1. 001AK2.06 001/FNP BANK/RO/C/A 2.9/3.2/Y 2007/N/2/CVR/SAT

Unit 1 is at 74% power and stable, and the following conditions occurred:

At 1000:

- Rod control is in AUTO.
- TI-408A, Tavg - Tref deviation, indicates 0°F and stable.
- Pressurizer level is stable.
- Reactor Power is approximately 75% and stable.
- Control Bank D step counters are at 144 steps.

At 1002:

- TI-408A, Tavg - Tref deviation, indicates approximately +2°F and rising.
- Pressurizer level is slowly rising.
- Pressurizer spray valves have throttled open.
- Reactor Power is approximately 76% and slowly rising.
- Control Bank D step counters are at 150 steps and rising at 8 steps per minute.
- There is no load change in progress.

Which one of the following describes:

1) the event in progress

and

2) the **NEXT** action that must be performed IAW AOP-19.0, Malfunction of Rod Control System?

A. 1) Inadvertent RCS boration;

2) Trip the reactor and enter EEP-0, Reactor Trip or Safety Injection.

B. 1) Inadvertent RCS boration;

2) Place the rod control mode selector switch to MANUAL and match Tavg with Tref by inserting rods.

C. 1) Uncontrolled Continuous Rod Withdrawal;

2) Trip the reactor and enter EEP-0, Reactor Trip or Safety Injection.

D.✓ 1) Uncontrolled Continuous Rod Withdrawal;

2) Place the rod control mode selector switch to MANUAL and verify that rod motion stops.

A - Incorrect. The first part is incorrect, since for an inadvertent boration, Tavg/Tref mismatch would be less than -1.5 (with rods to be moving outward) and power would be less than 75% instead of 76%. Plausible, since rods would be moving out and Tavg/Tref mismatch could be increasing (which would cause Przr level to rise and spray valves to throttle open) with an inadvertent boration. The second part is incorrect, but plausible. The stated action is the RNO if rods do not cease moving once they have been placed in manual IAW AOP-19. Also, a conservative action may be chosen to trip the reactor, but this would not be in accordance with AOP-19.0 for this situation, nor would it be necessary.

B - The first part is incorrect (see A). Second part is correct IAW AOP-19 for a continuous rod withdrawal (see D).

C - Incorrect. The first part is correct (see D). The second part is incorrect (see A).

D - Correct. A CRW is taking place as indicated by the Tavg/Tref meter value going up above +1.5 and continuing to increase. This shows rods should actually be moving to lower the high temperature, and the action is to place rods in Manual if they are stepping while in AUTO.

Technical Reference: **AOP-19 Malfunction of Rod Control, Version 26.0**

Previous NRC exam history if any: FNP 2007 NRC exam, but with different distractors (changed from inadvertent dilution to inadvertent boration in A & B). This is the only question in the bank that comes close to meeting this k/a (searched BOTH "KA" and "second KA" on "contains 001AK").

001AK2.06
001 Continuous Rod Withdrawal
AK2. Knowledge of the interrelations between the Continuous Rod Withdrawal and the following: (CFR 41.7 / 45.7) AK2.06 T-ave./ref. deviation meter 3.0* 3.1

Match justification: This question presents conditions indicating a Continuous Rod Withdrawal, and the Tavg/Tref meter value and trend is provided. To obtain the correct answer, a knowledge of the relationship between the CRW and the Tavg/Tref meter response is required.

Objective:
4. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing AOP-19, Malfunction of Rod Control System. (OPS-52520S06)

2. 001K2.02 001/MOD/RO/MEM 3.6/3.7//N/2/CVR/SAT

Which one of the following correctly describes components in the power flow path to the Reactor Trip Breakers?

The 600V __(1)__ supply the CRDM MG set supply breakers, then the __(2)__, then the Reactor trip breakers.

	(1)	(2)
A.	LCCs D and E	Motor Generator Sets, then the Power Cabinets
B.	MCCs A and B	Motor Generator Sets, then the Power Cabinets
C.✓	LCCs D and E	Motor Generator Sets
D.	MCCs A and B	Motor Generator Sets

A. Incorrect. Plausible because all parts of the correct answer are listed, but in an incorrect order. The power cabinet is actually downstream of the Reactor trip breaker.
B. Incorrect. MCC A & B are incorrect, but Plausible; these are also safety related 600V switchgear. See A for second part.
C. Correct. Per load list and FSD on Reactor Protection, A181007, Figure F-1.
D. Incorrect. MCC A & B are incorrect, but Plausible; these are also safety related 600V switchgear.

Each MG set consists of a 600V AC, 150 hp induction motor, a stainless steel flywheel, and a 260V AC, 3 phase synchronous generator located in the non-rad Aux bldg 139' elev. The motors for the MG sets are powered from two 600V load centers (LCs) (MG A is powered from LC D; MG B is powered from LC E). These motor supply breakers can be operated from the MG set control panels located in the rod control room (Aux bldg 121' elev.) or from the MCB.

Previous NRC exam history if any: 2005 Vogtle NRC exam under 001K2.01 (power supplies to MG SETS)

001K2.02
001 Control Rod Drive System
K2 Knowledge of bus power supplies to the following: (CFR: 41.7)
K2.02 One-line diagram of power supply to trip breakers 3.6 3.7

Match justification: The power flow to the Reactor trip breakers is examined here including the Busses that supply the CRDM MG. This power supply flow must be understood to correctly answer this question.

Objective:
1. **NAME AND IDENTIFY** the power supply for the following cabinets associated with the Reactor Protection System (RPS) to include those items found on Figure 12, 120 VAC Distribution (OPS-52201I04).

3. 003A2.03 001/NEW/RO/C/A 2.7/3.1/N/N/4/EDITORIAL/SAT

Unit 1 is at 25% power and the following conditions occurred:

At 1000:

- 1A RCP amps and motor winding temperature were observed to be rising while 1A RCS LOOP flow was decreasing.

At 1002:

- EF1, 1A RCS LOOP FLOW LO OR 1A RCP BKR OPEN, is in alarm.
- 1A RCP Handswitch indicating Green and Amber lights are LIT, the Red light is **NOT** LIT.
- AOP-4.0, Loss Of Reactor Coolant Flow, immediate actions have been completed.

RCS Temperatures are:

- 1A RCS LOOP Tavg is 537°F.
- 1B RCS LOOP Tavg is 553°F.
- 1C RCS LOOP Tavg is 553°F.

Which one of the following correctly describes the **CAUSE** of these indications and the **ACTION** required IAW AOP-4.0?

	CAUSE	ACTION
A.✓	Seized motor bearing	Trip the Reactor
B.	Sheared shaft	Trip the Reactor
C.	Seized motor bearing	Commence Normal Reactor Shutdown
D.	Sheared shaft	Commence Normal Reactor Shutdown

This is on the RO level, since TS 3.4.2 requires Mode 3 in 30 minutes for Tavg below Minimum Temperature for Criticality. AOP-4.0 requires a reactor trip in this situation at step 3 after immediate action steps 1 & 2.

A - Correct. As the motor bearing starts to seize, the amps go up and flow goes down until the breaker trips on overcurrent. Then, AOP-4.0 entry conditions are met ("This procedure is entered when forced RCS flow is lost in one or more loops and no reactor trip is required.") AOP-4.0 requires a reactor trip if any Tavg is < 541°F, and the stem gives 539°F for A Loop.

B - Incorrect. The first part is incorrect, but plausible. RCS flow would go down if the shaft sheared, but current would also go down instead of up. The second part is correct (see A).

C - Incorrect. The first part is correct (see A). The second part is incorrect, but plausible. It would be correct at this power level (<30%, P-8) if Tavg was not less than the Minimum Temperature for Criticality.

D - Incorrect. The first part is incorrect (see B). The second part is incorrect (see C).

AOP-4, Version 18.0
TS 3.4.2

Previous NRC exam history if any:

003A2.03

003 Reactor Coolant Pump System

A2 Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5/ 45.3 / 45/13)

A2.03 Problems associated with RCP motors, including faulty motors and current, and winding and bearing temperature problems 2.7 3.1

Match justification: This question requires knowledge of what type of motor malfunction would give the indications in the stem. The indications were given in the stem and the applicant is required to analyze and diagnose what malfunction would cause these indications to avoid backwards logic. This order was also required to allow choosing actions which are based on the indications. At FNP, Motor bearing temperature indication is not available, but motor winding temperatures are. Winding temperatures would go up due to the motor shaft and bearing seizing, so is included in the stem. The second part of the question and choices require knowledge of what action is required for the given set of indications.

Objective:

6. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Reactor Coolant Pumps (RCPS) components and equipment, to include the following (OPS-40301D07):

   - Normal Control Methods
   - Abnormal and Emergency Control Methods
   - Protective isolations such as high flow, low pressure, low level including setpoint
   - Fast dead bus transfer
   - Automatic actuation including setpoints
   - Actions needed to mitigate the consequence of the abnormality

4. 004G2.4.21 001/NEW/RO/C/A 4.0/4.6/N/N/2/EDITORIAL/SAT

A LOCA and LOSP has occurred on Unit 1, and the following conditions occurred:

- FRP-C.2, Response To Degraded Core Cooling, is in progress.
- CCW to ALL the RCPs thermal barriers has been lost.
- All charging pumps have tripped.
- All RCP's are secured.
- The five hottest CETCs are; 773°F, 779°F, 1023°F, 1252°F, 1508°F and all stable.
- All SG pressures are at 1000 psig.
- Off-Site Power is available.

Which one of the following states:

1) the FRP that must be in effect for the conditions given (FRP-C.2 Response To Degraded Core Cooling, **OR** FRP-C.1 Response To Inadequate Core Cooling),

and

2) whether the RCPs are required or **NOT** required to be started?

A. Enter FRP-C.1 RCPs are required to be started

B. Enter FRP-C.1 RCPs are **NOT** required to be started

C. Remain in FRP-C.2 RCPs are required to be started

D.✓ Remain in FRP-C.2 RCPs are **NOT** required to be started

A - Incorrect. The first part is incorrect, but is plausible. The fifth hottest core Core Exit Thermo Couple is not higher than 1200°F, so FRP-C.1 is not entered, but the highest three are >1200°F. Confusion may exist as to which of the highest thermocouples have to be >1200°F prior to entry into FRP-C.1. If the first part was correct, the second part would be correct also. A major difference between FRP-C.1 & C.2 is that in FRP-C.1, RCPs are started as a last resort even with no support conditions. In FRP-C.2 a RCP is started only if all support conditions are met.

B - Incorrect. The first part is incorrect (see A). The second part is incorrect, but plausible, since for FRP-C.2 and all other procedures it is correct. Confusion may exist as to whether or not FRP-C.1 directs starting the RCPs without support conditions when FRP-C.2, and all other procedures do not.

C - Incorrect. The first part is correct (see D). The second part is incorrect (see A).

D - Correct. The fifth hottest Core Exit TC is > 700°F but <1200°F. Therefore, FRP-C.2 is still in effect and FRP-C.1 is not entered. FRP-C.2 does not direct starting a RCP without support conditions, but FRP-C.1 does.

CSF-0, Critical Safety Function Status Trees, Revision 17
FNP-1-FRP-C.1, Response To Inadequate Core Cooling, Revision 17
FNP-1-FRP-C.2, Response To Degraded Core Cooling, Revision 17

Previous NRC exam history if any:

004G2.4.21
004 Chemical and Volume Control System
2.4.21 **Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.** (CFR: 41.7 / 43.5 / 45.12) RO 4.0 SRO 4.6

Match justification: RO level knowledge of the entry condition parameters and logic used to assess Red and Orange path (and logic) of the Critical Safety Functions is required to answer this question correctly. The Charging pumps in the CVCS system (which are also HHSI pumps during a LOCA) are provided as tripped which prevent all RCP support conditions from being met (along with a loss of CCW to the RCP Thermal barriers which is also listed in the stem). The second part of the question directly addresses the effect of the CVCS system on the procedure directions concerning starting or not starting RCPs.

Objective:

1. **EVALUATE** plant conditions and **DETERMINE** if entry into (1) FRP-C.1, Response to Inadequate Core Cooling; or (2) FRP-C.2, Response to Degraded Core Cooling; or (3) FRP-C.3, Response to Saturated Core Cooling is required. (OPS-52533C02)
2. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing (1) FRP-C.1, Response to Inadequate Core Cooling; (2) FRP-C.2, Response to Degraded Core Cooling; (3) FRP-C.3, Response to Saturated Core Cooling. (OPS-52533C06)

5. 004K1.04 001/NEW/RO/C/A 3.4/3/8/N/N/2/CVR/SAT

Unit 1 is at 100%, and the following conditions occurred:

- One Letdown orifice is on service.
- LK-459F, PRZR LVL, controller demand has failed high.

Which one of the following describes the effect on Charging Flow and RCP Seal Injection flows, **with no operator actions**?

	Charging Flow	Seal injection Flows
A.✓	Go up	Go Down
B.	Go up	Go up
C.	Go Down	Go up
D.	Go Down	Go Down

A - Correct. When FK-122 fails high, charging flow increases. This robs flow from the Seal injection lines and the Seal Injection flows go down. When Seal Injection flows go down, #2 seal flow and leakoff flow also goes down, since it is supplied by Seal Injection flow. When charging flow goes up, and letdown is unchanged, VCT level goes down. VCT pressure goes down due to expansion of the gas volume in the VCT. When pressure in the VCT goes down, #1 seal leakoff flow to the VCT goes up due to less back pressure.

B - Incorrect. The first part is correct (see A). The second part is incorrect, due to the immediate effect of Seal Injection decreasing due to charging flow being in parallel with Seal inj. Flow. Charging flow increasing robs flow from seal injection flow. Plausible, since the VCT pressure goes down as VCT level goes down and the Number 1 seal leak off does go up eventually due to the VCT pressure drop, but the seal injection flow does not go up.

C - Incorrect. Charging flow goes up due to the direct relationship between the master LK-459 Pressurizer level controller and the slave FK-122 controller, and the valve position of FCV-122. Plausible, since some of the MCB master slave controllers have an inverse relationship, such as PK-444A, PRZR PRESS REFERENCE controller and PK-444C & D, 1A & 1B LOOP SPRAY VLV controllers. The SPRAY VLV controller demands go up when the REFERENCE controller demand goes up. The second part is incorrect (see B). Plausible, since if the first part were correct, the second part would be correct also.

D - Incorrect. The first part is incorrect (see C). The second part is correct (see A). Plausible, since physical connections and the cause/effect relationships between the CVCS system and the RCPS may be misunderstood and confusion could exist as to the inverse relationship between the two flows.

FSD: CVCS/HHSI/ACCUMULATOR/RMWS A-181009
PID 175039 SH 6, CVCS chg & seal injection
REACTOR COOLANT PUMPS, OPS-62101D, OPS-52101D, OPS-40301D, STUDENT TEXT

Previous NRC exam history if any:

004K1.04

004 Chemical and Volume Control System

K1 Knowledge of the physical connections and/or cause-effect relationships between the CVCS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.04 RCPS, including seal injection flows 3.4 3.8

Match justification: The RCP #2 seal flow and the Seal Injection flows are both affected by the CVCS system during a Charging flow and/or VCT level/pressure transient. To correctly answer this question, knowledge of this relationship as well as the physical connections is required.

Objective:

4. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing AOP-100, Instrument Malfunction. (OPS-52521Q06).

2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the components associated with the Chemical and Volume Control System, to include the components found on Figure 3, Chemical and Volume Control System and Figure 4, RCP-Seal Injection System (OPS-40301F02).

6. 005A1.07 001/NEW/RO/C/A 2.5/3.1/N/N/3/EDITORIAL/SAT

A time stroke of Q1E11MOV8889, RHR TO RCS HOT LEGS ISO, in the open direction has been performed per STP-11.6, Residual Heat Removal Valves Inservice Test.

Open direction ACCEPTABLE STROKE TIME RANGE is 9.96 to 13.47 Sec.
Open direction MAXIMUM ALLOWABLE TIME is 16 Sec.

Stroke times obtained were as follows:

- **At 1000** First time stroke: 15.35 Secs
- **At 1005** Second time stroke: 15.52 Secs

Which one of the following describes MOV-8889 OPERABILITY IAW Technical Specifications and the actions that are required to be placed in a CR for these results IAW STP-11.6?

A✓ • MOV-8889 is OPERABLE

- Analysis of the time stroke results within 96 hours to determine if new stroke time is acceptable.

B. • MOV-8889 is INOPERABLE

- Analysis of the time stroke results within 96 hours to determine if new stroke time is acceptable.

C. • MOV-8889 is OPERABLE

- Repair or replacement of MOV8889.

D. • MOV-8889 is INOPERABLE

- Repair or replacement of MOV8889.

Changed question per comments to: "actions that are required to be placed in a" CR.

A - Correct. Tech Specs requires the time stroke to be less than the Maximum stroke time. This is stated as acceptance criteria in STP-11.6, Step 5.3.3.4 & 5.3.3.5, but outside of the Acceptable Stroke Time Range the valve must have a retest and an analysis if the stroke time is still outside of the Acceptable range but less than the maximum. The second part is correct per STP-11.6, Step 5.4.2.

B - Incorrect. The first part is incorrect, but plausible. The tech spec limit is the same as the maximum time for the valve stroke. A stroke time above the maximum does not meet acceptance criteria and requires declaring the valve inoperable, but above the acceptable range AND below the Max time meets acceptance criteria. Acceptable range may be confused with acceptance criteria. Not meeting acceptance criteria indicates inoperability due to TS requirements not being met. Also, if either of the tests were greater than the maximum, or if no retest was possible this choice would be correct. Second part is correct (see A).

C - Incorrect. First part is correct. Second part is incorrect but plausible. Writing a CR is required, but requiring repair or replacement of the valve is only required for a valve that has been required to be declared inoperable (no analysis in 96 hours, greater than MAX stroke time, or no retest possible and outside of the acceptable range). Analysis and possible resetting the baseline of the valve stroke time is allowed and required by the STP.

D - Incorrect. First part is incorrect (see B). Second part is incorrect (see C) but plausible. If the first part was correct, then this would be correct per STP-11.6, AND tech specs would not be met until the valve was repaired to allow time stroking in less than the Max allowed time.

STP-11.6 step 5.4 Version 36

5.4 In Table 1, compare Actual Stroke Times to Maximum Allowable Times and to the Acceptable Stroke Time Range and perform the following as applicable:

5.4.1 IF the Actual Stroke Time for a valve exceeds the Maximum Allowable Time, THEN perform the following:

1. Declare the valve inoperable.
2. Check the appropriate Technical Specifications, Technical Requirements Manual, and Fourth 10-Year Interval IST Program for corrective action requirements.

5.4.2 IF the Actual Stroke Time for a valve is outside the Acceptable Stroke Time Range AND does NOT exceed the Maximum Allowable Time, THEN perform the following:

1. Immediately retest the valve.
2. IF it is NOT possible to retest the valve, THEN declare the valve inoperable.
3. IF the valve is retested AND the second set of data is also outside the Acceptable Stroke Time Range, THEN perform the following:
   a. Submit a CR to have the data analyzed within 96 hours to verify that the new stroke time represents acceptable valve operation.
   b. Enter the CR number in Table 1.
   c. Initiate an Admin LCO to declare the valve inoperable if not analyzed within 96 hours.
4. IF the valve is retested AND the second set of data is within the Acceptable Stroke Time Range, THEN analyze the cause of the initial deviation and submit a CR to have the results documented in the Record of Tests.

5.4.3 IF any valve is declared inoperable, THEN perform the following:

1. Resolve the unacceptable condition by performing one of the following:
   - Repair the valve.
   - Replace the valve.
   - Analyze the associated valve stroke data to determine the cause of the deviation and whether valve operation is acceptable as is.
2. Prior to returning any valve to service following repair, replacement, or analysis, write a CR to request that ES issue new baseline data.

Previous NRC exam history if any:

005A1.07
005 Residual Heat Removal System
A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: (CFR: 41.5 / 45.5)
A1.07 Determination of test acceptability by comparison of recorded valve response times with Tech-Spec requirements 2.5 3.1*

Match justification: Recorded values of valve response times are given and the applicant is required to assess whether or not Tech Specs are met on the RO level of knowledge. The STPs are the mechanism with which ROs assess operability of valves per their stroke times. The "Tech-Spec requirements" are assessed in the valve stroke STPs with acceptance criteria being met or not met. This question provides a stroke time with a retest (directed by the procedure in this case) and the applicant must assess "Tech-Spec requirements" as to declaring inoperable or not (in the first part of the answers), and further actions per the STP (in the second part of the answers).

Objective:

1 **RECALL AND APPLY** the LCO and APPLICABILITY for Technical Specifications (TS) or TRM requirements, and the REQUIRED ACTIONS for 1 HR or less TS or TRM requirements, and the relevant portions of BASES that DEFINE the OPERABILITY and APPLICABILITY of the LCO associated with the Residual Heat Removal System components and attendant equipment alignment, to include the following (OPS-52101K01):

- 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
- 3.4.12, Low Temperature Overpressure Protection (LTOP) System
- 3.4.14, RCS Pressure Isolation Valve (PIV) Leakage
- 3.5.2, ECCS – Operating
- 3.5.3, ECCS – Shutdown
- 3.9.4, Residual Heat Removal (RHR) and Coolant Circulation - High Water Level
- 3.9.5, Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level
- 13.5.1, Emergency Core Cooling System (ECCS)

7. 005K4.03 001/NEW/RO/C/A 2.9/3.2/N/N/4/EDITORIAL/SAT

Unit 1 is performing a plant cooldown using the A Train RHR system, and the following conditions occurred:

- HIK-603A, 1A RHR HX DISCH VLV, controller demand is at 50%.
- FK-605A, 1A RHR HX BYP FLOW, controller is in **AUTO** with demand at 50%.

At 1000:

- HIK-603A demand setting is increased to 60%.

At 1005:

- RHR system flow is stable.

At 1010:

- FCV-605A is in its failed position due to an instrument air line break.

Which one of the following describes RHR flow at **1005**, and the position of FCV-605A at **1010**, **with no operator actions**:

RHR flow indicated on FI-605A at **1005** is (1) it was before **1000**,

and

the position of FCV-605A at **1010** is (2) ?

	At 1005 FI-605A, RHR HDR FLOW	**At 1010** FCV-605A, 1A RHR HX BYP FLOW
A.	higher than	Closed
B.	the same as	Open
C.	higher than	Open
D✓	the same as	Closed

Removed the flow rate from the stem and changed the stem to a fill in the blank question and each distracter from a flow rate to higher or same as terminology.

A - Incorrect. This first part is incorrect, since even thought the flow does initially go up, the FT senses this and the HX BYP FK demands the HX BYP FCV to close down to maintain the 3100 gpm initial flow. Plausible, since flow does go up initially. Also, if the BYP FCV is in manual which it normally is, this choice would be correct. The second part is correct (see D).

B - Incorrect. The first part is correct (see D). The second part is incorrect, but plausible. The valve fails closed on loss of air to maximize flow through the HX during a LOCA, but this valve could be confused with the HX discharge valve which fails open for the same reason.

C - Incorrect. The first part is incorrect (see A). The second part is incorrect (see B).

D - Correct. The design for the RHR HX BYP FCV is to operate in auto to maintain the total system flow rate constant while flow through the HX is adjusted with the potentiometer for the HX DISCH valve. The fail position of the valve is closed.

FSD: A181002, Residual Heat Removal-Low Head Safety Injection Functional System Description

3.15 RHR HEAT EXCHANGER DISCHARGE VALVES

5.1 RHR HEAT EXCHANGER BYPASS FLOW CONTROL

Previous NRC exam history if any:

005K4.03
005 Residual Heat Removal System
K4 Knowledge of RHRS design feature(s) and/or interlock(s) which provide or the following:
(CFR: 41.7)
K4.03 RHR heat exchanger bypass flow control 2.9 3.2

Match justification: The design features of normal cooldown operation of the RHR HX BYP FCV (in auto adjusting to maintain constant total flow vice adjusting to maintain constant valve position-first part of choices) and the design fail position of the valve (closed-second part of choices) must be understood to correctly answer both parts of this question.

Objective:

7. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Residual Heat Removal System components and equipment, to include the following (OPS-40301K07):

- Normal Control Methods
- Abnormal and Emergency Control Methods (Changes in system flow rates, Loss of control from the control room)
- Automatic actuation including setpoints (examples - Reactor Trip, SI, Phase A, LOSP/loss of all AC power)
- Protective isolations such as high flow, low pressure, low level including setpoint
- Protective interlocks
- Actions needed to mitigate the consequence of the abnormality

8. 006K6.13 001/FNP BANK/RO/MEM 2.8/3.1/N/N/4/CVR/SAT

A Small Break LOCA has occurred on Unit 1, and the following conditions occurred:

- A reactor trip and safety injection is in progress.
- 1A Charging Pump failed to auto start.
- 1C Charging Pump is the only charging pump running.
- RCS pressure is 1000 psig.

Which one of the following states the Safety Injection flow indication on FI-943, A TRN HHSI FLOW, **with no operator action**?

Safety Injection flow is approximately__________.

A. 0 gpm

B. 150 gpm

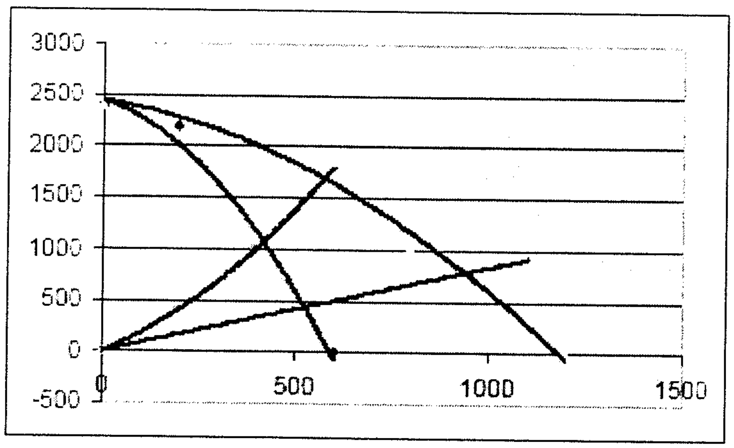
C✓ 450 gpm

D. 800 gpm

A - Incorrect. See C. Plausible since the meter is labeled “A train”, and during cold leg recirc this meter indicates only A train flow which is 0 gpm with no A train charging pump running. However, the trains are cross connected during the injection phase, and the B train pump flow is also indicated by this meter during the injection phase. Normal Charging flow indicated by FCV-122 indicates 0 for this condition.

B - Incorrect. See C. Plausible, since this is the approximate flow at normal RCS pressure with one charging pump. Also, it is the maximum charging indicated flow through the normal charging flow path at normal RCS pressure, but the normal charging flow path is isolated by the SI signal. Since the RCS pressure is less than NOP, the flow is greater than 150 gpm.

C - Correct. At ~1000 psig RCS Pressure, one HHSI (Charging) pump can provide about 450 gpm of flow. [Verified on simulator laptop, IC-73 SBLOCA from 100% power, 10,000 gpm leak. With RCS pressure at 1013 psig and one HHSI Pump tripped, HHSI flow on FI-943 was 440 gpm]. A knowledge of the exact value of charging flow from one pump at an RCS pressure of 1000 psig is not required to answer this question correctly. A knowledge of the characteristic pump curve for a centrifiugal pump and Charging pump capacity/capability at minimum is required. Also, knowledge of the system configuration in the injection phase of the LOCA (cross connected trains and both trains flow past the "A train" flow indicator).



graph shows single pump curve, parrellel pump (2 pumps) curve, and a generic System characteristic curves for SBLOCA and LBLOCA.

D - Incorrect. See C. Plausible, Two charging pumps could deliver 800 gpm into the RCS during a Large break LOCA if the RCS was at minimum pressure. This value is that which might be generally recalled from simulator observations for different conditions..

Previous NRC exam history if any:

006K6.13
006 Emergency Core Cooling System
K6 Knowledge of the effect of a loss or malfunction on the following will have on the ECCS:
(CFR: 41.7 / 45.7)
K6.13 Pumps 2.8 3.1

Match justification: This question requires knowledge of the effect on the ECCS system flow rate with one pump (HHSI) tripped.

Objective:

2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the components associated with the Emergency Core Cooling System, to include the components found on Figure 2, Accumulators, Figure 3, Refueling Water Storage Tank, and Figure 4, Emergency Core Cooling System (OPS-40302C02).

9. 007A3.01 001/FNP BANK/RO/C/A 2.7/2.9/N/N/3/EDITORIAL/SAT

Unit 2 is in Mode 5, and the following conditions occurred:

- PI-472, PRT PRESS, reads 7.5 psig and is rising slowly.
- LI-470, PRT LVL, reads 78% and is rising slowly.
- RCS Pressure is 225 psig.
- A Train RHR is aligned in the RCS Cooldown Operation IAW SOP-7.0, Residual Heat Removal System.
- It has been determined that V8708A, A Train RHR Pump suction relief valve, is leaking by the seat.

Which one of the following correctly states the impact on the PRT **with no operator action** and the required procedure actions to mitigate this condition per SOP-1.2, Reactor Coolant Pressure Relief System?

The PRT Pressure will reach a maximum pressure of __(1)__ psig,

and

to prevent reaching the PRT maximum pressure, the operator will be directed to __(2)__ per SOP-1.2, Reactor Coolant Pressure Relief System.

A. (1) 150 psig.

(2) pump down the PRT with the RCDT pump.

B. (1) 150 psig.

(2) gravity drain the PRT to the WHT.

C. (1) 100 psig.

(2) gravity drain the PRT to the WHT.

D✓ (1) 100 psig.

(2) pump down the PRT with the RCDT pump.

Deleted "and vent the PRT to #7 WGDT, if necessary" from due to it not being necessary.

A - Incorrect. Part 1 incorrect, but plausible, since RCDT relief is set at 150 psig. Part 2 is correct (see D).

B - Incorrect. Part one is incorrect (see A). Part 2 is incorrect, but plausible, since it would be correct IF the RCDT pumps were inoperable per SOP-1.2 step 4.3.3. Venting should not be necessary in this case due to the low energy of the RCS in mode 5 (<200°F), but the procedure does not address lowering pressure by just lowering level. Pressure is high because of level only, lowering level will also lower pressure and could be preferred, but lowering level by gravity draining should only be used if the RCDT pumps are inoperable.

C - Incorrect. Part 1 is correct (see D). Part 2 is incorrect (see B).

D - Correct. Both parts correct. RHR pump suction pressure is approx. the same as RCS pressure in this lineup: 225 psig per the stem. This makes it credible in that it could actually cause the rupture disc to break at it's setpoint of 100 psig per SOP-1.2 Step 3.5 "PRT pressure should be maintained < 100 psig to prevent rupture disc blowout."

The PRT has a N2 pressure established of approx. 0.5 to 3 psig to prevent formation of explosive gasses. This bubble will compress as level rises.

FNP-2-SOP-1.2, REACTOR COOLANT PRESSURE RELIEF SYSTEM, Version 30.0

<u>4.4 Reducing PRT Pressure</u>
4.4.1 Have Chemistry verify gas addition to the shutdown gas decay tank to be used for PRT venting (#7 or #8) is acceptable (e.g. H_2 < 4% and O_2 < 1% per CCP-203).

RCDT Relief valve pressure is 150 psig Per U259507.
PRT Rupture disks blows at 100 psig per SOP-1.2 STEP 3.5.

Per SOP-1.2:
4.3.3 Gravity Draining PRT to WHT
NOTES: • This method of draining the PRT should only be used if RCDT pumps are inoperable.

2-SOP-7.0, Residual Heat Removal System, Version 79.0

2-SOP-1.2, Reactor Coolant Pressure Relief System, Version 31
3.4 PRT level should be 68-78% during normal operation.
3.5 PRT pressure should be maintained < 100 psig to prevent rupture disc blowout.

4.3.2 Draining the PRT Using an RCDT Pump [Normal preferred method]

4.3.3 Gravity Draining PRT to WHT
NOTES: • This method of draining the PRT should only be used if RCDT pumps are inoperable.

Previous NRC exam history if any: n/a

007A3.01
007 Pressurizer Relief Tank / Quench Tank System
A3 Ability to monitor automatic operation of the PRTS, including: (CFR: 41.7 / 45.5)
A3.01 Components which discharge to the PRT 2.7* 2.9

Match justification: This question requires monitoring MCB PRT pressure indication and to know the pressure for automatic operation of the PRTS (at 100 psig the rupture disks ruptures). In this question, a component is discharging into the PRT (RHR Suction relief), and to answer this question, knowledge is required of what pressure will be indicated on the MCB prior to the PRT rupture disk automatically rupturing to relieve the pressure.

Objective:

2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the components associated with the Pressurizer System, to include the components found on Figure 3, Pressurizer and Pressurizer Relief Tank (OPS-40301E02).

10. 007EK2.02 001/NEW/RO/C/A 2.6/2.8/N/N/4/CVR/SAT .

Unit 1 was at 100% power, and the following conditions occurred:

- FRP-S.1, Response To Nuclear Power Generation – ATWT, is in progress.
- The Main Turbine was unable to be tripped from the MCB.
- A Safety Injection (SI) has **NOT** occurred.
- RCS pressure is 2230 psig.
- Tavg is 563°F.

Which one of the following describes the **immediate** effects if the Reactor Trip Breakers are opened locally at this time?

A. • The Block of an Auto SI will be allowed.
 • The Feed Water Reg Valves will trip closed.

B. • The Block of an Auto SI will be allowed.
 • The Steam Flow high setpoint will be reset.

C. • The Main Turbine will trip.
 • The Feed Water Reg Valves will trip closed.

D✓ • The Main Turbine will trip.
 • The Steam Flow high setpoint will be reset.

Added RCS pressure 2230 psig to ensure B is incorrect: IF <P-11, then PRZR LP SI can be reset (albeit not related to P-4): This was added to ensure there is ONLY 1 correct answer.

