Draft Submittal
(Pink Paper)

FARLEY RETAKE AUGUST 2004 EXAM

50-348 & 50-364/2004-301

AUGUST 24, 2004

DRAFT

Senior Reactor Operator Written Exam

1. 003K2.01 001

With Unit 2 at 25% power and 2A, 2B and 2C 4160V buses are powered from the Startup Transformers. 2B Startup Transformer trips due to Sudden Pressure. Which one of the following shows the correct Reactor Coolant Pump (RCP) status?

A. 2A RCP running, 2B and 2C RCPs tripped.

B. 2C RCP running, 2A and 2B RCPs tripped.

C. 2A and 2B RCPs running, 2C RCP tripped.

D✓ 2B and 2C RCPs running, 2A RCP tripped.

Unit 2 2B Startup Transformer supplies power to 4160 V bus 2A only
2A SU Xformer supplies power to the 2B and 2C 4160v bus.

Unit 1 1B Startup Transformer supplies power to 4160 V buses 1B and 1C

A, B and C. Incorrect - Each one of these lists a RCP that would be either tripped and is running or should be running and shows tripped.

D. Correct - **2B and 2C RCPs running, 2A tripped.**
This is the only correct combination.

1A S/U XFMR UV <2450V UV Relay (27) MB1

003 Reactor Coolant Pump Knowledge of Bus power supplies to the following:
K2.01 - RCPs

Identify the power supply for the following (OPS40301D04):
• Reactor coolant pumps (A, B and C)
Identify the power supply for each major electrical component associated with the Intermediate and Low Voltage AC Distribution System including (OPS40102B04):
• 4160V AC Buses

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

2. 004K4.15 001

While placing letdown on service, the OATC places the handswitch for HV-8149A, LTDN ORIF ISO, to the OPEN position. HV-8149A did not open. Which one of the following explains why HV-8149A did not open?

A. HV-8152, LTDN LINE CTMT ISO, is closed.

B. Pressurizer level on LT-459 is greater than 15%.

C✓ LCV-459 and 460, LTDN LINE ISO, are closed.

D. The LOCAL/REMOTE handswitch at the hot shutdown panel is in the REMOTE position.

A. Incorrect; 8152 being closed has no effect on the letdown orifice iso valves. valve.

B. Incorrect- Pzr level will keep the Letdown orifice isolation valves closed is < 15%

C. Correct; **LCV-459 and 460 , LTDN LINE ISO, are closed.**
these valves have to be open to allow 8149 A,B,C to open.

D. Incorrect; the statement is reversed, the LOCAL/REMOTE hs has to be in LOCAL for this to keep the valves from operating.

FROM OPS-52101F/40301F

Letdown Orifice Isolation Valves (8149A, B, and C)

Each orifice has its own air-operated isolation valve prior to rejoining the common letdown line. By opening different combinations of orifice isolation valves, the operator can control the amount of letdown flow.

To open 8149A, B, or C, all of the following conditions must be met:

1. LCV-459 and LCV-460 must be open.
2. Pressurizer level must be greater than 15%.

The orifice isolation valves **automatically close** if any of the following conditions exist:

1. **Pressurizer level decreases to less than 15%. (defeated if in "Local" at the HSP")**
2. **Either LCV-459 or LCV-460 closes. (defeated if in "Local" at the HSP")**
3. **A Phase A containment isolation signal ("T"-signal) occurs.**
4. **A loss-of-power or loss-of-air occurs.**

Location: CB 105' outside Biowall by letdown orifices.

004 Chemical and Volume Control Knowledge of the CVCS design feature and/or interlock(s) which provide for the following:
K4.15 – Interlocks associated with operation of orifice isolation valves.

Explain the purpose and operation including the design features and functions, capacities, and protective interlocks following major component associated with the Chemical and Volume Control System. (OPS40301F02)
Remotely Operated valves
- Letdown Orifice Isolation Valves (8149A, B, and C)

CVCS-40301F08 #2
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

3. 004K6.07 001

At normal operating conditions, the air supply line to the letdown line pressure control valve, PCV-145, was isolated to stop an air leak. FK-122 was immediately placed in manual. Which one of the following is the correct initial response of the listed parameter?

A. VCT level falls.

B. Pressurizer level rises.

C. TI - 143, LTDN DIVERT TEMP, falls.

D✓ TI -140, REGEN HX OUTLET TEMP, rises.

A. Incorrect - VCT level will rise due to increased LTDN flow.

B. Incorrect - PRZR level will fall with FCV-122 in Manual and LTDN flow increasing.

C. Incorrect - TI-143 is on the letdown side of the system just past the Ltdn HX, Ltdn flow to VCT prior to divert TCV. The TCV 3083 will open and cause cooling CCW to this area of the line. The case can be made that temp remains close to stable or a slight increase due to this valve opening, but definitely NOT falling.

D. CORRECT; **TI -140, REGEN HX OUTLET TEMP, rises.**
LTDN has increased while CHG flow has remained constant. The LTDN flow will not receive the same amount of cooling, so Regen hx outlet temp. will rise.

From OPS-52101F:
PCV-145 fails open on loss-of-power or loss-of-air. A manual bypass line is provided so that letdown can be maintained if maintenance is required on PCV-145.

004 Chemical and Volume Control - Knowledge of the effect of a loss or malfunction on the following CVCS components
K6.07 – Heat exchangers and condensers.

Describe the effect on the Chemical and Volume Control of a loss of an AC or DC bus or instrument air (OPS40301F06).
8. Predict and explain the following instrument/equipment response expected when performing Chemical and Volume Control System evolutions including the fail condition, alarms, trip setpoints. (OPS40301F08).
- TI-140 Regen Hx Outlet

CVCS-40301F07 #22
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

4. 005K5.05 001

Given the following plant conditions:

- The RCS is in solid plant condition.
- The residual heat removal (RHR) system is in service providing low pressure letdown.
- RCS pressure is 350 psig and stable.
- RCS temperature is stable.
- FCV-122 is in Manual control.
- PCV-145, LETDOWN PRESSURE Controller, is in AUTO.
- The OATC closes down on HCV-142, RHR TO LTDN HX, from 70% open to 50% open.

Which one of the following statements is correct?

A. Letdown pressure increases, PCV-145 automatically throttles SHUT to restore letdown pressure to its original value, and RCS pressure DECREASES.

B. Letdown pressure increases, PCV-145 automatically throttles OPEN to restore letdown pressure to its original value, and RCS pressure DECREASES.

C.✓ Letdown pressure decreases, PCV-145 automatically throttles SHUT to restore letdown pressure to its original value, and RCS pressure INCREASES.

D. Letdown pressure decreases, PCV-145 automatically throttles OPEN to restore letdown pressure to its original value, and RCS pressure INCREASES.

OPS 52101K
RCS pressure in the solid plant condition is controlled by the amount of charging and letdown. If charging and letdown are equal, RCS pressure is stable. More charging (or **less letdown) would increase pressure**; conversely, less charging (or more letdown) would reduce pressure. The operator controls letdown by adjusting HCV-142 and PCV-145 in the CVCS system. Charging is controlled by positioning FCV-122.

A B & D. Incorrect - Adjusting HCV 142 to the closed position restricts flow and will decrease ltdn pressure downstream. PCV 145 would throttle closed to try and raise ltdn pressure to the auto setpoint. Reduced letdown flow while maintaining charging at the same flow rate would cause an increase in RCS pressure

C. Correct -**Letdown pressure decreases, PCV-145 automatically throttles SHUT to restore letdown pressure to its original value, and RCS pressure INCREASES.**
per the above explanation

005 Residual Heat Removal Knowledge of the operational implications of the following concepts as they apply to the RHRS K5.05 – Plant response during "solid plant": pressure change due to the relative incompressibility of water.

Identify any special considerations such as safety hazards and plant condition changes that apply to the Residual Heat Removal System (OPS52101K04).
• Decrease in low pressure letdown flow
• HCV-142 fails closed
• HCV-142 fails open
List the automatic actions associated with the Residual Heat Removal System components and equipment during normal and abnormal operations including (OPS40301K07):
• Normal control methods
Describe the operation of PCV-145 during normal operations and solid plant pressure control including the conditions of air and electrical power failures (OPS52101F05).

RHR-40301K07 #21
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

5. 006K1.11 001

Concerning the Component Cooling Water (CCW) system, the following conditions exist:

- 1A CCW pump is tagged out to have the motor rebuilt.
- 1B and 1C CCW pumps are both in operation.
- B Train is the "On Service" train.

The 1C CCW pump has just tripped. Which one of the following is a list of loads that have **ALL** lost CCW flow due to the 1C CCW pump trip?

A. 1A Spent Fuel Pool Hx, RCP Thermal Barriers, 1B Charging Pump.

B.✓ 1A RHR Hx, 1B Spent Fuel Pool Hx, 1A Charging Pump.

C. 1A Spent Fuel Pool Hx, 1B Charging Pump, 1B RHR Pump Seal Cooler.

D. 1B Spent Fuel Pool Hx, 1A Charging Pump, Letdown Hx.

C CCW pump is the A train pump-
The B CCW pump is the onservice train and is carrying the misc. header.

A. Incorrect - Either supplied by the Misc. header and C CCW pump is not aligned there or supplied by the wrong train.

B. Correct - **1A RHR Hx, 1B Spent Fuel Pool Hx, 1A Charging Pump.**
All are A Train supplied by C CCW pump ESF loads

C. Incorrect - 1B RHR Pump Seal Cooler Supplied by B Train.

D. Incorrect - Supplied by Misc header. 1B Spent Fuel Pool Hx and 1A Charging Pump is correct.

OPS-52102G
The ESS loads consist of the following:
1. Charging pumps
2. Spent fuel pool heat exchangers
3. RHR heat exchangers
4. RHR pumps

The secondary heat exchanger loads consist of the following:
1. RCP oil coolers and thermal barrier heat exchangers
2. Reactor coolant drain tank (RCDT) heat exchanger
3. Excess letdown heat exchanger
4. Seal water heat exchanger
5. Letdown heat exchanger
9. Waste gas compressors
10. Sample system heat exchangers
11. Gross failed fuel detector

The CCW system provides cooling for the Train A and Train B emergency core cooling system components. The C CCW pump and the C heat exchanger are designated as Train A. The A CCW pump and the A CCW heat exchanger are designated as Train B. The B CCW pump and the B CCW heat exchanger can be aligned to either train.
CCW is normally lined up so that one CCW pump and one CCW heat exchanger is in operation supplying the on-service train. The on-service train is the one that supplies the secondary heat exchangers. The swing pump and heat exchanger (1B CCW pump and 1B HX) is normally aligned in standby to the on-service train with the heat exchanger outlet valve shut. The remaining pump and heat exchanger is valved into a closed loop with the redundant safety train. This train is idle and is designated as the off-service train. The off-service train CCW pump must be running before starting the off-service train charging pump or RHR pump.

006 Emergency Core Cooling Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following: K1.11 – CCWS.

Describe, when applicable, the Component Cooling Water System flow path to include all major components (OPS40204A05).
- Component Cooling Water Pumps
- Component Cooling Water Heat Exchangers including cooling medium

List all the engineered safety system (ESS) loads supplied by the component cooling water (CCW) system (OPS40204A13).
List all the secondary heat exchanger loads supplied by the component cooling water (CCW) system (OPS40204A14).
Describe what determines if train A or train B component cooling water (CCW) is the onservice train (OPS40204A15).

CCW-40204A05 #7

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	B B B B B B B B B B	Items Not Scrambled

6. 007EK2.03 001

Unit 1 is at 100% power.

Which one of the following groups of conditions will result in an automatic reactor trip, either directly or indirectly?

A.✓ RCP bus frequency (Hz): 57.1 (DA02), 56.9 (DB02), 56.8 (DC02).

B. Power range (%): 107 (N41), 108 (N42), 108 (N43), 109 (N44).

C. Pzr pressure (psig): 2376 (PT-455), 2381 (PT-456), 2386 (PT-457).

D. SG narrow range level (%): 81 (LT- 474), 82 (LT-475), 80 (LT-476).

A. Correct - **RCP bus frequency (Hz): 57.1 (DA02), 56.9 (DB02), 56.8 (DC02).** The SSPS signal is generated whenever an underfrequency condition (57 Hz, .25 second) exists as sensed on 1/2 of the RCP bus underfrequency (UF) detectors on 2/3 RCP buses. If reactor power is greater than P-7 (10% PWR), a reactor trip is also initiated by the SSPS. (TSLB2 2-1, 2-2, 2-3)

B. Incorrect; Power Range High Flux, High Setpoint NI-41,42,43,44 109% Rx Pwr 2/4 (TSLB2 11-1,11-2,11-3, 11-4)

C. Incorrect; Pressurizer High Pressure PI-455,456,457 2385 psig 2/3 (TSLB2 20-1,20-2,20-3)

D. Incorrect; If water level in a steam generator increases to 82 percent on 2/3 level instruments, a turbine trip signal and main feedwater isolation signal are initiated. At >35% power, 82% on 2/3 on 1/3 SG NR will trip the turbine which will trip the Rx. Tripping the turbine is a protective measure to ensure no damage occurs from moisture carry-over. Main feedwater is isolated so that no further water is added to the steam generator with the high-high level.

007 Reactor Trip - Stabilization - Recovery Knowledge of the interrelations between a reactor trip and the following: EK2.03 – Reactor trip status panel.

Evaluate plant conditions to determine if entry into EEP-0/ESP-0.0 is required. (OPS52530A02)

Braidwood 1 - 1998

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A A A A A A A A A A	Items Not Scrambled

7. 007K4.01 001

The PRT can be cooled two different ways after a PORV has lifted and increased the PRT level and temperature. The Systems Operator can use either the normal method or the alternate method. Which of the following lists both the normal method and the alternate method.

A. RCDT heat exchanger, Demin water fill and drain to the WHT.

B. Seal water heat exchanger, Demin water fill and drain to the WHT.

C✓ RCDT heat exchanger, Reactor make-up water fill and drain to the RHT's.

D. Seal water heat exchanger, Reactor make-up water fill and drain to the RHT's.

A. Incorrect- RCDT is correct, DW is not
B. Incorrect- SW Hx is not correct DW is not correct
C. Correct - **RCDT heat exchanger, Reactor make-up water fill and drain to the RHT's.** see below.
D. Incorrect - SW Hx is not correct

OPS-52101E
The PORVs and code safety relief valves do not relieve reactor coolant to the containment atmosphere. Instead, the relief valves discharge below the water level in the PRT. The PRT, which normally contains 70 percent water at ambient temperature, condenses and cools the discharge. The water in the PRT is heated by the discharge and can be cooled by the reactor coolant drain tank's (RCDT) heat exchanger and returned to the PRT through a spray header. An alternate method for cooling the PRT water involves spraying reactor makeup water into the PRT and draining the excess water to the waste disposal system. Refer to SOP-1.2 for actual procedure operations concerning cooldown of the PRT.

Then, the water in the PRT can be cooled by circulating through the RCDT heat exchanger via HV-8031. The water is circulated by the RCDT pump and returned through HV-7141 to the PRT. An internal spray header reintroduces the water into the PRT and condenses the remaining steam. The heat transfer capacity of the RCDT heat exchanger is sufficient to cool the contents of the PRT to 120°F within eight hours following a design relief actuation. The RCDT is returned to normal operation after completing the PRT cooldown. This is the preferred cooldown method since no liquid waste is produced as a result of the PRT cooldown.
As a backup method for cooling, cool reactor makeup water can be sprayed into the PRT via HV-8028 and 8030 and then drained to the recycle holdup tank via the RCDT pumps. Reactor makeup water is the source of water for initial fill and makeup to the PRT. This backup method will cool the PRT contents from an initial temperature of 200°F to 120°F in one hour.
The valves mentioned (HV-8028 and 8030) are controlled remotely from the MCB. HV-7141 is controlled from the liquid waste processing panel. All three valves fail closed on loss-of electrical power or -air.
The PRT is not designed to accept a continuous discharge from the Pressurizer. If the pressure in the tank exceeds 100 psig, two rupture discs relieve the PRT contents to the containment atmosphere. This creates the potential for increased containment pressure, temperature, humidity, and radiation levels. Depending on the PRT contents, the system which is relieving to the PRT, and the flow rate into the PRT, the adverse containment parameters can or will increase to an extent where significant problems with containment equipment and instrumentation could be encountered. The rupture discs have a relief capacity equal to the combined capacity of the Pressurizer code safety valves. The PRT design pressure and the rupture disc setpoints are twice the calculated pressure resulting from the maximum code safety valve discharge. This margin prevents deformation of the discs during releases less than design. The PRT and rupture disc holders also are designed for full vacuum to prevent tank collapse if the contents cool following a discharge.

007 Pressurizer Relief/Quench Tank Knowledge of the design feature(s) and/or interlock(s) which provide for the following: K4.01 – Quench tank cooling

Explain the purpose and operation including the design features and functions, capacities,
and protective interlocks of the following major component associated with the Pressurizer System (OPS40301E02):
• Pressurizer Relief Tank

Discuss the operation and alignment of each major component including precautions and
limitations of operation, and applicable procedures, associated with the Pressurizer System including (OPS40301E11):
• Pressurizer
• Pressurizer Relief Tank

PZR-40301E05 #1
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

8. 008A2.04 001

Unit 1 is at 100% power with A Train CCW on service. R-17B, CCW radiation monitor, has come into alarm. The following indications are received:

- LB3, RCP THRM BARR ISO HV-3184 AIR PRESS LO, annunciator has come in to alarm.
- HV-3184, CCW from RCP THRM BARR, closes.
- FH1, RMS HI-RAD, annunciator has come in to alarm.

Which one of the following is the proper action(s) of the Operating Team for this condition?

A. The CCW surge tank vent should be opened and closed once every eight hours and documented.

B. Declare R-17B INOPERABLE since the R-17B is from the off service train.

C. Establish excess letdown, remove normal letdown from service and swap the miscellaneous header to B Train.

D✓ Closely monitor seal injection flow, seal injection water temperatures and RCP lower bearing temperatures.

A. Incorrect - the actions of ARP FH1 have the operator do the following for a failed detector: the CCW surge tank vents should be cycled once every shift (eight hours) and documented in AutoLog.

B. Incorrect- R-17B is on A Train. Also R-17 is not in TS or the ODCM so there should be no question of operability.

C. Incorrect- The train is not swapped due to a rad monitor that comes in to alarm, especially with an indicated leak from the miscellaneous header. If the candidate did not know the RCP Thrm barr is on the Misc. Hdr, then swapping trains would be the appropriate action. This would cause more contamination of the system. R-17B is on A Train and this may be confusing to the student. The actions for letdown are from AOP-1.0 for a high R-17A/B alarm and is for leak detection.

D. Incorrect-**Closely monitor seal injection flow, seal injection water temperatures and RCP lower bearing temperatures.**
HV-3184 does not close on a R-17 alarm but does close on Hi press to protect against an RCS to thermal barrier leak. The actions contained are for a loss of CCW to the RCPs.

ARP 1.6 LOC. FH1
Step 2.17 IF **R-17A** OR **R-17B** alarms, THEN monitor CCW pump operation while the CCW surge tank vents are closed.
NOTE: IF CCW surge tank vents are closed for reasons other than an actual high radiation alarm, THEN with Shift Supervisor concurrence, the CCW surge tank vents should be cycled once every shift (eight hours) and documented in AutoLog.

ARP 1.6 LOC. FH1

The following actions will occur if a High Radiation Alarm is actuated on the associated Radiation Monitor.

c) R17A or B: (Component Cooling Water) closes Q1P17RCV3028 CCW SRG TANK VENT.

Lesson plan OPS 52102G/40204A
A high radiation signal on R-17A or B will deenergize their respective solenoid valves, allowing RCV-3028 to shut.
The surge tank normally vents to the auxiliary building nonradiation side through RCV-3028. In the event that R-17A or B senses a high radiation level, the vent valve closes automatically, minimizing the release of radioactivity to the auxiliary building. R17B is A train while R17A is B train related. Vacuum breakers are provided to prevent collapse of the tank or loss of system NPSH in the event of out-leakage during pump operation with the vent valve closed. The vacuum breakers lift at a setpoint of 1 psid or less to prevent maximum allowable vacuum of 1.69 psid.

ARP-3.1 LB3
IF VALVE 1-CCW-HV-3184 IS CLOSED DUE TO OTHER THAN OPERATOR ACTION, THEN ATTEMPT TO OPEN VALVE. IF VALVE WILL NOT OPEN, THEN CLOSELY MONITOR SEAL INJECTION WATER FLOWS, SEAL INJECTION WATER TEMPERATURES AND RCP LOWER BEARING TEMPERATURES.

008 Component Cooling Water Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use the procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.04 – PRMS alarm

Predict and explain the following instrument/equipment response expected when performing Component Cooling Water System evolutions including the fail condition, alarms, and trip setpoints as applicable (OPS40204A08).
• Radiation Monitors R-17A and B

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

9. 008G2.4.47 001

Given the following:

- Unit 1 has tripped from 100% power due to a loss of 4160V busses 1A, 1B and 1C.
- The operator initiated a Safety Injection due to rapidly falling pressurizer pressure.
- The following plant conditions were noted immediately after the Safety Injection:
  - Pressurizer level 35% and rising.
  - RCS pressure 1700 psig and falling.
  - Containment Pressure 0.2 psig and slowly rising.
  - R-2, 7, 11 and 12 are in alarm.

20 minutes later while in EEP-1, Loss of Reactor or Secondary Coolant , at the step to check RCS pressure on PI- 402A and 403A, the following conditions exist:

- Pressurizer level 95%.
- RCS pressure 2275 psig and rising.
- Containment Pressure 3.4 psig and rising.
- Ctmt sump level is increasing.

Which one of the following events is in progress?

A. RCS cold leg break.

B✓ Pressurizer steam space leak.

C. Pressurizer code safety leaking by.

D. Main Steamline break inside containment.

A. Incorrect - This is not correct due to the Pzr level increasing. A 3/8 inch RCS break is equivalent to 130 gpm and for a 1/2" break, RCS pressure would go no higher than 2000 psig in the first 70-80 min. Since RCS pressure dropped below 1700 psig which is indicative of a 2" break or larger; and since pressure rises to 2275 psig, it would not follow that an RCS cold leg break was the event.

B. Correct - **Pressurizer steam space leak.**
this type of leak has a trend indication of Pzr pressure increasing after the initial decrease, rad monitors in ctmt, Pzr pressure that drops at first and then rises as Pzr level rises to the top.

C. Incorrect - would not have early indications, immediately after the SI, of the ctmt pressure rising and rad monitors in alarm until the PRT ruptures.

D. Incorrect- rad monitor alarms inside ctmt indicate a primary problem.

008 Pzr vapor space accident.
G2.4.47 – Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Evaluate plant conditions and determine if transition to another section of EEP-1 or to another procedure is required. (OPS52530B08)

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

10. 008K3.03 001

Which one of the following is the limiting parameter for continued operation of the Reactor Coolant Pumps during a loss of Component Cooling Water?

A. RCP Motor Stator temperatures.

B.✓ RCP Motor Bearing temperatures.

C. RCP Lower Seal Water Outlet temperatures.

D. RCP Lower Seal Water Bearing temperatures.

A. Incorrect- Stator temps are affected by a Loss of SW, not CCW and by ctmt temp.

B Correct- **RCP Motor Bearing temperatures.**
Trip criteria is based on RCP Motor Bearing Oil Coolers, the bearing temperature will exceed 195°F in approximately 2 minutes.

C and D. Incorrect- FNP-1-ARP-1.4
6. CHECK RCP LOWER SEAL WATER BEARING AND SEAL WATER OUTLET TEMPERATURES -STABLE AND LESS THAN 225° F.

OPS-52101D
The oil coolers have been designed to maintain acceptable oil temperature with CCW inlet temperature as high as 105°F. Discharge from the oil coolers passes through a flow detector, FISL-3048A, B, and C, which gives MCB annunciation (CCW FLOW FROM RCP OIL CLRS LO-DD3), at a flow of 100 gpm. On a complete Loss of CCW Flow to RCP Motor Bearing Oil Coolers, the bearing temperatures will exceed 195°F in approximately 2 minutes. The flow then passes through two motor-operated isolation valves, one inside (MOV-3046) and one outside (MOV-3182) containment (130 foot elevation and 121 foot elevation respectively).

Reference:
ARP-1.4

SETPOINT: 100 + 10 GPM
- 0

ORIGIN:
1. Flow Switch (Q1P17FISL3048A-N)
2. Flow Switch (Q1P17FISL3048B-N)
3. Flow Switch (Q1P17FISL3048C-N)

D3
CCW FLOW
FROM RCP
OIL CLRS
LO

PROBABLE CAUSE

1. Loss of Component Cooling Water.
2. Loss of Component Cooling Water Flow to the RCP's due to Phase "B" isolation signal.
3. Improper valve lineup.

AUTOMATIC ACTION

NONE

IMMEDIATE ACTION

1. DETERMINE THE CAUSE OF THE ALARM.
2. IF A LOSS OF COMPONENT COOLING WATER HAS OCCURRED, THEN PERFORM THE ACTIONS REQUIRED BY FNP-1-AOP-9.0, LOSS OF COMPONENT COOLING WATER.
3. CLOSELY MONITOR THE RCP'S MOTOR BEARING TEMPERATURES.

NOTE: On a complete Loss of CCW Flow to RCP Motor Bearing Oil Coolers, the bearing temperature will exceed 195°F in approximately 2 minutes.

4. IF ANY RCP MOTOR BEARING TEMPERATURE EXCEEDS 195 F, THEN:
   a) TRIP THE REACTOR.
   b) STOP THE RCP.
   c) PERFORM THE ACTIONS REQUIRED BY FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.

AOP-9.0 Loss of Component Cooling Water show to secure the RCPs at 195°F if any RCP motor bearing temp exceeds that temp.

008 Component Cooling Water Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: K3.03 – RCP.

Evaluate plant conditions to determine if entry into AOP-9.0 is required. (OPS52520I02)
Explain the purpose and operation including the design features and functions, capacities,
and protective interlocks of the following components associated with the Reactor Coolant Pumps (OPS40301D02):
• Component Cooling Water

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

11. 009G2.4.4 001

Unit 1 is operating at 100% steady state power. The following conditions exist:

- Charging flow is 130 gpm in AUTO and EA2, CHG HDR FLO HI LO has come into alarm.
- VCT level is 25% and dropping.
- Two Letdown orifices are on service.
- Pressurizer level is decreasing and HB2, PRZR LVL DEV LO has come into alarm.
- Rad monitors R -11 and R -12 are trending upward.

Auto makeup starts. Pressurizer level is still falling. Which one of the following is the correct action to take next?

A. Attempt to identify the location of the leakage and isolate it.

B. Manually initiate a reactor trip, then a safety injection; Go to EEP-0, Reactor Trip or Safety Injection.

C✓ Enter AOP-1.0, RCS LEAKAGE, and maintain pressurizer level stable at the normal programmed level.

D. Perform a CVCS flow balance to determine the RCS leak rate and determine if a ramp-down to minimum load is required by Technical Specifications.

A. Incorrect - this may seem like a good idea, but the entry conditions for AOP-1.0 are met and that procedure would be entered.

B. Incorrect -AOP-1.0 will prioritize the actions to take in response to these symptoms. Since Pzr level is still falling, a rx trip and SI may seem to be the next appropriate action, however, since letdown is still on service and charging has not been increased, the team still does not have Rx trip criteria IAW AOP-1.0.

C. Correct - **Enter AOP-1.0, RCS LEAKAGE, and maintain pressurizer level stable at the normal programmed level.**
EA2, CHG HDR FLO HI LO, HB2, PRZR LVL DEV LO, unexplained rise in chg flow, reduction in VCT level and R-11/12 upscale all point to AOP-1.0 entry

D. Incorrect - This is done within the body of AOP-1.0, but should not be done prior to the entry to AOP-1.0 in this circumstance. This is done after stabilizing Pzr level to obtain a valid estimate of leak rate from the flow balance.

000009 Small Break LOCA / 3 G2.4.4 – Ability to recognize abnormal indications for system operating parameters which are entry- level conditions for emergency and abnormal operating procedures.

Evaluate plant conditions to determine if entry into AOP-1.0 is required. (OPS52520A02)

AOP-1.0-52520A05 #1
AOP-1.0-52520A04 #3

MCS	Time: 1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
				Answer:	C C C C C C C C C C	Items Not Scrambled

12. 010A1.03 001

Unit 1 is operating at 50% power. Given the following conditions:

- Pressurizer pressure is 2235 psig.
- Pressurized Relief Tank (PRT) pressure is 10.2 psig and rising.
- PRT temperature is 125°F and rising.
- PRT level is 81% and rising.
- One pressurizer PORV is blowing by its seat.

Assuming no operator action, which one of the following is correct?

A. The PRT level will increase beyond the maximum Technical Specification level.

B. The PRT hydrogen concentration will increase until an explosive mixture is created inside the PRT.

C✓ The PRT pressure will increase and then rapidly decrease, then containment sump level will rise.

D. The PRT temperature will decrease due to the cooling effect caused by the condensation of the safety valve leaking by.

A. Incorrect, There is no maximum Technical Specification level associated with the PRT even though it is a Tech Spec peice of equipment.

B. Incorrect, a nitrogen atmosphere exists in the PRT to prevent an explosive atmosphere.

C. Correct, **The PRT pressure will increase and then rapidly decrease, then containment sump level will rise.**
As water is added to the PRT and level rises, pressure will increase and if level and pressure gets high enough, the pressure could increase to 100 psig and cause the rupture disk to burst.

D. Incorrect, water will not be cold coming from the leaking valve and temperature will increase in the PRT. Liquid flashing to steam due to a pressure reduction has a cooling effect omn the surroundings. Steam condensing due to a heat sink has a heating effect on its heat sink (ie. the PRT).

The PRT is not designed to accept a continuous discharge from the Pressurizer. If the pressure in the tank exceeds 100 psig, two rupture discs relieve the PRT contents to the containment atmosphere. This creates the potential for increased containment pressure, temperature, humidity, and radiation levels. Depending on the PRT contents, the system which is relieving to the PRT, and the flow rate into the PRT, the adverse containment parameters can or will increase to an extent where significant problems with containment equipment and instrumentation could be encountered. The rupture discs have a relief capacity equal to the combined capacity of the Pressurizer code safety valves. The PRT design pressure and the rupture disc setpoints are twice the calculated pressure resulting from the maximum code safety valve discharge. This margin prevents deformation of the discs during releases less than design. The PRT and rupture disc holders also are designed for full vacuum to prevent tank collapse if the contents cool following a discharge.
The PRT is located on the 105 foot of containment inside the bioshield wall and normally contains 70 percent water with a 0.5 to 3 psig nitrogen atmosphere from the waste processing system nitrogen supply. The nitrogen blanket assures that no oxygen from the atmosphere leaks into the tank, therefore preventing the formation of an explosive hydrogen-oxygen mixture in the PRT. Hydrogen is present in the PRT after a discharge because hydrogen is used in the RCS to scavenge oxygen.

010 Pressurizer Pressure Control Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PZR PCS controls including: 010A1.03 PRT pressure and temperature

Explain the purpose and operation including the design features and functions, capacities, and protective interlocks of the following major component associated with the Pressurizer System (OPS40301E02):
- Pressurizer
- Pressurizer Relief Tank
- Code Safety Valves and PORVs

PZR-40301E02 #6

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

13. 010K5.02 001

Which one of the following represents the relief line temperature that would be indicated downstream of a leaking pressurizer PORV?
(Assume an ideal thermodynamic process, pressurizer pressure of 2235 psig, and PRT pressure of 20 psig.)

