bypasses are depressed for the same function then neither channel is in Bypass. Therefore "C" must be removed from Bypass before "B" is placed in Bypass

Again referring to the figure on page 75 , it can be seen that the Bypass switch contacts are arranged to automatically prevent bypassing any given parameter on more than one channel at the same time. When the Bypass switch is depressed in any single channel, the three remaining Bypass switches cannot energize their associated relays; but will, if depressed, remove all bypasses on that parameter. For example if the same parameter is bypassed on two or more channels at the same time, they will both be removed even though the Bypass push-buttons remain in.

When a system parameter is bypassed on any selected PPS channel, that bypass is applied to all Matrix Ladder contacts associated with that channel parameter. Enabling a Trip Channel Bypass therefore forces the PPS system to revert to a "two-out-of-three" logic configuration for the bypassed parameter.

The figure on page 76 illustrates a partial section of the “AB” RPS Matrix Ladder. When a Bistable Relay Card actuates/trips, it's contacts change state. Using A25 as an example, you can see that the contacts are in a tripped condition sending power to light the AB-1 lamp/indicator. By depressing the Bypass push-button, energizing Bypass relay K501, we close its contact. The now closed Bypass relay contact routes power from PS-9 around the tripped card contact to the next Bistable Relay Card A26 in the Ladder.

Indication of the status of any Trip Channel Bypass can be observed on the 4 PPS Remote Control Modules (RCM)in the Control Room. A picture of a RCM is shown on page 87. In addition, Control Room annunciators as well as indicator lamps located on each PPS Channel Bistable Control Panel are also provided to show Trip Channel Bypass status.

6.2.2 Operating Bypasses

Three Operating Bypasses are provided as part of the Plant Protection System to disable certain trips for the performance of testing while in a shutdown/cooled down condition and to allow plant startup or shutdown. The Operating Bypasses available are:

- High Logarithmic Power - Excore & PPS Logic
- Low RWT Level/Low PZR Press Bypass
- Hi/Low S/G Level Bypass

All of these Operating Bypass circuits are discussed in the following sections.

6.2.2.1 High Logarithmic Power - Excore & PPS Logic

The High Logarithmic Power - Excore & PPS Logic circuit provides contact logic for the following Operating Bypasses:

- DNBR Pretrip and Trip Bypass
- LPD Pretrip and Trip Bypass
- CEA Withdrawal Prohibit Bypass
- High Log Power Bypass

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

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- Knowledge of TS bases that is required to analyze TS required actions and terminology

Yes → SRO-only question

No ↓

Question not be linked to 1
55.43(b)(2) for SRO-only r

The question tests knowledge of 48 hour requirements contained within the action statements and Bases of LCO 3.3.1.1.
Bases knowledge makes this an SRO level question.

2015 NRC RO/SRO Exam

08-Nov-16

Bank: 2266 **Rev:** 0 **Rev Date:** 5/26/2015 2:04:15 **QID #:** 80 **Author:** foster

Lic Level: S **Difficulty:** 3 **Taxonomy:** H **Source:** MODIFIED NRC BANK QID #0441

Search 000057A218 **10CFR55:** 41.7 / 41.10 / 43.5 / 45 **Safety Function** 6

System Title: Loss of Vital AC Electrical Instrument Bus **System Number** 057 **K/A** AA2.18

Tier: 1 **Group:** 1 **RO Imp:** 3.1 **SRO Imp:** 3.1 **L. Plan:** A2LP-RO-RPS **OBJ** 10

Description: Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: - The indicator, valve, breaker, or damper position which will occur on a loss of power

Question:

ORIGINAL BANK QUESTION 2266 (2015 SRO Exam) was altered to create QID# 80 on the 2017 NRC Exam.

Consider the following:

- Unit 2 is operating at 100% power
- PPS Channel D Low SG Level Bistable has been declared inoperable due to spurious trips and has been placed in trip channel bypass
- AC power is now lost to the Low SG level bistable on channel "B" PPS
- I&C has been contacted to initiate troubleshooting and repair on PPS Channel B bistable
- Currently there is no low SG level bistable trip in on channel "D" PPS

Due to the loss of AC power and to comply with Tech Specs and allow I&C technicians to perform maintenance, the CRS should direct the CBOT to ________________________, and if the inoperable channel is bypassed for greater than ______ , the desirability of maintaining this channel in the bypassed condition shall be reviewed as soon as possible but no later than the next regularly scheduled OSRC meeting in accordance with the Quality Assurance Program Manual (QAPM).

A. remove channel D from bypass, place channel B in bypass, then place channel D in trip; 48 hours

B. place channel B in bypass, place channel D in trip, then remove channel D from bypass; 48 hours

C. remove channel D from bypass, place channel B in bypass, then place channel D in trip; 72 hours

D. place channel B in bypass, place channel D in trip, then remove channel D from bypass; 72 hours

QID use History

	RO	SRO
2005	☒	☒
2006	☒	☒
2008	☐	☐
2009	☐	☐
2011	☐	☐
2012	☐	☐
2014	☐	☐

Audit Exam History

2006	☒
2009	☐
2011	☒

Answer:

A. Correct

Notes:

A. Correct based on actions 2 and 3 of T.S. 3.3.1.1, the Tech Spec requires actions within 1 hour to align the PPS system in a safe configuration (not cause or prevent a trip). To allow I&C to perform maintenance the effected channel has to be placed in bypass. The CRS should direct this sequence is to prevent a plant trip, the PPS system is designed to not allow 2 channels to be placed in bypass at the same time, therefore to allow I&C to troubleshoot the B channel it has to be in bypassed. To comply with T.S. 3.3.1.1 the inoperable channels either have to be bypassed or placed in trip. With D channel in bypass it would have to be taken out of bypass then the B channel could be placed in bypass, then D channel would be tripped. The T.S. also directs with the number of channels OPERABLE one less than the Total Number of Channels, operation in the applicable MODES may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed for greater than 48 hours, the desirability of maintaining this channel in the bypassed condition shall be reviewed as soon as possible but no later than the next regularly scheduled OSRC meeting in accordance with the Quality Assurance Program Manual (QAPM).

B. Incorrect but plausible based on the applicants ability to recall and apply the requirements of T.S. 3.3.1.1 and time limits and the sequence required to prevent both channels being in a tripped condition at the same time without one channel being bypasses. If the RO depressed the B channel bypass switch first both channels (B and D) would come out of bypass (interlock to prevent more than one channel being in bypass at a time) When D channel is placed in trip the plant would trip due to the B channel is already tripped (2 channels in trip at the sam time without one being in bypass)
C. Incorrect based on the time required to comply with T.S. 3.3.1.1 which requires a review if conditions are to be maintained >48 hours not 72 ours. Plausible as the 72 hour time line is in several T.S. statements have a 72 hour time limit to complete actions. Also this is the correct sequence for bypassing/tripping the channels to prevent a plant trip.
D. Incorrect based on the time required to comply with T.S. 3.3.1.1 which requires a review if conditions are to be maintained >48 hours not 72 ours. Plausible as the 72 hour time line is in several T.S. statements have a 72 hour time limit to complete actions. Also plausible bases on the applicants ability to recall and apply the requirements of T.S. 3.3.1.1 and the sequence required to prevent both channels being in a tripped condition at th same time without one channel being bypasses. If B channel was bypassed first then both channels would come out of bypass then when D channel was placed in trip the plant would trip (2 channels in trip at the same time without one being in bypass)

References:

STM 2-63, Reactor Protection System, Rev. 12, section 6.2.1 pages 41 and 42
Unit 2 Tech Specs 3.3.1.1 Action 2 and Action 3

Historical Comments:

New for 2015 exam.
QID 0441 used on 2002 exam.

Question 81

Bank:	2411	**Rev:**	1	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	81	**Author:**	Burton		
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam				
Search	0000092430	**10CFR55:**	43.5	**Safety Function**	3						
Title:	Small Break LOCA	**System Number**	009	**K/A**	2.4.30						
Tier:	1	**Group:**	1	**RO Imp:**	2.7	**SRO Imp:**	4.1	**L. Plan:**	ASLP-RO-EPLAN	**OBJ**	6

Description: Emergency Procedures/Plan - Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

Question:

(REFERENCE PROVIDED)

Given the following sequence:

* Unit 2 is operating at 100% power.
* All 3 Charging pumps are running.
* Letdown is isolated.
* Containment pressure and humidity are slowly rising.
* Pressurizer level and pressure are lowering.
* A manual reactor trip is initiated.
* Pressurizer pressure stabilizes at 1500 psia.
* RCS T-cold is stable at 551°F.

This event should be classified as an ______ and the order of offsite notifications shall be ________ per OP-1903.010, EAL Classification.

A. NUE; State or Local agencies then the NRC

B. NUE; NRC then the State or Local agencies

C. Alert; State or Local agencies then the NRC

D. Alert; NRC then the State or Local agencies

Answer:

C. Alert; State or Local agencies then the NRC

Notes:

C is correct. Leakage is greater than capacity of the charging pumps since all 3 pumps are running and Pzr level and pressure are still lowering, making Alert the correct classification.
State and Local agencies must be notified within 15 minutes of classification and the NRC notification is after but within 1 hour.

A. Incorrect, NUE is plausible as SU7 is unidentified leakage > 10 gpm. Notifications are correct..

B. Incorrect, NUE is plausible as SU7 is unidentified leakage > 10 gpm. Notifications are wrong but plausible because NRC notification is required.

D. Alert is correct. Notifications are wrong but plausible because NRC notification is required but listed in the wrong order.

Meets the KA since the question conditions are a small break LOCA,, (no loss of subcooling) and requires knowledge of

the priority of off-site notifications.

References:

1903.010, (052) Classification (Attachments 1, 2 and 3 provided with exam)
1903.011-Y (043), Emergency Class Initial Notification Message
(All references verified current 11/10/16)
Modified copy of 1903.010 (Classification) is provided as a REFERENCE

Historical Comments:

To be used on the 2017 NRC Exam
Rev 1 - combined bullets so that there is only statement concerning Pzr level and pressure.
Verified that the modified copy of 1903.010, Classification (provided) does not provide information stating that the NRC is notified immediately after State and Local authorities. NO direct lookup.

QID-81
1 of 4

FISSION PRODUCT BARRIER
RCS

RCS Barrier EALs: RCB1 **OR** RCB2 **OR** RCB3 **OR** RCB4

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

1. RCS Leak Rate (RCB1)

Loss: RCS leak rate > available makeup capacity as indicated by:

Unit 1: Loss of adequate subcooling margin

Unit 2: RCS subcooling (MTS) can NOT be maintained at least 30 °F

Potential Loss:

Unit 1: UNISOLABLE RCS leak > 50 gpm with Letdown isolated

Unit 2: UNISOLABLE RCS leak > 44 gpm with Letdown isolated

Leakrate can be calculated at > 44 gpm therefore Alert is met

Basis:

Loss

This EAL addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak.

Potential Loss

This EAL is based on the apparent inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Makeup and Purification System (Unit 1) or the Chemical and Volume Control System (Unit 2).

Isolating letdown is a standard abnormal operating procedure action and may prevent unnecessary classifications when a non-RCS leakage path such as a Makeup and Purification System or CVCS leak exists. The intent of this condition is met if attempts to isolate Letdown are NOT successful. Additional charging pumps being required is indicative of a substantial RCS leak.

Reference Documents:

1. Unit 1 EOP 1202.013, Figure 1, *"Saturation and Adequate SCM"*
2. Unit 1 EOP Setpoint Document, Calculation 90-E-0116-07, Setpoint B.19
3. Unit 2 EOP 2202.009, *"Functional Recovery"*
4. Unit 2 EOP Setpoint Document, Calculation 90-E-0116-01
5. Unit 2 SAR Table 9.3-14, Charging Pumps Design Data

SYSTEM MALFUNCTION

SU7

Initiating Condition - NOTIFICATION OF UNUSUAL EVENT

RCS leakage

Operating Mode Applicability: Power Operations (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Example Emergency Action Level(s): (1 or 2)

Leakrate is unidentified and > 10 gpm
NUE is plausible

1. Unidentified or pressure boundary leakage > 10 gpm.

OR

2. Identified leakage > 25 gpm.

Basis:

With respect to this IC, RCS leakage is defined as a loss of RCS inventory due to a leak in the RCS or a supporting system that is not or cannot be isolated within 10 minutes. For example, isolation of the RCS Letdown (purification) system is a standard abnormal operating procedure action and may prevent unnecessary classifications when a non-RCS leakage path leak exists. However, the intent of this condition is met if attempts to isolate the RCS leak are NOT successful.

This IC is included as an NUE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified or pressure boundary leakage was selected as it is observable with normal Control Room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances).

Relief valve normal operation should be excluded from this IC. However, a relief valve that operates and fails to close per design should be considered applicable to this IC if the relief valve cannot be isolated.

The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. Steam generator tube leakage is identified leakage. In either case, escalation of this IC to the Alert level is via Fission Product Barrier Degradation (F) ICs.

ACTIONS FOR INITIAL NOTIFICATION

This form is used to notify the NRC, State and local governments of the following:

- Emergency Class Declaration
- Emergency Class Change (Upgrade or Downgrade)
- PAR Change

State and Local within 15 minutes
NRC immediately following

The Arkansas Department of Health (ADH) and other offsite response organizations **SHALL** be notified within **15 minutes** of any of the above events.

The Nuclear Regulatory Commission (NRC) **SHALL** be notified **immediately** following notification of the ADH and **SHALL NOT** exceed **1 hour** following the declaration of an emergency class.

ERDS must be initiated within 1 hour of the declaration of an **ALERT or higher** emergency class.

NOTE

The material contained within the symbols (*) throughout this form is proprietary or private information.

The Emergency Telephone Directory contains the emergency telephone numbers that you may need to complete this notification.

Computer generated Form 1903.011-Y may be used for notifications. The computer generated form is not an identical copy to the hard copy form, but contains all necessary information.

INSTRUCTIONS

1.0 Complete Initial Notification Message in accordance with Step 1.1 Computerized Notification Method **OR** Step 1.2 Manual Notification Method. (Computerized Notification Method preferred)

1.1 Computerized Notification Method

1.1.1 **IF** the Computerized Notification Method fails while performing notifications,
THEN go to the "Manual Notification Method" Step 1.2.

1.1.2 **IF** not already logged onto the notifications computer,
THEN perform the following:

A. Sign onto the computerized notification system computer using your Entergy logon ID and password. Control Room may use a generic ID and password.

B. Verify your computer is connected to a local or network printer in your area.
[Start]→[Settings]→[Printers and Faxes]

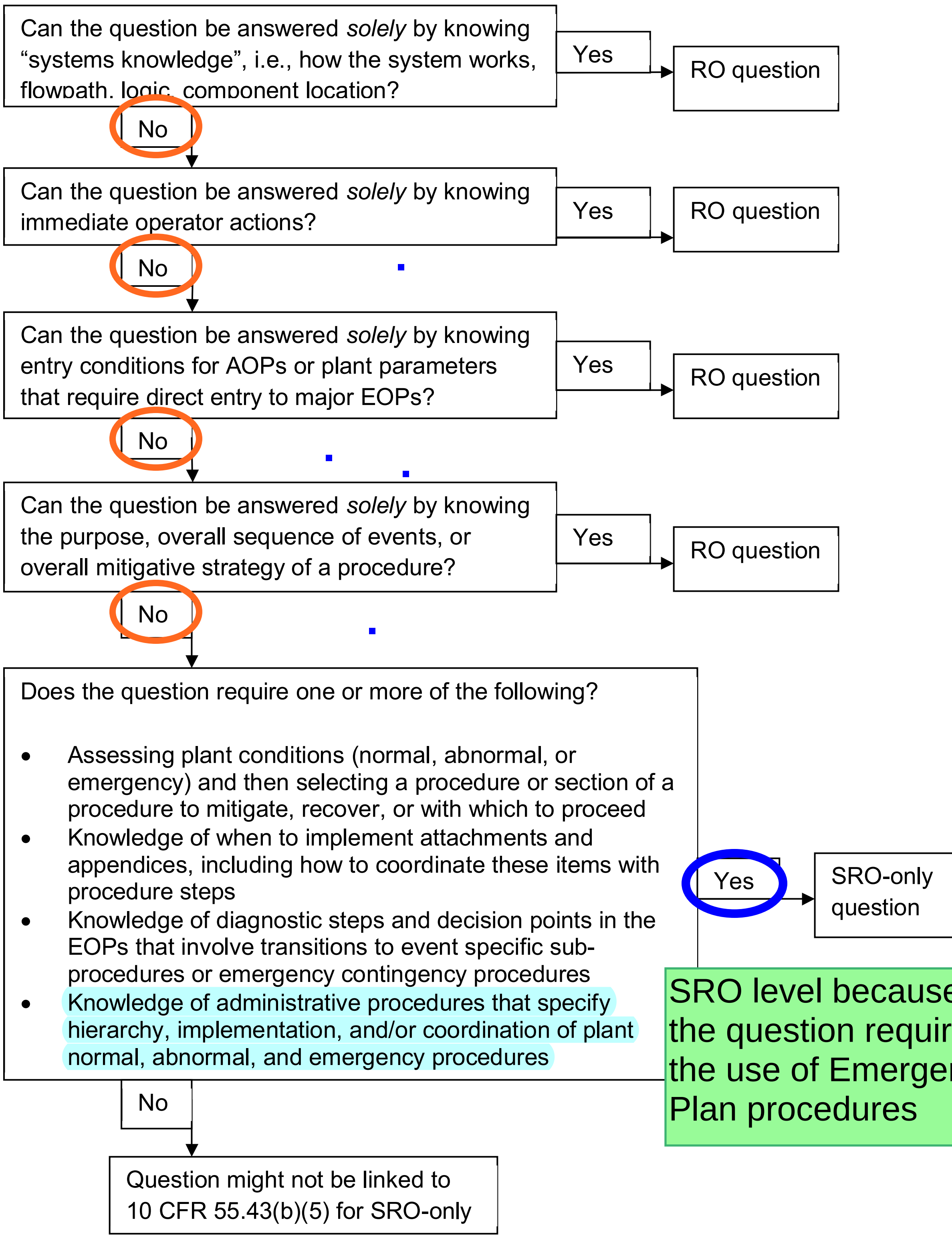
1.1.3 On the desktop double click the "EP Notification" icon **OR** select [Start], [(All) Programs], [EP Notifications], [EP Notifications Version XXXX] to start notification program.

1.1.4 Enter the appropriate data into the data fields for the Initial Notification Message. Use the [Tab] key (preferred) or mouse to navigate through the form. Refer to Emergency Class Notification Instructions page 8 of this form as needed.

1.1.5 **WHEN** the data fields are populated,
THEN press the [Create PDF only] button.

1.1.6 **IF** you receive an error message (i.e. "You have not correctly entered all the required data on Tab…"),
THEN review the form and make corrections.
Go to Step 1.1.5 above.

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



SRO level because the question requires the use of Emergency Plan procedures

Question 82

Bank:	2412	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	82	**Author:**	Larry Burton
Lic Level:	SRO	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	000069A202	**10CFR55:**	43.2			**Safety Function**	5		
Title:	Loss of Containment Integrity					**System Number**	069	**K/A**	AA2.02
Tier:	1	**Group:**	2	**RO Imp:**	3.9	**SRO Imp:**	4.4	**L. Plan:**	A2LP-RO-TS
OBJ	4								

Description: Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: - Verification of automatic and manual means of restoring integrity

Question:

Given the following:

* Unit 2 is "heating up" following a refueling outage.
* RCS pressure is 800 psia.
* RCS Tave is 260 degrees.
* RCS Sample valve, 2SV-5833-1 has failed OPEN and been declared Inoperable.

LCO 3.6.3.1, Containment Isolation Valves, _____ applicable under current plant conditions and the following actions will be required to satisfy the LCO 3.6.3.1 if OR when it is required.

A. is; Verify Operability and close the associated containment isolation valve.

B. is NOT; Verify Operability and close the associated containment isolation valve.

C. is; Deactivate and secure the associated containment isolation valve in the isolated position.

D. is NOT; Deactivate and secure the associated containment isolation valve in the isolated position.

Answer:

C. is; Deactivate and secure the associated containment isolation valve in the isolated position.

Notes:

LCO 3.6.3.1 is applicable in Mode 1-4, for the given conditions the Unit is in Mode 4.
LCO 3.6.3.1 condition b. requires that the penetration be Isolated within 4 hours by a deactivated automatic valve secured in the isolated position.

C is correct. Per the conditions, the unit is in Mode 4 so the LCO 3.6.3.1 applies and per the LCO the correct action is isolate the affected penetration by at least one deactivated valve secured in the isolation position

A is incorrect. Per the conditions, the unit is in Mode 4 so the LCO 3.6.3.1 applies. Closing an Operable containment isolation is plausible because this will satisfy LCO 3.6.1.1 Containment Integrity requirements

B. Is incorrect. Plant is in Mode 4, does not apply is plausible because two of the Containment system LCOs only apply in Modes 1-3. Closing an Operable containment isolation is plausible because this will satisfy LCO 3.6.1.1 Containment Integrity requirements.

D is incorrect. Plant is in Mode 4, does not apply is plausible because two of the Containment system LCOs only apply in Modes 1-3. Second part is correct.

KA Match - The second part of the question asks the manual means required to restore integrity per LCO 3.6.3.1. Maintain containment isolation is needed to ensure Containment Integrity

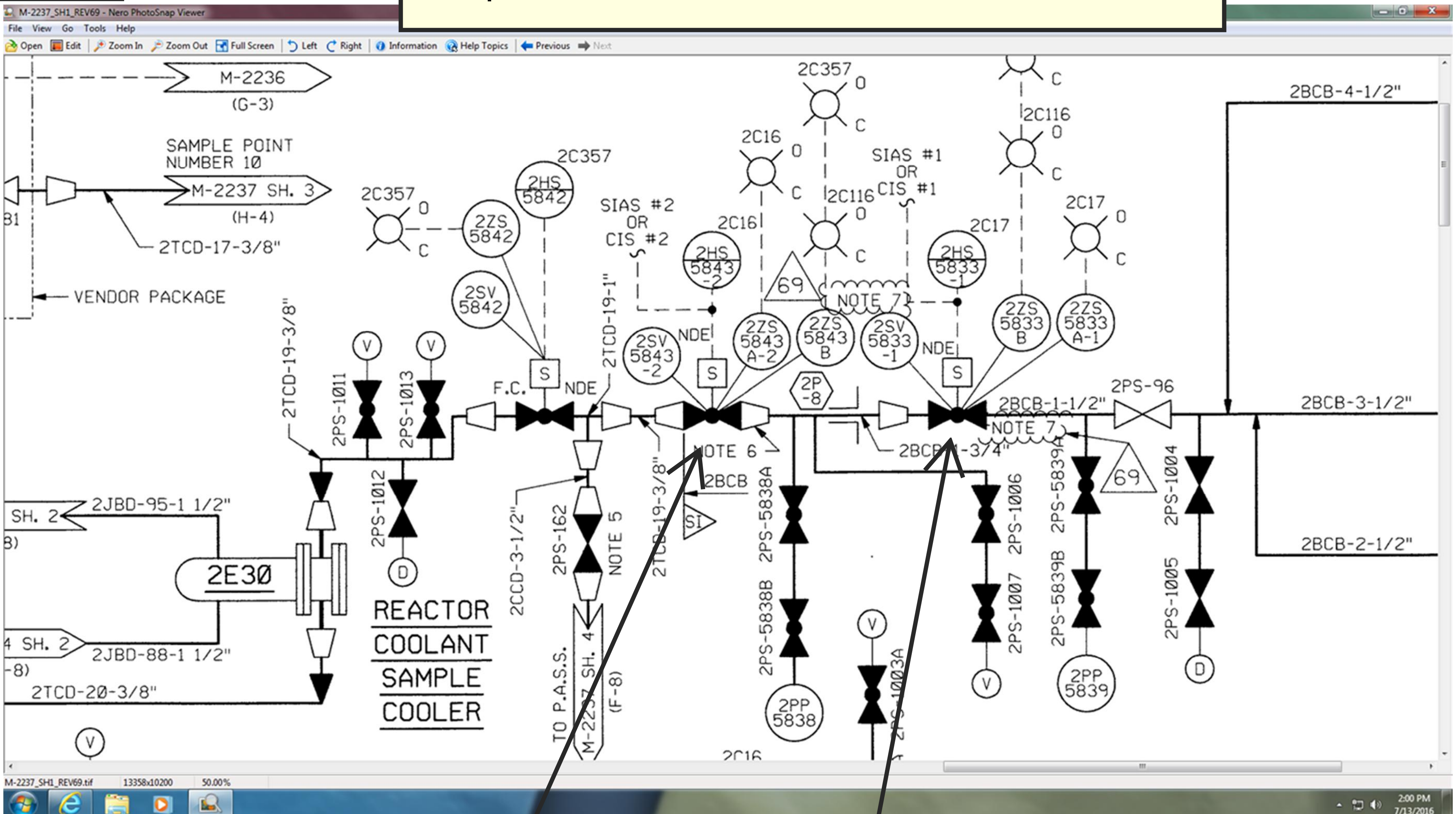
References:

LCO 3.6.3, Containment Isolation Valves
Mode 4 and Containment Integrity definitions.
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Rev 1 - removed attachment and changed wording in stem to say "associated containment isolation valve"

This picture is NO longer provided as part of the question stem.



Containment Isolation Valve

Failed sample valve

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.6.3.1.1 Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

* Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 300°F$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 300°F$
3. HOT STANDBY	< 0.99	0	$\geq 300°F$
4. HOT SHUTDOWN	< 0.99	0	$300°F > T_{avg} > 200°F$
5. COLD SHUTDOWN	< 0.99	0	$\leq 200°F$
6. REFUELING**	≤ 0.95	0	$\leq 140°F$

260 degrees is
Mode 4

* Excluding decay heat.

** Reactor vessel head unbolted or removed and fuel in the vessel.

Containment Integrity is satisfied by an Operable automatic isolation valve

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

1.8.2 All equipment hatches are closed and sealed,

1.8.3 Each airlock is OPERABLE pursuant to Specification 3.6.1.3,

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.8.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

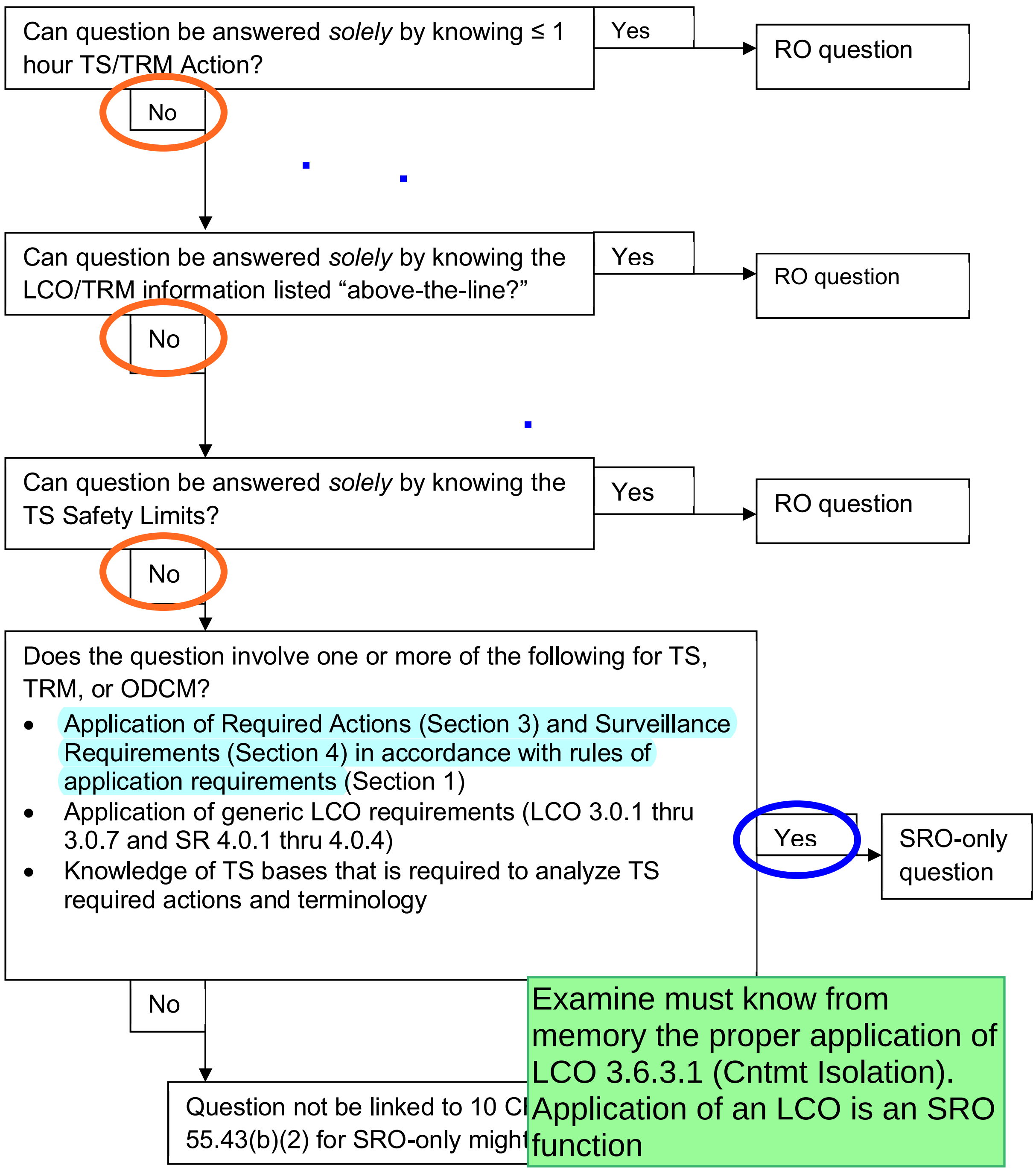
CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

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



Question 83

Bank:	2413	**Rev:**	2	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	83	**Author:**	Burton
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	000076A202	**10CFR55:**	43.5	**Safety Function**	9				
Title:	High Reactor Coolant Activity	**System Number**	076	**K/A**	AA2.02				
Tier:	1	**Group:**	2	**RO Imp:**	2.8	**SRO Imp:**	3.4	**L. Plan:**	A2LP-RO-TS
OBJ	4								

Description: Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: - Corrective actions required for high fission product activity in RCS

Question:

Given the following:

* Unit 2 is stable at 100% power.
* "LETDOWN RADIATION HI/LO" annunciator (2K12-A1) alarms.
* OP-2203.020, High Activity in RCS, has been entered.
* Chemistry reports that Dose Equivalent Iodine 131 activity is 80 µCi/gram.
* Aux. Building Area Radiation monitors are 0.2 R/hr and rising.
* RCS Letdown Gross monitor (2RITS-4806A) activity is slowly rising.