A - Incorrect. The first part is incorrect, since a block of SI is not an effect of opening the RT bkrs unless the SI has already initiated. Plausible, since if an SI had initiated this would be correct, and in many cases with an ATWT and NO Turbine trip an SI occurs, but the stem states that an SI has NOT occurred. The second part is incorrect also, since The Feed water Regulating Valves are only tripped closed by opening RT bkrs (P-4) in coincidence with a Low Tavg signal of 554°F. Since Tavg is still above 554°F, a FWIS will not occur immediately. Plausible, since on most reactor trips, a Low Tavg occurs due to steam dump operation very quickly after the Trip, but in this case Tavg is high due to the ATWT.

B - Incorrect. The first part is incorrect (see A). The second part is correct (see D).

C - Incorrect. The first part is correct (see D). The second part is incorrect (see A).

D - Correct. P-4 will trip the Main Turbine regardless of any other plant condition or parameter(s), and this will occur immediately when the Reactor Trip breakers are open. The Steam Flow high setpoint will be reset by P-4 immediately when the Reactor trip breakers are open regardless of any other plant parameter (the actuation of the steam flow MSIV isolation signal requires a Low Low Tavg: P-12, but resetting the signal occurs regardless of Tavg on a reactor trip. These are two of the several functions of P-4. The MCB handswitch which trips the Turbine directly did not work in this scenario (it operates the 20 AST-2 relay). P-4 operates the 20AST-1 and 20ET relays which open the interface valve and bleed EH fluid off of Throttle valves & Reheat Stop valves to trip main turbine per Figure 19 in the Student text for Main Turbine.

FNP-0-SOP-0.3, OPERATIONS REFERENCE INFORMATION, APPENDIX G, OPERATIONAL PERMISSIVES AND CONTROL INTERLOCKS, Version 39.0

Permissive

1. P-4 Reactor Trip Interlock

Source

Reactor Trip and Bypass Breakers

Setpoint

Breakers Open

Coincidence & Light Status

RTA & BYA Open or
RTB & BYB Open
No Light

Function

Prevents a rapid cooldown of primary system after a reactor trip.

1. Trips Turbine
2. Trips F.W. Reg Valves on Low Tavg
3. Seals in F.W. Reg Valve Trips from S.I. and S/G Hi Hi Level
4. Allows S.I. signal to be blocked after S.I. initiation
5. Resets Hi Stm Flow Setpoint
6. Arms steam dump system, enables plant trip controller and disables loss of load controller.

007EK2.02

007 Reactor Trip

EK2 Knowledge of the interrelations between a reactor trip and the following: (CFR 41.7 / 45.7)
EK2.02 Breakers, relays and disconnects 2.6 2.8

Match justification: To answer this question the applicant must know the normal relationship between the P-4 interlock and the reactor trip breakers and the various functions it accomplishes during a reactor trip.

Objective:

1. **RECALL AND 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 (RPS) and Engineered Safeguards Features (ESF) to include setpoint, coincidence, rate functions (if any), reset features, and the potential consequences for improper conditions to include those items in the following tables (OPS-52201I07):

   - Table 1, Reactor Trip Signals
   - Table 2, Engineered Safeguards Features Actuation Signals
   - Table 5, Permissives
   - Table 6, Control interlocks

5. **DEFINE AND EVALUATE** the operational implications of abnormal plant or equipment conditions associated with the operation of the Reactor Protection System (RPS) components and equipment to include the following (OPS-52201I09).
   - Normal Control Methods
   - Abnormal and Emergency Control Methods
   - Automatic actuation including setpoint (example SI, Phase A, Phase B, MSLIAS, LOSP, SG level)
   - Actions needed to mitigate the consequence of the abnormality

11. 008AA2.26 001/NEW/RO/C/A 3.1/3.4/N/N/2/EDITORIAL/SAT

Unit 1 has manually initiated a Safety Injection due to rapidly falling pressurizer pressure, and the following conditions occurred:

At 1000:

- Pressurizer level 35% and rising.
- RCS pressure 1700 psig and falling.
- PRT level is 73% and pressure is 5 psig and stable.
- TI-453, PORV downstream temperature, is 117°F.
- TI-453, Safety Valve downstream temperature, is 101°F.
- TI-453, Safety Valve downstream temperature, is 101°F.
- TI-453, Safety Valve downstream temperature, is 102°F.
- Containment Pressure 0.2 psig and slowly rising.
- The following radiation monitors are in alarm:
  R-2, CTMT 155 FT, R-7, SEAL TABLE, R-11, CTMT ATMOS, and
  R-12, CTMT GAS

At 1015:

Transition is made to EEP-1.0, Loss of Reactor or Secondary Coolant, and the following conditions exist:

- Pressurizer level 99% and rising.
- RCS pressure 1400 psig and rising.
- PRT level is 73% and pressure is 5 psig and stable.
- TI-453, PORV downstream temperature, is 138°F and rising.
- TI-455, Safety Valve downstream temperature, is 125°F and rising.
- TI-457, Safety Valve downstream temperature, is 125°F and rising.
- TI-459, Safety Valve downstream temperature, is 126°F and rising.
- Containment Pressure 0.96 psig and rising.
- Containment sump level is rising slowly.
- R-2, 7, 11 and 12 are still in alarm.

Which one of the following states **only** potential sources of the RCS leak indicated by the given conditions?

A. PORV leakby — Safety valve leakby

B. PORV leakby — PRZR Level upper tap break

C. PRZR Steam Space sample line break — Safety valve leakby

D✓ PRZR Steam Space sample line break — PRZR Level upper tap break

A - Incorrect. Both are incorrect, since the PRT parameters are unchanged after the event has been in progress for 15 minutes. If the PORVs or the Safeties had leaked by, the PRT parameters would be higher than they initially would. Plausible, since the downstream temperatures are higher than they were, but only slightly due to elevated ctmt ambient temp in the vicinity of the steam space break. If either of the PORVs or Safeties were leaking by, the tailpiece temperatures would be much higher than this.

B - Incorrect. The first part is incorrect (see A). The second part is correct (see D).

C - Incorrect. The first part is correct (see D). The second part is incorrect (see A).

D - Correct. These are both correct, since per the indications (przr level high and pressure low and rising due to going solid on SI flow) there is a steam space break. This choice has parts which are similar to incorrect choices which would be Przr Liquid Space sample and Przr Level lower tap, steam space sample and upper tap are both steam space penetrations.

Ran a 400 gpm steam space break from 100% power (IC-73) on the simulator laptop to validate these numbers.

Drawing PID: D-175037 SH 2

Previous NRC exam history if any:

008AA2.26

008 Pressurizer Vapor Space Accident

AA2. Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: (CFR: 43.5 / 45.13)

AA2.26 Probable PZR steam space leakage paths other than PORV or code safety 3.1 3.4

Match justification: This question presents a scenario with symptoms given of a steam space break. The indications have similarities to either a PORV or code safety leaking by OR another leakage path other than the PORV or code safeties, and some differences. In this case the applicant must correctly identify the potential sources of a a steam space break which for these symptoms must be other than a PORV or a code safety leaking by.

Objective:

1 **LABEL AND ILLUSTRATE** the Pressurizer System flow paths to include the components found on Figure 3, Pressurizer and Pressurizer Relief Tank (OPS-40301E05).

12. 008K3.01 001/MODIFIED/RO/C/A 3.4/3.5/N/N/3/REVISED/FIX

Unit 1 is at 100% power and the following occurred:

- TK-144, LTDN HX OUTLET TEMP controller, demand failed high.

Which one of the following describes the impact on the Letdown System Temperature, and what type of reactivity will be added to the core?

A. • Higher Letdown temperature.

  • Positive reactivity.

B✓ • Higher Letdown temperature.

  • Negative reactivity.

C. • Lower Letdown temperature.

  • Positive reactivity.

D. • Lower Letdown temperature.

  • Negative reactivity.

Changed second part of each distractor per suggestion to ask what type of reactivity would be added to the core vice what are the proper response actions. Updated Feedback accordingly.

A - Incorrect. The first part is correct (see B). The second part is incorrect (see B). Plausible, since it would be correct if temperature decreased (see D).

B - Correct. This controller demand goes up to raise temperature, which at 100% demand, sends a full closed signal to the CCW to the Letdown HX valve. Letdown temperature increasing will cause boron to be released from the demineralizers, thus adding negative reactivity.

C - Incorrect. This is incorrect since demand failing high causes the CCW to the Letdown HX valve to close (to raise Letdown Temperature). Plausible, since many valves open when demand goes to 100%. The second part is incorrect, but plausible. If the TK-144 valve went open and reduced letdown temperature, this would correct. Also, if the valve did go open, it would cause boron absorption in the Mixed Bed demineralizer due to the cooler Letdown temperature.

D - Incorrect. The first part is incorrect (see C). The second part is correct. See B.

Ran on Simulator Laptop (IC-73) AT 100%. DF1 was the first alarm to come in (less than 30 secs).

ARP-1.4, VERSION 48, DF1, LTDN TO DEMIN DIVERTED TEMP HI

3. Take manual control of LTDN HX Outlet Temp TK-144 and attempt to increase CCW flow to the Letdown Heat Exchanger.
4. Adjust charging or letdown flow as required to reduce the letdown flow temperature.
5. IF cause for the elevated temperature has been corrected, THEN refer to FNP-1-SOP-2.1, CHEMICAL AND VOLUME CONTROL SYSTEM PLANT STARTUP AND OPERATION to return TCV143 to DEMIN.
6. IF letdown temperature can NOT be reduced, THEN close LTDN ORIF ISO 45 (60) GPM Q1E21HV8149A, B, and C.

NOTE: Transients that will require boration or dilution should be avoided if letdown has been secured.

7. IF a ramp is in progress, THEN place turbine load on HOLD
8. Go to FNP-1-AOP-16.0, CVCS MALFUNCTION to address the loss of letdown flow.

ARP-1.4, VERSION 48, DF5, VCT TEMP HI,

4. Adjust charging or letdown flow as required to reduce the Letdown Flow Temperature.
5. Adjust LTDN HX Outlet Temperature < 111°F.

DRAWING D175039 SH 2

Previous NRC exam history if any:

008K3.01

008 Component Cooling Water System

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

K3.01 Loads cooled by CCWS 3.4 3.5

Match justification: This question presents a specific type of malfunction of the CCW system (Failure of the CCW to the Letdown HX control valve controller). To answer this question correctly, knowledge of the effect of this malfunction of the CCW system on the Load (Letdown) cooled by CCW is required. The effect is that LETDOWN temperature goes up due to the controller failure causing the CCW valve to the load (letdown) to go closed. The second part of the question and answers were added to gain 3 plausible but incorrect distractors.

Objective:

7. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the CCW System components and equipment, to include the following (OPS-40204A07):
   - Normal control methods
   - Abnormal and Emergency Control Methods
   - Automatic actuation including setpoint (example SI, Phase A, Phase B, High Radiation, LOSP)
   - Protective isolations such as high flow, low pressure, low level including setpoint
   - Protective interlocks
   - Actions needed to mitigate the consequence of the abnormality

13. 008K4.09 001/MOD/RO/C/A - 2.7/2.9//N/2/NO CHANGE OK/SAT

Unit 1 was operating at 100% power, and the following conditions occurred:

At 1000:

- A Train is the "On Service" train.
- 1B CCW pump is running and supplying loads in the on-service train.
- 1A CCW pump is running to support charging pump operations.
- 1C CCW pump is aligned and OPERABLE.

At 1005:

- A Safety Injection and LOSP occurred simultaneously.

Which one of the following combinations of CCW pumps will be running following the operation of the ESF sequencers, **with no operator actions**?

A.✓ 1A and 1C CCW pumps ONLY.

B. 1B and 1C CCW pumps ONLY.

C. 1A and 1B CCW pumps ONLY.

D. 1A and 1B and 1C CCW pumps.

A - Correct. 1B CCW pump is running on A train but will trip on Load shed. Then, the auto start circuitry starts up the non-swing, train related 1A & 1C pumps per CCW FSD Appendix A step 3.1.2.2, LOSP.

B - Incorrect. 1B CCW pump is running on A train and if there was no LOSP signal, the SI auto start circuitry would leave the 1B running and not start the pump on the same train. The opposite train pump is 1A, and not 1C. This is plausible since this is the opposite train and CCW has backward logic. normally 1A pump would be assigned to A train, but CCW is an exception to this general rule.

C - Incorrect. Plausible, since this would be correct with an SI and no LOSP.

D - Incorrect. Plausible, since the SW pumps would have all pumps including the swing running in the event of an SI if the swing pump was running to start with. For this CCW system alignment: 1B CCW pump is running on A train and 1A CCW pump is running on B train to start with. For an SI alone, 1B and 1A would be left running. For an LOSP, 1A and 1C would be started. However, the LOSP sequencer secures 1B prior to starting 1C.

FNP-1-SOP-23.0, Version 83.0

3.2 CCW is normally lined up so that

- One CCW pump and one CCW heat exchanger is in operation supplying the on-service train and the secondary heat exchangers.
- The remaining pump and heat exchanger are valved into a closed loop with the redundant safety train. The off-service train is normally in operation in modes 1–4 supplying the operating charging pump, with the non-operating SFP HX flowpath aligned and CCW to the RHR HX isolated. (Reference RER 1080944901)

3.1.1.3 SIAS

In the event of a SIAS with offsite power available, the on-service pump shall continue to operate, and the off-service (redundant) train-dedicated pump shall automatically start. Swing pump B shall continue to provide backup in the event of a fault trip of the dedicated pump in the train to which swing pump B is aligned, as above (Reference 6.7.11).

3.1.2 Swing Pump B on-Service, Dedicated Pump Available (Possible Alternate Alignment To Equalize Pump Wear)

3.1.2.1 On-Service Pump Trips

a. During normal plant operation, if on-service pump B trips due to a fault, the dedicated pump in the on-service (operational) train shall automatically start and supply component cooling water to the on-service component cooling heat exchanger (Reference 6.7.11).

3.1.2.2 LOSP

In the event of a LOSP with or without a SIAS, on-service pump B shall be shed and the two train dedicated pumps, C and A, shall be automatically sequenced onto the diesel generators (Reference 6.7.11).

Swing pump B shall provide backup in the event of a fault trip of the dedicated pump in the train to which swing pump B is aligned (Reference 6.7.11).

3.1.2.3 SIAS

In the event of a SIAS with offsite power available, on-service pump B shall continue to operate and the off-service train-dedicated pump shall automatically start. The dedicated pump in the on-service (operational) train shall continue to provide backup in the event of a fault trip of swing pump B (Reference 6.7.11).

Previous NRC exam history if any:

008K4.09
008 Component Cooling Water System
K4 Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following:
(CFR: 41.7)
K4.09 The "standby" feature for the CCW pumps 2.7 2.9

Match justification:

Objective:

2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the components associated with the CCW System, to include the components found on Figure 2, Component Cooling Water System, Figure 3, Secondary Heat Exchanger Header, and Figure 5, RCP-CCW & SW System (OPS-40204A02).

7. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the CCW System components and equipment, to include the following (OPS-40204A07):
   - Normal control methods
   - Abnormal and Emergency Control Methods
   - Automatic actuation including setpoint (example SI, Phase A, Phase B, High Radiation, LOSP)
   - Protective isolations such as high flow, low pressure, low level including setpoint
   - Protective interlocks
   - Actions needed to mitigate the consequence of the abnormality

14. 009EG2.1.23 001/NEW/RO/C/A 4.3/4.4/N/N/4/EDITORIAL/SAT

Unit 1 has experienced a Small Break LOCA, and the following conditions occurred:

At 1000:

- ESP-1.2, Post LOCA Cooldown and Depressurization, is in progress.
- Normal Charging has been established.

At 1010:

- CTMT Pressure is 6 psig and rising.
- Subcooling is 24°F and decreasing.
- PRZR Level is 28% and decreasing.

Which one of the following is the required action IAW ESP-1.2?

A. FK-122, CHG FLOW, must be adjusted to raise Przr level.

B. Place the SI ACTUATION switch to ACTUATE.

C. FK-122, CHG FLOW, must be adjusted to maintain current Przr level.

D✓ HHSI flow must be established and additional CHG PUMPs started.

A - Incorrect. The Fold Out Page requires reinitiating HHSI flow due to both Subcooling and PRZR Level being too low with adverse numbers "16°F{45°F} & 13%{43%}. Plausible, since this would be correct if the procedure step for maintaining pressurizer level was initiated with adverse numbers and 50% PRZR level was required, while forgetting about the FOP requirement to re-establish HHSI flow.

B - Incorrect. (see D). Plausible, since HHSI flow is needed, and it may seem more convenient to turn the SI switch vice going to the attachment to manipulate each component, but the FOP requires the attachment be used. This ensures that only the SI equipment and Phase A components desired are manipulated.

C - Incorrect. (see A). Plausible, since if Adverse numbers were not taken into account, this would be correct per step 20.2.1 of ESP-1.2.

D - Correct. The FOP requires this for these Subcooling and Przr level values. RO knowledge requires knowing the FOP requirements.

ESP-1.2, Revision 23

Previous NRC exam history if any:

009EG2.1.23
009 Small Break LOCA
2.1.23 **Ability to perform specific system and integrated plant procedures during all modes of plant operation.** (CFR: 41.10 / 43.5 / 45.2 / 45.6) RO 4.3 SRO 4.4

Match justification: This question requires knowledge of specific system and integrated plant procedures (ESP-1.2 Fold out Page) during a SBLOCA to answer correctly.

Objective:

6. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing ESP-1.2, Post LOCA Cooldown and Depressurization. (OPS-52531F06)

15. 010K1.03 001/NEW/RO/C/A - 3.6/3.7/N/N/4/REVISED/FIX

Unit 1 was at 28% power and the following conditions occurred:

- All PRZR Backup Heaters are in AUTO.
- A CVCS Malfunction has occurred.
- FK-122, CHG FLOW, has been placed in manual.
- PRZR level is at 36% and rising.

Which one of the following describes the operation of the Backup Heaters and the Spray valve controllers' demand **with no operator actions**?

All PRZR Backup Heaters will be ____(1)____

and

PK-444C and D, 1A and 1B LOOP SPRAY VLV controllers' demand go ____(2)____.

	(1)	(2)
A.✓	ON	Up
B.	ON	Down
C.	OFF	Up
D.	OFF	Down

Changed the second half of each question per the CE suggestion to eliminate low plausibility concerns. Now tests the demand response of the controller vice whether the spray valve opens or not.

A - Correct. First part: The CVCS Malfunction caused an insurge, which caused the PRZR level in increase. PRZR level program is 21.4-50.2% level from 547-573°F Tavg, so at 28% power, program level is 29.5% przr level. There is a 6.5% level deviation (>5%). Przr level >5% above the program level turns on all BU heaters which cause the pressure to go up more (after the water reaches the new higher saturation temperature).

Second part: The insurge caused the PRZR steam space to be compressed, which causes the Pressure go up. The pressure controller opens both spray valves until pressure stabilizes. While pressurizer level is increasing and all backup heaters are on, pressure will continue to increase and spray valves will continue to open.

B - Incorrect. First part correct (see A).

Second part incorrect. Plausible, since subcooled water has insurged into the pressurizer, and a cooler steam space would cause pressure to decrease, but the compression of the steam space in the pressurizer due to the increasing level raises pressure and overrides the cooler temperature of the pressurizer liquid which would tend to lower pressure.

C - Incorrect. First part incorrect. Plausible, due to the pressure going up. This automatically turns off all Backup heaters unless there is a > 5% high level deviation as in this case.

Second part is correct (see A).

D - Incorrect. First and second parts incorrect. Plausible, since an error in the second part (thinking that the subcooled water would drop pressure in the pressurizer) combined with a miscalculation of program level, or using the 100% value of program level, would indicate a PRZR level less than program and pressure low. These errors would cause this choice to be selected.

Second part is incorrect (see B).

ARP-1.8, Version 33.0
Drawing D175037 Sheet 2

Ran this malfunction on the simulator laptop: IC-38, 27% Power, when PRZR level increased to 6% above program level all Backup heaters were on and spray valve demand had increased from the initial value of 6.6% to 20% open. The spray valves continued to open further for several more minutes.

Previous NRC exam history if any:

010K1.03

010 Pressurizer Pressure Control System

K1 Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.03 RCS 3.6 3.7

Match justification: A CVCS malfuction in this question causes excess mass in the RCS which causes an insurge into the pressurizer. This causes a level deviation in the pressurizer which energizes the pressurizer heaters even though pressure is high, and opens spray valves due to the steam space compression and rising pressure, even though the insurge water is subcooled.

Objective:

11. Given a set of plant conditions, **LIST AND DESCRIBE** the actions/effects that will occur following a CVCS Malfunction with no operator action (OPS-52201H15).

16. 011K 5.05 001/NEW/RO/C/A 2.8/3.1/N/N/3/EDITORIAL/SAT

Unit 1 is at 100% power, and the following conditions exist:

- PRZR LVL CONT CH, LS/459Z, is in the "I/II LT459/60" position.
- A failure of the controlling pressurizer level transmitter has occurred.
- AOP-100, Instrumentation Malfunction, is in progress.
- Tavg is 573.0°F.
- FK-122, CHG FLOW, controller is in Manual.
- Przr level is 40% and rising.
- Charging flow is 125 gpm.
- Letdown flow is 130 gpm.

	1A	1B	1C
• Seal Injection flows are:	8.1 gpm	7.9 gpm	8.0 gpm
• Seal Leakoff Flows are:	2.9 gpm	3.0 gpm	3.1 gpm

Which one of the following is the:

1) approximate time that it will take for the Pressurizer level to get to program level **at the current rate in Manual control**,

and

2) the correct switch position for PRZR LVL CONT CH LS/459Z IAW AOP-100?

	Time	Switch position
A.	56 Minutes	I/III, LT459/61
B.	94 Minutes	I/III, LT459/61
C✓	56 Minutes	III/II, LT461/60
D.	94 Minutes	III/II, LT461/60

Changed due to confusion as to where the key information was written into the question. CE requests we walk him through this failure and show how indications are caused from this failure and stem conditions.

A - Incorrect. The time is correct (see C). The second part is incorrect, since LT-459 was the controlling channel, and it needs to be selected completely out by selecting III/II, III/II, LT461/60. This will place the remaining two operable LTs in service for Pressurizer level control. Plausible, since confusion may exist as to which of the two selected channels controls pressurizer level and which performs other control functions in relationship to the switch position. For example, if LT-460 was the failure this would be correct.

B - Incorrect. The time is incorrect, since the level of the pressurizer in percent is a volume affected by the specific volume at normal Operating Pressure Pressurizer Temperature (about 648°F). Plausible, since the pressurizer curve lists the change in level from 40-50% at 93 gals / %, but charging 100°F water of 56 gal volume will expand to 93 gallons for a 1% rise. The second part is incorrect (see A).

C - Correct. The time is correct, with a 100% program przr level of 50.2%, since a properly performed flow balance calculation shows that there is 10 gpm more charging into the RCS (in Charging and Seal inj minus the seal leakoff) than is leaving (in Letdown). Due to the specific volume of water at charging system temperature (about 100°F), which expands to pressurizer temperature (about 650°F), 56.3 gallons of charging water will equal 1% in the pressurizer (Per STP-9.0, RCS Leakrate determination). The controlling channel is given as failed, and that is LT-459 in the given switch position. The switch is directed by AOP-100 to be placed in a position that will not use channel I, LT-459.

(1 min/10 gals) * (56.3 gals/%) * (10%) = 56.3 mins
This gallons/% level relationship is also verified by steam table Specific Volume calculation (see below).

D - Incorrect. The time is incorrect (see B). The second part is correct (see C).

At 100°F, the specific volume of water is 0.016130 ft3/lbm per the steam tables, and it would expand to 0.02657 ft3/lbm @ pressurizer temperature of 648°F. The charging water of 56 gals/min will expand to 93.5 gals/min at PRZR temperature (which is 1% in the Pressurizer per Tank Curve 42). In approximately 56 minutes, at 10 gpm net Charging flow into the RCS, the pressurizer level will rise 10%.

Tank Curve: Unit 1 Volume II Curve 42 (Hot Calibrated)
40% level=4403.04 gals
50% level=5336 gals

93.5 gals/% PRZR level Hot calibrated (650°F)

Per Steam Table:

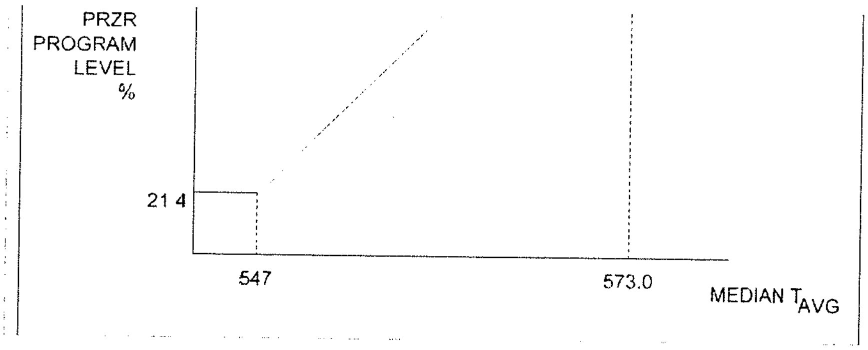
0.016130 ft^3/lbm @100°F

0.02249 ft^3/lbm @ 575°F

0.02657 ft^3/lbm @648°F

(Charging water expands in the RCS, which causes a pressurizer insurge, which in turn expands further in the pressurizer).

PRESSURIZER PRESSURE AND LEVEL CONTROL, OPS-62201H, OPS-52201H, ESP-52201H, Student Text, Figure 8



Previous NRC exam history if any:

011K 5.05

011 Pressurizer Level Control System

K5 Knowledge of the operational implications of the following concepts as they apply to the PZR LCS: (CFR: 41.5 / 45.7)

K5.05 Interrelation of indicated charging flow rate with volume of water required to bring PZR level back to programmed level hot/cold 2.8 3.1

Match justification: This question requires knowledge of determining what the net charging flow into the RCS is, and then determining the time for the pressurizer level return to program setpoint. The pressurizer level program value has been provided to ensure it is clear which program level is being used in this question. Program level changes each cycle and with changes in Tavg throughout each operating cycle, so it is provided. To obtain 3 plausible but incorrect distractors, a second part was added to test the system knowledge of the Pressurizer control system selector switch.

Objective:

5. **DEFINE AND EVALUATE** the operational implications of abnormal plant or equipment conditions associated with the operation of the Pressurizer Pressure and Level Control System components and equipment to include the following (OPS-52201H07):

- Normal Control Methods
- Abnormal and Emergency Control Methods
- Automatic actuation including setpoint, if applicable
- Protective Interlocks

Actions needed to mitigate the consequence of the abnormality

17. 012K2.01 001/FNP BANK/RO/C/A 3.3/3.7/N/N/3/NO CHANGE OK/SAT

A loss of 'A' Train Auxiliary Building 125V DC Bus has occurred on Unit 1.

If the plant experienced a problem which required manually tripping the reactor, which one of the following describes the effect (on any closed Reactor Trip and/or Bypass breakers) of placing the RX TRIP ACTUATION switch on the MCB to TRIP?

Placing the MCB handswitch in TRIP would ________ if they were closed.

A.✓ open ALL reactor trip and bypass breakers.

B. ONLY open the 'A' reactor trip breaker and the 'B' reactor trip bypass breaker.

C. ONLY open the 'B' reactor trip breaker and the 'A' reactor trip bypass breaker.

D. open BOTH reactor trip breakers but NOT open either reactor trip bypass breaker.

no change. Answer choices require more than one possible initial condition of RT & RTB bkr positions and tests the possible outcomes with a loss of one train of DC power.

A - Correct. Aux Building DC power is not required to trip open breakers, as long as the UV coils are deenergized by Solid State (SSPS). Voltage from SSPS feeds the 48V UV coils that will allow the trip breakers to open when power is removed (a trip signal deenergizes the UV coils). Loss of "A" train AB DC would prevent the closure of the A RTB & B RTBYP breakers, AND would prevent the shunt trip coils on the A RT & B BYP breakers from being energized to provide an additional trip signal. SSPS power is from the inverters which supply power from the Regulated AC, bypassing the inverters, if AB DC is lost.

B - Incorrect. See A. Plausible, since the Reactor Trip and Bypass breakers are operated and tripped by opposite trains. However, these two breakers are both operated by the B train aux building DC, and not the A train. Also, the Shunt trip coils operate to trip these breakers and the coils get power from AB DC (B train). However, the UV coils can still deenergize if needed and trip all of the reactor trip breakers. Confusion may exist as to which train of breaker is operated by which train of DC, AND as to which type of DC is needed to trip the breaker (UV coil 48V or Shunt Trip coil 125 V).

C - Incorrect. See A. Plausible, since the Reactor Trip and Bypass breakers are operated and tripped by opposite trains, AND these two breakers are both operated by the A train aux building DC. Also, the Shunt trip coils operate to trip these breakers and the coils get power from A train AB DC. However, the UV coils can still deenergize if needed and trip all of the reactor trip breakers. Confusion may exist as to which type of DC is needed to trip the breaker (UV coil 48V deenergizing or Shunt Trip coil 125 V energizing).

D - Incorrect. See A. Plausible, since the AB DC does supply the shunt trip coils, and only the local pushbutton energizes the Shunt trip coil to trip the Bypass breakers, so there is a difference in the way the trip breakers and the bypass breakers work for loss of AB DC. However, the UV coil will still trip all RT & BYP breakers if a manual trip is called for. Confusion may exist as to the redundant methods using the UV and Shunt Trip coils to trip the reactor.

Reactor Protection Functional System Diagram (FSD) A181007, section 3.3.2

Each circuit breaker shall be equipped with a 48 volt DC instantaneous undervoltage trip device and a 125 Vdc shunt trip device. (Reference 6.4.086) The Shunt Trip Attachment coil shall operate on 125 Vdc and function as a backup for the undervoltage trip device.

The first method of tripping the breaker (i.e., reactor trip or bypass breakers) is by a loss or drop of rated voltage to the Undervoltage Relay (UV). **The relay is normally energized from the 48 volt DC from the RPS.** When the voltage is removed by an automatic reactor trip signal, the relay is de-energized and releases the UV trip lever, which actuates the trip shaft, causing the breaker to unlatch from the closed position. The second method of tripping the trip shaft is by the shunt trip lever when the normally de-energized shunt trip (SHTR) coil is energized. **When energized, the SHTR coil is powered from the 125 volt DC system used to close the reactor trip and bypass breaker closing circuits.**
For the reactor trip bypass breaker, the SHTR relay is energized only by a manual pushbutton. After the reactor trip bypass breaker is opened, then a contact in series with the SHTR relay opens to de-energize the coil. Thus, the SHTR relay is only momentarily energized.

For the reactor trip bypass breaker, **the SHTR relay is energized only by a manual pushbutton.** After the reactor trip bypass breaker is opened, then a contact in series with the SHTR relay opens to de-energize the coil. Thus, the SHTR relay is only momentarily energized.

Train A of the reactor protection system powers the UV and Shunt Trip coils for RTA and BYB, and train B powers the UV and Shunt Trip coils for RTB and BYA per Reactor Protection Functional System Diagram (FSD) A181007, Figure F-1.

Previous NRC exam history if any:

012K2.01
012 Reactor Protection System
K2 Knowledge of bus power supplies to the following: (CFR: 41.7)
K2.01 RPS channels, components, and interconnections 3.3 3.7

Match justification: The 125V Aux Building DC busses supply the Reactor Trip Breakers and Bypass Breakers (RPS components). They provide power to the Reactor trip breakers for both closing power and one of the sources of power for tripping the breakers. To correctly answer this question, the power supplies to the Reactor Trip breakers must be understood, including the A train 125V Aux Building DC bus.

Objective:

2. **RELATE AND DESCRIBE** the operation of the Reactor Trip Breakers and Reactor Trip Bypass Breakers to include the operation of the following :(OPS-40302F02):

   Shunt Trip Coils

   Undervoltage Coils

1. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the following components associated with the Reactor Protection System (RPS) (OPS-52201I02):

   - Solid state protection system (SSPS) cabinets (A train/B train)
   - Input relay cabinets
   - Logic cabinets
   - Output relay cabinets
   - Safeguards test cabinets
   - Reactor trip breakers
   - Reactor trip bypass breakers

18. 012K6.03 001/MOD/RO/MEM 3.1/3.5/N/N/3/REVISED/FIX

Unit 2 is at 100% power, and the following conditions occurred:

- PT-455, PRZR PRESS, has failed off-scale HIGH.
- All bistables associated with PT-455 in Table 1 of AOP-100, Instrument Malfunction, have been placed in the trip condition per Tech Spec 3.3.1.

Which one of the following identifies the **MINIMUM** additional bistable channels required to meet the coincidence for RPS and ESF actuation logic to initiate any reactor trip and any safety injection on Pressurizer Pressure?

A **MINIMUM** of __(1)__ additional Pressurizer Pressure channel(s) failing **HIGH** will cause a Reactor Trip,

and

a **MINIMUM** of __(2)__ additional Pressurizer Pressure channel(s) failing **LOW** will cause a Safety Injection.

	(1)	(2)
A.✓	1	1
B.	1	2
C.	2	1
D.	2	2

changed "additional channels" to "additional bistable channels" and added "coincidence for" to make it clearer that either a "parameter exceeding a setpoint" or a "channel failure" that actuated a bistable was being asked about. Also, changed "or" to "and" in the question for consistency.

Added "All bistables associated with PT-455 in Table 1 of AOP-100, Instrument Malfunction, have been placed in the trip condition per Tech Spec 3.3.1" per CE suggestion.

The reason for the specific wording is that AOP-100 says to refer to Tech Specs for LCO requirements that exist. The AOP-100 does not say to take those actions. By not saying those actions have been taken sets up the candidate for two possible correct answers.