A. 162° F.

B. 228° F.

C.✓ 260° F.

D. 280° F.

A. Incorrect, 162 is sat temp is for 5 psia. 20 psig - 15 = 5 psia. A common mistake.

B. Incorrect, 228 is saturation temp for 20 psia. A mistake of not adding 15 psia.

C. Correct- **260°F.**
20 psig + 15 = 35 psia. For 35.427 psia / 260°F would be the appropriate temp. on table 1. Table 2 shows 30 psia = 250.34 and 40 psia = 267.25 psia.

D. Incorrect, 280 is a possible choice if the Mollier diagram is used and the lines are not followed exactly. A slight error on this diagram could result in this temp.

010 Pressurizer Pressure Control Knowledge of the operational implications of the following concepts as they apply to the PZR PCS: K5.02 – Constant enthalpy expansion through a valve.

Apply saturated and superheated steam tables in solving liquid-vapor problems (OPS30301C09).

Explain the purpose and operation including the design features and functions, capacities, and protective interlocks of the following major component associated with the Pressurizer System (OPS40301E02):
• Pressurizer
• Pressurizer Relief Tank
• Code Safety Valves and PORVs

PZR-52101E02 #3

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

14. 011A2.02 001

Given the following conditions:

- Unit 1 is at 50% power and ramping to 100% power.
- All automatic control systems are in their normal lineup.
- LS-459Z is selected to the I/II position.
- At 50% power, LT-459 fails as is.

Which one of the following describes the effect on the plant systems indicated and the operator actions required to mitigate the event?
(ASSUME no operator action is taken initially.)

A. Charging flow increases, and actual PZR level increases. HA2, PRZR LVL DEV HI B/U HTRS ON, will come into alarm. Stop the ramp, decrease charging flow in manual, and select LS-459Z to the I/III position.

B. Charging flow decreases, and actual PZR level decreases. HA3, PRZR LVL LO HTRS OFF LTDN SEC, will come into alarm. Stop the ramp, increase charging flow in manual, and select LS-459Z to the I/III position.

C✓ Charging flow increases, and actual PZR level increases. HB1, PRZR LVL HI, will come into alarm. Stop the ramp, decrease charging flow in manual, and select LS-459Z to the III/II position.

D. Charging flow decreases, and actual PZR level decreases. HB2, PRZR LVL DEV LO, will come into alarm. Stop the ramp, increase charging flow in manual, and select LS-459Z to the III/II position.

A. Incorrect - The controlling channel will stay at its present level and not change and the actual level will be increasing on the secondary channels. Since the setpoint is going up and level as read by the master controller remains the same, a -5% dev alarm will come in. HA2 will not come into alarm. The incorrect LS position is selected.

B. Incorrect - Charging flow will increase. This will not happen. If Pzr level did decrease, then this would happen at 15% level. The incorrect LS position is selected.

C. Correct - **Charging flow increases, and actual PZR level increases. HB1, PRZR LVL HI, will come into alarm. Stop the ramp, decrease charging flow in manual, and select LS-459Z to the III/II position.**
the setpoint will continually increase due to the Tavg input and the level will remain the same on the controlling channel. FCV-122 will see a need to raise chg flow to increase Pzr level. Since Pzr level as seen by the controlling channel remains constant, the signal will call for 122 to remain open until HB1 comes into alarm. The correct LS position is selected.

D. Incorrect - Charging flow will increase. The HB2 alarm will come into alarm but not for the reason given. The correct LS position is selected.

011 Pressurizer Level control Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS and (b) based on those predictions, use the procedures to correct, control, or mitigate the consequences of those malfunctions or operations:A2.02 - excessive charging

Predict and explain the following instrument/equipment response expected when performing Pressurizer Pressure and Level Control System evolutions including the fail condition, alarms, trip setpoints (OPS52201H08):
- LT-459
- LT-460
- LT-462

Describe the local actions needed to support plant operation during normal, abnormal and emergency conditions associated with the Pressurizer Pressure and Level Control System (OPS52201H09).

modified PZR PRS/LVL-52201H08 #51
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

15. 012A3.04 001

Unit 2 is at full power. Reactor Trip Breakers A (RTA) and B (RTB) are closed, Reactor Trip Bypass Breakers are open.

- 125V DC breaker 2B-16, "A" Reactor Trip switchgear control power to Bypass breaker and Reactor Trip breaker, has tripped open.

Which one of the following statements correctly describes how a loss of DC to the A Train reactor trip switchgear would effect the operation of Reactor Trip Breaker A from the control room?

A.✓ RTA would still open from either a manual or automatic signal.

B. RTA would immediately trip open because the shunt trip coil would de-energize.

C. RTA would not open in response to either an automatic or manual reactor trip signal.

D. RTA would not open in response to a manual reactor trip signal; an automatic trip would still open the breaker.

A. Correct- **RTA would still open from either a manual or automatic signal.** 125 V DC bus A allows the RTA shunt trip coil to energize on a Rx trip. With no DC available, the Rx trip Brkers will still open on a signal from SSPS A train by de-energizing the UV coil. With the loss of the DC, no RT brker will open immediately because SSPS is still energized and the breaker does not have a trip signal.

B. Incorrect - This will not cause a direct Rx trip due to SSPS still powered up and the only loss is DC to the STC Rx Trip.

C. Incorrect - It will open on both.

D. Incorrect - manual Rx trips cause the STC to be energized and the UV coil to be de-energized. A Rx trip will still operate on a loss of DC due to the UV coil. according to Table 6 of OPS-52103C, 125V DC distribution panel feeds to Rx trip swgr #1. According to the load list page F-51/52 LA-13 feeds 125V DC A Rx trip swgr control power to Byp brker & Rx trip bker. (2B-16)

012 Reactor Protection Ability to monitor automatic operation of the RPS, including: A3.04 – Circuit breaker.

Discuss the operation and alignment of each major component including precautions and limitations of operation, and applicable procedures, associated with the Reactor Protection System including (OPS52201I11):
• Reactor trip breakers (RTA and RTB)
• Bypass breakers (BYA and BYB)

RX PROT-52201I07 #28
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

16. 012K3.01 001

Given the following:

- Unit 1 is at 80% power.
- STP-33.0A, Solid State Protection System Train A Operability Test, is in progress.
- An operator calls the control room and reports the following:
  - Reactor Trip Breaker "A" is closed.
  - Reactor Trip Breaker "B" is open.
  - Bypass Breaker "B" is racked in and open.
  - Bypass Breaker "A" is racked in and closed.

Which one of the following describes the FINAL position of the control rods, the reason, and the required action?

A. No change in rod position as this is a normal configuration for this STP; continue with the STP in progress.

B.✓ All rods will be on the bottom due to loss of power to the CRDMs; enter EEP-0, Reactor Trip or Safety Injection.

C. All rods will be on the bottom due to a General Warning on both trains of SSPS; enter EEP-0, Reactor Trip or Safety Injection.

D. No change in rod position but all rods should be on the bottom due to an A Train Reactor Trip signal being generated but blocked in this configuration; manually trip the reactor and enter EEP-0, Reactor Trip or Safety Injection.

OPS-52201I
Two reactor trip breakers (RTA and RTB) are in series. The bypass breakers (BYA and BYB) are in parallel with reactor trip breakers RTA and RTB, respectively. Normally, the RTA and RTB reactor trip breakers are closed, and BYA and BYB bypass breakers are opened. Either RTA or RTB is capable of tripping the reactor.

A & D. Incorrect -since RTA and BYPA are closed and the other two are open, and these 2 are in parallel, there would be no power to the control rods and they will fall into the core.

B. Correct - **All rods will be on the bottom due to loss of power to the CRDMs, enter EEP-0, Reactor Trip or Safety Injection.**
no power due to configuration of the Rx trip breakers. Rods will fall into the core.

C. Incorrect - A general warning is generated from both RTB breakers being closed at the same time.

The 260V AC, 3 phase output from the MG sets is distributed to the four power cabinets through two, series-connected reactor trip breakers. Bypass breakers are connected in parallel with the reactor trip breakers to facilitate on-line testing of the protection system. These breakers are located in the rod control room in a cabinet adjacent to the MG set control cabinet. During a reactor trip, both the reactor trip breakers and the bypass breakers should open, removing the MG set output from the power cabinets and the DC hold cabinet. <u>De-energizing the power cabinets removes power from the CRDM coils, causing the control rods to drop</u>. (Refer to OPS-52201I for more information about the reactor trip and bypass breakers, reactor trip signals, and the Solid State Protection System.) In an emergency, these breakers may be required to be tripped locally.

An automatic reactor trip signal energizes the shunt coil and de-energizes the undervoltage coil within the two breakers (trip and bypass) associated with that protection train to trip the breakers. The reactor trip and bypass breakers can be manually operated from several locations. The reactor trip breakers are closed or tripped by a switch on the reactor controls section of the MCB. The CLOSE position allows the breakers to shut, while the TRIP position de-energizes the reactor trip and bypass breaker undervoltage (UV) coils and energizes each of the breaker's shunt (trip) coils.

De-energizing the undervoltage coil or energizing the breaker's shunt (trip) coil will cause the breakers to trip open. When the RTBs are closed, the UV coil is energized and the STR coil is de-energized.

012 Reactor Protection Knowledge of the effect that a loss or malfunction of the RPS will have on the following: K3.01 CRDS

Discuss the operation and alignment of each major component including precautions and limitations of operation, and applicable procedures, associated with the Reactor Protection System including (OPS52201I11):
- Reactor trip breakers (RTA and RTB)
- Bypass breakers (BYA and BYB)

Identify the power supply for each major electrical component associated with the Rod Control System including (OPS40204I04):
- Control Rod Drive Motor Generator Sets
- Logic Cabinet
- Power Cabinets
- DC Hold Cabinet

Discuss the operation and alignment of each major component including precautions and limitations of operation, and applicable procedures, associated with the Rod Control System including (OPS40204I11):
- Control Rod Drive Motor Generator Sets
- Reactor Trip & Bypass Breakers

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

17. 013K2.01 001

Given the following conditions on Unit 2:

- "A" train components are in service.
- 2A Charging Pump breaker was racked out for maintenance two hours ago.
- 2G 4160 volt bus has just been lost due to a fault.

Which one of the following states the ECCS pumps that are inoperable due to the fault?

A. 2B Charging Pump, 2A RHR Pump.

B. 2B Charging Pump, 2B RHR Pump.

C. 2C Charging Pump, 2A RHR Pump.

D✓ 2C Charging Pump, 2B RHR Pump.

A. Incorrect, both powered from 2F
B. Incorrect, 2B chg powered from 2F. 2B RHR powered from 2G
C. Incorrect, 2C chg powered from 2G. 2A RHR powered from 2F
D. Correct, **2C Charging Pump, 2B RHR Pump.**
both powered from 2G

013 Engineered Safety Features Actuation Knowledge of the bus power supplies of the following: K2.01 – ESFAS/safeguards equipment control.

Describe the effect on the Intermediate and Low Voltage AC Distribution of a loss of an AC or DC bus or instrument air (OPS40102B06).

LO/INT VOLT-40102B06 #3
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

18. 013K6.01 001

Given the following plant conditions:

- The operating crew has a loss of 1A 120V AC Vital Instrumentation Distribution Panel.
- PZR pressure transmitter PT-456, (Channel II), has just failed low.

Which ONE of the following describes the plant response?

A. Both trains of SSPS receive a safety injection signal AND both trains of ECCS equipment auto start.

B.✓ Both trains of SSPS receive a safety injection signal BUT only "B" train ECCS equipment auto start.

C. Only the "B" train SSPS receives a safety injection signal BUT both trains of ECCS equipment auto start.

D. Only the "B" train SSPS receives a safety injection signal AND only "B" train ECCS equipment auto start.

OPS-52201I
A. incorrect - Master relays would actuate since they are fed by the power supplies to the logic bay but slave relays for train A ECCS equipment cannot energize to actuate. Examinee may think the safeguards panel has redundant power supplies also.

B. correct - **Both trains of SSPS recieve a safety injection signal BUT only "B" train ECCS equipment auto start.**
Both trains of master relays will actuate since they are fed by the redundant power supplies of the logic bay, but A train ECCS will not start since only channel I supplies power to the safeguards panel to actuate A train slave relays.

C. incorrect - Examinee may be confused as to which relays have redundant power supplies. May think the master relays do not have redundant power supplies the slave relays do.

D. incorrect - Examinee may confuse the power supplies of master relays and think they are powered by the same power supply as the safeguards panel.

OPS-52103D Lesson plan pages 8,13,14,18,19, and figures 11,13, 12, 27
If an unsafe condition calls for safeguards actuation, the logic circuits will send a signal to the safeguards driver card. The card's output will increase from zero to 48V DC and will energize the required master relays for the specific safeguards actuation. The master relays energize their slave relays using 120V AC, which supply either AC or DC control power to ESF loads as appropriate.

The output relay cabinet contains master relays which are energized by the safeguards output cards. The master relay contacts apply power to a number of slave relays. One master relay contact can control one slave relay. The slave relay contacts, in turn, apply power to plant process equipment (e.g., pumps, solenoid valves, drive motors, power relay coils). The slave relays are coil and plunger type, AC-operated relays. The master relays are general purpose, DC-operated, four-pole relays. Two master relay locations are reserved for use as general warning relays. The relay test panel (located near the bottom of the cabinet) is used to check the master relay coil, contacts, and their associated slave relay coils by energizing the master relay coils which close contacts to apply a low DC voltage to the slave relay coil. (See Figures 7 and 27.)

013 Engineered Safety Features Actuation Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: K6.01 – Sensors and detectors.

Identify the power supply for the following cabinets associated with the Reactor Protection System (OPS52201I04):
- Input relay cabinets
- Logic cabinets
- Output relay cabinets

RX PROT-52201I04 #2
SQNP 2002

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

19. 014K5.01 001

Given the following information:

- Reactor Power is 100% RTP.
- All Control Bank D Rods are fully withdrawn - D230 Steps by Group Demand.
- Rod bank selector switch in manual.
- The in-hold-out switch is held in the "IN" position until the step counters count 5 steps IN.
- DRPI indication for Control Bank D does not change.

Which one of the following statements is true?

A. Rods definitely moved inward as indicated by the step counter change even though DRPI did not indicate rods moved.

B. Since rods did not move when 4 steps of rod movement was demanded, AOP-19, "Malfunction of Rod-Control System ," must be entered.

C✓ Rods probably moved inward as indicated by the step counter change. Rods will have to move in another step before DRPI indication will change.

D. Since DRPI indication did not change as expected when 4 steps of rod movement was demanded, operations should perform the control rod operability surveillance test.

<u>OPS-52201F, DIGITAL ROD POSITION INDICATION</u>
The display card converts the position signal to a conduction state of a unique LED. There are 40 LEDs, which indicate in six-step increments from rod bottom to 228 steps.

A. Incorrect. DRPI is the most reliable. Until DRPI changes, there is no assurance that rod motion is occurring due only to Group Demand.

B. Incorrect. AOP-19 entry conditions are for 12 steps misaligned.

C. Correct. **Rods probably moved inward as indicated by the step counter change. Rods will have to move in another step before DRPI indication will change.**
DRPI indicates in increments of 6 steps. If the 5 steps of rod motion was initiated from the rod position at which the DRPI indication changed during the last rod movement (the last rod movement was in the outward direction), it would take 6 steps back in the inward direction before DRPI changed again.

D. Incorrect. per above.

014 Rod Position Indication Knowledge of the operational implications of the following concepts as they apply to the RPIS: K5.01 – Reasons for differences between RPIS and step counter.

Discuss the operation and alignment of each major component including precautions and limitations of operation, and applicable procedures, associated with the Digital Rod Position Indication System including (OPS52201F11):
• Rod Position Detectors
• Data Cabinets
Explain the purpose and operation including the design features and functions, and protective interlocks of the following major components associated with the Digital Rod Position Indication System (OPS52201F02):
• Rod Position Detectors
• Rod Position Detectors

DRPI-52201F11 #8
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

20. 015/17G2.1.2 001

Unit 1 is at 20% reactor power and holding. During the transfer of 4160 buses 1A, 1B, and 1C from the Startup Transformer to the Unit Aux Transformer, Bus 1C does **not** transfer and is left de-energized.

What one of the following is the operators correct response to this event?

A. Re-energize the bus from the startup transformer; restart 1C RCP and continue power operation.

B. Do not exceed 30% reactor power; restore the inoperable loop to operable status within 72 hours, then continue with the ramp to 100% power.

C.✓ Proceed to hot standby within 6 hours; open the reactor trip breakers, re-energize the bus, and restart the RCP prior to Mode 2 entry.

D. Proceed to hot standby within one hour, then open the reactor trip breakers or shut down the rod drive MG sets within the next hour and initiate RCS cooldown per ESP-0.2, Natural Circulation Cooldown to Prevent Reactor Natural Circulation Cooldown to Prevent Vessel Head Steam Voiding.

AOP-4.0

B. Symptoms or Entry Conditions

This procedure is entered when forced RCS flow is lost in one or more loops and no reactor trip is required.

NOTE: Steps 7 and 8 must be completed within six hours of the loss of RCS flow
7 IF unit in Mode 1 or 2, THEN place unit in Mode 3 using FNP-1-UOP-3.1, POWER OPERATION and FNP-1-UOP-2.1, SHUTDOWN OF UNIT FROM MINIMUM LOAD TO HOT STANDBY.

8 WHEN unit in Mode 3 or 4, THEN verify all reactor trip and reactor trip bypass breakers open.

10 IF at least one RCP running, THEN go to FNP-1-UOP-2.1, SHUTDOWN OF UNIT FROM MINIMUM LOAD TO HOT STANDBY.

A and B. Incorrect - An operator would never restart a RCP at power and no procedure directs this.

C. Correct - **Proceed to hot standby within 6 hours; open the reactor trip breakers, re-energize the bus, and restart the RCP prior to Mode 2 entry.**
With the unit at 20% power and on hold, the operator is responsible to go to AOP-4.0 which will stabilize SGWL, have the unit ramped down to Mode 3 w/i 6 hours and RX trip bkrs opened. Then the unit is shutdown with UOP-2.1.

D - Incorrect - This is the wrong procedure to be in with RCP A & B running. No cooldown is required. the appropriate recovery strategy is S/D to mode 3, open the Rx trip breakers, repair the electrical problem, start the !c RCP and S/U.

000015/17 RCP Malfunctions / 4 G2.1.2 – Knowledge of operator responsibilities during all modes of plant operation.

Describe the sequence of major actions associated with AOP-4.0 (OPS52520D04).

AOP-4.0-52520D04 #7

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	C C C C C C C C C C	Items Not Scrambled

21. 015K3.01 001

Given the following:

- Unit 1 is operating at 30% reactor power.
- An I&C technician receives permission to perform a calibration on power range channel N-41.

The I&C technician pulls the instrument power fuses on N-42. Realizing a mistake was made, the technician reinserts the fuses for N-42 and then pulls the instrument power fuses for power range channel N-41. A reactor trip occurs.

Which one of the following is the cause of the reactor trip?

A. OPΔT trip

B.✓ PR rate trip

C. PR neutron flux high setpoint trip

D. PR neutron flux low setpoint trip

A. Incorrect - this B/S is not tripped by pulling the fuses.
The ARPs require the operator to pull the control power fuses for the failed channel. This is done in order to trip the associated bistables for the failed channel. This action gets all the PR reactor protection signals with the exception of the over-power .T and over-temperature .T bistables (reactor trip and rod stop).

B. Correct -**PR rate trip**
This B/S is locked in on the drawer until reset by the operator. It does come in on the pulled fuse and will not reset when the fuses are reinserted. When the other set of fuses are pulled, then a rate trip is received.

C. Incorrect - The Hi Flux trip B/S will come in whenthe fuse is pulled, but it will clear when the fuse is reinstalled. When the next fuse is pulled, the other B/S comes in but since the 1st cleared. nothing will happen.

D. Incorrect - This trip function is blocked >P-10.

Power Range (PR) Failure
Possible symptoms/indications of a PR channel failure include:
·PR UPPER DET HI FLUX DEV OR AUTO DEF annunciator (FB4)
·PR LOWER DET HI FLUX DEV OR AUTO DEF annunciator (FB5)
·PR HI FLUX - HI RNG RX TRIP ALERT annunciator (FC1)
·PR HI FLUX - LO RNG RX TRIP ALERT annunciator (FC2)
·PR HI FLUX RATE ALERT annunciator (FC3)
·PR CH DEV annunciator (FC5)
·PR OVERPOWER AUTO/MAN ROD STOP annunciator (FD2)
·Loss or erratic power range channel indication
·Improper overlap of IR and PR channels
·Power range trip status lights illuminated
·Rapid motion of control rods in automatic rod control (N-44 failure only)
Symptoms of a failed PR channel are likely to be multiple and obvious. The most difficult type of failure to detect is one that is failed as is; however, this failure is not likely to occur. The failed channel can be verified by performing a channel check, which compares all four PR channel readings. If the channel check reveals three readings that are similar and the fourth is drastically different, then further actions must be taken.

015 Nuclear Instrumentation Knowledge of the effect that a loss or malfunction of NIS will have on the following: K3.01 – RPS.

Predict and explain the following instrument/equipment response expected when performing Excore Nuclear Instrumentation System evolutions including the fail condition, alarms, and trip set points (OPS52201D08).
·Power Range Channels

EXCORE-52201D08 #10
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

22. 016G2.4.45 001

Unit 1 is at 100% power with all SG STM FLOW and FW FLOW selector switches in the channel IV position. PT-496, 1C SG PRESSURE transmitter, fails low. The following annunciators come into alarm by the time the Shift Supervisor makes it to the MCB:

- JA3, 1C SG LO LVL
- JA4, MS LINE PRESS LO ALERT
- JE3, 1C SG STM LINE HI DP ALERT
- JF3, 1C SG LVL DEV
- JG3, 1C SG FEED FLOW > STM FLOW

Which one of the following is the correct response to the above annunciators and the reason for that response?

A. Verify a reactor trip has initiated due to a Low Steam Generator Water Level.

B. Verify a reactor trip and safety injection has initiated due to a Low Main Steam Line Pressure.

C.✓ Take manual control and open 1C Feed Reg. Valve to prevent a Low Steam Generator Water Level reactor trip.

D. Take manual control and close 1C Feed Reg. Valve to prevent a High Steam Generator Water Level turbine trip.

A. Incorrect - JA1 comes in at 34% SGWL and the ARP has the operator take manual control of the FRV. A Rx trip has not occurred at this point. 28% is the Rx trip and that annunciator is not in alarm.

B. Incorrect - Since this is an ALERT annunciator, JA4 is telling the operator that 1 of 3 SG pressures is at 585 or less.

C. Correct - **Take manual control and open 1C Feed Reg. Valve to prevent a Low Steam Generator Water Level reactor trip.**
PT-496 fails low - indicated steam flow will fail in the same direction as the controlling S/G pressure transmitter because the pressure transmitter provides density compensation to the steam flow signal. Lower pressure translates into lower density of the steam which in turn means lower steam flow. The level control system will try to match lower steam flow with lower feed flow by closing the FRV.
This is also an input into MS line press low, stm line hi DP, and FF/SF annunciators. Since the FRV is affected and closes down, actual SGWL will drop until the FRV is taken to manual and opened or a Rx trip occurs.

D. Incorrect - the level is going low, not high. If the failure had been in the other direction, then the FRV would have gone full open and a HIGH SGWL would be the problem.

016 Non-nuclear Instrumentation G2.4.45 – Ability to prioritize and interpret the significance of each Annunciator or alarm.

Evaluate abnormal plant or equipment conditions associated with the Steam Generator Water Level Control System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52201B12).
Predict and explain the following instrument/equipment response expected when performing Steam Generator Water Level Control System evolutions including the fail condition, alarms, trip setpoints (OPS52201B08).
- Steam Flow
- Steam Pressure
- Steam Header Pressure
- Feed Pump Discharge Pressure
- Feedwater Flow
- Steam Generator Narrow Range Level

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

23. 017A1.01 001

Given the following temperatures from the CETC Monitor Display "HI" SUBMODE providing input to channel B subcooling monitor:

H11 = 600°F	H31 = 597°F
H12 = 603°F	H32 = 598°F
H21 = 602°F	H41 = 600°F
H22 = 605°F	H42 = 601°F

If T-sat for existing plant conditions is 649.64°F, which one of the following most closely represents the indicated subcooling from channel B subcooling monitor in the CETC mode?

A.✓ 45°F

B. 47°F

C. 49°F

D. 53°F

A. Correct - **45°F**
Highest Thermocouple is <u>H22 = 605</u>°F
T-sat = 649.64°F therefore: 649.6°F - 605°F = 44.64°F

B. Incorrect- 47°F would be correct if the average of the left column of Thermocouples were used to determine the subcooling
C. Incorrect- 49°F would be correct if the average of the allowed Thermocouples were used to determine the subcooling
D. Incorrect- 53°F would be correct if the lowest of the allowed Thermocouples were used to determine the subcooling

OPS52202E (in part)
The temperature margin and the pressure margin to saturation are both calculated for the three following areas:

1. RCS using resistance temperature detectors (RTDs) (in the RTD mode)
2. Highest thermocouple - core exit or upper head (in the individual valve display mode)
3. Highest CETC (in the CETC mode)

The inputs to the SMM are as follows per channel:

1. RTDs
   a. 3 T_{hot}
   b. 3 T_{cold}
2. Thermocouples (highest)
3. Pressurizer pressure (lowest compared to RCS pressure)
   a. Channel A - PT-455
   b. Channel B - PT-457
4. RCS pressure (lowest compared to pressurizer pressure)
   a. PT-402 - feeds each channel
   b. PT-403 - feeds each channel

The normal display mode for the SMM is the "CETC" mode. This displays the margin to saturation (°F) using the highest CETC (excluding the upper head) and the lowest pressure input.

017 In-core Temperature Monitor Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ITM system controls including: A1.01 – Core exit temperature

Discuss the operation and alignment of each major component including precautions and limitations of operation as well as applicable procedures associated with the Inadequate Core Cooling Monitor System including (OPS52202E11):
• HJTC sensors
• HJTC level displays
• Subcooled Margin Monitor (SMM)
• CETC Monitor
• ICCMS defeat panel
• Front Display Panel

ICCMS-52202E11 #15

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A A A A A A A A A A	Items Not Scrambled

24. 022A4.01 001

Given the following Plant Conditions:

- Unit 1 is operating at 100% power in the middle of summer.
- 1A, 1C, & 1D Containment Cooling fans are running in fast speed.
- 1B Containment Cooler has been turned OFF and SW has been isolated IAW BB1, CTMT CLR DRN LVL HI, for a suspected cooling coil leak.

BB3, CTMT AIR TEMP HI, has come into alarm and containment temperature has been verified to be 116°F. Which one of the following actions is allowed by procedure and will aid in keeping containment air temperature as low as possible until 1B Containment Cooler has been returned to service?

A. Reduce Service Water heat loads by securing unnecessary equipment associated with SW.

B. Reduce the heat load in containment by securing unnecessary equipment in containment.

C.✓ Verify all available Containment Coolers in fast speed with emergency SW supplied and verify all CTMT Dome Recirc Fans are in fast speed.

D. Shift all available Containment Coolers to slow speed with emergency SW supplied and verify all CTMT Dome Recirc Fans are in fast speed.

A. Incorrect. There is no direction for this and it would not help the situation anyway.

B. Incorrect. This might help if there was any unnecessary equipment in ctmt. however, no procedural guidance can be found to carry out this action.

C. Correct. **Place all available Containment Coolers in fast speed with emergency SW supplied and verify all CTMT Dome Recirc Fans are in fast speed.**
This the action recommended by ARP-1.2 BB3 and SOP-12.1, section 4.1

D. Incorrect. Slow speed fans will not remove as much heat and the Dome recirc fans are supposed to be in Fast speed IAW the ARP.

FNP-1-SOP-12.1 section 4.1 has the operator align emerg SW and the ARP sends the operator to SOP-12.1 and also to verify that the Dome fans are in fast

022 Containment Cooling
A4 Ability to manually operate and/or monitor in the control room: –A4.01 – CCS fans.

Evaluate abnormal plant or equipment conditions associated with the Containment Spray and Cooling System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS50102C02).

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

25. 022AA1.01 001

Given the following plant conditions:

- Unit 1 is operating at 100% power.
- Letdown Line isolation valve, LCV-460, has failed CLOSED.
- HA2, PRZR LVL DEV HI B/U HTRS ON, comes into alarm.
- PZR level is at 62% and slowly rising.

Which ONE of the following identifies the procedural actions that should be taken to address this transient if LCV-460 could not be re-opened?

A.✓ Close charging valve FCV-122; Place excess letdown in service.

B. Close charging valve FCV-122; Maintain seal flow between 8 - 13 gpm; Reduce power, as necessary, to maintain Pressurizer level less than 75%.

C. Close CHG PUMPS TO REGEN HX VALVES, MOV 8107 or 8108, to control charging flow; Place Back-up heaters OFF; Place excess letdown in service.

D. Close CHG PUMPS TO REGEN HX VALVES, MOV 8107 or 8108, to control charging flow; Reduce power, as necessary, to maintain Pressurizer level less than 75%.

AOP-16.0

A. Correct - **Close charging valve FCV-122; Place excess letdown in service.**
these are correct actions to be take in accordance with AOP-16 for a loss of CVCS letdown.

B. Incorrect - A ramp is not needed if excess letdown can be put on service.

C. Incorrect -These are the incorrect valves to control Pzr level and Applicant may also believe that the Backup heaters should be off, however with the level deviation (+5%) they should be on.

D. Incorrect - These are the incorrect valves to control Pzr level and a ramp is not needed.

000022 Loss of Rx Coolant Makeup /2 Ability to operate and/or monitor the following as they apply to the loss of RC pump makeup: AA1.01 – CVCS letdown and charging

Analyze plant indications to determine the successful completion of any step in AOP-16, Loss of Letdown (OPS52520K07).

SQNP 2002
AOP-16.0-52520K06 1

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

26. 025AK3.01 001

Unit 1 is in Mode 5, "COLD SHUTDOWN," with the following plant conditions:

- All CETC's read 195° F and are stable.
- All S/G's wide range levels are 94%.
- All RCP's are secured.
- RCS pressure is 325 psig and stable.
- Train 'A' RHR is in service.
- Train 'B' RHR is inoperable for repairs.
- All systems aligned in their normal configuration for the present plant conditions.

A loss of RHR has just occurred and cannot be restored. RCS temperature is rising.

Which ONE of the following is the preferred method for heat removal in accordance with FNP-1-AOP-12.0, "RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION?"

A. RWST gravity feed to RCS, spill through the Pressurizer PORVs.

B. Charging Pump injecting flow through the normal charging line, spill through the Pressurizer PORVs.

C✓ Natural Circulation RCS flow with the atmospheric relief valves open, AFW flow and SGBD established.

D. Reflux cooling to any S/G with level maintained in the Narrow Range.

A - Incorrect, This is an alternate method if CETC's >200 deg F and charging pumps are unavailable per AOP-12.0 but, it is not the preferred.

B - Incorrect, This is an alternate method if CETC's >200 deg F per AOP-12.0, step 26, but it is not the preferred.