This exceeds the LCO 3.4.8, Specific Activity, I-131 limit of ______ µCi/gram and based on these conditions, OP-2203.020 provides the Shift Manager direction to isolate _____________________.

A. 0.1; Letdown only

B. 1.0; Letdown only

C. 0.1; Letdown and RCP Bleedoff to VCT

D. 1.0; Letdown and RCP Bleedoff to VCT

Answer:

D. 1.0; Letdown and RCP Bleedoff to VCT

Notes:

D is correct. LCO 3.4.8, Specific Activity for I-131 is 1.0 µCi/gram. AOP 2203.020 step 9 contingency actions allows SM discretion to isolate Letdown on high Aux Building rad levels and then RCP Bleedoff if levels are still high.
LCO 3.7.1.4, Plant Systems, Activity limit for secondary I-131 is less than or equal to 0.1 µCi/gram. NOT Isolating RCP Bleedoff is plausible since this action is an action only taken if the RCP is secured as found in 2203.025, RCP Emergencies and will cause alarms in the CR.

A. is Incorrect but plausible because 0.1 is the secondary TS limit. Not isolating RCP Bleedoff is plausible because this is an action that is found in RCP Emergencies with a Loss of CCW and is only directed if the RCP is secured therefore the examine might assume CBO cannot be isolated to a running RCP.

B. Is incorrect but plausible since 1.0 is the correct LCO limit. Not isolating RCP Bleedoff is plausible because this is an action that is found in RCP Emergencies with a Loss of CCW and is only directed if the RCP is secured therefore the examine might assume CBO cannot be isolated to a running RCP. When the contingency action is taken then it must be completed in it's entirety. With a rising trend in the Aux Building the and the contingency implemented then isolating CBO is required.

C. Is Incorrect but plausible because 0.1 is the secondary TS limit and isolating both Letdown and RCP Bleedoff is directed by the AOP.

KA matches because the question is directly related to high activity in the RCS and the Corrective Actions within the associated AOP. Knowledge of LCO entry requirements will also direct plant shutdown

References:

LCO 3.4.8, Specific Activity
LCO 3.7.1.4, Plant Systems Activity
AOP-2203.020 (012), High Activity in RCS
AOP-2203.025 (018), RCP Emergencies
ACA-2203.012K (046), Annunciator 2K11 Corrective Actions
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Rev 1 - modified stem to add a trend to aux building monitor readings. Added information stating why B is not correct since the entire contingency step must be completed not just sub step A.
Rev 2 - removed "if radiation levels continue to rise" from distractors C and D

INSTRUCTIONS		CONTINGENCY ACTIONS	
*6.	**Check RCS activity limits <u>NOT</u> exceeded:**	*6.	**Refer to 1903.010, Emergency Action Level Classification to determine emergency classification.**

- RCS specific activity by grab sample:
  - Greater than 300 μCi/g Dose Equivalent I-131
  - Greater than 60 μCi/g Dose Equivalent I-131
  - Greater than 1 μCi/g Dose Equivalent I-131 **for more than 48 hours**
- RCS radiation levels:
  - Greater than 1000 mR/hr measured at 2TCD-19
  - Letdown Iodine 131 monitor (2RITS-4806B) activity greater than 0.65 E5 CPM
- RCS activity greater than 3100 μCi/g Dose Equivalent **Xe-133** by grab sample **for more than 48 hours**

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION

Note: The provisions of Specification 3.0.4.c are applicable to ACTION a and b.

a. With the DOSE EQUIVALENT I-131 not within limit:

   1. Verify DOSE EQUIVALENT I-131 ≤ 60 μCi/gm once every 4 hours, and

   2. Restore DOSE EQUIVALENT I-131 within limit within 48 hours.

b. With the DOSE EQUIVALENT XE-133 not within limit, restore DOSE EQUIVALENT XE-133 within limit within 48 hours.

c. With the requirements of ACTION a and/or b not met, or with DOSE EQUIVALENT I-131 > 60 μCi/gm, be in at least HOT STANDBY in 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1 Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 3100 μCi/gm once every 7 days.*

4.4.8.2 Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm:*

LCO limit

   a. once every 14 days, and

   b. between 2 and 6 hours after THERMAL POWER change of ≥ 15% RATED THERMAL POWER within a 1 hour period.

* Only required to be performed in MODE 1.

3/4.4.8 SPECIFIC ACTIVITY (continued)

The SGTR analysis assumes that offsite power is lost at the same time the tube rupture occurs. Thus, radioactively contaminated steam discharges to the atmosphere through the atmospheric dump valves or the main steam safety valves. The atmospheric discharge from the ruptured SG stops when the operator isolates the SG at 60 minutes into the event. The unaffected SG then removes core decay heat by venting steam until the cooldown ends and the Shutdown Cooling System (SDC) is placed in service.

The MSLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SG removes core decay heat by venting steam to the atmosphere until the cooldown ends and the SDC system is placed in service. The event ends when the RCS temperature reaches 212 °F and no further flashing in the affected SG can occur. The MSLB is assumed to result in an increase in the total SG tube leakage rate to 1 gpm.

APPLICABLE SAFETY ANALYSES (continued)

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 60.0 μCi/gm for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS Specific Activity satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO 3.4.8 limits

LCO

The iodine specific activity in the reactor coolant is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 3100 μCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will not exceed the applicable 10 CFR 50.67 requirements (Ref. 1).

The MSLB and SGTR accident analyses show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a MSLB or SGTR, lead to doses that exceed the applicable 10 CFR 50.67 requirements (Ref. 1).

APPLICABILITY

In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a MSLB or SGTR to within the applicable 10 CFR 50.67 requirements (Ref. 1).

In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be ≤ 0.10 μCi/gram DOSE EQUIVALENT I-131.

LCO limit for the secondary

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the specific activity of the secondary coolant system > 0.10 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-2.

INSTRUCTIONS	CONTINGENCY ACTIONS

***9. Check BOTH the following conditions exist:**

- Aux Building Area Radiation monitors less than 0.1 R/hr.
- RCS Letdown Gross monitor (2RITS-4806A) activity stable or lowering.

***9. At SM discretion perform the following:**

A. Isolate Letdown by verifying at least ONE Letdown Isolation valve closed:

- 2CV-4820-2
- 2CV-4821-1
- 2CV-4823-2 (least preferred)

B. Maintain PZR level within 5% of setpoint by cycling Charging pumps.

C. Record Charging Header data on 2202.010 Attachment 44, Charging Header Data.

D. IF plant shutdown required, THEN perform the following:

1) Refer to applicable reactivity plan.

2) Commence Plant shutdown using 2102.004, Power Operation.

E. Refer to 1903.010, Emergency Action Level Classification.

F. IF Letdown isolated AND Aux Building radiation high, THEN close RCP Bleedoff to VCT valves:

- 2CV-4846-1
- 2CV-4847-2

Procedure allows the SM discretion to isolate both Letdown and RCP Bleedoff

Additional Information -
If the contingency actions are taken then all steps must be completed A-F. Therefore Letdown only is not true with a rising trend in the Aux Building.

10. Notify RP to perform Aux Building area radiation surveys.

END

QID-83
6 of 8

ANNUNCIATOR 2K11

G-3

RCP BLEEDOFF FLOW HI/LO

1.0 CAUSES

1.1 Controlled Bleedoff flow ≥ 1.5 gpm or ≤ 0.6 gpm for any Reactor Coolant Pump:

- 2P-32A (2FS-6008)
- 2P-32B (2FS-6018)
- 2P-32C (2FS-6028)
- 2P-32D (2FS-6038)

Alarms occur if RCP Bleedoff is NOT between 0.6 and 1.5 gpm
Plausibility of A and B

2.0 ACTION REQUIRED

NOTE

Annunciator Reflash unit (2K420) is located in back of 2C14 and has the following applicable alarm indications:

- RCP 2P-32A Controlled Bleedoff Flow HI/LO
- RCP 2P-32B Controlled Bleedoff Flow HI/LO
- RCP 2P-32C Controlled Bleedoff Flow HI/LO
- RCP 2P-32D Controlled Bleedoff Flow HI/LO

2.1 Check RCP chart recorders, PMS/PDS trends, and Annunciator Reflash unit (2K420) as necessary to determine affected pump.

2.2 To allow reflash, perform the following:

- Acknowledge alarm by pressing “ACK ALM” on affected RCP chart recorder.
- Acknowledge RIS unit (2K420) in back of 2C-14.

* 2.3 Monitor affected RCP(s) CBO temperature and seal pressures on PMS/PDS.

2.4 Refer to RCP Emergencies (2203.025).

2.5 <u>IF</u> RCP BLEEDOFF FLOW HI/LO alarm becomes nuisance, <u>THEN</u> alarm can be locked in using Bypass switch located under 2K420 in 2C14 IAW Annunciator Removal From Service Or Modification Form 1015.001B, Conduct Of Operations (1015.001).

3.0 TO CLEAR ALARM

3.1 Restore Controlled Bleedoff flow of 0.6 to 1.5 gpm.

4.0 REFERENCES

4.1 E-2457-1

4.2 DCP-91-2012

INSTRUCTIONS | **CONTINGENCY ACTIONS**

(...inued)

CAUTION

Failure to isolate Controlled Bleedoff may result in failure of RCP seals.

4) **ISOLATE** Controlled Bleedoff as follows:

   a) **CLOSE** RCP Bleedoff to VCT valve (2CV-4847-2).

   b) **CLOSE** RCP Bleedoff Isolation to VCT valve (2CV-4846-1).

   c) **CLOSE** RCP Bleedoff Relief to Quench Tank valve (2CV-4856).

5) **IF** Reactor manually tripped, **THEN GO TO** 2202.001, Standard Post Trip Actions.

These steps are located within the Loss of CCW section of the RCP Emergencies AOP. Isolating RCP bleedoff usually associated with a loss of CCW and a secured RCP

NOTE

If a CIAS relay has actuated, it will be necessary to override CCW Containment Isolation valves. In this case, TS 3.0.3 will be applicable until a dedicated operator can be established.

D. **CHECK** the following valves open:

- CCW CNTMT Supply valve (2CV-5236-1).
- RCP CCW Return valve (2CV-5254-2).
- RCP CCW Return valve (2CV-5255-1).

D. **RESTORE** CCW to RCPs as follows:

1) **OPEN** CCW CNTMT Supply valve (2CV-5236-1).
2) **OPEN** RCP CCW Return valve (2CV-5254-2).
3) **OPEN** RCP CCW Return valve (2CV-5255-1).
4) **IF** ANY CCW pump running, **THEN GO TO** step 4.

E. **ENSURE** in-service CCW Surge Tank (2T-37A/B) level greater than 13% on 2LIS-5210/5214.

(Step 2 continued on next page)

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

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing immediate operator actions?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?
Yes → RO question
No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question
No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

SRO level because the question requires the application of Contingency steps. This is not the purpose, sequence or overall mitigating strategy of the procedure

Question 84

Bank:	2414	**Rev:**	3	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	84	**Author:**	Burton
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	00CA162244	**10CFR55:**	43.2	**Safety Function**	2				
Title:	Excess RCS Leakage	**System Number**	A16	**K/A**	2.2.44				
Tier:	1	**Group:**	2	**RO Imp:**	4.2	**SRO Imp:**	4.4	**L. Plan:**	A2LP-RO-ALEAK
OBJ	4								

Description: Equipment Control - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Question:

Given the following:

* Unit 2 is operating at 100% power.
* OP-2203.016, Excess RCS Leakage, has been entered due to indications of an RCS leak.
* Leakage is Pressure Boundary leakage into the Component Cooling Water system.
* Crew starts an additional Charging Pump.
* Letdown flow stabilizes at 60 gpm.

Per LCO 3.4.6.2 RCS Operational Leakage the unit must ____________. If a shutdown to HOT STANDBY is required, then AOP-2203.016 (Excess RCS Leakage) directs the use of __________ to perform the RCS boration IAW OP-2102.004, Power Operations.

A. be in at least HOT STANDBY within 6 hours; Exhibit 3, Normal RCS Boration at Power

B. be in at least HOT STANDBY within 6 hours; Attachment R, RCS Boration from the RWT or BAMT

C. reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours; Exhibit 3, Normal RCS Boration at Power

D. reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours; Attachment R, RCS Boration from the RWT or BAMT

Answer:

B. be in at least HOT STANDBY within 6 hours; Attachment R, RCS Boration from the RWT or BAMT

Notes:

Given the information that the leakage is Pressure Boundary, there is no time allowance to reduce the leakage prior to taking actions to be in Hot Standby.
Information is available in the stem to determined the leakrate is ~ 24 gpm

A. Plausible because this is the correct application of the LCO but wrong boration flowpath. This boration source is permitted if the leakrate is < 10 gpm. Examinee can determine from control board indications that the leakrate is ~ 24 gpm.

B. Correct application of the LCO, pressure boundary leakage. Be in at least HOT STANDBY within 6 hours; Attachment R, RCS Boration from the RWT/BAMT is correct since leakrate is greater than 10 gpm

C. Plausible because this is one of the applications of the LCO (unidentified leakage) Reduce the leakage rate to within limits within 4 hours or be in at least Hot Standby within the next 6 hours; Exhibit 3, Normal

RCS Boration at Power - This boration source is permitted if the leakrate is < 10 gpm. Examinee can determine that the leakrate is ~ 24 gpm, making this a plausible choice.

D. Plausible because this is one of the applications of the LCO (unidentified leakage) Reduce the leakage rate to within limits within 4 hours or be in at least Hot Standby within the next 6 hours; Correct boration source for a leakrate ≥ 10 gpm, examinee can determine leakrate ~ 24 gpm from control board indications.

KA Match - Examinee uses Control Room indications to verify RCS status determining size and general location of RCS leakage. Then takes actions that effect plant and system conditions.

References:

AOP-2203.016 (019), Excess RCS Leakage
AOP-2203.016 (017), Excess RCS Leakage Tech Guide
LCO 3.4.6.2 Reactor Coolant System Operational Leakage
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Rev. 1 Question was revised in response to NRC 10 question review as well as internal review.
Stem has been modified to state leakage is Pressure Boundary and no longer providing a reference
Rev 2 changed "Once a Shutdown is required" to "If a shutdown is required" and capitalized
HOT STANDBY in stem
Rev 3 - changed LOD from 4 to 3.

EXCESS RCS LEAKAGE

PURPOSE

This procedure provides actions for abnormal RCS leakage.

ENTRY CONDITIONS

ANY of the following conditions exist AND SDC NOT in operation:

Entry conditions are met

1. Unexplained drop in VCT level.
2. Unexplained Charging and Letdown flow mismatch.
3. "CNTRL CH 1/2 LEVEL LO" annunciator (2K10-G6/G7) in alarm.
4. "VCT 2T4 LEVEL HI/LO" annunciator (2K12-H5) in alarm.
5. Unexplained rise in CNTMT pressure, temperature, humidity, radiation, or sump level.
6. Unexplained rise in Quench tank level, pressure, or temperature.
7. Rising CCW activity or Surge tank level.
8. 2305.002, RCS Leak Detection calculation indicates excessive RCS leakage.

EXIT CONDITIONS

ANY of the following conditions exist:

1. RCS leakage greater than 44 gpm, Reactor tripped and 2202.001, Standard Post Trip Actions entered.
2. RCS leakage greater than 44 gpm, all CEAs inserted, and procedure directs user to 2202.010 Exhibit 8, Diagnostic Actions.
3. ANY SFSC acceptance criteria NOT satisfied.
4. ALL appropriate actions of the procedure have been performed.

5 Plant cooldown commenced and 2102.010, Plant Cooldown or 2203.013, Natural Circulation entered.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

Steps marked with (*) are continuous action steps.

Steps marked with (■) are floating steps.

1. **Open Placekeeping page.**

2. **Notify Control Board Operators to monitor floating steps.**

*3. **Determine RCS leakrate by ANY of the following:**

- Computer LKRT programs.
- Check PZR level stable and use Charging and Letdown mismatch minus Controlled Bleed Off.
- Check Letdown isolated and estimate RCS leak rate by total Charging flow minus Controlled Bleed Off.
- CNTMT Sump level rate of rise.

Leakrate is ~ 24 gpm
88 - 60 = 28 gpm
28 - 4 (CBO) = ~24 gpm

INSTRUCTIONS | **CONTINGENCY ACTIONS**

■ **9. IF RCS leakage greater than 44 gpm**
AND in Mode 3, 4, OR 5,
THEN perform the following:

A. IF ANY CEAS withdrawn,
THEN trip Reactor.

B. Actuate SIAS.

C. Actuate CCAS.

D. GO TO 2202.010 Exhibit 8,
Diagnostic Actions.

__10. Refer to 1903.010,__
Emergency Action Level Classification.

__11. IF location of leak known,__
THEN perform the applicable following steps (12.A through 12.J):

• CNTMT	step 12.A
• Primary to Secondary Leakage	step 12.B
• Quench Tank	step 12.C
• RDT	step 12.D
• RCP Seals	step 12.E
• CCW System	step 12.F
• CVCS	step 12.G
• SIT inleakage	step 12.H
• Vacuum Degasifier	step 12.I
• RCS Sample	step 12.J

Control room indications - Rising trend in Component Cooling Water.

INSTRUCTIONS	CONTINGENCY ACTIONS
12. (continued)	
D. Check Reactor Drain Tank parameters stable: • Level • Pressure	D. Perform the following: • IF RV HEAD LEAK OFF TEMP HI alarm (2K10-A5) in alarm, THEN perform 2203.012J, Annunciator 2K10 Corrective Action. • IF INNER GASKET LEAK alarm (2K11-D1/D3/D5/D7) in alarm, THEN perform 2203.012K, Annunciator 2K11 Corrective Action.
E. Check RCP Seals for proper staging.	E. Perform 2203.025, RCP Emergencies in conjunction with this procedure.
F. Check CCW System intact: 1) CCW Expansion tank levels stable. 2) CCW Radiation monitors stable.	F. IF indications of RCS leakage into CCW system exist, THEN commence Attachment A, RCS to CCW Leak Isolation.

(Step 12 continued on next page)

INSTRUCTIONS

***13. Check leakage within allowable limits, refer to TS 3.4.6.2, Reactor Coolant System Leakage.**

Use either if leakage < 10 gpm

Control Board indications show that leakrate is > 10 gpm therefore Attachment R is required

CONTINGENCY ACTIONS

***13. Perform the following:**

A. Continue efforts to locate and isolate leak.

B. IF plant shutdown required, THEN perform the following:

1) Refer to applicable reactivity plan.

2) IF leakage less than 10 gpm, THEN perform EITHER of the following using 2102.004, Power Operations:

- RCS boration using 2104.003, Chemical Addition, Attachment R, RCS Boration from the RWT or BAMT.
- RCS boration using 2104.003, Chemical Addition, Exhibit 3, Normal RCS Boration at Power.

3) IF leakage greater than or equal to 10 gpm, THEN perform RCS boration using 2104.003, Chemical Addition, Attachment R, RCS Boration from the RWT or BAMT using 2102.004, Power Operations.

C. WHEN Reactor shutdown AND 2202.001, Standard Post Trip Actions completed, THEN **GO TO** Step 16.

REACTOR COOLANT SYSTEM OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary to secondary leakage through any one steam generator (SG),

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. Leakage as specified in Table 3.4.6-1 for those Reactor Coolant System Pressure Isolation Valves identified in Table 3.4.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

This is Pressure Boundary leakage therefore ACTION a. applies

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE or any primary to secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and primary to secondary leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two valves* in each high pressure line having a non-functional valve and be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* These valves may include check valves for which the leakage rate has been verified, manual valves or automatic valves. Manual and automatic valves shall be tagged as closed to preclude inadvertent valve opening.

EXCESS RCS LEAKAGE

2203.016

AOP STEP:

*13. **Check leakage within allowable limits,
refer to TS 3.4.6.2, Reactor Coolant System Leakage.**

BASIS:

TS 3.4.6.2 permits continued plant operation as long as leakage rates are within the TS limits.

A. If leakage is in not within TS limits, the operator is directed to continue leak location/isolation efforts.

B. If plant shutdown is required, then one of the following will be used:

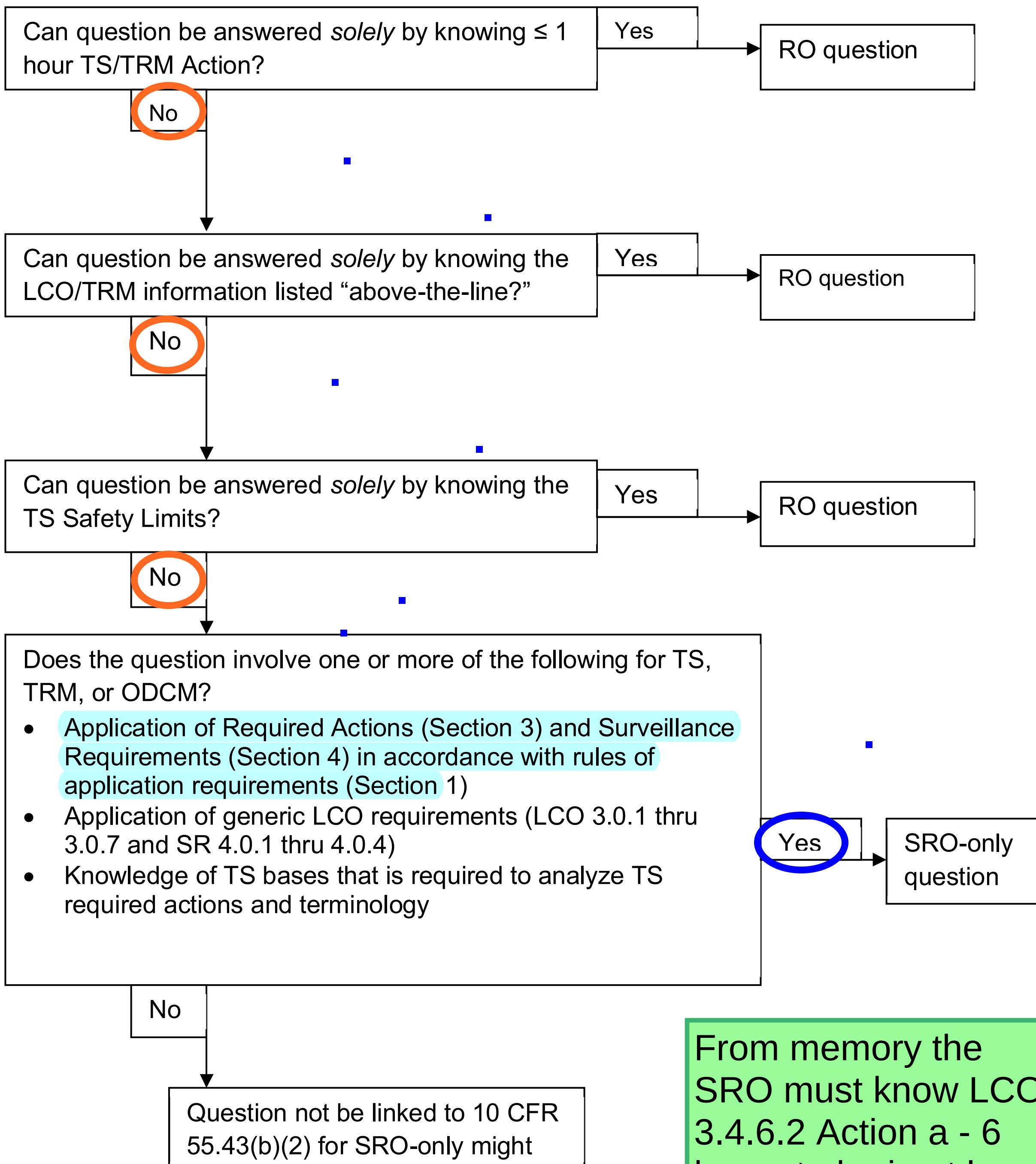
1) If leakage is less than 10 gpm, then either 2104.003 Attachment R (Boration from RWT or BAMT) or 2104.003 Exhibit 3 (Normal RCS Boration at Power) may be used for plant shutdown.

2) If leakage is greater than or equal to 10 gpm, then 2104.003 Attachment R (Boration from RWT or BAMT) is used for a more rapid plant shutdown.

SOURCE DOCUMENTS:

1 - ANO-2 Technical Specification 3.4.6.2.
2 - 2102.004, Power Operations.
3 - 2102.010, Plant Shutdown.
4 - LO-ALO-C-2002-0144-3. Provide specific guidance for power reductions with and without RCS leakage for boration sources in order of priority.

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



From memory the SRO must know LCO 3.4.6.2 Action a - 6 hours to be in at least Hot Standby

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

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

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

Choosing the correct boration source is a contingency step, 13.B.3.
Making this an SRO function

No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Question 85

Bank:	2415	**Rev:**	1	**Rev Date:**	7/6/2016	**2017 TEST QID #:**	85	**Author:**	Larry Burton
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	0000742240	**10CFR55:**	43.2			**Safety Function**	4		
Title:	Inadequate Core Cooling					**System Number**	074	**K/A**	2.2.40
Tier:	1	**Group:**	2	**RO Imp:**	3.4	**SRO Imp:**	4.7	**L. Plan:**	A2LP-RO-TS
OBJ	4								

Description: Equipment Control - Ability to apply Technical Specifications for a system.

Question:

(REFERENCE PROVIDED)

Given the following:

* Unit 2 is at 100% power on March 15th.
* At 0800, LPSI Pump 'B', 2P-60B, is OOS for scheduled maintenance.
* HPSI Pump 'C', 2P-89C, is aligned to the Green train.
* At 1200, NLO reports he has inadvertently broken a sight glass on HPSI Pump 'A', 2P-89A.
* NLO also reports there is a large amount of oil spreading on the floor.

Assuming no Operator actions have been taken and using the applicable Tech Spec maximum allowable action times, Unit 2 must be in Hot Standby no later than ______________?

A. 1800 on March 15th

B. 1900 on March 15th

C. 1800 on March 18th

D. 1400 on March 22nd

Answer:

C. 1800 on March 18th

Notes:

C. Is correct. There is an equivalent of a single HPSI and LPSI train available so condition b is satisfied.
12:00 plus 72 plus 6 is 1800 on the 18th.

A. Is incorrect but plausible because this is the time related to 3.0.3 spec (6 hours).

B. Is incorrect but plausible this is action c, 1 hour to recover or be in Hot Standby within the next 6 hours.

D. Is incorrect but plausible because this is action a., examine may consider that the 'C' HPSI satisfies the TS requirements and therefore remaining in action a is appropriate (7 days + 6 hours).