Also the RO does not have to memorize the actions of 72 hour LCO requirements of Tech Specs which this is

A - Correct. AOP-100 directs actions to place all bistables in trip, including the Low pressure RT and SI bistables. These actions provide 1/3 coincidence for both the RT and SI. One more Pressure channel failing low will provide the 2/3 coincidence for both the RT and SI.

B - Incorrect. The first part is correct (see A). The second part is incorrect, but plausible if the applicant does not realize that AOP-100 directs all bistables tripped. Since there is no High pressure SI, the failure alone of PT-455 High would not provide even 1/3 coincidence for an SI, so 3 additional failures would be required for an SI to occur.

C - Incorrect. First part is incorrect, but plausible since it is correct for a low pressure reactor trip prior to actions of AOP-100. However, the High pressure reactor trip has one channel already tripped, and one additional channel will give a reactor trip signal. The second part is correct (see A).

D - Incorrect. Both parts are incorrect, but plausible since for a low pressure reactor trip and low pressure SI it is correct prior to the actions of AOP-100. The applicant may not realize that all bistables are tripped in AOP-100, and forget about the high pressure Reactor trip.

Previous NRC exam history if any:

012K6.03

012 Reactor Protection System

K6 Knowledge of the effect of a loss or malfunction of the following will have on the RPS:

(CFR: 41.7 / 45/7)

K6.03 Trip logic circuits 3.1 3.5

Match justification: A channel failure in one direction (failing high) causes a loss of the potential for meeting coincidence in the opposite direction (failing low) from that channel. This is one way of losing a trip logic circuit for one of the channels. This question presents a scenario where one of the 3 required trip logic coincidence circuits for Pressurizer pressure is lost, and knowledge of the effect on the RPS system is required to answer the question.

Objective:

1. **RECALL AND 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 (RPS) and Engineered Safeguards Features (ESF) to include setpoint, coincidence, rate functions (if any), reset features, and the potential consequences for improper conditions to include those items in the following tables (OPS-52201I07):

   - Table 1, Reactor Trip Signals
   - Table 2, Engineered Safeguards Features Actuation Signals
   - Table 5, Permissives
   - Table 6, Control interlocks

19. 013K2.01 002/NEW/RO/MEM 3.6/3.8/N/N/3/CVR/SAT

A loss of B Train Auxiliary Building 125V DC Bus has occurred on Unit 1.

Which one of the following is the correct impact on B Train ESF Equipment control?

The B Train SI actuated MOVs __(1)__ automatically stroke upon an SI actuation, and

B Train ESF pumps __(2)__ be started in LOCAL at the HSP.

	(1)	(2)
A.	will	can
B.✓	will	can **NOT**
C.	will **NOT**	can
D.	will **NOT**	can **NOT**

A - Incorrect. The first part is correct (see B), however The second part is not correct, these breakers receive control power from B train DC and although there is an alternate control power that is placed into the circuit when in “LOCAL” at the HSP, it is also from B train DC.
Plausible, since B train has alternate control power and SOP-36.6, CIRCUIT BREAKER RACKING PROCEDURE, has numerous cautions about an additional Control Power source for the B train ESF Pumps. The existance of Alternate Control power may cause confusion as to the ultimate source of the alternate control power.

B - Correct. The B Train SI actuated (that would normally stroke upon an SI actuation signal) **MOVs** can be operated with or without DC power (a separate DC power source is provided to some of these valves if equipped with a disconnect for position indication only-- and these valves do not stroke automatically following an SI actuation because of the "normal" position of that disconnect--ie MOV8808B). The control power for operation comes from the **600V AC** supply for each MOV via transformer.
The control power for operation of the B Train ESF breakers is supplied from B Train 125V Aux Bldg DC. Although equipped with an alternate control power source, that power is also supplied form B train DC on another breaker with a different cable run (for Appendix R concerns).

C - Incorrect. The first part is incorrect (see B).
Plausible, since some of these MOVs are equipped with a DC power supply for indication (MOV8808B). Further plausibility is provided from many solenoid operated valves auto stroke after SI, and they usually require DC power (although for the opening)
The second part is also incorrect (see A).

D - Incorrect. The first part is incorrect (see C). The second part is correct (see B).

Previous NRC exam history if any:

013K2.01
013 Engineered Safety Features Actuation System
K2 Knowledge of bus power supplies to the following: (CFR: 41.7)
K2.01 ESFAS/safeguards equipment control . . . .
3.6* 3.8

Match justification: ESF equipment (pump) control requires DC for Pump breaker operation and breaker indication, even though the components themselves are powered from AC. ESF MOVs are powered from the 600V MCC AC and get control power for valve position indication from the same MCC AC source. To answer this question correctly knowledge of the power supplies for these ESF control functions is required.

Wrote this question to intentionally stay away from 120V vital AC Instrumentation power due to potential overlap with other questions on this exam.

Objective:

1 **NAME AND IDENTIFY** the Bus power supplies, for those electrical components associated with the Emergency Core Cooling System, to include those items in Table 4- Power Supplies (OPS-40302C04).

2. **RELATE AND DESCRIBE** the effect(s) on the Emergency Core Cooling System for a loss of an AC or DC bus, or a malfunction of the Instrument Air System (OPS-40302C06).

20. 013K5.01 001/NEW/RO/C/A - 2.8/3.2/N/N/3/CVR/SAT

Unit 2 was operating at 100% power, and the following conditions have occurred:

- PT-950, CTMT PRESS, has failed.
- PT-950, HI-3 bistable, is in the BYPASS condition.
- Subsequently, the 2D vital panel has become de-energized.

If a Large Break LOCA occurs, which one of the following describes:

1) the number of channels of Hi-3 bistables which will be actuated

and

2) the number of trains of containment spray (CS) that actuate automatically?

A.✓ 1) Two channels ONLY will be actuated.

2) One train ONLY will actuate.

B. 1) Two channels ONLY will be actuated.

2) Two trains will actuate.

C. 1) Three channels will be actuated.

2) One train ONLY will actuate.

D. 1) Three channels will be actuated.

2) Two trains will actuate.

A - Correct. The channel I bistable is bypassed and won't actuate. The channel IV Bistable won't actuate since it is deenergized by the loss of 1D vital 120V AC, and it is an energize to actuate bistable. Even though the coincidence for CS actuation would still be met with channel II & III, and both trains of SSPS would get the signal to actuate both trains of CS, the slave relays are deenergized in SSPS train B due to the loss of 1D vital 120V AC panel. This would prevent Train B CS from actuating.

B - Incorrect. The first part is correct (see A). The second part is incorrect, but plausible since both trains of SSPS get a signal to initiate CS, even with a loss of 1D 120V vital AC panel. The master relays call for an actuation on Both trains, but on B train the slave relays don't have power to start the loads and operate the valves for the ESF actuations.

C - Incorrect. The first part is incorrect, but plausible. For most bistables, when they lose power they deenergize to actuate. Containment spray is an exception to this general rule. Channel IV is thus deenergized and will NOT actuate. Examinee could also not realize that the bypass function (which prevents the bistable from actuating) is opposite the usual trip bistable function (which causes the bistable to trip) for a loop in maintenence. This would cause this choice to be selected. The second part is correct (see A).

D - Incorrect. Both parts are incorrect (See C & B).

FSD: A181007 REACTOR PROTECTION SYSTEM

2.2.2 The RPS system is housed in **two physically and electrically independent equipment <u>trains</u> (Train "A" and Train "B")**, typically referred to as the Solid State Protection System (SSPS) cabinets. (Reference 6.7.003)
2.2.3 Any single failure within the RPS system (sensor channel or actuation train) shall not prevent the redundant system actuation. **On loss of <u>channel or train</u> power the bistable shall be tripped. The only exception to the loss of channel or train power causing the bistable to trip is for Containment Spray and Containment Phase B Isolation where the bistables must energize to actuate.** (References 6.1.002, 6.1.41, 6.7.014,)

2.2.6 Instrument <u>channels</u> shall be powered from four separate independent AC instrument distribution panels. These panels shall be fed from four separate and independent Class 1E inverters. (References 6.1.023, 6.4.081, 6.4.091, 6.7.014, 6.7.016)

2.2.15 Except as noted below, all reactor trip and safeguards actuation **<u>channels</u>** shall be placed in the trip mode when the channel is out of service for any reason. The reactor trip and safeguards actuation **circuits noted below be administratively bypassed** for maintenance on a single channel.
1. Source range high neutron flux trip
2. Intermediate range high neutron trip
3. High 3 containment pressure actuation of containment spray

LOAD LIST: A-506250, Page G-39, 1D 120V Vital AC Dist PNL.

Previous NRC exam history if any:

013K5.01

013 Engineered Safety Features Actuation System

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

(CFR: 41.5 / 45.7)

K5.01 Definitions of safety train and ESF channel 2.8 3.2

Match justification: To answer this question correctly, knowledge is required of what constitutes a safety train, an ESF channel, and the operational implications of each must be understood.

Objective:

1. **RECALL AND 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 (RPS) and Engineered Safeguards Features (ESF) to include setpoint, coincidence, rate functions (if any), reset features, and the potential consequences for improper conditions to include those items in the following tables (OPS-52201I07):

   - Table 1, Reactor Trip Signals
   - Table 2, Engineered Safeguards Features Actuation Signals
   - Table 5, Permissives
   - Table 6, Control interlocks

21. 015AK2.10 001/NEW/RO/C/A - 2.8/2.8/N/N/2/EDITORIAL/FIX

Unit 1 is at 20% power and conditions are as follows:

At 1000:

	1A	1B	1C
• RCP amps :	670	680	690

At 1005:

	1A	1B	1C
• RCP amps:	670	680	0

- EF3, 1C RCS LOOP FLOW LO OR 1C RCP BKR OPEN, is in alarm.

Which one of the following describes the expected indications on 1A RCS LOOP and 1C RCS LOOP **stable** flow rates at 1010?

	1A RCS LOOP Flow rate	1C RCS LOOP Flow rate
A.	105%	0%
B.✓	105%	10%
C.	105%	< 0%
D.	100%	0%

changed C from 100% & 10% to 105% & <0% to enhance plausibility. Changed D to 100% & 0% per CE suggestion on phone. Took Stable out of each choice and put in stem.

A - Incorrect. The first part is correct (See B). The second part is incorrect, but plausible, since the RCP amps are 0, and it is tripped as indicated by the bkr light and amps. If it weren't for reverse flow caused by the discharge pressure of the other two pumps 0% would be correct.

B - Correct. Each of the two loops with forced flow provide some backflow through the tripped pump (approx 5% each for a total of 10%). The tripped pump has an indicated flow (approximately 10%) due to the flow indicator sensing a positive value of flow >0, even though the direction of flow is reversed.

C - Incorrect. The first part is correct (see B). The second part is incorrect, but plausible. If a misunderstanding of the type of flow indicator existed along with a proper understanding that there is going to be reverse flow in the idle loop, this choice would be selected.

D - Incorrect. Both parts are incorrect but plausible. The first part is incorrect but plausible due to the amps the RCP remaining unchanged. The second part is incorrect, but plausible. The 0gpm value would be correct for some types of flow instruments with backwards system flow, and would seem to be indicated by the amps and the fact that the 1C pump is not pumping any flow due to being tripped as indicated by annunciator indication. However the piping system of the RCS allows back flow into the C loop in this condition and flow is indicated on all three loop flowmeters.

Ran on simulator laptop to verify flows. No technical document was found which stated this characteristic in writing. Loss Of Reactor Coolant Flow, OPS-62520D OPS-52520D, Student Text– Version 2, listed this. Cvr 8-4-09

Previous NRC exam history if any:

015AK2.10

015 Reactor Coolant Pump Malfunctions

AK2. Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: (CFR 41.7 / 45.7)

AK2.10 RCP indicators and controls 2.8* 2.8

Match justification: This question provides indications which accompany a RCP trip and must be recognized as such. The knowledge of the interrelations between the RCP loss of flow and the flow indicators (and what to expect the RCP flow indicators to read after a RCP trip) must be used to obtain the correct answer.

Objective:

2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the components associated with the Reactor Coolant System (RCS) to include the components found on Figure 1, Reactor Coolant System (OPS-40301A02).

22. 022A1.01 001/NEW/RO/MEM 3.6/3.7/N/N/3/EDITORIAL/FIX

Unit 2 is at 100%, and the following conditions occurred:

At 1000:

- The Containment Cooling system is in the normal mode of operation per SOP-12.1, Containment Air Cooling System.
- Containment temperature is slowly rising.

At 1100:

- The crew has configured the containment cooling system per SOP-12.1.
- The emergency service water from CTMT coolers; MOVs 3024A, B, C and D are OPEN per SOP-12.1.
- Containment temperature is 115°F and slowly rising.

Which one of the following correctly states whether or not an ACTION STATEMENT is required to be entered for Tech Spec 3.6.5, Containment Temperature, and the required Containment Cooling Fans speed per SOP-12.1?

An ACTION STATEMENT for TS 3.6.5 is __(1)__ to be entered,

and

the Containment Cooling Fans are required to be operated in __(2)__ speed per SOP-12.1.

	(1)	(2)
A.	required	FAST
B.	required	SLOW
C.✓	NOT required	FAST
D.	NOT required	SLOW

Added "• Containment temperature is 115°F and slowly rising." per CE suggestion. Changed wording of question and first part of each choice for stem focus per CE suggestion.

A - Incorrect. The 110°F is incorrect for Ctmt temp limit, but is plausible, since 110°F is the temperature per the CTMT HI TEMP alarm ARP-1.2, BB3, to start all ctmt dome recirc fans in fast speed. The second part is correct per SOP-12.1 step 4.1.6 note.

B - Incorrect. The first part is incorrect (See A). The second part is incorrect but plausible, since Slow is the speed that the fans automatically shift to in a very high temperature LOCA environment. However, it is due to the humidity rather than the heat that they are shifted to slow to protect the fans. Under normal conditions, the lower humidity allows fast speed operation to remove more heat from containment.

C - Correct. 120 degrees F is the TS 3.6.5 limit. Fans must be operated in fast per note prior to step 4.1.6 normally to maintain ctmt less than this limit per SOP-12.1, Step 4.1.9-4.1.11, ver. 37.0.

D - Incorrect. The first part is correct (See C). The second part is incorrect (See B).

ARP-1.2, BB3, CTMT AIR TEMP HI, Version 44.0

OPERATOR ACTION

5. IF containment average air temperature is greater than 110°F, THEN verify containment dome recirc fans in service on fast speed.

Previous NRC exam history if any:

022A1.01

022 Containment Cooling System

A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: (CFR: 41.5 / 45.5)

A1.01 Containment temperature 3.6 3.7

Match justification: Question asks what the TS containment temperature limit is, and which controls of the Containment cooling system must be operated (fast or slow speed fans) to prevent exceeding the containment temperature limit.

Objective:

7. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Containment Ventilation and Purge System components and equipment, to include the following (OPS-40304A07):

- Normal control methods

Abnormal and Emergency Control Methods

Automatic actuation including setpoint (example SI, Phase A, Phase B, MSLIAS, LOSP, SG level)

Protective isolations such as high flow, low pressure, low level including setpoint

Protective interlocks

23. 022AK3.06 001/NEW/RO/MEM 3.2/3.3/N/N/4/CVR/SAT

Unit 1 Reactor has just tripped, and the following conditions occurred:

- All three RCPs have just tripped.
- All Charging has been lost.

Which one of the following correctly states the reason for maintaining CCW cooling flow to the Thermal Barrier HX in this condition?

Maintaining CCW cooling flow to the Thermal Barrier HX will prevent the RCP __(1)__ from starting to degrade due to overheating in as early as __(2)__.

	(1)	(2)
A.	#1 seal	2 minutes
B.✓	#1 seal	13 minutes
C.	lower radial bearing	2 minutes
D.	lower radial bearing	13 minutes

A - Incorrect. Incorrect, since the seal area takes time to void of the cooler water prior to allowing the hotter RCS water to the seal area. The WOG background document for ECP-0.0 gives 13 minutes for the time to degrade the RCP #1 seal after losing CCW to the thermal barrier and seal injection. The #1 seal degrading is the correct concern and reason, but the 2 minutes is incorrect. Plausible, because with loss of CCW to the motor oil coolers the upper and lower MOTOR bearing (but not the radial bearing) can overheat in a maximum of 2 minutes per UOP-1.1 Step 3.10 (P & L). And, the lower RADIAL bearing would heat up in the event that both Seal Injection and CCW Thermal Barrier cooling were lost, but that is not the limiting concern or reason.

B - Correct. The WOG background document for ECP-0.0 gives 13 minutes for the time to degrade the RCP #1 seal after losing CCW to the thermal barrier and seal injection.

C - Incorrect. The bearing is incorrect, since it is the #1 seal that is the limiting concern. Plausible, since the bearing will heat up in the condition given without CCW cooling to the Thermal Bearing, but the #1 seal is the limiting condition. The time is incorrect also, since the 13 minutes is given in the WOG background document for ECP-0.0. Plausible, since the 2 minutes would apply to an overheat RCP manual trip criteria in 2 minutes or less for a lower motor bearing if motor oil CCW cooling is lost (but not for lower RADIAL bearing).

D - Incorrect. The first part is incorrect (see C). The second part is correct (see B).

WOG Background Document FNP-0-ECB-0.0, LOSS OF ALL AC POWER, Plant Specific Background Information (pgs 39 & 40 of 88).

Isolating the RCP thermal barrier CCW return outside containment isolation valve prepares the plant for recovery while protecting the CCW system from steam formation due to RCP thermal barrier heating. Following the loss of all ac power, hot reactor coolant will gradually replace the normally cool seal injection water in the RCP seal area. As the hot reactor coolant leaks up the shaft, the water in the thermal barrier will heat up and potentially form steam in the thermal barrier and in the CCW lines adjacent to the thermal barrier. Subsequent automatic start of the CCW pump would deliver CCW flow to the thermal barrier, flushing the steam into the CCW system. If abnormal RCP seal leakage had developed in a pump, the abnormally high leakage rate could exceed the cooling capacity of the CCW flow to that pump thermal barrier and tend to generate more steam in the RCP thermal barrier CCW return lines. Isolating these lines prevents the potential introduction of this steam into the main portion of the CCW system upon CCW pump start. This keeps the main portion of the CCW system available for cooling equipment necessary for recovering the plant when ac power is restored.

Knowledge: 1. RCP seal integrity concerns following loss of ac power (See Subsection 2.1).
2. Analyses of RCP seal performance following a loss of all seal cooling estimate that **increased seal leakage may begin as early as 13 minutes due to seal degradation at high fluid temperatures.** It is important to establish sufficient backpressure in the seal leakoff line by isolating the seal return line before seal degradation occurs in order to limit RCP seal leakage. The time of 13 minutes was determined in WCAP-10541 as the time when "...**the lower pump internals volume will be completely purged and the seal area water temperature will be approaching the 550°F reactor coolant temperature.**"

FSD, CVCS/HHSI/ACCUMULATOR/RMWS, A-181009
2.2.3.2
This capability satisfies the seal water requirement for the RCP No. 1 seal. A portion of the seal injection flow (nominally 5 gpm per pump) enters the RCS through the labyrinth seals and the thermal barrier. This in-leakage precludes leakage of reactor coolant through the No. 1 seal during normal operation. The remainder of the seal injection flow (nominally 3 gpm) flows up the pump shaft, cooling the pump lower bearing and the No. 1 seal. The 5-micron filtration requirement is based upon RCP minimum seal clearances. (Reference 6.2.44)

FNP-1-SOP-1.1, Version 40.0
3.10 IF CCW flow to the RCP motor bearing oil coolers is lost, THEN pump operation may be continued until the motor upper or lower bearing temperature reaches 195°F (approximately 2 minutes after cooling water flow stops).

Previous NRC exam history if any:

022AK3.06

022 Loss of Reactor Coolant Makeup

AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: (CFR 41.5, 41.10 / 45.6 / 45.13)

AK3.06 RCP thermal barrier cooling 3.2 3.3

Match justification: To correctly answer this question, knowledge is required of the reason for requiring RCP thermal barrier cooling during a loss of all RCS makeup (which would include a loss of seal injection).

Objective:

2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the components associated with the Reactor Coolant System (RCS) to include the components found on Figure 1, Reactor Coolant System (OPS-40301A02).

4. **LABEL AND ILLUSTRATE** Reactor Coolant System (RCS) flow paths to include the components found on Figure 1, Reactor Coolant System (OPS-40301A05).

24. 026AA1.01 001/NEW/RO/C/A 3.1/3.1/N/N/2/REVISED/FIX

Unit 1 is in Mode 3, and the following conditions occurred:

- A loss of both Trains of CCW has occurred.
- AOP-9.0, Loss Of Component Cooling Water, is in progress.
- Attachment 1, Establishing Firewater Cooling to a Charging Pump, is in progress.

Which one of the following states the CCW temperature at which isolation of the CCW return from the RCP thermal barrier is required, and the number of charging pumps which will be aligned to Firewater Cooling IAW AOP-9.0, Attachment 1?

Isolating the CCW return from the RCP thermal barriers is required when CCW temperature is greater than (1) ,

and

Firewater Cooling will be aligned to (2) charging pump(s) IAW AOP-9.0, Attachment 1.

	(1)	(2)
A.	105°F	1
B.✓	130°F	1
C.	105°F	3
D.	130°F	3

Rewrite per CE suggestion.

A - Incorrect. First part is incorrect but plausible, since this is the temperature allowed during normal operation per SOP-23 Precautions and Limitations. The second part is correct (see B).

B - Correct. AOP-9.0 Attachment 1 Step 3.2 states that 130°F is the maximum CCW temperature allowed prior to isolating CCW return from RCP thermal barrier. Note prior to step 1 of Attachment 1 of AOP-9.0 states that aligning Firewater to one Charging pump is the purpose the attachment: The purpose of this attachment is to align the fire protection water system to replace CCW as the cooling water source for a CHG PUMP so that long term RCP seal injection and RCS makeup are available.

C - Incorrect. The first part is incorrect (see A). The second part is incorrect but plausible. CCW has been lost to all the charging pumps, aligning Firewater cooling to all three would be physically possible, and would seem desirable for maximum flexibility, but the procedure directs only one charging pump to be aligned to firewater.

D - Incorrect. The first part is correct (see B). The second part is incorrect (see C).

SOP-9.0, ATTACHMENT 1, ESTABLISHING FIREWATER COOLING TO A CHARGING PUMP

Page 3 of 12

3.2 WHEN CCW temperature greater than 130°F,
THEN isolate CCW return from RCP thermal barrier.

Previous NRC exam history if any:

026AA1.01

026 Loss of Component Cooling Water

AA1. Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: (CFR 41.7 / 45.5 / 45.6)

AA1.01 CCW temperature indications 3.1 3.1

Match justification: A loss of CCW is provided in the question, and rising temperature values are given. Knowledge is required to answer the question of how to operate as a result of the rising CCW temperature indications.

Objective:

2. **EVALUATE** plant conditions and **DETERMINE** if entry into AOP-9.0, Loss of Component Cooling Water is required. (OPS-52520I02)

6. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing AOP-9.0, Loss of Component Cooling Water. (OPS-52520I06).

25. 026K1.01 001/NEW/RO/MEM 4.2/4.2/N/N/3/CVR/SAT

A Unit 1 Safety Injection is in progress due to a Large Break LOCA.

Which one of the following describes the connection(s) between the RWST, A Train CS and ECCS pumps suction, and the operation of MOV-8827A and MOV-8826A, CTMT SUMP TO 1A CS PUMP valves?

A Train CS Pump, A Train HHSI Pump, and the A Train RHR Pump have __(1)__ suction header(s) penetrating the RWST,

and

the CS Sump suction valves __(2)__ automatically open on a LO-LO RWST condition.

	(1)	(2)
A.	separate	will **NOT**
B.	one common	will
C.	separate	will
D✓	one common	will **NOT**

A -. Incorrect. The first part is incorrect, but plausible since most of the safety related equipment has physical train separation for piping. The RWST is designed to minimize tank penetrations, and uses only one penetrations for suctions to all CS pumps, RHR pumps, and CVCS/HHSI pumps. The second part is correct.

B - Incorrect. The first part is correct. The second part is incorrect, but plausible since this would be correct for the RHR sump suctions which have the auto function described.

C - Incorrect. The first part is incorrect (See A). The Second part is incorrect (See B).

D - Correct. The RWST is designed to minimize tank penetrations, and uses only one penetrations for suctions to all CS pumps, RHR pumps, and CVCS/HHSI pumps. The CS Sump suction valves do not have the auto open feature, but the RHR sump suctions do.

Previous NRC exam history if any: n/a

026K1.01

026 Containment Spray System

K1 Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.01 ECCS 4.2 4.2

Match justification: The only physical connection between the CS system and the ECCS system is at the RWST suction of the pumps, which is tested in the first part of each choice. The second part of the distractor contrasts the design of the CSS with the ECCS system sump suction valves to provide symmetry and three plausible but incorrect distractors.

Objective:

1 **LABEL AND ILLUSTRATE** the Emergency Core Cooling System to include the components found on the following figures (OPS-40302C05):

- Figure 2, Accumulators
- Figure 3, Refueling Water Storage Tank and Figure 4, Emergency Core Cooling System
- The flow paths found on Figure 14, ECCS Injection Phase, Figure 15, ECCS Cold Leg Recirculation, Figure 16, ECCS Simultaneous Hot & Cold Leg Recirculation Normal, and Figure 17, ECCS Simultaneous Hot & Cold Leg Recirculation Alternate.

26. 027AK1.02 001/NEW/RO/C/A 2.8/3.1/N/N/3/CVR/SAT

Unit 2 is at 50% power, and PT-444, PRZR PRESS, pressure transmitter has failed to the **2230 psig** position.

Which one of the following describes the effects on PK-444A, PRZR PRESS REFERENCE controller, and the pressurizer liquid density due to this malfunction?

PK-444A controller demand goes __(1)__,

and

the density of the pressurizer liquid goes __(2)__.

	(1)	(2)
A.	down	up
B.	down	down
C.	up	up
D.✓	up	down

A - Incorrect. The first part is incorrect (see D). Plausible, since if the PT had failed 6 psig higher (above 2235 psig), the proportional integral controller would integrate the error signal down until the PORV 444B opened and the sprays opened. Also, the spray valve controllers are controlled by the "master" controller and when the pressure must be increased, the demand goes down. Confusion could exist as which controller function is being described. The second part is incorrect. Plausible, since the spray valve controllers are controlled by the "master" controller and when their demand goes up pressure goes down and the liquid density goes up. Also, steam space density does go up in this condition, and the liquid specific volume goes up (and specific volume, not density, is the value given in the steam table for the property of the liquid).

B - Incorrect. The first part is incorrect (see A). The second part is correct (see D).

C - Incorrect. The first part is correct (see D). The second part is incorrect (see A).

D - Correct. The Proportional/Integral PRZR PRESS controller senses a low pressure and the demand starts integrating higher and higher. This first causes the spray valves to close and the proportional heaters increase output. Then, the backup heaters energize. The pressurizer liquid heats up and expands (density goes down) due to the increased heat input into the pressurizer liquid. The integral part of the controller continues to add to the error signal and PORV-445A opens due to actual pressure increasing to 2235 on PT 445. The pressure cycles around the setpoint of the PORV at 2235 psig with a higher pressurizer liquid temperature.

Previous NRC exam history if any:

027AK1.02

027 Pressurizer Pressure Control System Malfunction

AK1. Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: (CFR 41.8 / 41.10 / 45.3)

AK1.02 Expansion of liquids as temperature increases 2.8 3.1

Match justification: To answer this question correctly, it must be recognized that for this particular malfunction of the PRZR Press control system, the pressurizer liquid heats up and expands due to pressurizer heaters energizing and sprays closing. The operational implications must also be understood in that this causes controller demand to go up (which would cause actual pressure go up until a PORV will lift: PORV-445A).

Objective:

2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the following components associated with the Pressurizer Pressure and Level Control System to include those items found on Figure 2, Pressurizer and Pressure Relief Tank, Figure 3, Pressurizer Pressure Protection and Control, and Figure 7, Pressurizer Level Protection and Control (OPS-52201H02).

5. **DEFINE AND EVALUATE** the operational implications of abnormal plant or equipment conditions associated with the operation of the Pressurizer Pressure and Level Control System components and equipment to include the following (OPS-52201H07):
   - Normal Control Methods
   - Abnormal and Emergency Control Methods
   - Automatic actuation including setpoint, if applicable
   - Protective Interlocks

   Actions needed to mitigate the consequence of the abnormality

27. 027K1.01 001/NEW/RO/MEM 3.4/3.7/N/N/2/CVR/SAT

Which one of the following correctly states how the Containment Spray System reduces radioactive iodine in the Containment atmosphere during a LOCA?

To enhance absorption of Iodine from the Containment atmosphere, the Containment Spray System sprays water from the __(1)__ at a pH of approximately __(2)__ .

	(1)	(2)
A.	containment sump	4.5
B.	RWST	7.5
C.✓	containment sump	7.5
D.	RWST	4.5

A - Incorrect. First part correct, see C.

second part - 4.5 pH is incorrect. Plausible, since the Borated water from the RWST in the injection phase is a pH of approx. 4.5 due to the 2300-2500 ppm borated water. However, the recirc phase begins the spray of the sump water which has the dissolved Tri-Sodium Phosphate in it, and the pH of that water is higher at 7.5 to 10.5.

B - Incorrect. The 7.5 is correct, but the RWST is acidic at a pH of 4.5 due to the high concentration of boric acid in. The low pH is not conducive to absorbing the iodine. The Iodine is absorbed during the recirc phase when the CS takes a suction on the Containment sump after the TSP has dissolved and raised the pH of the Spray water. Plausible, if confusion exists as to the need for the pH to be higher in order to absorb the Iodine out of the containment atmosphere.

C – Correct. The TSP in the Containment Sump dissolves in the Containment sump water during the injection phase, and raises the pH from about 4.5 to a range of 7.5-10.5. During the Containment Spray recirc phase, this causes the iodine in the containment atmosphere to be absorbed in the spray water and convert to a non-volatile form. Then, it stays in the sump water, and does not leak out of containment via any ctmt atmosphere leakage paths. Even though some iodine would be absorbed by the mechanical action of the spray water in the containment atmosphere, the higher pH enhances the effect, and the retention of the iodine in the sump water is due to the higher pH.

D - Incorrect. Both parts are incorrect (see A & C). Plausible, since the pH is correct for the RWST source, but this pH is not conducive to removing iodine from the containment atmosphere. Confusion may exist as to the exact mechanism of iodine removal by the CS system.

TS B3.5.6

FSD A181008, CS System
2.0 SYSTEM FUNCTIONAL REQUIREMENTS

The safety-related function of the CSS is to reduce the containment building pressure and temperature following a LOCA or high-energy line rupture and to reduce airborne fission products in the containment atmosphere following a LOCA.

During the injection phase, the CSS pumps are aligned to take suction off the RWST. When the RWST reaches low-low level, the spray pumps operate in the recirculation mode from the containment sump. Operator action to perform realignment of the CSS pumps to sump recirculation must be completed within 130 seconds of reaching the RWST low-low level setpoint. Completion of this operator action in 130 seconds ensures sufficient volume remains in the RWST to ensure adequate pump NPSH is available and to prevent vortexing in the RWST (References 6.3.020, 6.7.039). Trisodium phosphate (TSP) filled baskets in the recirculation area of containment provide iodine absorption and retention in the containment sump solution (References 6.2.001, 6.3.001, 6.7.001).

As the RCS inventory combined with ECCS solution accumulates in the recirculation sump, the rising water level dissolves the TSP crystals in the baskets (References 6.7.033 and 6.7.034).

The spray water is maintained at a pH level of approximately 4.5 during injection. During recirculation, a pH of approximately 7.5 enhances the absorption of the airborne fission product iodine, retains the iodine in the containment sump solution, and minimizes potential for chloride induced stress corrosion cracking (References 6.1.001, 6.2.001, 6.3.001, 6.3.017).

The development of the iodine removal coefficient is a function of the characteristics of the CSS. The design value of the iodine removal coefficient is 10 hr^{-1}. This coefficient is based on one CSS pump operating at a flow rate of 2,200 gpm, and a spray fall height of 110 ft (References 6.3.018, 6.7.003).

3.1 CSS PUMPS

3.1.1 Basic Functions

Post-LOCA, the CSS pumps shall deliver borated water from the RWST during the injection mode, water from the containment sump and trisodium phosphate from the TSP baskets during the recirculation mode, to the containment spray ring headers (References 6.2.001, 6.7.033, 6.7.034).

Previous NRC exam history if any: Wrote a new question and intentionally stayed away from the 2008 nrc exam question on k/a 027G2.1.27, CS&COOL-40302D02 17 to prevent going over the limit of 4 RO questions from the previous 2 NRC exams.

027K1.01
027 Containment Iodine Removal System
K1 Knowledge of the physical connections and/or cause-effect relationships between the CIRS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
K1.01 CSS 3.4* 3.7*

Match justification: To answer this question correctly, the physical connections to the Iodine Removal and the CSS (only connected during the CS recirc phase taking a suction from the Sump instead of the RWST), and the knowledge of the TSP (iodine removal) cause-effect on the CSS of adjusting the pH FROM 4.5 TO 7.5 or greater is required.

Objective:

1 **LABEL AND ILLUSTRATE** the Emergency Core Cooling System to include the components found on the following figures (OPS-40302C05):

- Figure 2, Accumulators
- Figure 3, Refueling Water Storage Tank and Figure 4, Emergency Core Cooling System
- The flow paths found on Figure 14, ECCS Injection Phase, Figure 15, ECCS Cold Leg Recirculation, Figure 16, ECCS Simultaneous Hot & Cold Leg Recirculation Normal, and Figure 17, ECCS Simultaneous Hot & Cold Leg Recirculation Alternate.

2 **LABEL AND ILLUSTRATE** the Containment Spray and Cooling System flow paths, to include the components found on Figure 2, Containment Cooling System, Figure 3, Containment Spray System and Figure 4, Service Water to Containment Coolers (OPS-40302D05).