C - Correct, **Natural Circulation RCS flow with the atmospheric relief valves open, AFW flow and SGBD established.**
This is the preferred method per AOP-12 with a loss of both RHR pumps.

D - Incorrect- this would only occur when primary coolant is not in the SG U-tubes.

025 Loss of RHR System / 4 Knowledge of the reasons for the following responses as they apply to the Loss of RHRS AK3.01 – Shift to alternate flowpath

Evaluate plant conditions to determine if any system components need to be operated while performing AOP-12.0. (OPS52520L06)
Evaluate plant conditions and determine if transition to another section of AOP-12.0 or to another procedure is required. (OPS52520L08)

Source: Byron 2000-301
2001 nrc exam
AOP-12.0-52520L08 #1
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

27. 026AA2.01 001

Given the following Unit 2 plant conditions:

- Operating at 100% power, for the last month.
- All systems are lined up for normal operation.
- While in AUTO rod control, control bank "D" starts stepping in slowly.

A tube leak in which ONE of the following places will cause this response?

A. the letdown heat exchanger.

B✓ the seal water heat exchanger.

C. a RCP thermal barrier heat exchanger.

D. the on-service CCW heat exchanger.

OPS-52520I AOP-9.0 lesson plan
If the surge tank level is falling due to leakage, the leaking train can be identified by personnel being dispatched or when surge tank level drops below 30 inches, the height of the divider
Potential sources of outleakage include:
- Spent fuel pool heat exchanger
- Charging pump oil coolers
- RHR heat exchanger (if normally closed, MOV-3185 A or B are open)
- RCP oil coolers
- Excess letdown heat exchanger (if excess letdown secured)

If the CCW surge tank level is increasing, then the source of the inleakage could potentially
be from one of the following depending on plant conditions:
- Letdown heat exchanger (if letdown on service)
- RCP thermal barriers
- RHR heat exchanger (if on service)
- Reactor makeup system (if normally closed, valves leaking by)
- Demineralized water system (if normally closed, valves leaking by)
- SW system (if SW discharge pressure higher than CCW discharge pressure)
- RCDT heat exchanger (if at least one RCDT pump running)
- Primary and secondary sample coolers (if sampling in progress)
- GFFD sampling assembly

A. Incorrect -CCW cools the letdown heat exchange but a leak would not result in control rods stepping in. The Ltdn Hx is at a higher pressure than CCW and the leak would be from the Ltdn to CCW.

B. Correct - **the seal water heat exchanger.**
CCW cools the seal water heat exchanger and a leak would result in slowly diluting the RCS causing the control rods to move in. The seal water Hx inlet pressure runs approx. 35-40 psig, based on PI-134 reading and VCT pressure. CCW pressure is approx 85-100 psig.

C. Incorrect - CCW cools the RCP thermal barrier heat exchanger but a leak would not result in control rods stepping in. The thermal barrier Hx is at a higher pressure than CCW and the leak would from the RCS to CCW.

D. Incorrect - CCW is cooled by service water but a leak in the heat exchanger will not cause inward rod motion. No inter connection between SW and RCS.

026 Loss of CCW Ability to determine and interpret the following as they apply to the Loss of CCW: AA2.01 – Location of a leak in the CCWS.

Evaluate abnormal plant or equipment conditions associated with the Component Cooling Water System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52102G02).
Identify any special considerations such as safety hazards and plant condition changes that apply to the Component Cooling Water System (OPS52102G04).
List all the secondary heat exchanger loads supplied by the component cooling water (CCW) system (OPS40204A14).

CCW-52102G02 #14

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

28. 026K4.01 001

The Containment Spray system is required and is operating due to a Large Break LOCA on Unit 1. RWST level has decreased to 4.5 feet.
Which one of the following is correct concerning CTMT SUMP TO CS PUMP, MOVs 8826 and 8827 A / B?

A. Automatically open on a lo-lo level in the RWST if a CS actuation signal is present.

B. Automatically open on a lo-lo level in the RWST if an SI signal is present.

C. Can be manually opened only after the CS actuation signal is reset.

D✓ Can be manually opened at any time from the MCB.

A and B. Incorrect - CS valves do not roll open automatically. This is only for RHR sump suction valves.

C. Incorrect -CS actuation signal does not have to be reset to open these valves manually. CS 8820A/B MOVs have to have the CS actuation signal reset prior to opening these valves.

D. Correct - **Can be manually opened at any time from the MCB.**
These valves can be opened at any time.

OPS-52102C
The valves, normally in a closed position, remain closed during the injection phase of the spray period when the spray pumps take a suction from the RWST. When the water level in the RWST is down to the low-low level, the operator opens the sump isolation valves and closes MOV-8817A and B, at which time the spray recirculation phase commences, and the spray pumps take a suction from the containment sump. Control switches for these valves are provided on the main control board.
OPS-52101K
Containment Sump to **RHR System** Isolation Valves (8811A and B; 8812A and B)
The containment sump to RHR system isolation valves are motor-operated valves that are controlled from the MCB. They align the RHR system suction to the containment sump for longterm cooling of the RCS after a LOCA. The valves will automatically open on a lo-lo level (50,000 gal) in the RWST if a safety injection signal is present.

026 Containment Spray Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: K4.01 – Source of water for CSS, including recirculation phase after LOCA.

Explain the purpose and operation including the design features and functions, capacities, and protective interlocks of the following major components associated with the Containment Spray and Cooling System (OPS40302D02):
• Containment Spray Pump Sump Suction Isolation Valves (MOV-8826A and B,MOV-8827A and B)

MCS Time: 3 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

29. 027AA2.15 001

Given the following:

- Unit 2 is at 75% power.
- Tavg is 565° F.
- PI - 444 PRZR PRESS reads greater than 2500 psig.
- PI - 445 PRZR PRESS reads 2180 psig and decreasing.
- HC1, PRZR PRESS HI-LO, is in alarm.

Which one of the following actions should be taken next?

A. Close PORV-445A and its associated block valve.

B. Turn on all heaters and commence a rapid power reduction.

C✓ Take manual control of the pressurizer spray valves, heaters and PORV's.

D. Trip the reactor and then trip 2A and 2B RCP's and implement EEP-0, REACTOR TRIP OR SAFETY INJECTION.

HC1 ARP
*A- Incorrect - PORV – 445A will not be open for a PT-444 failure. It will be closed due to PT-445 being at 2180 psig. If the PORV were to be stuck open, this would be a correct action.

*B- Incorrect – This is not the appropriate action taken for a loss of pressure. A rapid power reduction would cause an insurge and raise pressure but more controlled methods are used.

*C- Correct – **Take manual control of the pressurizer spray valves, heaters and PORV's.**
correct per HC1. PI-444 reads off-scale high and must be recognized as failed. RCS pressure could not get to >2500 psig with all its PORVs and safeties open and operating normally. It is sending a signal to the Htrs, sprays and 444B to lower pressure in AUTO. Taking manual control is the appropriate response.

*D- Incorrect – This is done if it is a mechanically stuck open spray valve prior to reaching 2000 psig.

Channel PT-444 provides pressure input for the master pressure controller PK-444A which controls the following components:
1. PORV PCV-444B control
2. Backup heater control
3. Variable heater control
4. Spray valve PCV-444C & 444D control
5. Controller output high alarm

444 failing High will give the following: PORV PCV-444B opens, Backup heaters turn off, Variable heaters turn off, Spray valves open, Plant pressure decreases rapidly, PORV PCV-444B closes at P-11 setpoint, Plant pressure continues to decrease due to spray valves still open and a Plant trip and SI should occur on low RCS pressure.

If a channel failure causes a PORV to open, closing the PORV block valve (MOV-8000A/B) for the corresponding PORV will isolate the flow path to the pressurizer relief tank (PRT) and RCS pressure should stop decreasing. If the failed channel causes spray valves to open, manually closing the spray valves will terminate the pressure decrease. Since pressurizer heaters turning on or off will affect plant pressure more slowly than the opening of PORVs or spray valves, your priority should be toward controlling the PORV and spray valve effect on RCS pressure.

000027 Pressurizer Pressure Control System Malfunction / 3 AA2.15 – Actions to be taken in PZR pressure instrument fails high.

Describe the local actions needed to support plant operation during normal, abnormal and emergency conditions associated with the Pressurizer Pressure and Level Control System (OPS52201H09).
Predict and explain the following instrument/equipment response expected when performing Pressurizer Pressure and Level Control System evolutions including the fail condition, alarms, trip setpoints (OPS52201H08):
- PT-444
- PT-445

PZR PRS/LVL-52201H09 #1
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

30. 028K2.01 001

Unit 1 has just lost power to 600V Motor Control Center 1B. Which one of the following components will not have power?

A.✓ 1B Post LOCA Hydrogen Recombiner.

B. 1B Containment Cooler Fan - High speed.

C. HHSI TO RCS CL Isolation Valve, MOV-8803B.

D. 1B Accumulator Discharge Isolation Valve, MOV-8808B.

A. Correct - **1B Post LOCA Hydrogen Recombiner.**
OPS-52102D
B. The recombiners receive power from separate vital electrical power trains.
Recombiners A and B are powered from 600V MCC A and B,
respectively.

B. Incorrect - LCC B is the power supply.

C. Incorrect - MCC V; valve is outside CTMT.

D. Incorrect - MCC V; valve is inside CTMT.

028 Hydrogen Recombiner and Purge Control Knowledge of bus power supplies to the following: K2.01 – Hydrogen recombiners.

Identify the power supply for each major electrical component associated with the Post LOCA Atmospheric Control System including (OPS40302E04):
• Hydrogen Recombiners

modified POST LOCA-40302E04 #1
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

31. 029EK2.06 001

FRP-S.1, Response To Nuclear Power Generation/ ATWT, has the operator open the reactor trip and bypass breakers locally. Opening the reactor trip and bypass breakers causes which one of the following to occur?

A.✓ A Main Turbine trip, arms the Steam Dumps, and places the plant trip controller into control of the Steam Dumps.

B. The rods to drop into the core, automatically blocks the Safety Injection signal, and closes the FRV and Bypass valves.

C. The rods to drop into the core, seals in the feedwater isolation signal with a lo-lo Tavg signal in, and trips the SGFPs.

D. Resets the high Steam flow setpoint to 40 percent, automatically starts the AFW pumps, and closes the FRV and Bypass valves.

A. Correct-**A Main Turbine trip, arms the Steam Dumps, and places the plant trip controller into control of the Steam Dumps.**
This is a few of the functions of P-4 permissive.

B. Incorrect- P-4 does not automatically block an SI signal but allows the operator to block the SI signal after a time delay, does not auto close the FRV and Bypass valves unless temp is < 554°F, Lo Tavg.

C. Incorrect - does not trip the SGFPs and does not technically seal in the FW isolation signal on a LO tavg, but definitely does not seal in on a Lo-lo Tavg.

D. Incorrect - Does not auto start the AFW pumps. does not always close the FRV and bypass valves, only in conjunction with a LO Tavg.

OPS--52201I
P-4 Permissive
The P-4 permissive is generated when both the reactor trip breaker and the bypass breaker, which physically bypasses it, are open. Train A of the reactor protection system uses RTA and BYA, and train B uses RTB and BYB. The following are functions of P-4:
1. Causes a turbine trip
2. Closes main feedwater regulating valves and feedwater bypass valves if low Tavg (554°F) is also present
3. Seals in feedwater isolation signal from safety injection or steam generator high-high water level
4. Resets high steam flow setpoint to 40 percent
5. Arms steam dumps on a plant trip, defeats the output of the load rejection controller, and places the plant trip controller into control.
6. Allows operator block of the safety injection signal after a time delay
This last feature ensures that the reactor is tripped and that all emergency core coolant system (ECCS) loads are started before the operator overrides what could be a spurious actuation signal. The block does not prevent the operator from reinitiating safety injection through use of either manual safety injection actuation switch.

000029 ATWS / 1 Knowledge of the interrelations between an ATWS and the following:
EK2.06 – Breakers, relays, and disconnects

This question relates to the above KA due to being a function of S.1 action step 6 and the functions that P-4 does for the operator derived from a signal due to the tripping of that breaker, and functions that will help mitigate the ATWT.

Describe the operation and function of the following reactor trip signals, permissives, control interlocks, and engineered safeguards actuation signals associated with the Reactor Protection System and Engineered Safeguards Features to include setpoint, coincidence, rate functions (if any), reset features, and the potential consequences improper conditions (OPS52201I07):
• All reactor trip signals
• All permissive signals (P-4, P-6, P-7, P-8, P-9, P-10, P-11, P-12, P-13, and P-14)

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

32. 033A3.02 001

A refueling accident has occurred in the Unit I spent fuel pool, resulting in a large release of contaminants from the damaged assembly. The following plant conditions also exist:

- R-2, 7, and 27A & B, CTMT RAD MONITORS, indicate pre-event readings.
- R-5, SFP RM is alarming.
- R-10, PRF DISCH, is trending up but not in alarm.
- R-14 and 21, VENT STACK GAS, is trending up but not in alarm.
- R-24A & B, CTMT PURGE, indicate pre-event readings.
- R-25A & B, SFP VENT, is alarming but slowly trending down.

Upon evaluation of the above conditions, you can determine that:

A.✓ The spent fuel pool area release is being filtered by the PRF system and monitored by R-10.

B. The spent fuel pool area release is being filtered by the SFP exhaust filter discharged to the Main exhaust plenum.

C. A second event resulting in release of radioactivity has occurred in the penetration rooms as evidenced by R-10 trending up.

D. Penetration room atmosphere has been affected by the accident in the spent fuel pool as evidenced by R-10, R-14 and R-21 trending up.

D175045, ARP-1.6 LOC FH5

A. Correct, **The spent fuel pool area release is being filtered by the PRF system and monitored by R-10.**
R-10 monitors PRF discharge. PRF normally takes a suction a suction on the SFP and discharges to the vent stack. Vent stack monitors R-14 and 21 are also trending up.

B. Incorrect, R-25A & B alarming trips the SFP exhaust fan and closes the SFP dampers.C. Incorrect, PRF takes a suction on SFP in this type of accident and NOT on the penetration rooms in this case.

D. Incorrect, PRF takes a suction on SFP in this type of accident and NOT on the penetration rooms in this case.

Fuel Handling Accident
During normal operation, the penetration room filtration system is aligned to automatically process the exhaust air from the spent fuel pool upon receiving an actuation signal initiated by either high radiation or low flow in the spent fuel pool exhaust system. Either of these signals will cause the spent fuel pool system to automatically isolate and the penetration room filtration system to energize.
The recirculation and exhaust fan exhaust dampers (HV-3356A and B, HV-3357A and B) open automatically upon energization of their respective fan to allow filtered air flow.
The train exhaust ducts join to form a common, monitored discharge duct, which routes the air to the vent stack. Radiation monitor (RE-10) alarms locally to inform the operator of a possible filter unit malfunction. **It should be noted that during this type of accident, the penetration room suction dampers (MOV-3362A and B) and recirculation dampers (MOV-3361A and B) remain closed.**

033 Spent Fuel Pool Cooling Ability to monitor automatic operation of the SFP cooling system including: A3.02 – Spent fuel leak or rupture

Predict and explain the following instrument/equipment response expected when performing Auxiliary Building Ventilation System evolutions including the fail condition, alarms, and trip setpoints (OPS40304B08).
- RE-14
- RE-21
- RE-22
- RE-25A and B

Evaluate abnormal plant or equipment conditions associated with the Auxiliary Building Ventilation System and determine the local actions needed to mitigate the consequence of the abnormality (OPS40304B12):

AUX BLDG VT-52107B02 #2
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

33. 034K1.01 001

During a refueling outage on Unit 1, the refueling team is removing a fuel assembly from the core. The manipulator crane operator receives a red OVERLOAD light.

This OVERLOAD cutoff limit is required by the Technical Requirements Manual to cut out on a load greater than or equal to __________ , and protects which one of the following components from excessive lifting force in the event they are inadvertently engaged during lifting operations?

A.✓ 2850 lbs.; Core internals and pressure vessel.

B. 2000 lbs.; Core internals and pressure vessel.

C. 2850 lbs.; Baffle plates and fuel assembly grid straps.

D. 2000 lbs.; Baffle plates and fuel assembly grid straps.

A. Correct - **2850 lbs.; Core internals and pressure vessel.**
This is IAW the TRM and TRM bases.

B. Incorrect - incorrect weight

C. Incorrect - incorrect components.

D. Correct - incorrect weight and components.

Technical Requirement 13.9.3, Manipulator Crane
The OPERABILITY requirements for the manipulator cranes ensure the following. The manipulator cranes will be used for movement of control rods and fuel assemblies. Each crane has sufficient load capacity to lift a control rod or fuel assembly. **The core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.**

FHP-5.13 page 6 1.1.16 was also used.

FHP-3.0 Sec 4.11 Approximate weight in pounds of fuel assemblies and inserts.

Fuel Assembly (empty)	- 1467
RCC	- 149
20 Rod BPRA	- 44
16 Rod BPRA	-39
12 Rod BPRA	- 34
Thimble Plugging Device	- 14
New Fuel Handling Tool	75

1467+39+75 = 1581 $\pm$ 200 = **1381 to 1781#**

034 Fuel Handling Equipment Knowledge of the physical connections and/or cause effect relationships between the fuel handling system and the following systems:
K1.01 – RCS
This was selected because the TRM bases shows that the Manipulator crane OPERABILITY requirements protect the core internals and the Rx vessel which is part of the RCS. We could not tie the RCS to the fuel handling system any other way.

Explain the purpose and operation including the design features and functions, capacities, and protective interlocks of the following major components associated with the Fuel Storage, Handling and Refueling System (OPS40305B02):
• Manipulator Crane

Discuss the operation and alignment of each major component including precautions and limitations of operation, and applicable procedures, associated with the Fuel Storage, Handling and Refueling System including (OPS40305B11):
• Manipulator Crane
• Hoist Control
• Manipulator Crane Checkout

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

34. 037AA1.08 001

Unit 1 is at 100% power with the following plant conditions:

• All Pressurizer heaters are energized.
• Letdown flow is 75 gpm.
• Charging flow is 105 gpm in Auto.
• S/G levels are constant.
• Tavg/Tref are matched.
• Pressurizer level is 48% and decreasing.
• VCT level is 22% and decreasing.
• R-70A, SG TUBE LEAK DET, is trending up.
• R-15, SJAE EXH, is trending up.

Which one of the following actions would be correct for the above plant conditions?

A. Reduce charging flow to match letdown flow.

B.✓ Close the letdown orifice isolation valve and raise charging flow.

C. Commence a manual makeup to the VCT and reduce charging flow.

D. Raise charging flow and align the charging pump suction to the RWST.

Maintain pressurizer level stable at normal programmed value by one or both of the following:
· Control charging flow.
· Reduce letdown
Maintain VCT level greater than 20%.
Verify reactor makeup system - IN AUTOMATIC.
OR
2.2 Manually control reactor makeup system using FNP-1-SOP-2.3, CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM.

A - Incorrect, This action would cause Pzr level to continue to decrease at a faster rate and is not IAW AOP-2.0.

B - Correct, **Close the letdown orifice isolation valve and raise charging flow.** This would allow the operator to stabilize Pzr level at the normal programmed level or determine if trip/SI criteria were met.

C - Incorrect, While a manual makeup might be something good to do, it is not IAW the first step of AOP-2.0. This action would cause Pzr level to continue to decrease at a faster rate and is not IAW AOP-2.0.

D - Incorrect, This is the actions of AOP-1.0 if charging cannot keep up with the VCT level decrease. In AOP-2.0, if the VCT level cannot be maintained > 20%, then the Rx is tripped/SI. Roll over to the RWST is not an option in AOP-2.

<u>OPS-52520B</u>
PRZR Level Maintenance
The team is directed to maintain PRZR level stable at the programmed value. Actions specified include manual adjustment of FCV-122, reduction of letdown flow, and actions to maintain adequate suction flow to the charging pumps. Maintaining adequate suction flow includes monitoring of the reactor makeup system and verification that makeup flow is sufficient to maintain VCT level within the program band. If the leak rate is greater than the capability of the reactor makeup system (120 gpm), then the operator is directed to trip the reactor and manually safety inject.

000037 Steam Generator Tube Leak / 3 Ability to operate and/or monitor the following as they apply to the SGT leak: AA1.08 – Charging flow indicator.

State the basis for and apply all cautions, notes, and actions associated with AOP-2.0 (OPS52520B03)

MCS	Time:	1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
					Answer:	B B B B B B B B B B	Items Not Scrambled

35. 039K5.05 001

Brittle fracture of the reactor vessel (RV) is most likely to occur during a ____________ of the reactor coolant system (RCS) when RCS temperature is ____________ the RV reference temperature for nil-ductility transition (RTNDT).

A. heatup; above.

B. heatup; below.

C. cooldown; above.

D✓ cooldown; below.

A and B. Incorrect -Heatup is not as limiting as cooldown
C. Incorrect - cooldown is correct. The RCS temp is most limiting BELOW the RTndt

D. Correct - **cooldown; below.**
The most limiting factors are cooldown and below the RTndt.

3.4.3 bases, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS
All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.
The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).
Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.
The LCO establishes *operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB).* **The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel.** The limits do not apply to the pressurizer, which has different design characteristics and operating functions.
10 CFR 50, Appendix G (Ref. 2) requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G (Ref. 3).
Enlarging the nil ductility reference temperature (RTNDT) as exposure to neutron fluence advances reflects the neutron embrittlement effect on the material toughness.
The actual shift in the RTNDT of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a non isolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause non ductile failure of the RCPB, an unanalyzed condition. Reference 1 the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

PWR thermodynamics chapter 10

High temperatures are characterized by ductile fracture; low temperatures by brittle fracture. As the temperature of a metal is reduced, the transition from ductile fracture to brittle fracture occurs in a narrow temperature range.

The temperature below which the metal fails by brittle fracture is called the nil-ductility transition temperature (NDTT). Another name often used is the reference temperature for nil-ductility transition (RTNDT). This is more commonly referred to as simply the reference temperature (RTNDT). Some metals, notably carbon steel, have an RTNDT that is of practical concern in the design and operation of nuclear power plants.

039 Main and Reheat Steam Knowledge of the operational implications of the following concepts as they apply to the MRSS: K5.05 – Bases for RCS cooldown limits

Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Reactor Coolant System (OPS52101A01).

- TS 3.4.2 RCS Minimum Temperature for Criticality
- TS 3.4.3 RCS Pressure and Temperature (P/T) Limits

Define and describe the following terms: (OPS30901J08)

a. Nil-ductility transition temperature (NDTT)

b. Reference temperature for nil-ductility transition (RTNDT)

BRITT FRAC-30901J06 #6

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9

Answer: D D D D D D D D D D Items Not Scrambled

36. 041K4.17 001

A plant startup has just been completed. The steam dump system is in the T_{avg} mode, and the turbine load is being raised from 200 MW to 400 MW. At approximately 40% power, the plant experiences a reactor trip/turbine trip due to a complete loss of vacuum.

Which one of the following describes how the Steam Dump system would react to the given situation?

A. Steam dumps would arm from C-7 and actuate from the loss-of-load controller.

B. Steam dumps get an arming signal from P-4 and actuate from the plant trip controller.

C. Steam dumps would arm from C-7 but would not actuate from the loss-of-load controller.

D✓ Steam dumps get an arming signal from P-4 but would not actuate from the plant trip controller.

A. Incorrect - Due to the reactor trip, P-4 would arm the Stm dumps. The plant trip controller would be the correct controller.

B. Incorrect - Due to the loss of vacuum, C-9 will NOT be available and the dumps will be blocked from operating.

C. Incorrect - Due to the reactor trip, P-4 would arm the Stm dumps. The plant trip controller would be the correct controller. Due to the loss of vacuum, C-9 will NOT be available and the dumps will be blocked from operating.

D. Correct - **Steam dumps get an arming signal from P-4 but would not actuate from the plant trip controller.**
They will arm but not actuate.

Arming Signals
There are three arming signals (Figure 13) associated with the blocking solenoid valves.
They are:
1. Turbine Loss-of-Load (C-7)
2. Plant Trip (P-4)
3. STM PRESS mode of operation
During normal plant operation in the TAVG mode, all blocking solenoid valves are deenergized, which prevents all steam dump valve operation. When the magnitude of a turbine load rejection exceeds 15 percent as indicated by the rate of change in turbine impulse chamber pressure from PT-447, the C-7 arm signal energizes the steam dump valve blocking solenoid valves. The source of this signal (PT-447) is different from the pressure signal (PT-446) used to generate Tref for the loss-of-load controller and the automatic rod control circuit. This signal separation is important because it removes the possibility that low failure of a single impulse pressure transmitter would cause unwanted steam dump actuation. Failures of PT-446 have previously been discussed and, as was pointed out, resulted only in a change in valve demand signal. PT-447 failures result only in the generation of an arming signal with no demand signal being generated. PT-447 failing low may result in arming of the dumps but would not result in actual steam dump operation unless TAVG was high. IF C-7 is received, operations management expectation is that C-7 will not be reset. Failure of PT-447 high would result in the inability to generate the loss-of-load arming signal. The loss-of-load setpoint is 15% with a 120 second time constant. When a loss-of-load is sensed then the loss-of-load bistable trips and energizes the steam dump valve blocking solenoid valves. This energization allows the loss-of-load controller to position the eight condenser dump valves.

There is one control interlock signal (Figure 13) associated with the steam dump valve blocking solenoid valves. This control interlock signal and one arming signal and no blocking signals are required to actuate the blocking solenoid valves. Any different combination of signals blocks the solenoid valves and prevents steam dump valve operation.

041 Steam Dump/Turbine Bypass Control Knowledge of the SDS design feature (s) and/or interlock(s) which provide for the following: K4.17 Reactor Trip

List the automatic actions associated with the Steam Dump System components and equipment during normal and abnormal operations including (OPS52201G07):
- Steam dump valves
- Steam dump system solenoid-operated three-way valves
- High-1 and High-2 trip bistables
- Plant trip controller
- Loss of load controller, C-7
- Condenser available, C-9
- Low-Low TAVG signal, P-12

STM DUMP-52201G07 #15
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

37. 054AK1.02 001

Which one of the following is the reason for feeding S/Gs < 100 gpm with AFW when SG water level is less than 12% wide range?

A. Prevents excessive cooldown rates to the Reactor coolant system.

B✓ Prevents excessive thermal stress to Steam Generator components.

C. Ensures adequate feedwater inventory is available for complete restoration of normal cooling.

D. Reduces the potential for water hammer when restoring a Reactor Coolant Pump to service.

REF: OPS-52533F Lesson Plan, page 14 -16
A. Incorrect - the cooldown rate is not a concern at this time. The concern is getting and maintaining a wet, moist atmosphere in the SG and to slowly cool the Hot, Dry SG to fill it up and to prevent the secondary shell to cone transition joint and on the feedwater nozzle from failing.

B. Correct- **Prevents excessive thermal stress in steam generators.**
as shown below

C. Incorrect - CST level and feedwater inventory is not a concern in this event.

D. Incorrect- Starting a RCP and waterhammer is not a concern.

FRP-H.5
CAUTION : *To minimize thermal stress,* AFW flow to any affected SG with a wide range level less than 12%{31%} should be limited to less than 100 gpm.

FRP-H.1
NOTE: IF it is necessary to feed a hot, dry SG(s) [RCS hot leg temperature > 550 F AND SG wide range level < 12%{31%}], THEN it (they) should be fed one at a time at a flow rate of 20 gpm to 100 gpm until RCS hot leg temperature falls to less than 550 F. IF bleed and feed is imminent OR bleed and feed is in progress and RCS temperatures are rising, THEN there is no limit on the feed flow rate.

FRP Background Document
BASIS:
If feed flow to a SG is isolated and the SG is allowed to dry out, subsequent reinitiation of feed flow to the SG could create significant thermal stress conditions on SG components.
Maintaining a minimum verifiable feed flow (i.e., (S.04) gpm) to the SG allows the components to remain in a "wet" condition, thereby minimizing any thermal shock effects if feed flow is increased.
In addition, if SG wide range indication shows that an appreciable water mass exists (i.e., greater than (N.01)% [(N.02)% for adverse containment], in the SG, then reestablishment of AFW should not cause significant thermal stresses. However, if SG wide range level is less than (N.01)% [(N.02)% for adverse containment], then the SG may have dried out. For this condition, AFW flow should be established very slowly (at a rate not to exceed 100 gpm) to any SG in this condition. This should be maintained until wide range water level indication rises above (N.01)% [(N.02)% for adverse containment].

Because the heat removal capability of one steam generator is always greater than decay heat, it is advisable to reestablish feedwater to only one steam generator regardless of the size of the plant or number of loops. Prior to establishing feed to a hot dry steam generator, it is recommended that the capability to isolate the steam generator be verified prior to initiating feed flow, especially if core overheating has occurred. Thus, if a failure in a steam generator occurs due to excessive thermal stresses, the failure is confined to one steam generator that can be isolated to minimize the consequences resulting from any releases that occur.

The initiation of cold feedwater flow to a hot, dry steam generator is not anticipated to result in the failure of the primary system pressure boundary. An evaluation of the impact of cold water addition on the steam generator tubes, initially at 800°F, has been performed. The results of the evaluation indicate that substantial margin to failure of the steam generator **tubes** exists due to the ductile properties of Inconel.
A similar evaluation was performed for the steam generator **tube sheet**. In this evaluation, it was assumed that the tube sheet was initially at 800°F, and the feedwater at the feed nozzle was 50°F. In the perforated portion of the tube sheet, the presence of the holes acts as insurance against brittle fracture. Substantial crack growth cannot occur across the tube sheet because of the holes. If a crack should develop, it can only grow until it intersects a hole. Therefore, it was concluded that fracture of the tube sheet under these conditions is not a concern.
The worst condition that may be reasonably postulated to occur as a result of feeding a hot, dry steam generator is a fault of the steam generator at the shell to cone transition joint or at the feedwater nozzle.
In the first case, the steam generator would depressurize to the containment at a point that is above the top of the uppermost steam generator tube. Thus, the steam generator could still function in the heat removal mode in this case.
For a fault in the steam generator feedwater nozzle (which would also result in a depressurization of the steam generator to the containment), the failure could result in the inability to supply additional feedwater to the steam generator, thus negating the recovery of feedwater for decay heat removal via that steam generator.

APE054 (CE/E06) Loss of Main Feedwater Knowledge of the operational implications of the following concepts as they apply to Loss of MFW: AK1.02 – Effects of feedwater introduction on dry S/G.

State the basis for all cautions, notes, and actions associated with FRP-H.1/2/3/4/5 (OPS52533F03).

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

38. 054AK3.04 001

If the **ONLY** running SGFP trips at 40% power, which one of the following statements is correct?

A. Reactor power must be reduced to < 35% quickly and the main turbine tripped to minimize the RCS temperature increase.

B. Reactor power must be reduced to < 35% quickly and the main turbine tripped to reduce steam demand.

C. The reactor should be manually tripped, then the main turbine is tripped in order to minimize the RCS cooldown.

D✓ The reactor should be manually tripped to conserve S/G inventory for adequate secondary heat sink and decay heat removal.