KA Match - This is the application of TS for a reduction of ECCS equipment which directly relates to Inadequate Core Cooling.

LCO is provided for this question because of the time frame involved 72 hours to fix and 6 hours to Hot Standby. Also the inclusion of LCO 3.0.3 as a distractor, not provided. In addition the real answer to this question is explained in the Bases, not provided. This combination of facts and requirements is beyond a memory requirement.
Concerning the other two LCO related questions one asks for an entry condition or limit (primary chemistry) and the other

asks do you know that for pressure boundary leakage the unit must be promptly placed in cold shutdown. These would be direct lookups in the LCO were provided.

References:

LCO 3.5.2 and Bases
LCO 3.0.3
(All references verified current 11/10/16)
LCO 3.5.2 provided as a REFERENCE.

Historical Comments:

To be used on the 2017 NRC Exam
Rev 1- Added "Assuming no Operator actions have been taken" to the stem

RGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS – $T_{avg} \geq 300°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each sub-system comprised of:

a. One OPERABLE high-pressure safety injection (HPSI) train,

b. One OPERABLE low-pressure safety injection (LPSI) train, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 1, 2 and 3 with pressurizer pressure ≥ 1700 psia.

ACTION:

a. With one ECCS subsystem inoperable due to an inoperable LPSI train, restore the inoperable train to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

b. With one or more ECCS subsystems inoperable due to conditions other than “a” above and 100% of ECCS flow equivalent to a single OPERABLE HPSI and LPSI train is available, restore the inoperable train(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

c. With less than 100% ECCS flow equivalent to either the HPSI or LPSI trains within both ECCS subsystems, restore at least one HPSI train and one LPSI train to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

d. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the NRC within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

Actions a and c are both plausible as they are contained within the LCO.

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Limiting Conditions for Operation (LCO) and ACTION requirements shall be applicable during the OPERATIONAL MODES or oth[...] except as provided in LCO 3.0.2 and 3.0.8[...]

3.0.3 entry is not required because Actions of 3.5.2 cover the ECCS train conditions

3.0.2 Adherence to the requirements of the Limi[...] ACTION within the specified time interval [...] specification, except as provided in Specif[...] Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

3.0.3 In the event a Limiting Condition for Operation and/or associated ACTION requirements cannot be satisfied because of circumstances in excess of those addressed in the specification within 1 hour, action shall be initiated to place the unit in a mode in which the specification does not apply by placing it, as applicable, in at least HOT STANDBY within 6 hours, in at least HOT SHUTDOWN within the next 6 hours, and in at least COLD SHUTDOWN within the following 24 hours unless corrective measures are completed that permit operation under the permissible ACTION statements for the specified time interval as measured from initial discovery or until the reactor is placed in a MODE in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specification.

3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONs to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONs or that are part of a shutdown of the unit.

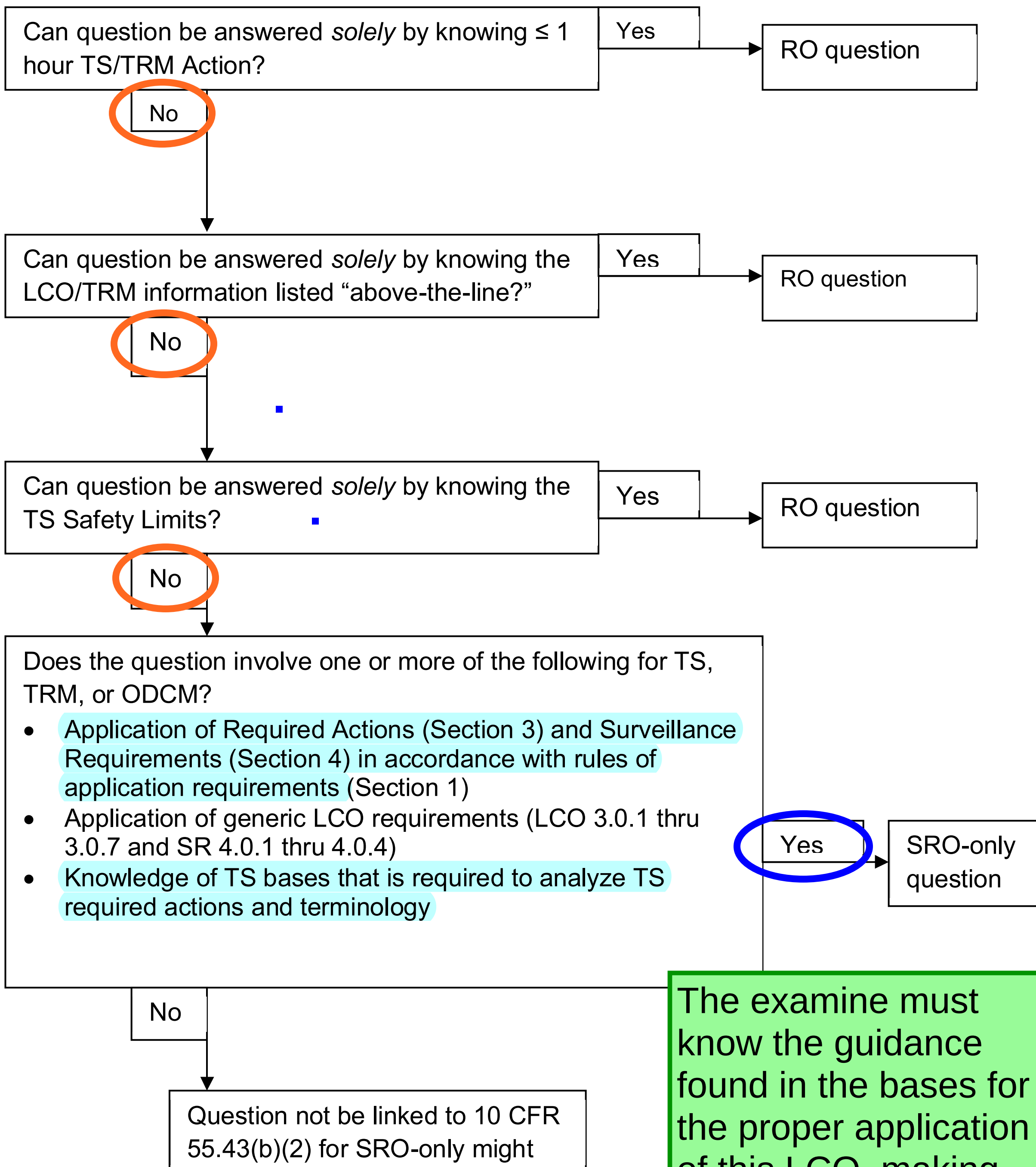
3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS (continued)

In accordance with the NRC Safety Evaluation, *"generally the LPSI AOT will not be entered unless these actions are satisfied. However, it should be recognized that unforeseen circumstances may arise that prohibit complying with these actions."* In addition, it is standard operational practice to verify redundant train OPERABILITY, along with the required support systems, prior to removing any TS component from service, regardless of the length of time a TS component will be removed from service. Therefore, if the redundant LPSI train is not OPERABLE, the maintenance activity will not be performed.

In Action "b", if one or more HPSI or LPSI trains are inoperable except for reasons other than Action "a" and at least 100% of the ECCS flow equivalent to at least one of the individual HPSI and LPSI trains is available, the individual ECCS trains may be inoperable for up to 72 hours. The 72-hour AOT is based on a reasonable amount of time to effect repairs. A HPSI or LPSI train is inoperable if it is not capable of delivering its design flow to the RCS. The individual components within a HPSI or LPSI train are inoperable if they are not capable of performing their design function, or if supporting systems are not available. Due to the redundancy of trains within the ECCS subsystems, the inoperability of one component in a train does not necessarily render the ECCS incapable of performing its function. Similarly, the inoperability of two different components, each in a different HPSI or LPSI train does not make the ECCS subsystem inoperable as long as at least one HPSI and LPSI train is capable of performing its required safety function to deliver at least 100% of its ECCS flow equivalent. This allows increased flexibility in plant operations when components in opposite trains are inoperable.

QID-85
4 of 4

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



The examine must know the guidance found in the bases for the proper application of this LCO, making this an SRO question

ECCS SUBSYSTEMS – $T_{avg} \geq 300°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each sub-system comprised of:

a. One OPERABLE high-pressure safety injection (HPSI) train,

b. One OPERABLE low-pressure safety injection (LPSI) train, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 1, 2 and 3 with pressurizer pressure ≥ 1700 psia.

ACTION:

a. With one ECCS subsystem inoperable due to an inoperable LPSI train, restore the inoperable train to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

b. With one or more ECCS subsystems inoperable due to conditions other than "a" above and 100% of ECCS flow equivalent to a single OPERABLE HPSI and LPSI train is available, restore the inoperable train(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

c. With less than 100% ECCS flow equivalent to either the HPSI or LPSI trains within both ECCS subsystems, restore at least one HPSI train and one LPSI train to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

d. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the NRC within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

Question 86

Bank:	2416	**Rev:**	3	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	86	**Author:**	Burton
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	0030002406	**10CFR55:**	43.5	**Safety Function**	4				
Title:	Reactor Coolant Pump System (RCPS)	**System Number**	003	**K/A**	2.4.6				
Tier:	2	**Group:**	1	**RO Imp:**	3.7	**SRO Imp:**	4.7	**L. Plan:**	A2LP-RO-EFRP	**OBJ**	4

Description: Emergency Procedures/Plan - Knowledge of EOP mitigation strategies.

Question:

Given the following:

* Unit 2 is operating at 100% power.
* 'B' EFW Pump (2P-7B) is OOS for emergent maintenance.
* A seismic event occurs.
* Condenser vacuum rapidly degraded and indications are off scale high.
* A manual reactor trip is initiated.
* Bus 2A1 faults and cannot be re-energized.
* 'A' and 'B' Steam Generator pressures are being controlled by ADVs.
* Pressurizer pressure is 1380 psia and stable.
* Pressurizer level is 10% and slowly rising.
* RCS T-hot is 547°F.
* Containment radiation levels are rising.
* 'A' EFW Pump (2P-7A) trips on overspeed.
* SPTAs are complete.
* All 4 RCPs are running.

The crew should transition to ______________ and ___________________ .

A. OP-2202.009, Functional Recovery; secure ONLY 2 RCPs

B. OP-2202.009, Functional Recovery; secure all running RCPs

C. OP-2202.003, Loss of Coolant Accident; secure ONLY 2 RCPs

D. OP-2202.003, Loss of Coolant Accident; secure all running RCPs

Answer:

B. OP-2202.009, Functional Recovery; secure all running RCPs

Notes:

B is correct. LOCA is the obvious event in progress due to the pressurizer and RCS conditions given in the stem. The LOCA is greater than capacity of the charging pumps but less than HPSI flow since pressurizer level and pressure are recovering. The containment rad levels increasing confirm this is a LOCA and not an ESD or SGTR. There is also a Loss of Feedwater occurring due to the loss of vacuum and the 2A3 bus making this a dual event and requires FRP entry. Stopping all RCPs is correct due to the loss of Feedwater but leaving two RCPs in operation is plausible for the RCS pressure and MTS requirements being met. The LOCA in this question is in the range of 200-300 gpm range based on given conditions. Greater than charging pump capacity but < than HPSI flow since Pressure level and pressure are recovering.

A. is incorrect. FRP is correct as stated above in "B". Stopping only 2 RCPs is plausible because there is no loss of subcooling based on RCS parameters.

C. is incorrect. LOCA is plausible because of the RCS and containment conditions, the Loss of Feed must be deduced due to the loss of vacuum. Stopping only 2 RCPs is plausible because it is the correct action for the current RCS temp and pressure if the loss of feed is not considered.

D. is incorrect. LOCA is plausible as stated in "C" and stopping all RCPs is correct. If the examine chooses LOCA stopping all 4 RCPs could be correct if there is a loss of subcooling based on RCS temp and pressure.

Question meets the KA as it requires knowledge EOP mitigation strategies involving the RCPs

References:

EOP-2202.003 (15), LOCA
EOP-2202.009 (18), FRP
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Rev 2 - strengthened LOCA explanation in the notes, no change to question.
Rev 3 - added "Bus 2A1 faults and cannot be re-energized" to de-energize AFW pump 2P75
Changed stem to 'B' EFW OOS for maintenance.
De-energizing 2A3 would have also been an overlap issue with Q-51

LOSS OF COOLANT ACCIDENT

PURPOSE

This procedure provides operator actions which must be accomplished in the event of a Loss of Coolant Accident (LOCA). These actions are necessary to ensure that the plant is placed in a stable, safe condition. The goals are to mitigate the effects of a LOCA, isolate the break (if possible), and establish long term core cooling using the Safety Injection System or the Shutdown Cooling System. This procedure achieves these goals while maintaining continuous adequate core cooling and minimizing radiological releases to the environment.

ENTRY CONDITIONS

ANY of the following may be present:

1. PZR level lowering (for a break in the PZR, level may be rising).
2. Lowering RCS pressure.
3. SIAS automatically actuated.
4. Rising CNTMT pressure, temperature, radiation, humidity, and sump level.
5. Rising Quench Tank level, temperature, or pressure.
6. RCS MTS lowering.

All these conditions make LOCA plausible

EXIT CONDITIONS

EITHER of the following conditions exist:

1. ANY LOCA SFSC acceptance criteria NOT satisfied.
2. LOCA EOP has accomplished at le
   - SDC system entry conditions sa
   - RCS long term core cooling esta
   - RCS leakage isolated and plant

Additional Information -
The LOCA in this question is in the range of 200-300 gpm range based on given conditions. Greater than charging pump capacity but < than HPSI flow since Pressure level and pressure are recovering.

INSTRUCTIONS	CONTINGENCY ACTIONS
*7. **Verify the following for any operating RCP:**	
A. CSAS NOT actuated.	A. Secure ALL RCPs.
B. Proper seal staging.	B. IF three or more stages failed, THEN secure the affected pump.
■8. **Check CCW flow aligned to RCPs.**	■8. Perform the following: A. IF CCW system available, THEN restore CCW to RCPs using 2202.010 Attachment 21, Restoration of CCW to RCPs. B. IF CCW system NOT available, THEN secure CCW system using 2202.010 Attachment 6, Securing CCW and ACW.
■9. **Check RCS pressure greater than 1400 psia.** Guidance with RCS pressure < 1400 psia. assists plausibility of RCP status	■9. Perform the following: A. IF RCS pressure less than 1400 psia, THEN perform the following: 1) Verify maximum of ONE RCP running in EACH loop. 2) IF RCP 2P32A or 2P32B stopped, THEN verify associated PZR Spray valve in MANUAL and closed. B. IF NPSH requirements violated OR RCS MTS less than 30°F, THEN perform the following: 1) Stop ALL RCPs. 2) Verify BOTH PZR Spray valves in MANUAL and closed.

FUNCTIONAL RECOVERY

PURPOSE

This procedure provides actions for events where diagnosis is NOT possible, or where emergency guidance is NOT available.

ENTRY CONDITIONS

ANY of the following conditions exist:

1. ANY event in progress which can NOT be diagnosed as a single event.
2. Actions taken have NOT satisfied SFSC acceptance criteria.
3. Entry is directed by Diagnostic Actions

EXIT CONDITIONS

1. Acceptance criteria for ALL success paths in use are satisfied.
2. Event diagnosis completed and a recovery procedure determined using Exhibit 8, Diagnostic Actions.

Entry conditions for FRP are met. Loss of MFPs is not directly given but can be diagnosed due to the loss of Condenser vacuum

INSTRUCTIONS	CONTINGENCY ACTIONS
■ **34. Restore ESF/Non-ESF systems post-SIAS using 2202.010 Attachment 51, Post ESFAS Actuation System Restoration.**	
■ **35. Check at least ONE intact SG available for Heat Removal by EITHER of the following:** • Level 10% to 90% [20% to 90%] AND FW available. • Level being restored AND total FW flow of 485 gpm or greater.	■**35. <u>IF</u> NEITHER SG available, <u>THEN</u> perform the following:** A. Stop ALL RCPs. B. Verify BOTH PZR Spray valves in MANUAL and closed. C. Close BOTH SG Blowdown Isolation valves: • 2CV-1016-1 • 2CV-1066-1 D. **GO TO** Step 37.
With no MFW/AFW/EFW the SGs are NOT available and RCPs must be stopped	
36. GO TO Step 42.	

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

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing immediate operator actions?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?
Yes → RO question
No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question
No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

SRO determines procedure transition. Also uses plant conditions to determine mitigating strategy in regards to RCP operation.

Question 87

Bank:	2417	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	87	**Author:**	Burton	
Lic Level:	SRO	**Difficulty:**	4	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam			
Search	0100002411	**10CFR55:**	43.5	**Safety Function**	3					
Title:	Pressurizer Pressure Control System (PZR PCS)	**System Number**	010	**K/A**	2.4.11					
Tier:	2	**Group:**	1	**RO Imp:**	4.0	**SRO Imp:**	4.2	**L. Plan:**	A2LP-RO-PZR	**OBJ** 2.4

Description: Emergency Procedures/Plan - Knowledge of abnormal condition procedures.

Question:

Consider the following

* Unit 2 is in Mode 3 preparing for a reactor startup when the following occurs.
* ATC reports RCS pressure is 2250 psia and rising.
* Crew is taking actions IAW OP-2203.028, Pzr Systems Malfunction.
* Pzr Spray valves 2CV-4651 and 2CV-4652 indicate closed.

Pressurizer Pressure LCO 3.2.8. will have to be entered if RCS pressure exceeds a MINIMUM of ______ psia and if PZR spray valves will NOT open manually, then OP-2203.028 directs using ___________ to mitigate these conditions.

A. 2275; OP-2103.005, Pressurizer Operations

B. 2275; OP-2202.010, Attachment 48, RCS Pressure Control

C. 2340; OP-2103.005, Pressurizer Operations

D. 2340; OP-2202.010, Attachment 48, RCS Pressure Control

Answer:

B. 2275; OP-2202.010, Attachment 48, RCS Pressure Control

Notes:

2275 psia is the TS limit for pressurizer pressure, 2340 psia is the high Pzr pressure alarm setpoint.
Step 4 of the AOP directs the use of standard attachment 48 if spray valve will not open.
Step 5 of the AOP directs the use of Pressurizer Operations if backup Charging Pumps operation is an issue.

B. Is correct the upper LCO limit is 2275 psia and the AOP directs using attachment 48.

A. Is incorrect. The LCO limit is correct, using Pressurizer Operation is wrong but plausible because it's use is directed within the AOP if Charging pumps are the issue.

C. Is incorrect. The pressure is wrong but plausible because this is the high pressure alarm setpoint, using Pressurizer Operation is wrong but plausible because it's use is directed within the AOP if Charging pumps are the issue.

D. Is incorrect. The pressure is wrong but plausible because this is the high pressure alarm setpoint, however the procedure use is correct.

KA Match - Knowledge of the transition procedures within the AOP meets the KA requirement, Knowledge of AOP procedures.

References:

LCO 3.2.8 Pressurizer Pressure
AOP-2203.28 (013), PZR Systems Malfunction
ACA-2203.012J (043), Annunciator 2K10, E-6 (Control Channel Pressure Hi-Lo)
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Rev 1 - capitalized MINIMUM in stem

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

Steps marked with (*) are continuous action steps.

1. Check the following criteria:

Spray valves will not open per stem

A. IF any PZR spray valve failed open, THEN **GO TO** Step 2.

B. IF any PZR spray valve failed closed, THEN **GO TO** Step 4.2203.028, Pzr Systems

C. Check “RRS TROUBLE” annunciator (2K10-H2) clear.
   - C. **GO TO** Step 5.

D. Check "CNTRL CH 1/2 PRESSURE HI/LO" annunciators (2K10-E6/E7) clear.
   - D. **GO TO** Step 6.

E. Check the following PZR level annunciators clear:
   - "CNTRL CH 1/2 LEVEL LO" 2K10-G6/G7
   - "CNTRL CH 1/2 LEVEL HI" 2K10-J6/J7
   - E. **GO TO** Step 7.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

4. **<u>IF</u> ANY PZR Spray valve failed closed, <u>THEN</u> perform the following as needed:**

A. Place affected PZR Spray valves in MANUAL:

- 2CV-4651
- 2CV-4652

Directed to perform Attachment 48 if Pzr Spray valves will not control pressure

B. Throttle PZR Spray valves as necessary to maintain RCS pressure 2025 psia to 2275 psia.

B. <u>IF</u> PZR Spray valves will <u>NOT</u> open, <u>THEN</u> maintain RCS pressure 2025 psia to 2275 psia with Aux Spray using 2202.010, Attachment 48, RCS Pressure Control.

INSTRUCTIONS

5. **Check "RRS TROUBLE" annunciator (2K10-H2) clear.**

Distracters A and C

CONTINGENCY ACTIONS

5. **IF malfunction caused PZR level setpoint to change, THEN perform the following:**

   A. Perform the following for Letdown Flow controller (2HIC-4817):

      1) Place controller in MANUAL.

      2) Adjust output to control PZR level within 5% of setpoint.

   B. Manually control Charging pumps.

   C. Manually operate PZR heaters.

   D. IF Remote Auto PZR Level setpoint incorrect,
   THEN place PZR Level controller in LOCAL AUTO and adjust setpoint based on T_{AVE} refer to 2102.004 Attachment E, Pressurizer Level Program.

   E. WHEN Letdown Flow controller (2HIC-4817) automatic and manual signals matched,
   THEN restore controller to AUTO using 2104.002, Chemical and Volume Control.

   F. IF failure prevents backup Charging pump operation,
   AND backup Charging pump required,
   THEN defeat stop interlock using 2103.005, Pressurizer Operations.

   G. IF Letdown Radiation monitor isolated due to high temperature
   AND Letdown HX Outlet temperature lowered to less than 140°F,
   THEN restore Letdown Radiation Monitor flow by opening Letdown Rad Monitor Isolation, 2CV-4804 (2HS-4804).

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PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The average pressurizer pressure shall be maintained between 2025 psia and 2275 psia.

2275 psia is the LCO high limit

APPLICABILITY: MODE 1.

ACTION:

With the average pressurizer pressure exceeding its limits, restore the pressure to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The average pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

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ANNUNCIATOR 2K10

E-6

CNTRL CH 1 PRESSURE HI/LO

Distractors C and D

1.0 CAUSES

1.1 High - ≥ 2340 psia RCS pressure (2PS-4626A)

1.2 Low - ≤ 2100 psia RCS pressure (2PS-4626A)

2.0 ACTION REQUIRED

2.1 Check Pzr Pressure Ch 1 controller (2PIC-4626A) status.

2.2 Validate alarm using trending capability of PMS/PDS and available RCS pressure indications.

- IF alarm invalid
  OR lowering RCS pressure for plant cooldown/evolutions,
  THEN no further action required.

2.3 IF reducing pressure due to Safety valve leakage,
THEN no further action required.

2.4 Refer to PZR Systems Malfunction (2203.028).

2.5 Refer to Tech Spec 3.2.8.

3.0 TO CLEAR ALARM

3.1 Restore RCS pressure to between 2100 psia and 2340 psia.

4.0 REFERENCES

4.1 E-2456-4

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

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

No

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

No

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

No

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

This question requires assessment of plant conditions and determining which procedure to transition to.

No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Question 88

Bank:	2418	**Rev:**	3	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	88	**Author:**	Larry Burton
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	005000A203	**10CFR55:**	43.5	**Safety Function**	4				
Title:	Residual Heat Removal System (RHRS)	**System Number**	005	**K/A**	A2.03				
Tier:	2	**Group:**	0a	**RO Imp:**	2.9	**SRO Imp:**	3.1	**L. Plan:**	A2LP-RO-LMFR
OBJ	8								

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - RHR pump/motor malfunction

Question:

Consider the following:

* Unit 2 entered a refueling outage 2 days ago.
* LPSI Pump 'B', 2P-60B, is aligned and providing SDC flow.
* RCS Tc is 220°F and is being slowly lowered.
* RCS pressure is 250 psia and stable.

NOW

* LPSI Pump 'B' motor faults and trips.
* SDC SUCTION PRESS HI, annunciator (2K07-D7) is in alarm.
* SDC FLOW HI/LO annunciator (2K07-C8) is in alarm.
* RCS temperature is 225°F and slowly rising.
* RCS pressure is 305 psia and slowly rising.
* SDC RCS Isolation MOV, 2CV-5084-1, is closed.

Based on these conditions the CRS should ___________________ and this procedure will direct the CRS to __________________ in order to open SDC RCS Isolation valve 2CV-5084-1.

A. transition to the Lower Mode Functional, OP-2202.011; reduce RCS pressure with Aux. Spray

B. implement the Loss of Shutdown Cooling, OP-2203.029; reduce RCS pressure with Aux. Spray

C. transition to the Lower Mode Functional, OP-2202.011; ensure the 2CV-5084-1 handswitch is in CLOSE and remove the associated power supply fuse located in front of panel 2C21

D. implement the Loss of Shutdown Cooling, OP-2203.029; ensure the 2CV-5084-1 handswitch is in CLOSE and remove the associated power supply fuse located in front of panel 2C21

Answer:

B. implement the Loss of Shutdown Cooling, OP-2203.029; reduce RCS pressure with Aux. Spray

Notes:

Unit 2 entered the outage 2 days ago so any loss of SDC, would cause a significant rise in both RCS pressure and temperature. Per the Loss of Shutdown Cooling procedure when pressure rises above 300 psia then at least one of the RCS Isolation MOVs must be closed to prevent over pressurization of SDC components. There is nothing in the stem that will lead the examine to believe that the other train, LPSI "A" is not available for SDC operations so the LMFRP is not required.

B. is correct, Loss of SDC is directed by the ACAs and none of the contingency actions within the AOP direct exiting to the Lower Mode FRP. The AOP directs lowering RCS pressure to less than 300 psia

using Aux Spray in order to open the Isolation MOVs and start a SDC pump.

A. is Plausible because LMFRP is directed for a loss of power and other conditions within the AOP if the current actions are not mitigating the event. Reducing RCS pressure is correct.

C. is Plausible because LMFRP is directed for a loss of power and other conditions within the AOP if the current actions are not mitigating the event. Removing the fuse is plausible since these steps are directed by the AOP if the cause of the Isolation MOV closing is due to a transmitter failure.

D. is Plausible because, Loss of SDC is directed by the ACAs and none of the contingency actions within the AOP direct exiting to the Lower Mode FRP. Removing the fuse is plausible since these steps are directed by the AOP if the cause of the Isolation MOV closing is due to a transmitter failure.

Meets the KA because SDC is lost due to an RHR pump failure and the question asks for both the itigating procedure and required actions.

References:

EOP-2202.011 (013), LMFRP
AOP-2203.029 (020), Loss of Shutdown Cooling
ACA-2203.012G (033), Annunciator 2K07 (C-8 and D-7) Corrective Action
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Rev. 1 This question was part of the 10 question review and had several comments, I would like to discuss further. - I have made no changes other than to elaborate comments in the reference pages supporting HR-4 as correct. The "RNO" column directs closing the SDC isolation valves it makes no effort to restore SDC.
Rev 2 - based on NRC comment, I have replaced the original question and created this version.
Rev 3 - Added "should" to stem and removed can and must from distractors.

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ANNUNCIATOR 2K07

C-8

SDC FLOW HI/LO

1.0 CAUSES

1.1 Shutdown Cooling flow outside of variable alarm setpoints (2FIC-5091).

2.0 ACTION REQUIRED

2.1 Verify SDC flow being controlled within desired band.

2.2 Verify RCS level adequate to prevent vortexing for present SDC flow IAW Draining the Reactor Coolant System (2103.011), Exhibit 1.

2.3 IF due to failure in SDC system, THEN GO TO Loss Of Shutdown Cooling (2203.029).

2.4 IF Shutdown Cooling Flow less than minimum required by OPS-B40, THEN refer to the following:

- Tech Spec 3.4.1.3
- Tech Spec 3.9.8.1
- Tech Spec 3.9.8.2

Low flow alarm - GO TO Loss of Shutdown Cooling

2.5 Adjust alarm setpoints as needed IAW guidelines of Unit 2 SDC Control (1015.008).

3.0 TO CLEAR ALARM

3.1 Restore SDC flow to within alarm setpoints (2FIC-5091).

4.0 REFERENCES

4.1 E-2455-4

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ANNUNCIATOR 2K07

D-7

SDC SUCTION PRESS HI

1.0 CAUSES

1.1 Pressure between SDC RC Isolations (2CV-5086-2 and 2CV-5038-l) above variable alarm setpoint (2PIS-5088).

2.0 ACTION REQUIRED

2.1 Verify RCS pressure being controlled in desired band.

2.2 IF alarm caused while energizing LPSI Pump Suction/Discharge Pressure Transmitter (2PI-5039A),
THEN no further action required.