2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the components associated with the Containment Spray and Cooling System to include the components found on Figure 2, Containment Cooling System, Figure 3, Containment Spray System and Figure 4, Service Water to Containment Coolers and the following (OPS-40302D02):

- Containment Cooler Service Water Inlet Isolation Valves (MOV-3019A, B, C, and D)
- Trisodium Phosphate Baskets

28. 029EA1.06 001/MOD/RO/C/A 3.2/3.1/N/N/2/EDITORIAL/SAT

Unit 2 has experienced an Anticipated Transient Without Trip (ATWT) and the following plant conditions occurred:

- Charging Flow is 68 gpm.
- 2A Boric Acid Transfer Pump is tagged out.
- Safety Injection has **NOT** actuated at this time.
- IAW FRP-S.1, Response to Nuclear Power Generation – ATWT, the UO is establishing Emergency Boration.
- 2B Boric Acid Transfer Pump tripped when it was started.

Which one of the following states:

1) the required actions to establish an emergency boration flow path,

and

2) the required action(s) for FK-122, CHG FLOW controller, IAW FRP-S.1?

A. 1) Open V185, MAN EMERG BORATION valve, AND open FCV-113A, BORIC ACID TO BLENDER valve.

2) Place FK-122 in MAN and maintain current demand.

B.✓ 1) Open LCV-115B and D, RWST TO CHG PUMP valves, AND close LCV-115C and E, VCT OUTLET ISO valves.

2) Place FK-122 in MAN **AND** raise demand.

C. 1) Open V185, MAN EMERG BORATION valve, AND open FCV-113A, BORIC ACID TO BLENDER valve.

2) Place FK-122 in MAN **AND** raise demand.

D. 1) Open LCV-115B and D, RWST TO CHG PUMP valves, AND close LCV-115C and E, VCT OUTLET ISO valves.

2) Place FK-122 in MAN and maintain current demand.

defined "ONLY" and removed "minimum" from stem per CE suggestion

A - Incorrect. The first part is incorrect, since the Manual Emergency Borate flowpath will not work in this situation. Neither BAT pump is available, and at least one is required to use either the normal emergency or manual emergency borate flowpath. Plausible, since in other situations with a loss of the normal emergency borate flowpath, the MANUAL emergency borate flowpath would be used per FRP-S.1 Step 4.3 RNO. The second part is incorrect due to the charging flow being less than required (92 gpm) with the RWST boration flowpath aligned, and the charging will have to be increased by adjusting FK-122 in the raise direction per FRP-S.1, Step 4.6 & 4.7.3. Plausible, since the charging flow is greater than required for the normal emergency or manual emergency borate flowpath (40 gpm) per FRP-S.1, step 4.6.

B - Correct. Per FRP-S.1 Steps 4.2.1 RNO, the RWST boration flow path will be aligned due to the inability to start either BAT pump. The flow from the RWST to the RCS is required to be > 92 gpm per step 4.6, thus the charging demand must be raised in manual.

C - Incorrect. The first part is incorrect (see A). The second part is correct (see B).

D - Incorrect. The first part is correct (see B). The second part is incorrect (see A).

FRP-S.1 Revision 25

Previous NRC exam history if any:

029EA1.06

029 Anticipated Transient Without Scram (ATWS)

EA1 Ability to operate and monitor the following as they apply to a ATWS: (CFR 41.7 / 45.5 / 45.6)

EA1.06 Operating switches for normal charging header isolation valves 3.2* 3.1

Match justification: To answer this question the applicant must know what to do with the operating switch for normal charging header isolation valve (Operate & monitor: FCV-122 Controlled by FK-122 controller), in response to the flow indication on FI-122 in the given situation during an ATWS.

Objective:

6. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing (1) FRP-S.1, Response to Nuclear Power Generation/ATWT; (2) FRP-S.2, Response to Loss of Core Shutdown. (OPS-52533A06)

29. 032AK3.02 001/NEW/RO/C/A 3.7/4.1/N/N/3/EDITORIAL/SAT

Unit 1 is at 100%, and the following conditions occurred:

- Intermediate Range Channel N-35 lost compensating voltage.
- I&C is called to investigate.
- Prior to any action by I&C, a reactor trip occurs.

Which one of the following describes the Source Range NI detectors response after the trip, and the required actions IAW ESP-0.1, Reactor Trip Response?

Source Range Instruments will __(1)__; and they must be manually __(2)__.

A. (1) automatically energize prematurely

(2) de-energized until approximately 5 minutes post-trip.

B. (1) automatically energize prematurely

(2) de-energized until approximately 15 minutes post-trip.

C. (1) **NOT** automatically energize when required

(2) energized approximately 5 minutes post-trip.

D.✓ (1) **NOT** automatically energize when required

(2) energized approximately 15 minutes post-trip.

Deleted "to prevent damage to the detectors" from second bullet of A & B per CE suggestion. Ran on Simulator Laptop to validatate time.

A - Incorrect. Plausible, since examinee may believe loss of compensating voltage will make IR power read lower than actual and energize the Source range NIs above the power level that they are normally operated at. UOP-1.2 (step 5.18) & UOP-1.3 direct the Source Range NIs deenergized above P-6, IR>10E-10 amps, and applicant may believe there is similar guidance in ESP-0.1 for a premature energizing of the Source Range NIs. The second part is incorrect, since the decay into the Source range is at ~1/3 dpm for about 6 decades, and thus takes about 15 minutes. Plausible, since confusion may exist between the decay into the source range from the power range and the limit on power ascension rate of 1 DPM from procedures to travel the required five decades.

B - Incorrect. The first part is incorrect (see A). The second part is correct (see D).

C - Incorrect. The first part is correct (see D). The second part is incorrect (see A).

D - Correct. Loss of compensation is under compensated, which means that IR power will read higher than actual. SR automatic energization requires 2 of 2 IR detectors < P-6. Power drops immediately after a trip approximately one decade from 100% to approximately 7% (even though NIs indicate 0%). Then, it decays at approximately 1/3 dpm for 5 decades to 10E-10 amps in the IR. Assuming -1/3 DPM SUR for about 5 decades, ~15 minutes post trip is when the Source Range is required, and automatically energized if both IR NI detectors are working properly. Per ESP-0.1, Step 12; since P-6 interlock should have reinstated the SR NI high voltage power, "verify source range detectors energized" is directed. "Verify" means take action to accomplish it if it didn't already happen. At this time, the other Intermediate range that is reading correctly will indicate that the power level is below P-6, and the source range NIs must be energized manually.

Ran on simulator laptop from 100% power Middle of Life (IC73). Tripped from 100% power, and 12 minutes 48 seconds later the source ranges energized.

ESP-0.1, Reactor Trip Response, Revision 29

12 Monitor nuclear instrumentation.

12.1 [CA] WHEN intermediate range indication less than 10^{-10} amps OR BYP & PERMISSIVE P-6 light off, THEN verify source range	12.1 IF no source range detector energized, THEN within one hour verify adequate shutdown margin using FNP-1-STP-29.1, detectors – ENERGIZED SHUTDOWN MARGIN CALCULATION (TAVG 547??F), or FNP-1-STP-29.2, SHUTDOWN MARGIN CALCULATION (TAVG <547F OR BEFORE THE INITIAL CRITICALITY FOLLOWING REFUELING).

Previous NRC exam history if any:

032AK3.02

032 Loss of Source Range Nuclear Instrumentation

AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Source Range Nuclear Instrumentation: (CFR 41.5,41.10 / 45.6 / 45.13)

AK3.02 Guidance contained in EOP for loss of source-range nuclear instrumentation . 3.7* 4.1

Match justification: This has a IR channel malfunction that causes the SR instruments to be de-energized at a time they should be energized and the procedural guidance and time when the SR instruments will be energized by the operator, for the reason that they should have energized and need to be energized.

Objective:

6. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing ESP-0.1, Reactor Trip Response. (OPS-52531B06)

30. 035K3.02 005/NEW/RO/C/A 4.0/4.3/N/N/4/NO CHANGE OK/SAT

Unit 1 has experienced a Loss of Offsite Power and a Tube Rupture on the 1A SG, and the following conditions exist:

- RCS cooldown at the maximum obtainable rate is in progress IAW EEP-3, Steam Generator Tube Rupture.
- INTEGRITY Critical Safety Function Status Tree has turned ORANGE due to the 1A RCS LOOP cold leg temperature dropping rapidly.

Which one of the following describes the reason the 1A RCS LOOP cold leg temperature has dropped rapidly?

1A RCS Loop flow has ______________________

A. increased, moving the cold 1A SG U-tube water past the T_{COLD} instruments.

B. restarted, causing a sudden rise then rapid drop in temperature as the stagnant water from the hot leg is flushed through the loop.

C. reversed, causing the cold water from 1B and 1C loops to pass over the T_{COLD} instruments in 1A loop.

D.✓ stopped, allowing the cold Safety Injection water to pass over the T_{COLD} instruments.

A - Incorrect. During E-3 max rate cooldown under natural circ conditions, the ruptured SG loop flow will stagnate and may reverse, but NOT increase since the ruptured SG is not steamed, the differential temperature causing the Thermal Driving Head will be lost in that loop.

Plausible: performing a cooldown increases the TDH for the intact loops and the cooldown for the intact loops would result in colder SG U-tube water to pass the T_{COLD} instruments.

B - Incorrect. SEE A. Loop flow is expected to stall not restart.

Plausible: Initiating the cooldown, would restart or improve the intact SG loop flows. Also, after the termination of the cooldown/depressurization there is expected to be a minor restoration of flow in A RCS loop during the recovery actions and subsequent stabilization procedures.

C - Incorrect. 1A RCS LOOP flow will stop, however T_{cold} of the active loops will not be sufficiently low to cause integrity to be challenged in the inactive loop, otherwise the Integrity status tree would be VALID and thier temperatures would ALSO result in an ORANGE INTEGRITY condition.

Plausible: a flow reversal is discussed in the occurrance of this condition and the 1B & 1C loops temperatures are lower than the 1A RCS loop.

D - Correct. The basis for step 6.4 CAUTION-1 warns the operator to not enter FRP-P.1 if caused from the LOOP with the Ruptured SG. This is because SI flow reversal will likely occur in the ruptured Loop and "result in the indicated cold leg temperature (due to the location of the cold leg RTD)" to decrease. The flow stagnation in the 1A RCS loop, combined with the SI Flow into the loop, and a leak in the 1A SG would cause an accumulation of the Cold SI (RWST) water to accumulate between the SI thermal sleeve and the SG, causing a >100°F/hr cooldown (which is in all loops due to operator action) AND <250°F (285°F unit 2)-- an ORANGE path condition on FRP-P.

EEB-3.0, ver 1, pg 31:
Basis for step 6.4 CAUTION-1

"If the RCS is being cooled down on natural circulation during a steam generator tube rupture event, **reverse flow through the ruptured loop during the cooldown or when the pressurizer PORV is opened to depressurize the RCS is possible and could cause the SI flow path in the ruptured loop to change. This change in the SI flow path could result in an indicated cold leg temperature (due to the location fo the cold leg RTD) that decreases to the point that the symptoms for FR-P.1 would occur.** This false indication would only be seen in the ruptured loop since it is essentially stagnant while th either loops are circulated by natural circulation. When the PORV is closed, the flow paths are expected to change and the indicated cold leg temperature should increase resulting in the symptoms disappearing. When SI is terminated, the indicated cold leg temperature would increase if it did not do so earlier resulting in the symptoms for FR-P.1 no longer being present. This is an expected condition and the operator should only monitor the F-0.4, Integrity Status Tree for information purposes."

Previous NRC exam history if any: None

035K3.02

035 Steam Generator System

K3 Knowledge of the effect that a loss or malfunction of the S/GS will have on the following:

(CFR: 41.7 / 45.6)

K3.02 ECCS 4.0 4.3

Match justification: the effect of ECCS flow into the loop due to implementing E-3 is a stagnation/flow reversal in A loop, which is indicated by a rapid drop in indicated loop cold leg temperature. This rapid drop in temperature results in FRP-P.1 being ORANGE, and understanding the mitigation strategy IMPACT **on ECCS flowpath ensures that the cooldown would not be erroneously terminating.** Terminating the cooldown due to the indications presented here, would complicate stabilization of the plant.

Objective: OPS-52530D03; State and Explain the basis for all Cautions, Notes and Actions associated with EEP-3 [...].

31. 037AA2.08 001/NEW/RO/C/A 2.8/3.3/N/N/2/REVISED/FIX

Unit 1 is at 12% power, and the following conditions exist:

- R-15A, SJAE EXH, has a red Low Alarm light LIT, and is reading 10 cpm.
- 1A SG has developed a 10 gpm tube leak.

Which one of the following radiation monitors will provide the **EARLIEST** indication of the 1A SG Tube leak?

A. R-15A, SJAE EXH, alarm.

B.✓ R-19, SGBD SAMPLE, alarm.

C. R-23B, SGBD TO DILUTION, alarm.

D. R-70A, 1A SG TUBE LEAK DET, alarm.

Replaced "R-15A is failed" with indications that R-15A is failed. Used R-15 in place of the least plausible distractor (R-60A) in the choices per CE suggestion.

A – Incorrect. R-15A is inoperable based on given idications. 10 cpm is the low end of the meter scale, and the low alarm indicates that the instrument is indicating below normal background value. Plausible, since R-15A is normally the first indication of a SGTL, and if these indications are not recognized as an indication of inoperability, this answer could be chosen.

B - Correct. R-19 is continuously monitoring the SGBD system sample stream and will be the first indication of an alarm considering only the 4 choices given.

C - Incorrect. R-23B will only alarm after the SGBD surge tank starts filling with the contaminated SGBD water. The tank is maintained 50% full, and there will be a diluting effect at first. Downstream of this tank is R-23B in a flow stream going to the environment. R-19 samples undiluted SGBD water continuously, and would alarm sooner than R-23B. Plausible, since R-23A samples blowdown water at the inlet of the Surge Tank and is undiluted SGBD water. Due to the higher flowrate of SGBD (about 130 gpm) than the sample stream, it alarms sooner than the R-19 alarm for a particular SGTL event. Confusion may exist as to the difference between the choice for R-23B and R-23A which would alarm prior to R-19 (as seen on the simulator during SGTL events).

D - Incorrect. R-70A alarm setpoints are not valid at this power level. The R-70s shift automatically from the gpd Mode to the ME mode below 20% power, and the alarm functions are set for gpd. Plausible, since it is on the Steam line at the outlet of the SG, and alarms first before any other Radiation monitor in the event of a SGTL above 20% reactor power.

FNP-1-AOP-2.0, Steam Generator Tube Leakage, Version 33.0

B. Symptoms or Entry Conditions

I. Enter this procedure when RCS tube leakage is indicated by high secondary activity on any of the following radiation monitors or by sample results.

a. R-15 SJAE EXH **[listed here to show the nomenclature in the procedure]**

b. R-15B or R-15C TURB BLDG VNTL

c. R-19 SGBD SAMPLE **[listed here to show the nomenclature in the procedure]**d. R-23A SGBD HX OUTLET

e. R-23B SGBD TO DILUTION **[listed here to show the nomenclature in the procedure]**

f. R-70A, R-70B or R-70C 1A(1B,1C) SG TUBE LEAK DET

g. SG sample results indicate primary to secondary leakage for any SG greater than or equal to the normal alarm setpoint for annunciator FG1, SG TUBE LEAK ABOVE SETPT.

Previous NRC exam history if any:

037AA2.08

037 Steam Generator Tube Leak

AA2. Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: (CFR: 43.5 / 45.13)

AA2.08 Failure of Condensate air ejector exhaust monitor 2.8 3.3

Match justification: A scenario is given with a SGTL and a failed Condensate air ejector exhaust monitor (R-15A), which is normally the first indication of a SGTL. To answer this question correctly, determining whether or not the indications allow the R-15A to indicate properly or not, and how the failed R-15 applies to the SGTL is required. I. E., since R-15A is failed, and it is no longer the first indicaiton of a SGTL, which is the first indication?

Objective:

2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the components associated with the Radiation Monitoring System to include those items in Table 4- Remote and Local Indications and Controls (OPS-40305A02).

5. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Radiation Monitoring System components and equipment, to include the following (OPS-40305A07):

- Normal control methods
- Abnormal and Emergency Control Methods
- Automatic actuation
- Protective isolations
- Protective interlocks
- Actions needed to mitigate the consequence of the abnormality

32. 038EA2.07 001/NEW/RO/C/A 4.4/4.8/N/N/3/REVISED/FIX

Unit 1 has been manually Tripped and Safety Injected from 14% power IAW AOP-2.0, Steam Generator Tube Leakage. The following conditions exist:

At 1000:

- DA-07, 1A 4160V BUS SUPP FROM 1A S/U XFMR, breaker tripped open.
- AFW Flow has been secured to all SGs.

SG Pressures:	SG NR levels:
- 1A 990 psig	- 1A 61%
- 1B 995 psig	- 1B 61%
- 1C 980 psig	- 1C 60%

At 1010:

- EEP-0.0, Reactor Trip And Safety Injection, is in progress at the step to diagnose the event.

SG Pressures:	SG NR levels:
- 1A 980 psig	- 1A 61%
- 1B 985 psig	- 1B 73%
- 1C 970 psig	- 1C 50%

Which one of the following correctly describes the event in progress based on the MCB indications?

A.✓ 1B SG is ruptured, but 1A and 1C are intact.

B. 1A and 1B SG are ruptured, but 1C SG is intact.

C. 1B SG is ruptured and 1C SG is faulted, but 1A SG is intact.

D. 1A and 1B SG are ruptured and 1C SG is faulted.

Created a time line and adjusted pressures slightly different than each other. Removed the stated trends, since the applicant must determine the trend from the values and the time line given. Made SG pressures slightly lower at 1010 than at 1000 indicating a normal condition of slightly lower SG pressure after feeding the SG's with cooler AFW and a cooldown of the RCS to 547°F due to the decay heat reduction and steam dump operation after a trip. Since there is no steam leak, pressures could not be lowered more at 1010. Steam dumps will be maintaining Tavg at 547°F, and MSIVs are open to equalize pressures.

VALIDATION on 2/10/10 revealed a flaw in question: A steam leak is not PRECLUDED by any of the indications available, and a LEAK would (could) NOT be evaluated in a SGTR event until the MSIVs are close or if the cooldown could not be stopped changed the stem to evaluate the status of the generators with regard to intact/faulted/ruptured instead. FEEDBACK review identified changes that need to be made to match current question.

A - Correct. 1B SG has an uncontrollable rise in lvl, therefore ruptured.
1A SG level is stable but elevated due to the loss of 1A RCP.
1C SG level is low since it (with 1B) is steaming.
NO SG pressure is not falling in an uncontrollable manner therefore no fault. (MSIVs OPEN--> 3 SG pressures will fall together > 10 psig, IF MSIVs CLOSED (due to EEP-0 step 7 RNO, then any single SG would be significantly lower in pressure if faulted).

B- Incorrect. See A; Plausible: Correct if NO RCP trip had occurred. No AFW flow to all SG, but only 1C level lowering. One might believe that 1A and 1B SG were ruptured if they did not recognize that 1A RCP is not running.

C - Incorrect. See A: Plausible: Correct if 1C pressure were falling at an uncontrolled rate.

C - Incorrect. See A: plausible: the pressure is reduction occurring on all SG caused by short term cooling after the trip from AFW flow and decay heat reduction, but not enough to indicate a fault.

D - Incorrect. See A: Plausible, would be correct if A RCP were running and SG pressures were falling uncontrollably (or <50 psig).
since A level is elevated may presume ruptured= not intact.
Since B lvl IS rising uncontrollably, it is ruptured and NOT intact..

Previous NRC exam history if any:

038EA2.07

038 Steam Generator Tube Rupture

EA2 Ability to determine or interpret the following as they apply to a SGTR: (CFR 43.5 / 45.13)

EA2.07 Plant conditions, from survey of control room indications 4.4 4.8

Match justification: Control Room indications are given which could indicate a SGTR in two SGs and a Steam leak in one SG under slightly different conditions than given. With one tripped RCP one SG level is high due to not steaming and not due to a SGTR. One SG is high due to a SGTR. One SG level is low and dropping due to being the only intact SG producing steam, even though the other intact SG lvl is stable (due to the tripped RCP). The applicant must correctly evaluate all these indications and diagnose the event to be a SGTR in one SG only.

Objective:

3. **LIST AND DESCRIBE** the sequence of major actions, when and how continuous actions will be implemented, associated with (1) EEP-0, Reactor Trip or Safety Injection and (2) ESP-0.0, Rediagnosis. (OPS-52530A04)

33. 039K4.05 001/FNP BANK/RO/MEM 3.7/3.7/N/N/2/CVR/SAT

Which one of the following adequately describes the setpoint of the steam line flow for the High Main Steam Line Flow with Low-Low T_{avg} MSIV isolation?

A. Increases linearly from 40% to 110% steam flow as power increases from 0% to 100%.

B. Increases linearly from 20% to 110% steam flow as power increases from 0% to 100%.

C. Constant 20% steam flow up to 10% power; then increases linearly to 110% flow as power increases from 10% to 100%.

D✓ Constant 40% steam flow up to 20% power; then increases linearly to 110% flow as power increases from 20% to 100%.

A - Incorrect. Plausible, since the numbers are the same as the values for the correct setpoint, but the Constant value of 40% steam flow limit from 1%- 20% is left out of this choice.

B - Incorrect. Plausible, since the numbers are the same as the values for the correct setpoint, but the Constant value of 40% steam flow limit from 1%- 20% is left out of this choice.

C - Incorrect. Plausible, since this choice correctly states that there is a constant value of Steam Flow setpoint up to a certain power, but the constant value and associated power level are incorrect. The setpoint at 100% power is correct.

D - Correct.

OPS-52201K

A higher than expected steam flow from the steam generators, along with a decreasing T_{avg}, is another indication of a steam break that will shut the MSIVs.

The high steam flow set point is varied with turbine power by P_{imp}. The set point is 40 percent steam flow from 0 percent to 20 percent turbine power. It then increases linearly from 20 percent turbine power to 100 percent turbine power where the set point is 110 percent steam flow.

2/3 steam lines reaching the set point and T_{avg} below the P-12 set point will shut the MSIVs. It requires only one of the two, density-compensated steam flow detectors per steam line to reach the set point to actuate the MSIV closure with a one second time delay. This main steam line isolation is not able to be blocked or bypassed.

Previous NRC exam history if any:

039K4.05

039 Main and Reheat Steam System

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

(CFR: 41.7)

K4.05 Automatic isolation of steam line 3.7 3.7

Match justification: Other MSIAS are on exam so this type of question was selected to avoid double jeopardy.

Objective:

6. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Main and Reheat Steam System components and equipment, to include the following (OPS-40201A07):

   - Normal control methods
   - Abnormal and Emergency Control Methods
   - Automatic actuation including setpoint (example SI, Phase A, Phase B, MSLIAS, LOSP, SG level)
   - Protective isolations such as high flow, low pressure, low level including setpoint
   - Protective interlocks
   - Actions needed to mitigate the consequence of the abnormality

34. 041K4.14 001/NEW/RO/C/A 2.5/2.8/N/N/4/CVR/SAT

Unit 1 was at 26% power and 180 MWe, and the following conditions occurred:

- The reactor tripped.
- The "A" Reactor Trip Breaker failed to open.

Which one of the following correctly states the arming signal for the Steam Dumps, and the RCS temperature maintained by the Steam Dumps?

The Steam Dumps are armed due to the __(1)__;

and

RCS temperature will be controlled at __(2)__.

(1)	(2)
A. P-4 signal	547°F
B. P-4 signal	551°F
C✓ Loss of Load signal	547°F
D. Loss of Load signal	551°F

A - Incorrect. The first part is incorrect but plausible, since the reactor did trip and if A RT bkr would have opened this would be correct. Most functions of the P-4 Permissive come from both trains, but this function comes only from A train. The second part is correct, and is the result of the B train P-4 signal which is present as normal with the B train RT bkr open.

B - Incorrect. The first and second part are incorrect but plausible, since this choice would be correct for a B train RT bkr failing to open. A train P-4 would arm the Steam Dumps and the LOL controller would stay in the circuit (since the B train P-4 did not shift controllers to the Plant Trip mode) to control Tavg 4°F higher than no load Tavg (547+4=551).

C - Correct. The A train P-4 did not arm the steam dumps, and the Loss of Load did (the loss of load was 20% instantaneously, and thus greater than the LOL arming setpoint of 15% with a 120 second time constant). The B train P-4 shifted the controllers from the LOL to the Plant Trip controller which maintains a constant no load Tavg of 547°F.

D - Incorrect. The first part is correct (see C). The second part is incorrect, but plausible, since it would be correct for a loss of load or reactor trip with the B train RT bkr open instead of the A.

Previous NRC exam history if any: None

041K4.14

041 Steam Dump System and Turbine Bypass Control

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

(CFR: 41.7)

K4.14 Operation of loss-of-load bistable taps upon turbine load loss 2.5* 2.8

Match justification: The loss-of-bistable arms the steam dump when the loss of load magnitude is greater than the variable setpoint in a given time. The Reactor trip overrides the arming of the loss of load due the A RT bkr P-4 signal for a normal reactor trip. This question requires knowledge the loss of load setpoint and the times that it does and does not arm the Steam dumps. It also requires knowledge of what controller (Plant Trip or Loss of Load) is in the circuit under different conditions than normal. Several design features and interlocks relating to and affecting the Loss of Load interlock must be understood to correctly answer this question.

Objective:

5. **DEFINE AND EVALUATE** the operational implications of abnormal plant or equipment conditions associated with the operation of the Steam Dump System components and equipment to include the following (OPS-52201G07):

- Normal Control Methods (Steam dump valves)
- Abnormal and Emergency Control Methods (Steam dump valves, Steam dump system solenoid-operated three-way valves)
- Automatic actuation including setpoint (High-1 and High-2 trip bistables)

Protective isolations (Plant trip controller, Loss of load controller, C-7)

- Protective Interlocks (Condenser available, C-9, Low-Low T_{AVG} signal, P-12)

Actions needed to mitigate the consequence of the abnormality

35. 045A1.05 001/NEW/RO/C/A 3.8/4.1/N/N/2/REVISED/FIX

The following conditions existed on Unit 1:

- This is the first plant startup after a refueling outage.
- Moderator Temperature Coefficient (MTC) is at the BOL limit allowed by the Core Operating Limits Report (COLR).
- Unit 1 was at 30% power, BOL.
- Control Rods are in Manual.

The following event occurred:

- The Main Turbine was manually tripped.

Which one of the following is the **initial** response of RCS Tavg and Reactor Power, **with no operator actions**?

Tavg __(1)__ and Reactor Power __(2)__.

	(1)	(2)
A.✓	increases	increases
B.	increases	decreases
C.	decreases	increases
D.	decreases	decreases

Added verbiage to say the plant is at BOL so the candidate knows there is +MTC per CE suggestion to enhance plausibility of one of the distractors. Re formatted the stem for clarity with this new information.

A - Correct. Steam pressure goes up, causing Tcold to go up. This causes Tavg to go up. This causes a positive reactivity due to the Positive MTC present below 100% power during the first power ascension after a refueling outage per Step 2.3 of the COLR: MTC shall be less than or equal to +0.7 x 10-4 ?k/k/°F for power levels up to 70 percent RTP with a linear ramp to 0 ?k/k/°F at 100 percent RTP.

.

B - Incorrect. The first part is correct (see A). The second part is incorrect (see A). Plausible, since during most of the operating cycle between refueling outages MTC would be negative and this choice would be correct.

C - Incorrect. The first part is incorrect (see A). Plausible, since the Steam Dumps and/or SG Atmospheric relief valves will open to reduce SG pressure and Tavg, but prior to the Steam Dumps and/or SG Atmospherics opening, Tavg will go up. The second part is correct (see A).

D - Incorrect. The first part is incorrect (see C). The second part is incorrect (see B). This choice would be the correct response for temperature after the initial response, and the correct for power after the initial power ascension to 100% after the refueling outage.

TS U1 COLR Cycle 23:

2.3 Moderator Temperature Coefficient (Specification 3.1.3)

2.3.1 The Moderator Temperature Coefficient (MTC) limits are:

The BOL/ARO-MTC shall be less than or equal to +0.7 x 10_{-4} ?k/k/°F for power levels up to 70 percent RTP with a linear ramp to 0 ?k/k/°F at 100 percent RTP.

Previous NRC exam history if any:

045A1.05

045 Main Turbine Generator System

A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: (CFR: 41.5 / 45.5)

A1.05 Expected response of primary plant parameters (temperature and pressure) following T/G trip3.8 4.1

Match justification: This question requires knowledge of the expected response (predicting the changes in the parameters) of primary plant parameters of Temperature and Reactor Power following the Turbine Trip. Written with 2 parts to facilite including 3 wrong but plausible distractors.

Objective:

15. Describe the operation of the reactor's inherent control systems as they function to re-establish a steady-state condition for the following transients: (OPS52701A15)
    a. 10% step load increase
    b. 50% load rejection
    c. 10% ramp increase
    d. Entry into the power range
18. Determine the final condition of the plant for various transients assuming no operator response (OPS52701A19).

36. 051AA1.04 001/BANK/RO/MEM 2.5/2.5/N/N/2/EDITORIAL/SAT

Unit 1 was at 100% power and the following conditions exist:

- AOP-8.0, Partial Loss Of Condenser Vacuum, is in progress.
- A rapid power reduction per AOP 17.0, Rapid Load Rejection, was completed.
- Condenser Vacuum is stable.
- FE1, CONT ROD BANK POSITION LO, is in alarm.
- A normal boration is in progress per SOP-2.3, Chemical And Volume Control System Reactor Makeup Control System.

Which one of the following states:

1) whether or not SS permission is required prior to the Control Rod **insertion** during the downpower IAW NMP-OS-001, Reactivity Management Program,

and

2) is an Emergency Boration required IAW the ARP for FE1?

	Control Rod Insertion	**Boration**
A.	SS permission is **NOT** required.	Emergency boration is required.
B.✓	SS permission is **NOT** required.	Emergency boration is **NOT** required.
C.	SS permission is required.	Emergency boration is required.
D.	SS permission is required.	Emergency boration is **NOT** required.

Added a normal boration is in progress to the stem and changed the second part of the question to: is an Emergency Boration required...

A - Incorrect. Permission not required on insertion. Emergency Boration required until the Rod Bank LO-LO Limit alarm clears. With Rod Bank LO Limit alarm present, a normal boration is required until the LO Limit alarm clears so this is plausible for the student to confuse the the two requirements for normal or emergency boration.

B - Correct. Per NMP-OS-001 and ARP-1.6 FE1 & FE2. (See below)

C - Incorrect. Permission not required on insertion. Plausible, since it is always required to get SS permission for all positive reactivity additions, and it is expected to get permission when there is time to do so even for negative reactivity additions. However, for responding to a transient to stabilize the plant no permission is required to insert negative reactivity of any type. Emergency Boration is required until the Rod Bank LO-LO Limit alarm clears. With Rod Bank LO Limit alarm present, a normal boration is required until the LO Limit alarm clears so this is plausible for the student to confuse this.

D - Incorrect. Permission not required on insertion. Emergency Boration required until the Rod Bank LO-LO Limit alarm clears. With Rod Bank LO Limit alarm present, a normal boration is required until the LO Limit alarm clears so this is plausible for the student to confuse this.

NMP-OS-001, Reactivity Management Program, Version 13.0
6.3.8.1 During transient conditions that require a rapid reduction in reactor power, operators may take actions to insert negative reactivity that are outside the amounts discussed in the reactivity brief and without SS concurrence.

ARP-1.6, FE1 Annunciator: CONT ROD BANK POSITION LO, **Version 64.0**

5. Borate [NORMAL BORATION] the Control Bank "OUT" as necessary using the Boron Addition Nomographs. {CMT 0008900}

ARP-1.6, FE2 Annunciator: CONT ROD BANK POSITION LO-LO, **Version 64.0**

2. Emergency borate the reactor coolant system in accordance with FNP-1-AOP-27.0, EMERGENCY BORATION. **{CMTs 0008555, 0008900}**

Previous NRC exam history if any:

051AA1.04

051 Loss of Condenser Vacuum

AA1. Ability to operate and / or monitor the following as they apply to the Loss of Condenser Vacuum: (CFR 41.7 / 45.5 / 45.6)

AA1.04 Rod position 2.5* 2.5*

Match justification: The question presents a plausible scenario where a rapid power reduction is in progress in accordance with AOP-17. The examinee has to determine the correct procedural guidance given for control rod operation during insertions and withdrawals. The question pertains to whether SS permission is required for insertions and the predicted rod position when Emergency Boration may be terminated.

Objective:

6. State the actions that the UO and/or OATC have the authority to perform in addition to being responsible to the Shift Supervisor (OPS52303H10).

37. 054AG2.1.7 001/NEW/RO/MEM 4.4/4.7/N/N/3/CVR/SAT

A Unit 1 SGFP trip has occurred from 100% power, and the following conditions exist:

- AOP-13.0, Condensate And Feedwater Malfunction, is in progress.
- The operator is at the step to "Verify automatic operation of the Feedwater Regulating Valves adequate".
- SG NR levels are as follows:
  - 1A 34% Rising
  - 1B 33% Rising
  - 1C 36% Rising

Which one of the following is the correct method of controlling the Main Feed Regulating valves (MFRVs) during this transient IAW AOP-13.0?

Place each MFRV controller in manual at __(1)__ SG NR Level, __(2)__ .

	(1)	(2)
A.✓	55%	match steam flow and feed flow, and then place the controller back in automatic.
B.	55%	and then immediately place the controller back in automatic.
C.	65%	match steam flow and feed flow, and then place the controller back in automatic.
D.	65%	and then immediately place the controller back in automatic.