OPS-52520M and AOP-13.0
A. Incorrect - this is the reason and action for 1 SGFP running and 1 tripped at >60% power. Reactor power must be reduced quickly to minimize the RCS temperature increase. The temperature increase will occur because reactor power will be greater than turbine power as the turbine load is reduced. In this case there is no SGFP running.
B. Incorrect - This is correct in that a Turbine trip at 40% power would cause a Rx trip but this is not the correct action per our procedures. Check Reactor Power - LESS THAN 35% is a step that is an immediate operator action step and is reached only if there are no SGFPs available. If reactor power is less than 35% (P-9), then the turbine can be tripped without getting a reactor trip. The turbine is tripped to **reduce steam demand** in this case.

C. Incorrect - This is not correct for the same reason as above. The turbine will automatically trip at this power when the Rx is tripped. Tripping the turbine would minimize the RCS cooldown only if Rx power was less than turbine power.

D. Correct - **The reactor should be manually tripped to conserve S/G inventory for adequate secondary heat sink and decay heat removal.**
The Rx is tripped first which should trip the Mn turbine. This is done b/c the SGWL will go to 28% rapidly with the Mn turbine on line and the Rx at power.
If SG < 28% go to EEP-0.
If steam generator level cannot be maintained above 28% (narrow range), then the reactor should be tripped **to maximize the steam generator water inventory available for removing decay heat from the reactor.** A reactor trip should occur if steam generator levels are allowed to reach 28%. The reactor is tripped first (prior to tripping the turbine) to avoid getting into the worst case situation for a loss of feedwater.

000054 (CE/E06) Loss of Main Feedwater / 4 Knowledge of the reasons for the following responses as they apply to the Loss of MFW: AK3.04 – Actions contained in EOPs for loss of MFW.

State the basis for and apply all cautions, notes, and actions associated with AOP-13.0 (OPS52520M03)

modified from AOP-13.0-52520M03 #3
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

39. 055EA2.01 002

Given the following plant conditions:

- A station blackout has occurred.
- ECP-0.0, Loss of All AC Power, has been implemented.
- No Emergency power is available.

Which ONE of the following describes the position to which the Pressurizer PORVs, Main Steam atmospheric reliefs, the TDAFW pump shut off valves (HV-3235A and B) and the TDAFW pump steam admission valve (HV-3226) fail upon loss of instrument air?

A. PORVs- closed, atmospheric reliefs-closed, TDAFW steamline shutoff valves-closed in 2 hours, TDAFW pump steam admission valve - closed in 2 hours.

B.✓ PORVs-closed, atmospheric reliefs-closed, TDAFW steamline shutoff valves-closed in 2 hours, TDAFW pump steam admission valve - open.

C. PORVs-open, atmospheric reliefs-closed in 2 hours, TDAFW steamline shutoff valves-closed in 2 hours, TDAFW pump steam admission valve - open.

D. PORVs-closed in 2 hours, atmospheric reliefs-closed, TDAFW steam line shutoff valves-closed, TDAFW pump steam admission valve - closed in 2 hours.

AOP-6.0

A. Incorrect, TDAFW pump steam admission valve 3226 fails open.

B. Correct - **PORVs-closed, atmospheric reliefs-closed, TDAFW steamline shutoff valves-closed in 2 hours, TDAFW pump steam admission valve - open**

C. Incorrect, the atmospheric relief valves do not have an accumulator and will fail close immediately. The PORVs do not have an accumulator and will fail close immediately.

D. Incorrect the PORVs do not have an accumulator and will fail close immediately, the TDAFW pump steam line shutoff valves have a 2 hour accumulator. TDAFW pump steam admission valve 3226 fails open.

FROM AOP-6.0

Number	Name	Manual Operator	Fail Position
Q2B31V053 (2-RC-HV-445A)	PRZR PORV	NO	CLOSED
Q2B31V061 (2-RC-PCV-444B)	PRZR PORV	NO	CLOSED
Q2N11PV3371A	MAIN STM ATMOS PRESS REL VLV	YES	CLOSED

(2-MS-PCV-3371A)	2A		
Q2N11PV3371B	MAIN STM ATMOS PRESS REL VLV	YES	CLOSED
(2-MS-PCV-3371B)	2B		
Q2N11PV3371C	MAIN STM ATMOS PRESS REL VLV	YES	CLOSED
(2-MS-PCV-3371C)	2C		
Q2N11V001A (2-MS-HV-3369A)	MAIN STM ISO VLV SG 2A	NO	CLOSED
Q2N12V001A-A	STM LINE 2B TO TDAFWP SHUT OFF	YES	CLOSED
(2-AFW-HV-3235A)	VLV		
Q2N12V001B-B	STM LINE 2C TO TDAFWP SHUT OFF	YES	CLOSED
(2-AFW-HV-3235B)	VLV		

From the AFW FSD concerning HV-3235A and B
3.14.2.3 Each valve shall be provided with an air reservoir to ensure that these valves can be opened in emergency conditions. This air reservoir shall have a capacity of 13.4 imperial gallons (2.15 ft3), which is sufficient to open valve HV-3235A or B from a closed position and hold it open for 2 hours (References 6.5.001, 6.7.016,6.7.042, 6.7.048, 6.7.049, 6.7.050).
OPS-52102H
These valves are normally supplied with air from the instrument air system but can also be supplied from the emergency air compressors if necessary. HV-3235A and 3235B fail close on loss of air or power, however an air reservoir is provided that will hold these valves open for a nominal two hours. **HV-3226 fails open or loss of air and power**. Valve HV-3226 is located on the non-rad side, 100 foot elevation and HV-3235A, B are located on the non-rad side, 127 foot elevation, in the MSVR.

000055 Station Blackout / 6 Ability to determine or interpret the following as they apply to a station blackout: EA2.01 – Existing valve positioning on a loss of instrument air system

Describe the effect on the Auxiliary Feedwater of a loss of an AC or DC bus or instrument air (OPS-40201D06).
Describe the effect on the Pressurizer of a loss of an AC or DC bus or instrument air (OPS40301E06).
Evaluate abnormal plant or equipment conditions associated with the Main and Reheat Steam System and determine the local actions needed to mitigate the consequence of the
abnormality (OPS40201A12).

slightly modified - AOP-6.0-52520F07 #2 added 3226 position failure for the question to make it a level 2-3 difficulty Vs. a 1.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

40. 056AK1.01 001

The plant was operating at 10% Reactor power when a Loss of Off-Site Power caused the RCP's to trip. All core-exit thermocouples are operable. ESP - 0.2, Natural Circulation Cooldown to Prevent Reactor Vessel Head Steam Voiding, is being performed.

Which one of the following describes the indication that could be used to verify natural circulation cooling is adequate?

A.✓ Steam Generator pressures stable or falling.

B. RCS cold leg temperature stable or rising.

C. Steam Generator water levels stable or rising.

D. Sub Cooled Margin Monitor indication stable or falling.

Answer A is Correct:

A. Steam Generator pressures stable or lowering

Per ESP 0.2 Step 1.10 RNO

```
1.10 Verify adequate natural
     circulation.

     a) Check SG pressure stable
        or falling.

     b) Check SUB COOLED MARGIN
        MONITOR indication
        greater than 16°F
        subcooled in CETC mode.

     c) Check RCS hot leg
        temperatures stable or
        falling.

        RCS HOT LEG TEMP
      [] TR 413

     d) Check core exit T/Cs
        stable or falling.

     e) Check RCS cold leg
        temperatures at
        saturation temperature
        for SG pressure.

        RCS COLD LEG TEMP
      [] TR 410

     f) IF natural circulation
        NOT adequate,
        THEN dump steam at a
        faster rate.

     g) Begin taking natural
        circulation logs.

     h) Continue efforts to
        establish RCP support
        conditions.
```

Choice B: Incorrect - Tc is checked at saturation for the SG and would be stable or falling.

Choice C: Incorrect - SGWL are not checked in the list and would be most affected by feed vs. steam rate.

Choice D: Incorrect - SCMM is checked greater than 16°F in CETC mode and it would be stable or increasing during natural circulation.

000056 Loss of Off-site Power / 6 Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite power: AK1.01 – Principle of cooling by natural convection

Analyze plant indications to determine the successful completion of any step in ESP-0.2/0.3/0.4 (OPS52531C07).

Modified slightly from question ESP-0.2.3.4-52531C07 12.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

41. 057AA1.04 001

Given the following plant conditions:

- Unit 1 is holding at 25% power due to problems with DEH during a plant startup.
- Rod control is in AUTO, with Bank D rods at 174 steps.
- VCT level transmitter, LT-115, failed low 30 minutes ago.
- I&C is troubleshooting Power Range Nuclear Instrument N-43 because of a blown fuse and all actions required by ARP FC3 have been completed.

Which ONE of the following conditions will occur if power is lost to the 1C 120V AC Vital Bus?

A. A reactor trip will occur.

B. An Auto makeup will commence and continue until secured by the Plant Operator.

C. Automatic rod withdrawal is blocked but the operator can still pull rods in MANUAL.

D✓ A boration of the RCS will begin due to LCV-115B and D, RWST to CHG PUMP, opening.

see figure 5 of 120 VAC lesson plan ops52103D
This describes what happens to LT-115 if LT-112 was failed low and then 120v AC bus A is lost. The exact same event occurs if 120v AC were to be lost to cabinet 7 and LT-115 was failed low. LCV-115B/D would roll open and LCV-115 C/E would go closed.

A - Incorrect; A reactor trip would occur if another PRNI channel, other than N-43, had already been placed in a tripped condition. 120v vital panel C powers N-43.

B - Incorrect; This almost happens if 120v vital panel A were lost. FCV-113A and 114B opens and the BAT & RMW pumps start, but neither FCV-114A or 113B opens so the flow path is not complete.

C - Incorrect; Auto rod control is not affected by N-43 when the conditions of FC3 have been completed. Auto rod control would be a factor if it were not already failed and placed in the trip condition. If the rod block was in affect, rods could not be withdrawn in Auto or Manual.

D - Correct; **A boration of the RCS will begin due to LCV-115B and D, RWST to CHG PUMP, opening.**
A boration of the RCS will occur since power was lost to 1C 120V AC Vital Bus with LT-115 failed low causing LCV-115B, RWST to CHG PUMP, to open and LCV-115C, VCT Outlet ISO, to close. (Aux Safeguards Cabinet B)
Valves LCVs 115D and E are powered from Aux Safeguards Cabinet B.

Prints referenced:D177603, 7377D81, 7378D38, D177303, D177604D177631, D177602

The 7300 cabinets (7300 racks) are fed power as shown below:

7300 Cabinet	Normal Power	Alternate Power
7300 Cabinet 1 and 5	120V Vital AC Instrumentation Panel A	120V Regulated AC Panel C
7300 Cabinet 2 and 6	120V Vital AC Instrumentation Panel B	120V Regulated AC Panel D
7300 Cabinet 3 and 7	120V Vital AC Instrumentation Panel C	120V Regulated AC Panel E
7300 Cabinet 4 and 8	120V Vital AC Instrumentation Panel D	120V Regulated AC Panel F

000057 Loss of Vital AC Inst. Bus / 6 Abilty to operate and/or monitor the following as they apply to the Loss of Vital AC instrument bus:AA1.04 – RWST and VCT valves.

Evaluate abnormal plant or equipment conditions associated with the 120 Volt AC Distribution System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52103D02).

modified 120 VAC-52103D02 #9

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

42. 058AA1.03 001

Plant conditions are as follows:

- Unit 1 is operating at 95% power.
- Technical Specification 3.8.4, A.C. Sources - Operating, is in effect due to 1B Battery Output Breaker, LB-18, being open while an inoperable battery cell is being replaced.

Prior to reclosure of the B Train battery output breaker, a loss of AC electrical power is experienced on the 1G 4160V bus. Which one of the following will occur?

A. 1B Diesel Generator will start and reenergize the 1G 4160V bus.

B. 1B Diesel Generator will start, but the output breaker will NOT close.

C✓ 1B Diesel Generator will NOT start, a reactor trip and safety injection will occur.

D. 1B Diesel Generator will NOT start, a reactor trip will occur, a safety injection will NOT occur.

A and B. Incorrect - 1B DG will NOT start due to DC not available to the air start solenoids. Also there is no DC ABT associated with 1B DG.

C. Correct - **1B Diesel Generator will NOT start, a reactor trip and safety injection will occur.**
the Rx trip is due to loss of RCP brker indication due to the loss of C vital panel. 1 loop out of 3 will cause a Rx trip to occur at 95% power. The SI is due to Ctmt pressure 2/3 B/Ss loss of power TSLB 1 1-3 and 1-4 for PC 953B and 952 B.

D. Incorrect - partially correct, however an SI will occur from one train (A Train). B Train will not occur due to the loss of D vital panel.

00058 Loss of DC Power / 6 Ability to operate and/or monitor the following as they apply to the Loss of DC power: AA1.03 – Vital and battery bus components. Assess abnormal plant or equipment conditions associated with the DC Distribution

Describe the effect on the 120 Volt AC Distribution of a loss of an AC or DC bus or instrument air (OPS40204F06).
Evaluate abnormal plant or equipment conditions associated with the 120 Volt AC Distribution System and determine the local actions needed to mitigate the consequence of the abnormality (OPS40204F12).

DC DIST-40204E08 #2

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

43. 059A2.11 001

The unit is at 100% power when PT-464, Steam Header Pressure, fails **HIGH**.
Which one of the following will be the correct plant response and operator response to this failure?

A. SGFP speed will increase; main FRVs start to open; take manual control of all three main FRVs and close as necessary to control feed flow.

B. SGFP speed will decrease; main FRVs start to close; take manual control of all three main FRVs and open as necessary to control feed flow.

C.✓ SGFP speed will increase; main FRVs start to close; take manual control of SGFP master controller and decrease speed of both SGFPs as necessary to control feed flow.

D. SGFP speed will decrease; main FRVs start to open; take manual control of SGFP master controller and increase speed of both SGFPs as necessary to control feed flow.

A. Incorrect, Sensed Feed header to Steam header DP will be lower than program, and SGFP speed will increase. FRV's will tend to close to offset a flow error signal and an actual rising SG level (level error). Opening FRVs for this failure will not in itself correct the problem.

B. Incorrect, Sensed Feed header to Steam header DP will be lower than program, and SGFP speed will increase. FRV's will tend to close to offset a flow error signal and an actual rising SG level (level error). Opening the FRVs will not correct this problem.

C. Correct, **SGFP speed will increase; main FRVs start to close; take manual control of SGFP master controller and decrease speed of both SGFPs as necessary to control feed flow.**
Feed - steam = DP. steam increases = DP decr. below program and speed increases. Sensed Feed header to Steam header DP will be lower than program, and SGFP speed will increase. As speed increases and SGWL increases, FRVs will go closed. Decreasing the speed of the SGFPs will take care of this problem properly.
The action IAW ARPs is to take manual control of appropriate SG FRV or SGFP speed as required. C is the only correct answer based on SGFP SPEED AND FRV direction. taking control of the SGFPs in this case will cause the desired results.

D. Incorrect, Sensed Feed header to Steam header DP will be lower than program, and SGFP speed will increase. FRV's will tend to close to offset a flow error signal and an actual rising SG level (level error).

059 Main Feedwater Ability to (a) predict the impacts of the following malfunctions or operations on the MFW, and (b) based on those predictions, use the procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.11 – Failure of feedwater control system.

Predict and explain the following instrument/equipment response expected when performing Steam Generator Water Level Control System evolutions including the fail condition, alarms, trip setpoints (OPS52201B08).
- Steam Flow
- Steam Pressure
- Steam Header Pressure

Evaluate abnormal plant or equipment conditions associated with the Steam Generator Water Level Control System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52201B12).

modified from SGWLC-52201B08 #30 and 26
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

44. 061G2.1.12 001

Unit 1 is at 100% power, steady state when HV-3235B, TDAFWP STM SUPP FROM 1C SG, is declared Inoperable. Five (5) days after HV-3235B is declared Inoperable, it is discovered that the 1A MDAFW Pump is Inoperable and has been for the past 2 days.

Assuming that HV-3235B is restored in the next 2 hours, how long is allowed in order to restore the 1A MDAFW Pump to operable status?

References Provided

A. 6 hours from the restoration of HV-3235B to operable status.

B✓ 72 hours from the discovery of 1A MDAFW Pump inoperability.

C. 24 hours from the discovery of 1A MDAFW Pump inoperability.

D. 10 Days from the discovery of HV-3235B inoperability.

Provide Tech Spec 3.7.5 for reference. (**NOT** a 1 hour or less and is an RO question)

A. Incorrect - When HV-3235B is declared Inoperable, Condition A is entered. Condition B is entered when the MDAFW pump is discovered to be inoperable, but this specific combination does not require entry into Condition C.

B. Correct - **72 hours from the discovery of 1A MDAFW Pump inoperability.** After HV-3235B is restored to operability, then the only condition still applicable would be condition B, and 72 hours is more restrictive than the 10 days from the initial discovery of the failure to meet the LCO.

C Incorrect - The 72 hours begins from time of discovery.

D Incorrect - not correct because the 72 hours is more limiting than the 10 day limit, the most limiting in this case per the bases.

061 Auxiliary/Emergency Feedwater G2.1.12 – Ability to apply technical specifications for a system.

Identify and apply any Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Auxiliary Feedwater System (AFW) including (OPS-52102H01):
Technical Specification 3.7.5, Auxiliary Feedwater (AFW) System

AFW-52102H01 #11

MCS	Time:	1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
					Answer:	B B B B B B B B B B	Items Not Scrambled

45. 062AK3.02 001

Given the following plant conditions on Unit 2:

- Service Water (SW) Pumps 2A, 2B, 2D and 2E are running.
- 2C SW pump has been aligned to B Train.
- SW pump "2C" spare pump selector switch has been placed in the "2D" position in preparation for tagging 2D SW pump out.

A spurious Safety Injection occurs. Assuming the ESF sequencers operate properly, which ONE of the following describes the operating status of the SW Pumps following the ESF sequencer operation?

	Pump 2C	Pump 2D	Pump 2E
A.✓	ON	ON	ON
B.	ON	OFF	OFF
C.	OFF	ON	OFF
D.	ON	OFF	ON

A - Correct, **ON** **ON** **ON**
C SW pump will start due to the position of the Pump Spare Selector switch, D & E SW pumps will continue running since an LOSP has not occurred.

B - Incorrect, D and E SW pumps continue running since no LOSP or other trip.

C - Incorrect, 'E' pump continues running and C SW pump will start due to the position of the Pump Spare Selector switch.

D - Incorrect, D SW pump continues running and C SW pump will start due to the position of the Pump Spare Selector switch.

OPS-52102F/40101B/ESP-52102F
One of the train-oriented pumps, (A or B for train A, D or E for train B) can be backed up by pump C. A key-interlock allows pump C to back up one train-oriented pump at any one time. Since only one key is available for the two spare pump selector switches, the key can be removed only when the switch is in the C position (A/C/B or D/C/E). The key must be inserted to change the position of the spare pump selector switch.
An autostart from the Emergency Safeguard System (ESS) or LOSP sequencer starts the train-oriented service water pumps, provided that both spare pump selector switches are in the C position. Otherwise, the C pump will autostart in place of the train-oriented pump selected on the switch. In the case of an ESS start with no LOSP, there could potentially be five (5) service water pumps running. For this to occur, the dedicated pump must already be running with the C pump selected to replace it at the time of the SI without LOSP.

000062 Loss of Nuclear Svc Water / 4 Knowledge of the reasons for the following responses as they apply to the Loss of SW: AK3.02 – The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS.

List the automatic actions associated with the Service Water System components and equipment during normal and abnormal operations including (OPS40101B07):
• Automatic actuation including setpoint (example SI, Phase A, LOSP)

modified from SW-40101B07 #1
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

46. 062K2.01 001

Which one of the following describes the NORMAL, EMERGENCY, and ALTERNATE power supplies to Emergency 4160V AC Bus 1J?

	NORMAL	EMERGENCY	ALTERNATE
A.	S/U 2B	2B DG	S/U 2A
B.	S/U 1A	1B DG	S/U 1B
C.	S/U 1A	2C DG	S/U 1B
D✓	S/U 1B	1B DG	S/U 1A

A - Incorrect, Incorrect - for unit 2.

B - Incorrect- correct DG, wrong S/U for both

C - Incorrect, wrong S/U for both and DG is incorrect.

D - Correct, **S/U 1B** **1B DG** **S/U 1A**

OPS-52103B

Bus G, the heart of train B, normally receives power from S/U transformer B with an alternate supply from S/U transformer A and an emergency supply from emergency diesel generator 1B or 2B, for Unit I and Unit II, respectively.

062 AC Electrical Distribution Knowledge of bus power supplies to the following:
K2.01 – Major system loads

Explain the purpose and operation including the design features and functions, capacities, and protective interlocks for the following major components associated with the Intermediate and Low Voltage AC Distribution System (OPS40102B02):
• 4160V AC Buses and breakers
Identify the power supply for each major electrical component associated with the Intermediate and Low Voltage AC Distribution System including (OPS40102B04):
• 4160V AC Buses

LO/INT VOLT-40102B06 #41

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

47. 063G2.4.10 001

Unit 1 is at 100% power. Preparations are underway to remove the 1A battery from service for maintenance. Battery charger 1B is out of service, battery charger 1C is aligned to 125 VDC Bus 1B.

When the 1A battery output breaker, LA-05, is opened, the following annunciators actuate on the Emergency Power Board Annunciator Panel W:

- WC2, 1A 125V DC BUS UV OR GND
- WC3, 1A 125V DC BUS BATT BKR 72-LA05 TRIPPED
- WC4, 1A BATT CHG FAULT OR DISC

Which one of the following describes a possible cause for the annunciators described above?

A.✓ Battery charger 1A has no voltage output.

B. Battery charger 1C has no voltage output.

C. Excessive voltage is being supplied from battery charger 1A.

D. Excessive voltage is being supplied from battery charger 1C.

Reference: ARP-2.2 Panel WC2,3,4
OPS-52103C lesson plan Table 2
A is correct.

A. Battery charger A has ceased to produce voltage.

SOP-37.1 has the operator make sure a battery charger is on service prior to taking the battery out of service. When the Battery output breaker is opened, LA-05, WC3 will come into alarm due to the b contact from breaker LA05. WC 2 shows either a low voltage condition or a ground. In this case it would be low voltage due to no current. WC4 shows a fault with the battery chgr which could be many different failures, one of which would be loss of AC output.

B. Incorrect - wrong bus. 1C Battery charger is in service but on Bus 1B not 1A.

C. incorrect - not consistent with batt chgr annunciators. Annunciator reads UV or GND and has nothing to do with excessive voltage.

D. incorrect - not consistent with DG or batt chgr annunciators AND on wrong bus

063 DC Electrical Distribution G2.4.10 Knowledge of Annunciator response procedures.

System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52103C02).
Assess abnormal plant or equipment conditions associated with the DC Distribution System and determine the local actions needed to mitigate the consequence of the abnormality (OPS40204E12).

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

48. 064A3.01 001

The 1C Diesel Generator will **auto start** on an Undervoltage condition on which one of the following busses?

A.✓ 2H 4160 volt bus.

B. 1G 4160 volt bus.

C. 1F 4160 volt bus.

D. 2J 4160 volt bus.

OPS-52102I
The emergency diesels will auto start for a safety injection (SI) signal on the associated unit or an under-voltage (UV) condition for their specific bus on the associated unit. The specific buses that the emergency diesels can supply power to are those buses that are used for UV sensing for auto start. 1B and 2B diesels can supply 1G and 1J, and 2G and 2J, respectively. 1-2A diesel can supply 1F or 2F. 1C diesel can supply 1H or 2H.

A. Correct - **The 2H 4160 volt bus.**
an undervoltage condition on the bus that 1C DG normally supplies is 1H or 2H bus.

B. Incorrect - wrong train

C. Incorrect - wrong bus

D. Incorrect - wrong train

064 Emergency Diesel Generator Ability to monitor automatic operation of the D/G system, including: A3.01 – Automatic start of compressor and ED/G

List the automatic actions associated with the Diesel Generator and Auxiliaries System components and equipment during normal and abnormal operations including (OPS40102C07):
• Normal control methods
• Automatic actuation including set point (example SI, Phase A, Phase B, MSLIAS, LOSP, SG level)
• Protective isolations such as high flow, low pressure, low level including set point
• Protective interlocks

Modified DG-40102C07 #2
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

49. 067AA1.06 001

MH1, FIRE, annunciator on the Unit 1 Main Control Board (MCB) has come into alarm. On the control room fire panel, "Aux Bldg El 139 W Side Det 1A-39, 43, 46, 53 1A-55, 59, 106", alarm light is lit. The Rover reports that the window for 1A-59 at the PYR-A-LARM panel is lit.

What is the **minimum** action required to ensure that MCB annunciator, MH1, FIRE, will alert the operator to a fire in one of the remaining fire detection zones?

A. The "Aux Bldg El 139 W Side Det" alarm on the control room fire panel must be acknowledged and the alarm for the annunciator for MH1, FIRE, must be acknowledged.

B.✓ The "Aux Bldg El 139 W Side Det" alarm on the control room fire panel must be acknowledged and the Reflash Panel alarm for detection system 1A-59 must be acknowledged.

C. The "Aux Bldg El 139 W Side Det" alarm on the control room fire panel must be acknowledged.

D. The Reflash Panel alarm for detection system 1A-59 must be acknowledged.

A, C and D. Incorrect -
B. Correct - **The "Aux Bldg El 139 W Side Det" alarm on the control room fire panel must be acknowledged and the Reflash Panel alarm for detection system 1A-59 must be acknowledged.**
The minimum action that is required is to reset both the CR fire panel and the reflash panel to be able to get another alarm from the remaining systems for that alarm. If just the reflash panel is reset then the fire panel in the CR will not see an alarm and cause MH1 to come into alarm. If just the Fire panel in the CR were reset, then the reflash

from occurring.

OPS-52108K - 40103D-ESP-52108K
FIRE ALARM PANELS
Basic Arrangement
Several fire alarm panels working in conjunction with each other to ensure that in the event of a fire, plant personnel can effectively identify its location. The PYR-A-LARM panel (also referred to as the "Pyro" panel) on the 121' non rad side of the Unit 1 and Unit 2 Auxiliary Buildings provides a local audible alarm if a fire occurs as well as a visual indication of which fire protection system is affected.
Alarms from the PYR-A-LARM panel input into a fire alarm panel located in the control room. An alarm on the control room panel causes MCB annunciator MH1, FIRE, to alarm.
Some alarms on the control room fire alarm panel receive several inputs from the PYR-A-LARM panel. Because of this, local reflash panels are provided on the 121' elevation near the PYR-ALARM panel to allow individual alarms which input to common fire alarm panel alarms to be acknowledged. In order for the MCB annunciator MH1 to reflash if another alarm occurs, the fire alarm panel alarms must be acknowledged. Fire alarm panel alarms with multiple inputs cannot be acknowledged until individual alarms are acknowledged on the reflash panel(s). It is important to remember this fact when responding to a fire alarm (MH1). Alarms on both the fire alarm panel and reflash panel (if applicable) must be acknowledged in order to ensure that all fire detection system alarms are received in the control room.
Example: One of the alarms on the fire panel in the control room is "Aux Building Elev. 139' West Side". This one alarm receives input from 12 separate detection systems. If smoke were detected outside the 139' cable spreading room, detector system 1A-59 would come into alarm. This would set off the alarm at the PYR-A-LARM panel, light the 1A-59 window on the PYR-A-LARM panel, and cause the Aux Building Elev. 139' West Side window to alarm on the control room fire panel, which in turn causes MCB annunciator MH1 to alarm.
In order to allow one of the remaining 11 detection systems, which input to the 139' West Side alarm, to alert the operator to another problem the following must occur:
• The Reflash Panel alarm for detection system 1A-59 must be acknowledged - usually the Rover's responsibility.
• The Aux Building Elev. 139' West Side alarm on the control room fire panel must be acknowledged - usually the unit operator's responsibility.
• Once these actions are completed, annunciator MH1, FIRE, will clear and will alarm again if additional smoke/fire is detected.

FNP-0-SOP-0.4
3.0 Pyrotronics Alarm Panel
3.1 This section assumes that there are no inoperable fire barriers involved with the alarming fire detection zone.
3.2 WHEN a fire detection zone alarms on the Pyro Panel, THEN the panel should be checked immediately.
CAUTION: The local reflash unit must be acknowledged to allow reflash on the control room fire alarm panel for any zones on the local reflash unit that subsequently alarm.
3.2.1 The System Operator (SO) will determine which zone is in alarm and acknowledge the local reflash unit (gray box – N1/2V43L004 or L005), if applicable.
4.0 Main Control Room Fire Alarm Panel
CAUTION: The MCB alarm will not reflash unless all alarms are acknowledged on the fire alarm panel.
4.1 WHEN a fire alarm is received on the control room fire alarm panel, THEN the PO should immediately check the alarm panel to determine the affected fire alarm zone and acknowledge the panel.
4.2 The PO should dispatch the SO to the Pyro Panel to determine the alarming fire detection zone and acknowledge the local reflash unit (gray box – N1/2V43L004 or L005), if applicable.

NOTE: Only the system that is in alarm is inoperable as long as the associated local reflash unit has been acknowledged, however, all zones within that system are inoperable when any zone of that system is in alarm. IF a system is in alarm and a local reflash unit is NOT acknowledged, THEN all systems common to a shared alarm window on the BOP Fire Annunciator Panel are inoperable.

000067 Plant Fire On-site / 8 Ability to operate and/or monitor the following as they apply to the Plant Fire on site AA1.06 – Fire alarm

Describe action(s) the Plant Operator and System Operator must take if the Control Room Alarm panel goes into alarm (OPS 40502L04).
State the operation and alignment of each major component including precautions and limitations of operation, and applicable procedures, associated with the Fire Protection System including (OPS40103D11): • Fire Alarm Panels

FIRE PROT-52108K02 #1
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

50. 068AK2.01 001

Units 1 and 2 were operating at 100% power when a toxic gas problem in the control room caused the operators to evacuate and establish control at the Hot Shutdown Panel (HSP).

Which one of the following statements correctly describes the operation of Unit 1 RCP seal injection (SI) flow, Charging flow and Letdown flow from the correct HSP?

A. RCP SI flow can be isolated/restored from B HSP; Charging flow can be isolated/restored from B HSP; and Letdown flow can be isolated/restored from B HSP.

B. RCP SI flow can be isolated/restored from B HSP; Charging flow can be isolated/restored from B HSP; and Letdown flow can be throttled from C HSP.

C.✓ RCP SI flow can be throttled from A HSP; Charging flow can be throttled from A HSP; and Letdown flow can be isolated/restored from A HSP.

D. RCP SI flow can be throttled from A HSP; Charging flow can be throttled from A HSP; and Letdown flow can throttled from C HSP.

OPS- 52202D Lesson Plan
A,B and D. Incorrect

C. Correct - **RCP SI flow can be adjusted manually from A HSP, Charging flow can be adjusted manually from A HSP, and Letdown flow can only be isolated/restored from A HSP.**
All three can only be controlled from A HSD panel. RCP SI FCV-186 and charging flow FCV-122 can be taken to manual and using a potentiometer flow can be throttled in manual. Letdown flow can only be secured and cannot be controlled or throttled. The letdown orifice isolation valves will only secure flow.

000068 (BW/A06) Control Room Evac. Knowledge of the interrelations between control room evacuation and the following: AK2.01 – Auxiliary shutdown panel layout.