2.3 Reset (2PIS-5088) alarm setpoint to 285 psig or less and as low as needed to provide early indication of rising RCS pressure using 2104.004, Shutdown Cooling System Attachment N SDC Pump Suction Pressure 2PIS-5088.

2.4 IF RCS pressure rises to greater than 350 psia,
THEN perform the following:

2.4.1 Secure in-service Shutdown Cooling pump.

2.4.2 Close the following SDC RC Isol valves:

- 2CV-5084-1 (2HS-5084-1)
- 2CV-5038-1 (2HS-5038-1)
- 2CV-5086-2 (2HS-5086-2)

High Pressure alarm directs entry of Loss of Shutdown Cooling

2.4.3 GO TO Loss of Shutdown Cooling (2203.029).

3.0 TO CLEAR ALARM

3.1 Reduce pressure in LPSI Suction Line below Alarm Setpoint.

4.0 REFERENCES

4.1 E-2455-4

LOSS OF SHUTDOWN COOLING

PURPOSE

This procedure provides operator actions for a loss of SDC due to pump trips, isolable leaks, Temperature/Flow controller malfunctions and pressure switch failures.

ENTRY CONDITIONS

ANY of the following conditions exist during SDC operations with fuel in the core:

1. "LOSS OF SDC SUCTION" annunciator (2K07-A7) in alarm.
2. "SDC FLOW HI/LO" annunciator (2K07-C8) in alarm.
3. "CET TEMP HI" annunciator (2K07-D8) in alarm.
4. "RCS LEVEL HI/LO" annunciator (2K07-A8) in alarm.
5. "SDC SUCTION PRESS HI" annunciator (2K07-D7) in alarm.
6. Indication of erratic SDC flow.
7. Unexplained lowering RCS Level.
8. Unexplained rise in CET, RVLMS ATS, RCS T_H, or SDC temperatures.
9. Low or erratic LPSI Header pressure (2PI-5092).
10. "LPSI 2P60A/B BREAKER TRIP" annunciator (2K06-B2/2K05-B2) or "CNTMT SPRAY 2P35 A/B BREAKER TRIP" annunciator (2K06-B1/2K05-B1) in alarm.
11. Unexplained rises in CNTMT Building or Auxiliary Building Sump levels with lowering RCS Level.
12. Unexplained rises in Quench tank, RDT or RWT levels with lowering RCS level.
13. "SW HEADER LOOP 1/2 PRESS LOW" annunciator (2K06-A6/2K05-A6) in alarm.
14. Loss of Instrument Air.
15. Loss of 125v DC to bus 2D22.
16. Indications of RCS to SW leak.
17. "LPSI PUMP MOTOR AMPS HI/LO" annunciator (2K07-C7) in alarm.
18. "LPSI PUMP SUCT PRESS HI/LO" annunciator (2K07-B7) in alarm.
19. "LPSI DISCH HEADER PRESS HI/LO" annunciator (2K07-B8) in alarm.

INSTRUCTIONS	CONTINGENCY ACTIONS
2. **CHECK** EITHER 4160V Non-Vital AC bus 2A1 or 2A2 energized.	2. **CHECK** EITHER 4160V Vital AC bus energized. A. **PERFORM** the following: 1) **CLOSE** ALL LPSI Injection MOVs previously open for SDC flow. 2) **THROTTLE** open ONE LPSI Injection MOV for approximately 2 seconds. 3) **ENSURE** RCS level adequate, **REFER TO** 2202.010 Attachment 33, RCS Level. * 4) **CONTROL** RCS cooldown rate within TS limits, **REFER TO** 2202.010 Attachment 8 RCS Cooldown Table. 5) **START** SDC pump, **REFER TO** 2104.004, Shutdown Cooling System. 6) **RAISE** flow to greater than 2400 gpm by opening LPSI Injection MOVs. 7) **GO TO** 2202.011, Lower Mode Functional Recovery. B. **IF** SDC cannot be restored, **THEN GO TO** 2202.011, Lower Mode Functional Recovery.

This is one of numerous contingency steps within the procedure directing entry into Lower Mode FRP. However conditions are not met; no Loss of Power

3. **OPEN** Placekeeping Page and **RECORD** the following:

   A. Event start time _______.

   B. RCS temperature ________°F.

4. **NOTIFY** Control Board Operators to monitor floating steps.

INSTRUCTIONS	CONTINGENCY ACTIONS
5. **INITIATE** 2202.010 Attachment 32, CNTMT Evacuation Checklist.	
6. **COMMENCE** plotting heatup rate every 15 minutes using Form 1015.016I, RCS/PZR Temperature VS Time.	
*7. **CHECK** RCS pressure less than 300 psia.	*7. **PERFORM** the following: A. **SECURE** the operating SDC pump. B. **CLOSE** at least ONE SDC RCS Isolation MOV: • 2CV-5038-1 • 2CV-5084-1 • 2CV-5086-2 C. **GO TO** Step 9.

With RCS pressure > 300 psia it is required to close at least 1 SDC-RCS Isolation MOV

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

- If the Vessel Head is removed, CETs and ATS will NOT be operable.
- If SDC flow NOT indicated, Shutdown Cooling recorder, (2TR-5097) and computer points (T5096/T5095) indications are invalid.
- T_H RTDs temperature indication may be influenced by SIS injection water temperatures.

*11. **MONITOR** RCS temperature with ANY of the following:

- Core Exit Thermocouples (CETs)
- ATS from ANY of the following:
  - From SPDS Computer Points (CV2EXITA/CV2EXITB)
  - From RVLMS using 2105.003, Reactor Vessel Level Monitoring System Operations
- T_H RTDs
- LPSI Pump Discharge temperature computer point (T5096)
- LPSI Pump Header temperature computer point (T5095)
- Shutdown Cooling Recorder, (2TR-5097)

12. **CHECK** SDC pump running.

12. **PERFORM** the following:

A. **IF** refueling in progress, **THEN STOP** ALL fuel movement.

B. **IF** RCS dilution in progress, **THEN STOP** dilution.

C. **GO TO** Step 18.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

16. **CHECK** SDC available as follows:

    - SDC flow (2FIC-5091) 2400 gpm or greater.
    - SDC flow through SDC HX.
    - SW aligned to SDC HX.

    16. **PERFORM** the following:

    A. **IF** refueling in progress,
    THEN STOP ALL fuel movement.

    B. **IF** RCS dilution in progress,
    THEN STOP dilution.

    C. **GO TO** 2202.011,
    Lower Mode Functional Recovery.

17. **EXIT** this procedure.

18. **NOTIFY** SM to refer to the following:

    - 1903.010,
      Emergency Action Level Classification
    - ANO-2 Technical Specifications and Technical Requirements:
      - TS 3.1.1.3, Boron Dilution
      - TRM 3.1.7,
        Borated Water Sources-Shutdown
      - TS 3.4.1.3,
        Reactor Coolant System-Shutdown
      - TS 3.4.9.1, Pressure/Temperature Limits
      - TS 3.8.1.2, Electrical Power System-Shutdown
      - TS 3.8.2.2,
        AC Distribution-Shutdown
      - TS 3.9.8.1,
        Shutdown Cooling-ONE Loop
      - TS 3.9.8.2,
        Shutdown Cooling-Two Loops

INSTRUCTIONS

19. **CHECK** ALL SDC RCS Isolation MOVs open:

- 2CV-5038-1
- 2CV-5084-1
- 2CV-5086-2

Direction to lower pressure with Aux spray

Distractors C and D

CONTINGENCY ACTIONS

19. **PERFORM** the following:

A. **IF** RCS pressure greater than 300 psia, **THEN REDUCE** RCS pressure with Aux Spray using 2202.010 Attachment 48, RCS Pressure Control.

B. **IF** EITHER RCS pressure instrument (2PI-4623-1/ 2PI-4623-2) failed high, **THEN PERFORM** the following:

1) **PLACE** associated SDC RCS Isolation valve handswitch to CLOSE:

- 2HS-5084-1 for failure of 2PI-4623-1
- 2HS-5086-2 for failure of 2PI-4623-2

2) Locally **REMOVE** associated power supply fuse located in front panel of 2C21:

- 2PI-4623-1, "2PWR-4623-1"
- 2PI-4623-2, "2PWR-4623-2"

(Step 19 continued on next page)

INSTRUCTIONS | **CONTINGENCY ACTIONS**

19. (continued)

OPEN the SDC RCS isolation MOVs after corrective actions are taken

C. **ENSURE** ALL SDC RCS Isolation MOVs open per ANY of the following as required:

- Control Room handswitches (preferred):
  - 2CV-5084-1 (2HS-5084-1)
  - 2CV-5038-1 (2HS-5038-1)
  - 2CV-5086-2 (2HS-5086-2)
- Local handswitches at the breaker:
  - 2CV-5084-1 (2HS-5084-1A at 2B51-G2)
  - 2CV-5038-1 (2HS-5038-1A at 2B52-E5)
  - 2CV-5086-1 (2HS-5086-2A at 2B62-E5)
- Local manual operation:
  - "SHUTDOWN COOLING RC ISOLATION 2CV-5084-1" (located in CNTMT)
  - "SHUTDOWN COOLING RC ISOLATION 2CV-5038-1" (located in USPPR)
  - "SHUTDOWN COOLING RC ISOL 2CV-5086-2" (located in CNTMT)

INSTRUCTIONS	CONTINGENCY ACTIONS

CAUTION

Starting a SDC pump with RCS level less than 24 inches on RCS Refueling Level indicator (2LI-4791/2LI-4792) or 371 ft 1.5 inches on "RCS LOCAL LEVEL INDICATOR" may cause SDC pump cavitation and gas binding of SDC train.

INSTRUCTIONS	CONTINGENCY ACTIONS
23. **CHECK** SDC pump running. Steps to restore SDC	23. **PERFORM** the following: A. **CLOSE** ALL LPSI Injection MOVs previously open for SDC flow. B. **THROTTLE** open ONE LPSI Injection MOV for approximately 2 seconds. C. **ENSURE** RCS level adequate, **REFER TO** 2202.010 Attachment 33, RCS Level. * D. **CONTROL** RCS cooldown rate within TS limits, **REFER TO** 2202.010 Attachment 8 RCS Cooldown Table. E. **START** SDC pump, **REFER TO** 2104.004, Shutdown Cooling System. F. **RAISE** flow to greater than 2400 gpm by opening LPSI Injection MOVs. G. **<u>IF</u>** unable to start SDC pump, **<u>THEN</u> GO TO** 2202.011, Lower Mode Functional Recovery.
24. **RETURN TO** Step 13.	

END

INSTRUCTIONS	CONTINGENCY ACTIONS
19. Check SDC available.	**19. <u>IF</u> SDC <u>NOT</u> available, <u>THEN</u> GO TO Step 27.**
20. Check SDC RCS Isolation valves open: • 2CV-5038-1 • 2CV-5084-1 • 2CV-5086-2 LMFRP contains the same steps to remove fuse. Distractors C and D	**20. Perform the following:** A. <u>IF</u> RCS pressure greater than 300 psia, <u>THEN</u> reduce RCS pressure with Aux Spray using Attachment 48, RCS Pressure Control. B. <u>IF</u> either RCS pressure instrument (2PI-4623-1/ 2PI-4623-2) failed high, <u>THEN</u> perform the following: 1) Place associated SDC RCS Isolation valve handswitch to CLOSE. 2) Locally remove associated power supply fuse located in front panel of 2C21. • 2PI-4623-1, "2PWR-4623-1" • 2PI-4623-2, "2PWR-4623-2" C. Verify ALL SDC RCS Isolation valves open: • 2CV-5038-1 • 2CV-5084-1 • 2CV-5086-2

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

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Assessing plant conditions then choosing the correct mitigating contingency action or procedure section to perform is an SRO function.

Question 89

Bank:	2419	**Rev:**	1	**Rev Date:**	10/17/2016	**2017 TEST QID #:**	89	**Author:**	Simpson
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	Modified NRC Exam Bank QID 2161		
Search	026000A207	**10CFR55:**	43.5			**Safety Function**	5		
Title:	Containment Spray System (CSS)					**System Number**	026	**K/A**	A2.07
Tier:	2	**Group:**	1	**RO Imp:**	3.6	**SRO Imp:**	3.9	**L. Plan:**	A2LP-RO-ELOCA
OBJ	19								

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the CSS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Loss of containment spray pump suction when in recirculation mode, possibly caused by clogged sump screen, pump inlet high temperature exceeded cavitation, voiding), or sump level below cutoff (interlock) limit containment spray

Question:

Given the following:

* OP-2202.003, Loss of Coolant Accident has been entered.
* HPSI Pump 2P-89C is OOS for scheduled maintenance.
* HPSI Pump 2P-89B tripped and will not restart.
* HPSI Pumps 2P-89A is running.
* Cntmt Spray Pumps 2P-35A and 2P-35B are running.
* RAS has actuated, all automatic and manual actions are complete.

Standard Attachment 43, ECCS/CSS Pump monitoring states that Containment Sump blockage would be indicated by _________________ and ________________ handswitch(es) would be placed in PTL if Early Termination conditions are met.

A. changing containment sump level; either Cntmt Spray Pump 'A' or 'B' ONLY

B. changing containment sump level; either Cntmt Spray Pump 'A' or 'B' and HPSI Pump 'A'

C. unstable flow and discharge pressure on running ESF pumps; either Cntmt Spray Pump 'A' or 'B' ONLY

D. unstable flow and discharge pressure on running ESF pumps; either Cntmt Spray Pump 'A' or 'B' and HPSI Pump 'A'

Answer:

C. unstable flow and discharge pressure on running ESF pumps; either Cntmt Spray Pump 'A' or 'B' only

Notes:

(1) Attachment 43 states the indications of a blocked Cntmt sump which include unstable or lower than expected flow and discharge pressure. LOCA procedure has the operators monitor CNTMT sump levels making this a valid distractor.

(2) Cntmt sump levels, which SI pumps are running and Cntmt parameters determine which pumps will be stopped. With only one HPSI pump in service, then stopping only one Containment Spray pump is correct.

C. Is Correct - sump blockage indication and stopping only CS pumps is correct.

A. Is Incorrect - changing containment sump levels is plausible because monitoring for sump levels is directed in the LOCA procedure. Stopping only one of the CS pumps is correct.

B. Is Incorrect - changing containment sump levels is plausible because monitoring for sump levels is directed in the LOCA procedure. Stopping either CS pump and the HPSI is plausible because step 23 addresses stopping a HPSI pump.

D. Is Incorrect - indications of Containment Sump blockage is correct. Stopping either CS pump and the HPSI is plausible because step 23 addresses stopping a HPSI pump.

KA Match - This question requires knowledge of the indications of a blocked Cntmt sumps which would cause a loss of CS pump suction as well as the correct application of procedures to correct this condition

References:

EOP-2202.010 (023), Standard Attachment 43
EOP-2202.003 (015), Loss of Coolant Accident
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam.
The parent question is QID-2161 from the 2014 Retake Exam.
Modifications include asking the indications of "Sump blockage" instead of which procedure to enter and changed which pumps to be secured to a single CS pump only.

I made the following changes. Based on the NRC "10 question review".
Rev 1 - Question has been changed from "new" to modified.
(1) deleted several bullets from the stem.
(2) Added the word only to part 2 of 'A' and 'C'".
(3) Changed water level to sump level in 'A' and 'B'.
(4) Fixed the KA statement.

INSTRUCTIONS	CONTINGENCY ACTIONS

CAUTION

Circulation of high activity reactor coolant through Aux Building will raise dose rates to personnel.

■**21.** **WHEN "RWT LEVEL LO LO RAS PRETRIP" annunciator (2K06-A9) in alarm,**
THEN perform the following:

A. Notify Control Board Operators to monitor for CNTMT Sump blockage using 2202.010 Attachment 43, ECCS/CSS Pump Monitoring.

Direction to monitor for containment sump blockage using attachment 43

B. Locally verify ESF pump room doors closed and dogged.

C. Locally verify the following valves closed:

- "A ESF PUMP ROOM DRAIN LINE ISOLATION" 2ABS-5
- "B ESF PUMP ROOM DRAIN LINE ISOLATION" 2ABS-6

ATTACHMENT 43
ECCS/CSS PUMP MONITORING

Monitor all available indications for signs of a loss of ECCS pump suction, i.e. CNTMT Sump blockage, as indicated by any of the following (listed in order of likely occurrence):

a) Unstable or lower than expected HPSI or CS flow

b) Unstable or lower than expected HPSI or CS pump discharge pressure

c) Lower than expected HPSI or CS pump suction pressure, low suction pressure alarm

d) Unstable or lower than expected HPSI or CS pump motor current

e) Raised HPSI or CS pump noise

If there are indications of a reduction in NPSH or pump performance, operator should review parameter trends and attempt to diagnose what is happening. For example: an individual pump in distress, a valve or system component failure or sump screen blockage. Accurate diagnosis of these occurrences under accident conditions is difficult and will require the operator to rely heavily on knowledge, experience and training. None of the available indications will provide a 100% conclusive diagnosis.

Attachment 43 -- indications of CNTMT Sump Blockage

INSTRUCTIONS	CONTINGENCY ACTIONS
***17. Terminate LPSI flow as follows:**	
A. Check the following criteria satisfied: • RCS pressure greater than 200 psia. • RCS pressure controlled.	A. **GO TO** Step 18.
B. Place BOTH LPSI pumps in PTL.	
C. Override and close LPSI Injection MOVs.	
***18. Monitor LPSI termination criteria satisfied for duration of event.**	***18. IF RAS NOT actuated AND LPSI termination criteria NOT satisfied, THEN restore LPSI flow as necessary to maintain termination criteria satisfied.**
19. IF possible, THEN initiate action to refill the RWT by ANY of the following: • **Normal makeup per 2104.003, Chemical Addition** • **Makeup from Holdup tanks per 2104.006, Fuel Pool Systems** • **Makeup from SFP per 2104.006, Fuel Pool Systems** • **Instruct TSC to consider RWT Refill strategy per SAMG.**	Containment conditions that are monitored in a LOCA event that can also cause a loss of suction to ECCS pumps **CAUTION** **RAS Initiation during LOCA outside CNTMT may result in loss of suction to HPSI and CNTMT Spray pumps.**
***20. Monitor for RWT level lowering with corresponding rise in CNTMT Sump level.**	***20. IF LOCA occurring outside CNTMT, THEN perform the following:** A. Perform 2202.010 Attachment 22, LOCA Outside CNTMT Isolation. B. Verify CIAS actuated on PPS inserts.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

Optimal Early Termination lineup configuration would include one CNTMT Spray pump and one HPSI pump with the same power supply, with a common sump suction line.

*23. **Verify Early HPSI Termination as follows:**

A. Check indication(s) of CNTMT Sump Blockage using 2202.010 Attachment 43, ECCS/CSS Pump Monitoring.

A. **GO TO** Step 24.

B. IF all available HPSI trains in service, THEN initiate Early HPSI Termination as follows:

1) Verify ONE HPSI train secured by placing HPSI pump (2P89A/B/C) in PTL.

Only HPSI pump 'A' is running per stem, therefore do NOT stop HPSI pump

*24. **Verify Early CNTMT Spray Termination as follows:**

*25. **GO TO Step 25.**

IF All of the following are TRUE:

- CNTMT Spray operating
- Indication(s) of CNTMT Sump Blockage using 2202.010 Attachment 43, ECCS/CSS Pump Monitoring

THEN place EITHER CNTMT Spray pump (2P35A/B) in PTL.

Both CS pumps are running per stem, therefore stop 1 CS

*25. **Instruct TSC to consider Alternate RCS Injection strategy.**

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

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Determining the recovery actions as found in steps 23/24 of the LOCA EOP is an SRO function. These actions are not initial actions or an overall mitigative strategy of an EOP

For All QID In Exam Bank

17-Nov-16

Bank:	2161	**Rev:**	1	**Rev Date:**	7/10/2014 10:41:2	**QID #:**	89	**Author:**	Simpson
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW		
Search	026000A207	**10CFR55:**	41.5 / 43.5 / 45.3 / 45.1	**Safety Function**	5				
System Title:	Containment Spray System (CSS)	**System Number**	026	**K/A**	A2.07				
Tier:	2	**Group:**	1	**RO Imp:**	3.6	**SRO Imp:**	3.9	**L. Plan:**	A2LP-RO-SPRAY
OBJ	8								

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the CSS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Loss of containment spray

Question:

Consider the following:

- Unit 2 experienced a large break LOCA 4 hours ago
- ESF pump suctions are aligned to the Containment sump
- HPSI pumps 2P-89A and B are running
- Spray Pumps 2P-35A and B are running
- RAS now actuates

30 minutes later the ATC reports:

- "B" Spray header flow, discharge pressure, and amps are fluctuating and containment
- sump blockage is indicated per Standard Attachment 43
- Containment pressure stable at 24 psia

The CRS should direct securing ____________ using ______________________ .

A. "B" Spray pump (2P-35B) ONLY;
OP-2202.003, Loss of Coolant Accident section 3 "Unisolated LOCA"

B. "B" Spray pump (2P-35B) AND "B" HPSI pump (2P-89B);
OP-2202.003, Loss of Coolant Accident section 3 "Unisolated LOCA"

C. "B" Spray pump (2P-35B) ONLY;
OP-2202.010, Standard Attachments, Att. 45 "CSAS Reset"

D. "B" Spray pump (2P-35B) AND "B" HPSI pump (2P-89B);
OP-2202.010, Standard Attachments, Att. 45 "CSAS Reset"

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐

Audit Exam History

2006	☐
2008	☒

Answer:

B. Correct

Notes:

B. Correct: The LOCA EOP has the operators monitor for Containment Sump Blockage (Standard Attachment 43) when the RAS pretrip alarm is received (LOCA EOP, section 3, step 21). "B" Spray pump is cavitating by the combination of header flow, pressure, and amps fluctuating. Attachment 43 will direct the CBOT/ATC to inform the CRS (step 7) if there is any indication of Sump Blockage. The CRS will use the LOCA EOP (section 3 step 23 and 24) to direct both "B" train ESF pumps to be secured to prevent pump damage.

A. Incorrect: only securing "B" spray pump is plausible due to it is currently the only effected pump but the LOCA EOP directs securing BOTH ECCS pumps supplied by that side of the containment sump.

C. Incorrect: LOCA EOP directs securing BOTH ECCS pumps supplied by the effected side of the

containment sump not just one. Attachment 45 is plausible in that the actions for securing spray pumps is listed, however, containment pressure has to be below 22.5 psia to allow reset.

D. Incorrect: The LOCA EOP will direct securing BOTH the "B" Spray pump and HPSI pump when containment sump blockage is indicated not Att. 45. Attachment 45 is plausible in that the actions for securing spray pumps is listed, however, containment pressure has to be below 22.5 psia to allow reset.

References:

OP-2202.003, Loss of Coolant Accident, Rev 014, section 3, step 21 page 38 of 74, step 23 and 24 page 43 of 74
EOP-2202.003, Loss of Coolant Accident Tech Guide, Rev 014, section 3, step 23 pages 115 and 116 of 140, step 24 page 117 of 140
OP-2202.010, Standard Attachments, Rev 022, Att. 43 page 131 of 204
Lesson Plan A2LP-RO-SPRAY objective 8: Given a Containment Spray Actuation, evaluate the Containment Spray system response to determine if the system has responded as designed and what actions, if any are required to make any necessary corrections

Historical Comments:

Rev 1: added "B" Spray pump discharge pressure, and amps are fluctuating and containment sump blockage is indicated per Standard Attachment 43 to the stem and expanded the explanations for the distracters.

This is the parent question that has been modified to create Q #89 on the 2017 SRO exam.
Modified version asks for the indications of "Sump Blockage" and only a single CS pump is secured not both pumps.

Question 90

Bank:	2420	**Rev:**	3	**Rev Date:**	12/8/2016	**2017 TEST QID #:**	90	**Author:**	Burton
Lic Level:	SRO	**Difficulty:**	4	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	022000A204	**10CFR55:**	43.5	**Safety Function**	5				
Title:	Containment Cooling System (CCS)	**System Number**	022	**K/A**	A2.04				
Tier:	2	**Group:**	1	**RO Imp:**	2.9	**SRO Imp:**	3.2	**L. Plan:**	A2LP-RO-ELOCA
OBJ	6								

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the CCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Loss of service water

Question:

Given the following:

* Unit 2 is operating at 100% power.
* Service Water Loop 2 has been isolated due to a piping rupture.
* Pressurizer level and pressure are rapidly lowering.
* An automatic reactor trip occurs.
* RCS pressure is 1100 psia and lowering.
* Containment pressure is 24 psia and rising.
* SG pressures are 1000 psia and stable.
* OP-2202.003, LOCA has been entered.
* 'B' train Containment Spray flow is 1200 gpm.

NOW

* Bus 2A3 faults and cannot be re-energized.

Containment Coolers 2VSF-1C and 2VSF-1D are running with _______ and the CRS _________.

A. no cooling flow; must transition to OP-2202.009, FRP and implement CTPC-3, CNTMT Spray

B. normal chill water flow; must transition to OP-2202.009, FRP and implement CTPC-3, CNTMT Spray

C. no cooling flow; may remain in OP-2202.003, LOCA and implement Exhibit 13, Miscellaneous Containment Building Ventilation

D. normal chill water flow; may remain in OP-2202.003, LOCA and implement Exhibit 13, Miscellaneous Containment Building Ventilation

Answer:

A. no cooling flow; must transition to OP-2202.009, FRP and implement CTPC-3, CNTMT Spray

Notes:

Cntmt coolers 1C/1D have power from the 'B' class buses, 2A4 is energized from offsite power.No service water is available to the 1C/1D coolers due to Loop 2 rupture 1A/1B have no power due to the loss of the 2A3 bus. Cntmt coolers start and normal chill water isolates on CCAS signal at 18.3 psia in Cntmt. The crew must transition to the FRP due to NO adequate CS flow.
Miscellaneous CNTMT Building ventilation is plausible since is listed in the LOCA procedure.

A. No cooling flow and transition to the FRP are both correct.

B. Is Incorrect - there is no chill water flow because it isolated at 18.3 psia in containment but plausible because there was flow up to that point, also the stem does not state that a CCAS has actuated but the examine should be able to assess conditions and realize it did. FRP is the correct procedure.

C. Is Incorrect - No cooling water is correct because chill water isolated at 18.3 psia in containment and the SW Loop 2 rupture. LOCA is Plausible due to the indication of an RCS break and the guidance within LOCA to implement Exhibit 13, but without adequate CS flow the FRP is required.

D. is Incorrect - there is no chill water flow because it isolated at 18.3 psia in containment but plausible because there was flow up to that point, also the stem does not state that a CCAS has actuated but the examine should be able to assess conditions and realize it did. LOCA is Plausible due to the indication of an RCS break and the guidance within LOCA to implement Exhibit 13, but without adequate CS flow the FRP is required.

KA matches since part one gives the effect on the Containment Cooling System due to the loss of Service Water and the second part addresses the procedures used to mitigate the consequences of these conditions.

References:

2202.003 (015), LOCA
2202.009 (018), FRP
STM 2-09 (016), Containment Cooling and Purge System
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Question was revised in response to NRC 10 question review.
Rev 1 - In response to the 10 question review the question has been modified to make FRP the correct answer. QID-11 has been modified so there is no cueing.
Rev 2 - Added LOCA entry to stem then changed answers to "may remain in LOCA" or "must transition to FRP". Also changed order shortest to longest.
Rev 3 - removed the word Vital from stem.

LOSS OF COOLANT ACCIDENT

PURPOSE

This procedure provides operator actions which must be accomplished in the event of a Loss of Coolant Accident (LOCA). These actions are necessary to ensure that the plant is placed in a stable, safe condition. The goals are to mitigate the effects of a LOCA, isolate the break (if possible), and establish long term core cooling using the Safety Injection System or the Shutdown Cooling System. This procedure achieves these goals while maintaining continuous adequate core cooling and minimizing radiological releases to the environment.

ENTRY CONDITIONS

ANY of the following may be present:

1. PZR level lowering (for a break in the PZR, level may be rising).
2. Lowering RCS pressure.
3. SIAS automatically actuated.
4. Rising CNTMT pressure, temperature, radiation, humidity, and sump level.
5. Rising Quench Tank level, temperature, or pressure.
6. RCS MTS lowering.