A - Correct. Per step 1.8 and the "D. Operational Concern" note of AOP-13.0. The swell and the response time of the MFRV controller and valve necessitates taking the controller to manual at 55% (before 65% - program level) and matching feed and steam flows prior to taking it back to automatic. This prevents a high high level Turbine trip at 82% level which would occur due to the large Feed Flow Steam flow mismatch if feed flow is not reduced prior to 65%.

B - Incorrect. The first part is correct (see A), but the second part is incorrect (see D).

C - Incorrect. First part is incorrect, since manual control must be taken at 55% level instead of 65%. Plausible, since doing this at 65% would seem adequate, since that is the level desired to maintain. This could be chosen if the magnitude of the effects of swell and the response time of the controllers is not taken into account. By waiting until 65% to place the controller in manual, the feed flow is high enough that the time to reduce it combined with the expansion of the cooler feed water after getting to the SG can cause excessively high SG levels and a high high SG level trip of the Turbine and SGFPs and a FWIS. The second part is correct (see A).

D - Incorrect. The first part is incorrect (see C). The second part is incorrect, since placing in AUTO after taking to manual would not correct the high feed flow and lower the feed flow/steam flow mismatch quickly enough to prevent excessively high SG levels. Plausible, since taking the controller to manual will reset the windup and decrease the controller response time to a level transient, and this is an important part of the procedure guidance reason to go to manual. Also, AUTO control is preferred to manual when adequate for the magnitude of the transient. In smaller SG level transients, going to manual to reset the windup and then allowing AUTO to control the SG level is preferred.

FNP-1-AOP-13.0, Condensate And Feedwater Malfunction, Version 29.0
D. Operational Concerns
1 In the SG level recovery phase, the SG level will start increasing due to the feedwater flow being higher than steam flow and due to swell. If manual action is not taken before the SG reaches normal operating level, the combined effect of swell and additional feed flow may result in SG Hi-Hi Level Turbine and SGFP trip and Feedwater Isolation. Taking manual control and reducing the demand resets the level controller and flow controller integration circuits (i.e. windup) and makes the flow controller output track the associated driver card output.

1.8 Closely monitor steam generator narrow range levels.

- [ ] WHEN a SG narrow range level recovers to approximately 55%, THEN verify its main feedwater regulating valve controllers in MANUAL.
- [ ] Match feed flow with steam flow.
- [ ] Return feedwater regulating valves to AUTO.

Previous NRC exam history if any:

054AG2.1.7
054 Loss of Main Feedwater
2.1.7 **Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.**
(CFR: 41.5 / 43.5 / 45.12 / 45.13) RO 4.4 SRO 4.7

Match justification: to answer this question correctly, evaluation of plant performance and operational judgement of how to operate the Main Feed Regulating Valves in the given transient condition is required. Instrument interpretation is also included in that with the SG level as low in the Narrow range as they are, MFRV controllers will windup to maximum output by the time the SG level is at normal level of 65%, and this operating characteristic must be taken into account to get the correct answer.

Objective:

4. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing AOP-100, Instrument Malfunction. (OPS-52521Q06).

38. 055EK3.02 001/MOD/RO/MEM 4.3/4.6/N/N/2/REVISED/FIX

ECP-0.0, Loss of All AC Power, directs the operator to:

- Dump steam from intact SGs at maximum controllable rate.

Which one of the following states the **primary reason** for dumping steam from the intact SGs at the maximum controllable rate, and states the **primary reason** for closing the Atmospheric Relief Valves (ARVs) when SG pressure is at 200 psig?

Steam is dumped from the SGs at the maximum controllable rate to __(1)__,

and

the ARVs are closed at 200 psig SG pressure to prevent __(2)__.

	(1)	(2)
A.	minimize RCS inventory loss	losing PRZR level.
B.✓	minimize RCS inventory loss	N2 injection.
C.	maximize TDAFW pump flow	losing PRZR level.
D.	maximize TDAFW pump flow	N2 injection.

Added second part per CE suggestion.

A - Incorrect. This is plausible, since cooling down and reducing the volume of the RCS would reduce RCS pressure, and reducing SG tube d/p is desirable, but this is not the reason for the depressurization.

second part is not correct per the note in ECP-0 prior to the step to depressurize the SGs. NOTE: Reduction of intact SGs pressure should continue even if pressurizer level is lost or reactor vessel head voiding occurs.

B - Correct. This is the Background Document basis for this step: 16.4 in ECB-0.0. (See below).

ERG StepText: The SGs should be depressurized at maximum rate to minimize RCS inventory loss.

Purpose: To inform the operator of the desired rate for depressurization of steam generators

Basis: The intact **steam generators should be depressurized as quickly as possible, to minimize RCS inventory loss...**

second part is correct per FNP-0-ECB-0.0 - Once the target SG pressure is reached, the SG PORVs and AFW flow should be controlled to maintain SG pressure at the target value until ac power is restored. The target SG pressure for Step 16 should ensure that RCS pressure is above the minimum pressure to preclude injection of accumulator nitrogen into the RCS.

C - Incorrect. Plausible, since the ECP-0.0 note prior to step 4 does remind of the 2 hour limit on air accumulator supply and UPS power supply, and the heat sink provided by the TDAFW pump is the main source of core cooling with no AC power. Long term, the need to remove decay heat would extend beyond the 2 hours, and a

hour period. However, the background document requires only SG level of >31% narrow range on one SG for an adequate heat sink.
second part not correct - see A above

D - Incorrect. Plausible, since cooling the RCS would eventually cool the vessel head, and without CRDM fans running would be the main cooling for the head. However, the short term as stated in a note in the procedure is that head voiding may be caused by the depressurization.
second part correct - see B above

ECP-0.0, Loss Of All Ac Power, Revision 22
****
CAUTION: **The TDAFWP will become unreliable within 2 hours following a loss of all AC power, unless power is restored**. This will occur due to a loss of air to the steam supply valves and a loss of control power from the UPS.
****

4 Verify total AFW flow GREATER THAN 395 gpm. | 4 Verify proper AFW alignment.

16.4 Dump steam from intact SGs at maximum controllable rate.

FNP-0-ECB-0.0

***Section:* Procedure**

Unit 1 ERP Step: 16.4 **Unit 2 ERP Step:** 16.4 ***ERG Step No:*** *16 NOTE-1*
ERP StepText: Dump steam from intact SGs at maximum controllable rate.
ERG StepText: The SGs should be depressurized at maximum rate to minimize RCS inventory loss.
Purpose: To inform the operator of the desired rate for depressurization of steam generators
Basis: The intact **steam generators should be depressurized as quickly as possible, to minimize RCS inventory loss**, but within the constraint of controllability. Controllability is required to ensure that steam generator pressures do not undershoot the specified limit. For the reference plant, the operator can control the secondary depressurization from the control room. In this case, maximum rate means steam generator PORVs full open. For plants that must control the secondary depressurization by local actions, maximum rate must be determined by the control room and local operators based on plant conditions and available communications. A slower rate is acceptable for locally controlled secondary depressurization. See Subsection

FNP-0-ECB-0.0

***Section:* Procedure**

Unit 1 ERP Step: 3 **Unit 2 ERP Step:** 3 ***ERG Step No:*** *3*

ERP StepText: Verify RCS isolated.
ERG StepText: *Check If RCS Is Isolated*
Purpose: To ensure all RCS outflow paths are isolated
Basis: A check for RCS isolation is performed to ensure that RCS inventory loss is minimized. The valves itemized are those in major RCS outflow lines that could contribute to rapid depletion of RCS inventory.... **Following completion of this step, the only RCS inventory leakage path should be the RCP controlled leakage seals....** The secondary depressurization in Step 16 will minimize RCS inventory loss by reducing RCS pressure which will terminate or minimize relief valve flow. For example, reducing RCS pressure to 400 psig would permit the letdown line relief valve to close and would minimize flow through the excess letdown relief valve.

Knowledge: Need to minimize RCS inventory depletion during loss of all ac power event to maximize time to core uncovery.

Previous NRC exam history if any:

055EK3.02
055 Station Blackout
EK3 Knowledge of the reasons for the following responses as the apply to the Station Blackout:
(CFR 41.5 / 41.10 / 45.6 / 45.13)
EK3.02 Actions contained in EOP for loss of offsite and onsite power 4.3 4.6

Match justification: This question requires knowledge of a response in the EOP (ECP-0.0) which is required to minimize inventory loss during a Station Blackout.

Objective:

3. **STATE AND EXPLAIN** the basis for all Cautions, Notes, and Actions associated with (1) ECP-0.0, Loss of All AC Power; (2) ECP-0.1, Loss of All AC Power Recovery, Without SI Required; (3) ECP-0.2, Loss of All AC Power Recovery, With SI Required. (OPS-52532A03)

39. 057AG2.4.49 001/NEW/RO/C/A 4.6/4.4/N/N/3/CVR/SAT

Unit 1 is ramping down for an outage at 2 MW/min. The following conditions occurred:

At 1000:

- Reactor power is 25%.
- The Reactor Makeup Control System is aligned for repetitive batch borations.
- A boration is **NOT** currently in progress.
- LK-112, LTDN TO VCT FLOW, has been adjusted to maintain 45% level in the VCT.
- LT-112B and LT-115, VCT LVL, meters both indicate 45%.

At 1001 the following occurs:

- 1A 120V Vital AC Instrumentation Panel is de-energized due to an electrical fault.

Which one of the following is the correct operator response to these conditions?

A. Secure BOTH Reactor Makeup Water Pumps.

B. Realign the Reactor Makeup system to AUTO.

C✓ Realign LCV-115A, VCT HI LVL DIVERT VLV, to the VCT.

D. Increase the ramp rate to control Tavg within the limits of AOP-17, Rapid Load Reduction.

A - Incorrect. The RMW pumps and Boric Acid Transfer pumps don't start unless the RMW control is in Auto when the 1A 120V bus fails. In the conditions given, the system is aligned per repetitive batch borations, which is common during a ramp, and the failure of the 1A 120V bus will not give an auto makeup. Plausible, since if the RMU system was in Auto, OR if an applicant believed that AUTO makeup occurred immediately in the borate mode when the 1A 120V vital panel failed, this choice would be selected.

B - Incorrect. Aligning to AUTO is incorrect since it would cause an AUTO makeup to occur regardless of VCT level, and it would not automatically stop. Also,in the repetitive batch Boration alignment, no BAT pump starts and no valves open to cause a boration. Plausible, since confusion between the effects of the 120V vital panel failure in each switch position may exist. In AUTO, the failure causes an Auto makeup to commence. However, in BORATE, an automatic Boration does NOT commence. Failures of other 3 vital instrumentation panels (1B, 1C, 1D) do not cause auto makeups with the control switch in AUTO, so the effects of each of the four 120V Vital panels may be misunderstood. This choice would be selected if an applicant thought that a Boration from the BAT occurred immediately with the switch in BORATE, but that an auto makeup did not occur with the switch in the AUTO position.

C - Correct. The LCV115A does divert letdown to the RHT, and will not automatically divert back to the VCT regardless of VCT level per **ARP for WD1:**

- **VCT Hi Lvl Divert Valve - Q1E21LCV115A diverts to the RHT if in auto.**

In addition, an auto makeup will not occur with the switch in BORATE. Prompt action to realign LETDOWN to the VCT must occur to stop the letdown diversion prior to realignment of the RWST to Chg pump suction valves which would cause a significant boration and reactivity event and reactor power transient at the end of life if it occurred.

D - Incorrect. This is incorrect since Boration from the RWST does not occur immediately requiring this action unless LT-112 is out of service when the 1A 120V vital panel fails. Plausible, since it could occur if LT-112 had also failed or was out of service per the FNP-0-ARP-2.2 WD1:

"• If LT 112 VCT level is out of service, RWST to Chg Pump Suction Valves –Q2E21LCV115B & D open."

Ran on SIMULATOR from 100% (IC73):

IN repetitive boration:
No BAT pump starts, the B RMW pump does not start (A RMW pump is normally running all the time), and no valves open (113A, 113B, 114A, & 114B)

In AUTO:
O/S BAT PUMP STARTS, B RMW PUMP STARTS, 114B RMW TO THE BLENDER modulates open, 113A BORIC ACID TO BLENDER modulates open, 113B MKUP TO CHG PUMP SUCTION HDR opens.

Previous NRC exam history if any:

057AG2.4.49
057 Loss of Vital AC Electrical Instrument Bus
2.4.49 **Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.** (CFR: 41.10 / 43.2 / 45.6)
RO 4.6 SRO 4.4

Match justification: **The Chief examiner was consulted** about the difficulty in meeting this k/a with a discriminatory question on the very few immediate actions that are committed to memory. Most "immediate actions" that are performed "without reference to procedures" have been eliminated in recent years. He recommended using actions that must be performed promptly to avoid adverse effects to the plant, whether a procedure directs them or not. This question requires knowledge of the immediate effects of this loss of vital AC Electrical Instrument Bus that would require actions very quickly to mitigate or prevent undesired plant effects. Also, to answer this question correctly, it requires knowledge from memory without a reference being provided such as immediate operator actions. This question fits these criteria, and thus matches the original intent of the K/A. A set of conditions in which a loss of a 120V Vital AC electrical Instrument Bus is given for which prompt actions must be taken to prevent a large boration which would cause a large undesired transient at EOL core conditions. A significant transient, and likely a manual trip requirement, would occur if actions are not taken.

Objective:

2. **STATE AND EXPLAIN** any special considerations such as safety hazards and plant condition changes that apply to the 120 Volt AC Distribution System (OPS-52103D04).

40. 059A2.04 001/MOD/RO/C/A 2.9/3.4/N/N/3/REVISED/FIX

Unit 1 was at 100%, and the following conditions occurred:

- The reactor was tripped on simultaneous loss of BOTH SGFPs.
- All AFW was subsequently lost.
- RCS Bleed and Feed is in progress in accordance with FRP-H.1, Response To Loss Of Secondary Heat Sink.
- Core Exit Thermocouples have reached 575°F and are falling.
- 1A SGFP has been started.
- SG Wide Range Levels are:
  - 1A= 8%
  - 1B= 8%
  - 1C= 10%

Which ONE of the following describes:

1) the MAXIMUM number of SGs that may be fed at the same time

and

2) the Feed Flow limit, if any, required for feeding the SGs IAW FRP-H.1?

A. 1) Feed ALL SGs at once.

2) No feed flow limit.

B. 1) Feed ALL SGs at once.

2) Limited to between 20-100 gpm.

C. 1) Feed ONE SG ONLY at a time.

2) No feed flow limit.

D✓ 1) Feed ONE SG ONLY at a time.

2) Limited to between 20-100 gpm.

Changed second parts of each to eliminate using required valves and used flow rate limit as the new second part per CE suggestions.

A - Incorrect. First part incorrect but plausible, since this would be correct if CETCs were less than 550°F, OR if CETCs were their given value and rising instead of falling. Second part is incorrect but plausible, since it would be correct per note prior to step 5 if Temps were <550 OR SG WR levels were >12% in at least two SGs. Note prior to FRP-H.1 Step 5: "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."

B - Incorrect. First part incorrect (see A). Second part correct (see D).

C - Incorrect. The first part is correct (see D). Second part is incorrect (see A).

D - Correct. Note prior to FRP-H.1, step 5 states: " **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.**

FRP-H.1, Revision 26

Previous NRC exam history if any:

059A2.04
059 Main Feedwater (MFW) System
A2 Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)
A2.04 Feeding a dry S/G 2.9* 3.4*

Match justification: The only affects on the MFW system for feeding a hot and dry SG are in a loss of heat sink where the MFW system is required to feed the SG (FRP-H.1). This question requires knowledge of the flow path and flow rate required in the MFW system (impacts of feeding a dry S/G on the MFW system) in the given condition per the procedure.

Objective:

6. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing (1) FRP-H.1, Response to Loss of Secondary Heat Sink; (2) FRP-H.2, Response to SG Overpressure; (3) FRP-H.3, Response to SG High Level; (4) FRP-H.4, Response to Loss of Normal Steam Release Capabilities; (5) FRP-H.5, Response to SG Low Level. (OPS-52533F06)

41. 059G2.1.19 001/NEW/RO/C/A 3.9/3.8/N/N/2/EDITORIAL/SAT

Unit 1 is at 100%, and the following conditions exist:

- A Computer point for 6A FW HTR Dump valve position is in the alarm condition.
- On the IPC, the following indications are observed:
  - 6A FW Heater level indicates 12" and stable.
  - V910A, 6A FW Heater level Dump valve, symbol indicates solid red on the system mimic.
  - V909A, 6A FW Heater level Drain valve, symbol indicates solid green on the system mimic.

Which one of the following explains the given indications, and states the isolation signal which will isolate the 5A FW HTR extraction steam valve, V502A, 5A FW HTR ES ISO?

1) 6A FW HTR has a __(1)__

and

2) 5A FW HTR Extraction Steam will isolate on high __(2)__.

	(1)	(2)
A.	tube leak	Level
B.	tube leak	Pressure
C✓	failed open dump valve controller	Level
D.	failed open dump valve controller	Pressure

Added "in the alarm condition" at the CE suggestion.

A - Incorrect. The first part is incorrect, since the FW HTR is actually empty. Below 14" tubes are uncovered and the htr is actually empty with steam blowing by to the HDT. Plausible, since indication is at 12" and the level indication is steady. A tube leak would cause this indication in other heaters (except for 6A & B) except for the drain valve being closed. If a tube leak were causing this indication, the drain would be open also. The second part is correct. Plausible even when combined with the incorrect first part, since it is correct for a HTR malfunction which causes steam flow through the 6A FW Htr to the HDT, but not for high liquid flow to the HDT. Often with a transient in the FW HTR strings, other heaters are affected. Confusion may exist as exactly how 6A FW HTR will affect the HDT and 5A FW HTR to cause the extraction valve to close.

B - Incorrect. The first part is incorrect (see A). The second part is incorrect. Plausible, since the number 6 heater can cause a high level in the HDT which would cause a high pressure in the 5A FW heater, however, there is no high pressure isolation in the 5A HTR, only a high level. Also, the 5A heater does not normally have any level, and the incorrect assumption could be made that it does not have a high level isolation . The 6A FW heater has a d/p isolation signal, and confusion could exist as to the 5A isolation signals.

C - Correct. Cautions in ARP-1.10, for HTR HI LEVEL and in SOP-20 state that if the 6A HTR level drops below 14" (18-19" is normal level controlled by the drain valve), then the tubes will be uncovered and the tank will empty. Even when empty, the tank will indicate 12". The dump being open below the normal level of 18-19" and the drain being closed is indication that the dump valve controller is failed to demand full open, or at least controlling at too low of a level. The same cautions state that with the 6A FW HTR empty, steam will blow by to the HDT, pressurize the HDT AND the 5A FW HTR, and extraction steam will isolate on high 5A FW HTR level.

D - Incorrect. The first part is correct (see C). The second part is incorrect (see B). Plausible, since it may be understood that below 14" the heater empties and blows steam to the HDT instead of subcooled liquid, but confusion may still exist as to the relationship between the HDT pressure going up, causing the 5A FW HTR pressure to go up, and the 5A FW HTR extraction steam isolating on high level. Also, the 5A heater does not normally have a level, and the incorrect assumption could be made that it does not have a high level isolation either, but could have a pressure isolation.

ARP-1.10, KC4, FW HTR OR DRN TK LVL HI, **Version 64.0**

AUTOMATIC ACTION

If level is not stabilized, extraction steam to the HP and LP heaters will automatically isolate. See Table 1 for information next page.

CAUTION: DO NOT let #6 FW HTR level trend below 14". If the tubes are uncovered level will indicate 12", but steam is actually blowing by and pressurizing the HDT. This results in level increase in the #5 FW HTR due to its [IN]ability to drain to the HDT. On high level, the #5 FW HTR extraction steam will close. (AI 2008202332)

FNP-1-SOP-20.0, FEED WATER HEATER EXTRACTION, VENT AND DRAIN SYSTEM, **Version 53.0**

3.7 If the tubes in #6 FW HTR are uncovered level will indicate 12", but steam is actually blowing by and pressurizing the HDT. This results in a level increase in the #5 FW HTR due to its inability to drain to the HDT. On high level, the #5 FW HTR extraction steam will close. (AI 2008202330)

Previous NRC exam history if any:

059G2.1.19
059 Main Feedwater System
2.1.19 **Ability to use plant computers to evaluate system or component status.**
(CFR: 41.10 / 45.12) RO 3.9 SRO 3.8

Match justification: This question requires the evaluation of plant computer points for some parameters for a FW system component (6A FW Heater) to determine component status of two FW system components: 6A FW HTR actual level and 5A FW HTR extraction valve resulting status.

Objective:

6. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Main and Reheat Steam System components and equipment, to include the following (OPS-40201A07):
   - Normal control methods
   - Abnormal and Emergency Control Methods
   - Automatic actuation including setpoint (example SI, Phase A, Phase B, MSLIAS, LOSP, SG level)
   - Protective isolations such as high flow, low pressure, low level including setpoint
   - Protective interlocks
   - Actions needed to mitigate the consequence of the abnormality

42. 061G2.2.37 001/FNP BANK/RO/C/A 3.6/4.6/N/N/3/EDITORIAL/SAT

At 1000 the following plant conditions exist on Unit 1:

- A TECH SPEC required shutdown was in progress due to BOTH 1A and 1B SW pumps inoperable and unavailable (not running).
- 1C SW pump is aligned to B Train.

At 1005 the following events occur:

- A seismic event caused a loss of BOTH SGFPs, a leak in the A Train SW header **and** a tear in the CST at the bottom of the tank.
- CST level is at 5 ft. and decreasing.

Given the above conditions, which one of the following correctly describes the actions for establishing AFW per SOP-22.0, Auxiliary Feedwater System, for the available AFW pumps?

SOP-22.0 will direct establishing SW to the suctions of ________________

A. the 1A and 1B MDAFW pumps with SW valve alignments made from the main control room **ONLY**.

B. the TDAFW and the 1B MDAFW pumps with SW valve alignments made from the main control room **ONLY**.

C. ALL AFW pumps with SW valve alignments made from in the plant **AND** from the main control room.

D.✓ the TDAFW pump with SW valve alignments made from in the plant **AND** from the main control room, and the 1B MDAFW pump with SW valve alignments made from the main control room **ONLY**.

changed the stem for the question to enhance stem focus per CE suggestion.

A - Incorrect. Correct for B MDAFW pump only, and incorrect for A MDAFW pump. B has a source from B train SW with MCR valve alignments only. Even though B MDAFW pump has a suction source from B train SW, A MDAFW pump has no procedurally allowed suction source from A train SW per SOP-22. However, the system would allow cross connecting trains to supply both A and/or B MDAFW from B train SW. Plausible, since in this emergency, it could be incorrectly assumed that maximum flexibility would be written into the procedures to allow this option to cool the SGs and Core.

B - Incorrect. Correct for B MDAFW Pump. TDAFW has a source of suction from A train only aligned to enable supplying with only MCR MOV operations. Plausible, since the TDAFW suction can be supplied from B train AFW with manual valve alignments outside of the Control Room.

C - Incorrect. Correct for B MDAFW and the TDAFW Pumps. Incorrect for A MDAFW pump. Plausible, since the system is versatile enough to allow cross connecting trains to supply both A and/or B MDAFW from B train SW. In this emergency, it could be incorrectly assumed that maximum flexibility would be written into the procedures to allow this option to cool the SGs and Core.

D - Correct. Procedurally, B MDAFW (with MCR valve alignments only) and TDAFW pump (with in-plant and MCR valve alignments required) are both able to use B train SW as an auxiliary suction source.

SOP-22, AUXILIARY FEEDWATER SYSTEM, Version 64.0

Previous NRC exam history if any:

061G2.2.37
061 Auxiliary / Emergency Feedwater System
2.2.37 **Ability to determine operability and/or availability of safety related equipment.**
(CFR: 41.7 / 43.5 / 45.12) RO 3.6 SRO 4.6

Match justification: Determining availability is more on the RO level than determining operability (other than determinations of low discriminatory value such as pump trips on overcurrent, no power to start pumps, etc.). This question requires knowledge of which of the AFW pumps are available from only one train of their alternate suction source of Service Water when their primary suction source, the CST, is not available. It also requires knowledge of how the pumps will become available from their alternate suction source procedurally and by location of the valve manipulations.

Objective:

7. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of AFW System components and equipment to include the following (OPS-40201D07):

- Normal Control Methods
- Abnormal and Emergency Control Methods
- Automatic actuation including setpoints (examples - SI, Phase A, Phase B, MSLIAS, LOSP or SG level)
- Actions needed to mitigate the consequence of the abnormality

43. 062A1.01 001/NEW/RO/C/A 3.4/3.8/N/N/4/CVR/SAT

Unit 1 has had a LOSP followed by a SBLOCA, and the following conditions exist:

- The 1-2A DG is tagged out.
- The 1B DG is tripped.
- The 1C DG is supplying power to the A Train busses.
- The load on the 1C DG is 2.860 MW.

Which one of the following describes whether or not the 2000 amp hour rating on the 1C DG will be exceeded if the 1B PRZR HTR GROUP BACKUP is energized?

IF the PRZR HTR GROUP BACKUP is energized, THEN the 1C DG 2000 hour load rating __(1)__ be exceeded,

and

energizing the 1B PRZR HTR GROUP BACKUP __(2)__ allowed IAW EEP-1, Loss of Reactor or Secondary Coolant.

	(1)	(2)
A.	will	is
B.✓	will	is **NOT**
C.	will **NOT**	is
D.	will **NOT**	is **NOT**

Note: in this alignment the 1C DG has been manually aligned to the 1F bus.

A - Incorrect. The first part is correct (see B). The Second part is incorrect (see B). Plausible since the procedure does state that in an emergency, the design of the electrical system has determined that a slight overload may exist after a LOSP and a LOCA. This is acceptable as long as the 2000 hour rating is not exceeded. Also, the Basis of TS 3.8.3 allows the 2000 hour limit to be exceeded for up to 300 hour per year but EEP-1 forbits exceeding the 2000 hour limit. Confusion may exist as to which one or if both the continuous and/or the 2000 ratings may be exceeded for a period of time. Also, the continuous load limit and the 2000 hour rating values may not be remembered properly. However, the procedure states that MANUAL loading above EITHER the continuous or 2000 rating is not allowed. Confusion could exist as to what the 2000 hour load allows, i.e. it does not allow overloading above the limit for any period of time as the continuous load limit does.

B - Correct. The continuous load limit is 2.850 MW, and the 2000 hour load limit is 3.100 MW. 2.860+0.30=3.16MW > continuous load limit. EEP-1 APP 4, Step 1 caution states Do NOT manually load diesel generators above 2000 hr. load limit. . Per EEP-1, Att. 4: "...continuous load rating limit (i.e. **2.85 MW for small DGs**, **4.075 MW for large DGs**). Under these circumstances, **diesel generator loading may be raised not to exceed the 2000 hour load rating limit (i.e. 3.1 MW for small DGs**, **4.353 MW for large DGs...)**.

C - Incorrect. The first part is incorrect. Plausible, since the continuous load limit and the 2000 hour rating may not be remembered properly, and/or the load in MW of the przr heaters may be confused with other smaller loads. The large DGs have a higher load limit of 4.075 & 4.353 MW for continuous and 2000 hr rating respectively. The second part is incorrect (see A).

D - Incorrect. The first part is incorrect (see C). The second part is correct (see B). Plausible, since the continuous load liming is already exceeded, and manual loading above the continuous load is not allowed, even though automatic loading above the continuous load limit is allowed in an emergency.

FSD, Diesel Generator System

3.1.6 Interface Requirements

The only time during operation (other than design basis accidents) that the diesel is intentionally loaded above its continuous rating is during Technical Specification surveillance testing when the diesels are loaded to their 2000h ratings (4353 KW for the large diesels and 3100 KW for the small diesels).

APPENDIX B
STATIC LOADING OF THE DIESEL GENERATORS
B.2.0 INTRODUCTION

During some design basic events, diesel generator 1C is loaded above its continuous rating by less than 5%. However, this calculated loading above the continuous rating is acceptable since the diesel loading still meets the criteria contained in Position C.2 of Safety Guide 9 (Reference 6.7.028).

G.4.3 Potential Diesel Generator Overload

The potential exists for DG overload if the LOSP is followed by a LOCA after step 2 of the LOSP sequencer has been energized. In those cases, the DG will be loaded with the Reactor Cavity Cooling Fan (13 Kw) and the CRDM fan (84 Kw) in addition to the ESS loads, and the operator may have to remove selected loads if the DG is loaded above its rated capacity. This situation does not constitute a concern given the existing guidance in the plant procedures which provides the operator with guidance for reducing the DG loading if it is above rated capacity.

EEP-1 LOSS OF REACTOR OR SECONDARY COOLANT Revision 29
ATTACHMENT 4
VERIFYING 4160 V BUSSES ENERGIZED

1 Verify 4160 V busses energized.

****

CAUTION: IF a DG is already operating above its continuous load rating, THEN additional manual loads should not be added. Unanticipated plant emergency conditions may dictate the need to load the emergency diesel generators above the continuous load rating limit (i.e. **2.85 MW for small DGs**, 4.075 MW for large DGs). Under these circumstances, **diesel generator loading may be raised not to exceed the 2000 hour load rating limit (i.e. 3.1 MW for small DGs**, 4.353 MW for large DGs). Diesel loading should be reduced within the diesel generator continuous load rating limit as soon as plant conditions allow.

****

****

CAUTION: To prevent diesel generator overloading, at least 0.3 MW of diesel generator capacity must be available prior to energizing a group of pressurizer heaters.

****

1.7.4 RNO Energize pressurizer heater group 1B as required.

Previous NRC exam history if any:

062A1.01
062 A.C. Electrical Distribution
A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: (CFR: 41.5 / 45.5)
A1.01 Significance of D/G load limits 3.4 3.8

Match justification: In this question parameters are provided which must be evaluated to predict if the DG will be overloaded if a load is manually started. The size of the load in MWs must be known and the load limit must be know to predict if the load may be started and if it will exceed the design limits. The significance of the load limits must be understood, since the continuous limit may be exceeded in an emergency without expected damage to the DG, but the 2000 hour load limit may not be exceeded for any reason.

Objective:

2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the components associated with the Diesel Generator and Auxiliaries System, to include the following (OPS-40102C02):
   a. PC2 Diesel, including capacity
   b. FM Diesel, including capacity

44. 063A3.01 001/FNP BANK/RO/C/A 2.7/3.1/Y 2007/N/4/CVR/SAT

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

- 1A Battery Charger is on service.
- EM personnel are doing preventative maintenance on the 1A battery.

The following indications and alarms are received:

- The UNIT 1 AUX BLDG DC BUS - A TRN GROUND DET white light comes on momentarily and then goes OFF.
- WC3, 1A 125V DC BUS BATT BKR 72-LA05 TRIPPED
- WC2, 1A 125V DC BUS UV OR GND alarms and clears.

Which ONE of the following describes the status of the indications on the EPB for the 1A DC BUS and the 1A and 1B Inverters?

1A DC BUS VOLTAGE reads approximately ____(1)____

1A and 1B INVERTER AMPERES are reading approximately ____(2)____

A. (1) 0 DC VOLTS.

(2) 25 amps and being powered from the bypass source.

B. (1) 0 DC VOLTS.

(2) 0 amps and being powered from the normal source.

C. (1) 125 DC VOLTS.

(2) 0 amps and being powered from the bypass source.

D✓ (1) 125 DC VOLTS.

(2) 25 amps and being powered from the normal source.

explanation

When the Battery output breaker is opened, LA-05, WC3 will come into alarm due to the b contact from breaker LA05. WC2 shows either a low voltage condition or a ground. In this case it would be a ground.

The battery output breaker has opened due to a ground on the battery and when it opens WC2 clears. The annunciators provide indication that the breaker opened and the white light provides indication of the ground. For this set of circumstances, the battery is no longer aligned to the bus and the battery charger is carrying the load. The indications will remain normal and the inverters will have normal indications. The inverters will not swap to the bypass source and will still be powered from the BC.

A - Incorrect. 0 DC volts on the 1A DC bus indicates the bus is de-energized. The bus still has power from the Batt. chger. The inverters will be powered from the BC or the normal supply and will indicate 25 amps. If it were to swap to the bypass source, it would still have amp readings, but if the manual bypass switch were to be placed in the bypass position, then the amps would be 0 amps.

B - Incorrect. 0 is not correct for both. Normal is correct.

C - Incorrect. 125 is correct. 0 is not correct and bypass is not correct.

D - Correct. 125 is correct and 25 is correct from the normal source.

DWNG: D177082 sheet 1

Previous NRC exam history if any: 2007 FNP NRC exam, this question is the only one in the bank tied to this K/A

063A3.01

063 D.C. Electrical Distribution

A3 Ability to monitor automatic operation of the DC electrical system, including: (CFR: 41.7 / 45.5)

A3.01 Meters, annunciators, dials, recorders, and indicating lights 2.7 3.1

Match justification: It meets the KA in that it tests the ability to determine the proper readings on the EPB for an abnormal condition based on the indications and alarms received (white light and annunciators). The automatic portion of the KA is the breaker opening on an overcurrent condition.

Objective:

6. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the DC Distribution System components and equipment, to include the following (OPS-40204E07):

- Normal control methods
- Abnormal and Emergency Control Methods
- Automatic actuation including setpoint
- Protective isolations
- Protective interlocks
- Actions needed to mitigate the consequence of the abnormality

45. 064A4.03 001/NEW/RO/MEM 3.2/3.3/N/N/3/REVISED/FIX

The 1B DG is being paralleled with the grid for surveillance testing, and prior to closing the 1B DG output breaker, the Synchroscope is turning **fast** in the FAST direction.

Which one of the following states:

1) the component with the **highest frequency** (the 1B DG output or 1G 4160V Bus), and

2) which **switch** must be turned to adjust frequency prior to closing the output breaker?