Explain the purpose/design features of the Hot Shutdown Panels System OPS40204G01
Describe the physical in plant location of the major components associated with the Hot Shutdown Panels System (OPS40204G03):
- HSP-A
- HSP-B
- HSP-C
- HSP-D
- HSP-E
- HSP-F
- HSP-G

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

51. 068K6.10 001

A #2 Waste Monitor Tank release to the environment is in progress in accordance with a Liquid release permit and SOP-50.1, Appendix 2, "Waste Monitor Tank #2 Release to the Environment."

- Annunciator FH2, RMS CH FAILURE, alarms.
- The LOW ALARM light is found to be illuminated on R-18, Liquid Waste Processing Monitor, on the RMS panel in the control room.

Which one of the following describes the system response?

A. #2 WMT pump trips.

B.✓ RCV -18, Liquid Waste Release Valve, automatically shuts.

C. No automatic action occurs, the systems operator will have to secure the release.

D. #2 WMT pump trips and RCV -18, Liquid Waste Release Valve, automatically shuts.

FH2
The radiation monitors fail to a "High Radiation" conditions on loss of instrument and/or control power that will result in actuation of associated automatic functions.
OPS-52106A
Liquid Waste Discharge Valve (RCV-018)
The control valve is operated by a three-position handswitch (CLOSE/AUTO/OPEN, spring return to AUTO from OPEN). In the AUTO position, the valve will automatically shut on a high radiation level from RE-18. Valve position lights indicate above the switch. The valve fails closed on a loss-of-power or loss-of-air.
OPS-52106D
The LOW ALARM indicator illuminates when the RMS input signal has failed or the RMS malfunctions.

A - Incorrect; #2 WMT pump does not trip on a hi rad alarm

B - Correct; **RCV -18, Liquid Waste Release Valve, automatically shuts.**

C - Incorrect; RCV-18 automatically shuts on a HI rad alarm.

D - Incorrect; #2 WMT pump does not trip on a hi rad alarm.

068 Liquid Radwaste Knowledge of the effect of a loss or malfunction on the following will have on the liquid radwaste system: K6.10 – Radiation monitors

• List the automatic actions associated with the Radiation Monitoring System components and equipment during normal and abnormal operations including (OPS40305A07):
• Evaluate abnormal plant or equipment conditions associated with the Radiation Monitoring System and determine the local actions needed to mitigate the consequence of the abnormality (OPS40301F12).
• Identify any special considerations such as safety hazards and plant condition changes that apply to the Liquid and Solid Waste System (OPS52106A04).

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

52. 069AK2.03 001

Unit 1 is in a refueling outage and the following conditions exist:

- RCS water level is 129' 5".
- Both trains of RHR have been lost.
- AOP-12.0, Residual Heat Removal System Malfunction, has been initiated.
- Containment airlock interlocks have been restored.
- Containment Closure has been set.

Which one of the following conditions represents a loss of Containment Integrity for this condition?

A.✓ The equipment hatch is closed and held in place with 3 bolts with no visible air gaps.

B. An operator enters containment to perform emergency actions and leaves the inner airlock door open.

C. 1A Atmospheric relief valve is removed and has a blind flange installed on the bonnet in place of the valve.

D. With HV-3045 and HV-3184, CCW FROM RCP THRM BARR, valves closed, maintenance is performed which allows HV-3045 to be cycled.

STP-18.4
A - Correct; **The equipment hatch is closed and held in place with 3 bolts with no visible air gaps.**
the equipment hatch is required to have 4 bolts installed, one in each quadrant IAW FNP-0-MP-38.0 PER stp-18.4.

B - Incorrect; One airlock door can be opened at a time.

C - Incorrect; This represents a proper way to establish Containment Closure.

D - Incorrect; for ctmt closure to be met, only one valve in this line needs to be closed IAW STP-18.4

Containment Closure is defined as: a) equipment hatch is closed and held in place by a minimum of four (4) bolts, b) a minimum of one (1) door in each air lock is closed, c) each penetration providing access from the containment atmosphere to the outside atmosphere shall be closed by a containment isolation valve OR a containment isolation blind flange (closure by other valves or blind flanges may be used if they are similar in capability to those provided for containment isolation AND have been evaluated on a case-by-case basis). This is the definition of Closure required to ensure that containment is an effective barrier to the release of radioactive material.

STP-18.4 APP A; PAGE 3
NOTE • Intact as used for refueling integrity, means that the valve, relief valve, indicator, or pressure transmitter will prevent air to air passage from containment.
• IF all outside containment components for a penetration are in the required position, THEN integrity is satisfactory for that penetration and inside containment integrity is not required.
• IF all inside containment components for a penetration are in the required position, THEN integrity is satisfactory for that penetration and outside containment integrity is not required. Consider tagging boundaries inside containment if establishing mid loop integrity simultaneously.

000069 (W/E14) Loss of CTMT Integrity / 5 Knowledge of the interrelations between the Loss of CTMT integrity and the following: AK2.03 – Personnel access hatch and emergency access hatch.

Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Containment Structure and Isolation System (OPS52102A01).
• 1.6 Containment Integrity – Definition

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

53. 073K1.01 001

Which one of the following automatic actions will occur upon receipt of a high radiation alarm on the listed instrument?

A. R-19 (SGBD sampling) closes dilution discharge valve, 1-BD-RCV-23B.

B. R-23B (SG blowdown processing) shuts SGBD heat exchanger discharge valve, 1-BD-FCV-1152.

C. R-25A (fuel handling building HVAC) starts the fuel building HVAC supply and exhaust fans.

D✓ R-35A (control room HVAC) isolates computer room supply and return HVAC dampers.

FH1 annunciator Response Procedure
A. Incorrect -R-19 Steam Generator Blowdown/Sample Isolates sample lines 3328, 3329, 3330

B. Incorrect - R-23B SG Blowdown Surge Tank Discharge Closes RCV-23B, R-23A closes FCV-1152.

C. Incorrect - R-25A/R-25B* Spent Fuel Pool Ventilation Trip fuel bldg supply and exhaust fans; closes SFP HVAC supply and exhaust dampers; starts associated trains of penetration room filtration.

D. Correct - **R-35A (control room HVAC) isolates computer room supply and return HVAC dampers.**
R-35A*/R-35B* Control Room Vents (Control Room Ventilation Room) Isolates control room supply and return from computer room AHU, closes utility exhaust dampers, and shifts TSC ventilation to recirc (filtration mode)

073 Process Radiation Monitoring Knowledge of the physical connections and/or cause -effect relationships between the PRM system and the following systems: K1.01 – Those systems served by PRMs.

• List the automatic actions associated with the Radiation Monitoring System components and equipment during normal and abnormal operations including (OPS40305A07):
• Explain the purpose and operation including the design features and functions, and protective interlocks for the following major component associated with the Radiation Monitoring System (OPS40305A02):
• Steam Generator Liquid Sample Monitor (R-19)
• Steam Generator Blowdown Processing System Monitor (R-23A)
• Steam Generator Blowdown Discharge Radiation Monitor (R-23B)
• Containment Purge Exhaust Flow Gas Monitors (R-24A and B)
• Spent Fuel Pool Exhaust Flow Gas Monitors (R-25A and B)
• Victoreen Airborne Monitor (R-35A and B)

RMS-40305A07 #21
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

54. 073K3.01 001

Which one of the following is correct concerning the sample requirements for an effluent release with R-18 inoperable?

A. No release can be made until R-18 is returned to service.

B. A grab sample taken and analyzed after the release is initiated and once per hour while the release is in progress.

C.✓ At least two independent samples are analyzed for each Batch release prior to discharging the Waste Monitor Tank.

D. At least one independent sample is analyzed for each Batch release prior to discharging the Waste Monitor Tank and a grab sample taken hourly while the release is in progress.

ODCM page 2-1, 2-3, 2-4, 2-7, 2-9
Table 2-1 #1a. ACTION #28 page 2-4 ACTION 28 a.and Table 2-3 says Batch.

A. Incorrect – release is permitted

B. Incorrect – No grab sample is required per Table 2-3 and sample requirements per action 28 call for 2 independent samples

C. Correct – **At least two independent samples are analyzed for each Batch release prior to discharging the Waste Monitor Tank.**
per ODCM and Page 39 of 052106D

D. Incorrect - No grab sample is required per Table 2-3 and sample requirements per action 28 call for 2 independent samples

073 Process Radiation Monitoring Knowledge of the effect that a loss or malfunction will have on the following: K3.01 – Radioactive effluent releases

Discuss the operation and alignment of each major component including precautions and limitations of operation, and applicable procedures, associated with the Radiation Monitoring System including: (OPS40305A11).
• Waste Processing System Liquid Effluent Monitor (R-18)

LIQ SD WAST-40303A11 #7

MCS	Time:	1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
					Answer:	C C C C C C C C C C	Items Not Scrambled

55. 076A2.02 001

Unit 1 is holding for chemistry at 33% reactor power. The OATC is swapping Service Water (SW) pumps IAW SOP-24.0. Immediately after the SW pump swap has been completed, the OATC checks all isolation MOV's for SW to the Turbine Building open. ALL SW to the Turbine Building MOV's indicate CLOSED.

- AF5, SW TO TURB BLDG A OR B TRN FLOW HI, comes into alarm

The OATC attempts to open ALL MOV's for SW to the Turbine Building.
Only MOV-514, SW TO TURB BLDG ISO B TRAIN, and MOV-516, SW TO TURB BLDG ISO A TRAIN, indicate OPEN.

Which one of the following describes the condition of the Service Water System and the appropriate actions?

A.✓ Both Trains of SW to the Turbine Building are isolated; Trip the turbine; Go to AOP-3.0,TURBINE TRIP BELOW THE P-9 SETPOINT.

B. Both Trains of SW to the Turbine Building are isolated; Trip the reactor and the turbine; Go to EEP-0, REACTOR TRIP OR SAFETY INJECTION.

C. One Train of SW to the Turbine Building is isolated; Remain on line, monitor Turbine Building component temperatures and begin securing Turbine Building equipment as necessary.

D. One Train of SW to the Turbine Building is isolated; Remain on line and check generator hydrogen temperature less than 46°C, continue in AOP-7.0, LOSS OF TURBINE BUILDING SERVICE WATER.

Service water is supplied to the turbine building from the service water supply headers through MOVs-514, 515, 516, and 517. These valves are operated from the main control board and will automatically close upon receipt of a T-signal (phase A isolation) or a high DP (11 psid) corresponding to a flow through the valve exceeding 17,600 gpm. A high flow in either train will isolate both trains of service water to the turbine building.
A condition highly likely at FNP is while swapping SW pumps, the high DP condition of 11 psid can exist. (past OE) This will cause a loss of SW to the TB and is the reason for the steps to immediately proceed in SOP-24.

A - Correct, **Both Trains of SW to the Turbine Building is isolated; Trip the turbine; Go to AOP-3.0,TURBINE TRIP BELOW THE P-9 SETPOINT.**
With these two valves open and the other valves closed, there would be no SW flow to the TB. IAW AF5 and AOP-7.0, IA steps 1, If SW cannot be immediately restored and if Rx power is < 35%, trip the turbine go to AOP-3.0.

B - Incorrect, This is the action if >35% power.

C - Incorrect, Since both trains are isolated, there would be no need to continue in AOP-7.0. These are the next procedural steps if both trains were isolated and the Rx was shutdown.

D- Incorrect - Since both trains are isolated, there would be no need to continue in AOP-7.0. These are the next procedural steps if these conditions exist.

The operator is directed to check at least one service water train aligned to turbine building.
If neither train is aligned, the operator is directed to restore at least one train of service water flow to the turbine building.
If at least on train of service water flow cannot be immediately restored and reactor power is greater than 35%, the operator is directed to trip the reactor. If reactor power is less than 35%, and neither train of service water can be restored to the turbine building, the operator is directed to trip the turbine and perform AOP-3, "Turbine Trip Below The P9 Setpoint". If the reactor is tripped, EEP-0, "Reactor Trip Or Safety Injection" is entered and steps 3, 4, and 5 of this procedure should be performed if sufficient personnel are available. Manually tripping, if reactor power is greater than 35%, will allow the operator to secure other equipment (main turbine, main generator, SGFPs, condensate pumps, and heater drain pumps) that was required for power operation.

076 Service Water Ability to (a) predict the impacts of the following malfunctions or operations on the SWS, and (b) based on those predictions, use the procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.02 – Service water header pressure

- Analyze plant indications to determine the successful completion of any step in AOP-7.0. (OPS52520G07)
- Evaluate plant conditions to determine if entry into AOP-7.0 is required. (OPS52520G02)
- Evaluate plant conditions and determine if transition to another section of AOP-7.0 or to another procedure is required. (OPS52520G08)
- Describe the sequence of major actions associated with AOP-7.0. (OPS52520G04)

AOP-7.0-52520G04 #2

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

56. 076AA2.02 001

Given the following conditions:

- A rapid load reduction from 100% power to 65% power was performed approximately 3 hours ago.
- FG5, GFFD SYS TRBL, is in alarm.
- R-50, GFFD radiation monitor is in alarm.
- R-4, 1C CHG PUMP RM Area Radiation Monitor, is in alarm.
- Chemistry confirms RCS Gross Specific Activity exceeds the limits of Tech Spec 3.4.16, RCS Specific Activity, due to failed fuel.

The OSS directs plant shutdown required by Tech Specs to be performed. Which one of the following actions is designed to limit the release of radioactivity in the event of a subsequent SG tube rupture?

A. The ruptured SG MSIV's are closed.

B.✓ The RCS is cooled down below 500°F.

C. The ruptured SG Atmospheric relief valve setpoint is verified at 8.25 in AUTO.

D. The standby mixed bed demineralizer is placed in service and letdown flow raised to 120 gpm.

A. Incorrect - The ruptured MSIV and bypass valves are closed for a SGTR but not for this reason. The atmospherics and code safety reliefs can still release contamination if they reach the setpoint.

B. Correct -**The RCS is cooled down below 500°F.**
per TS bases 3.4.16
RCS Specific Activity B 3.4.16
APPLICABILITY
In MODES 1 and 2, and in MODE 3 with RCS average temperature > or equal to 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR or MSLB to within the acceptable site boundary dose values.
For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

C. Incorrect - The ruptured SG Atmospheric relief valve setpoint is verified at 8.25 in AUTO for a SGTR but not for this reason.

D. Incorrect - IAW AOP-32, the Standby demins are placed on service and letdown flow increased to reduce RCS crud, not for a SGTR. It would not be placed in service for this event.

000076 High Reactor Coolant Activity / 9 Ability to operate and/or monitor the following as they apply to the High RCS activity: AA2.02 – Corrective actions required for high fission product activity in RCS.

State the basis for all cautions, notes, and actions associated with AOP-32.0 (OPS52521J03).
Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Gross Failed Fuel Detector (OPS52106E10). • Technical Specification 3.4.16, RCS Specific Activity

MCS	Time: 1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
				Answer:	B B B B B B B B B B	Items Not Scrambled

57. 078A3.01 001

An Instrument Air header line break has occurred in the Unit 1 100' Lower Equipment Room. Which one of the following describes the position of V-902, Instrument air dryer bypass valve, V-903, Instrument air to Turbine Building isolation valve, and V-904, Instrument air to Service Building isolation valve, when instrument air pressure reaches 60 psig?

	V-902	V-903	V-904
A.	CLOSED	CLOSED	CLOSED
B.	CLOSED	OPEN	CLOSED
C.	OPEN	CLOSED	OPEN
D.✓	OPEN	OPEN	OPEN

ARP- 1.10 and OPS- 52108A

A. Incorrect - v902 will be open

B. Incorrect - v902 will be open

C. Incorrect - v903/904 will be closed

D. Correct - **OPEN** **OPEN** **OPEN**
v902 will open to bypass the air dryer and V903 and 904 will remain open until pressure decreases below 45 and 55 psig, respectively.
If the instrument air header pressure decreases to 70 psig, the air drying unit bypass valve, V-902, opens (refer to Figure 3). If pressure drops to 55 psig (Unit I only), V-904 closes, isolating instrument air to the service building. At 45 psig, instrument air to the turbine building and out-side areas is isolated by closing V-903 in an attempt to provide the auxiliary building and containment with instrument air.

078 Instrument Air Ability to monitor automatic operation of the IAS including:
A3.01 - air pressure

List the automatic actions associated with the Compressed Air System components and equipment during normal and abnormal operations including (OPS40204D07):
• Normal control methods
• Automatic actuation including setpoints for selective isolation on decreasing header pressure

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

58. 078A4.01 001

A rupture occurred in the service air header. Air pressure decreased and resulted in the isolation of service air. After the isolation, instrument air header pressure returned to normal.

Which ONE of the following describes what the air pressure indicators on the Main Control Board, PI-4004A, "SVC AIR PRESS," and PI-4004B, "INST AIR PRESS," should indicate?

A. 0 psig on both PI-4004A and PI-4004B.

B. 90 -100 psig on both PI-4004A and PI-4004B.

C. 90 -100 psig on PI-4004A and 0 psig on PI-4004B.

D✓ 0 psig on PI-4004A and 90 -100 psig on PI-4004B.

A - Incorrect, Once the break is isolated, by automatic closure of V901, in the service air header the pressure in the instrument air header will return to normal so PI-4004B will indicate normal instrument air pressure.

B - Incorrect, PI-4004A can not indicate pressure due to the rupture in the service air header.

C - Incorrect, This is the opposite of what is true.

D - Correct, **0 psig on PI-4004A and 90 - 100 psig on PI-4004B.**
Since V901 does not automatically reopen PI-4004B will indicate normal air pressure and due to the break in the service air header this line will depressurize so PI-4004A will indicate 0 psig.

078 Instrument Air Ability to manually operate and/or monitor in the control room:
A4.01 – Pressure gauges.

Predict and explain the following instrument/equipment response expected when performing Compressed Air System evolutions including the fail condition, alarms, trip setpoints (OPS40204D08).

COMP AIR-40204D08 #1
Source: Farley Bank Question #O52108A03014
2001 nrc exam

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

59. 103A1.01 001

Given the following plant conditions:

- Unit 1 is at 100% power
- Containment Mini-Purge supply and exhaust fans are tagged out for maintenance.
- A small air leak inside containment is causing a slow rise in containment pressure.
- Containment pressure is currently 1.7 psig.

In order to ensure containment pressure is maintained below Technical Specification maximum pressure, containment pressure will have to be reduced.
Containment should be vented...

A.✓ using the Post-LOCA vent system.

B. into the piping penetration room using Containment Purge Exhaust Bleed MOV-2788A.

C. using a containment sample point lineup to a filtered poly bottle in the 139' electrical penetration room.

D. into the piping penetration room using Containment Mini-Purge exhaust manual fill line bleed valve, V285B and a filtered poly bottle.

A. Correct - **using the Post-LOCA vent system.**
IAW SOP-10 Appendix 1, the only appropriate means to vent the ctmt is to go thru the plant vent stack for a monitored release.

B. Incorrect - this is a plausible vent point, however, these valve are required to be powered down and capped at power.

C. Incorrect - there is no procedural guidance and no vent rig available to hook up to the sample connection in the 139'. This sample point is for a removable sample vessel for post loca ctmt sampling.

D. Incorrect - This is a normally closed and seal wired manual vent valve on the 48" CP line in the 121' PPR. This should not be used due to ctmt isolation requirements in modes 1-4.

103 Containment Ability to predict and/or monitor changes in parameters associated with operating the containment system controls including: A1.01 – Containment pressure, temperature, and humidity

Describe the local actions needed to support plant operation during normal, abnormal and emergency conditions associated with the Post LOCA Atmospheric Control System (OPS40302E09).
Describe the local actions needed to support plant operation during normal, abnormal and emergency conditions associated with the Containment Ventilation and Purge System (OPS40304A09).
Evaluate abnormal plant or equipment conditions associated with the Containment Ventilation and Purge System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52107A02).

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

60. G2.1.2 001

Complete the following sentence with words from the choices below that make it a correct statement.
In cases of emergency, _(1)_ personnel are authorized to depart from plant procedures provided that such departure is necessary to prevent _(2)_.

	(1)	(2)
A.✓	All	damage to the facility
B.	Only Operations	radiation release to the public
C.	All Licensed	unplanned ESF actuation
D.	Only On Shift	an automatic reactor trip

A. Incorrect - Licensed personnel are included in the list but ALL personnel are authorized IAW AP-6. The reason is incorrect per AP-6.

B. Incorrect - Licensed personnel are included in the list but ALL personnel are authorized IAW AP-6. The reason is incorrect per AP-6.

C. Correct - **All** **Prevent damage to the facility**
AP-6
NOTE: • In the following section, "emergency" should be defined as any condition where the plant is in a transient, unstable, or degraded condition or any condition threatening personnel safety, where operator action is required to stabilize the plant and/or mitigate the transient/condition OR an actual declared emergency per EIP-9.0 exists.

4.1 Emergency Condition
4.1.1 In cases of emergency, **all personnel are authorized** with approval of Shift Supervisor, if time allows, to depart from plant procedures (provided that such departure is necessary) to prevent:
• Injury to personnel,
• Danger to the public,
• Damage to the facility,
• Stabilize and/or mitigate a transient.
Authorized Deviations
Nuclear power plants are licensed to operate under the requirements set forth in the government's Code of Federal Regulations (CFR). Title 10 part 50 (10CFR50) of this code specifies the rules governing the licensing of commercial nuclear power plants. A sub section of part 50, section 50.54, specifies conditions applicable to this operating license. A sub section of 10CFR50.54, 50.54(x), contains a provision that allows operation outside of the license conditions and/or Technical Specifications. It states that, "*A licensee may take reasonable action that departs from a license condition or technical specification in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with the license condition and technical specifications that can provide adequate or equivalent protection is immediately apparent.*"
An "emergency" should be defined as any condition where the plant is in a transient, unstable, or degraded condition or any condition threatening personnel safety where operator action is required to stabilize the plant and/or mitigate the transient/condition OR an actual declared emergency per EIP-9.0 exists.
The next section in the code, 50.54(y), qualifies the use of an authorized deviation. It requires that the authorization or approval come from a licensed senior operator before the deviation can occur.

D. Incorrect - not an emergency as defined by AP-6 and ALL personnel are authorized.

G2.1.2 Knowledge of operator responsibilities during all modes of plant operation

Explain management's expectations associated with authorizing procedural deviations (OPS52303E01).

AP-6-40502E01 #3
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

61. G2.1.22 001

Given the following condition on Unit 1:

- Tavg = 300°F
- Keff = .98
- Rated Thermal Power = 0%

Which one of the following is the OPERATIONAL Mode?

A. Mode 2

B. Mode 3

C.✓ Mode 4

D. Mode 5

References: TECH SPEC 1.1
C. Correct - **Mode 4**
Reference:
Tech Spec 1.1 Definitions

MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

Table 1.1-1 (page 1 of 1)

MODES

MODE TITLE REACTIVITY % RATED AVERAGE
CONDITION THERMAL REACTOR COOLANT (keff) POWER(a)
TEMPERATURE (°F)

1	Power Operation	0.99	> 5	NA
2	Startup	0.99	5	NA
3	Hot Standby	< 0.99	NA	350
4	**Hot Shutdown[b]**	**< 0.99**	**NA**	**350 > Tavg > 200**
5	Cold Shutdown[b]	< 0.99	NA	200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

G2.1.22 Ability to determine Mode of Operation

Define the terms listed in Technical Specifications/Technical Requirements Manual (OPS52302A01).

INTRO TS-52302A01 #20

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

62. G2.1.33 001

Which one of the following scenarios would require entering an ACTION statement per Technical Specifications or the Technical Requirements Manual?

A.✓ In MODE 2, Accumulator Tank 1A level indicates 68% on both indicators.

B. In MODE 4, 1A Boric Acid Tank boron concentration is reported by Chemistry to be 7290 ppm.

C. In MODE 1, Accumulator Tank 1A boron concentration is reported by Chemistry to be 2290 ppm.

D. In MODE 6, 1A Boric Acid Tank volume indicates 25%, 1B Boric Acid Tank indicates 5% and the RWST indicates 35 feet.

TRM 13.1.7; TS 3.5.1

A. Correct, **In MODE 2, Accumulator Tank 1A level indicates 68% on both indicators.**
SR 3.5.1.2 Verify borated water volume in each accumulator is greater than or equal to 7555 gallons (31.4%) and less than or equal to 7780 gallons (58.4%).

B. Incorrect - This is required to be between 7000-7700 ppm boron.

C. Incorrect - SR 3.5.1.4 Verify boron concentration in each accumulator is greater than or equal to 2200 ppm and less than or equal to 2500 ppm. Meets TS requirements.

D. Incorrect, In MODE 5 and 6 there is only a need for 2000 gallons of water
DG4
SETPOINT: 65% Level BAT
ORIGIN: 1. Level Bistable LB-106C from Level Transmitter LVL
(N1E21LT106) LO-LO
NOTE: • The minimum required borated water volume for Modes 1 through 4 is 11,336 gallons (52% level in the BAT).
• The minimum required borated water volume for Modes 5 and 6 is 2,000 gallons (approximately 6% level in the BAT).
Added the RWST to this distracter b/c the TRM has boric acid system and RWST operable or enter the action statement.

G2.1.33 Ability to recognize indications for system operating parameters which are entry level conditions for Tech Specs.

Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Emergency Core Cooling System (OPS52102B01).
• 3.5.1 Accumulators
• 3.5.2 ECCS—Operating
• 3.5.3 ECCS—Shutdown
• 3.5.4 Refueling Water Storage Tank (RWST)
• 2.1.1 Reactor Core Safety Limits
Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Boric Acid System (OPS52101I01).
• TRM 13.1.2, Boration Flow Path - Shutdown
• TRM 13.1.3, Boration Flow Path - Operating
• TRM 13.1.6, Borated Water Source - Shutdown
• TRM 13.1.7, Borated Water Sources - Operating

Modified from ECCS-52102B01 #12
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

63. G2.2.13 001

An OPERATING PERMIT TAG present on a component means that:

A. The component position can NOT be changed – The equipment is in operation.

B. The component position can NOT be changed – The equipment is isolated for maintenance.

C✓ The component position can be changed - The equipment is under the control of a Tagout Holder for the purposes of position alignment, testing, or maintenance.

D. The component position can be changed – The equipment has been designated as an isolation boundary for personnel safety and will be isolated in the tagout process.

Equipment Clearance and tagging lesson plan
S-GE-LP-400

A. Incorrect - the purpose of a OPERATING PERMIT TAG is to allow operation of the equipment by one tagout holder.

B. Incorrect - the purpose of a OPERATING PERMIT TAG is to allow operation of the equipment by one tagout holder.

C. Correct - **The component position can be changed - The equipment is under the control of a Tagout Holder for the purposes of position alignment, testing, or maintenance.**

D. Incorrect - The first part is true, however, this is not an isolation boundary

Operating Permit Tagout (Figure 3)

Operating Permit Tagouts (OP) allow the operation of components turned over to a Tagout Holder for testing, minor maintenance, or configuration control. An Operating Permit provides a means of controlling equipment configuration without limiting the operation of the equipment. As a result, Operating Permit Tags will not be hung on equipment that is also danger tagged. Operating Permit Tags are used with this tagout type and will include the reason for the tagout (i.e. MOV testing, LLRT, configuration control etc.). Only a single Tagout Holder is allowed to be signed on to an Operating Permit Tagout at any one time. The standards that accompany Operating Permit Tags and Operating Permit Tagouts include the following:

- *Operations Permit Tags permit operation of a component on which a tag is placed. The individual signed on as Tagout Holder is accountable for the component's status.*
- *No more than one Operations Permit Tag shall be hanging on a component at the same time.*
- *Danger Tags and Operations Permit Tags shall not co-exist on the same component.*
- ***Personal Danger Tags (PDTs) or Maintenance Locks are permitted on Operating Permit Tags***

G2.2.13 Knowledge of tagging and clearance procedures.

Define the following terms: (S-GE-LP-400-T03-L01)

Alternate Boundary

Alternate Release

Clearance

Caution Tag

Danger Tag

Operating Permit Tag

TAG-SGELP400-T03-L01 #3

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

64. G2.2.26 001

What prevents raising a spent fuel assembly out of the water with the new fuel elevator?

A.✓ Administrative controls and a mechanical stop, if installed.

B. The elevator will not raise with the weight of a spent fuel assembly.

C. The SFP bridge crane limit switches prevents inserting a spent fuel assembly into the elevator.

D. The length of the spent fuel handling tool prevents inserting a spent fuel assembly into the elevator.

Reference: FHP-5.17

3.0 Precautions and Limitations

3.1 Never attempt to raise a fuel assembly in the elevator without the permission of both the **Shift Supervisor AND the Fuel Handling Supervisor.**

3.2 IF the mechanical stop is installed, THEN the geared limit switch should stop the elevator prior to making contact with the stop. The mechanical stop is a fail-safe device.

3.3 Have **HP concurrence** before raising the New Fuel Elevator.

A. Correct per the above [The new fuel elevator has the capability to lift a fuel assembly out of the water.]

B, C, D. Incorrect - Elevator is capable of lifting the weight of the fuel assembly and the handling tool hooked to the crane is used to move new fuel from the elevator to the spent fuel pool

G2.2.26 Knowledge of refueling administrative requirements.

Discuss the operation and alignment of each major component including precautions and limitations of operation, and applicable procedures, associated with the Fuel Storage, Handling and Refueling System including (OPS40305B11):
- New Fuel Bridge Crane
- New Fuel Monorail Hoist
- New Fuel Elevator

REFUEL/STOR-40305B07 #24.
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

65. G2.3.1 001

Which one of the following pertain to exposure limits for ambulance personnel while transporting accident victims?

A. Exposure is limited to 3 rem under all circumstances.

B.✓ An ambulance attendant may receive up to 25 rem to save a life.

C. An ambulance attendant is limited to nonoccupational exposure limit of 500 mrem.

D. Exposure must be limited to 7.5 rem if the number of attendants is such that personnel cannot be rotated.

<u>EIP-11.0 page 3</u>
7.0 Exposure Limits for Ambulance Personnel
7.1 Rotate attendants to maintain dose ALARA, if there is an adequate number of attendants so that rotation will not endanger the patient.
7.2 Limit dose to 5 REM if a higher dose is not required to save a life.
7.3 Limit dose to 25 REM if necessary to save a life.

A. Incorrect - 3 rem is not correct for this case.

B. Correct - **An ambulance attendant may receive up to 25 rem to save a life.**
Per EIP-11

C. Incorrect - this is not an occupational dose limit for ambulance personnel. This is the limit for a female who has voluntarily accepted the limit while pregnant.

D. Incorrect - EIP-11 says to rotate attendants to maintain dose ALARA if there are adequate number of attendants and will not endanger the patient. 5 rem is the Max limit unless saving a life requires up to 25 rem.

G2.3.1 Knowledge of 10 CFR 20 and related facility radiation control requirements.

Describe the duties and responsibilities of the Recovery Manager.(OPS53002A11)
Evaluate emergency reports involving injured personnel, use plant procedures/references to determine the appropriate emergency responses. (OPS53002A10) (EIP-11.0).

EPIP OVER-53002A011 #1

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

66. G2.3.10 001

When an individual in the plant discovers an area radiation monitor has a valid alarm and does not know the reason, the Health Physics Manual requires which one of the following?

A. Call HP to verify the high alarm is valid and meet HP at the alarming monitor.

B. Evacuate the affected area and make an announcement to clear the area.

C. Call HP after trying to RESET the alarming radiation monitor.

D✓ Evacuate the affected area and notify the control room.

A. Incorrect - There is no requirement to call HP and it is contrary to the guidance in the HP manual to remain in the area. The CR will call all supporting groups as needed.