Conditions given in stem making LOCA plausible.

EXIT CONDITIONS

EITHER of the following conditions exist:

1. ANY LOCA SFSC acceptance criteria NOT satisfied.
2. LOCA EOP has accomplished at least ONE of the following:
   - SDC system entry conditions satisfied.
   - RCS long term core cooling established.
   - RCS leakage isolated and plant stabilized.

SAFETY FUNCTION STATUS CHECK

SAFETY FUNCTION | **ACCEPTANCE CRITERIA**

NOTE

Meeting the provisions of Condition 1 or Condition 2 will satisfy the Safety Function.

8. CNTMT Temperature and Pressure Control

Condition 1 not met; Cntmt pressure is 24 psia and rising in the stem

"B" Containment Spray pump is 1200 gpm and "A" CS pump has no power.

Due to CTPC Safety Function not being met - FRP entry is required

CONDITION 1

A. CNTMT temperature less than 235°F.

B. CNTMT pressure less than 23.3 psia.

C. Hydrogen less than minimum detectable concentration (1).

CONDITION 2

A. At least ONE of the following:

1) Two available CNTMT Cooling fans operating in Emergency Mode.

2) ONE CNTMT Spray header with flow 1875 gpm or greater.

3) Required combination of CNTMT Spray Systems and CNTMT Cooling Fans running in EMERGENCY MODE.

B. CNTMT pressure less than 73.7 psia.

C. Hydrogen less than lower flammability concentration (1).

(1) Hydrogen concentration acceptance criteria may be omitted until Hydrogen Analyzers are in service and providing data.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

20. **Verify ALL available miscellaneous CNTMT Building ventilation operating using 2202.010 Exhibit 13, Miscellaneous Containment Building Ventilation.**

Plausibility for the LOCA answer

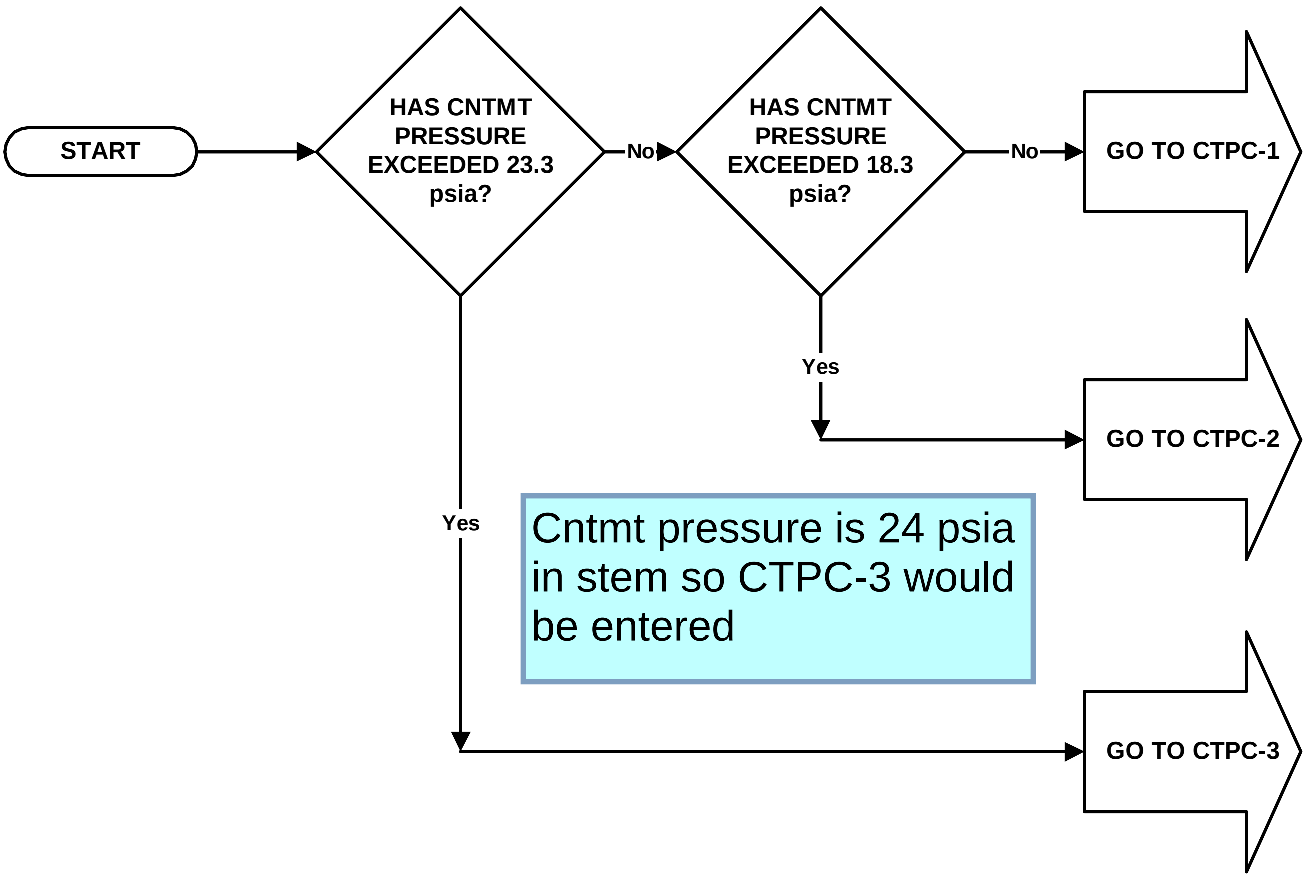
■21. **Check ALL AC and Vital DC buses energized.**

■21. **Perform the following:**

A. IF ANY AC bus de-energized, THEN commence power restoration using 2202.010 Attachment 11, Degraded Power.

B. IF ANY Vital DC bus de-energized, THEN commence power restoration using 2203.037, Loss of 125v DC.

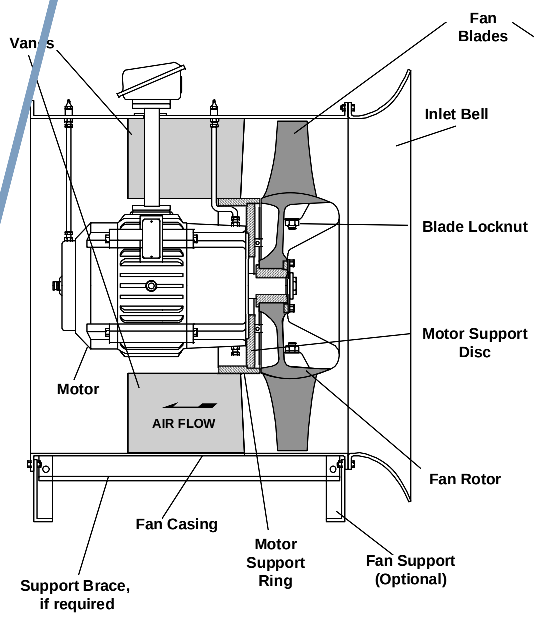
CTPC DECISION TREE

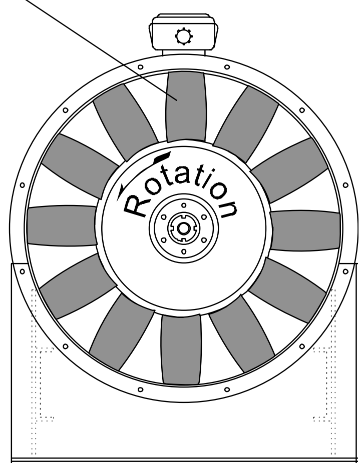


2.8 Cooling Unit Fans, 2VSF-1A through D

Each of the containment cooling units is equipped with a vaneaxial fan driven by a 75 HP motor. The fan, of course, provides the motive force for the ventilation airflow. Each fan is rated at 27,000 cubic feet per minute (CFM).

An axial fan is a fan that produces airflow parallel to the axis of the fan rotation. An axial fan discharge air pattern rotates in a spiral which is less efficient flowing through long lengths of ductwork because the turbulent flow causes too much of a heat gain to the air just from flowing through the duct.





Vane-Axial Fan

A vaneaxial fan is an axial fan mounted in a cylinder that has air-guide vanes attached to it. The air-guide vanes straighten out the rotating spiral air flow of the fan which improves the efficiency of the unit when discharging into long lengths of duct work because the laminar (smooth) flow does not pick up as much heat from flowing through the duct.

18.3 psia in Containment, fan will have started

The fans are operated from the control room with a two position (START-STOP) spring return to normal handswitch. When the handswitch is taken to START the fan starts and when it is taken to STOP the fan stops as long as no CCAS signal is present.

The fans also automatically start if their handswitch is NOT in PULL TO LOCK under the following conditions:

- IF the bus is powered from off-site or its respective Diesel Generator AND,
- CCAS is actuated.

When the above conditions are satisfied, a time delay relay provides contact closure to start the fan in 18.2 seconds. The time delay allows a sequencing of loads to prevent overloading the bus that could occur if all of the loads started at the same time.

In other words, the vital 4160 volt bus that is powering up the MCC is either being powered by its associated emergency diesel generator or its normal power supply. It cannot be cross tied to the other trains vital 4160 volt bus or to the alternate AC generator. If it is, the fan does not automatically start.

These fans **CANNOT** be overridden in an accident condition. The only way to secure the fans following a CCAS is to take their handswitch to PULL TO LOCK. If the handswitch were taken to STOP during a CCAS, the fan would stop but, as soon as the

supply is outside the containment, it is required by technical specifications to have a back up overcurrent protection device.

Bypass damper power supply information is listed below.

Power supply - fan has power to be running

Cooling Unit	Bypass Damper	Power Supply/Backup Overcurrent Protection	Control Room Handswitch Location
2VSF-1A	2UCD-8203-1	2B53 brkr G1 2B53 brkr A5	2C-17
2VSF-1B	2UCD-8209-1	2B53 brkr G2 2B53 brkr A6	2C-17
2VSF-1C	2UCD-8216-2	2B64 brkr D4 2B64 brkr B3	2C-16
2VSF-1D	2UCD-8222-2	2B64 brkr E4 2B64 brkr C2	2C-16

2.7 Service Water Cooling Coils, 2VCC-2A through D*

The next "piece" of the cooling unit to discuss is the Service Water cooling coils, 2VCC-2A through D. Each cooling unit is equipped with eight service water cooling coils. Each cooling coil has 192 finned tubes for heat transfer.

During accident conditions service water is lined up to each containment cooling group at greater than or equal to 1250 gpm.

Although the actual technical specification limit on service water flow to the group is 1250 gpm, there is a different operability limit specified in the surveillance requirement due to fouling of the service water cooling coils. This value changes as a result of service water flow testing and should be obtained from the procedure.

Each containment cooling unit group (remember a group has two units) has its own service water supply and return header and since they penetrate the containment, each header has an isolation valve. These valves automatically open upon receipt of a containment cooling actuation signal (CCAS) or a main steam isolation signal (MSIS).

The supply header to the group that contains cooling units 2VSF-1A and 1B has a single isolation valve outside of the containment, 2CV-1511-1, located in the upper south piping penetration room (USPPR). This header penetrates the containment at penetration 2P-20. The return header also has a single isolation valve outside of containment, 2CV-1519-1. This valve is also located in the USPPR. This header penetrates the containment at penetration 2P-21.

The supply header to the group that contains cooling units 2VSF-1C and 1D has a single isolation valve outside of the containment, 2CV-1510-2. This header penetrates the containment at 2P-55. The return header also has a single isolation valve outside

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

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

No

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

No

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

No

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

No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

→ Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

The need to transition to the FRP to address the loss of a Safety Function makes this an SRO question.

Question 91

Data for 2017 NRC RO/SRO Exam

19-Jan-17

Bank:	2421	**Rev:**	0	**Rev Date:**	10/14/2016	**2017 TEST QID #:**	91	**Author:**	Larry Burton
Lic Level:	SRO	**Difficulty:**	4	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	045000A212	**10CFR55:**	43.2	**Safety Function**	4				
Title:	Main Turbine Generator (MT/G) System	**System Number**	045	**K/A**	A2.12				
Tier:	2	**Group:**	2	**RO Imp:**	2.5	**SRO Imp:**	2.8	**L. Plan:**	A2LP-RO-TS	**OBJ**	4

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the MT/G System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Control rod insertion limits exceeded (stabilize secondary)

Question:

Given the following:

* Unit 2 is operating at 100% power.
* Main Turbine net output is 975 megawatts.
* CEAs are being used to dampen ASI oscillations per OP-2102.004, Power Operation.
* Group 6 CEAs are 130 inches withdrawn.
* Group P CEAs are 136 inches withdrawn.
* Both CEACs are Operable.

NOW

* Annunciators 2K04-J5 and J6 "CEAC 1/2 CEA DEVIATION" come into alarm.
* Group 6, CEA-47, CEA Pulse Counter is 130 inches and CEAC positions are 121 inches withdrawn.
* Group P, CEA-23, CEA Pulse Counter and CEAC positions all indicate 0 inches.
* Crew has entered OP-2203.003, CEA Malfunction.

Prior to any Operator actions Main Turbine load ___________ and the CRS will enter 2203.003, CEA Malfunctions and direct a _____________________ for the given CEA positions.

A. remains constant; Plant shutdown using OP-2102.004, Power Operations

B. remains constant; Reactor trip, and performance of OP-2202.001, SPTAs

C. initially lowers; Plant shutdown using OP- 2102.004, Power Operations

D. initially lowers; Reactor trip, and performance of OP-2202.001, SPTAs

Answer:

C. initially lowers; Plant shutdown using OP-2102.004, Power Operations

Notes:

As stated in the CEA Malfunctions Tech Guide, RCS temperature will lower to add positive reactivity this lower temperature will have the effect of lowering Main Turbine load. Remaining constant is plausible because ANO maintains the turbine on the "Potentiometer" which keeps the control valves in a set position.
Plant Shutdown is correct because the 2nd CEA deviation is > 7 inches but less than 19 inches.
Reactor trip is plausible because there are multiple deviating CEAs and one of them is > 19 inches.

C. Is Correct. Main Turbine load will lower due to the lowering RCS temperatures counteracting the dropped CEA. Plant Shutdown is correct because multiple CEAs have "slipped" and only one of them is fully inserted.

A. Is Incorrect - Main Turbine load remaining constant is wrong but plausible because Unit 2 operates with the Turbine controlled by the potentiometer which has the effect of keeping the Control Valves is a set position, this can be interrupted as constant load. Plant Shutdown is correct because multiple CEAs have "slipped" and only one of them is fully inserted.

B. Is Incorrect - Main Turbine load remaining constant is wrong but plausible because Unit 2 operates with the Turbine controlled by the potentiometer which has the effect of keeping the Control Valves is a set position, this can be interrupted as constant load. Tripping the plant and performing SPTAs is wrong because only one CEA is deviating > 19 inches.

D. Is Incorrect - Main Turbine load will lower due to the lowering RCS temperatures counteracting the dropped CEA. Tripping the plant and performing SPTAs is wrong because only one CEA is deviating > 19 inches.

KA Matches because:
(1) addresses the effect on MT/G of a CEA inserting beyond limits.
(2) addresses the procedure which will be used to mitigate the condition, CEA Malfunctions

References:

AOP-2203.003 (023), CEA Malfunctions
AOP-2203.003 (023), CEA Malfunctions Tech Guide;
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam

CEA MALFUNCTION

2203.003

Lower RCS temperature means lower saturation pressure and Main Turbine load will lower

AOP STEP:

■18. **ADJUST** Turbine load to match T_{AVE} within 2°F of T_{REF}.

BASIS:

A dropped or slipped CEA will add negative reactivity, resulting in a lowering of RCS temperature during power operation. The RCS temperature lowering adds positive reactivity, counteracting the effect of the dropped or slipped CEA. Reducing Turbine load will raise T_{AVE} and lower T_{REF}, returning T_{AVE} back to its programmed value and balancing both primary and secondary plants.

SOURCE DOCUMENTS:

1. 2102.004, Power Operations.

AOP STEP:

*19. **CHECK** RCS T_C 542 to 554.7°F using CPC PID 5, 6, 160, or 161.

BASIS:

A dropped or slipped CEA will add negative reactivity, resulting in a lowering of RCS temperature during power operation. This step checks T_C within range specified in TS 3.2.6. The contingency action refers the operator to TS 3.2.6, providing corrective action when T_C is out of range.

SOURCE DOCUMENTS:

1. Technical Specification 3.2.6, Reactor Coolant Cold Leg Temperature.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

17. (continued)

D. **NOTIFY** I&C to commence CEA troubleshooting. (Refer to TS 3.1.3.1.c, CEA Position.)

NOTE

A CEA is considered untrippable with CEDMCS operable AND CEA immovable.

E. **IF** ANY CEAs determined to be untrippable, **THEN** **GO TO** Step 38.

F. **GO TO** Step 37.

■18. **ADJUST** Turbine load to match T_{AVE} within 2°F of T_{REF}.

*19. **CHECK** RCS T_C 542 to 554.7°F using CPC PID 5, 6, 160, or 161.

*19. **REFER TO** TS 3.2.6, Reactor Coolant Cold Leg Temperature.

*20. **CHECK** RCS pressure 2025 to 2275 psia.

*20. **PERFORM** the following:

A. **ENSURE** PZR Pressure Control system restoring RCS pressure to setpoint.

B. **REFER TO** TS 3.2.8, Pressurizer Pressure.

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

- CEA misalignment is defined as a CEA misaligned from its associated Group by outward deviation more than 5 inches or inward deviation more than 7 inches.
- For the purpose of defining CEA operability to satisfy TS LCO requirements, a CEA is inoperable under the following conditions:
  1. CEA is known to be untrippable or immovable as a result of excessive friction or mechanical interference (TS 3.1.3.1.a).
  2. CEA is known to be immovable as a result of CEDMCS malfunction (TS 3.1.3.1.b and 3.1.3.1.c).
  3. CEA is misaligned from ANY other CEA in its group by greater than 7 inches and can NOT be aligned (TS 3.1.3.1.d).
  4. CEA can NOT be exercised within the maximum TS surveillance time requirements of TS 4.1.3.1.2.
  5. Shutdown Bank CEA withdrawn to less than its full out position except for surveillance testing (TS 3.1.3.5).

4. **REFER TO** the following:
   - Tech Specs
   - Attachment E, Tech Spec Two Hour Actions

*5. **<u>IF</u>** ANY CEAs immovable
<u>AND</u> aligned,
<u>THEN</u> GO TO Step 37.

*6. **<u>IF</u>** TWO **<u>OR</u>** MORE CEAs misaligned by greater than 19 inches,
<u>THEN</u> PERFORM the following:

   A. **TRIP** Reactor.

   B. **GO TO** 2202.001, Standard Post Trip Actions.

Action for two CEAs deviating greater than 19 inches. Distractors B and D

INSTRUCTIONS | **CONTINGENCY ACTIONS**

*7. **IF** TWO **OR** MORE CEAs misaligned by greater than 7 inches,
THEN PERFORM the following:

Per the stem one CEA is fully inserted and the other 9 inches making these conditions true.

A. **COMMENCE** Plant shutdown at greater than 14 %/hr using 2102.004, Power Operation.

B. **REFER TO** TS 3.1.3.1.e., CEA Position.

NOTE

T_{AVE} computer point numbers that may be used include T-AVG and T4617-B.

8. **RECORD** the following:

- Start time __________
- Pre-misalignment Rx power __________
- T_{AVE} change __________

9. **CHECK** Reactor startup in progress.

9. **IF** Reactor startup **NOT** in progress,
THEN GO TO Step 17.

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5 of 5

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

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question r
10 CFR 55

Determining the recovery actions as found in the AOP is an SRO level task. This AOP has multiple corrective actions for a number of malfunctions therefore the stated actions are neither initial or an overall mitigating strategy.

Question 92

Bank:	2422	**Rev:**	1	**Rev Date:**	10/12/2016	**2017 TEST QID #:**	92	**Author:**	Coble		
Lic Level:	SRO	**Difficulty:**	4	**Taxonomy:**	H	**Source:**	BANK ANO-2 2011 SRO				
Search	0350002418	**10CFR55:**	43.5	**Safety Function**	4						
Title:	Steam Generator System (S/GS)	**System Number**	035	**K/A**	2.4.18						
Tier:	2	**Group:**	2	**RO Imp:**	3.3	**SRO Imp:**	4.0	**L. Plan:**	A2LP-RO-ESGTR	**OBJ**	15
Description:	Emergency Procedures/Plan - Knowledge of the specific bases for EOPs.										

Question:

Given the following:

* Unit 2 has tripped from full power due to a Steam Generator Tube Rupture.
* 'A' Steam Generator has been diagnosed as the ruptured SG.
* RCPs 2P-32A and 2P-32D are running.
* SG 'A' has been isolated.
* Cooldown and depressurization of the 'A' SG has commenced.
* All other system and components function as designed.

At this time, OP-2202.004, Steam Generator Tube Rupture, requires a level band of _______ % in the ruptured SG and the basis for this level is to ensure SG tubes are _______________.

A. 10 to 38; partially uncovered to cool the steam space of the 'A" SG

B. 10 to 38; covered to prevent release of gaseous activity from the RCS

C. 22.2 to 45; partially uncovered to cool the steam space of the 'A" SG

D. 22.2 to 45; covered to prevent release of gaseous activity from the RCS

Answer:

A. 10 to 38; partially uncovered to cool the steam space of the 'A" SG

Notes:

Step 36 of the SGTR EOP has a step to maintain SG level 45 to 90% to limit any radioactive release. The bases for the 45% is to keep the SG tubes covered.
Step 19 checks level >22.2% (EFAS setpoint) making a band of 22.2 to 45%, plausible
Step 50 of the SGTR EOP, specifies 10 to 38% level band on the faulted SG level band during cooldown and depressurization with RCPs running.
Level is lowered to 10 to 38% to allow uncovering of the SG tubes thus transferring latent heat of the hot steam to the cooler RCS.

A. is Correct - both the level band and reason are correct.

B. Is Incorrect - level band is correct and the reason is plausible because early in a SGTR event level is maintained above the top of the tubes to prevent a release.

C. Is Incorrect - level band is wrong but plausible because the bounding values are both numbers found within SGTR procedure. (22.2% is the EFAS setpoint and 45% is found throughout the procedure). Havng the tubes uncovered is the correct reason.

D. Is Incorrect - level band is wrong but plausible because the bounding values are both numbers found within SGTR procedure. (22.2% is the EFAS setpoint and 45% is found throughout the procedure). The

reason is plausible because early in a SGTR event level is maintained above the top of the tubes to prevent a release.

KA matches because the question asks the bases for EOP guidance as it relates to the SGs. SGTR and the associated bases for the level bands meets this requirement.

References:

EOP-2202.004 (014), SGTR EOP Steps 36 and 50.
TG-2202.004 (014), SGTR Tech Guide
(All references verified current 11/10/16)

Historical Comments:

NRC Exam Bank # 1808 was used on the ANO-2 2011, SRO exam.
Minor alterations include wording of the stem.
Changed the order of the answers making A correct instead of B.
Changed the incorrect level band from 20-45% to 22,2-45%
To be used on the 2017 NRC Exam
Rev 1 - Verified no overlap with Operating exam. Level band was made from levels found within the SGTR EOP

INSTRUCTIONS	CONTINGENCY ACTIONS
18. IF SG Blowdown aligned to flume, THEN perform the following: A. Verify SG Blowdown Isolation valves closed: • 2CV-1016-1 • 2CV-1066-1 B. Secure SG Blowdown lineup to flume and realign SG Blowdown to SU BD DI using 2106.008, Steam Generator Operations.	
■ **19. Check SG levels greater than 22.2%.** Plausibility number	■ **19. IF SG levels less than 22.2%, THEN perform the following:** A. Verify EFAS actuated on PPS inserts. B. IF MSIS prevents feeding intact SG, THEN override and establish EFW flow 2202.010 Attachment 46, Establishing EFW Flow. C. IF total EFW flow less than 485 gpm, THEN commence EFAS verification using 2202.010 Attachment 7, EFAS Verification.

INSTRUCTIONS	CONTINGENCY ACTIONS
■ **36. Maintain ruptured SG level 45% to 90% as follows:**	
A. Maintain ruptured SG level less than 90% by performing the following as needed:	
1) Minimize primary to secondary break flow:	
a) Maintain RCP NPSH and RCS MTS, refer to 2202.010 Attachment 1, P-T Limits.	
b) Depressurize RCS to within 50 psi of ruptured SG using 2202.010 Attachment 48, RCS Pressure Control.	
2) Check Condenser available.	2) <u>IF</u> Condenser <u>NOT</u> available, <u>THEN</u> perform the following: a) Steam ruptured SG using SDBCS Upstream ADV or Upstream ADV Isolation MOV. b) **GO TO** Step 36.B.
3) Throttle SG Blowdown rate on ruptured SG to maintain level using 2202.010 Attachment 49, Steam Generator Blowdown Restoration.	
4) Open MSIV Bypass valve to steam ruptured SG.	

Plausibility number

(Step 36 continued on next page)

STEAM GENERATOR TUBE RUPTURE

2202.004

EOP STEP:

*36. **Maintain ruptured SG level 45% to 90% as follows:**

EPG STEP:

*27.

SUPPLEMENTARY INFORMATION:

12.

DEVIATION? Yes

BASIS FOR DEVIATION:

45% is to maintain the tubes covered limiting any release. plausibility of C and D

EOP substeps were created to separately check for ruptured SG level less than the maximum level and greater than the minimum level (4) because these two conditions require totally different contingency actions. The EOP uses the plant specific post accident SG level range (4). This range implements the direction of the EPG Bases which also requires that the Steam Generator tubes remain covered (6).

A previous EOP Step provides guidance on RCS Pressure control. IF RCPs are running, then RCS pressure is maintained in a narrow band above minimum RCP NPSH (1).
If RCPs are not running, then pressure is maintained in a narrow band, within 50 psia above ruptured SG pressure.

If tube inleakage is excessive, steaming of the ruptured SG via the MSIV Bypass Valve may be necessary to maintain level less than the maximum (4). Another method provided is to reestablish blowdown using an attachment (8). Contingency actions address overfill by lowering RCS pressure in the same manner as a previous EOP step which deals with minimizing break flow. RCS pressure is reduced to less than the MSSV setpoint to prevent possible water discharge through a main steam safety valve (7). Also, specific ANO-2 concerns regarding filling the Main Steam lines with water (3) are addressed.

If the ruptured SG level cannot be maintained greater than the minimum (4), then the contingency action ensures the SG is isolated using a standard attachment (5). This attachment will isolate all steam vent and drain paths to prevent ruptured SG inventory loss. A 2nd contingency allows for RCS inleakage to raise level in the ruptured SG.

INSTRUCTIONS	CONTINGENCY ACTIONS

***50.** **Cooldown and depressurize ruptured SG by performing ANY of the following methods (listed in order of preference).**

CAUTION

Allowing backflow of 2% SG level to RCS will reduce boron concentration about 1%.

A. SG Backflow to RCS (preferred):

1) IF ANY RCP running, THEN maintain ruptured SG level 10% to 38% as follows:

Correct level control band

a) Determine RCS Boron reduction using 2202.010 Attachment 39, SG Backflow Log.

b) Maintain RCS pressure within 50 psi of ruptured SG as RCP NPSH requirements allow using 2202.010 Attachment 48, RCS Pressure Control.

c) Lower RCS pressure below ruptured SG pressure to allow SG inventory to flow into RCS using 2202.010 Attachment 48, RCS Pressure Control.

d) Control RCS pressure within 50 psi of ruptured SG to maintain SG level 10% to 38% using 2202.010 Attachment 48, RCS Pressure Control.

(Step 50 continued on next page)

STEAM GENERATOR TUBE RUPTURE

2202.004

EOP STEP:

*50. **Cooldown and depressurize ruptured SG by performing ANY of the following methods (listed in the order of preference):**

EPG STEP:

*40.

This band allows for partial exposure of the tubes

SUPPLEMENTARY INFORMATION:

14.

DEVIATION? Yes

BASIS FOR DEVIATION:

The EOP, as does the EPG, prioritizes the different methods available for cooldown of the ruptured SG. EPG method "a" is not used in the EOP. This method would produce more waste water and cause more dilution of the RCS than method "b". ANO-2 chose to implement SG tube exposure as the primary method of depressurization for this reason.

The EOP uses SG tube exposure and RCS backflow as the preferred method to cool and depressurize the ruptured SG since this method is relatively fast, requires very little secondary makeup to the affected SG, and is fairly easy to control. RCS backflow should only be used as a means to cool and depressurize an isolated SG if one or more RCPs are operating. RCP operation mixes the incoming SG inventory with the borated RCS water, thus minimizing the dilution effect of the incoming secondary inventory by preventing the formation of a "slug" of unborated water in the RCS. SG level is lowered to allow exposure of the U-tubes (3), which will allow steam to condense on the cooler tube, surfaces.