	(1)	(2)
A.	1G Bus Frequency	VOLTAGE ADJUST VOLTS/MVARS
B.	1G Bus Frequency	GOVERNOR MOTOR SPEED/MW
C.	1B DG Frequency	VOLTAGE ADJUST VOLTS/MVARS
D✓	1B DG Frequency	GOVERNOR MOTOR SPEED/MW

Changed the second parts due to plausibility concerns in two distractors. Made the second parts test which switch to use to adjust frequency due to previous NRC exam mistake on the EPB observed by the CE (applicant on previous NRC exam inadvertently use the voltage adjust switch to adjust frequency).

A - Incorrect. Both parts are incorrect, and are the exact opposite of the correct. Plausible, since confusion could exist as to the relationship in this scenario between the two frequencies (which is running and which is incoming) and which switch to use to adjust frequency. Also, the first part of this choice would be correct for this indication if the DG was on the bus and the Grid was being paralleled on after an LOSP.

B - Incorrect. The first part is incorrect (see A). The second part is correct (see D.)

C - Incorrect. The first part is correct, (see D). The second part is incorrect (see A).

D - Correct. The oncoming generator must be at a higher frequency output to turn the Synchroscope in the clockwise (fast) direction. It must be going slow in the "fast" direction prior to closing the breaker per STP-80.1 Step 5.9. In this case, the DG must be slowed down by going to LOWER on the Speed switch until the synchroscope is traveling slower in the same direction.

STP-80.1 Version 47

Previous NRC exam history if any:

064A4.03
064 Emergency Diesel Generators
A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
A4.03 Synchroscope 3.2 3.3

Match justification: To correctly answer this question, knowledge is required of monitoring the synchroscope operation, and interpreting what information the synchroscope is communicating. Interpreting this information from the syncroscope properly is required to determine how to adjust DG speed adjust switch for paralleling the output breaker safely.

Objective:

6. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Diesel Generator and Auxiliaries System components and equipment, to include the following (OPS-40102C07):

- Normal control methods
- Abnormal and Emergency Control Methods
- Automatic actuation including setpoint (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
- Actions needed to mitigate the consequence of the abnormality

46. 065AK3.04 001/NEW/RO/C/A 3.0/3.2/N/N/2/REVISED/FIX

A complete loss of instrument air has occurred on Unit 1, and the following conditions exist:

- The Reactor was tripped from 100% power.
- The TDAFW pump auto started.
- BOTH MDAFW pumps failed to start.
- SG NR Levels are slowly trending up and read:
  1A: 27%, 1B: 29%, 1C: 30%
- Instrument Air is expected to be lost for the next 4 hours while repairs are made.

Which one of the following describes the MAXIMUM required time to align Emergency Air compressors for the TDAFW system and the reason?

Alignment of the Emergency Air Compressors to the TDAFW components IAW SOP-62 is required within a MAXIMUM of _(1)_

and

is required in order to _(2)_.

	(1)	(2)
A.	1 hour	ensure adequate heat sink
B.✓	2 hours	ensure adequate heat sink
C.	1 hour	prevent exceeding Tech Spec cooldown limits
D.	2 hours	prevent exceeding Tech Spec cooldown limits

Changed both parts of the answer choices and the question per CE suggestions.

A – Incorrect. The fist part is incorrect, but plausible, since the correct time is 2 hours and confusion may exist as to the exact time of the design of the air accumulators. The second part is correct (see B).

B – Correct. Per AOP-6.0, Version 35, Step 8 below. For the first two hours after the loss of air, the installed accumulators maintain the steam admission valves open, but after that the emergency air compressors must be started to supply them with air to maintain them open, maintain TDAFW pump speed, and to maintain AFW flow to the SGs for a heat sink. With only one AFW pump available and SG levels below 31%, at least 395 gpm TDAFW flow is required for a heat sink.

AOP-6.0, Version 35, Step 8

8 Maintain SG narrow range levels between 35-69%.

****

CAUTION: The TDAFW Pump steam admission valves will fail closed within two hours if emergency air is not aligned.

****

8.1 A/ER WHEN TDAFW Pump is started, THEN vary TDAFW Pump Speed to control AFW flow.

TDAFWP SPEED CONT
[ ] SIC 3405 adjusted

8.1 RNO IF the TDAFW Pump steam admission valves fail closed, THEN align emergency air using FNP-1-SOP-62.0, EMERGENCY AIR SYSTEM.

C- Incorrect. The first part is incorrect (see A). Second part is incorrect, but plausible, since the SG levels are trending up and close to the levels at which AFW flow can be secured, and at Beginning of Core Life cooldown would be excessive with full TDAFW Flow. Also, throttling with the jacking devices for the FCVs would be an option to limit excessive cooldown if necessary. However, AOP-6.0 would direct reducing TDAFW pump speed from the MCB Pot to control the amount of AFW Flow to the SGs. With no Instrument air for 4 hours, decay heat would require steaming the SGs and makeup from the only source of AFW to the SGs would continue to be required even after SG levels were > 31% NR.

D - Incorrect. The first part is correct (see B). The second part is incorrect (see C).

AFW FSD, A181010

3.14.7.1 The emergency air system shall provide a backup air source for these valves as a means of additional reliability in admitting steam to the turbine for TDAFW pump operation (References 6.7.049, 6.7.050).

3.14.7.2 The instrument air system shall supply clean, dry air at a range of 80 to 100 psig to the air reservoir for steam supply isolation valve operation (Reference 6.5.001).

3.22 TDAFW PUMP UPS SYSTEM
TPNS No. QN23L001-AB (UPS) and QN23E001-AB (Battery)

3.22.1 Basic Function
The UPS system shall be designed to provide an uninterruptible source of 120 V ac and 125 V dc power supply for control of the TDAFW pump turbine drive (QN23P003), steam admission valve (QN12HV3226), steam supply isolation valves (QN12HV3235A, B), instrument air valves (NN12SV3412A, B) and the TDAFW pump discharge flow control valves (QN23HV3228A, B, C) for a minimum period of 2 hours considering loss of both offsite and the backup diesel power to the UPS (Reference 6.7.074).

D175035L (Emer air compressors)
D175033 sh. 2 (D-9) Air supply to SG Atmosph & TDAFWP Stm Admission valves

Previous NRC exam history if any:

065AK3.04

065 Loss of Instrument Air

AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: (CFR 41.5,41.10 / 45.6 / 45.13)

AK3.04 Cross-over to backup air supplies 3.0 3.2

Match justification: AOP-6, Knowledge of the reasons for Cross-over to backup air supplies [1C or 2C AC, N2 to PORVs, Emerg. AC's for SG ATMOSPERICS] as they apply to Loss of Instrument Air. This question asks for the action and reason for the action on loss of instrument air. The reason for starting the Emergency air compressors in the given scenario is that SG makeup is needed for a heat sink and the other options will not provide an adequate heat sink for 4 hours.

Objective: AOP-6.0

1 **STATE AND EXPLAIN** the operational implications for all Cautions, Notes, and Actions associated with AOP-6.0, Loss of Instrument Air. (OPS-52520F03)

2. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing AOP-6.0, Loss of Instrument Air. (OPS-52520F06)

47. 068K6.10 001/MOD/RO/C/A 2.5/2.9/N/N/3/REVISED/SAT

A Unit 1 #2 Waste Monitor Tank release to the environment is in progress IAW a Liquid Waste release permit and SOP-50.1, Appendix 2, Waste Monitor Tank #2 Release to the Environment.

- FH2, RMS CH FAILURE, comes into alarm.
- R-18, LIQ WASTE DISCH, is indicating normal on the Radiation Monitoring system console and on the recorder for R-18, RR0200.
- The HIGH Alarm and LOW Alarm red lights are illuminated on the R-18 drawer.
- The control power fuse holder is found to be illuminated on R-18.

What effect, if any, does this condition have on RCV-18, and the #2 WMT pump?

A. 1) RCV-18 will **NOT** automatically close.

2) #2 WMT pump will **NOT** automatically trip.

B.✓ 1) RCV-18 will automatically close.

2) #2 WMT pump will **NOT** automatically trip.

C. 1) RCV-18 will **NOT** automatically close.

2) #2 WMT pump will automatically trip.

D. 1) RCV-18 will automatically close.

2) #2 WMT pump will automatically trip.

Added "on the R-18 drawer" to specify where the red lights are illuminated. Changed fuse is lit to fuse "holder" is lit.

Changed second part of all distractors to enhance plausibility per CE suggestions. This change makes the second part meet the k/a in addition to the first part: effect of R-18 loss on the Liquid radwaste system.

A - Incorrect. First part is incorrect, since the control power has been lost and will initiate the automatic action the same as if a valid high rad alarm condition existed. Plausible, since no high alarm condition exists, and the meter is reading a normal reading. The second part is correct (see B).

B - Correct. Control power being lost causes a fail safe automatic function of closing RCV-18. The second part is correct since R-18 does not trip the WMT pump.

C - Incorrect. The first part is incorrect (see A). The second part is incorrect since WMT will not automatically trip. Plausible, since stopping the release is an automatic control function of RE-18, and when the WMT level gets to a predetermined level the release is stopped by tripping the WMT pump. However, WMT low level trips the WMT but high radiation only closes RCV-18. SOP-50.1, Step 2.5.16 "Verify pump tripped at low level set point or secure pump."

D - Incorrect. The first part is correct (see B). The second part is incorrect, (see C).

FNP-1-SOP-50.1, Version 66.0

3.2 IF R-18 becomes inoperable while discharging liquid waste to the river, THEN the discharge shall be stopped immediately and the Shift Support Supervisor notified.

FNP-1-ARP-1.6, RMS CH FAILURE, FH2, Version 64.0

1. The radiation monitors fail to a "High Radiation" condition on loss of instrument and/or control power that will result in actuation of associated automatic functions. Refer to annunciator FH1 for automatic actions.

FNP-1-ARP-1.6, RMS CH FAILURE, FH1, Version 64.0

4.18 IF **R-18** alarms with high liquid effluent activity possible, THEN verify any liquid waste release is secured and refer to FNP-1-SOP-50, LIQUID WASTE PROCESSING SYSTEM for potential problems with the liquid waste system.

Previous NRC exam history if any:

068K6.10

068 Liquid Radwaste System

K6 Knowledge of the effect of a loss or malfunction on the following will have on the Liquid Radwaste System: (CFR: 41.7 / 45.7)

K6.10 Radiation monitors 2.5 2.9

Match justification: The only Radiation monitor for which a loss or malfunction would affect the Liquid Radwaste System is R-18. This question requires knowledge of how a failure of R-18, during a liquid waste release, would affect the Liquid Radwaste system. In order to produce 3 plausible but incorrect distractors, the the auto function of the WMT pump/low tank level trip are part of each choice in the second part.

Objective:

5. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Radiation Monitoring System components and equipment, to include the following (OPS-40305A07):

- Normal control methods
- Abnormal and Emergency Control Methods
- Automatic actuation
- Protective isolations
- Protective interlocks
- Actions needed to mitigate the consequence of the abnormality

48. 069AA2.01 001/FNP BANK/RO/MEM 3.7/4.3/Y 2007/N/2/CVR/SAT

Which one of the following conditions represents a loss of containment integrity and would cause entry into Tech Spec 3.6.1, Containment?

A. Mode 3 and one of the Personnel Airlock doors will not close.

B.✓ Mode 4 and Integrated Leak Rate test determines that leakage is not within limits.

C. Mode 5 and it is discovered that the Phase 'B' isolation valve for CCW to the RCPs, will not close.

D. Mode 6 and the Equipment Hatch is held in place by 4 bolts ONLY.

A - incorrect. Both doors inop would be a loss of Containment Integrity, this is just an inop of one of the doors in the Personnel Airlock. Plausible because one of two series valves at a containment penetration makes containment integrity LCO not met.

B - correct. Surveillance requires ILRT to be within limits for Containment Integrity to be set.

C - incorrect. because Containment Integrity is not required in Mode 5, plausible because the valve is part of a containment penetration that would affect integrity in modes 1-4.

D - incorrect. 4 bolts meets the minimum requirement for Containment Closure in Mode 6, but not containment integrity in the modes that containment integrity is required.

TS 3.6.1

Previous NRC exam history if any: 2007 FNP NRC EXAM, this is the only question in the FNP bank tied to this k/a.

069AA2.01

069 Loss of Containment Integrity

AA2. Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: (CFR: 43.5 / 45.13)

AA2.01 Loss of containment integrity 3.7 4.3

Match justification: This question requires knowledge of the ability to determine IF Ctmt integrity is lost or met in different modes IAW Tech Specs. Mode applicability (1-4) & one hour or less tech specs (one or more air locks with one door inoperable) are RO level Knowledge.

Objective: OPS-52102A-1

49. 071A4.01 001/NEW/RO/MEM 2.7/2.2/N/N/3/EDITORIAL/SAT

Unit 1 is 60% power at EOL, and the following conditions exist:

- #1 RHT is On Service and level is 10%.
- LK-112, LTDN TO VCT FLOW, has been adjusted to maintain the VCT at 40% level in AUTO.
- A manual makeup of 400 gallons at a rate of 40 gpm has been set up as the unit ramps up in power.
- #2 RHT is Off Service and level is 50%.
- #2 RHT gas space under the bladder is being educted IAW SOP-2.4, Chemical And Volume Control System Boron Recycle System, using the step entitled "#1 (#2, #3) RHT venting with RHT GAS SAMPLE PANEL."

Which one of the following describes:

1) the indication of LCV115A, VCT HI LVL DIVERT VLV, in response to the manual makeup

and

2) the indication that the educting of #2 RHT is complete or almost complete IAW SOP-2.4?

A.✓ 1) LCV115A will indicate RED (VCT) **AND** WHITE (HU TANK) lights LIT.

2) Gas panel annunciator #23, RHT EDUCTOR LO PRESS, comes into alarm when educting is **almost** complete.

B. 1) LCV115A will indicate RED (VCT) **AND** WHITE (HU TANK) lights LIT.

2) PCV-251, RHT Eductor Suction Line Pressure Control valve, indication will change from red light LIT to green light LIT when educting **is** complete.

C. 1) LCV115A will indicate RED (VCT) light LIT, WHITE (HU TANK) light **NOT** LIT.

2) Gas panel annunciator #23, RHT EDUCTOR LO PRESS, comes into alarm when educting is **almost** complete.

D. 1) LCV115A will indicate RED (VCT) light LIT, WHITE (HU TANK) light **NOT** LIT.

2) PCV-251, RHT Eductor Suction Line Pressure Control valve, indication will change from red light LIT to green light LIT when educting **is** complete.

VCT = 15 gal/%. If VCT level at 20%, then a 400 gallon add will result in level rise to 46%. An EOL ramp up, it is operationally valid to add 400 gallons for Xe buildup.

On the LCV115A lights on the MCB the following is written:
RED (VCT) **AND** WHITE (HU TANK) This is the reason this is provided in the distracter.

A - Correct. LCV 115A is a three way valve that has only a red and a white light indicated on the MCB, and unlike most other valve indications of the MCB, it has no green light. The red light is on when the valve is not fully diverted to the RHT and the white light is on when the valve is not fully aligned to the VCT. The red light indicates at least some flow going to the VCT, and the white light indicates at least some flow going to the RHT. When the valve is in mid position, both lights are on, such as is the case with a 40 gpm makeup. Verified on the simulator laptop computer. The second part is correct per 4.9.14 and associated NOTE of SOP-2.4. This alarm indicates 7" vacuum, and is an indication that educting is almost complete. Educting must be secured by 20" vacuum.

B - Incorrect. The first part is correct (see A). The second part incorrect, since PCV-251, IF it were open in this lineup, would close at 7" vacuum automatically, but the educting can continue to a higher vacuum than 7" in this educting lineup. Plausible, since PCV-251 would automatically close at 7" and secure the educting if the installed piping for educting and installed valve (PCV-251) was used for the educting flowpath (per Step 4.8.7 of SOP-2.4). However, in the educting flowpath specified in the stem, using a portable vacuum pump and discharging directly to the plant vent, PCV-251 is not in the flowpath, and allowance is made by the procedure to go to a vacuum of 20" vice 7".

C - Incorrect. The first part is incorrect (see A). The red light only would be lit if all water was flowing to the VCT. Plausible, since this choice would be chosen if confusion existed as to which valve position the red indicates (normally open on other valve indications). Also, the white light may not be understood. It may be thought to be a full divert indication instead of a partial OR full divert indication. This is a non-standard light indication arrangement, since most valves are not three way, and have red and green lights for open and close indications. For example, the TCV143 hi letdown temperature divert valve HS white light is for the VCT position and is next to the LCV115A HS, which has the white light for the RHT position and the red light for the VCT position. The second part is correct (see A).

D - Incorrect. The first part is incorrect (see C). The second part is incorrect (see B).

A-181009, CVCS/HHSI/ACCUMULATOR/RMWS
3.14.1 Basic Functions
Valve LCV-115A controls the amount of letdown flow diverted to the RHTs on high VCT level. Valves LCV-115B,C,D and E are actuated to isolate the VCT on low level and provide a suction source for the charging pumps from the RWST.

3.14.2 Functional Requirements
Modulate Divert (LT-112) -- This setpoint shall start diversion of letdown to the RHTs via valve LCV-115A. The valve shall modulate open as required, based on maintaining the level at the modulate divert setpoint. (References 6.2.1, 6.2.9, 6.2.58, 6.3.32, and 6.3.13)

FNP-1-SOP-2.5, RCS Chemical Addition, VCT Gas Control, And Demineralizer Operation, Version 67.0
Appendix 2 Flushing Cation and Mixed Bed Demineralizers to the RHT's

4.2.5 Commence blended or batch makeup to the VCT equal to desired RCS boron concentration per FNP-1-SOP-2.3 CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM.

FNP-1-ARP-13.1, Boron Recycle Processing Panel, Version 8.0
Annunciator window 23, Recycle Holdup Tank Eductor Lo Press

SOP-2.4, Chemical And Volume Control System Boron Recycle System, Version 55.0

4.1 **CAUTION: RHT may overflow if level is allowed to exceed 50% without venting RHT.**

4.1.1.10 Monitor RHT level as follows:

1. Frequently check indicated level for expected level rise.

NOTES: • On service RHT is normally swapped prior to exceeding 50% level.
• SS approval required to exceed 50% level.

2. IF intentionally filling an RHT >50% THEN, verify RHT has been educted within the last 30 days prior to exceeding 50%.
3. WHEN transferring water to an RHT with indicated level > 50% THEN, check bladder pressure approximately every 30 minutes. (Ref. OR 2-96-329)
4. IF bladder pressure > 0.5 psig THEN, educt (vent) RHT using desired section of this procedure.

4.8 #1 (#2, #3) RHT venting with WASTE GAS SYSTEM.

4.9 #1 (#2 #3) RHT venting with RHT GAS SAMPLE PANEL.

4.12 #1 (#2, #3) RHT pressure check for gas buildup under bladder using temporary gage or installed instrumentation.

Previous NRC exam history if any: none

071A4.01
071 Waste Gas Disposal System
A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
A4.01 Valve to put the holdup tank into service; indications of valve positions and tank pressure .2.7* 2.2*

Match justification: The Holdup tank at FNP is the Recycle Holdup Tank (RHT), which is filled (put into service) when LCV-115A, Letdown divert valve to the RHT, shifts to the divert position. LCV-115A position is indicated in the Control Room. This question requires knowledge of the how this valve indicates the position during a continuous dilution evolution. The tie to the Waste Gas system is that RHT pressure can build up under the bladder from gasses coming out of solution, and the question requires knowledge of how to monitor in the control room while operating the holdup tank, and the gas system during educting the holdup tank, at the given water level to educt the gas space under the bladder to mitigate the effects of the pressure accumulation (the actual RHT pressure guage is local in the plant, and not remote in the control room). An RHT pressure alarm is used for indication of when to secure the educting, and it indicates both locally at the Gas panel and at a common Gas and Liquid Waste alarm in the control room. Applying this k/a to FNP is challenging, and making it a discriminating question that isn't trivia is also challenging, but this has been accomplished in this question.

Discussed with lead examiner 9-21-09 that the only holdup tank at FNP is the

Recycle Holdup Tank that received liquid RCS water, and the only connection between the holdup tank and the Waste gas system is when educting the gas space under the bladder (required at FNP when the tank gets to 50% level). The only Control Room indications are for the valve which diverts letdown flow to the RHT placing the holdup tank in service) and the common MCB alarm which alarms when the Educting is almost complete. Diverting flow to the RHT will, as RHT level rises to 50%, require educting with the waste gas system OR the new method of using a portable vacuum pump discharging to the Plant Vent stack. The original method of educting is not normally used (educting with the Waste Gas Compressors to a waste gas decay tank), although it is still installed, trained on, and in the procedure. The old method could still be used at any time, but a portable vacuum pump is normally used to educt the tank directly to the Plant Vent Stack.

Objective:

7. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Waste Gas System components and equipment, to include the following (OPS-40303B07):

- Normal control methods

Abnormal and Emergency Control Methods

Automatic actuation including setpoint (example SI, Phase A, Phase B, MSLIAS, LOSP, SG level)

Protective isolations such as high flow, low pressure, low level including setpoint

Protective interlocks

Actions needed to mitigate the consequence of the abnormality

2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the components associated with the Chemical and Volume Control System, to include the components found on Figure 3, Chemical and Volume Control System and Figure 4, RCP-Seal Injection System (OPS-40301F02).

50. 072G2.1.27 001/NEW/RO/MEM 3.9/4.0/N/N/2/EDITORIAL/FIX

Which one of the following states **ONLY** correct purposes and/or functions of the Area Radiation Monitoring System (ARMS)?

A.✓ • Provide indication of Loss of RCS barrier indication for Emergency Event Classification.

• Provide indication for when Adverse Numbers are used while performing Emergency Response Procedures.

B. • Provide indication of Loss of RCS barrier indication for Emergency Event Classification.

• Provide control functions to shift ventilation to recirculation in the event of high radioactivity.

C. • Provide control functions to isolate liquid effluent processes in the event of high radioactivity.

• Provide indication for when Adverse Numbers are needed while performing Emergency Responce Procedures.

D. • Provide control functions to isolate liquid effluent processes in the event of high radioactivity.

• Provide control functions to shift ventilation to recirculation in the event of high radioactivity.

Changed the correct answer to be two parts which were less general per CE suggestions. Used these two parts with incorrect but plausible parts for other distractors.

A - Correct. Per FSD: Radiation Monitoring System, A-181015. The FSD lists the subcategories of the entire RMS system, of which the ARMS is a subset along with PERMS and the Atmosphere Radiation Monitoring System. Each section in the FSD under 3.1, Area Monitors 3-1 describes the functions of each ARMS monitor. Knowing which RMS monitors are ARMS monitors and the different functions of each monitor is required to answer this question correctly.

The first part is correct: R-2 & R-7 are ARMS monitors and they are both used to indicate a loss of fission product barrier per EIP-9.2 Figure 1, Fission Product Barrier Evaluation Modes 1, 2, 3, and 4.

The second part is correct: R-27 is an ARMS monitor and they are used to indicate when adverse numbers are used in the ERGs per SOP-0.8, Version 18

3.18 Use of Adverse Containment Values

Due to the affect of environmental conditions on instrumentation, the ERP's are written such that some operator decisions are based on conditions within containment. The following guidelines are used to determine the continued applicability of adverse values:

- If adverse values were entered due to containment pressure > 4 psig, then adverse values remain in effect until pressure is < 4 psig and the applicable instruments have been channel checked.

- If adverse values were entered due to containment radiation > 10^5 R/hr, then adverse values remain in effect until the integrated radiation dose is verified to be less than 10^6 Rad as indicated on IPC display 3C3.

B - Incorrect. The first part is correct (see A). The second part is incorrect. Plausible, since one of the Area radiation monitors stops the ventilation fans (Low Level Rad Waste Building area monitors), but it does not shift to recirc. Other area monitors such as R-1A & R-1B, CR & TSC respectively, have ventilation that shifts to recirc on high radiation, but due only to R-35A & 35B, and not due to any ARMS monitor.

C - Incorrect. The first part is incorrect. Plausible, since other parts of the RMS system than the ARMS have this function (PERMS), but not the ARMS. The second part is correct (see A).

D - Incorrect. Both parts are incorrect (see C & B).

FSD: Radiation Monitoring System, A-181015

1.1 SYSTEM OVERVIEW

The RMS is designed to perform three basic functions:

- Provide warning of any radiation hazard that could develop.
- Provide advance warning of a plant malfunction that could lead to a health hazard or plant damage.
- Provide a warning of any potential inadvertent release of radioactivity to the environment.

The RMS at Farley Nuclear Plant is divided into three subsystems:

- Area Radiation Monitoring System (ARMS)
- Process and Effluent Radiation Monitoring System (PERMS)
- Atmosphere Radiation Monitoring System

Previous NRC exam history if any:

072G2.1.27
072 Area Radiation Monitoring System
2.1.27 **Knowledge of system purpose and/or function.** (CFR: 41.7) RO 3.9 SRO 4.0

Match justification: Knowledge of what the purposes and functions of the ARMS part of the Radiation monitoring system is required to answer this question. Knowledge is also required of which part of the Radiation monitoring system is comprised of the Area Radiation monitoring system, and what it's purpose(s) and function(s) is(are)-as opposed to the purposes and functions of the PERMS and/or "Atmosphere Radiation Monitoring System". There are only 3 functions/purposes of the RMS system, and all of them apply to the ARMs. To obtain distractors that were plausible but wrong, functions of parts of the RMS were used that were not part of the ARMs subsystem. Chose "shift ventilation to recirc and "secures a liquid release" to ensure the distractors were plausible but definitely wrong (these are accomplished by RMS, but NOT by ARMs). ARMs does provide functions to secure a ventilation system (R-60s) and to secure an airborne release (R-60s).

Objective:

1. **STATE AND EXPLAIN** the purpose and/or function(s) of the Radiation Monitoring System (OPS-40305A01)

51. 073K3.01 001/NEW/RO/C/A 3.6/4.2/N/N/2/NO CHANGE OK/SAT

Given the following plant conditions:

- Unit 1 is in Mode 5.
- Spent Fuel is being moved in preparation for a refueling outage.
- R-25A, SFP VENT, radiation monitor loses instrument power.

Which one of the following describes:

1) the Train(s) of PRF RECIRC and EXH fans that automatically start,

and

2) whether or not manual action is required to OPEN HV-3538A, SFP to 1A PRF SUPPLY DMPR?

A. 1) Only A train starts;

2) Manual action is required.

B. 1) Only A train starts;

2) Manual action is **NOT** required.

C. 1) Both trains start;

2) Manual action is required.

D✓ 1) Both trains start;

2) Manual action is **NOT** required.

A - Incorrect. 1A PRF system will be started directly due to the R-25A rad monitor failure, but as a result of the SFPR Differential pressure, the 1B PRF train is expected to start also.
HV3538A should already be open for the given plant conditions. Action is not required to align it.

Plausible:
1) 1A train PRF is directly started from R-25A; a separate start signal is provided to the 1B Train PRF system.
2) HV3538A & B are required to be verified open following an autostart of the PRF system per P&L 3.3 of SOP-58.0. They do not automatically open. Applicant may be aware that the dampers don't automatically operate with an alarm on R-25A or on low d/p SFP ventilation but be confused on the normal position. These dampers are required to be automatically closed during a post LOCA alignment, and any time the PRF is started up not aligned to the SFP. Applicant may be confused about when this manual operation is required.

A - Incorrect. See A for discussion and plausibility.
C - Incorrect. See A for discussion and plausibility.
D - Correct. See A for discussion.

REFERENCES:

SOP-45.0, ver 36.0, P&L 3.5 "The radiation monitors fail to a "High Radiation" conditions on a loss of instrument and/or control power that will result in actuation of associated automatic functions. [...]"

ARP-1.6, vers 64.0, FH1 and FH5; R-25A & B automatically trips the SFP Supply AHU, both EXH Fans and closes the supply and exhaust dampers. And starts the associated train PRF. Additionally, the unaffected train penetration room filtration system will start due to Low ΔP in the spent fuel pool room.

SOP-58.0, ver 70.0, Step 3.9: "PRF system auto start form R-25A or R-25B requires operator action to verify open SFP TO 1A PRF SUPPLY DMPR, [...] or SFP TO 1B PRF SUPPLY DMPR,[...]."

TSR 3.7.12 (REQUIRED during SFP movement in the SFPR) VERIFY **two PRF trains aligned** to the SFPR.

Previous NRC exam history if any: (MODIFIED? NEW?)
MODIFIED FROM AUX BLDG VT-40304B07 015 ---- 2006 NRC
MODIFIED FROM AUX BLDG VT-62107B01 004 ---- none
MODIFIED FROM RMS-40305A07 003 ---- 2001 NRC

073K3.01
073 Process Radiation Monitoring System
K3 Knowledge of the effect that a loss or malfunction of the PRM system will have on the following:
(CFR: 41.7 / 45.6)
K3.01 Radioactive effluent releases 3.6 4.2

Match justification:
- R-25A & B require PROCESS flow (SFP system flow) to be operable therefore they are considered PROCESS RADIATION MONITORs.
- SFP HVAC effluent is discharged to the Plant Vent stack via the Aux Bldg Main Exh fans and the plenum. PRF discharges directly to the Plant Vent Stack, therefore these systems can be considered Radioactive Effluent Release paths.
- R-25A failure is indicated in the stem which impacts (affects) that radioactive effluent release.

Somewhat related to a Simulator Scenario (#2) failure on this exam, but the Scenario has a R-25A **hi alarm** (instead of an instrument power failure), with auto SFP Ventilation isolation defeated, which will cause only **one** train of PRF to auto start initially (the other won't auto start until the SFP ventilation is manually secured). This question, tests an **instrument power failure** with no failure of the SFP ventilation isolation which secures SFP ventilation and the low d/p across the SFP ventilation fans starts **both** PRF trains initially. This question tests the knowledge of how a failure of one R-25 affects the PRF system with no failure of SFP ventilation system to auto secure.

Objective: OPS-40304B02; Relate and Identify the operational characteristics including design features, capacities and protective interlocks for the components associated with the Auxiliary Building Ventilation Systems [...]

52. 074EA1.28 001/NEW/RO/C/A 3.7/3.9/N/N/3/REVISED/FIX

The following plant conditions exist on Unit 1:

- FRP-C.2, Response to Degraded Core Cooling, is in progress.
- All ECCS disconnects have been closed.
- B Train SI could **NOT** be reset, MLB 1 11-1 remains LIT.

The crew is at the step to close all SI accumulator discharge valves.

- The OATC has placed all handswitches for the SI Accumulator Discharge Isolation valves in the CLOSE position.
- The following are the indications available on the MCB:

1B ACCUM DISCH ISO Q1E21MOV8808B		1C ACCUM DISCH ISO Q1E21MOV8808C	
WH-LIT		WH-DRK	
GR-DRK	R-LIT	GR-DRK	R-LIT

LEGEND:

WH = WHITE
R = RED
GR = GREEN

LIT = ILLUMINATED
DRK = NOT ILLUMINATED

Neither of the valves are responding to MCB switch manipulation.

Which one of the following describes the reason MOV-8808B and MOV-8808C did **NOT** close?

	MOV-8808B	MOV-8808C
A.	Supply breaker has tripped open.	B Train SI is NOT reset.
B.✓	B Train SI is NOT reset.	Supply breaker has tripped open.
C.	RCS pressure is too high.	Supply breaker has tripped open.
D.	Supply breaker has tripped open.	RCS pressure is too high.

added: "handswitches for the" SI Accumulator valves per CE suggestion for clarity. Removed 2 bullets that were unnecessaary and added bullet 2 where the crew is at in the procedure. This eliminates the plausibility concern for RCS pressure being too high with SGs already at 100 psig. I cahnged the way the question looks to aid the candidate in recognizing what has occurred and what is occurring at this time so no confusion exists. Also changed the stem to better reflect the question being asked.

A - Incorrect. 1) The white light LIT indicates control power availability to the MOV, and normally would be sufficient to allow for operation; MOV-8808B can not be closed until after the SI signal is also reset, the SI signal "locks out" a close signal to these valves. (K603 relay)

2) MOV-8808C is an A train component, B train SI does not block its closure via the (K603) contact. Furthermore, the Breaker is tripped as indicated by the white light.

Plausible: 1) Many switches on the MCB are equipped with an AMBER light above the position indication to indicate a tripped supply breaker, this light NOT lit indicates "proper" operation.

2) Both trains of SSPS recieve information from single indications within the field -- this concept could be reversed thinking that both trains of SSPS provide input to all 3 components to allow closure (ex: as both trains of 125VDC are required for MFIV opening or SD operation). Also, since plant is equipped with 3 Accumulators train alignment is necessary to differentiate the impact from SI not being reset.

B- Correct. 1) MOV-8808B can not be closed until after the B Train SI signal is **also** reset, the SI signal "locks out" a close signal to these valves. (K603 relay).

2) MOV-8808C has experienced a loss of control power the motor power contacts as indicated by the loss of the White light.

C - Incorrect. 1) Following the SG depressurization, RCS pressure is very likely well below the 2000 psig auto-open signal. Furthermore, this signal does not prevent closure, it would only re-open the valve if were closed ≥2000 psig with an SI signal present.

Plausible: If pressure was >2000 psig and SI was not reset, then upon closure, an auto-open signal would be present; K628 relay (>P-11) and K603 relay (SI) would re-open the valve. If incorrectly applied CETs to RCS temp, and a saturated condition assumed then pressure in the RCS could be presumed to be very high-- not considering the SG cooldown to 100 psig.

2) this part is correct see B#2.