B. Incorrect - an announcement would notify the control room but is not required and not the proper way of doing business. The CR will make an announcement.

C. Incorrect - Calling HP is not the requirement. Also resetting the rad monitor is not the job of any individual until HP has surveyed the area. The CR will contact HP as necessary.

D. Correct - **Evacuate the affected area and notify the control room.**

HP manual Page 2
3.1.8 **Leave the area and then notify the Control Room if you find any permanently installed plant fixed area radiation monitor or any permanently installed plant fixed air monitor alarming.** If it can be determined that the alarm is not due to a radiological hazard (e.g. Maintenance, Calibration Repair, Outstanding Deficiency Report, etc.), then exiting area and notification is not required.
OPS-52106D
Airborne Radiation Monitoring System (Figures 15 through 17)
The Victoreen airborne detectors (R-30 through 34) are completely self-contained, off-line units with no control room instrumentation or indication. The units have positive displacement pumps similar to the Westinghouse APD (Figure 16).
All instrumentation is incorporated at the local control panels. If an alarm condition exists at one of these panels, the control room operator would be dependent on a report from someone in the area near the monitor.

G2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

Describe how the rad monitoring system helps to protect the health and safety of plant workers and the public. (ESP52106D08)

RMS-40305A11 #6
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

67. G2.4.17 001

Which one of the following terms is a continuing action term used in Emergency Response Procedures that directs an operator to operate appropriate components to control a specific parameter within the bounds of a procedure?

A. Adjust

B. Establish

C. Implement

D✓ Maintain

SOP-0.8, ATTACHMENT 1, GLOSSARY OF TERMS AND ACTION VERBS

A. Incorrect, Definition of ADJUST: "To change a specified feature so that a specified parameter meets specified value."
B. Incorrect, Definition of ESTABLISH: "To make arrangements for a stated condition. "
C. Incorrect, Definition of IMPLEMENT: "To begin and maintain a prescribed course of action. "
D. Correct, Definition of **MAINTAIN**: "To operate appropriate components to control a specific parameter to a procedure requirement. This implies a continuing action. "

G2.4.17 Knowledge of EOP terms and definitions.

Apply the rules of usage for the ERP's and (FRPs) (OPS52301B09).

INTRO ERP-52301B09 #9

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

68. G2.4.27 001

A Unit One LOSP has occurred as the result of a fire. No 'A' Train diesels will start. The 2C air compressor is not available. All other systems operated as per design. 'A' Train CCW is the on-service train.
Which one of the following methods SHOULD be used from the MCB to keep pressurizer level from rising rapidly uncontrolled?

A. Close MOV-8107.

B✓ Close MOV-8108.

C. Close FCV-122.

D. Stop B train charging pumps.

AOP-29.1
A. Incorrect; MOV-8107 will have no power available. FU-F3

B. Correct; MOV-8108 should be powered from FV-E3.

C. Incorrect - FCV-122 will have no air to operate with the loss of non-vital power and loss of A-train power: A & C a/c's.

D. Incorrect, AOP-29.1 the only available charging pump must be kept running to maintain seal injection for RCP seal cooling.

G2.4.27 Knowledge of Fire in the Plant procedure.

Evaluate plant conditions to determine if any system components need to be operated while performing AOP-29.0. (OPS52521E06)

AOP-29.0-52521E06 #4

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

69. G2.4.43 001

An Alert has been declared. The Southern LINC ENN was used to try to inform Georgia and Alabama of the Alert. The shift support supervisor has been able to obtain adequate response from Georgia, but <u>NO</u> agency in Alabama has responded. Based on the lack of response from Alabama, the shift support supervisor should:

A. contact the on-call ED for resolution.

B.✓ try to contact appropriate personnel in Alabama using the commercial telephone lines.

C. continue using the ENN to contact Alabama; it is the only authorized method for initial notification.

D. log the attempts made; no further action is required because a reasonable attempt has been made to contact Alabama personnel.

A. Incorrect - the verbal notification Form is clear on what to do and the ED should not be called on to resolve this.

B. Correct - **Try to contact appropriate personnel in Alabama using the commercial telephone lines.**
EIP-9.0 Guidelines verbal notification form
3 PTT and announce on the ENN **"Please prepare to receive a GENERAL EMERGENCY, RED initial notification message with acknowledgment"**, then slowly read the GE initial notification form over the ENN. Release the PTT after reading two or three lines to allow individuals to respond.
4. Have the agencies contacted above, acknowledge receipt of the message and fill in the checkbox on previous page when they do.
<u>5. If any required agency could not be contacted on the ENN, then use numbers listed below or in FNP-0-EIP-8.3 to contact them by any available means as soon as possible.</u>
6. If the display does not show "WIDE AREA, FEP ENN" when group is pressed, press the button with the square until the top line is indicated, then press the arrow buttons until "WIDE AREA" is displayed, then press the button under OK. Press the button with the square until the second line is indicated then press the arrow buttons until "FEP ENN" is displayed, then press the button under OK

C. Incorrect - If the ENN is not working or not being answered then other means are used to contact the agencies.

D. Incorrect - see above #5

G2.4.43 Knowledge of emergency communications systems and techniques.

Explain management's expectations associated with On-Shift Response to plant emergencies.(OPS53002A02)
Describe each on-shift operations staff position responsibilities. (OPS53002A03)

EPIP OVER-40501A02 #3

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

70. W/E01EK1.2 001

Which one of the following describes when ESP-0.0, Rediagnosis, may be used?

A.✓ EEP-1, Loss of Reactor or Secondary Coolant , is in use at step 3 to check intact SG water levels, but the team is unsure of whether to continue in this procedure or transition to EEP-2, Faulted Steam Generator Isolation.

B. ESP-1.3, Transfer to Cold Leg Recirculation, is in use at the step to stop both RHR pumps, but the team is unsure of whether to continue in this procedure or transition to FRP-H.1, Response to Loss of Secondary Heat Sink.

C. EEP-3, Steam Generator Tube Rupture, is in use and the team has placed normal charging on service, but the team is unsure of whether to continue in this procedure or transition to ESP-3.1, Post-SGTR Cooldown using Backfill.

D. FRP-S.1, Response to Nuclear Power Generation/ATWT, is in use and the reactor and turbine trip have been verified, no SI is required, but the team is unsure of whether to continue in this procedure or transition back to EEP-0, Reactor Trip or Safety Injection.

REFERENCE SOP-0.8

A. Correct- **EEP-1, Loss of Reactor or Secondary Coolant , is in use at step 3 to check intact SG water levels, but the team is unsure of whether to continue in this procedure or transition to EEP-2, Faulted Steam Generator Isolation.**
EEP-1 in use at step 3 means the team has exited e-0 and still have a SI in progress. If they are unsure of what to at this point, ESP-0.0 could be used.

B. Incorrect - ESP-1.3 should never be exited to go to another procedure once it has been implemented IAW the notes and rules of usage with ESP-1.3.

C. Incorrect - There is no SI in progress and No SI required so this would be an improper exit strategy.

D. Incorrect - no SI is required, and it is intended for use after E-0 has been exited.

OPS-52301B
There is one unique procedure in the ERP set that has no entry symptoms or transitions. It is used purely as an operator aid and is entered based upon operator judgment. The procedure is ESP-0.0, REDIAGNOSIS. It is intended to be used after departure from EEP-0, but only if SI has been actuated or is required, and a RED or ORANGE path FRP is not in progress. It provides reassurance to the operator that he is in the correct ERP or provides the necessary transition instruction to get to the correct ERP for the existing symptoms. Anticipated operator need for this rediagnosis procedure is limited in the ERPs because many of the transitions needed to respond to new symptoms are included on the foldout pages. However, its presence can be reassuring to an operator after making several consecutive transitions due to rapidly changing conditions.
Once the ERP network has been entered, the user is directed to other ERPs by transition steps. ESP-0.0, REDIAGNOSIS, may be entered at any time after exiting EEP-0, REACTOR TRIP OR SAFETY INJECTION, when a safety injection is in progress or is required and no red or orange path FRP is being implemented. This procedure is entered based on the user's judgment and is designed to help him determine which ERP should be implemented if any confusion develops.

W/EO1 Rediagnosis Knowledge of the operational implications of the following concepts as they apply to the (Rx Trip or SI/Rediagnosis) EK1.2 – Normal, abnormal and emergency operating procedures associated with Reactor Trip or Safety Injection/ Rediagnosis.

Apply the rules of usage for the ERP's and (FRPs) (OPS52301B09).
Given a specific plant condition, state the entry points for using the Emergency Response Procedures (ERPs) (OPS52301B05).

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

71. W/E03G2.4.2 001

The Unit 2 operating crew is currently executing ESP-1.2, Post LOCA Cooldown and Depressurization, and is at the step to verify 2C air compressor in service.
The following conditions exist:

- A dual unit LOSP occurred just prior to the Unit 2 small break LOCA.
- Unit 1 has had no other problems and is currently cooling down IAW the applicable procedures.
- 2C air compressor is tagged out for PMs.

Which one of the following identifies the minimum actions necessary to start the 2A Air Compressor?

A. Verify DH07, 1C DG Output Breaker for Unit 2 closed; then energize 2G 600 V LC from the normal supply.

B. Close DF13, 2F 4160 V bus tie to to 2H 4160 V bus; Verify 2G 600 V LC energized from the normal supply.

C. Verify the SI signal is reset; Verify DH07,1C DG Output Breaker closed; then energize 2G 600 V LC from the normal supply.

D✓ Verify the SI signal is reset; Reset the B2F sequencer; close DF13, 2F 4160 V bus tie to to 2H 4160 V bus; then energize 2G 600 V LC from the normal supply.

Due to the fact that U-2 is in ESP-1.2, Unit 2 had to have an SI. Since U-2 had an SI and a dual Unit LOSP was part of the event, U-2 now has DG 1-2A on the F 4160V bus. U-1 did not have an SI so it would have 1C DG on the H bus. The candidate will have to realize which bus has which DG. The candidate will have to remember that to close DF-13, the sequencer has to be reset and the SI signal has to have been reset also.

A. Incorrect - 1C DG is not aligned to U-2
B. Incorrect - this breaker will not close until the sequencer has been reset and 2G 600 V LC will not be energized when this process is complete.
C. Incorrect - 1C DG is not aligned to U-2
D. Correct - **Verify the SI signal is reset; Reset the B2F sequencer; close DF13, 2F 4160 V bus tie to to 2H 4160 V bus; then energize 2G 600 V LC from the normal supply.**
this is the procedurally correct answer and will allow DF13 to close, and allow the operator to close 2G 600 V LC breakers.

Lesson plan for sequencers OPS-52103F
B1G, B2G, B1F, and B2F Sequencer SIAS Memory Logic
The B1G and B2G sequencer SI memory logic circuits (Figure 30) are identical. If Unit I receives a B train SI, the solid state protection system (SSPS) will start diesel generator 1B, and the B1G ESS sequencer will run. The B1G SI will remain locked in until reset by the operator. The manual reset can only be accomplished at the local B1G sequencer panel by depressing the ESS STOP RESET push button.
If Unit II receives a B train SI, SSPS will start diesel generator 2B, and the B2G ESS sequencer will run. The B2G SI will remain locked in until reset by the operator. The manual reset can only be accomplished at the local B2G sequencer panel by depressing the ESS STOP RESET push button.
The B1F and B2F sequencer SI memory logic circuits operate in conjunction with each other, whereby they determine which unit has the highest priority for emergency electrical power from diesel generator 1-2A.
If Unit I receives an SI, SSPS will start diesel generator 1-2A, and the B1F ESS sequencer will run. Diesel generator 1-2A would only supply Unit I if a subsequent dual-unit LOSP occurred. The memory circuit prevents the Unit II ESS sequencer from running until the Unit I ESS sequencer is manually reset by the operator. The manual reset can only be accomplished at the local sequencer panel by depressing the Unit I ESS sequencer ESS STOP RESET push button.
If Unit II receives an SI, SSPS will start diesel generator 1-2A, and the B2F ESS sequencer will run. Diesel generator 1-2A would only supply Unit II if a subsequent dual-unit LOSP occurred. In this case, the B2F memory circuit prevents the Unit I ESS sequencer from running until the SI on Unit II is manually reset.
The possibility of both units receiving simultaneous SI actuation signals is very remote. If this should occur, the unit that received the SI first would have priority for power from diesel generator 1-2A first. When that unit's memory circuit was reset by the operator, the second unit's memory circuit would remember that an SI had been received. Diesel generator 1-2A would then be available for the second unit, and the associated ESS sequencer for the second unit would run. **The sequencer reset push button should not be depressed unless specifically called for by an applicable operating procedure.**

W/E03 LOCA Cooldown - Depress
G2.4.2 – Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions. NOTE: The issue of setpoints and automatic safety features is not specifically covered in the systems sections.

Evaluate plant conditions to determine if any system components need to be operated while performing ESP-1.2 (OPS52531F06).

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

72. W/E05EK1.3 001

A loss of ALL feedwater has occurred on Unit 1. The team is implementing FRP-H.1, Response to Loss of Secondary Heat Sink, and the following conditions exist:

- No AFW pump can be started.
- Attachment 1, MAIN FEEDWATER BYPASS VALVES AUTOMATIC CLOSURE DEFEAT, has been completed.
- 1A SGFP has just been Latched.
- All SG wide range levels are at 30% and dropping at 1% per minute.

GB5, STM LINE LO PRESS RX TRIP SI, annunciator comes into alarm. Which one of the following is the correct action to be taken at this time?

A. There is no flow path available to the SG's, go immediately to bleed and feed.

B. There is no flow path available to the SG's, continue efforts to get AFW flow available to the SG's.

C. Wait 60 seconds, reset the SI signal, latch the SGFP, and continue efforts to commence feeding the SG's.

D.✓ Wait 60 seconds, reset the SI signal, and continue efforts to commence feeding the SG's with the condensate pumps.

References: FRP-H.1 and OPS-52532F
A. Incorrect - the conditions for bleed and feed are not met and the condensate pumps are still available.

B. Incorrect; there is one flow path available, the condensate pumps. If no other flow path were to exist, then this would be correct.

C. Incorrect; the SGFP will not be available due to the MSIVs going closed. If the MSIVs were open, this would be the preferred method. The step to BLOCK the low steam line press SI is later in the procedure after the feeding of the SGs has started. The Low steam line press SI is still active at this point.
<u>FRP-H.1</u>

7.8 IF SI has NOT actuated since reactor trip, THEN reset FW ISO.	7.8 Verify SI reset. [] MLB-1 1-1 not lit [] MLB-1 11-1 not lit

NOTE: If SI has not actuated since Reactor Trip, defeating the feedwater isolation signal to main feedwater regulating bypass valves will ensure the main feedwater flow path remains open. A subsequent SI will still cause the trip of an operating SGFP.

D. Correct; **Wait 60 seconds, reset the SI signal, and continue efforts to commence feeding the SG's with the condensate pumps.**
Attachment 1 does not prohibit the SGFP from tripping, however, it can be reset. With no steam pressure available to the SGFPs due to the STM LINE LO PRESS RX TRIP SI and subsequent MSLI, the next available method in the procedure is condensate flow.
<u>OPS-52532F</u>
The feedwater isolation signal is defeated using Attachment 1. (Maintenance has already been directed to start this step.) When the only feedwater isolation signal is P-4 and low Tave, the isolation can be reset with the main control board (MCB) push buttons. When an SI has occurred, it must be reset and jumpers installed to defeat the feedwater isolation. The jumpers will defeat all three feedwater isolations: SI, Hi-Hi steam generator level, and P-4 with low Tave. A note associated with this step states that if SI has not actuated since reactor trip, defeating the feedwater isolation signal to main feedwater regulating bypass valves will ensure the main feedwater flow path remains open. Also, <u>a subsequent SI will still cause the trip of an operating SGFP.</u>

W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4
Knowledge of the operational implications of the following concepts as they apply to the Loss of secondary heat sink: EK1.3 – Annunciators and conditions indicating signals, and remedial actions associated with the Loss of Secondary Heat Sink.

Analyze plant conditions to determine if actuation or reset of any engineered safety features actuation signal is necessary (OPS52533F05).
Evaluate plant conditions to determine if any system components need to be operated while performing FRP-H.1/2/3/4/5 (OPS52533F06).
Evaluate plant conditions and determine if transition to another section of ECP-2.1 or to another procedure is required. (OPS52532F08)

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

73. W/E09EA2.1 002

Unit I is being cooled down per ESP-0.2, Natural Circulation Cooldown to Prevent Reactor Vessel Head Steam Voiding. Proper natural circulation conditions have been verified. Natural circulation logs are being taken with present RCS hot leg temperature indicating 495°F. Per earlier steps in the procedure, the crew is operating to maintain the following:

- Cooldown rate less than or equal to 25°F/hr.
- Narrow range steam generator level 61-69%.
- Condensate storage tank level presently at 9 feet.
- Pressurizer level 19-24%.
- RCS pressure reduction and cooldown must continue.
- Only one CRDM fan is available and operating.
- RVLIS indicates 100% UPPER HEAD AND 100% UPPER PLENUM.

Pressurizer level suddenly increases to 55%. Which one of the following describes the appropriate actions?

A. FRP-I.3, Response to Voids in Reactor Vessel and increase the cooldown rate.

B. Remain in ESP-0.2, Natural Circulation Cooldown to Prevent Reactor Vessel Head Steam Voiding and increase the cooldown rate.

C✓ Transition to ESP-0.3, Natural Circulation Cooldown with Allowance for Reactor Vessel Head Steam Voiding (With RVLIS) and increase the cooldown rate.

D. Transition to ESP-0.4, Natural Circulation Cooldown with Allowance for Reactor Vessel Head Steam Voiding (Without RVLIS) and increase the cooldown rate.

References: ESP-0.2/0.3/0.4

A. Incorrect, never go to FRP-I.3 when in ESP-0.2 or .3.

B. Incorrect, RVLIS is Operable but there is a need for an increased cooldown rate due CST level and with RVLIS available, this would be the procedure to go to to increase the cooldown rate. ESP-0.2 will not allow the CDR to be increased with 1 CRDM fan down.

C. Correct. **Transition to ESP- 0.3, Natural Circulation Cooldown with Allowance for Reactor Vessel Head Steam Voiding (With RVLIS) and increase the cooldown rate.**
This is appropriate due to the need to Increase the CDR and voids are forming and pressure reduction must continue.

D. Incorrect, voiding indicated but RVLIS is available.

OPS- 52531C
The objective of ESP-0.3 and 0.4 is to continue plant cooldown and depressurization to cold shutdown conditions that allow for the potential formation of a void in the upper head region due to a need for a rapid cooldown. ESP-0.3 and 0.4 will only be entered from ESP-0.2 and only after the first 10 steps of ESP-0.2 have been completed. Therefore, natural circulation has been verified and cooldown and depressurization initiated per ESP-0.2. These procedures assume no accident is in progress and RVLIS is or is not available for ESP-0.3 and ESP-0.4, respectively.
ESP-0.3 permits a natural circulation cooldown and depressurization of the RCS while allowing, controlling, and monitoring the formation of voids in the vessel. Although there is no analysis to support the techniques employed in ESP-0.3, specific symptoms are relied upon to indicate that void growth is restricted to the upper head/upper plenum region above the top of the hot legs so as not to disrupt the natural circulation flow. By maintaining RCS subcooling, controlling pressurizer level, and monitoring RVLIS indication, a maximum cooldown rate of 100°F/hr and a continued RCS depressurization are permitted. If during the cooldown and depressurization RVLIS indicates that void growth is approaching the hot legs, RCS depressurization is stopped and the operator is instructed to repressurize the RCS to collapse the void.

At no time during the performance of ESP-0.3 or ESP-0.4 is it appropriate to transition to FRP-I.3, Response to Voids (a yellow path procedure) in the reactor vessel and perform a head venting operation. If FRP-I.3 were used to vent the vessel head at this time, the steam void would not be eliminated. As pressure decreased due to venting, more water would flash to steam in the head region. The void size would remain essentially constant, and the net result is a loss of inventory. Therefore, FRP-I.3 should not be used when cooling down and depressurizing the system with ESP-0.3 or ESP-0.4. If FRP-I.3 is entered, a caution at the beginning of the procedure will direct the operator out of FRP-I.3.

W/E09 Natural Circ. Ability to determine and interpret the following as they apply to the NC operations: EA2.1 – Facility conditions and selection of appropriate procedures during abnormal and emergency operations

Evaluate plant conditions and determine if transition to another section of ESP-0.2/0.3/0.4 or to another procedure is required (OPS52531C08).

Analyze plant indications to determine the successful completion of any step in ESP-0.2/0.3/0.4 (OPS52531C07).

ESP-0.2.3.4-52531C08 #4
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C A A B C D C A D Scramble Range: A - D

74. W/E12EK2.2 001

During operation at 100% power, Unit 1 experiences a large steam line break on the common steam header in the MSVR.

- None of the MSIV's will close manually or automatically.
- Access to the MSVR is hazardous due to the steam leak.

The Operators have entered ECP-2.1, Uncontrolled Depressurization of ALL Steam Generators, and have throttled flow to **ALL** SG's to 25 gpm.

Which one of the following describes the actions to be taken if the RCS hot leg temperatures are subsequently increasing?

A. Increase AFW flow to 395 gpm or transition to FRP-H.1, Response to Loss of Secondary Heat Sink.

B. Take no action to prevent RCS heat up. Maintain AFW flow rate of 25 gpm per SG until narrow range level is 31% or greater.

C. Stop all Reactor Coolant Pumps. Increase AFW flow to all SGs not to exceed 100 gpm per SG.

D✓ Increase AFW flow, as necessary, to stabilize RCS hot leg temperatures.

OPS-52532F

A. Incorrect - The procedure in a NOTE above Step 5 says to only transition to H.1 if 395 gpm is not available, NOT due to deliberate operator actions.

B. Incorrect - The procedure has the operator take action if RCS HOT LEG temps increase on TR 413.

C. Incorrect - There is no limit on the AFW flow rate to each SG in ECP-2.1. There is a limit in FRP-H.1 and H.5. Also there is no guidance to secure RCPs in this event except when RCP criteria is not met and HHSI flow is > 0 gpm. It is preferable to have RCPs running to maintain normal pressure control, prevent formation of stagnant water in the upper head region, minimize potential PT shock challenges and to minimize operator actions that would require restarting the RCPs.

D. Correct - **Increase AFW flow, as necessary, to stabilize RCS hot leg temperatures.**
Step 5.3 has the operator stabilize RCS hot leg temperatures by controlling AFW.

W/E12- Steam Line Rupture - Excessive Heat Transfer / 4 Knowledge of the inter relations between the UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS and the following: EK2.2 – Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Evaluate plant conditions to determine if any system components need to be operated while performing ECP-2.1. (OPS52532F06)
State the basis for all cautions, notes, and actions associated with ECP-2.1 (OPS52532F03)

Turkey Point 3 1997
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

75. W/E13EK3.4 001

The Unit 1 operating crew is currently executing EEP-1, Loss of Reactor or Secondary Coolant.
Plant conditions are as follows:

- Containment pressure is 4.5 psig and slowly decreasing.
- RCS temperature is 585°F.

- SG Pressures:

1A SG - 1200 psig; 1B SG - 1190 psig; 1C SG- 1250 psig

- SG NR Levels:

1A SG - 25%; 1B SG - 30%; 1C SG = 98%

FRP-H.2, Response to Steam Generator Overpressure, has been entered. For the existing plant conditions, the team should:

A✓ NOT release steam from 1C SG to prevent equipment or piping damage.

B. NOT release steam from any SG to prevent excessive cooldown of the RCS.

C. open 1C Atmospheric to drop pressure below 1070 psig to reseat the code safeties.

D. use the TDAFW pump from 1C SG to drop pressure below 1130 psig to gain control of AFW flow to all SGs.

A. Correct - **NOT release steam from 1C SG to prevent equipment or piping damage.**
In this case with SGWL > 91% on 1C SG, steam should not be released from that SG until an overfill evaluation has been performed.

B. Incorrect - While the team would exit H.2 to go to H.3 for 1C SG, the other two SGs would have the pressure in them reduced IAW H.2 when H.3 was exited. H.3 would isolate flow to 1C SG and align blowdown to reduce level, then transition back to H.2 to reduce pressure in the other two SGs.

C. Incorrect - This could cause damage to that SG and has nothing to do with cooldown limits of the RCS.

D. Incorrect - This is another path to reduce SG pressure in 1C SG if level was < 91%, incorrect reason for aligning the TDAFWP.

FRP-H.2 Step 3. **Check affected SG(s) narrow range level - LESS THAN 91%{76%}**
For this condition RNO Step 3 directs the operator to Go to FNP-1-FRP-H.3, RESPONSE TO STEAM GENERATOR HIGH LEVEL.

CAUTION : To prevent equipment or piping damage, steam should not be released from a SG whose narrow range level is greater than 91%{76%}.

FRP-H.3 Caution prior to step 1
CAUTION : Steam should not be released from any SG whose narrow range level has exceeded 91%{76%} until an overfill evaluation has been performed. Blocking main steam line hangers and draining affected steam lines should be considered.

W/E13 Steam Generator Over-pressure / 4 Knowledge for the reasons for the following responses as they apply to the SG overpressure: EK3.4 – RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

State the basis for all cautions, notes, and actions associated with FRP-H.1/2/3/4/5 (OPS52533F03).
Describe the sequence of major actions associated with FRP-H.1/2/3/4/5 (OPS52533F04).
Evaluate plant conditions to determine if any system components need to be operated while performing FRP-H.1/2/3/4/5 (OPS52533F06).
Evaluate plant conditions and determine if transition to another section of FRP H.1/2/3/4/5 or to another procedure is required (OPS52533F08).

FRP-H-52533F08
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

1. 003G2.1.11 001

Unit 1 is at 85% power, ramping to 95% power. During rod movement to adjust Tavg, the following indications are received:

- FE3, ROD AT BOTTOM, annunciator is in alarm.
- Control Bank D, rod F-6, DRPI rod bottom light is lit.
- Reactor power dropped and returned to the initial value.
- Tavg dropped and remained lower than the initial value.

After the Team adjusts Turbine load to restore RCS Tavg, the following plant conditions exist:

- Delta I is (-14%).
- Bank D is at 176 steps.
- Reactor power is 80%.

Which one of the following satisfies Technical Specifications for this event and gives the reason?

REFERENCES PROVIDED

A. Reduce power to 74% in the next 30 minutes so that the Heat Flux Hot Channel Factor is not exceeded.

B. Reduce power to 48% in the next 30 minutes so that the Heat Flux Hot Channel Factor is not exceeded.

C✓ Restore the dropped rod to 165 steps in the next hour so that Local Xenon Redistribution will not be significant.

D. Restore the dropped rod to 148 steps in the next hour so that Local Xenon Redistribution will not be significant.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question.

REFERENCE PROVIDED: Axial Flux difference Limits as a Function of Rated Thermal Power for RAOC figure 3 (page 12 OF 13 of COLR) and Rod Bank Insertion Limits Versus Rated Thermal power Figure 1 (page 10 OF 13 of COLR).

A - Incorrect; The ramp would be correct for 3.1.4 REQUIRED ACTION B.2, due to not meeting rod restoration time limits. While this may be correct for the second part of the TS, by itself it will not satisfy the TS action statement that must be accomplished: REQUIRED ACTION A. Also, the reason is not correct for the time requirement for TS 3.1.4. This is the basis for the limits on AFD.

B - Incorrect; Required power and time would be correct at 100% power, but the limit is already met without any ramp at 80%. Reducing power to <50% is TS 3.2.3 30 min. time requirement to ramp for AFD being outside the limits of the COLR, (doghouse). In this case the delta I is inside the doghouse but outside the normal operating band. Not the correct TS for this situation. Also, the reason is not correct for the time requirement for TS 3.2.3. This is the basis for the limits on AFD.

C - Correct; **Restore the rod to 165 steps in the next hour so that Local Xenon Redistribution will not be significant.**
IAW TS 3.1.4, Rod group alignment limits & BASIS.

LCO 3.1.4 All shutdown and control rods shall be OPERABLE, with all individual indicated rod positions within 12 steps of their group step counter demand position.

B. One rod not within alignment limits. B.1 Restore rod to within alignment limits. 1 hour

Bases:
B.1
When a rod becomes misaligned, it can usually be moved and is still trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

D - Incorrect; 148 steps is the limit for Rod insertion Limit per TS 3.1.6. The time limit to restore control banks w/i limits is 2 hours, so the answer meets the time requirement for that TS however, this would not satisfy the Rod Alignment TS 3.1.4. The 148 steps comes from the Figure 1 limit in the COLR of 148 steps for 80% power. This is the bases for LCO 3.1.4.

000003 Dropped Control Rod / 1
SRO –G2.1.11 – Knowledge of less than one hour technical specification action statements for systems.

Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Rod Control System (OPS52201E01).

- 3.1.4, Rod Group Alignment Limits
- 3.1.5, Shutdown Bank Insertion Limits
- 3.1.6, Control Bank Insertion Limits
- 3.1.7, Rod Position Indication
- 3.1.8, PHYSICS TESTS Exceptions

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

2. 005A2.03 001

The following plant conditions exist:

- Unit 1 is in Mode 5.
- The Pressurizer manway is removed.
- RCS loops are not filled.
- RCS level is at the flange.
- SG nozzle dams are installed.

1B RHR pump has been declared INOPERABLE. The operating crew immediately starts the 1A RHR pump and it trips on overload. In addition to initiating action to restore 1B RHR pump to operable status and placing it on service, which one of the following must also be done IMMEDIATELY and the reason for this action?

A. Align a boration flow path to maintain stable reactivity conditions.

B. Verify locked closed the unborated water source valves to preclude any power escalation.

C. Ensure that at least two SGs have a WR level >75% to ensure an alternate heat removal path.

D✓ Suspend all operations involving reduction in RCS boron concentration to ensure the margin to criticality is not reduced.

Meets 10 CFR 55.43 (b) 5 and 2 requirements for SRO level question.

A. Incorrect - A step in AOP 12 has the operator align a TRM boration flow path if level is < 123'3" to raise level. The reason was taken from the TRM 13.1 bases for an operable boration flow path.

12 Check two independent RCS level indications - GREATER THAN 123 ft 3 in.	12 Raise RCS level. 12.1 Notify personnel in containment that RCS level will be raised. 12.2 Align Technical Specification {Technical Requirement Manual} boration flow path. 12.3 Raise RCS level to greater than 123 ft 3 in.

B. Incorrect - This is the correct reason for a loss of source range channels inoperable.

C. Incorrect - The SGs are not available in this case. AOP-12 has the operator use the SGs if manways are installed, SG secondary handhole covers are installed and nozzle dams are removed. The reason for SGs being used is covered in a NOTE above the step in AOP-12 that says a secondary heat sink will reduce RCS heat up and pressurization rate to provide more time for recovery actions.