A caution was added with information developed by Combustion Engineering for a boron dilution correlation factor for backflow into the RCS (4). This correlation provides operators with a "thumb rule" to quickly estimate the minimum expected boron concentration after a given backflow evolution (5). This provides information contained in supplementary information #14.

The EOP prioritizes other methods for cooldown and depressurization of the ruptured SG in order of preference. These methods include steaming of the isolated steam generator to the main condenser, feed and bleed using feedwater and the blowdown system (6, 7, 8), ambient cooling, and steaming the ruptured SG to atmosphere. These steps provide the plant specific actions to accomplish these actions, and are consistent with the EPG. This method minimizes the spread of contamination by preventing offsite releases. Steaming the ruptured SG to atmosphere should only be performed if the ruptured SG's pressure is preventing RCS depressurization to SDC conditions. This is consistent with the EPG Bases (1) for this step and ensures the steam dump method is used only as a last resort.

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

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

Yes → RO question

No ↓

Can the question be answere
immediate operator actions?

No ↓

Can the question be answere
entry conditions for AOPs or
that require direct entry to ma

No ↓

This question requires knowledge of 1 of the 5 cooldown options found in step 50 of SGTR.
Knowing that the SG is going to be cooled down is RO knowledge but details of the method is SRO level.
Also requires Tech Guide knowledge of why a specific SG level band is used.

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

Yes → RO question

No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

2011 NRC RO/SRO Exam

08-Nov-16

Bank:	1808	**Rev:**	0	**Rev Date:**	9/2/2010 1:55:24	**QID #:**	99	**Author:**	Coble
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	F	**Source:**	Modified IH Exam Bank OPS2-11534		
Search	1940012418	**10CFR55:**	43.1	**Safety Function**	4				
System Title:	Generic	**System Number**	GENERIC	**K/A**	2.4.18				
Tier:	3	**Group:**	1	**RO Imp:**	3.3	**SRO Imp:**	4.0	**L. Plan:**	A2LP-RO-ESGTR
OBJ	9								

Description: Emergency Procedures/Plan - Knowledge of the specific bases for EOPs.

Question:

ORIGINAL BANK QUESTION 1808 (2011 SRO Exam) was altered to create Q #92 on the 2017 NRC Exam.

Given the following:

* Unit 2 has tripped from full power due to a Steam Generator Tube Rupture.
* 'A' Steam Generator has been diagnosed as the ruptured SG.
* SG 'A' has been isolated.
* RCPs 2P-32A and 2P-32D are running.
* Cooldown and depressurization of the 'A' SG has commenced.
* All other system and components function as designed.

During this time, the level in the ruptured SG should be maintained between _______________% and the basis for this level is to ensure SG tubes are ______________________.

A. 10 to 38; covered to prevent release of gaseous activity from the RCS.

B. 10 to 38; partially uncovered to cool the steam space of the 'A" SG.

C. 20 to 45; covered to prevent release of gaseous activity from the RCS.

D. 20 to 45; partially uncovered to cool the steam space of the 'A" SG.

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐
2009	☐	☐
2011	☐	☑

Audit Exam History

2011	☐

Answer:

B. 10 to 38; partially uncovered to cool the steam space of the 'A" SG.

Notes:

Step 35 of the SGTR EOP has a step to maintain SG level 45 to 90% to limit any radioactive release. The basis for the 45% is to keep the SG tubes covered. However in Step 49 of the SGTR EOP, the process of cooling down the isolated SG begins and level is lowered to 10 to 38% to allow uncovering of the SG tubes thus transferring latent heat of the hot steam to the cooler RCS. C and D are incorrect because they list the wrong level to maintain. A and C are incorrect because they list the wrong basis.

References:

EOP 2202.004, SGTR EOP, Revision 10, Steps 35 and 49, pages 22,29.
TG 2202.004, SGTR EOP Tech Guide, Revision 10, Step 35 and 49, pages 52 and 70.

Historical Comments:

Question 93

Bank:	2423	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	93	**Author:**	Larry Burton
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	H	**Source:**	NEW FOR 2017 NRC Exam		
Search	0160002212	**10CFR55:**	43.2	**Safety Function**	7				
Title:	Non-Nuclear Instrumentation System (NNIS)	**System Number**	016	**K/A**	2.2.12				
Tier:	2	**Group:**	2	**RO Imp:**	3.7	**SRO Imp:**	4.1	**L. Plan:**	ASLP-RO-SURV
OBJ	1A								
Description:	Equipment Control - Knowledge of surveillance procedures.								

Question:

(Picture of Pzr level instruments attached to this question)

Given the following:

* Unit 2 is operating at 100% power.
* Pressurizer level instrument, 2LI-4627-2B, Channel 2, was declared Inoperable and removed from service three (3) shifts ago due to erratic readings.
* I & C has completed all required maintenance/rework activities.
* OPSLOG B18 = Post Accident Instrumentation.
* OPSLOG B19 = Remote Shutdown Instrumentation.
* 2LI-4627-1 = Channel 1 Pressurizer Level Instrument.
* 2LI-4627-2AN = Channel 2 Pressurizer Level Instrument.

In order to restore Operability of PZR Level Instrument 2LI-4627-2B, a channel check must be performed and documented using ____________________________ and the channel check must be performed using ________________________ PZR Level Instrument(s).

A. OPSLOG B18 ONLY; 2LI-4627-1 ONLY

B. OPSLOG B19 ONLY; 2LI-4627-2AN ONLY

C. OPSLOG B18 and OPSLOG B19; 2LI-4627-2AN ONLY

D. OPSLOG B18 and OPSLOG B19; 2LI-4627-1 and 2LI-4627-2AN

Answer:

D. OPSLOG B18 and OPSLOG B19; 2LI-4627-1 and 2LI-4627-2AN

Notes:

D is correct - Pressurizer level instrument, 2LI-4627-2B is both a PAM and Remote shutdown instrument therefore both LCOs 3.3.3.5 and 6 retest/surveillance must be satisfied and documented by the performance of both OPLOGs 18 and 19. PAM uses 2LI-4627-1 for comparison and RSDP uses 2LI-4627-2AN for it's comparison. Only 1 instrument to cross check is plausible because it would be true for any instrument not PAM or Remote shutdown related. Same logic applies for the validating procedure.

A. is wrong it only identifies the Control Room instrument (PAM).

B. is wrong both instruments but only one (PAM) retest document. Plausible because this instrument is a RSDP instrument.

C. is wrong it lists only 1 instrument but is correct for both retest documents (PAM & RSDP). It is plausible that there are instruments on the RSDP that are also PAM.

KA Match - Pressurizer level is a non-nuclear system and the question requires knowledge of the OPLOGs which are used to satisfy the surveillance requirements to restore Operability.

References:

OPSLOG18 - Post Accident Instrumentation
OPSLOG19 - Remote Shutdown Instrumentation
LCO 3.3.3.5 - Remote Shutdown Instrumentation
LCO 3.3.3.6 - Post-Accident instrumentation
STM-2-03-1 (017), Pressurizer Pressure and Level Control
Picture of PZR instruments in Control Room proved as a HANDOUT.
(All references verified current 11/10/16)

Picture of PZR Instrument provided

Historical Comments:

To be used on the 2017 NRC Exam
Rev 1 - placed distractors in correct order. Changed the stem from "Remote Shutdown Panel " to "Channel 2". Changed B to a single procedure and instrument. Changed C to both Surveillances on a single instrument this is plausible that a RSDP instrument is also PAM required.

ENTERGY

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ARKANSAS NUCLEAR ONE

DATE______________

AREA/POINTS	INSTRUMENT (Note 2)		MAX	OPER	UNITS	INDICATION		DIFF
	RED	GREEN	DIFF	DIFF		RED	GREEN	
NOTE - If any indication out of spec, then refer to Tech Spec 3.3.3.6. (Note 2)								
A S/G LEVEL	2LI-1079-1	2LI-1079-2	20	38	INCH			
B S/G LEVEL	2LI-1179-1	2LI-1179-2	20	38	INCH			
A S/G PRESSURE	2PI-1041-1	2PI-1041-2	60	104	PSIA			
B S/G PRESSURE	2PI-1141-1	2PI-1141-2	60	104	PSIA			
PRESSURIZER PRESSURE	2PI-4624-1B	2PI-4624-2	120	192	PSIA			
PRESSURIZER LEVEL	2LI-4627-1	2LI-4627-2B	3	4				
PRESSURIZER RELIEF TEMP	2TIS-4630	2TIS-4631	////////	//////				
CONTAINMENT PRESSURE WIDE RANGE AND NORMAL DESIGN RANGE (Note 1)	2PI-5605-1	2PI-5606-2	6	10	PSIA			
CONTAINMENT FLOOD LEVEL (P11402)	2LI-5645-1	2LI-5646-2	6	10	INCH			
RWT LEVEL	2LI-5636-1	2LI-5637-2	3	4	%			
EFW TO A S/G FLOW (Note 3)	2FIS-0710-1	2FIS-0718-2	////////	///////	GPM			
EFW TO B S/G FLOW (Note 3 & 4)	2FIS-0717-1	2FIS-0713-2	////////	///////	GPM			
RCS SUBCOOL MONITOR	2XI-4612-3	2XI-4612-4	8	10	°F			

TS 3.3.3.6, Post Accident Monitoring, Requires 2LI-4627-1

VERIFY THE FOLLOWING INSTRUMENTS OPERABLE PER APPLICABLE REQUIREMENT.

	INSTRUMENT		REQUIREMENT	INITIALS		//////
	RED	GREEN		RED	GREEN	//////
RCS SUBCOOL MONITOR (N/A in Modes 4, 5, and 6)	2XI-4612-3	2XI-4612-4	NOT FLASHING			N/A
CNTMT RAD MONITOR	2RITS-8925-1	2RITS-8925-2	GREEN LIGHT ON			N/A
PZR RELIEF VALVE AUDIO MONITOR	2VYI-4633-1	2VYI-4634-1	COMPLETE 2105.011 SUP 1			N/A
CONTAINMENT SUMP LEVEL TRANSMITTERS	2LT-5645-1	2LT-5646-2	LOCAL POWER SWITCH IN ON AND POWER LIGHTS LIT			N/A

Notes:

(1) 2PI-5605-1 AND 2PI-5606-2 satisfy the Tech Spec requirements for Normal Design Range and Wide Range (Table 3.3-10, Action 1 and 2).

(2) IF indicator suspect/OOS, BUT sensor signal available for readout, THEN sensor signal reading may be used per 1015.003B section 7.0.

(3) NUREG TS and Bases indicate the intent is two per loop per SG, so loss of any one of four requires TS entry.

(4) Dixson indicators will flash the bottom led segment on loss of signal, verify bottom led NOT flashing when reading 0

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ENTERGY

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DATE _______________

AREA/POINTS	INSTRUMENT CR NOTE 3	INSTRUMENT 2C80	MAX DIFF	OPER DIFF	UNITS	INDICATION CR	INDICATION 2C80	DIFF
LOG POWER	**2JI-9007-2B**	**2JI-9007-2AN**	0.6 DEC	1.0 DEC	%			(NOTE 1)
S/U CHANNEL POWER (not reliable > 1E5 CPS)	**2JR-9000 (RED)**	**2JI-9000A**	0.6 DEC	1.0 DEC	CPS			(NOTE 1)
T-COLD LOOP A	**2TR-4615**	**2TI-4615A**	5	7	°F			
T-COLD LOOP B	**2TR-4715**	**2TI-4715A**	5	7	°F			
PRESSURIZER PRESSURE	**2PI-4624-1B**	**2PI-4624-1AN**	120	192	PSIA			
PRESSURIZER LEVEL	**2LI-4627-2B OR L4627-2**	**2LI-4627-2AN**	3					
SDC FLOW	**2FIC-5091**	**2FI-5091**	200					
A S/G LEVEL	**2LR-1033**	**2LI-1033**	4	8	%			
B S/G LEVEL	**2LR-1133**	**2LI-1133**	4	8	%			
A S/G PRESSURE	**2PI-1041-1**	**2PI-1041A-1N**	60	104	PSIA			
B S/G PRESSURE	**2PI-1141-1**	**2PI-1141A-1N**	60	104	PSIA			
A CST LEVEL	**2LR-0605 (RED)**	**2LI-1937** (NOTE 2)	5	10	%			
B CST LEVEL	**2LR-0605 (BLUE)**	**2LI-1977** (NOTE 2)	5	10	%			
Q CST LEVEL	**2LIS-0727-2A**	**2LIS-0727-2N** (NOTE 2)	0.8	1.0	FEET			

LCO 3.3.3.5, Remote SD Panel requires 2-LI-4627-2AN

						INDICATION 2C14	INDICATION Local	
TCB STATUS	/////////////	/////////////	/////////////	/////////////	OPEN/ CLOSE			/////////////

AREA/POINTS	INSTRUMENT CR	INSTRUMENT LOCAL	MAX DIFF	OPER DIFF	UNITS	INDICATION CR	INDICATION LOCAL	DIFF
NOTES-Compare the following local indications to the applicable Control Room Instrument.								
A S/G LEVEL	**2LR-1033**	**2LI-1033A (MFRV)**	4	8	%			
A S/G LEVEL	**2LR-1033**	**2LI-1033B (USPP)**	4	8	%			
B S/G LEVEL	**2LR-1133**	**2LI-1133A (MFRV)**	4	8	%			
B S/G LEVEL	**2LR-1133**	**2LI-1133B (UNPP)**	4	8	%			

NOTE 1: Difference = Logarithm of the higher number minus the logarithm of the lower number.

NOTE 2: As per TS 3.3.3.5 bases, "With regard to CST level, the required Remote Shutdown Panel indication is that CST level indication associated with the CST aligned to the EFW system."

NOTE 3: IF indicator suspect/OOS, BUT sensor signal available for readout, THEN sensor signal reading may be used for channel checks against 2C80 indicators per 1015.003B section 7.0.

1.0 IF any indication Out of Spec, THEN refer to Tech Spec 3.3.3.5. __________

2.0 REMARKS: ____

PERFORMED BY:________________________ **REVIEWED BY**: ______________________ (S/M) **DATE**: ____________

FORM TITLE:	FORM NO.	REV.
REMOTE SHUTDOWN CHANNEL CHECKS	**OPS-B19**	**05/31/05**

RUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE
1. Logarithmic Neutron Channel	2C80	10^{-8} – 200%
2. Startup Channel	2C80	1 – 10^{6} cps
3. Reactor Trip Breaker Indication	-	OPEN-CLOSE
4. Reactor Coolant Cold Leg Temperature	2C80	0 - 600°F
5. Pressurizer Pressure	2C80	0 – 3000 psia
6. Pressurizer Level (This is instrument 2LI-4627-2B)	2C80	0 – 100%
7. Steam Generator Pressure	2C80	0 – 1200 psia
8. Steam Generator Level	2C80 and Local (at EFW Valves Control)	0 – 100%
9. Shutdown Cooling Flow Rate	2C80	0 – 8000 gpm
10. Condensate Storage Tank Level	2C80	0 – 100%

POST-ACCIDENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

As shown in Table 3.3-10.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-10.

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

Must have both Pzr level instruments

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Containment Pressure (Normal Design Range)	2	1
2. Containment Pressure (High Range)	2	2
3. Pressurizer Pressure	2	1
4. Pressurizer Water Level	2	1
5. Steam Generator Pressure	2/steam generator	1
6. Steam Generator Water Level	2/steam generator	1
7. Refueling Water Tank Water Level	2	1
8. Containment Water Level – Wide Range	2	2
9. Emergency Feedwater Flow Rate	1/steam generator	1
10. Reactor Coolant System Subcooling Margin Monitor	1	1
11. Pressurizer Safety Valve Acoustic Position Indication	1/Valve	1
12. Pressurizer Safety Valve Tail Pipe Temperature	1/Valve	1
13. In Core Thermocouples (Core Exit Thermocouples)	2/core quadrant	1
14. Reactor Vessel Level Monitoring System (RVLMS)	2	3, 4

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ZR Level Control System Annunciators

The Pressurizer Level Control System also causes several annunciators to alarm if level should reach the alarms respective setpoints. The following table lists each annunciator associated with the Pressurizer Level Control System as well as the parameters that will cause the alarm.

WINDOW NUMBER	DESCRIPTION	CAUSES
2K10-F6	Cntrl CH 1 Level Lo-Lo	• <29% on 2LC-4627-1N • Control Circuit Power Breaker Open 2Y1-7
2K10-G6	Cntrl CH 1 Level Lo	• > 5.2% level deviation below program 2LC-4627-1AN • Control Circuit Power Breaker Open 2Y1-7
2K10-H6	Cntrl CH 1 Level Hi-Hi	• > 13.2% Level deviation above program 2LS-4627-1N
2K10-J6	Cntrl CH 1 Level Hi	• > 4.5% Level deviation above program 2LC-4627-1BN • Control Circuit Power Breaker Open 2Y1-7
2K10-F7	Cntrl CH 2 Level Lo-Lo	• < 29% on 2LC-4627-2N • Control Circuit Power Breaker Open 2Y2-7
2K10-G7	Cntrl CH 2 Level Lo	• > 5.2% Level deviation below program 2LC-4627-2AN • Control Circuit Power Breaker Open 2Y2-7
2K10-H7	Cntrl CH 2 Level Hi-Hi	• > 13.2% Level deviation above program 2LS-4627-2N
2K10-J7	Cntrl CH 2 Level Hi	• > 4.5% Level deviation above program 2LC-4627-2BN • Control Circuit Power Breaker Open 2Y2-7

3.4 System Indications

The instrumentation associated with the Pressurizer Level Control System also provides indication of pressurizer level to the operators. Both Channel A and B provide pressurizer level indication on 2C04. 2LT-4627-1 feeds 2LI-4627-1 while 2LT-4627-2 inputs to 2LI-4627-2B.

Pressurizer level transmitter 2LT-4627-2 also provides pressurizer narrow range level indication to the Remote Shutdown Panel 2C80 instrument 2LI-4627-2AN.

A 2-pen recorder on 2C04 tracks the Pressurizer Level Setpoint from the RRS as well as the selected level channel. This recorder is designated 2LR-4628. The red pen on this recorder is the pressurizer level setpoint while the blue pen is the selected level channel 4627-A or B.

Attachment for Question 93

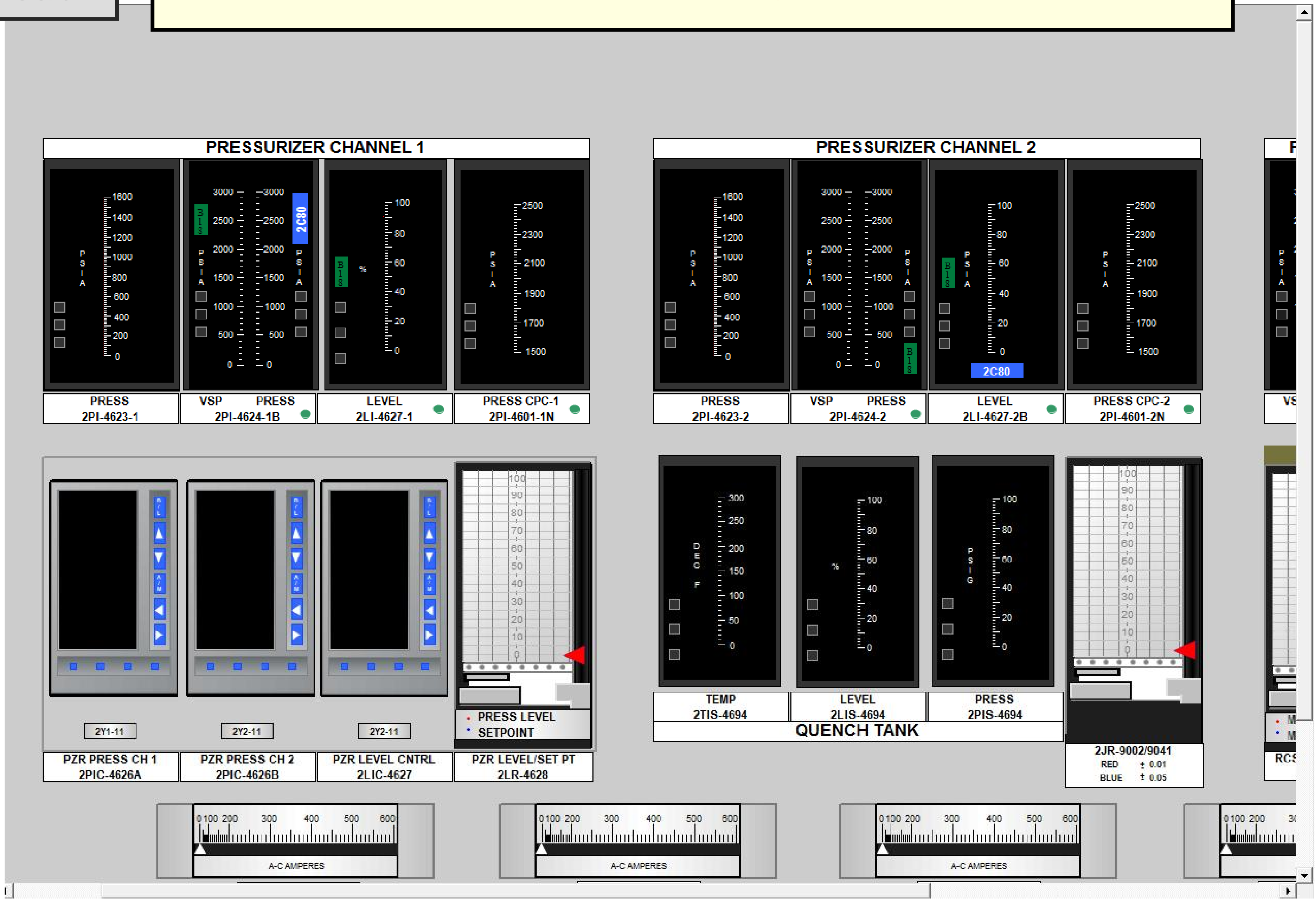

**Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2)
(Tech Specs)**

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- Knowledge of TS bases that is required to analyze TS required actions and terminology

Yes → SRO-only question

Knowing Survelliance Requirments is an SRO responsibility

No ↓

Question not be linked to 10 CFR 55.43(b)(2) for SRO-only might

Question 94

Bank:	2424	**Rev:**	2	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	94	**Author:**	Burton
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	1940012141	**10CFR55:**	43.7	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.1.41				
Tier:	3	**Group:**	1	**RO Imp:**	2.8	**SRO Imp:**	3.7	**L. Plan:**	A2LP-RO-FH
OBJ	4								

Description: Conduct of Operations - Knowledge of the refueling process.

Question:

Consider the following:

* Core reload is in progress.
* A fuel assembly is having insertion difficulties.
* Reactor Fuel Bridge Operator requests a fuel shuffle sequence change.

Per OP-2502.001, Refueling Shuffle, both the "SRO in charge of fuel handling" and __________ permission is required to authorize this change?

A. Operations Manager's

B. Shift Manager's

C. Shift Outage Manager's

D. Reactor Engineering's

Answer:

D. Reactor Engineering's

Notes:

D is correct: per the Refueling Shuffle procedure Section 11 Fuel Shuffle Sequence Change Guidelines - Authorization must be obtained from the following prior to making changes: SRO in charge of fuel handling - Reactor Engineering

A - is incorrect but plausible as the Operations Manager is tasked with oversight of the outage and must be contacted for fuel damage. Inspection.

B - is incorrect but plausible because the Shift Manager is responsible Control Room activities and holds an active SRO license.

C - is incorrect but plausible as the SOM is tasked with oversight of the outage including the "refueling" process.

KA Match - knowledge of a "Fuel Shuffle Request" is knowledge of the Refueling process.

References:

OP-2502.001 (052), Refueling Shuffle
EN-FAP-OU-105 (05), Refueling Outage Execution
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam

Rev 1 - corrected procedure number in stem
Rev 2 - re-formatted the question adding "SRO in charge of refueling to the stem" and asked
who's additional approval is required for a fuel shuffle.

6.0 LIMITS AND PRECAUTIONS

6.1 SRO in charge of fuel handling shall directly supervise and maintain oversight of all CORE ALTERATIONS.

- He shall have no other concurrent responsibilities during this operation (i.e. Spotter or fuel move communications). (10CFR50.54(m)(2)(iv))

6.2 Fuel/CEA movements shall only be performed by qualified Fuel Handlers.

6.3 If fuel damage suspected during fuel handling process, the SRO in charge shall contact Reactor Engineering and Operations Manager to determine whether suspected assembly should be inspected.

- Inspection determines if assembly is suitable for loading or if an alternate should be selected.

6.4 Containment polar crane shall NOT travel over/above the Refueling canal without permission of SRO in charge of fuel handling.

6.5 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in storage pool. Tech Spec 3.9.7.

6.6 Loading/Unloading

Reactor Engineering and OPS Manager must be contacted for potentially damaged fuel
Distractor "A"

- Sequence may b ion and handling equip
- Changes in cyc only affect sequence of ev y do NOT require written approval.
  - These changes shall be recorded in “Chronological Log”, Attachment B of this procedure.

6.7 Following cautions should be used to minimize possibility of inadvertent collisions between refueling machine hoist box or spreader and such items as core support barrel:

- Refueling machine operator should exercise caution when handling fuel in vicinity of core support barrel wall.
- Hoist box and spreader should be at full up position when moving into or out of the core region.

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11.0 FUEL SHUFFLE SEQUENCE CHANGE GUIDELINES

11.1 IF EITHER of the following applicable,

- Change required to Fuel Shuffle Sequence that affect destination or orientation of an assembly
- Changes due to handling problems or equipment malfunctions requires temporary placement of a fuel assembly in a location or orientation that is NOT the final location

THEN use following guidelines to make changes:

11.1.1 Authorization must be obtained from the following prior to making changes:

- SRO in charge of fuel handling
- Reactor Engineering

11.1.2 Authorizing personnel are responsible for ensuring completion of all applicable documentation.

11.1.3 Use “Nuclear Fuel Transfer Report Modification” (1022.012R) for the following:

- Record requested transfer change
- Record approval of transfer change
  - – Required refue telecom but sh personnel no l
- Transfer can NOT approved.

SRO in charge of Fuel Handling and Reactor Engineering must approve Fuel Shuffle sequence change

11.1.4 IF changes in sequence affects destination of Fuel Assembly being located in Spent fuel Pool,
THEN storage requirements must be met IAW Storage, Control, and Accountability of Special Nuclear Material (1022.012).

11.1.5 IF changes in sequence involve CORE ALTERATIONS,
THEN extreme caution required to verify correct mast orientation on refueling machine for subsequent steps of Item Control Area (ICA) Transfer Form (EN-NF-200, Att. 9.1).

- Act on 10/30 minute rule issues turned over by the Work Management Center (WMC) group supervision or received directly from the field.
- Address work implementation issues beyond WMC resolution capabilities.
- Address critical path, near critical path, or refueling operations impact concerns / issues.
- Attend OCC turnover and status update meetings as scheduled.
- Ensure lessons learned are captured and incorporated, as necessary, to improve outage performance.
- Provide oversight and coaching to Outage Team personnel specifically related to the following areas:
- Recognition and communication of any perceived high risk activities to OCC management.
- Setting of boundaries on activities which could place the plant in a high risk condition.
- Recognition of the need to stop and review any activity which is not proceeding as planned.

2.3.1 Outage Control Center Standard Positio

Shift Outage Manager has numerous responsibilities during the outage. Plausibility of "C"

[1] Minimum requirements for staffing the OCC an listed below.

2.3.1.1 SHIFT OUTAGE MANAGER

[1] Responsible for overall outage execution including ensuring critical path, near critical path and major projects are maintained on track.

[2] Direct the Outage Control Center staff as necessary. Responsible for hour-to-hour conduct of outage.

[3] Ensure shutdown risk is evaluated each shift.

[4] Establish outage priorities for the station.

[5] Verify prompt action is taken to address outage performance gaps and emergent issues

[6] Ensure that OCC leads look ahead 48 hours in the schedule to ensure major outage evolutions are ready for implementation.

[7] Ensure the Operations Shift Manager (SM) is kept informed of major upcoming schedule activities and emergent issues.

[8] Chair OCC turnover meetings and OCC shift briefings.

[9] Provide oversight and coaching for outage personnel.

[10] Recognize and communicates any perceived high risk activities to the OCC Team.