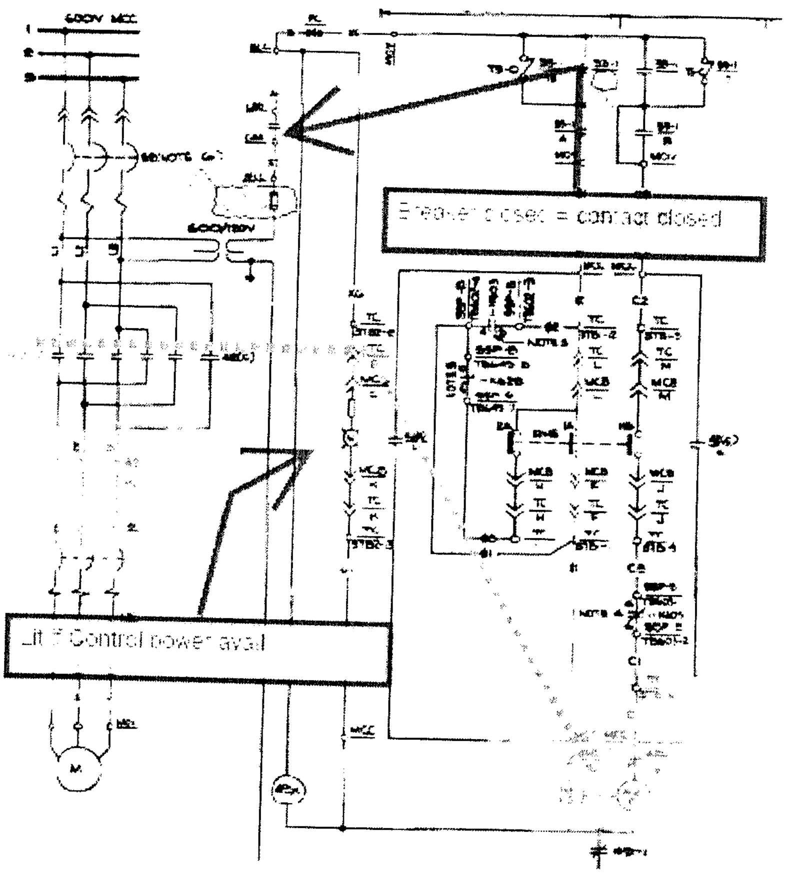
D - Incorrect. 1) See A.

2) equivalent to discussion in C for the wrong train component.

RHS	BLK	CONT	CLOSE	AUTO	OPEN
AUTO CLOSE & OPEN	FRONT	A			X
		B	X		
	2 REAR	A		X	

REMOTE HANDSWITCH

SPRING RETURN TO AUTO



Previous NRC exam history if any: NONE

074EA1.28
074 Inadequate Core Cooling
EA1 Ability to operate and monitor the following as they apply to a Inadequate Core Cooling:
(CFR 41.7 / 45.5 / 45.6)
EA1.28 Core flood tank isolation valve controls and indicators 3.7* 3.9*

Match justification:
- the ability to operate: knowledge of the system interlocks and controls is required to ensure suscessful operation of a component. (the operation is already completed as an initial condition)
- ability to monitor: Given the indications, the examinee must evaluate response to validate completion success.
- AS they apply to Inadequate core cooling: actions required by FRP-C.2 in a "degraded" core cooling situation (which is a condition resulting from inadequate cooling) -- equivical to C.1 actions

Objective: OPS-52533C07
Analyze plant conditions and determine the successful completion of any step in FRP-C.2

53. 075A2.03 001/NEW/RO/C/A 2.5/2.7/N/N/4/CVR/SAT

Unit 1 is at 20% power, and the following conditions occurred:

At 1000:

- Main Condenser pressure is 1.3 psia and degrading.
- AOP-8.0, Partial Loss Of Condenser Vacuum, is in progress due to an air ejector malfunction.

At 1010:

- Main Condenser vacuum has degraded to 12 psia.
- AOP-13.0, Condensate And Feedwater Malfunction, has been entered.

Which one of the following describes the Circulating Water (CW) outlet temperature at 1010 as compared to earlier, and the action(s) required by AOP-13.0 ?

At 1010 CW outlet temperature is __(1)__ than **at 1000**,

and

AOP-13.0 requires __(2)__.

	(1)	(2)
A.	higher	tripping the reactor
B.	lower	reducing power to approximately 2%
C.	higher	reducing power to approximately 2%
D✓	lower	tripping the reactor

A - Incorrect. The first part is incorrect, since the Steam dumps are not able to arm with vacuum worse than 8" Hg Vacuum (10.78 psia), the SGFPs trip at 5.9 psia (12 in Hga), and the Main Turbine trips at 21" Hg (4.41 psia), Plausible, since it would be correct if Steam dumps armed and operated as usual. The second part is correct (see D). Plausible, even when combined with the first part, since the Main Condenser pressure at which the Control interlock C-9 for blocking Steam Dump operation on high pressure (low vacuum) is higher (a lower vacuum) than for a trip of the SGFPs and Main Turbine.

B - Incorrect. The first part is correct (see D). The second part is incorrect, since AOP-13 directs tripping the reactor with a loss of both SGFPs >5%. Plausible, since the choice would be correct if initial power was 5% or less per AOP-13.0 Step 2.2 A/ER "Reduce reactor power to approximately 2%". AOP-13.0 until recently allowed reducing power to the capacity of AFW for both SGFPs tripped from a power level as high as 35% to prevent a reactor trip.

C - Incorrect. The first part is incorrect (see A). The second part is incorrect (see B).

D - Correct. The first part is correct, since no steam is condensing in the Main Condenser from the Main Turbine, SGFPs, or from the Steam dumps. Due to the degraded vacuum, the steam dumps don't arm or operate, the Main Turbine Trips, the SGFPs trip, and the CW outlet temperature approaches the inlet (decreases from it's value when steam was condensing in the condenser). The SG atmospherics and Safeties will open as necessary to maintain SG pressure less than the design, and thus control Tavg. The second part is correct per AOP-13.0 step 2.1: for power above 5%, with a trip of both SGFPs, trip the reactor.

Per FNP-0-SOP-0.3, Version 39.0, APPENDIX G:
8" Hg Vac. Minimum to arm SDs, and 10.78 psia max pressure to arm SDs.

Per ARP-1.10, KK1, TURB COND VAC LO alarm, Version 64.0:
NOTE: IF condenser vacuum decreases to 21" Hg (4.41 psia), THEN a turbine trip occurs.

Per AOP-8.0, Partial Loss Of Condenser Vacuum, Version 24.0:
NOTE:
• Main turbine trip will occur at 4.41 psia (9 in Hga)
• SGFP trip will occur at 5.9 psia (12 in Hga).

Procedures:

AOP-3.0
AOP-3.0, & AOP-13.0
AOP-3.0, AOP-13.0, & EEP-0

FNP-1-AOP-13.0, Condensate And Feedwater Malfunction, Version 29.0
2 A/ER: Check Both SGFPs - TRIPPED
2 RNO: Proceed to step 3 OBSERVE CAUTION prior to step 3.

2.1 A/ER Check Reactor Power - LESS THAN 5%.
2.1 RNO: Trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.

2.2 A/ER Reduce reactor power to approximately 2%.

FNP-1-AOP-3.0, Turbine Trip Below P-9 Setpoint, Version 16.0

FNP-0-SOP-0.3, Version 39.0, APPENDIX G,
8" Hg Vac. Minimum to arm SDs, 10.78 psia max pressure to arm SDs

AOP-8.0, Partial Loss Of Condenser Vacuum, Version 24.0
• Main turbine trip will occur at 4.41 psia (9 in Hga)
• SGFP trip will occur at 5.9 psia (12 in Hga)

Previous NRC exam history if any:

075A2.03

075 Circulating Water System

A2 Ability to (a) predict the impacts of the following malfunctions or operations on the circulating water system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

A2.03 Safety features and relationship between condenser vacuum, turbine trip, and steam dump . . 2.5 2.7*

Match justification: This question presents a loss of vacuum and requires knowledge of how this affects the Circ Water system in this set of conditions: i.e. at this vacuum, the steam dumps are prevented from opening to protect the main condenser from overpressure. This causes CW temp to be affected differently than if they opened after the turbine trip as they normally would. The second part of the question and each choice requires knowledge of the procedure action for this condition (on the RO level).

Objective:

3. **STATE AND EXPLAIN** the operational implications for all Cautions, Notes, and Actions associated with AOP-8.0, Partial Loss of Condenser Vacuum. (OPS-52520H03).
5. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing AOP-8.0, Partial Loss of Condenser Vacuum. (OPS-52520H06).

54. 076A3.02 001/NEW/RO/MEM 3.7/3.7//N/3/CVR/SAT

A Safety Injection and loss of B train Start Up transformer occurred on Unit 2.

Which one of the following is the position of the Turbine Building Service Water Supply Isolation Valves?

Valve nomenclature is listed below:
SW TO TURB BLDG ISO B TRN V514
SW TO TURB BLDG ISO A TRN V515
SW TO TURB BLDG ISO A TRN V516
SW TO TURB BLDG ISO B TRN V517

	V515/V517	V514/V516
A.	Throttled	Throttled
B.	Closed	Throttled
C.	Throttled	Closed
D.✓	Closed	Closed

A - Incorrect. This would be correct for LOSP on both trains with no SI, but in this case there is an SI on both trains and an LOSP on only one train.

B - Incorrect. The first part is correct because of the SI on A train. The second part is incorrect since the SI isolates both trains of valves regardless of the LOSP signal on the B train. This would be correct for an LOSP on B Train with no SI, and an SI on A train with or without an LOSP on A train, but in this case the SI isolates both trains of SW valves.

C - Incorrect. This is the exact opposite of B and may be chosen if confusion existed as to the automatic action of the SW to the TB valves due to the two signals: SI and LOSP.

D - Correct. The SI Closes the valve on both trains, even though the LOSP alone (with no SI) would throttle the valves on the respective train. With an SI and and LOSP, the valves close to ensure sufficient cooling flow to the Emergency DGs.

SW FSD A-181001

3.44 TURBINE BUILDING SERVICE WATER SUPPLY ISOLATION VALVES

TPNS Nos.	Unit 1	Unit 2
Train A -	Q1P16V515	Q2P16V515
	Q1P16V516	Q2P16V516
Train B -	Q1P16V514	Q2P16V514
	Q1P16V517	Q2P16V517

3.44.1 Basic Functions

Redundant Turbine Building Service Water Supply Isolation Valves **automatically isolate the nonessential turbine building service water loads upon receipt of a Phase A Containment Isolation Signal (T-signal)** and/or excess turbine building service water flow rate. This action is required to ensure adequate service water flow to safety-related equipment during accident modes.

The Turbine Building Service Water Supply Valves **provide a second, throttling function during a Loss of Offsite Power event.** Specifically, the valve operators automatically position the valve to 16 degrees in the open direction upon receipt of a LOSP signal. This throttled, or mid-stroke, position serves to provide a limited amount of cooling water to the Turbine Bldg. to support the cooldown of the secondary side of the plant. This action simultaneously serves to automatically provide increased cooling water to the Emergency Diesel Generators during the LOSP event. Plant operator actions are still required to isolate the Turbine Building within fifteen minutes to provide cooling water for long term operation of the diesels.

Previous NRC exam history if any:

076A3.02
076 Service Water System
A3 Ability to monitor automatic operation of the SWS, including: (CFR: 41.7 / 45.5)
A3.02 Emergency heat loads 3.7 3.7

Match justification: Ability to monitor automatic operations of the Service Water system including: emergency heat loads. This question requires knowledge of automatic operation of the SW system in an emergency as it automatically operates to reduce SW to non-vital loads to conserve SW for cooling the emergency DGs by throttling on an LOSP, and isolating the TB SW Supply on an SI.

Objective:

6. **ANALYZE** plant conditions and **DETERMINE** the successful completion of any step in AOP-10.0, Loss of Service Water. (OPS-52520J07).

55. 077AA1.02 001/BANK/RO/MEM 3.8/3.7/N/N/3/NO CHANGE OK/SAT

The following plant conditions exist on Unit 1:

- 100% power.
- All systems are aligned normally.
- Generator reactive load is currently at "0" MVARs.
- ACC has notified the plant that system voltage problems require UNIT 1 to establish maximum allowable incoming reactive load (MVARS in).

Which one of the following:

1) identifies the administrative limit on incoming reactive load (MVARS in) IAW UOP-3.1, Power Operation,

and

2) the proper switch which will be used to establish maximum allowable incoming reactive load?

	(1)	(2)
A.	-200 MVARs	Manual Voltage Adjust Switch
B.	- 200 MVARs	Auto Voltage Adjust Switch
C.	-300 MVARs	Manual Voltage Adjust Switch
D.✓	-300 MVARs	Auto Voltage Adjust Switch

A - Incorrect. 1) -200 MVARS is the limit of the MCB MVAR meter; UOP-3.1 step 3.3.3 states that -300 MVAR is the administrative limit. Also stated in SOP-36.8 4.8.2.2.
Plausible: This is a limit of the MCB meter, and could be perceived to be the administrative limit for operation.

2) Operation of the Manual voltage adjust is incorrect per SOP-28.1, see caution before 4.7.14 and 4.23.1; operating the Manual voltage adjust while in auto changes the base for Auto and if it were to auto shift to manual, the transient could result in generator damage/rx trip.
Plausible: A manual adjustment is being made, and the Manual Voltage Adjustment Switch is manipulated if the voltage regulator were in TEST or OFF; this switch is also manipulated to ZERO VM4098.

B - Incorrect. 1) See A for discussion and plausibility.
2) this is the correct switch to manipulate.

C - Incorrect. 1) this is the correct Administrative limit per UOP-3.1 (ver 101) 3.3.3
2) See A #2 for discussion and plausibility.

D - Correct. 1) UOP-3.1, SOP-36.8 both establish this limit for MVARs to prevent the auto adjuster from going to its mechanical stop.
2) This is the correct switch manipulation per SOP-28.1 for adjusting MVARS.

UOP-3.1, Version 104.0
SOP-36.8, Version 14.0

Previous NRC exam history if any:
Sequoyah 2009 question 17 (RO NRC EXAM)

077AA1.02
077 Generator Voltage and Electric Grid Disturbances
AA1. Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.5 and 41.10 / 45.5, 45.7, and 45.8)
AA1.02 Turbine / generator controls.. 3.8 3.7

Match justification:

- Grid disturbance: identified in stem, causing the initiating queues for the operator to "OPERATE and/or MONITOR Turbine/generator controls--- Grid disturbances result in Alabama Control Center (ACC) or Power Coordination Center (PCC) to request FNP to accept max vars in.
- The examinee must know limitations to the operation of the voltage regulator and select the correct switch to manipulate the Generator output.

Objective:
OPS-52105C05; Recall and discuss the P&Ls, Notes, and Cautions found in SOP-28.1.

OPS-40202B16; State and explain how the [...] main generator parameters are controlled, including thier limits, and the adverse effects on the main generator of excessive VARS OUT or VARS IN.

56. 078K2.02 001/NEW/RO/MEM 3.3/3.5/N/N/2/CVR/SAT

Which one of the following correctly states the power supplies to the 1A and 1B Emergency Air Compressors?

A.✓ 1A and 1B 600V MCC

B. 1A and 1B 600V LCC

C. 1D and 1E 600V LCC

D. 1U and 1V 600V MCC

A - Correct. Per the load list A-506250, Rev. 12, Pages F – 94 & G – 78.

B - Incorrect. Plausible, since the voltage is correct and these are safety related switchgears.

C - Incorrect. Plausible, since the voltage is correct and these are safety related switchgears.

D - Incorrect. Plausible, since the voltage and type of switchgear is correct: safety related 600V Motor Control Center (MCC).

LOAD LIST, A-506250, Rev. 12
Pages **F – 94 & G - 78**

Previous NRC exam history if any:

078K2.02
078 Instrument Air System
K2 Knowledge of bus power supplies to the following: (CFR: 41.7)
K2.02 Emergency air compressor 3.3* 3.5*

Match justification: FNP has 2 Emergency Air Compressors which Air to operate SG Atmospheric Relief valves if Instrument air is unavailable. This question requires knowledge of the bus power supplies for the two Emergency air compressors.

Objective:

1 **NAME AND IDENTIFY** the Bus power supplies, for those electrical components associated with the Compressed Air System, to include those items in Table 1- Power Supplies (OPS-40204D04).

57. 103A4.03 001/MOD/RO/MEM 2.7/2.7/N/N/3/HBF/SAT

The following plant conditions exist on Unit 1:

- A LOCA has occurred.
- Containment Pressure is 30 psig and decreasing.
- All required actuations have occurred.

Which one of the following describes the **MINIMUM conditions** if any, **AND** actions required to reset **'B' train** PHASE B CTMT ISO (MLB-3 6-1)?

A. 1) Containment Pressure must be lowered to less than the HI-3 setpoint prior to reset.

2) BOTH Train A and B CS RESET pushbuttons, and BOTH Train A and B PHASE B CTMT ISO RESET pushbuttons must be depressed.

B. 1) Containment Pressure must be lowered to less than the HI-3 setpoint prior to reset.

2) The B Train PHASE B CTMT ISO RESET pushbutton ONLY must be depressed.

C✓ 1) Phase B can be reset regardless of Containment Pressure.

2) The B Train PHASE B CTMT ISO RESET pushbutton ONLY must be depressed.

D. 1) Phase B can be reset regardless of Containment Pressure.

2) BOTH Train A and B CS RESET pushbuttons, and BOTH Train A and B PHASE B CTMT ISO RESET pushbuttons must be depressed.

A Incorrect. 1) Phase B is equipped with a memory retentive latching relay when actuated on HIGH-3 signal. This latching relay allows for a reset of the signal before clearing the initiating signal.
2) Containment Isolation Phase B signal does not require Cnmt Spray signal to be RESET. These signals, because of the memory retentative relays are mutually exclusive of one another for RESET.
plausibility: 1) SI, Phase A, and Phase B components can not be repositioned without first clearing the originating signal; as is true for the TDAFW pump UV & LO-Level auto-start signals.
2) CNMT Spray actuation requires the operation of two switches per train; Also, the Phase B and Cnmt Spray actuation signals are actuated by the same signal and it is feasible that one might believe that one must be reset before the other.

B Incorrect. 1) See A part 1 for discussion and plausibility
2) See C part 2 for discussion.

C Correct. 1) The latching relays allow resetting the Phase B signal without clearing the actuating condition. See A part 1 for discussion.
2) This is correct. Depressing a single RESET pushbutton for EACH train will reset the PHASE B containment isolation signals, AND the B train pushbutton resets the B train signal.

D Incorrect. 1) See C part 1 2) See A part 2.

Previous NRC exam history if any:

103A4.03
103 Containment System
A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
A4.03 ESF slave relays . . . .
2.7* 2.7*

Match justification:
Operation of ESF slave relays -- resetting Phase B signal by depressing the RESET pushbuttons on the MCB, Depressing these pushbuttons resets the sealed in signal on the CNMT ISOL valves allowing them to be repositioned to support recovery actions.

Objective: 52201I07: Recall and describe the operation and function of the following [...] ESF to include setpoint, [...] and reset features [...].

58. 103K3.03 001/NEW/RO/C/A 3.7/4.1/N/N/3/CVR/SAT

Refueling on Unit 2 is in progress, and the following conditions exist:

- The Containment Equipment Hatch is open and hoses and electrical cables are routed through the Hatch.
- Both Main Personnel Airlock doors are open.
- The Inner Airlock door is inoperable and cannot be closed.
- CTMT Main Purge is in operation.
- R-24A and R-24B are discovered to be inoperable.

Which one of the following describes whether or not Fuel movement may continue in Containment, and the reason?

A.✓ CORE ALTERATIONS must be stopped immediately, because R-24A and R-24B are inoperable.

B. CORE ALTERATIONS must be stopped immediately, because the Inner Airlock door cannot be closed.

C. CORE ALTERATIONS must be stopped immediately, because the Containment Equipment Hatch is open.

D. CORE ALTERATIONS may continue in the current condition, because all penetrations are capable of being isolated with manual actions.

A- Correct. Either the dampers must be closed or the auto isolation feature must be operable to allow CORE ALTERATIONS. R-24A & R-24B provide automatic isolation on high radiation sensed in the CTMT purge system.

B - Incorrect. Only one of the two airlock doors must be capable of closing. Plausible, since at least one airlock door must be capable of closing, but in this case, the outer door is still capable of being closed.

C - Incorrect. The equipment hatch may be open as long as it is capable of being closed and held in place with 4 bolts in two hours. The hoses and cables give added plausiblity to this distractor, since they would prolong the time that it would take to close the hatch, but they are allowed to be routed through the hatch as long as they have quick disconnects, isolation valves, and blue ownership tags to facilitate closing the hatch in two hours or less if needed for containment closure per STP-18.4, Containment Mid-Loop And/Or Refueling Integrity Verification And Containment Closure, Step 5.2.4 Version 33 and UOP-4.1, Controlling Procedure For Refueling, version 51.

D - Incorrect. Plausible, since it is correct for two of the three penetrations listed (Equipment hatch and Personnel Hatch), but the Main purge has an additional requirement of automatic isolation capability per TS 3.9.3, Containment Penetrations, during refueling. R-24A & R-24B being inoperable defeats the required automatic isolation feature. Until the Purge dampers are closed, core alts must stop. Also plausible, since if the dampers were manually closed, the fuel movement could continue, but the choice states "in the current condition" which includes Main purge in operation per the stem. Fuel movement cannot continue with Main purge in operation, but could the dampers were manually closed per TS 3.3.6, Condition C.

TSs

3.3.6 Containment Purge and Exhaust Isolation Instrumentation, Amendment No. 146 (Unit 1) & Amendment No. 137 (Unit 2)

3.9.3 Containment Penetrations, Amendment No. 178 (Unit 1) & Amendment No. 171 (Unit 2)

Previous NRC exam history if any:

103K3 03

103 Containment System

K3 Knowledge of the effect that a loss or malfunction of the containment system will have on the following:
(CFR: 41.7 / 45.6)

K3.03 Loss of containment integrity under refueling operations. 3.7 4.1

Match justification: This question provides conditions which must be recognized as either a loss of Refueling integrity or allowed for refueling integrity. There is some normal conditions allowed by refueling integrity (but not by Containment integrity in modes 1-4) and one condition that must be recognized as a loss of refueling integrity in Mode 6. Knowledge is required that an automatic iso ckt must be operable for ctmt purge valves OR they must be closed for refueling integrity. The effect of the conditions which must be known to answer this question is that Core alterations are prohibited.

Objective:

1 **RECALL AND APPLY** the LCO and APPLICABILITY for Technical Specifications (TS) or TRM requirements, and the REQUIRED ACTIONS for 1 HR or less TS or TRM requirements, and the relevant portions of BASES that DEFINE the OPERABILITY and APPLICABILITY of the LCO associated with the Containment Structure and Isolation System components and attendant equipment alignment, to include the following (OPS-52102A01):

- 1.6 Containment Integrity – Definition

3.6.1 Containment

3.6.2 Containment Air Locks

3.6.3 Containment Isolation Valves

3.6.4 Containment 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).

59. G2.1.17 001/NEW/RO/MEM 3.9/4.0/N/N/2/REWRITE/FIX

Which one of the following demonstrates **UNACCEPTABLE** communication IAW ACP-1.0, Plant Communications?

A. During a ramp per UOP-3.1, Power Operation;

the SS speaking to the UO: "Jim, raise turbine load by 10 megawatts at 1 megawatt per minute."

UO: "Raise turbine load by 10 megawatts at 1 megawatt per minute".

SS: "That's correct."

B. While swapping on service CCW heat exchangers per SOP-23.0, Component Cooling Water System;

the UO speaking to the Rover on the phone: "Check one bravo CCW heat exchanger outlet isolation valve, Q1P17V008 Bravo, closed."

Rover: "Check one bravo CCW heat exchanger outlet isolation valve, Q1P17V008 Bravo, closed."

UO: "That's correct."

C. In preparation for a Waste Gas Release per SOP-51.1, Waste Gas System Gas Decay Tank Release;

the Shift Radio Chemist speaking to the Chemistry Technician: "Joe, Sample Unit 1 Waste Gas Decay Tank #2 for release."

Chemistry Technician: "Sample Unit 1 Waste Gas Decay Tank #2."

Shift Radio Chemist: "Yes, sample #2 Waste Gas Decay Tank on Unit 1."

D✓ While performing EEP-0.0, Reactor Trip and Safety Injection;

the SS speaking to the OATC: "Check Containment Pressure has remained less than 27 psig."

OATC: "Yes, Containment Pressure is 1.6 psig."

Rewrote in the NOT question format due to difficulty in finding 3 plausible but incorrect choices if the question asks "which is correct?" per CE suggestion.

A - Incorrect. This is correct communication per ACP-1.0, Appendix 1, example 1. Changed OATC to SS for this choice since it would be highly unlikely that the direction to ramp Up and add positive reactivity would come from anyone other than the SS. Plausible, because the UO is not directing the repeat back, but is not required to, even though the SS must direct the initial communication when other people are near by. Also,

B - Incorrect. This is correct communications per ACP-1.0. The phonetics are normally required for all letters, except for in a TPNS and some other exceptions listed in

STEP 5.1.3 & & APPENDIX 1 of ACP-1.0. The example given is correct in that it does not require a directed communication since no one else is present, it has a unit designator for the "1"B CCW valve (and a "1" in the TPNS), and the B on the end of the TPNS has a phonetic pronunciation (Bravo). That is what is required by ACP-1.0 Appendix 1 page 5 of 5. Plausible to think this is incorrect since not all the TPNS letters are spelled out phonetically.

C - Incorrect. . This is correct communications per ACP-1.0 as described in APP 1, example 2. Plausible that the applicant may think this is unacceptable due to thinking there are different requirements outside of the Operations department for communications, or that this does not meet the standard for 3 way communication. Also, it is a paraphrase repeat back which meets the standard in ACP-1. The applicant may think it should be a more verbatim repeat back, however, the repeat back contained all the key elements of the task that is being directed to be performed.

D - Correct. This is an unacceptable communication technique due to a 3 way communication being required for the transmission of operational data. This may be mistaken for informational communication per ACP-1, Step 3.5, which would not require a 3 way, but it is the transmission of technical data with which the Shift Supervisor will make a procedure decision. Per ACP-1.0 step 5.1, this requires a 3 way communication.

ACP-1.0, Version 5.0

3.5 **Informational Communications** – Personnel holding informational briefings, summaries, announcements or status discussions. Closed loop communications are not generally utilized. Personnel should refrain from giving operational orders during this type of communications but may elect to close the loop on selected portions.

5.0 General Communications Guidelines

5.1 ...**Three-way communication is the expected standard** for communications involving directions, operations **or transmission of technical data**. Refer to Appendix 1 of this procedure for examples of effective Three-way communication techniques.

5.1 ...The following communication techniques are provided to ensure information is transmitted and received effectively.

5.1.3 Be specific when identifying equipment. Identify equipment by using noun name, TPNS designation, or approved abbreviation such as RWST for the refueling water storage tank. Approved abbreviations are found in FNP-0-AP-25, EQUIPMENT IDENTIFICATION. **Specify if it is Unit 1, Unit 2 or shared equipment**. (NOTE: If the unit designation is included in the equipment name, i.e. 2A charging pump, that is acceptable.) Being specific ensures that the information is detailed enough so that the correct component is identified. **When a long string of alpha-numeric characters is being spoken such as a TPNS number it is acceptable to only use the phonetic alphabet for the last letter in the sequence.** See Appendix 1 examples for clarification.

5.1.4 Ensure the intended individual receives the message. **When more than two people involved in the same task are in the immediate area then use the name or title of the intended receiver** prior to the message or instruction.

5.1.6 As the receiver of an instructional communication, the individual should acknowledge receipt of the communication by providing feedback in the form of repeatback or paraphrase. Verbatim repeatbacks are not required, but may be useful in some activities.

APPENDIX 1, Page 1 of 5 Version 5.0
COMMUNICATIONS EXAMPLES
Face-to-face Communications in the Control Room

EXAMPLE 1
Inter-operator Communications to Effect a Turbine Load Increase:

Reactor Operator (RO) to Unit Operator (UO):
"Jim, raise turbine load by 10 megawatts at 1 megawatt per minute."

Response by UO:
"Raise load 10 megawatts at 1 megawatt per minute."

Acknowledgement by RO:
"That's correct."

EXAMPLE 2
Communications Between the Shift Radio Chemist and Chemistry Technician concerning sampling need:

Shift Radio Chemist to Chemistry Technician:
"Joe, Unit 1 Waste Gas Decay Tank #2 needs to be sampled for release."

Response by Chemistry Technician:
"Sample Unit 1 Waste Gas Decay Tank #2."

Acknowledgement by Shift Radio Chemist:
"Yes, sample #2 Waste Gas Decay Tank on Unit 1."

ACP-1.0, APPENDIX 1, Page 5 of 5, Version 5.0
These are examples of how the phonetic alaphabet should be used:
Stating a TPNS designation:
Written: Q1E21V009C
Spoken: Q1E21V009 Charlie

Previous NRC exam history if any:

G2.1.17

G2.1.17 **Ability to make accurate, clear, and concise verbal reports.** (CFR: 41.10 / 45.12 / 45.13) RO 3.9 SRO 4.0

Match justification: This question requires knowledge of the techniques which the applicant is required to utilize in verbal communications which ensure accurate, clear, and concise verbal reports. ACP-1.0, an FNP Administrative Control Procedure, has a section titled: 5.1 "The following communication techniques are provided to ensure information is transmitted and received effectively". This question requires knowledge of these ACP listed techniques.

Objective:

1. Explain the importance of maintaining professional communications when using plant communications equipment (OPS40502C01).
2. Outline management's expectations for communications (OPS40502C02).
3. Explain the purpose of and the method for conducting three-way communications (OPS40502C03).

60. G2.1.45 001/NEW/RO/MEM 4.3/4.3/N/N/4/HBF/SAT

The following plant conditions exist on Unit 1:

AT 1000:

- N-41, N-42, N-43, and N-44, PR Nuclear Power, indicate 100% power.
- Main Generator Load is 901 MW.
- All SG steam flows are 4.1 X 10^6 lbs/hr.

AT 1010:

The crew identifies PK-3371A, 1A SG Atmospheric Relief Valve Controller, failed to 100% demand.

- The UO places PK-3371A in manual and lowers the demand to 0%.
- The crew suspects the Atmospheric Relief Valve has not closed.

Which one of the following sets of stable plant parameters indicates that PCV-3371A has **remained OPEN**?

		Stm Flow **(X 10^6 lbs/hr)**		
	MW	1A SG FI-474	1B SG FI-484	1C SG FI-494
A.	840 MW	4.970	4.100	4.100
B.	840 MW	4.404	4.398	4.398
C.	880 MW	4.500	4.100	4.100
D✓	880 MW	4.254	4.248	4.248

Knowledge:

an Atmospheric Relief has a design capacity of 3% total steam flow.
Distractor uses Safety valve design capacity of 7.6% total steam flow.

Stm flow indication: 1A SG will demonstrate a minutely higher steam flow than the other 2 SGs only because of the head loss which occurs in the Cross-over piping (42" line in MSVR).

INITIAL Steam flow @100% =12.3 Mlbh]/ 3) =4.1 X 10^3 lbs/hr

- 3% steam flow = 12.3 Mlbh/100%*3%= .369 Mlbh
- 7% steam flow = 12.3 Mlbh/100%*7%= .861 Mlbh

Fundamental: Pascal's LAW states that the pressure exerted within a system is felt equally and undeminished throughout that system. Pascal's Law is designated for a closed system, but the fundamental is relatively consistent in an open system when conservation of Mass Flow is maintained.

A - Incorrect. This is the MW loading expected after a steam leak of 7%. An Atmospheric relief capacity is only 3%. Additionally, since the MSIVs are open, the steam flow will be balanced between the three SGs, with only a minor difference in A SG steam flow due to headloss.

$$\left(\frac{7\%}{1085psig}\right)\cdot 750psig = 4.84\%$$ resulting in a MW load of: $$\left(\frac{901MW}{100\%}\right)\cdot 95.16\% = 857MW$$

ADDED a 20 MW loss to increase desparity between answers.

B - Incorrect. This is the MW loading and steam flows expected after a steam leak of ~7%.

C - Incorrect. Although the MW load is correct, the steam flow would be shared by all SGs since cross-connected.

D - Correct. the design capacity of an Atmospheric relief valve is 3% total steam flow at 1035 psig; Since 100% power Steam pressure is 750 psig, then the Impact from a failed open ARV will be

$$\left(\frac{3\%}{1035psig}\right)\cdot 750psig = 2.17\%$$ resulting in MW reduction of: $$\left(\frac{901MW}{100\%}\right)\cdot 97.73\% = 881MW$$

Validated on Laptop Sim 9/23/09 with IC-73 -- actual MW load is 880 MW and steam flow 1A = 4.2, 1B=4.2, 1C=4.1 (smoothed homepage view) instrument view has swing variance but avg around 4.25 on all instrumentation)

The Steam flow will be shared by all Steam generators: With only a minor variation noted in the 1A SG steam flow noticable.

Previous NRC exam history if any:
N/A

G2.1.45
2.1.45 **Ability to identify and interpret diverse indications to validate the response of another indication.** (CFR: 41.7 / 43.5 / 45.4) RO 4.3 SRO 4.3

Match justification:

- PC-3371A/B/C from the MCB requires evaluation of diverse indications to validate success since there are no Position indicators on the MCB.
- the indication that must be validated with diverse indications is the MA station 0% demand.
- The indications that must be interpreted are Steam flow and turbine load.

Objective:

OPS-40201A02; relate and identify the operational characteristics including design features, capacities, and protective interlocks for the components associated with the Main and Reheat Steam System [...].

OPS-52521O07; Analyze plant conditions and Determine the successful completion of any step in AOP-14 [...].

61. G2.1.9 001/NEW/RO/MEM 2.9/4.5/N/N/2/EDITORIAL/SAT

Which one of the following lists **ONLY** those personnel in which BOTH are required to request permission to enter the Control Room **At-the-Controls Area** from the OATC IAW NMP-OS-007-001, Conduct of Operations Standards and Expectations?