D. Correct - **Suspend all operations involving reduction in RCS boron concentration to ensure the margin to criticality is not reduced.**
This is the correct reason from Bases and is listed in TS 3.4.8. It is also a step in AOP-12.0. **10 Suspend any boron dilution in progress. (IN 91-54)**

005 Residual Heat Removal A2 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: SRO –A2.03 – RHR pump / motor malfunction

1. Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Residual Heat Removal System (OPS52101K01).
- 3.4.3, RCS Pressure and Temperature (P/T) Limits
- 3.4.6, RCS Loops – MODE 4
- 3.4.7, RCS Loops - MODE 5, Loops Filled
- 3.4.8, RCS Loops - MODE 5, Loops Not Filled

RHR-52101K01
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

3. 006G2.2.25 001

During a normal surveillance, the following RWST parameters are determined:

- Volume = 481,000 gallons
- Temperature = 45°F
- Boron concentration = 2200 ppm

Corrective action must be taken because:

A. Boric acid may crystalize and partially block ECCS flow in the core.

B✓ The reactor Shutdown Margin may be less than analyzed following an accident.

C. Insufficient cooling capacity is available for both ECCS and containment spray recirculation flow.

D. Excessive caustic stress corrosion to the extent that environmentally qualified equipment and instrumentation are damaged.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question.

From TS 3.5.4 Bases:
This LCO ensures that:
a. The RWST contains sufficient borated water to support the ECCS during the injection phase;
b. **Sufficient water volume exists in the containment sump to support continued operation of the ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling**;
and
c. **The reactor remains subcritical following a LOCA**. **Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.**

A. Incorrect; RWST temperature is satisfactory in the condition given. see below.
In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of ≥35°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature.

B. Correct; **The reactor Shutdown Margin may be less than analyzed following an accident.**
The low limit of 2300 ppm is not met in the condition given. See above.

C. Incorrect; The low limit of RWST Volume is met. See above

D. Incorrect; This would be correct if the RWST Boron concentration was too high, but it is satisfactory in the condition given.

006 Emergency Core Cooling
SRO-G2.2.25 – Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Emergency Core Cooling System (OPS52102B01).
• 3.5.1 Accumulators
• 3.5.2 ECCS—Operating
• 3.5.3 ECCS—Shutdown
• 3.5.4 Refueling Water Storage Tank (RWST)
• 2.1.1 Reactor Core Safety Limits

Modified Bank Question: ECCS-52102B01 53

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

4. 007G2.4.4 001

Unit 1 is in Mode 3 with the following conditions:

- The shutdown banks are withdrawn.
- A cooldown IAW Technical Specification 3.4.10 due to an inoperable PRZR code safety valve is in progress.

A complete Loss of Site Power occurs. All emergency equipment operate normally. The operators stabilize the plant per ESP-0.1, REACTOR TRIP RESPONSE. The switchboard operator reports it will be 12 hours before off-site power will be restored.

What is the correct course of action?

A.✓ Transition to ESP-0.2, Natural Circulation Cooldown to Prevent Reactor Vessel Head Steam Voiding.

B. Transition to ESP-0.3, Natural Circulation Cooldown with Allowance for Reactor Vessel Head Steam Voiding.

C. Transition to UOP-2.1, Shutdown of Unit from Minimum Load to Hot Standby, start a RCP as soon as possible, then transition to UOP-2.2, Shutdown of Unit from Hot Standby to Cold Shutdown.

D. Maintain the plant in a stable condition per ESP-0.1, Reactor Trip Response until a RCP is started. Then transition to UOP-2.1, Shutdown of Unit from Minimum Load to Hot Standby.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

ESP-0.1
Cooldown to ≤325°F in 12h 15m is required per TS. 3.4.10. For this reason, waiting 12 hours to start a RCP prior to starting a cooldown is not an option. A natural circ cooldown is required. The last step of ESP-0.1 directs a cooldown per ESP-0.2 IF no RCPs are runninng, AND Cooldown Required which is the case. Exceeding the limits (≤ 25°F/ HR) of ESP-0.2 is not required (cooldown from 547°F to 300°F @ 21°F/hr would take 11.8 hours), therefore, ESP-0.3 does not have to be entered.

A. Correct. **Transition to ESP-0.2, Natural Circulation Cooldown to Prevent Reactor Vessel Head Steam Voiding.**
ESP-0.1 will be completed and the last step directs to ESP-0.2. A natural circ cooldown will be conducted to comply with the tech spec action statement.

B. Incorrect. ESP-0.3 is never entered until after the first 10 steps of ESP-0.2 are complete. Then, it is only entered if the cooldown limits of ESP-0.2 must be exceeded (25°F/min) which is not the case.

C. Incorrect. ESP-0.1 will be completed, and would direct to UOP-2.1 ONLY if a RCP was running.

D. Incorrect. Stopping in ESP-0.1 instead of continuing on in the procedure violates the rules of procedure usage per FNP-0-SOP-0.8. Correct procedure useage dictates proceeding in ESP-0.1, THEN transitioning either to UOP-2.1 (only if any RCP running) OR ESP-0.2 (if no RCP running).

000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1
SRO –G2.4.4 – Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

2. Evaluate plant conditions to determine if entry into ESP-0.1 is required. (OPS52531B02)

Modified two distracters and the time in the stem (changed 8 hours to 12hours).
FNP Bank question ESP-0.1-52531B08 01

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

5. 011EA2.01 001

Given the following Plant Conditions:

- Unit 1 has experienced a LOCA.
- SI actuation occurred and all systems responded as per design.
- The immediate operator actions have just been completed.
- Containment pressure is at 5 psig and rising.
- FI-943, HHSI flow, reads 500 gpm.
- RCS subcooling is 22° F and dropping.

Which one of the following describes the correct operator response to this situation?

A. Trip all running RCPs to aid the mitigation of the accident by adding less pump heat to the core.

B. Do not trip the running RCPs to aid the mitigation of the accident by maintaining forced cooling to the core.

C.✓ Trip all running RCPs because tripping them will have little or no adverse effect on the mitigation of the accident.

D. Do not trip the running RCPs unless directed to do so later in the procedure because support conditions are met.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

Ref: EEP-0, ERG Background document, Executive Volume, Generic Issues, RCP Trip/Restart criteria

Foldout page criteria

1 Monitor RCP criteria.

1.1	Greater than 16 °F{45 °F} subcooled in CETC mode.	1.1	IF HHSI flow greater than 0 gpm, THEN stop all RCPs.

A- Incorrect; RCP's are tripped by fold out page criteria. Pump heat added to the core in a Large Break LOCA is negligible. Both HHSI, Accumulators, & LHSI flow is maximized to cool the core, and there is no coolant in the loops with which the RCPs will produce and transfer pump heat to the core.

B - Incorrect; RCP's are tripped by fold out page criteria. Forced RCP cooling is maintained without support conditions in other accident contingencies such as severely degraded core cooling. In this condition, SI flow provides all necessary cooling to the core.

C - Correct; **Trip all running RCPs because they have lost support conditions and tripping them will have little or no effect on the mitigation of the accident.** EEP-0 fold out page criteria has been met. Under Adverse containment conditions subcooling $\leq$16{45}, RCP trip criteria is met. From ERG background document: "For large break LOCAs, the operation of the RCPs has little if any effect during mitigation and recovery".

D - Incorrect; RCP's are tripped by fold out page criteria. The Fold Out Page subcooling criteria combined with SI Flow present directs securing RCPs. In a LBLOCA, 27# in containment is soon met and the RCPs are again directed by procedure to be secured because of loss of support conditions.

000011 Large Break LOCA / 3
EA2 Ability to determine or interpret the following as they apply to a Large Break LOCA:
SRO –EA2.01 – Actions to be taken, based on RCS temperature and pressure saturated and superheated.

9. Using appropriate ERPs, recognize the symptoms and perform the procedural guidance to
mitigate the consequences of the conditions to include (OPS52504A09):
a. Reactor trip (w/wo stuck rod)
b. Safety injection (steam break, SGTR, LOCA, or Spurious)
c. Anticipated transient without trip (ATWT)
d. Reactor trip recovery
e. SI termination
f. Loss of secondary heat sink
g. Loss of forced flow

Modified Bank Question: E-0/ESP-0.0-52530A06 #2

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

6. 016A2.02 001

Unit 1 is at 100% power. I & C personnel are performing a routine STP on TE-144, Letdown Heat Exchanger outlet Temperature Element. They discover a loose electrical connection on the RTD and during investigation power is lost to the RTD.

The OATC reports that TI-144, Letdown Heat Exchanger Outlet Temperature, is indicating off scale low.

The consequences of power lost to the RTD is an RCS ____1____ and the Shift Supervisor needs to direct the OATC to _____2_____.

A. 1. Dilution 2. secure letdown to slow the dilution.

B✓ 1. Boration 2. use MCB TK-144 in Manual Control and increase CCW flow.

C. 1. Dilution 2. use MCB TK-144 in Manual Control and decrease CCW flow.

D. 1. Boration 2. direct a systems operator to take local control of TCV-3083, LTDN HX CCW TEMP CONTROLLER, and raise Letdown Temperature.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

From SOP-23.0:
CAUTION: CCW temperature should be maintained as stable as possible due to the effects on reactivity due to changes in letdown temperature. Also, changing CCW temperature could affect RCP oil levels which could cause level annunciators to come in.

The Temperature input to the Letdown Temperature controller, TK-144, failed low. The controller senses a lower temperature and sends a signal to the CCW valve to close down to provide less cooling to raise the temperature of Letdown. When Letdown temperature goes up, the demineralizers have less affinity for boron, and some of the boron in the demineralizers is released. This is a boration effect.

Manual control of the controller is prefered, and a small band should be given to the operator to prevent reactivity excursions.

ARP'S DF5 & DF1 both direct taking manual control of TK-144 when needed to control temperature. They also direct securing Letdown only if necessary.

A. Incorrect. a dilution will not occur, the securing of letdown is an option but only if necessary.

B. Correct. **1. Boration 2. use MCB TK-144 in Manual Control and increase CCW flow.**

Due to the TE failing low the TCV will close down to increase CCW temp. The increase in CCW temp will cause a boration and manual control of TCV will have to be accomplished in order to decrease CCW temp and stabilize temp.

C. Incorrect. a dilution will not occur, if it did then this would be the correct answer

D. Incorrect. There is NEW procedural guidance to take manual control of the CCW valve. There is also guidance to set up and control the bypass around TCV-3083 but there are briefs required and it is a coordinated effort. Also the operator would lower letdown temperature, not raise it.

016 Non-nuclear Instrumentation
A2 Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS;and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
SRO –A2.02 – Loss of power supply.

8. Predict and explain the following instrument/equipment response expected when performing Component Cooling Water System evolutions including the fail condition, alarms, and trip setpoints as applicable (OPS40204A08).
- Surge Tank Level
- CCW Heat Exchanger Outlet Temperature (TI-3042)
- Letdown Temperature Out of Letdown Heat Exchanger (TI-144)
- CCW Heat Exchanger CCW Flow (FI-3043)
- Radiation Monitors R-17A and B

2. Evaluate abnormal plant or equipment conditions associated with the Component Cooling Water System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52102G02).

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

7. 022G2.4.10 001

Given the following Plant Conditions conditions:

- Unit 1 is at 100% power.
- A Train is on service.
- Two Letdown Orifices are on service.
- 1A Charging Pump is running.

The following alarms come in:

- DD1, RCP SEAL INJ FLOW LO
- DE1, REGEN HX LTDN FLOW DISCH TEMP HI
- EA2, CHG HDR FLOW HI-LO
- EB1, CHG PUMP OVERLOAD TRIP

Which one of the following is correct?

A. Start 1A CCW pump and 1C Charging pump. No Charging system surveillance is required since both trains of charging are operable.

B. If seal injection cannot be restored immediately, trip the reactor and secure all RCP's to protect the RCP Seals.

C✓ Start 1B Charging Pump. Perform surveillance to verify A Train Charging Discharge Isolations are open with power removed to ensure a safety injection flow path is maintained.

D. Increase Charging flow by opening FCV-122 in manual to lower Letdown Regenerative Heat Exchanger temperature until DE1, REGEN HX LTDN FLOW DISCH TEMP HI, is clear to protect the letdown regenerative heat exchanger.

This question meets 10CFR 55.43 (b) 2 for SRO level questions.

A Incorrect. Starting the 1B charging pump is directed by the ARP. Management expectations dictate starting a piece of equipment that should have auto started but did not even if no procedure directs it. IF B Charging pump did not start as directed by the ARP, starting the other train CCW and charging pump would be appropriate. B charging pump SHOULD have started, and the surveillance IS required because A Charging Pump is inoperable (even though both trains are operable).

*B Incorrect. Per DD1, RCP SEAL INJ FLOW LO, ARP: this would be correct if CCW to the thermal barrier was lost in conjunction with the loss of Seal Injection. As long as CCW to the thermal barrier is available, RCP Seal injection is not required for the short term. Loss of seal injection is a long term concern due to unfiltered water flowing through the seals can degrade them over time.

*C Correct. **Start 1B Charging Pump. Perform surveillance to verify A Train Charging Discharge Isolations are open to ensure a safety injection flow path is maintained.**
Per EB1, CHG PUMP OVERLOAD TRIP, ARP Automatic Action: B should auto start if A trips and is aligned to the A train. A train is on service, so B charging pump is by convention aligned to the A train. ARP directs verifying automatic action and if necessary take manual actions. Management expectations is: attempt to manually start any piece of equipment that should have auto started but did not. Tech Spec 3.5.2 SR 3.5.2.1 requires the 8132A & 8132B MOVs open to ensure an SI flowpath whenever the A Charging pump is inoperable.

*D Incorrect. Per DE1, REGEN HX LTDN FLOW DISCH TEMP HI, ARP: Charging or Letdown flow is adjusted to lower the Letdown flow temperature to below 380° F. In this situation, there is no charging pump running as indicated by the alarms that are in. Adjusting FK-122 open alone will not supply any more charging flow. When a charging pump is running, FCV-122 will supply charging flow even in auto. Manual is not necessary or required.

000022 Loss of Rx Coolant Makeup / 2
SRO –G2.4.10 – Knowledge of annunciator response procedures.

2. Evaluate abnormal plant or equipment conditions associated with the Chemical and Volume Control System and determine the integrated plant actions needed to mitigate the
consequence of the abnormality (OPS52101F02).

MCS	Time:	1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
					Answer:	C C C C C C C C C C	Items Not Scrambled

8. 026A2.08 001

Given the following conditions on Unit 2:

- The plant was at 100% power when the 2A S/G Main Steam line ruptured inside containment.
- All systems actuated as per design.
- Containment pressure spiked to 33 psig and is now continuing to decrease slowly.
- The crew has entered ESP-1.1, SI Termination.

Which one of the following provides both the correct procedure and plant parameters which allow for securing the Containment Spray (CS) system?

A. ESP-1.1, SI Termination. Containment pressure is 12 psig and the RWST level is 10.5 feet and continuing to decrease.

B✓ ESP-1.3, Transfer to Cold Leg Recirc. Containment pressure is 15 psig and CS has been aligned for recirculation flow for 10 hours.

C. ESP-1.2, Post LOCA Cooldown and Depressurization. Containment pressure is 12 psig and CS has been aligned for recirculation flow for 7.5 hours.

D. EEP-1.0, Loss of Reactor or Secondary Coolant. Containment pressure is 15 psig, RWST level is <5.5 feet and it has been 8 hours since the initiation of the event.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

The requirement to secure CS is found in EEP-1, ESP-1.3 AND ESP-1.1 (WHEN, THEN statements). It is defined as "WHEN CS recirculation flow has been aligned for at least 8 hours and ctmt pressure is less than 16 psig THEN stop both CS pumps."

A - Incorrect, ESP-1.1 does provide guidance, but ESP-1.3 would be in progress at 4.5 ft in the RWST to commence placing the CS on recirc per the foldout page. In ESP-1.3, at 4.5 ft, IF transfer to recirc is not imminent, THEN the CS pumps are secured. This guidance is not in ESP-1.1. It takes about 1.5 hours to lower the RWST to 4.5 ft with both spray pumps running with full flow: 2 x 2600 gpm/(60min/hr)= ~1.5hrs. This is when the CS is initially placed on recirc. The requirement of 8 hours on recirc could not have been met.

B - Correct, **ESP-1.3, TRANSFER TO COLD LEG RECIRC. CS has been aligned for recirculation flow for 10 hours and containment pressure is 15 psig.**
ESP-1.3, does provide guidance; Containment pressure is <16# and the time on recirc is > 8 hours.

C - Incorrect, ESP-1.2 does not have guidance, and would not be entered for this event unless there was a LOCA along with the Steam Break. The 7.5 hours of operation applys to HHSI/LHSI transferring from Cold Leg recirc to Simultaneous hot/cold leg recirc, but is not long enough to meet the 8 hour minimum for operation of CS on recirc prior to securing CS.

D - Incorrect, EEP-1 does provide guidance, but 8 hours from the initiation of the event is not long enough to have 8 hours of CS recirc time elapsed.

026 Containment Spray
A2 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:
SRO –A2.08 – Safe securing of containment spray (when it can be done).

2. Evaluate abnormal plant or equipment conditions associated with the Containment Spray
and Cooling System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS50102C02).

Modified from Bank Questions:
EEP-1-52530B07 #10 Source: Farley NRC Exam 2000-301 2001 nrc exam
EEP-1-52530B07 #11
ESP-1.3/.4-52531G06 #4
original EEP-1-52530B07 #10:

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

9. 028G.2.1.12 001

With Unit 1 at 100% power, the following events have occurred:

- On April 2, at 0515, the 1A Post Accident Hydrogen Recombiner was declared INOPERABLE.
- On April 5, at 0200, the 1B Post Accident Hydrogen Recombiner was declared INOPERABLE.

Which one of the following meets the required action and completion time of Technical Specifications?

Reference provided

A. Verify the Post Accident Vent system is available within 1 hour; Restore Both Recombiners to OPERABLE status by April 30.

B. Verify Reactor Cavity Hydrogen Dilution Fans are available within 1 hour; Restore Both Recombiners to OPERABLE status by April 9.

C✓ Verify the Post Accident Vent system is available within 1 hour; Restore one Recombiner to OPERABLE status by April 11 <u>and</u> the other Recombiner by April 30.

D. Verify Reactor Cavity Hydrogen Dilution Fans are available within 1 hour; Restore at least one Recombiner to OPERABLE status by April 10 <u>or</u> be in MODE 3 by April 30.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question.

TECHNICAL SPECIFICATION 3.6.7 PROVIDED

A. Incorrect. This would meet the 30 day completion time and required actions for one Recombiner inoperable Per condition A., but subsequent to the first Recombiner becoming inoperable, a second inoperable Recombiner requires entering Action B. One recombiner has to be operable by April 12.

B. Incorrect. This would meet only the B.2 Required action and completion time, but not the B. 1. Reactor Cavity Hydrogen Dilution Fans are not correct.

C. Correct. **Verify the Post Accident Vent system is available within 1 hour; Restore one Recombiner to OPERABLE status by April 11 and the other Recombiner by April 30.**
On April 2, the condition A is entered. On April 5, condition B is entered. Administrative verification that "the Post Accident Vent System is operable" is required within one hour (see TS Basis for definition of "hydrogen control function"=Post Accident Containment Vent System). One recombiner has to be back in 7 days, 7+5 = April 12. if not then SD to mode 3. The other has to be back in 30 days from the time one was declared INOPERABLE, May 2.

D. Incorrect. Reactor Cavity Hydrogen Dilution Fans operable do not meet the B.1 required action. If one recombiner is not operable by April 12. If not then in 6 hours on April 12 be in mode 3.

028 Hydrogen Recombiner and Purge Control
SRO –G2.1.12 – Ability to apply technical specifications for a system.

1. Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Post LOCA Atmospheric Control System (OPS52102D01).
- 3.3.3 Post Accident Monitoring (PAM) Instrumentation
- 3.4.15 RCS Leakage Detection Instrumentation
- 3.6.7 Hydrogen Recombiner
- 3.6.8 Hydrogen Mixing System
- 3.6.9 Reactor Cavity Hydrogen Dilution System
- 13.3.5 Electric Hydrogen Recombiners Instrumentation and Control Circuits

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

10. 033G2.2.22 001

Given the following Plant Conditions on Unit One:

- A Reactor startup is in progress.
- Reactor Power is stable and critical at 5x10-8 amps.
- Intermediate Range (IR) channel, N-36 fails low.

Which one of the following statements is correct?

Reference provided

A✓ Power may be raised to 100% power in this condition. Diverse Nuclear Instrument trips protect the reactor without N-36.

B. Reactor Shutdown is required due to N-36 failed. The IR NIs are the only protection against a Continuous Rod Withdrawal accident at this power.

C. Mode 1 entry is not allowed until N-36 is repaired and declared operable. Diverse Nuclear Instrument trips allow operation at this power level as long as one IR NI is operable.

D. Power must be lowered to the source range. Operation at a higher power level is not allowed in this condition. SR NI trips are the only other trips which protect the reactor from reactivity addition accidents during a startup.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question.

TECHNICAL SPECIFICATION 3.3.1 PROVIDED

Tech Spec 3.3.1 & Bases

A. Correct. **Power may be raised to 100% power in this condition. Diverse Nuclear Instrument trips protect the reactor without N-36.**

Per TS 3.3.1, with a loss of one IR CHANNEL >P-6 & ,<P-10, power must be raised above P-10 or lowered below P-6 in two hours. SR, IR, & PR lo setpoint hi flux NI trips all protect the reactor from positive reactivity accidents subcritical.

B. Incorrect. Reactor shutdown would be required if BOTH IR NIs were lost. Also, the PR NIs protect against a CRW Accident at this power.

C. Incorrect. Mode one entry is allowed as long as power is above P-10 or less than P-6 in two hours. Operation at this power level is not allowed longer than two hours.

D. Incorrect. Operation at a higher power IS allowed. Source Range NI trips are NOT the only NI trips other than the Intermediate Range NI trips which protect the reactor from reactivity addition accidents during a startup.

000033 Loss of Intermediate Range NI / 7
SRO –G2.2.22 – Knowledge of limiting conditions for operations and safety limits.

7. Identify and apply the following Technical Specifications or TRM requirements, including
the bases and attendant equipment, associated with the Excore Nuclear Instrumentation
System (OPS52201D10).
·3.2.3 Axial Flux Difference (AFD)
·3.2.4 Quadrant Power Tilt Ratio (QPTR)
·3.3.1 Reactor Trip System Instrumentation
·3.3.4 Reactor Shutdown System Instrumentation and Controls
·3.9.2 Refueling Operations Instrumentation

Modified FNP Bank question: EXCORE-52201D10 66

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

11. 036AA2.02 001

You are the Shift Supervisor (SS).

- Unit 1 is in Mode 6, Refueling.
- Core reload is occurring in containment and fuel movement is in progress.

The SRO in charge of fuel handling reports to you that the fuel assembly has impacted the seal ring at the hold down clamp.

- Annunciator EH2, "SFP LVL HI-LO," has just alarmed.
- The Refueling Cavity watch reports that the refueling cavity level is lowering rapidly.

Which one of the following describes the initial action in accordance with AOP-30.0, "REFUELING ACCIDENT?"

A. Ensure the SRO in charge of fuel handling evacuates all personnel from Containment and the Spent Fuel Pool room.

B✓ Ensure the SRO in charge of fuel handling places any fuel assembly in transit in a safe location.

C. Initiate action to place the Control Room Emergency Filtration/Pressurization System (CREFS) in service.

D. Restore the Reactor Internals to the reactor vessel.

Meets 10 CFR 55.43 (b) 6 requirements for SRO level question.

A - Incorrect, Only non-essential personnel need to be evacuated the refueling crew is needed to safely store the fuel assembly in transit.

B - Correct, **Ensure the SRO in charge of fuel handling places any fuel assembly in transit in a safe location.**

Initial response is to secure the fuel assembly in transit, worst case the SRO has 164 minutes to safely store the fuel assembly. Makeup flow capacity from one train of RHR is sufficient to maintain refueling level above the level where uncovery of the fuel assembly would occur.

C - Incorrect, This is the ATTACHMENT 1 actions of AOP-30 which is done in parallel with the remainder of AOP-30 after the fuel assembly is directed to be secured.

D - Incorrect, This can not be done due to the fuel assembly being in transit.

Step 3 IF level is falling, THEN place internals back in the vessel, or at the lowest point in the refueling cavity.
Since fuel movement is in progress, the internals would have to be on the stand in the cavity. Procedures do not permit the internals to be moved with fuel in transit.

000036 Fuel Handling Accident / 8
AA2 Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: SRO –AA2.02 – Occurrence of a fuel handling incident.

3. Determine the appropriate action for given conditions during performance of UOP-4.1, [CONTROLLING PROCEDURE FOR REFUELING] (OPS40503B03).

Bank Question: AOP-30.0-52521H04
2001 nrc exam
2003 nrc exam

MCS	Time:	1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
					Answer:	B B B B B B B B B B	Items Not Scrambled

12. 040G2.1.20 001

Given the following conditions on Unit 1:

- 1B MDAFW Pump Tagged Out.
- Unit 1 Safety Injection has occurred.
- Dual Unit LOSP has occurred.
- 1-2A DG Tripped on Low Lube Oil Pressure.

EEP-2, FAULTED STEAM GENERATOR ISOLATION, is now in progress.

- All MSIVs are closed.
- S/G Pressures are:
  - 1A 950 psig & stable
  - 1B 605 psig & decreasing
  - 1C 556 psig & decreasing

At the step: "Isolate all faulted SGs", with regard to the TDAFW Pump isolation, which one of the following is correct?

A. Isolate steam supply from only the 1B Main Steam line to the TDAFW Pump.

B✓ Isolate steam supply from only the 1C Main Steam line to the TDAFW Pump.

C. Isolate steam supplies from the 1B & 1C Main Steam lines to the TDAFW Pump.

D. Do NOT isolate steam supply from either the 1B or 1C Main Steam lines to the TDAFW Pump.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

EEP-2:
On a Dual Unit LOSP with a Unit One Safety injection, 1/2A DG supplies power to 1F 4160V Bus. 1C DG supplies Unit Two A train bus. When 1/2A DG trips, Unit One has no A Train 4160V ESF Power, and A MDAFW Pump is deenergized. B MDAFW Pump is T/O. The TDAFW Pump is the only pump available to supply AFW flow for a Heat Sink.

Note prior to step 1 states: "If more than one SG appears faulted, the SG with the lowest pressure should be isolated first".

Note prior to step 4.4 states: "At least one steam supply should remain aligned if TDAFWP required".

WOG ERG Background document for the TDAFW Steam supply isolation in EEP-2.0 states: "If the TDAFW Pump is the only source of feed flow to the steam generators (I.E., no other MD AFW Pumps or other operable pumps are capable of providing feed flow to the SGs), then isolation of its steam supply line may degrade system conditions and result in a transition to FRP-H.1. Therefore, this isolation must not be performed".

A. Incorrect. B Main steam line is NOT the lowest pressure of any faulted SG, therefore C would be isolated first. B would not be isolated due to the note requiring at least one Steam supply unisolated when the TDAFW Pump is the only AFW supply available.

B. Correct. **Isolate steam supply from only the 1C Main Steam line to the TDAFW Pump.**
The TDAFW Pump is the only AFW Supply available. Isolating the lowest pressure SG (C) steam supply to the TDAFW Pump is directed by EEP-2. B Steam supply to the TDAFW pump is NOT isolated even though the B SG is faulted to maintain a SG Heat Sink and prevent entering FRP-H.1.

C. Incorrect. This would be true if either MDAFW Pump was available to provide >395 gpm flow. Since there is no other source of AFW than the TDAFW Pump, one steam supply is maintained unisolated even though both B & C SGs are faulted.

D. Incorrect. Only one steam supply is required to operate the TDAFW Pump at full capacity. Per EEP-2, the lowest pressure SG Steam supply to the TDAFW pump is isolated first. The other is left unisolated to supply the only source of AFW to maintain a heat sink and prevent transition to FRP-H.1.

000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture - Excessive Heat Transfer / 4
SRO –G2.1.20 – Ability to execute procedure steps.

7. Analyze plant indications to determine the successful completion of any step in EEP-2 (OPS52530C07).

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

13. 056AA2.09 001

Given the following conditions on Unit 1:

- CTMT CLR FANS A TRAIN SEL SWITCH is selected to 1A.
- CTMT CLR FANS B TRAIN SEL SWITCH is selected to 1C.
- ALL CTMT CLR FANS are running in FAST.

A Design Base Steam Line Break has just occurred inside Containment with a subsequent LOSP. 1B DG tripped on overspeed, all other equipment actuated per design. As containment pressure exceeds 34 psig, the 1A CTMT CLR FAN trips on overload.

Which one of the following statements is correct concerning the action required by EEP-0, Reactor Trip or Safety Injection. Based on the equipment running after the action is complete, what is the effect on Containment Pressure and Temperature?

A. Start 1C CTMT CLR FAN, Containment Pressure and Temperature will exceed design limits.

B. No action is required due to the required number of CTMT CLR FANS running, Containment Pressure and Temperature will exceed design limits.

C. Start 1D CTMT CLR FAN, Containment Pressure and Temperature will NOT exceed design limits.

D✓ Start 1B CTMT CLR FAN, Containment Pressure and Temperature will NOT exceed design limits.

This question meets 10CFR 55.41(b) 2 for SRO level questions.

TS Bases 3.6.6 **During a DBA, a minimum of one containment cooling train with a single OPERABLE fan unit and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits** (Ref. 3). Additionally, **one containment spray train is also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis**. To ensure that these requirements are met, two containment spray trains and two containment cooling trains with a single OPERABLE fan unit per cooling train with at least 600 gpm SW flow must be OPERABLE. Therefore, in the event of an accident, at least one train in each system operates, assuming the worst case single active failure occurs.

Bases also says that a SLB is the worst case scenario, the highest peak pressure is 52 psig and highest temperature is 367°F. Both meet the intent of the design bases. The design air temp. is 367°F B3.6.5-2 and pressure is 54 psig B3.6.4-1.

In this question the candidate would have to recognize the following:
- the required actions of EEP-0 which are to Verify at least one containment fan cooler per train - STARTED IN SLOW SPEED
- during an LOSP with 1B DG tripped only one CTMT CLR FAN would be running and it would be the selected one on the train with power.
- no fan will start when the 1A CTMT CLR FAN trips as happens in FAST speed.
- CTMT CLR FANs 1 C&D are powered from the B Train and B Train has no power.
- That 1B CTMT CLR FAN is powered from A Train and A Train has power.
- That it takes only 1 CS pump and 1 Fan to keep peak pressure and temperature below design limits.

A. Incorrect, 1C CTMT CLR FAN will NOT start due to the loss of power to that train. This not a complete action due to E-0 says to verify 1 CTMT CLR FAN running in each train and no fan is running so the action is not complete when 1B is available. If this were done and not recognized that 1C CTMT CLR FAN was not running, then both would exceed their limits with no fan operating and one CS pump running IAW TS Bases.

B. Incorrect, No CTMT CLR FAN is running at present so action is required IAW EEP-0, both would exceed their limits with no fan operating and one CS pump running IAW TS Bases.

C. Incorrect, Since 1D CTMT CLR FAN will NOT start, Both could exceed their limits.

D. Correct, **Start 1B CTMT CLR FAN, Containment Pressure and Temperature will NOT exceed design limits.**
One CS pump and one fan will ensure neither of the parameters exceed their design limits.

000056 Loss of Off-site Power / 6
AA2 Ability to determine and interpret the following as they apply to the Loss of Offsite Power: SRO –AA2.09 – Operational status of reactor building cooling unit.