[11] Set boundaries on activities which could place the plant in a high risk condition. Recognize the need to stop and review any activity which is not proceeding as planned.

[12] Reports to the Outage Director.

[13] While the OCC is staffed during refueling outage execution, the SOM is expected to notify the Corporate Duty Manager on significant deviations from the planned schedule. For example, if the site scrams during start-up or an emergency action level (EAL) is entered.

[14] Ensures the 6 Hour Status Update Report is distributed as detailed in Attachment 7.8.

2.3.1.2 EMERGENT WORK MANAGER

[1] Manage outage scope change process once OCC is established as operational through outage completion in accordance with EN-FAP-OU-104, Refueling Outage Scope Identification and Control.

[2] Verify that the Scope Change Form (SCF) initiation section information is accurate. This includes man-hour estimates, dose estimates and cost estimates.

[3] Provide guidance and ownership of emergent work scope until the work is approved and placed into the outage schedule. The Emergent Work Manager will provide direction to and interface with planners, schedulers, tagging coordinator, and shop coordinators to ensure that scope changes are processed efficiently to support Outage Schedule requirements. The Emergent Work Manager will facilitate Site Lead team notification of all scope changes on a daily basis.

[4] Interface with Lead Outage Scheduler to ensure all approved outage scope additions/deletions are promptly integrated into or removed from the refueling outage schedule no later than 24 hours after approval.

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

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

Knowledge of Administrative procedures specifically Refueling is an SRO responsibility

No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Question 95

Bank:	2425	**Rev:**	2	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	95	**Author:**	Burton
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	1940012304	**10CFR55:**	43.4	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.3.4				
Tier:	3	**Group:**	1	**RO Imp:**	3.2	**SRO Imp:**	3.7	**L. Plan:**	ASLP-RO-EPLAN
OBJ	4								

Description: Radiological Controls - Knowledge of radiation exposure limits under normal or emergency conditions.

Question:

Given the following:

* General Emergency has been declared.
* TSC has been activated.
* Repair activities need to be performed on LPSI Pump 'A', 2P-60A.
* LPSI 'A', 2P-60A, has been deemed valuable property.
* Dose readings in the repair area are 120 Rem/hr.

Per OP-1903.033, Protective Action Guidelines for Rescue/Repair & Damage Control Teams, the ____________ is responsible to AUTHORIZE the exposure for this repair and the MAXIMUM amount of time the repair team can remain in the area is _______________.

A. Emergency Plant Manager; 2.5 minutes

B. Emergency Plant Manager; 5.0 minutes

C. Radiological Coordinator; 2.5 minutes

D. Radiological Coordinator; 5.0 minutes

Answer:

B. Emergency Plant Manager; 5.0 minutes

Notes:

Limit for valuable property is 10R, dose rate is 2R/minute so team can stay 5 minutes. Shift Manager, Emergency Plant Manager and Emergency Director have authorization for exceeding 10CFR exposure limits. 2.5 minutes is based of life saving exposure of 5 R/year limit.

A. Is Incorrect - EPM typically assumes this duty after TSC activation. Time is wrong but plausible as this is based on the 5 rem yearly limit.

B. Is Correct - EPM is correct and time is correct.

C. Is Incorrect - identifies an incorrect person of responsibility but plausible as this is a dose issue which would normally be the Radiological Coordinator's responsibility. Time is wrong but plausible as this is based on the 5 rem yearly limit.

D. Is Incorrect - identifies an incorrect person of responsibility but plausible as this is a dose issue which would normally be the Radiological Coordinator's responsibility. Time is wrong but plausible as this is based on the 5 rem yearly limit.

KA Match - This question requires the knowledge of emergency exposure limits

References:

1903.033 (024), PAG for Rescue/Repair & Damage Control Teams
(Reference verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Rev 1 - changed stem to after TSC is activated. This changes the correct answer to EPM and makes the Radiological Coordinator plausible.
Rev 2 - changed the wrong limit from 25 rem to 5 rem. Changed order making B correct.

4.0 DEFINITIONS

4.1 Emergency Direction and Control - Overall direction of facility response which must include the non-delegable responsibilities for the decision to notify and to recommend protective actions to Arkansas Department of Health personnel and other authorities responsible for offsite emergency measures. The direction of facility operations to mitigate accident consequences remains with the Emergency Plant Manager in the Technical Support Center and/or the Shift Manager in the Control Room.

4.2 Emergency Response Organization (ERO) - The ... composed of the IRS, the Technical ... Operational Support Center Staff, t... the Emergency Operations Facility Staff ...vide manpower and other resources necess... response to an emergency situation.

Shift Manager performs this function prior to activation of the TSC, the EPM after activation

5.0 RESPONSIBILITY AND AUTHORITY

5.1 The Shift Manager, Emergency Plant Manager (EPM) or Emergency Director (ED) is responsible for approving personnel exposures exceeding the limits of 10CFR20 under the conditions specified in this procedure. After activation of the TSC, the Emergency Plant Manager will typically assume the responsibility for approving in-plant personnel exposures exceeding 10CFR20 limits.

5.2 The Emergency Plant Manager is responsible for the overall development and implementation of rescue/repair and damage control plans. He shall direct the Maintenance Coordinator to develop those plans as appropriate and shall direct the OSC Manager to implement the formulated plans.

5.3 The Maintenance Coordinator is responsible for the development of repair and damage control plans under the direction of the Emergency Plant Manager. He shall provide the OSC Manager with recommendations ...oped by the TSC staff. He shall also report all results to the ...ency Plant Manager.

Has radiological responsibilities making this a good distractor

...SC Manager is responsible for implementation of rescue/repair and ...e control plans. He shall ensure that appropriate rescue/repair ...amage control teams are selected, briefed upon the specific ...tives of the mission, and that the progress of the teams is tracked. He shall report all results to the Emergency Plant Manager.

5.5 The Radiological Coordinator is responsible for providing oversight for all Health Physics activities and for ensuring that the Emergency Plant Manager is informed of current radiological conditions.

5.6 The RAD Coordinator is responsible for providing Health Physics coverage for rescue/repair and damage control operations. He is responsible for directing onsite monitoring and decontamination and shall also provide radiological protection information for rescue/repair team briefings. He will report all results to the OSC Manager.

Protection of "valuable property" - per stem

Dose limit* (Rem TEDE)	Activity	
5	All	
10	Protecting valuable property	Lower dose not practicable
25	Life saving or protection of large populations	Lower dose not practicable
>25	Life saving or protection of large populations	Only on a voluntary basis to persons fully aware of the risks involved (refer to Attachment 1 of this procedure for health risks).

Yearly Dose Limit

* Workers performing services during emergencies should limit dose to the lens of the eye to <u>three times</u> the listed value and doses to any other organ (including skin and body extremities) to <u>ten times</u> the listed value.

6.1.4 Rescue/repair and damage control personnel shall perform their duties in the most safe and efficient manner possible. Once their operations have been completed, they shall follow self-monitoring and personnel decontamination procedures as specified by the RAD Coordinator.

6.2 ACTIONS

NOTE
Prompt medical attention shall take precedence over RP procedures for a seriously injured individual.

6.2.1 Emergency Medical Team may enter Radiologically Controlled Areas without SRDs or Alarming Dosimeters during a “Personnel Emergency” as long as an RP Technician is providing radiological instructions and is monitoring dose rates and time in the area.

6.2.2 Personnel selected for the rescue/repair and damage control teams should report to the OSC (unless otherwise instructed) for their briefing.

6.2.3 The rescue/repair and damage control team leader shall function under the direction of the Shift Manager/OSC Manager.

C. Facility licensee procedures required to obtain authority for design and operating changes in the facility. [10 CFR 55.43(b)(3)]

Some examples of SRO exam items for this topic include:

- 10 CFR 50.59 screening and evaluation processes.
- Administrative processes for temporary modifications.
- Administrative processes for disabling annunciators.
- Administrative processes for the installation of temporary instrumentation.
- Processes for changing the plant or plant procedures.

Section IV provides an example of a satisfactory SRO-only question related to this topic.

D. Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. [10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include:

- Process for gaseous/liquid release approvals, i.e., release permits.
- Analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures.
- Analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits.

SRO-only knowledge should not be claimed for questions that can be answered *solely* based on RO knowledge of radiological safety principles; e.g., RWP requirements, stay-time, DAC-hours, etc.

E. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

This 10 CFR 55.43 topic involves both 1) assessing plant conditions (normal, abnormal, or emergency) and then 2) selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. One area of SRO level knowledge (with respect to selecting a procedure) is knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose.

The applicant's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item, for example:

QID-95
4 of 4

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

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

Knowledge of Administrative procedures is an SRO responsibility

No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Question 96

Bank:	2426	**Rev:**	0	**Rev Date:**	8/4/2016	**2017 TEST QID #:**	96	**Author:**	Wright
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	F	**Source:**	NRC Exam Bank 1705		
Search	1940012211	**10CFR55:**	43.3	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.2.11				
Tier:	3	**Group:**	1	**RO Imp:**	2.3	**SRO Imp:**	3.3	**L. Plan:**	A2LP-SRO-MNTC
OBJ	7								

Description: Equipment Control - Knowledge of the process for controlling temporary design changes.

Question:

Consider the following:

* System Engineering has determined that connecting a contaminated and non-contaminated system with a rubber hose would be a Temporary Modification

Before the hose is installed between these two systems, who must authorize installation AND what is the MINIMUM number of check valve(s) that must be installed in series to satisfy procedural requirements per EN-DC-136, Temporary Modifications?

A. Shift Manager; 1 check valve

B. Shift Manager; 2 check valves

C. System Engineering Manager; 1 check valve

D. System Engineering Manager; 2 check valves

Answer:

B. Shift Manager; 2 check valves

Notes:

Engineering Manager is plausible because this position owns the Temp Modification process and has other responsibilities within the procedure. (1) check valve is plausible because (1) backflow preventer meets requirements, also a single check valve can be a " boundary isolation" for a clearance.

B. Is Correct - Shift Manager has authorization responsibility and 2 check valves is the procedural requirement.

A. is Incorrect. Shift Manager has the authorization but 2 check valves are required as stated above.

C and D are incorrect. The SM authorizes and 2 check valves are required. Plausible. Because the System Engineering Manager owns the Temp Mod process. 1 check valve is plausible because the tagging procedure does allow the use of one check for a boundary and can also satisfy Containment Isolation criteria.

KA Match - Knowing who must authorize and the conditions of a Temp Modification is the "Knowledge" of the process.

References:

EN-DC-136 (013), Temporary Modifications
EN-OP-102 (018), Protective and Caution Tagging
LCO 3.6.3.1 Bases

(All references verified current 11/10/16)

Historical Comments:

NRC Exam Bank Question #1705 was used on the 2009 NRC Exam
Minor alternations include changing order making D instead of B correct.
Changed WRONG answers from 3 to 1 check valve for plausibility.
To be used on the 2017 NRC Exam

4.8 TECHNICAL REVIEWER / DESIGN VERIFIER:

[1] Verifies Temporary Modifications have been designed properly.

[2] Verifies Temporary Modifications maintain adequate plant safety and reliability.

[3] Verifies proper specification of Temporary Modification testing requirements (post-installation & removal).

[4] Ensures completeness of the package.

4.9 SHIFT MANAGER: [FSAR 13.1.2.3.10 Para 3 s1]

[1] Ensures that an Operations Department review is performed of each Temporary Modification package prior to installation.

[2] Ensures Temporary Modifications are reviewed to determine if the Temporary Modification is considered a compensatory measure (operational) th… tracked via a Condition Report. If the determination is made that the … Modification is a compensatory measure (operational) ensure that a … is issued.

Shift Manager authorizes the Temp Mod

[3] Authorizes installation and removal of Temporary Modifications.

[4] Maintains the Temporary Modification Log of installed Temporary Modifications. [ANSI N18.7 5.2.6 s 15]

[5] Verifies that Control Room Drawings are annotated and the Temporary Modification Integrated Drawing List is properly maintained, if used.

[6] Ensures that shift personnel are aware of the installed Temporary Modifications.

[7] Ensures narrative log is updated to track Temporary Modi… removed during their shift.

Has responsibilities but not approval authority

[8] Maintains overall administrative control of installed Tempo…

4.10 MANAGER, SYSTEMS & COMPONENTS: [COM-92-04427]

[1] Owns the site Temporary Modification process.

[2] Designates Temporary Modification Owner for each Temporary Modification.

[3] Communicates the number and status of Temporary Modifications to the site and advocates timely removal of Temporary Modifications.

4.10 cont.

[4] Ensures that FSAR updates are submitted for Temporary Modifications that remain in place beyond the periodic FSAR update cycle.

4.11 TEMPORARY MODIFICATION OWNER:

[1] Tracks Temporary Modifications for applicability and advocates timely removal.

4.12 TRAINING MANAGER:

[1] Ensures that lesson plans and the simulator are updated as necessary for Temporary Modifications.

4.13 VERIFIER: [ANSI N18.7 5.2.6 s14] [ACT-92-05152]

[1] Certifies that the installation and removal of Temporary Modifications has been performed as designed. This activity can be performed by any of the following individuals:

- Operator
- Installation or Removal Supervisor
- Engineer
- Personnel qualified to ANSI N45.2.6 Level I, II, or III

4.14 **Site IT - OT Cyber Security Lead:**

[1] Is a member of the Site Manager, Information Technology**/**Operational Technology (IT **-** OT) organization and is responsible for:

(a) Fulfilling the role of Cyber Security Specialist at the site to perform cyber security assessments of CDAs.

(b) Performing activities such as CDA identification, CDA design modification, control of portable digital media to CDAs, cyber incident response, and self-assessment, per cyber security implementing procedures.

5.0 DETAILS

5.1 PRECAUTIONS AND LIMITATIONS

[1] It is expected that Temporary Modifications should be short in duration, and few in number.

[2] Electrical jumpers used for testing purposes that are removed and installed from the same terminal points during the test are typically equipped with a properly sized switch to provide for positive breaking of the jumper circuit. Electrical jumpers used for testing purposes should be used in conjunction with approved procedures. [ACTS 5245]

[3] Electrical jumpers should be clearly identified (e.g. tagging, contrasting colors, or sufficient length) to be distinguishable from permanent wiring. [A-3771] [A-15360] [P-3768]

[4] Mechanical jumpers should be assessed to avoid the potential for cross-connecting a contaminated system with a non-contaminated system (e.g., hydro lazing or flushing).
IF cross connecting is required,
THEN two check valves in series or a back flow preventer should be installed to prevent the contamination of the clean system (reference 10CFR50 Appendix A GDC 60).

2 check valves

[5] Independence between redundant channels in protective systems should be maintained in accordance with site-specific procedures for cable routing requirements.

[6] Temporary Modifications should be designed and installed considering the design requirements related to redundant channels and should not have the potential to [...]dant channels from a single failure. [LER-99-015-01]

[7] Temporary Modifications, where the installation has been completed, that are left unattended should be tagged using Temporary Modification Tags including those installed by an approved procedure. Temporary Modifications are considered unattended if line of sight is not maintained. [ACTS 1679] [LER-99-015-01]

> NOTE
> Tagging and logging is not required for implementation of emergency or off-normal/abnormal operating procedures.

[8] The use of alligator or bulldog clips should be limited to circuits that cannot be de-energized at both end points due to an automatic action. Alligator or bulldog clips **SHALL NOT** be used in safety related circuits or within a seismic panel. [ACTS 5223]

[9] Temporary Modification Packages which have been installed (or received Operations Department approval) under a previous revision of this procedure, may be installed or remain in place. Temporary Alteration Packages approved prior to the initial effective date of this procedure shall not be left in place after the completion of the following refueling outage. [A-11604]

2.5 Vents and drains must also be opened as required to release dangerous pressures or harmful substances. One vent or drain should normally be Danger Tagged open to depressurize systems and to prevent them from re-pressurizing. There will be cases when this is not possible due to system designs.

2.6 Valves (MOV, AOV, Solenoid, and Manual Valve) with hand wheels shall be Danger-tagged when the valve provides isolation from process fluids. If a valve is inaccessible, can be made inaccessible due to changing radiological conditions, or is in an area with restricted access due to High Radiation, then it would be acceptable not to tag the inaccessible valve when the following conditions are met:

- The component can be verified that it is in the required position by using alternate methods.

AND

- The inaccessible valve will not have work performed on it and will remain inaccessible for the duration of the Tagout.

AND/OR

- A Caution tag may be installed at entrance to area with inaccessible valve with information that valve is part of Tagout boundary and entrance to area must be approved by the Shift Manager. For overhead valves in general areas, the area under the valve can be flagged off with the Caution tag installed at flagging.

2.7 Non-Manual valves (MOV, AOV and Solenoid) without valve operator Danger tagged when the valve provides fluids. This is to prevent the operator from being remov These Non-Manual valves should not normally be usec valves are available**.** If the valve is inaccessible then re for actions on inaccessible valves.

Single check valve can be used as an Isolation Boundary Plausibility of A and C

2.8 The use of check valves for isolation boundaries shall be avoided if at all possible.

If a Tagout requires the use of a check valve as an isolation boundary, then Shift Manager concurrence must be obtained and all individuals working under the Tagout shall clearly understand the potential risks of this type of isolation boundary. These risks will be listed in the hazards section of the Tagout.

2.9 When blank flanges are used as boundary isolations, they shall be Danger tagged.

2.10 When valves that have reach rods are tagged, the tag shall be placed on the reach rod hand wheel and an additional tag may be hung on the valve. If the reach rod is disconnected or broken then hang the tag on the valve and hang a Caution tag on the reach rod hand wheel indicating the reach rod is broken.

3/4.6.3 CONTAINMENT ISOLATION VALVES (continued)

With one or more CIVs inoperable in one or more penetrations, the method of penetration isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Examples of isolation barriers that meet this criterion are a closed and de-activated automatic reactor building isolation valve, a closed manual valve, a blind flange, and a check valve (inside containment) with flow through the valve secured. Note that if both CIVs in a given penetration are inoperable while the penetration remains open, LCO 3.0.3 is also applicable until the penetration is isolated by at least one isolation barrier as described above, except as permitted under the aforementioned LCO Note.

Single check valve can be used for containment Isolation
Plausibility of A and C

e), the plant must be brought to a MODE in this status, the plant must be brought to at hin the following 6 hours. Remaining within the he plant risk in MODE 4 is similar to or lower hnical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. These Actions are modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

Unless there is reason to believe the seating capability of the affected valve(s) has degraded, no verification of leakage through the penetration is required. If seat degradation is suspected, a vent or drain within the penetration boundary may be used to verify sufficient seating has taken place. Any noted leak-by should be evaluated in accordance with the Containment Leakage Rate Testing Program of Specification 6.5.16. CIVs closed due to inoperabilities in the respective penetration must be verified to remain in the isolated position once every 31 days in accordance with Surveillance Requirement 4.6.1.1.a.

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

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

Knowledge of Administrative procedures is an SRO responsibility

Does the question require one or more

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

09 NRC RO/SRO Exam

15-Aug-16

Bank:	1705	**Rev:**	1	**Rev Date:**	7/24/2009 7:02:11	**QID #:**	96	**Author:**	Jim Wright
Lic Level:	S	**Difficulty:**	2	**Taxonomy:**	F	**Source:**	New		
Search	1940012211	**10CFR55:**	43.3	**Safety Function**					
System Title:	Generic	**System Number**	GENERIC	**K/A**	2.2.11				
Tier:	3	**Group:**	1	**RO Imp:**	2.3	**SRO Imp:**	3.3	**L. Plan:**	A2LP-SRO-MNTC
OBJ	7								

Description: Equipment Control - Knowledge of the process for controlling temporary design changes.

Question:

ORIGINAL BANK QUESTION 1705 (2009 SRO Exam) was altered to create QID# 96 on the 2017 NRC Exam.

Given the following:

* System Engineering has determined that connecting a contaminated and non-contaminated system with a rubber hose would be a Temporary Modification

Before the hose is installed between these two systems, who must authorize installation AND what is the minimum number of check valves that must be installed in series to satisfy procedural requirements?

A. System Engineering Manager; 2 check valves

B. Shift Manager; 3 check valves

C. System Engineering Manager; 3 check valves

D. Shift Manager; 2 check valves

QID use History

	RO	SRO
2003	☐	☐
2005	☐	☐
2006	☐	☐
2008	☐	☐
2009	☐	☑

Audit Exam History

2009	☐

Answer:

D. Shift Manager; 2 check valves

Notes:

A and C are incorrect because the System engineering manager cannot authorize Temporary Modification installation- Engineering is responsible to ensure that the Temporary Modification is acceptable for safe plant operation. B is incorrect because 2 check valves not 3 are required for mechanical hose jumper installation.

References:

Maintenance Procedure EN-DC-136 4.9 [2] REV 3
Maintenance ProcedureEN-DC-136 5.0 [4] REV 3
Lesson Plan ASLP-SRO-MNTC Rev 4 Objective 7: Describe the operations department responsibilities associated with installation and restoration of a temporary modification package.

Historical Comments:

Question 97

Bank:	2427	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	97	**Author:**	Burton
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	1940012427	**10CFR55:**	43.5	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.4.27				
Tier:	3	**Group:**	1	**RO Imp:**	3.4	**SRO Imp:**	3.9	**L. Plan:**	A2LP-RO-AFIRE
OBJ	2								

Description: Emergency Procedures/Plan - Knowledge of "fire in the plant" procedure.

Question:

Given the following:

* Due to a fire, the Unit 2 CRS has entered OP 2203.034, Fire or Explosion.
* Unit 2 Control Room has become uninhabitable due to the fire.

Due to the subsequent Unit 2 Control Room evacuation, the _______ will be responsible to continue with OP 2203.034, Fire or Explosion procedure and if the Fire Brigade requires additional assistance the __________ Fire Department will be requested per OP-2203.034, Fire or Explosion.

A. Unit 1 SM; London

B. Unit 1 SM; Russellville

C. Unit 2 SM; London

D. Unit 2 SM; Russellville

Answer:

A. Unit 1 SM; London

Notes:

A is correct. Per section 2 (Protected Area) of the Fire and Explosion AOP if the Control Room is rendered uninhabitable then direction is provided to turn over performance of the AOP to Unit 1 Control Room.
Later in section 2 the SRO is directed to call 911 and notify the Dispatcher that ANO needs assistance from the London Fire Department which has a letter of agreement to respond to the plant

B is incorrect. U1 Control Room is correct/plausible, but Russellville FD is wrong. However, the Russellville FD is a larger FD just across the lake and can be called for additional assistance making it plausible.

C is incorrect. U2 Control Room SM is wrong, but the London FD is correct/plausible. The Unit 2 SM will enter OP 2203.014, Alternate Shutdown AOP, report to the TSC and direct the actions from there so it is plausible that the Shift Manager could also continue the Fire and Explosion AOP. The CRS at this point will be taking actions in the field per the Alternate Shutdown AOP

D. is incorrect . U2 Control Room SM and Russellville FD are both wrong. The Unit 2 SM will enter OP 2203.014, Alternate Shutdown AOP, report to the TSC and direct the actions from there so it is plausible that the Shift Manager could also continue the Fire and Explosion AOP. Also, the Russellville FD is a larger FD just across the lake and can be called for additional assistance making it plausible.

KA Match - matches the KA since question requires procedural knowledge of procedural guidance contained within the Fire or Explosion AOP.

References:

AOP-2203.034 (019), Fire or Explosion
AOP-2203.014 (032), Alternate Shutdown
Arkansas Nuclear One - Emergency Plan (041)
(All references verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Rev 1 - removed "large" from the stem and "control room" from the distractors.

*11. **<u>IF</u>** fire is on Unit 1 **<u>AND</u>** ANY of the following occur:

- Unit 1 Control Room rendered uninhabitable.
- Fire in Control Room threatens immediate damage to major portions of vital controls.
- Fire in Cable Spread Room threatens immediate damage to significant number of cables.

<u>THEN</u>:

A. **DIRECT** Unit 2 Control Room to perform remainder of this procedure.

B. **GO TO** 1203.002, Alternate Shutdown.

*12. **<u>IF</u>** fire is on Unit 2 **<u>AND</u>** ANY of the following occur:

Unit 1 is required to take over this AOP

- Unit 2 Control Room rendered uninhabitable.
- Fire in Control Room threatens immediate damage to major portions of vital controls.
- Fire in ANY of the below listed areas threatens immediate damage to significant number of cables.

2098-L	Cable Spreading Room
2199-G	Unit 2 Control Room
2150-C	Core Protection Calculator Room (elevation 404)
2136-I	Radiation Protection (RP) Office (elevation 386)
2137-I	Upper South Electrical Penetration Room (includes Hot Tool Room and Hot I&C Shop)
2119-H	Control Room Printer Room
2098-C	Core Protection Calculator Room (elevation 372)

<u>THEN</u>:

A. **DIRECT** Unit 1 Control Room to perform remainder of this procedure.

B. **GO TO** 2203.014, Alternate Shutdown.

*15. **<u>IF</u>** Fire Brigade requires additional assistance,
<u>THEN</u>:

{P-646}

A. Using an outside line, **NOTIFY** 911 Dispatcher of the following:

- ANO needs assistance from London Fire Department
- Type of fire (structural, electrical, etc.)
- Location of fire
- Size of fire (as much description as possible)
- Access point if other than north gate
- Fire department is to coordinate with ANO Fire Brigade Leader

London FD is the first assistance asked for

B. **NOTIFY** Security to dispatch TWO Security Officers to receive and direct Fire Department personnel.

C. **<u>IF</u>** fire located in radiological controlled area,
<u>THEN</u> REQUEST Radiation Protection to provide the following:

- Anti-C's
- Dosimetry
- Continuous RP escort
- Brief description of access route, fire area, and egress route from fire.

D. **REQUEST** additional Emergency Teams as needed.

16. **<u>IF</u>** ANY injured personnel,
<u>THEN</u> REFER TO 1903.023, Personnel Emergency.

17. **<u>WHEN</u>** fire zone number and name are known,
<u>THEN</u> INFORM Fire Brigade Leader.

18. **<u>IF</u>** opposite Unit STA available,
<u>THEN</u> DIRECT opposite unit STA to implement the Pre-Fire Plan.

2.2.9 Civil Air Patrol (CAP)

Provides aerial reconnaissance, and logistic support for aerial monitoring and search and rescue activities.

2.2.10 Other State and Local Resources

For Federal agencies responding to an emergency at ANO, the Bill and Hillary Clinton National Airport/Adams Field located in Little Rock is the nearest airport with commercial flight service. The Russellville Airport can accommodate Falcon 10 jet traffic or its equivalent.

The Russellville Airport is the closer of the two located approximately 8 miles from the site. The State of Arkansas Backup SEOF is located at the Entergy Arkansas, Inc. Russellville Service Center. This facility is referenced in the State and county emergency plans.

2.3 LAW ENFORCEMENT SUPPORT ORGANIZATIONS

Support from local, State, and/or Federal law enforcement agencies may be required to supplement the Arkansas Nuclear One Emergency Response Organization and to direct emergency efforts conducted beyond the ANO property boundary. The agencies listed below assist in the event of an emergency at Arkansas Nuclear One. Letters of agreement with these agencies are contained in Appendix 1. Interfaces between the Initial Response Staff, the Emergency Response Organization, and support groups are shown in Figures B-3, B-8 and B-10.

2.3.1 Sheriff Departments (Pope, Yell, Johnson, and Logan Counties)

When requested, the Sheriff Departments aid in security, traffic control, evacuations, emergency transportation and provide backup communications as required. The support of the Pope, Yell, Johnson, and Logan County Sheriff Departments are coordinated through the Arkansas Department of Emergency Management.

2.3.2 Arkansas State Police

Other FD such as Russellville will be asked for help as needed. Making them a plausible choice

The Arkansas State Police provides traffic contro
communications support where necessary.

2.4 FIRE SUPPORT ORGANIZATIONS

The local fire organization which has agreed to support the Arkansas Nuclear One Emergency Response Organization is listed below. A letter of agreement with this agency is contained in Appendix 1. Interfaces between the Initial Response Staff, Emergency Response Organization and the local fire department are shown in Figures B-3, B-8, and B-10.

2.4.1 London Fire Department

When requested, the London Fire Department dispatches men and equipment to assist in fighting fires. The London Fire Department assists the Fire Brigade, as required. The London Fire Department coordinates the support efforts of other fire departments in the area.

ALTERNATE SHUTDOWN

PURPOSE

This procedure provides alternate shutdown capability to comply with 10 CFR 50.48, Fire Protection, and to mitigate the consequences of a significant fire in any one of the fire zones listed below.