List the automatic actions associated with the Containment Spray and Cooling System components and equipment during normal and abnormal operations including (OPS40302D07):
- Normal control methods
- Automatic actuation including setpoint (example SI, Phase-B, LOSP) and the effect of selecting the containment cooler control to local.
- Protective isolations such as high flow, low pressure, low level including setpoint
- Protective interlocks

Modified Bank Question CS&COOL-52102C01 #2
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

14. 076A2.01 001

Unit 1 is operating at 50% power with 'A' Train on service. The 1C SW Pump is tagged out for motor replacement.

A fire is reported on the 1K 4160 volt bus and the plant operators deenergize 1K 4160 volt bus. CCW FROM CCW HX TEMP, TI-3042C, is elevated slightly and rising slowly. Which one of the following is the procedurally correct response to this situation?

A. Reduce power to < 35%, remove the Main Turbine from service and perform AOP-7.0, Loss of Turbine Building Service Water, in conjunction with AOP-10.0, Loss of Service Water.

B. Start a CCW pump and charging pump in the nonaffected train, secure affected train charging pump, then ramp the Main Turbine as necessary to maintain hydrogen temperature < 46°C.

C. Trip the reactor and perform EEP-0, Reactor Trip or Safety Injection, stop all RCP's, and perform AOP-4, Loss of Reactor Coolant Flow in conjunction with ESP-0.1, Reactor Trip Response.

D.✓ Start a CCW pump and charging pump in the nonaffected train, secure affected train charging pump, and swap on service trains of CCW.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.
AOP-10.0

A. Incorrect - removing the main turbine from service is not addressing the problem and there is no procedural guidance to do this at this time.

B. Incorrect - Swapping on-service trains is partially correct, but the turbine is not ramped. At 46°C the turbine is now tripped. We used to ramp at 48°C.

C. Incorrect - The motor temps are not high (195°F) enough to require a rx trip at this point. AOP-4 would only be performed if the Plant was initially in mode 3 (AOP-10.0 Step 12.2.3 RNO).

D. Correct - **Start a CCW pump and charging pump in the nonaffected train, secure affected train charging pump, and swap on service trains of CCW.**
This is the correct response because enough time is allowed before RCP temps increase to 195°F to mitigate the loss of SW and prevent the need to trip the reactor. It will take time to heat up the Onservice train of CCW, and AOP-10.0 takes that into account.

076 Service Water
A2 Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: SRO –A2.01 – Loss of SWS.

2. Evaluate abnormal plant or equipment conditions associated with the Service Water System and determine the integrated plant actions needed to mitigate the consequence of the abnormality (OPS52102F02).

Modified Bank Question: AOP-10.0-52520J08 02

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D D D D D D D D D D	Items Not Scrambled

15. G2.1.12 001

Given the following conditions:

- Plant is at 100% power.
- RCS pressure is 2235 psig and Tavg is 573°F.
- The RHR discharge relief valve is leaking to the PRT at a rate of 2.5 gpm.

Which one of the following describes the type of leakage and the action required by Technical Specifications? (Assume all other systems are operating normally and no other RCS leakage)

A. Identified leakage that requires shutdown.

B. Unidentified leakage that requires shutdown.

C.✓ Identified leakage, but does not require shutdown.

D. Unidentified leakage, but does not require shutdown.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question.

a. Incorrect - examinee may confuse unidentified leakage limits with identified leakage limits. Does not require a shutdown.

b. Incorrect - examinee may confuse unidentified leakage limits with identified leakage limits. Does not require a shutdown.

c. Correct - **Identified leakage, but does not require shutdown.** leakage from this relief valve to the PRT is identified leakage however the limit is 10 gpm and thus does not require shutdown.

d. Incorrect - nto unidentified leakage.

Conduct of Operations 2.1.12 Ability to apply technical specifications for a system.

1. Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Reactor Coolant System (OPS52101A01).
- TS 3.4.13 RCS Operational LEAKAGE

slightly modified Bank Question: INTRO TS-52302A03 46

MCS Time: 5 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

16. G2.1.32 001

Given the following plant conditions:

- The "A" RCP is being prepared for startup.
- RCS cold leg temperature (T-cold) is 130°F.
- Pressurizer cold cal level (LT-462) is 90%.
- Steam Generator metal temperatures are all between 185 -190°F.
- Seal injection is in service with supply temperature at 100°F.

Which one of the following is the reason the "A" RCP should NOT be started?

A. The RCS heatup rate limit will be exceeded.

B.✓ The RCS pressure limits for low temperature will be exceeded.

C. The Pressurizer spray nozzle will cooldown rapidly causing it to crack.

D. The Steam Generator Tube sheet will cool down rapidly and cause Tube sheet damage.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question.
The reason for this precaution and limitation originates in Tech Spec Basis.

UOP-1.1 P&L 3.1.9.5
A RCP shall not be started with one or more of the RCS cold leg temperatures ≤ 325°F unless the pressurizer water volume is less than 24% wide range cold pressurizer level indication or the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

OPS-52510A page 7
Verify cold pressurizer level <24% or all steam generator shell temperatures <50°F above the associated RCS cold leg temperature.
If the water in the secondary side of the steam generators is hotter than the water in the reactor vessel, the RCS will experience an increasing pressure transient when the first RCP is started. As the cooler water in the reactor vessel passes through the steam generators, heat will be transferred from the SGs to the RCS. The expansion of the reactor coolant will cause a pressure increase. If either of the above conditions are met, the severity of the increasing pressure transient will be minimized.

Page 5
A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures < 325°F unless the pressurizer water volume is less than 24% wide range cold pressurizer level indication or the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.
With the steam generators hotter than the rest of the RCS, starting an RCP causes the relatively cool RCS water to be heated up as it passed through the steam generator. As the water heats up it causes a rapid pressure rise. If there is a bubble in the pressurizer, a sufficient surge volume would exist to absorb the transient.

3.4.6 TS BASIS:
Note 2 requires that the secondary side water temperature of each SG be < 50°F above each of the RCS cold leg temperatures or that the pressurizer water volume is less than 770 cubic feet (24% of wide range, cold, pressurizer level indication) before the start of an RCP with any RCS cold leg temperature = 325°F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

A. C. & D. - Incorrect. There is at most a 60-90°F difference between the RCS and S/G temperature, and this is less than the 100°F which is the amount of temperature change allowed in any 60 minute period. Even if the RCS and the S/Gs reached thermal equilibrium within 60 minutes, no cool down or heat up limits will be violated. The RCS would heat up due to the hotter S/G water, the S/G tubesheet would cool down, and the spray nozzle would probably not change temperature much due to the water near the top of the pressurizer not changing temperature much during the insurge. The most significant effect would be the RCS heat up causing an insurge to the pressurizer which would cause the pressurizer to go solid and rapidly increase RCS pressure at low temperature when the metals are the most brittle.

B. - Correct. **The RCS pressure limits for low temperature will be exceeded.** Tech Spec 3.4.6 Basis describes the reason for this precaution and limitation. "This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started".

1. Conduct of Operations
G2.1.32 Ability to explain and apply all system limits and precautions

1. Identify conditions during performance of UOP-1.1 that might result in equipment damage or degradation (OPS52510A01).

bank question: UOP1.1-52510A01 #1

MCS	Time:	1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
					Answer:	B B B B B B B B B B	Items Not Scrambled

17. G2.2.17 001

During a refueling outage an SO reports a leak on a HHSI flange inside containment located in an exclusion area. The FIN team supervisor states that his team can tighten the flange and stop the leak as TOOLPOUCH WORK.

Is this activity acceptable for the FIN team to perform and why/why not?

The activity is...

A. Acceptable as long as the maximum torque limits are **not** exceeded and a special RWP is obtained.

B. Acceptable as long as the work will **not** affect EQ, seismic, or ASME Code qualification of the flange.

C✓ NOT acceptable because TOOLPOUCH WORK is not allowed in areas that require support beyond that required in the actual conduct of work.

D. NOT acceptable because TOOLPOUCH WORK cannot be performed on flanges that are in the RCA.

Meets 10 CFR 55.43 (b) 4 requirements for SRO level question.

AP-52 Appendix 2

TOOLPOUCH WORK is defined as work that can be conducted without detailed written Instructions and without overall plant scheduling.
section 4.0 table
Flanges
Tighten to stop leakage (within maximum torque limits)

Section 1.0 of appendix 2
TOOLPOUCH WORK: (Refer to Figure 1 for process flow.)

- Is NON-OPERATIONAL IMPACT WORK within a worker's qualification and skill level that can be safely done on the spot without written work instructions.
- Requires no initiating work documents.
- Does not require support beyond that required in the actual conduct of work. Tools, consumables, or a Routine Radiation Work Permit are examples of appropriate **TOOLPOUCH WORK** support.
- Can be performed by one or more qualified workers, and may or may not require post maintenance documentation (section 1.3).

2.9 The consequences of error, potential for high radiation exposure, personnel injury, or equipment damage are minimal.

A & B. Incorrect- This is not acceptable because the flange is located in an exclusion area and this job requires a special RWP

C. Correct - **NOT acceptable because TOOLPOUCH WORK is not allowed in areas that require support beyond that required in the actual conduct of work.**

this is not acceptable to be done by the FIN team as TOOLPOUCH WORK because work is not allowed in an area where high radiation exposure is possible per App. 2 step 2.9, determining if the job is TOOLPOUCH WORK.

D. Incorrect- it is not acceptable but the reason given is incorrect. It is permissible to do TOOLPOUCH WORK in the RCA under a routine RWP AND PER STEP 4.0 TABLE, FLANGES can be tightened as long as it is not tightened past the max. torque limits.

2. Equipment Control
2.2.17 Knowledge of the process for managing maintenance activities during power operations.

9. Describe the work control process and associated program interfaces, including Toolpouch Work (for example, tagging, radiation protection, foreign material exclusion, fire protection, and industry safety (ESP52303N09).

BANK QUESTION: PLT WK CONT-40502N09 07
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

18. G2.2.28 001

A reactor core re-load is in progress on Unit 1 with ten assemblies loaded in the core. A fuel assembly is currently being lowered into the core. As the on-coming Shift Supervisor you have been given the following information during turnover:

- Both Source Range detectors ARE indicating counts.
- High Flux at Shutdown alarms are BLOCKED due to spurious alarms.
- Source Range counts are audible in containment.
- Containment Purge is in service.
- One Reactor Operator is in the Unit 1 Control Room.
- The SFP upender operator has left the SFP area to use the restroom.

After reviewing the above information, your direction to the refueling SRO is to suspend Core Alterations because of which one of the following?

A. Containment Purge is required to be secured.

B. The High Flux at Shutdown alarm should be in service.

C.✓ The SFP upender operator is required to be on the headset.

D. There should be at least two licensed Reactor Operators in the Unit 1 Control Room.

Meets 10 CFR 55.43 (b) 6 requirements for SRO level question.

A - Incorrect, 3.6.3 is only applicable in Modes 1-4

B - Incorrect, Not required to be in service at this time per UOP-4.1 P&L 3.3.

C - Correct, **The SFP upender operator is required to be on the headset.**
The SFP upender operator is required by FHP-1.0 P&L 3.22 and 3.23.3 while moving fuel.

D - Incorrect, Even though two ROs are required to be in the Control Room during a plant startup (Mode 2), only One RO is required to be in the Control Room during refueling.

2. Equipment Control
G2.2.28 Knowledge of new and spent fuel movement procedures.

3. Determine the appropriate action for given conditions during performance of UOP-4.1 (OPS40503B03).

Modified Bank Question: INTRO TS-52302A03
removed the equipment hatch verbage due to double jeopardy with RO question on eq. hatch

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	C C C C C C C C C C	Items Not Scrambled

19. G2.3.4 001

The following Plant dose histories exist for four operators: (No dose has been received from other sites)

Operator	Paul	Andrew	Josh	Mark
Deep Dose Equivalent (DDE)	1.897 rem	1.929 rem	1.888 rem	1.861 rem
Shallow Dose Equivalent (SDE)	23 mrem	118 mrem	39 mrem	120 mrem
Committed Dose Equivalent (CDE)	1.668 rem	1.845 rem	1.767 rem	1.819 rem
Committed Effective Dose Equivalent (CEDE)	64 mrem	17 mrem	69 mrem	89 mrem

An activity in containment requires 2 operators to work in an area with a dose rate of 140 mrem/hr for 20 minutes.

Which one of the following sets of operators would you, as the SRO, assign to perform this activity so that neither operator would EXCEED their annual Administrative Dose Limit (ADL) for Total Effective Dose Equivalent (TEDE) while performing this activity?

A. Josh and Mark

B. Paul and Andrew

C.✓ Andrew and Mark

D. Paul and Josh

Meets 10 CFR 55.43 (b) 4 requirements for SRO level question.

changed the names and answer from B to C and reworded to show the 2 operators who would **not** exceed the limit rather than the 2 operators that would exceed the limit.

Page 7 of FNP-0-M-001 HP MANUAL shows operators to have an ADMIN Limit of 2000 mrem/yr for TEDE.

TEDE = CEDE + DDE
Paul 1.897 + .064 = 1.961 rem
Andrew 1.929 + .017 = 1.946 rem
Josh 1.888+ .069 = 1.957 rem
Mark 1.861+ .089 = 1.95 rem

140 mr/hr for 20 min = 2.33333 mr/min X 20 min = 46.67 mr

Paul = 1961 mr + 47 mr = 2008 mr
Andrew = 1946 mr + 47 mr = 1993 mr
Josh = 1957 mr + 47 mr = 2004 mr
Mark = 1950 mr + 47 mr = 1997 mr

C. Correct - **Andrew and Mark**

3. Radiation Control
G2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

9. Determine the actions to be taken by the re-entry personnel if dose rates during re-entry exceed the dose limits given. (OPS40105B9*). (EIP-14.0)

4. Given an emergency scenario determine if a reentry, relocation or movement is required evaluate the conditions and determine dose limits for personnel involved (OPS53002D04)

Bank Question: EPIP PERS-40501B09

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C C C C C C C C C C Items Not Scrambled

20. G2.3.9 001

You are the Unit 2 Supervisor. After a refueling outage, Unit 2 is being made ready to support power operations. The Unit is currently in Mode 5, Cold Shutdown. The crew is performing FNP-2-UOP-1.1, "STARTUP OF UNIT FROM COLD SHUTDOWN TO HOT STANDBY," RCS temperature is 195° F.

A member of the current license class is in the control room for the On-the-Job training portion of class. He asks you why the Containment Main Purge system must be secured prior to raising the RCS temperature above 200° F.

Which one of the following describes the correct response to the trainee's question?

A.✓ The minipurge duct work has much smaller isolation butterfly valves then the main purge duct work and are much more likely to shut and provide positive isolation of containment.

B. The minipurge ducts are small enough to be equipped with PAC filter assemblies which remove airborne activity, the main purge exhaust ducts are too large to be equipped with a PAC filter assembly.

C. The minipurge duct work is much smaller providing less flow to the plant vent stack therefore, if a release were to occur and a single isolation valve fails to close, its environmental effects would be minimized.

D. The minipurge isolation butterfly valves are designed to close within 5 seconds of receiving a isolation signal, the main purge isolation butterfly valves do not receive an isolation signal since they are already shut.

Meets 10 CFR 55.43 (b) 2 requirements for SRO level question.

Tech Spec Basis 3.6.3:
Shutdown Purge System (48-inch purge valves CBV-HV-3198A, (continued) 3198D, 3196, 3197)
The Shutdown Purge System operates to supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. **Because of their large size, the 48-inch purge valves are not qualified for automatic closure from their open position under DBA conditions. Therefore, the 48-inch purge valves are normally maintained closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.**
Minipurge System (8-inch purge valves CBV-HV-2866C, 2866D, 2867C, 2867D)
The Minipurge System operates to: b. Maintain radioactivity levels in the containment consistent with occupancy requirements with continuous system operation; and b. Equalize internal and external pressures with continuous system operation. **Since the valves used in the Minipurge System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3, and 4.**

A - Correct - **The minipurge duct work has much smaller isolation butterfly valves then the main purge duct work and are much more likely to shut and provide positive isolation of containment.**
per TS 3.6.3 BASIS. SEE ABOVE

B - Incorrect, Both Main and mini purge containment exhaust is sent through the PAC filtration unit.

C - Incorrect, Plant safety analysis assumes that all containment isolation valves close or are closed at the time an event occurs. If the minipurge valves were assumed to pass any flow from containment to the vent stack during an accident, they would be required to be closed in modes 1-4.

D - Incorrect, The containment isolation signal is sent to all butterfly valves in the containment ventilation system and all valves are designed to be quick closing. During an outage when the main purge valves are open, they would get an auto close signal and actuate to isolate containment in the event of high radiation in the ductwork.

3. Radiation Control
2.3.9 Knowledge of the process for performing a containment purge.

1. Identify and apply the following Technical Specifications or TRM requirements, including the bases and attendant equipment, associated with the Containment Structure and Isolation System (OPS52102A01).
• 1.6 Containment Integrity – Definition
• 3.6.1 Containment
• 3.6.2 Containment Air Locks
• 3.6.3 Containment Isolation Valves
• 3.6.4 Internal Pressure
• 3.6.5 Containment Air Temperature
• 13.6.1 Containment Ventilation System leakage Rate
• 13.8.1 Containment Penetration Conductor Overcurrent Protective Devices (Unit 2 Only).

3. Identify any special considerations such as safety hazards and plant condition changes that apply to the Containment Structure and Isolation System (OPS52102A04).

Bank Question number: UOP1.1-52510A02 02
Source: New 2001 nrc exam: Modified wording slightly to change "A recently licensed operator is the OATC performing his first startup" to "A member of the current license class is in the control room for the On-the-Job training portion of class".
MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

21. G2.4.21 001

The following conditions exist on Unit 1:

- Operators are responding to a reactor accident.
- The SI headers have been damaged resulting in a complete loss of injection to the core.
- Upon transitioning to EEP-1.0, Loss of Reactor or Secondary Coolant, the STA reports that temperatures seen by all core exit thermocouples (CETC's) are increasing rapidly.
- The five hottest CETCs read between 850°F and 875°F.
- Intermediate range SUR is oscillating from zero to +0.5 dpm.
- Containment Pressure is 55 psig.

Which one of the following describes the correct operator response for these conditions? Operators should:

A.✓ transition to FRP-Z.1, Response to High Containment Pressure.

B. transition to FRP-S.1, Response to Nuclear Power Generation/ATWT.

C. transition to FRP-C.2, Response to Inadequate Core Cooling.

D. continue in EEP-1.0. If the five hottest CETCs exceed 1200°F, then transition to FRP-C.1, Response to Inadequate Core Cooling.

This Question meets 10CFR55.43(b)(5)

References: CSF-0.0 and FRP-C.1

A. Correct. **transition to FRP-Z.1, Response to High Containment Pressure.**
This is the correct transition. A red path does exist for High Containment pressure, and there are no other red paths at this time.

B. Incorrect, Intermediate range detectors above zero would be an orange path on FRP-S.1, the red path on High Containment pressure is of higher priority.

C. Incorrect, There is an orange path on Core Cooling, but the Red path takes priority over all orange paths.

D. Incorrect, the correct transition to FRP-C.1 is the fifth hottest CETC > 1200°F, but staying in EEP-1 with two orange paths and one red path in is not an option.

4. Emergency Procedures / Plan
2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions including: (1) Reactivity control; (2) Core cooling and heat removal, (3) Reactor coolant system integrity, (4) Containment conditions, (5) Radioactivity release control.

11. Using the foldout page and memory (entry conditions) recognize the symptoms and perform the procedural guidance of the FRPs during orange and red path plant conditions (OPS52504A11).

Modified from FRP-C-52533C09 01

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

22. W/E02EA2.1 001

Plant conditions on Unit 1 are as follows:

- An inadvertent safety injection has occurred.
- ESP-1.1, SI Termination, in progress.

The pressurizer goes solid when ESP-1.1 is entered and then the 1A Pressurizer Safety valve sticks open. After Normal Charging is aligned with FK-122 demand at 100% and RHR Pumps secured, Subcooling Margin Monitor trends down to 15°F and continues to fall.

The SRO should:

A. Manually SI and stay in ESP-1.1, SI Termination.

B. Manually SI and go to EEP-0, Reactor Trip or Safety Injection.

C. Realign HHSI flow per Attachment 3 and go back to EEP-0, Reactor Trip or Safety Injection.

D✓ Realign HHSI flow per Attachment 3 and go to EEP-1.0, Loss of Reactor or Secondary Coolant.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

ESP-1.1

A. Incorrect. Realigning HHSI (but NOT via manual SI) and staying in ESP-1.1 would be an option per the Fold Out page of ESP-1.1 if subcooling was not maintained. but after securing all SI flow and going on normal charging, EEP-1 is the directed procedure transition per the RNO Column when solid plant and when subcooling cannot be maintained.

B. Incorrect. This could be correct in other plant conditions and in other emergency procedures when a safety injection is needed to raise Pressurizer level and maintain subcooling. In the stated conditions, Manual SI is not an option, since there is both fold out page and procedure step guidance to direct realignment of HHSI flow per attachment 3 if needed. EEP-0 actions which could help mitigate this condition have already been performed, and no value would be added by repeating those steps.

C. Incorrect. With a solid plant, a note prior to step 9.2.3 states that subcooling vice pressurizer level must be maintained by adjusting charging flow. If adjusting charging flow is not sufficient to control Subcooling, 11.1 RNO directs Establish HHSI flow using ATT. 3. The procedure user should then go to EEP-1.0, NOT BACK TO EEP-0.

D. Correct. **Realign HHSI flow per Attachment 3 and go to EEP-1.0, Loss of Reactor or Secondary Coolant.**

With a solid plant, a note prior to step 9.2.3 states that subcooling vice pressurizer level must be maintained by adjusting charging flow. If adjusting charging flow is not sufficient to control Subcooling, 11.1 RNO directs Establish HHSI flow using ATT. 3 and go to EEP-1. By this point in the procedure, subcooling was maintained until LHSI pumps were secured. This indicates that the cooldown and depressurization has been performed by the LOCA, and ESP-1.2 is not needed. EEP-1 is the proper recovery procedure for a LOCA with LHSI operation.

W/E02 SI Termination / 3

EA2 Ability to determine and interpret the following as they apply to the (SI Termination) SRO –EA2.1 – Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

8. Evaluate plant conditions and determine if transition to another section of ESP-1.1 or to another procedure is required. (OPS52531E08)

MCS	Time: 1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
				Answer:	D D D D D D D D D D	Items Not Scrambled

23. W/E04EA2.1 001

Unit 1 was operating at 100% power when a Small Break Loss Of Coolant Accident (SBLOCA) caused a plant trip and SI actuation.

- SI and Phase A Containment Isolation have actuated per design.
- The crew has implemented EEP-0, Reactor Trip or Safety Injection and EEP-1, Loss of Reactor or Secondary Coolant.
- A LOCA outside containment is indicated so the crew has transitioned to and is performing steps in ECP-1.2, LOCA Outside Containment.
- The crew has just completed the step to isolate RCP seal injection.

The crew observes RCS Pressure is no longer dropping, and is now rising.

Which one of the following describes the required actions in accordance with ECP-1.2 that must be taken at this point?

A.✓ Go to EEP-1, Loss Of Reactor Coolant Or Secondary Coolant.

B. Direct HP to perform radiation surveys in the auxiliary buildings.

C. Go to ESP-1.2, Post LOCA Cooldown and Depressurization.

D. Immediately transition to ECP-1.1, Loss Of Emergency Coolant Recirculation.

This question meets 10CFR55.43(b)(5)

A - Correct; **Go to EEP-1, Loss Of Reactor Coolant Or Secondary Coolant.**
EEP-1 is entered since the leak has been isolated. Pressure rising in the RCS is the symptom of the leak being successfully isolated that the next procedure step uses to direct the transition to EEP-1.0

B - Incorrect; If the leak in NOT isolated, and before transitioning to ECP-1.1, HP is sent to the aux building to perform surveys to possibly identify the leak location (ECP-1.2 Step 3.14).

C - Incorrect; No procedural guidance to do this. ESP-1.2 is likely the procedure transition after EEP-1.0.

D - Incorrect; Only after all possible leak locations are checked and the leak is still NOT isolated does the crew transition to ECP-1.1, this is performed after HP performs surveys in the aux building (ECP-1.2 step 3.15).

W/E04 LOCA Outside Containment
EA2 Ability to determine and interpret the following as they apply to the (LOCA outside CTMT) SRO-EA2.1 – Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

8. Evaluate plant conditions and determine if transition to another section of ECP-1.2 or to another procedure is required (OPS52532E08).

Modified ECP-1.2-52532E08 02
Initial conditions changed to make a different distractor correct.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled

24. W/E06EA2.2 001

FRP-C.1, "Response to Inadequate Core Cooling" has been implemented.

The following plant conditions exist on Unit 1:

- ECCS flow could not be established by any means.
- The secondary depressurization was performed at the maximum rate IAW FRP-C.1.
- All RCS hot leg temperatures are stable at 346°F.
- All SG pressures are at 90 psig.
- SG levels are 8% WR.
- Core exit thermocouple temperatures are approximately 900°F.

Which one of the following action(s) should the crew perform next:

A. Exit FRP-C.1 and enter FRP-C.2, "Response to Degraded Core Cooling".

B. Ensure ALL reactor coolant pumps running regardless of support conditions.

C. Continue SG depressurization to atmospheric pressure, isolating the accumulators at RCS pressure < 150 psig.

D✓ Stop SG depressurization, isolate accumulators, then depressurize SGs to atmospheric pressure.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.

WOG Background Document for FRP-C.1 Major Action Catagories:
1) Establish Safety injection Flow To the RCS
2) Rapidly Depressurize SGs to Depressurize RCS
3) Start RCPs and Open All RCS Vent Paths to Containment

FRP-C.1 Step 14.3 thru 14.5
Check ALL intact SG pressures less than or equal to 100 psig.

When ALL intact SG pressures less than or equal to 100 psig and RCS HL temps less than 350°F, THEN stop the pressure reduction.

Step 15 isolates the accumulators
Step 16 turns off any running RCP
Step 17 reduces pressure in all Intact SGs to atmospheric pressure

A. Incorrect. Even though the red path is clear now (<1200 CETC) and the orange path exists (>700 CETC), finishing the FRP-C.1 is required by procedure rules of usage since it has been entered. This point in the procedure does not transition the user to procedure and step in effect. If it did, transition to FRP-C.2 would be appropriate.

B. Incorrect. < 100# & < 350°F 2/3 Hot Leg temp. the accumulators are isolated. The purpose of the depressurization is to dump the accumulators in attempt to cool the core BEFORE the last resort of starting RCPs regardless of support conditions. IF CETC >1200°F after dumping and isolating Accumulators, then the RCP's are started one at a time vice all at once. If after any one RCP is started, the CETC Temps are < 1200°F, the non-running RCPs are not started.

C. Incorrect. Depressurizing to atmospheric pressure is not performed until the accumulators are isolated, to prevent non-condensable gas from injection into the RCS and further degrading cooling capablility.

D. Correct. **Stop SG depressurization, isolate accumulators, then depressurize SGs to atmospheric pressure.**
This is part of the attempt to lower RCS pressure to get accumulators to dump, isolate accumulators prior to injecting excessive N2 into the RCS, then continue to reduce RCS pressure in order to allow max LHSI flow for core cooling.

W/E06 Inad. Core Cooling
EA2 Ability to determine and interpret the following as they apply to the (Degraded Core Cooling) EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

7. Analyze plant indications to determine the successful completion of any step in FRPC. 1/C.2/C.3 (OPS52533C07)

BANK Question: FRP-C-52533C04 06
MCS Time: 1 Points: 0.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D D D D D D D D D Items Not Scrambled

25. W/E11EA2.2 001

The crew is responding to a LOCA outside containment. The reactor was tripped and SI was manually actuated. The team has completed procedure ECP-1.2, "LOCA Outside Containment," and transitioned to ECP-1.1, "Loss of Emergency Coolant Recirculation," since they were unable to isolate the leak.

- RWST level has trended down to 12.2 feet.

Which one of the following describes the correct actions to take in ECP-1.1 under these conditions?

A.✓ Start makeup to the RWST, initiate an RCS cooldown at less than 100°F/hr, minimize ECCS flow, and reduce RCS pressure.

B. Immediately initiate an RCS cooldown at the maximum rate possible, start makeup to the RWST, and establish one train of ECCS flow to maintain subcooling >66°F.

C. Verify all containment cooling units running in slow speed, initiate an RCS cooldown at the maximum rate possible to 200°F, and depressurize the RCS to allow RHR to be placed on service.

D. Verify all containment cooling units are running in fast speed, verify one containment spray pump is running, and control RCS temperature and pressure to maintain subcooling >66°F.

Meets 10 CFR 55.43 (b) 5 requirements for SRO level question.
ECP-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION OPS-52532D

A. Correct - **Start makeup to the RWST, initiate an RCS cooldown, minimize ECCS flow and reduce RCS pressure.**
The following criteria are the high level actions needed to be successful in ECP-1.1
Makeup to the RWST is necessary
Inventory in the RWST is a concern for recovery from a loss of ECR capability. Makeup is added to the RWST to extend the time the SI pumps and containment spray pumps (if operating) can take suction from the RWST and provide core cooling to the RCS.
Begin Cool Down to Cold Shutdown
The purpose is to begin a controlled RCS cool down to cold shutdown temperature using a preferred or alternate method with a specified maximum cool down rate. Shutdown margin should be monitored during RCS cool down using Curve 61 and/or 61A.
The objective is to reduce the overall temperature of the RCS coolant and metal to reduce the need for supporting plant systems and equipment required for heat removal. The maximum cool down rate of 100°F/hr will preclude violation of the integrity status tree, thermal shock limits.
Stop SI Pumps
To reduce flow into the RCS, the low-head injection pumps and all but one high-head pump are stopped. Satisfaction of conditions for SI termination implies that control can be maintained by the operator without all of the ECCS pumps running. In this step, all but one high-head pump are stopped and placed in standby for future use.
Reduce RCS Pressure to Reduce Subcooling
This step is performed to decrease RCS pressure to the lowest pressure possible without losing adequate subcooling. The RCS pressure reduction is done to decrease RCS break flow. The RCS should be depressurized until RCS subcooling indicates between 16°F (45°F) and 26°F (55°F) on the Subcooled Margin Monitor in CETC mode. A second criterion for stopping the pressure reduction is PRZR level greater than 73% (50%).

B. Incorrect - The CDR should be at 100°F/hr or less to preclude violation of the integrity status tree, thermal shock limits. ECCS flow is reduced to one train, but subcooling is maintained low (from 16°-26° per step 25.3). RNO Column for SCMM < 66°F stops all HHSI & RHR pumps and aligns chg thru the normal flow path.

C. Incorrect - The CDR should be at or less than 100°F/hr to preclude violation of the integrity status tree, thermal shock limits. When 200°F is reached, RHR is not started up if not already in service. Steam Dumps or Atmospherics are used to Cool Down Further.

D. Incorrect - All CTMT coolers will be started in slow speed and CS pumps are not necessary for this event: LOCA OUTSIDE Containment. Also the mitigation strategy is to decrease RCS temp and pressure to minimize Subcooling (which reduces break flow) and reduce ECCS flow (which conserves RWST inventory).

W/E11 Loss of Emergency Coolant Recirc. / 4
EA2 Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation)
SRO –EA2.2 – Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

7. Analyze plant indications to determine the successful completion of any step in ECP-1.1 (OPS52532D07).

Modified Bank Question: ECP-1.1-52532D04 #1 significantly changed the answer and all distracters. The stem remains the same.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A A A A A A A A Items Not Scrambled