ENTRY CONDITIONS

ANY of the following conditions exist:

1. Fire in the Control Room that renders the Control Room uninhabitable.

2. Fire in the Control Room that threatens damage to a major portion of vital controls.

NOTE

A severe fire exists when ANY of the following conditions are present:

- **Smoke in zone prevents assessing fire status.**
- **Door/room is hot preventing access to zone.**
- **Fire in cable tray affecting several cables.**
- **Fire in 4160v bus or 480v load center.**

3. Confirmed fire in ANY of the fire zones listed below that is, or becomes severe.

2098-L	Cable Spreading Room
2199-G	Unit 2 Control Room
2150-C	Core Protection Calculator Room (elevation 404)
2136-I	Radiation Protection (RP) Office (elevation 386)
2137-I	Upper South Electrical Penetration Room (includes Hot Tool Room and Hot I&C Shop)
2119-H	Control Room Printer Room
2098-C	Core Protection Calculator Room (elevation 372)

EXIT CONDITIONS

1. Plant control is re-established from the Control Room.

OR

2. An Alternate Shutdown Cooldown has been successfully completed.

NOTE

- **Operator actions should NOT be delayed for Radiation Protection, Security or any other concerns.**
- **Operators are to take actions to mitigate the event, and these actions should be taken as safely as possible. However, prompt response to the emergency may not support the use of some Personal Protective Equipment (PPE). Delaying accident response could lead to a worse condition than not wearing the PPE. If time allows, PPE should be utilized, but not at the expense of delaying critical actions.**
- **Normal exposure limits apply except as approved by SM or Emergency Coordinator.**
- **Elevators should NOT be used when performing this procedure.**
- **Prompt completion of these actions overrides all other procedures, technical specifications or verbal directions other than those from Operations Management.**

5. **The following designated personnel GO TO the Alternate Shutdown Cabinet and obtain assigned equipment carrier (if applicable) for performance of their assigned section:**
   - CRS - Section 4, CRS Follow-up Actions.
   - EO - Section 5, EO Follow-up Actions.
   - RO 1 - Section 6, RO 1 Follow-up Actions.
   - RO 2 - Section 7, RO 2 Follow-up Actions.
   - AO – Section 8, AO Follow-up Actions.
   - ASDO – Section 9, ASDO Follow-up Actions.

6. Shift Manager perform the following:

A. Notify the following personnel to report to the Technical Support Center (TSC) and perform assigned sections as applicable:

- Shift Technical Advisor (Section 3, STA Follow-up Actions)
- Notifications Communicator (Unit 1)

B. Obtain a hand held radio.

C. Perform Radio check of hand-held radio (Channel 5).

NOTE
A copy of 2203.014, Alternate Shutdown, is located in the TSC.

D. Evacuate to the TSC with this procedure.

E. Perform Section 2, SM Follow-up Actions.

Responsibilites of the Shift Manager in the Alternate Shutdown procedure
Plausibility of C and D

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

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Ques
10 CF

SRO level because the questions asked are NOT entry conditions, overall sequence or mitigating strategy of the AOP. The examine will have to know decision points within the AOP. Also requires knowledge of the Emergency procedures

Question 98

Bank:	2428	**Rev:**	1	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	98	**Author:**	Burton
Lic Level:	SRO	**Difficulty:**	2	**Taxonomy:**	H	**Source:**	Modified QID-2285 NRC Bank		
Search	1940012120	**10CFR55:**	43.5	**Safety Function**					
Title:	Generic	**System Number**	GENERIC	**K/A**	2.1.20				
Tier:	3	**Group:**	1	**RO Imp:**	4.6	**SRO Imp:**	4.6	**L. Plan:**	A2LP-RO-EFRP
OBJ	2								
Description:	Conduct of Operations - Ability to interpret and execute procedure steps.								

Question:

Given the following:

* Unit 2 has tripped from 100% power.
* OP-2202.009, Functional Recovery procedure has been entered.
* The assessment of safety functions has been completed and is provided below.
  - Maintenance of Vital Auxiliaries (AC) is Challenged.
  - RCS Inventory Control is Challenged.
  - Containment Isolation is Jeopardized.
  - RCS Pressure Control is Jeopardized.
  - All other Safety Function acceptance criteria are "Satisfied".

Per step 13 of the FRP, which of the following is the correct order for success path implementation?

A. RCS Pressure Control, Cntmt Isolation, RCS Inventory Control, MVAC (AC)

B. RCS Pressure Control, Cntmt Isolation, MVAC (AC), RCS Inventory Control

C. CNMT Isolation, RCS Pressure Control, RCS Inventory Control, MVAC (AC)

D. CNMT Isolation, RCS Pressure Control, MVAC (AC), RCS Inventory Control

Answer:

B. RCS Pressure Control, Cntmt Isolation, MVAC (AC), RCS Inventory Control

Notes:

Per step 13 of the FRP, success paths should be initiated in the following order according to safety function hierarchy and
1) Jeopardized
2) Challenged
3) Satisfied

B. Is Correct - RCS Pressure Control, CNMT Isolation, MVAC (AC), RCS Pressure Control. Jeopardized then Challenged and in the correct hierarchy.

A. Is Incorrect but plausible because Jeopardized are first and in the correct hierarchy, Challenged are in the wrong hierarchy.

C. Is Incorrect but plausible because Jeopardized are first but in the wrong hierarchy, Challenged are also in the wrong hierarchy.

D. Is Incorrect but plausible because Jeopardized are first but in the wrong hierarchy, Challenged are in the correct hierarchy,

KA Match - Based on facility conditions the examinee must select success path to be performed and the correct order to be initiated.

References:

EOP-2202.009 (018), Functional Recovery Procedure and Tech Guide
(Reference verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Parent question is QID-2285 from the 2015 NRC exam.
This question has been modified with a different set of safety functions that are jeopardized, challenged or met therefore creating a new answer.
Rev 1 - removed the attachment and added the conditions to the stem. Made an additional Safety Function jeopardized and changed the question to a 2x2 format.

INSTRUCTIONS	CONTINGENCY ACTIONS
***13.** **Check ALL Safety Function acceptance criteria satisfied.** Hierarchy of the success paths	***13. Perform the following:** A. Determine appropriate success paths using Success Path Decision Trees. B. Initiate success paths for ALL Safety Functions in the following order: 1) Jeopardized. 2) Challenged. 3) Satisfied. C. IF higher priority Safety Function jeopardized AND lower priority safety function success path in progress, THEN **GO TO** appropriate success path for highest priority safety function in jeopardy. D. IF it is determined that a de-energized electrical bus is needed to satisfy a SAFETY FUNCTION, THEN restore power to affected bus using 2202.010 Attachment 11, Degraded Power E. WHEN success path implemented for EACH Safety Function, THEN **GO TO** Step 14.

FRP STEP:

*13. **Check ALL Safety Function acceptance criteria satisfied**.

EPG STEP:

*9. Perform operator instructions for each success path.

DEVIATION? NO

STEP BASIS:

Both the operators and Shift Technical Advisor are required to continually verify that all safety function acceptance criteria are being satisfied. This step provides guidance for the implementation of the success paths.

Contingency action 'A' references the Success Path Decision Trees. The Decision Trees are flowcharts which use plant conditions or equipment availability to guide the operator to the appropriate success path.

Contingency action 'B' provides the direction to initiate the success paths for jeopardized, challenged, and satisfied safety functions.

Contingency action 'C' provides guidance on proper usage of higher and lower priority success paths. This step addresses the possibility of a success path being in use and a higher priority safety function being jeopardized.

Contingency Action “D” directs the operator to an attachment (1) if an electrical bus is needed to satisfy a SAFETY FUNCTION.

Contingency Action “E” directs the operator to go to the next step once a Success Path has been implemented for each Safety Function.

SOURCE DOCUMENTS:

1. 2202.010, Standard Attachments, Attachment 11, Degraded Power.

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

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

Selecting the correct order to address the Safety Functions is selecting the procedure sections to mitigate the event and is an SRO function

Can the question be answered *solely* b
the purpose, overall sequence of even
overall mitigative strategy of a procedu

No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Data for 2015 NRC RO/SRO Exam

Bank: 2285 **Rev:** 1 **Rev Date:** 7/6/2015 **QID #:** 99 **Author:** Coble

Lic Level: S **Difficulty:** 2 **Taxonomy:** F **Source:** MODIFIED NRC BANK QID #1709

Search 1940012406 **10CFR55:** 41.10 / 43.5 / 45.13 **Safety Function**

System Title: Generic **System Number** GENERIC **K/A** 2.4.6

Tier: 3 **Group:** 1 **RO Imp:** 3.7 **SRO Imp:** 4.7 **L. Plan:** A2LP-RO-EFRP **OBJ** 4

Description: Emergency Procedures/Plan - Knowledge of EOP mitigation strategies.

Question:

Consider the following:

- During the use of the Functional Recovery Procedure, an assessment of safety functions has been completed
- Reactivity Control safety function has been diagnosed as "Challenged"
- RCS/Core Heat Removal safety function has been diagnosed as "Challenged"
- RCS Inventory Control safety function has been diagnosed as "Jeopardized"
- All other safety function acceptance criteria are "Satisfied"

In which order would the CRS implement the success paths for each safety function?

A. Reactivity Control; RCS/Core heat removal; RCS Inventory Control

B. RCS Inventory Control; RCS/Core heat removal; Reactivity Control

C. Reactivity Control; RCS Inventory Control; RCS/Core heat removal

D. RCS Inventory Control; Reactivity Control; RCS/Core heat removal

Ans

D. (

This is the parent question that was modified to create Q #98 on the 2017 SRO Exam. Modified question has a different set of safety functions that are jeopardized, challenged or met therefore creating a new answer

Notes:

D. Correct: Jeopardized safety functions are always addressed first by order of hierarchy. Challenged safety functions are always addressed after all actions for any jeopardized safety functions are completed unless a lesser safety function is needed t accomplish the higher jeopardized safety function. Satisfied safety functions are addressed after all jeopardized and challenge safety functions have been addressed.
A, B and C are incorrect but plausible as the applicant has to recall the information above and address the safety functions in order

References:

OP-1015.021, ANO-2 EOP/AOP User Guide, Rev 013, steps 4.40.4, 4.40.13, and 4.40.18.
OP-2202.009, Functional Recovery Procedure, Rev 017, entry section steps 13 and 14 pages 15 and 16 of 271and Success Pat Tracking Page page 271 of 271

Historical Comments:

QID #1709 used on 2009 NRC Exam
Incorporated comments form NRC preview mwf 5/26/15
Rev 1 - Incorporated NRC review comments mwf 7/6/15

Question 99

Bank:	2429	**Rev:**	2	**Rev Date:**	12/7/2016	**2017 TEST QID #:**	99	**Author:**	Burton		
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam				
Search	1940012444	**10CFR55:**	43.5			**Safety Function**					
Title:	Generic					**System Number**	GENERIC	**K/A**	2.4.44		
Tier:	3	**Group:**	1	**RO Imp:**	2.4	**SRO Imp:**	4.4	**L. Plan:**	ASLP-RO-EPLAN	**OBJ**	12

Description: Emergency Procedures/Plan - Knowledge of emergency plan protective action recommendations.

Question:

Protective Action Recommendations (PARs) are required to be provided during ____________ Emergencies and PARs must be reassessed within a MINIMUM of every _________________.

A. ONLY General; 15 minutes

B. ONLY General 30 minutes

C. General and Site Area; 15 minutes

D. General and Site Area; 30 minutes

Answer:

A. ONLY General ; 15 minutes

Notes:

A is correct. The Emergency Class Initial Notification Message Form 1903.011-Y states that PARS are required for General Emergencies but not for NUE, Alert, or SAE emergency classifications. 30 minutes is plausible because this is the maximum allowed time to complete notifications from the time conditions are available in the Control Room. (15 to evaluate + 15 to complete notifications)

B. Only General is correct. 30 minutes is incorrect but plausible as stated in "A".

C. Is incorrect as PARS are not required for SAEs but plausible as plant or localized evacuations could be required for Gen. Emergencies or SAEs based on the plant conditions and 15 minutes is correct.

D. Is incorrect as PARS are not required for SAEs and the time to reassess is 15 minutes, 30 minutes is plausible as stated in "A" above.

This question matches the K/A because the applicant must have knowledge of when to provide emergency plan PARs and how often PARs are re-assessed knowledge of Protective Action Recommendations".

References:

OP-1903.011 (052), Emergency Response/Notifications,
Initial Notification Message 1903.011-Y Form (043)
(References verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam
Rev 1 - added minimum to stem.
Rev 2 - capitalized MINIMUM

INITIAL NOTIFICATION MESSAGE

Use this form for **Emergency Class Declarations, Changes (Upgrade or Downgrade) or Protective Action Recommendations (PAR's).**

1. **MESSAGE NUMBER**: _______

2. **MESSAGE**:

This is ______________________________________ at Arkansas Nuclear One. My
(Communicator's name)

phone number is (479) 858-_______.

This is ☐ **AN ACTUAL EVENT** ☐ **A DRILL**

☐ **A NOTIFICATION OF UNUSUAL EVENT was DECLARED**
☐ **An ALERT was DECLARED**
☐ **A SITE AREA EMERGENCY was DECLARED**
☐ **A GENERAL EMERGENCY was DECLARED**

on ☐ **UNIT 1** ☐ **UNIT 2** on ___________ at ________ based on
(Date) (Time)

EAL No. __________ **Description:** ____

____

The **wind** is **AT** _____ **miles per hour** and **FROM** _______ **degrees**.
(Degrees must be between 0 & 360)

☐ There is **NO GASEOUS RADIOACTIVE RELEASE** taking place at this time due to this event.

☐ There is **A GASEOUS RADIOACTIVE RELEASE** due to this event, which
☐ **does** ☐ **does not** exceed federally approved operating limits. (RASCAL indicates NUE or higher emergency class)

Recommended Protective Actions are:
IF NUE, Alert or SAE
☐ **NONE AT THIS TIME**

IF General Emergency or as directed by person with ED&C
☐ **EVACUATE ZONES: G H I J K L M N O P Q R S T U**
☐ **SHELTER ZONES: G H I J K L M N O P Q R S T U**
☐ **Remainder of the EPZ to go indoors: G H I J K L M N O P Q R S T U**
☐ **Beyond 10 Mile EPZ.** ☐ **Evacuate** ☐ **Shelter sectors** __________________ **out to** _____ **miles.**

Comments: ____

____

More information will follow shortly.

3. **APPROVAL Print and Sign:**____
☐ **Shift Manager** ☐ **Emergency Director]**

QID-99
2 of 5

SAE

This form is intended to be used by the **EMERGENCY DIRECTOR** when a Site Area Emergency has been declared and the ED has the responsibility for Emergency Direction and Control. Steps may be performed by another individual if directed by the person with ED&C unless noted.

1.0 **IF** this is a security event AND entry conditions for Procedure 1203.048, “Security Event”, are met,
THEN ERO and offsite notifications are performed in accordance with Procedure 1203.048.

2.0 Site Area Emergency Declared: Unit_______ Time_______ Date_______

****EMERGENCY CLASSIFICATION / PLANT EVACUATION ANNOUNCEMENT SHOULD BE MADE WITHIN 15 MINUTES OF THE DECLARATION****

3.0 EAL No.______ Description: ____

____

Examine may believe that if a Site Evacuation is required then so is a PAR. Plausibility of C and D

4.0 Announce emergency class de

5.0 Direct the EOF Offsite Comm
Form 1903.011-Y, "Emergency Class Initial Notification Message".

6.0 **HAS a plant evacuation been performed?**

NO – Then go to step 7.0.

YES – Then perform the following:

6.1 **IF** the EPM is available,
THEN request the EPM perform Step 10 and Step 11 of Form 1903.011Q, “SAE Plant Evacuation / Personnel Accountability Actions EPM Checklist”.

5.1.1 Go to Step 6.3.

6.2 Request the Shift Manager to perform Step 16 and 17 of Form 1903.011P, “SAE Emergency Direction and Control Checklist Shift Manager”.

6.3 Dial 199 and make the following announcement:

"Attention all personnel. Attention all personnel. This is (state name and title). A Site Area Emergency has been declared on Unit _____ (One/Two) based upon (state EAL condition). Emergency response personnel report to your designated assembly areas."

ATTACHMENT 6
PROTECTIVE ACTION RECOMMENDATIONS (PARs)
FOR GENERAL EMERGENCY

Discussion

This attachment provides instructions for the assessment and initiation of Protective Action Recommendations (PARs) following the declaration of a General Emergency classification. Offsite response agencies shall be notified of Protective Action Recommendation within 15 minutes. Revisions to Protective Action Recommendations may be based upon:

- Current plant conditions
- Projected offsite dose assessment
- Forecasted/actual wind shifts

Evacuation is the preferred method for protecting the public within the ANO 10-mile Emergency Planning Zone (EPZ) as a result of a radiological emergency event at ANO. However, some circumstances may warrant a protective action of “shelter” when evacuation cannot be performed due to impediments and/or severe weather conditions. Individuals responsible for determining PARs at ANO should consider all circumstances when developing protective actions.

In the event of a “shelter” PAR, coordinate with ADH to develop a plan for transitioning out of this protective action as soon as possible. This is especially of concern during weather extremes since the public is advised to shut down ventilation systems.

The Arkansas Department of Health (ADH) will be notified of the ANO protective action recommendations and are responsible for determining and issuing a Protective Action Advisory (PAA) to the County Judges (Conway, Johnson, Logan, Pope and Yell counties). Arkansas law places the responsibility for issuing protective actions to the public with the County Judges which will have both a Protective Action Recommendation and a Protective Action Advisory available for decision making. At a General Emergency classification, the Arkansas Department of Health, at a minimum, will issue a default Protective Action Advisory of “evacuate a 5-mile radius and evacuate 5-10 miles downwind and the remaining EPZ to remain indoors and listen to emergency broadcasts”. At a General Emergency classification, ANO, at a minimum, will issue a default Protective Action Recommendation (PAR) of “evacuate a 2-mile radius and evacuate 2-5 miles downwind and the remaining EPZ to remain indoors and listen to emergency broadcasts”. The ADH Protective Action Advisory encompasses a larger area than that recommended by federal guidance and the ANO General Emergency classification PAR. Be aware of this difference between the ANO protective action recommendation and the ADH protective action advisory should a question arise. ANO PARs meet all of the EPA/NRC recommended regulatory guidance and are consistent with the rest of the nuclear industry.

Guidance Involving Wind Shifts within the 10-mile EPZ

If wind shifts are occurring or are predicted to occur within the 10-mile EPZ, guidance is provided on PAR No. 6 within this attachment.

Use of the PAR Flowchart in Attachment 6

A PAR Flowchart is included on Pages 3 and 4 of this attachment. This flowchart should be used initially starting on Page 3 and at the beginning of each subsequent PAR evaluation (page 4) to help determine the correct PAR to issue based on plant conditions, release status, evacuation impediments and offsite dose assessment.

ATTACHMENT 6
PROTECTIVE ACTION RECOMMENDATIONS (PARs)
FOR GENERAL EMERGENCY

PAR No. 1
EVACUATE

NOTE
State and local governments must be notified within **15 minutes** of PARs or changes to PARs using Form 1903.011-Y.

1. Entry Conditions

General Emergency Declared

2. Recommend the following Protective Action Recommendations:

Recommend **evacuation** of 2 mile radius and 2-5 miles downwind. Recommend the remainder of the 10 mile EPZ to go indoors and listen to the emergency broadcast for this event. Include any previously evacuated zones with this PAR. **DO NOT** change any previously evacuated zones to "shelter" or "go indoors" on this PAR.

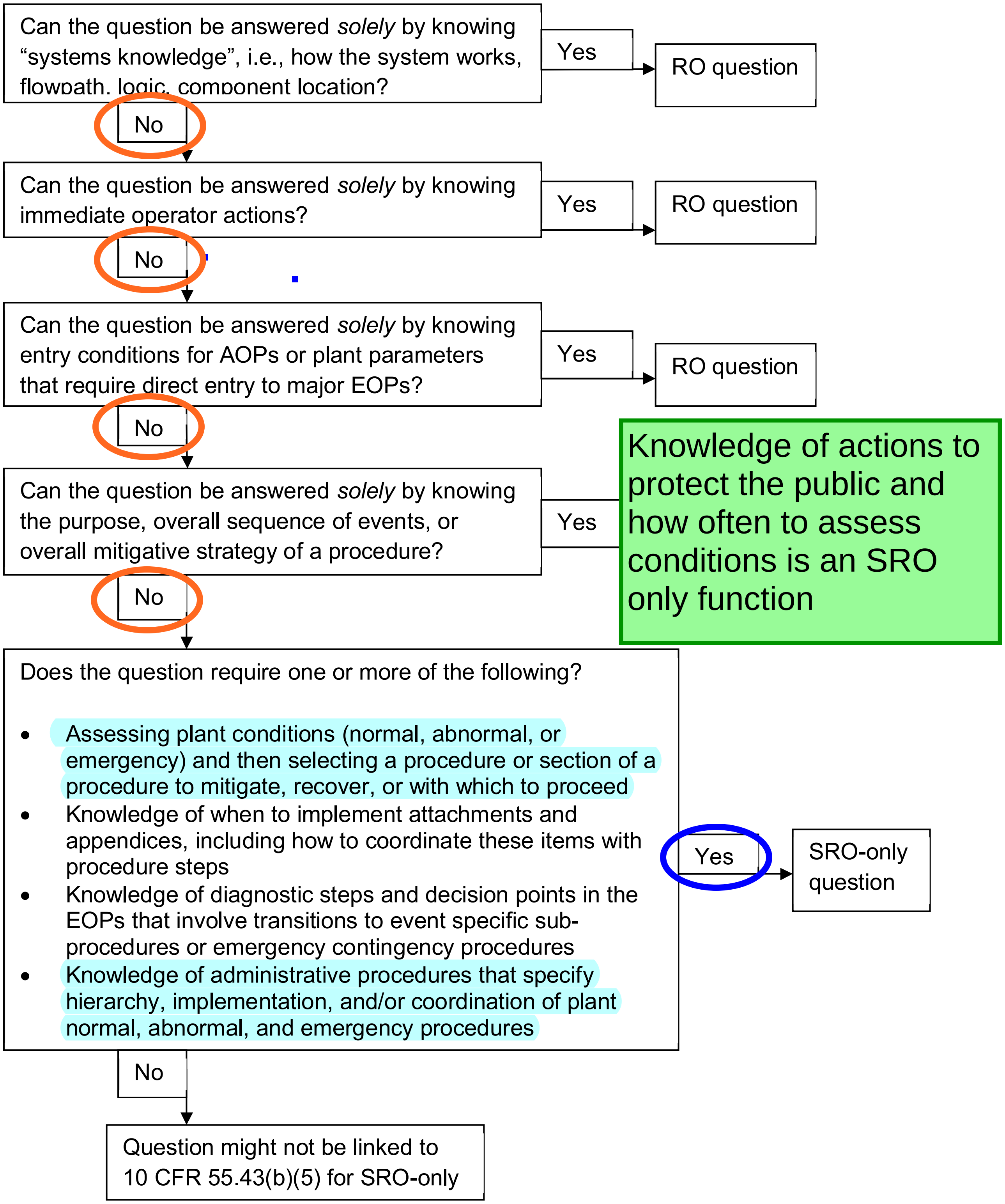
Determine the affected zones for the PAR from the chart given below.

Wind Direction (from)	Evacuate Zones	Zones "to go indoors"
348.75 to 11.25	G U	H I J K L M N O P Q R S T
11.25 to 33.75	G R U	H I J K L M N O P Q S T
33.75 to 56.25	G R U	H I J K L M N O P Q S T
56.25 to 78.75	G R U	H I J K L M N O P Q S T
78.75 to 101.25	G N O R	H I J K L M P Q S T U
101.25 to 123.75	G N O R	H I J K L M P Q S T U
123.75 to 146.25	G K N O	H I J L M P Q R S T U
146.25 to 168.75	G K N O	H I J L M P Q R S T U
168.75 to 191.25	G K N	H I J L M O P Q R S T U
191.25 to 213.75	G K	H I J L M N O P Q R S T U
213.75 to 236.25	G K	H I J L M N O P Q R S T U
236.25 to 258.75	G H K	I J L M N O P Q R S T U
258.75 to 281.25	G H K	I J L M N O P Q R S T U
281.25 to 303.75	G H K U	I J L M N O P Q R S T
303.75 to 326.25	G H U	I J K L M N O P Q R S T
326.25 to 348.75	G H U	I J K L M N O P Q R S T

3. Reassess PARs every **15 minutes** until downgrade or recovery phase is entered.

Every 15 minutes

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



Question 100

Bank:	2430	**Rev:**	0	**Rev Date:**	8/15/2016	**2017 TEST QID #:**	100	**Author:**	Burton
Lic Level:	SRO	**Difficulty:**	3	**Taxonomy:**	F	**Source:**	NEW FOR 2017 NRC Exam		
Search	1940012242	**10CFR55:**	43.2			**Safety Function**			
Title:	Generic					**System Number**	GENERIC	**K/A**	2.2.42
Tier:	3	**Group:**	1	**RO Imp:**	3.9	**SRO Imp:**	4.6	**L. Plan:**	A2LP-RO-TS

OBJ 2

Description: Equipment Control - Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Question:

Given the following:

* Unit 2 is operating at 100% power.
* It is discovered that a 31 day surveillance was not performed on a Tech Spec component.
* The surveillance was due 10 days ago.

Which of the following actions is/are required per Surveillance Requirement 4.0.3?

Component must be declared Inoperable _________

A. at time of discovery.

B. within a maximum 24 hours from time of discovery.

C. within a maximum of 31 days from time of discovery provided a risk evaluation has been performed and risk impact is managed.

D. within a maximum of 38.75 days from time of discovery provided a risk evaluation has been performed and risk impact is managed.

Answer:

C. within a maximum of 31 days from time of discovery provided a risk evaluation has been performed and risk impact is managed.

Notes:

C. is correct - Component remains Operable but must be declared Inoperable if the surveillance is not completed within 31 days from time of discovery provided a risk evaluation has been performed and the risk impact has been managed per SR 4.0.3.

A. Is incorrect but plausible because a missed surveillance could be inoperable if SR 4.0.3 is not used. 4.0.1 states failure to perform a SR within the specified interval shall be failure to meet the LCO.

B. Component must be declared Inoperable within a maximum 24 hours from time of discovery. If the surveillance were missed outside of the 25% then this looks like a good answer per SR 4.0.3, making this plausible.

D. Component must be declared Inoperable within a maximum 31 days from time of discovery provided a risk evaluation has been performed and the risk impact has been managed. SR 4.0.2 does not apply to a missed SR. 38.75 is the 25% extension of 31 days making this a plausible choice.

KA Match - Operability/Inoperability and SRs fall within application of Tech Specs

References:

Surveillance Requirements 4.01, 4.0.2 and 4.0.3
(Reference verified current 11/10/16)

Historical Comments:

To be used on the 2017 NRC Exam

QID-100
1 of 3

Distractor A

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the Surveillance. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified interval shall be failure to meet the LCO except as provided in 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

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

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

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

4.0.4 Entry into a MODE or other spe
when the LCO's Surveillances h
as provided in SR 4.0.3. When
met, entry into a MODE or othe
made in accordance with LCO 3

Allowed 24 hours or the limit of the specified interval. This make 'B' wrong

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONs or that are part of a shutdown of the unit.

SR 4.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 4.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Emergency Feedwater (EFW) pump turbine maintenance during refueling that requires testing at steam pressures > 700 psi. However, if other appropriate testing is satisfactorily completed, the EFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

b. High Pressure Safety Injec ystem
functional tests at a specifi ctorily
completed, startup can pro eration
to reach the specified pres .

There is nothing within the Bases that allows the use of the 25% extension for a missed SR

SR 4.0.2

SR 4.0.2 establishes the requirements for meeting the specified Frequency interval for Surveillances and any ACTION with an AOT that requires the periodic performance of the ACTION on a “once per…” interval. For those SR intervals established on a STAGGERED TEST BASIS, the 25% extension is applied to the stated frequency divided by the number of trains/channels associated with the system. For example, given an SR on a four channel system required to be performed TA (every 123 days) on a STAGGERED TEST BASIS, the 25% extension is applied on a per-channel basis (123/4 x 1.25 = 38 days, 10½ hours).

SR 4.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities). It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval.

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 4.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 4.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations.

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

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

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

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- Knowledge of TS bases that is required to analyze TS required actions and terminology

Yes → SRO-only question

Application of SR 4.0.1 thru 4.04 is an SRO function.

No

Question not be linked to 10 CFR 55.43(b)(2) for SRO-only might