A. NRC Inspectors, Plant Manager

B. Operations Superintendents, Site VP

C. Operations Manager, Chemistry Foreman

D✓ Reactor Engineer, Health Physics Foreman

A - Incorrect. NRC inspector and Plant Manager are incorrect per NMP-OS-007-001, step 6.11.2.1. Plausible, since not all of management or NRC/INPO observers are exempt from getting permission, but the resident NRC inspector and the Plant Manager are both exempt.

B - Incorrect. Operations Superintendents & Site VP are incorrect per NMP-OS-007-001, step 6.11.2.1. Plausible, since not all of management are exempt from getting permission, but the Operations Superintendents & Site VP are both exempt.

C - Incorrect. Operations Manager is incorrect per NMP-OS-007-001, step 6.11.2.1. Chemistry Foreman is correct since the Chemistry Foreman is NOT exempt and must get permission prior to entry.

D - Correct. Correct per NMP-OS-007-001, step 6.11.2.1.

NMP-OS-007-001, Version 5.0

6.11.2.1 Access Protocol

Personnel who are exempt from requesting permission to enter the ATCA include:

- Site VP
- Plant Manager
- Operations Manager and Operations Superintendents
- On-duty shift operating crew, including the STA
- NRC Inspectors

All others wishing to enter the ATCA **must obtain permission from a licensed operator on shift**

Previous NRC exam history if any:

G2.1.9

2.1.9 **Ability to direct personnel activities inside the control room.** (CFR: 41.10 / 45.5 / 45.12 / 45.13) RO 2.9* SRO 4.5

Match justification: To answer this question correctly, knowledge is required of who needs to obtain permission to enter the control room At the Controls Area and who does not. The control room staff must know this to control access to minimize distractions and properly direct activities inside the control room per NMP-OS-007-001 Step 6.11.1: "Access to the main control room is managed so operators are not distracted from properly monitoring plant parameters."

Objective:

6. Describe Management's expectations for Control Room Formality (OPS40502C06).

7. Describe the "at the controls area," and explain the controls associated with accessing this area (OPS40502C07).

62. G2.2.3 001/FNP BANK/RO/MEM 3.8/3.9/N/N/2/CVR/SAT

Both Units are operating at 22% power with the following conditions:

- Both units 4160V Busses A, B and C are powered from their respective Startup Transformers.

AT 1000 the following occurs:
Due to Severe Weather, the 1B Startup Transformer and 2A Startup Transformer became de-energized.

Which one of the following states **ALL** of the Reactor Coolant Pumps (RCPs) which will still be **running** after the event?

	Unit 1	Unit 2
A.✓	1A RCP	2A RCP
B.	1A RCP	2B and 2C RCPs
C.	1B and 1C RCPs	2A RCP
D.	1B and 1C RCPs	2B and 2C RCPs

A - Correct. **2A RCP and 1A RCP.**
Since 2A and 1B SU xformer trips this means 2B and 2C and 1B and 1C RCPs will be tripped. Therefore the 2A and 1A RCPs will be running.

B - Incorrect. Plausible, since this would be correct if the unit 2 SU XFMR power configuration was the same as for Unit 1.

C - Incorrect. Plausible, since this would be correct if the unit 1 SU XFMR power configuration was the same as for Unit 2.

D - Incorrect. Plausible, since this the RCPs that lose power, not which ones are still running after the others lose power. This is for the 2B SU and the 1A SU tripping.

Unit 2 - 2A SU Transformer supplies power to the 2B RCP on the 2B 4160V Bus and the 2C RCP on the 2C 4160v bus.
Unit 1 - 1B Startup Transformer supplies power to the 1B RCP on the 2B 4160v bus and 1C RCP on the 2C 4160v bus.
This is a difference between the units in that U-2 has a different SU xformer supplies to the RCP busses than U-1.

Unit 1:

S/U XFMR	4160V Bus
1A	1A
1B	**1B**
1B	**1C**
1A	1D
1B	**1E**

Unit 2:

S/U XFMR	4160V Bus
2B	2A
2A	**2B**
2A	**2C**
2A	**2D**
2B	2E

Previous NRC exam history if any:

G2.2.3

2.2.3 (multi-unit license) **Knowledge of the design, procedural, and operational differences between units.** (CFR: 41.5 / 41.6 / 41.7 / 41.10 / 45.12) RO 3.8 SRO 3.9

Match justification: This is a difference in power supplies to the RCPs on each unit and tests the knowledge of those differences.

Objective:

1 **NAME AND IDENTIFY** the Bus power supplies, for those electrical components associated with the Intermediate and Low Voltage AC Distribution System, to include the following (OPS-40102B04):

- 4160V AC Buses
- 600V Load Control Centers
- 600\480\208V Motor Control Centers

63. G2.2.36 001/NEW/RO/C/A 3.1/4.2/N/N/3/EDITORIAL/SAT

Unit 1 is in Mode 6. Fuel Movement inside containment is in progress, and the following conditions occurred:

At 1000:

- The 1B DG is tagged out for Maintenance.
- The 1A RHR pump is in operation.
- The 1B RHR pump is in standby.

At 1005:

- DG15, 1B SU XFMR to 1G 4160V BUS, tripped open.

Which one of the following correctly states whether or not the Tech Specs listed below are met?

- 3.8.2 AC Sources—Shutdown
- 3.9.4 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

	TS 3.8.2	TS 3.9.4
A.✓	is met	is met
B.	is met	is **NOT** met
C.	is **NOT** met	is met
D.	is **NOT** met	is **NOT** met

A - Correct. Only one offsite transmission line is required in Mode 6 during refueling, so with the 1A SU XFMR still operable, the TS 3.8.2 is met. Only one train of RHR is required to be operable and in operation at this refueling water level, so with the 1A RHR still operating and in operation, the TS 3.9.4 is met.

B - Incorrect. The first part is correct (see A). The second part is incorrect, but plausible, since this choice would be correct in modes 1-3, or in this mode with a lower refueling cavity water level (per TSs 3.5.2 & 3.9.5), in which two RHR Pumps are required.

C - Incorrect. The first part is incorrect. Plausible, since in modes 1-4 it would be correct per TS 3.8.1. The second choice is correct (see A).

D - Incorrect. The first choice is incorrect (see C). The second choice is incorrect (see B).

Previous NRC exam history if any:

G2.2.36

2.2.36 **Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.**
(CFR: 41.10 / 43.2 / 45.13) RO 3.1 SRO 4.2

Match justification: The maintenance activity is a DG Tagged out, and it causes a degraded power source condition due to less redundancy. A scenario is provided which changes the number of available trains of decay heat removal (RHR) pumps, and the effect on limiting conditions for operations must be determined.

Objective:

1 **RECALL AND APPLY** the LCO and APPLICABILITY for Technical Specifications (TS) or TRM requirements, and the REQUIRED ACTIONS for 1 HR or less TS or TRM requirements, and the relevant portions of BASES that DEFINE the OPERABILITY and APPLICABILITY of the LCO associated with the Residual Heat Removal System components and attendant equipment alignment, to include the following (OPS-52101K01):

- 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
- 3.4.12, Low Temperature Overpressure Protection (LTOP) System
- 3.4.14, RCS Pressure Isolation Valve (PIV) Leakage
- 3.5.2, ECCS – Operating
- 3.5.3, ECCS – Shutdown
- 3.9.4, Residual Heat Removal (RHR) and Coolant Circulation - High Water Level
- 3.9.5, Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level
- 13.5.1, Emergency Core Cooling System (ECCS)

64. G2.3.13 001/FNP BANK/RO/C/A 3.4/3.8/Y 2008/N/3/CVR/SAT

Unit 1 is at 100% power, and the following conditions occurred:

- Containment mini-purge supply and exhaust fans are running.

R-11, CTMT ATMOS, has come into alarm. It is reading 8000 cpm.

The following radiation monitors are trending up:

- R-12, CTMT GAS
- R-2, CTMT 155 FT
- R-7, SEAL TABLE

Which one of the following are the actions that the OATC is required to take for this condition IAW annunciator response procedure FH1, RMS HI-RAD?

A.✓ • Check pressurizer level and VCT level stable.

• Secure containment mini-purge fans.

B. • Ensure ALL containment mini-purge dampers have automatically closed.

• Secure containment mini-purge fans.

C. • Check pressurizer level and VCT level stable.

• Verify ARDA has auto started.

D. • Ensure ALL containment mini-purge dampers have automatically closed.

• Verify ARDA has auto started.

A - Correct. First part is correct since the operator actions for all rad monitors coming into alarms states: IF RCS leakage is possible then perform actions of FNP-1-AOP-1.0, RCS LEAKAGE, per step 3.5.

Second part is correct since the actions of FH1 say to IF high activity in containment is possible, THEN consider securing containment purge / minipurge (refer to FNP-1-SOP-12.2 CONTAINMENT PURGE AND PREACCESS FILTRATION SYSTEM. It also says: Perform actions of AOP-1.0 (which secures purge), per step 4.11.

B - Incorrect. First part is not correct since the containment mini-purge dampers do not close on R-11 hi rad but do close on high radiation from R-24 which monitors ctmt atmosphere when the minipurge system is running.

Second part is correct - see A above.

C - Incorrect. First part is correct- see A above.

Second part is not correct since ARDA does not start on an R-11 signal but does auto start on R-29, 15C, 60A,B,C,D and R-14, 21, 22.

D - Incorrect. First part is not correct (see B). Second part is not correct - see C above

ARP-1.6 Ver. 64 FH1 and FH4

FH1 has the operator:
2. Insure that any automatic actions, associated with the alarmed channel, have occurred.

For R-11 actions it says:
IF high activity in containment is possible, THEN consider securing containment purge / minipurge. AOP-1 will also have this fan secured when it is entered by procedural guidance and due to entry conditions with all the above rad monitors in alarm.
Plausible since auto actions of some rad monitor does cause these auto actions to occur, just not these.

Previous NRC exam history if any: 2008 NRC Exam, this is the only question in the exam bank that meets the k/a and all NUREG 1021 Rev. 9 Supp. 1 standards.

G2.3.13

2.3.13 **Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.** (CFR: 41.12 / 43.4 / 45.9 / 45.10) RO 3.4 SRO 3.8

Match justification: This question asks for actions to be done by the control room operators and these actions are guided by procedure. Automatic actions of all the rad monitors are common misconceptions, and actions to take are found in the ARP. This ARP has guidance that is both generic in nature and specific to this one radiation monitor. The ARP directs actions and sends to the RCS leak AOP which directs more actions.

Objective:

2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the components associated with the Radiation Monitoring System to include those items in Table 4- Remote and Local Indications and Controls (OPS-40305A02).

5. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Radiation Monitoring System components and equipment, to include the following (OPS-40305A07):
   - Normal control methods
   - Abnormal and Emergency Control Methods
   - Automatic actuation
   - Protective isolations
   - Protective interlocks
   - Actions needed to mitigate the consequence of the abnormality

65. G2.3.15 001/NEW/RO/MEM 2.9/3.1/N/N/3/GTO/CVR/SAT

A Plant Operator is assigned to use a portable RAM 100 frisker during an emergency entry.

Which one of the following describes the:

1) radiation that the frisker detects

and

2) the required checks prior to use IAW RCP-208, Operation and Calibration of MGP Instruments RAM 100 Count Rate Meter?

A.✓ 1) Beta-gamma ONLY.

2) Ensure the daily response check is current and conduct a battery check.

B. 1) Beta ONLY.

2) Ensure the instrument responds properly to a known reference source and calibrate the instrument.

C. 1) Beta-gamma ONLY.

2) Ensure the instrument responds properly to a known reference source and calibrate the instrument.

D. 1) Beta ONLY.

2) Ensure the daily response check is current and conduct a battery check.

A - Correct. Per RCP-208, Step 5.0, 5.4, & 5.5. (See below)

B - Incorrect. Beta is incorrect, but plausible, since this is memory level and confusion may exist as to which type of radiation is detected. The second part is incorrect due to the calibration not being required prior to every use. Plausible, since a calibration check is required prior to every use, but not a calibration. "Ensure the instrument responds properly to a known reference source" is correct.

C - Incorrect. The first part is correct (see A). The second part is incorrect (see B).

D - Incorrect. The first part is incorrect (see B). The second part is correct (see A).

FNP-0-RCP-208, OPERATION AND CALIBRATION OF MGP INSTRUMENTS RAM 100 RATE METER, version 5.0.

5.0 Operation and Response Check

The instrument must be response checked daily or prior to use whichever is less frequent.

5 Ensure the instrument calibration is current as indicated by the calibration sticker.

5.4 The probe will detect a beta-gamma field in CPM.

5.5 Ensure that the instrument responds properly to a known reference source.

Previous NRC exam history if any:

G2.3.15

2.3.15 **Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.**
(CFR: 41.12 / 43.4 / 45.9) RO 2.9 SRO 3.1

Match justification: This question requires knowledge of a portable survey instrument that a licensed operator may use for personnel monitoring in an emergency.

Objective: OPS-30401A G2.3.5 and 2.3.15 objectives

66. G2.3.4 001/NEW/RO/MEM 3.2/3.7/N/N/3/REVISED/FIX

The Unit 1 Radside System Operator is required to enter the RCS Filter cubicle to operate a valve with a broken reach rod and hang a clearance. The Radside System Operator's year-to-date TEDE is 1.5 Rem.

- The RCS Filter cubicle dose rate is 1 R/hr.
- It will take 40 minutes to perform this job.

Which one of the following states the 10CFR20 TEDE radiation exposure limits, and whether or not the worker will exceed the FNP Admin Limits per FNP-M-001, Health Physics Manual?

The 10CFR20 TEDE exposure limit is __(1)__ Rem/year.

This worker __(2)__ exceed the FNP Admin Limits in FNP-M-001.

	(1)	(2)
A.	10 Rem	will
B.	10 Rem	will NOT
C.✓	5 Rem	will
D.	5 Rem	will NOT

A - Incorrect. 1) Incorrect value for Total Effective Dose Equivalent.
2) See C.2; Plausible: The value stated is that of the EIP-14 limit for emergency dose requirements for protecting plant equipment.

B - Incorrect. 1) See A.1 2) since the cumulative TEDE recieved would exceed 2R/yr, the Administrative Exposure limit would be exceeded. Plausible: If one assumes the admin limit is to be in excess of the 2 R /yr. Which would coincide with mistaking the TODE limit of 20R/yr.

C - Correct. 1) This is the TEDE annual limit of 10CFR20.
2) The total TEDE received by this worker will be in excess of 2R/yr therefore, the Administrative Exposure limit would be exceeded.
Calculation:

$$\left(\frac{40\,\text{min}}{}\right)\left(\frac{1hr}{60\,\text{min}}\right)\left(\frac{1R}{hr}\right) = 0.67R$$

$$0.67R + 1.5R = 2.17R$$

D - Incorrect. 1) See C.1 2) See B.2-- Plausibility: IF Admin limit presumed >2.17R/yr.

Previous NRC exam history if any:

G2.3.4

2.3.4 **Knowledge of radiation exposure limits under normal or emergency conditions.** (CFR: 41.12 / 43.4 / 45.10) RO 3.2 SRO 3.7

Match justification: The question asks what are the radiation limits for the emergency conditions of protecting valuable equipment and saving a life.

Objective:

6. **LIST AND IDENTIFY** the individuals who can authorize re-entry into an evacuated area. (OPS40501B06).

67. G2.4.13 001/MOD/RO/MEM 4.4/4.6/N/N/3/REVISED/FIX

Unit 1 has just experienced a Reactor Trip, and the following conditions exist:

- The Operator at the Controls (OATC) and the Unit Operator (UO) are in the Control Room.
- Performance of EEP-0.0, Reactor Trip or Safety Injection, Immediate Operator Actions (IOAs) from memory is in progress.

Which one of the following states the roles assigned to each Plant Operator **during** the performance of IOAs of EEP-0.0 IAW SOP-0.8, Emergency Response Procedure User's Guide?

The OATC is required to perform IOAs and while the IOAs are being performed is required to __(1)__ any ESF components if auto actuations failed to occur as designed,

and

the UO will __(2)__

A. (1) manually align

(2) perform IOAs.

B. (1) manually align

(2) **NOT** perform IOAs, but will ensure the OATC performs the IOAs correctly.

C. (1) complete the IOAs of EEP-0.0 before manually aligning

(2) perform IOAs.

D✓ (1) complete the IOAs of EEP-0.0 before manually aligning

(2) **NOT** perform IOAs, but will ensure the OATC performs the IOAs correctly.

Rewrote per CE suggestion to ensure on the RO level.

A - Incorrect. The first part is incorrect, but plausible. It may seem that the OATC, while performing immediate operator actions (IOAs), should actuate any ESF components which should have actuated, but SOP-0.8 prohibits this until after reporting to the SS that IOAs are complete. The second part is incorrect per SOP-0.8, Step 3.7. Plausible, since this would have been correct a few months prior to this exam, and it is a recent change.

B - Incorrect. The first part is incorrect (see A). The second part is correct (see D).

C - Incorrect. The first part is correct (see D). The second part is incorrect (see A & D).

D - Correct. Neither operator is allowed to manually align any ESF component prior to the completion of immediate operator actions from memory, even though after completion is reported to the SS, ESF compnents which should have manually aligned are directed to be aligned manually by SOP-0.8, Step 3.7. The UO is not perform immediate operator actions while the OATC performs them, but is to ensure the OATC correctly performs the actions, and the UO is to station himself to trip the turbine per step 2 of the EEP-0.0 immediate actions if it does not automatically trip.

SOP-0.8, Version 18

3.7 The immediate actions in EEP-0, FRP-S.1, and ECP-0.0 **will be performed, in order, by the OATC. If available, the UO will ensure performance of the immediate actions are done correctly** and will take action as needed to trip the turbine in the event an automatic turbine trip does not occur. Typically, the UO will station himself at the Turbine Panel to allow for an immediate turbine trip in the event the turbine does not trip. When the operator has finished his/her immediate actions and reported completion to the Shift Supervisor, the shift supervisor will verify performance of the actions using the applicable ERP. **It is expected for the operator to perform manual actions to address failed ESF component actuations and to address foldout page items after the immediate actions are performed.** Early operator actions should not occur until after the immediate actions are verified by the Shift Supervisor. Following verification of immediate actions, the Shift Supervisor will proceed expeditiously to implement subsequent actions.

Previous NRC exam history if any:

G2.4.13

2.4.13 **Knowledge of crew roles and responsibilities during EOP usage.**
(CFR: 41.10 / 45.12) RO 4.0 SRO 4.6

Match justification: A recent change (policy changed during the Simulator training portion of the current Initial License Class) to the Ops Policy for crew member roles and responsibilities during the performance of immediate operator actions during EOP usage must be understood to answer this question correctly.

Objective:

6. **LIST AND IDENTIFY** the responsibilities of individual using a procedure (OPS-40504A07).

68. G2.4.16 001/NEW/RO/C/A 3.5/4.4/N/N/3/NO CHANGE OK/FIX

Unit 1 was at 100% power when a Large Break LOCA and a subsequent LOSP occurred. The following conditions exist:

- The crew is performing the actions of ECP-0.0, Loss of ALL AC Power.
- Attempts to restore power to any 4160V bus from any source per the step, "Restoration of power to any emergency bus", have all been **unsuccessful.**
- ALL Core Exit Thermocouples (CETCs) read 725°F and are increasing.

Which one of the following is the required procedural flowpath?

A.✓ Continue in ECP-0.0 until power is restored to at least one emergency bus.

B. Continue in ECP-0.0 until instructed to monitor CSF status trees.

C. Immediately transition to FRP-C.2, Response to Degraded Core Cooling, from any step of ECP-0.0.

D. Immediately transition to FRP-C.1, Response to Inadequate Core Cooling, from any step of ECP-0.0.

Rewrote per CE suggestion to ensure on the RO level.

A - Correct. Remaining in ECP-0.0 until power is restored is correct since there are nn CSFs required to be entered while in ECP-0 per the caution at the beginning of the procedure.

CAUTION: Critical safety function status trees should be monitored for information only. No function restoration or other procedure should be implemented during a loss of all AC power.

B - Incorrect. there is no step in ECP 0 that directs monitoring CSFs.

C - Incorrect. See A above

D - Incorrect. See A above

SOP-0.8, Emergency Response Procedure User's Guide, Version 18.0

4.2 Applicability [of the CSFSTs: FRPs]

The user should begin monitoring the CSFSTs when directed by EEP-0 or upon transition from EEP-0. The CSFSTs are not monitored initially because the ERPs are already directing the initial action required to protect the barriers. **If the user enters ECP-0.0, the CSFSTs should be monitored for information only.** The Function Restoration Procedures assume that at least one train of safeguards busses is available. If all AC power has been lost, ECP-0.0 will provide the appropriate actions to protect the barriers.

Previous NRC exam history if any:

G2.4.16

2.4.16 **Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.**
(CFR: 41.10 / 43.5 / 45.13) RO 3.5 SRO 4.4

Match justification: This question requires knowledge of the EOP hierarchy to answer correctly. The EOP heirarchy involves the FRPs being implemented as the highest priority procedures except in certain cases (early in E-0 and during loss of all AC they are not implemented), also, in the event of SAMG entry requirements, there are only a few entry points from the ERG network, and the transition is directed at the specific procedure steps. When SAMG entry is required, it takes priority over the EOPs and FRPs.

Comment: RO Knowledge of basic high level EOP priorities is being tested with this question.

Objective:

1. **EVALUATE** plant conditions and **DETERMINE** if entry into (1) ECP-0.0, Loss of All AC Power; and/or (2) ECP-0.1, Loss of All AC Power Recovery, Without SI Required; and/or (3) ECP-0.2, Loss of All AC Power Recovery, With SI Required is required. (OPS-52532A02)
2. **LIST AND DESCRIBE** the sequence of major actions, when and how continuous actions will be implemented, associated with (1) ECP-0.0, Loss of All AC Power; (2) ECP-0.1, Loss of All AC Power Recovery, Without SI Required; (3) ECP-0.2, Loss of All AC Power Recovery, With SI Required. (OPS-52532A04)
3. **ANALYZE** plant conditions and **DETERMINE** the successful completion of any step in (1) ECP-0.0, Loss of All AC Power; (2) ECP-0.1, Loss of All AC Power Recovery, Without SI Required; (3) ECP-0.2, Loss of All AC Power Recovery, With SI Required. (OPS-52532A07)

69. WE01EG2.2.2 001/MOD/RO/MEM 4.6/4.1/N/N/2/NO CHANGE OK/SAT

An inadvertent Safety injection has occurred on Unit 1. ESP-1.1, SI Termination, was in progress when the following conditions occurred:

- MLB-1 1-1 and 11-1 lights are NOT LIT.
- Pressurizer level is dropping rapidly.
- SG narrow range water levels are:
  - -1A SG 42% ↑.
  - -1B SG 30% ↓.
  - -1C SG 31% ↓.
- All SG pressures are decreasing rapidly.
- All Main Steam Isolation Valves (MSIVs) are open.
- Ctmt pressure is 14 psig and increasing.

Which one of the following states:

1) the allowable actions to be taken per SOP-0.8, Emergency Response Procedure User's Guide,

and

2) the procedure to implement **IF** the crew is not sure of the procedural transition?

A.✓ 1) Close the MSIVs;

2) Enter ESP-0.0, Rediagnosis.

B. 1) Isolate all AFW to 1A SG;

2) Enter ESP-0.0, Rediagnosis.

C. 1) Close the MSIVs;

2) Re-enter EEP-0, Rx Trip and Safety Injection.

D. 1) Isolate all AFW to 1A SG;

2) Re-enter EEP-0, Rx Trip and Safety Injection.

A - Correct. 1) Operating MSIVs is appropriate since pressure is approaching 16.2 psig automatic actuation setpoint: SOP-0.8, ver 18.0 "IF the condition is recognized in sufficient time, crews are expected to take manual actions prior to reaching the automatic setpoint for [...] MSIV isolation.

2) SOP-0.8 states that ESP-0.0, may be entered any time after exiting E-0, when SI is in progress OR IS REQUIRED [and no CSFs are Challenged]. Since the crew is uncertain of what action is to be taken, ESP-0.0 is the appropriate action.

B - Incorrect. 1) Per SOP-0.8, Step 3.3.7 Early actions may be taken since the immediate operator actions are complete. 3.3.8, Early actions to isolate the 1A SG may be taken since the SG is obviously ruptured, and the procedure will subsequently isolate it for the optimal recovery strategy, BUT NOT until the level is above the tubes, as indicated by the adverse numbers level of 48% minimum NR level. EEP-3, which will direct isolating all AFW at step 4, gives the minimum SG level at which the AFW flow can be isolated, and minimum level has been attained in the given conditions (1A SG level is 42%<48% minimum). Plausible, even though in this situation, the level is not high enough to secure all AFW to the SG, but at or above 31% NR level for non-adverse containment conditions or at 48% for the given adverse containment conditions (ctmt >4 psig) it would be appropriate to isolate all AFW Flow and secure feeding 1A SG.

2) Correct See A #2

C - Incorrect. 1) See A #1 2) Re-entering EEP-0 is inappropriate since E-0 has already been exited; Plausible: PER SOP-0.8 section 4.4, "IF plant conditions degrade during recovery from reactor trip without safety injection, EEP-0.0 should be reentered and immediate actions performed prior to transition from ESP-0.1 to any FRP. Also, ESP-1.1 Fold out page gives direction to go to both EEP-2 (SG Fault) and EEP-3 (SGTR), and EEP-0 may be chosen to provide a priority on which procedure to use for mitigation based on the EEP-0 Diagnostic steps.

D - Incorrect. 1) See B #1 2) See C#2

EEP-3, Steam Generator Tube Rupture, Revision 24
NOTE: [CA] Maintaining ruptured SG(s) narrow range level greater than 31%{48%} prevents SG depressurization during RCS cooldown.
4 [CA] WHEN ruptured SG(s) narrow range level greater than 31%{48%}, THEN perform the following.

FNP-0-SOP-0.8, Emergency Response Procedure User's Guide, Version 18.0
3.7 Immediate Actions {CMT 0007770}
Early operator actions should not occur until after the immediate actions are verified by the Shift Supervisor.
3.8 Manual Operator Actions and Early Operator Actions
3.8.3 Crews may take early operator action when the step will mitigate the consequence of the event **but not interfere with optimal recovery strategies.** (Examples include: securing all but one condensate pump and calling for backup cooling to be aligned, taking manual control of Auxiliary Feedwater flow, restoring instrument air to containment, etc) The Shift Supervisor will be notified prior to the commencement of early operator action.

Previous NRC exam history if any:
2006 NRC exam-- FNP bank E-0/ESP-0.0-52530A02 012

WE01EG2.2.2
E01 Rediagnosis
2.2.2 **Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.**
(CFR: 41.6 / 41.7 / 45.2) RO 4.6 SRO 4.1

Match justification:
- Requires examinee to recognize the allowable actions and expectation to be taken without procedure per the User's guide and identify proper entry into ESP-0.0, Rediagnosis.

Objective: OPS-52530A05; Analyze plant conditions and DETERMINE if actuation or reset of any ESFAS is necessary.

OPS-52530A02; Evaluate plant conditions and determine if entry into [...] ESP-0.0 [...]is required.

70. WE04EA2.2 001/MOD/RO/MEM 3.6/4.2/N/N/3/EDITORIAL/SAT

The Unit 1 crew has transitioned to ECP-1.2, LOCA Outside Containment.

Which one of the following correctly states the Cold Leg Injection path which is isolated, and the parameter used to determine if the break is isolated IAW ECP-1.2?

The __(1)__ Cold Leg Injection path is isolated,

and

RCS __(2)__ rising is used to determine if the break is isolated IAW ECP-1.2.

	(1)	(2)
A.✓	RHR	Pressure
B.	RHR	Subcooling
C.	HHSI	Pressure
D.	HHSI	Subcooling

Added: "Cold Leg injection path" per CE suggestion, changed format to fill in the blank.

A - Correct. Per ECP-1.2 Steps 3.1 & 3.2.

B - Incorrect. The first part is correct (see A). The second part is incorrect per ECP-1.2. Plausible, since with temperature constant, subcooling would be going up with RCS pressure going up, but the temperature and trend is not given, nor is subcooling used per ECP-1.2.

C - Incorrect. First part is incorrect. Plausible, since it is a penetration into containment which is unisolated during a safety injection the same as the RHR injection to the cold leg. However, the procedure does not direct isolating this flowpath. The second part is correct (see A).

D - Incorrect. Both parts are incorrect (see C & B).

ECP-1.2 Version 7

Previous NRC exam history if any:

WE04EA2.2
E04 LOCA Outside Containment
EA2. Ability to determine and interpret the following as they apply to the (LOCA Outside Containment) (CFR: 43.5 / 45.13)
EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments. RO 3.6 SRO 4.2

Match justification: Knowledge of the LOCA outside containment procedure is required as related to isolating the potential leak sources and the indications which are used to determine the leak is isolated. RCS leakage in a TS limitations in the facility's license, and the procedure directed leak isolation will maintain the RCS leakrate within the limits.

Objective:

3. **LIST AND DESCRIBE** the sequence of major actions, when and how continuous actions will be implemented, associated with ECP-1.2, LOCA Outside Containment. (OPS-52532E04)

4. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing ECP-1.2, LOCA Outside Containment. (OPS-52532E06)

5. **ANALYZE** plant conditions and **DETERMINE** the successful completion of any step in ECP-1.2, LOCA Outside Containment. (OPS-52532E07)

71. WE05EK1.1 001/NEW/RO/MEM 3.8/4.1/N/N/4/EDITORIAL/SAT

Given the following plant conditions for Unit 1:

- A Train is On Service.
- The Operators have implemented FRP-H.1, Response to Loss of Secondary Heat Sink.
- RCS feed and bleed criteria was met and a manual Safety Injection was initiated IAW FRP-H.1.
- 1C Charging pump is tripped.
- PRZR PORV, PCV-445A, will **NOT** open.

Which one of the following describes the **MINIMUM** action(s) required to provide adequate core cooling?

A. Open one PORV.

B. Open four Reactor Vessel Head vents.

C.✓ Open one PORV AND Open four Reactor Vessel Head vents.

D. Open one PORV AND Open four Reactor Vessel Head vents, AND place 1B Charging pump on B train and start 1B Charging pump.

added the word four to each distractor with Reactor Vessel Head vents.

A - Incorrect. Both PORVs are required per FRB-H.1 of the background documents for FRP-H.1, Loss of heat sink Function Restoration Procedure.

B - Incorrect. One PORV and all Head vents must be open to provide an adequate heat sink if one PORV cannot be opened.

C - Correct. As stated in FRB-H.1 ver 2.0 , for ERP step 17 basis (below), the function provided by the second PORV capacity is cooling:

"[...] If both PRZR PORVs are not maintained open, the RCS **may not depressurize sufficiently to permit adequate feed of subcooled SI flow to remove core decay heat**. If core decay heat exceeds RCS bleed and feed heat removal capability, the RCS will repressurize rapidly, further reducing the feed of subcooled SI flow and resulting in a rapid decrease of RCS inventory. [...] IF a low pressure water source can not be aligned [to at least one intact SG], a SG **should not** be depressurized in order to minimize the risk of tube creep rupture [...]."

Safety Capacity: From TS B2.4.10, each safety is capable of 345,000 lb/hr Assuming that each HHSI pump can deliver 600 gpm each at 2485 psig, then the following calculation (not adjusting for Temperature correction which lowers the gal/lbm #) below still provides sufficient relief capacity to prevent integrity failure of the RCS due to overpressure conditions.

$$\left(\frac{1200 gal}{\min}\right)\left(\frac{7.48 lbs}{gal}\right)\left(\frac{60 \min}{hr}\right) = 538560 lbs/hr$$

17.3 Open both PRZR PORVs.	17.3 Perform the following. 17.3.1 Open all available PORV's. 17.3.2 Open reactor vessel head vent valves. RX VESSEL HEAD VENT OUTER ISO [] Q1B13SV2213A [] Q1B13SV2213B RX VESSEL HEAD VENT INNER ISO [] Q1B13SV2214A [] Q1B13SV2214B

D - Incorrect. Starting a second pump is not required, since one pump will deliver the required flow. Plausible, since two PORVs are required, and it may be incorrectly assumed that in this case both HHSI pumps are required. Starting a second pump impacts pump discharge pressure, and may be construed to improve the overall pump head capacity. Although starting a second centrifugal pump aligned in a parrallel configuration with another causes discharge pressure to rise, the shutoff head of both pumps remains unchanged. This action only alters the current operating point on the pump curve, and does not improve the overall capability of the pumps, as RCS pressure approaches the Safety valve setpoint, SI flow is continualy degraded.

Previous NRC exam history if any:

WE05EK1.1

E05 Loss of Secondary Heat Sink

EK1. Knowledge of the operational implications of the following concepts as they apply to the (Loss of Secondary Heat Sink) (CFR: 41.8 / 41.10, 45.3)

EK1.1 Components, capacity, and function of emergency systems. RO 3.8 SRO 4.1

Match justification:

Knowledge of the PORV's function and capacity requirements of the PORV during bleed and feed operations

Objective:

2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the components associated with the Pressurizer System, to include the components found on Figure 3, Pressurizer and Pressurizer Relief Tank (OPS-40301E02).

72. WE08EK1.1 001/NEW/RO/C/A 3.5/3.8/N/N/2/REVISED/FIX
Given the following plant conditions for Unit 1:

- A failure has occurred on TE-412, Loop A T_{AVG}, resulting in a constant output equivalent to 561°F.
- A failure has occurred on TE-422, Loop B T_{AVG}, resulting in a constant output equivalent to 570°F.
- The Reactor is manually tripped.

Which one of the following states the protective feature(s) that will prevent a Pressurized Thermal Shock (PTS) condition from developing, **with no operator actions**?

A. P-4, Reactor Trip Interlock.

B. Main Steam Line Isolation on High Flow **AND** Main Steam Line Isolation on Low Pressure.

C✓ Main Steam Line Isolation on Low Pressure (Main Steam Line Isolation on High Flow will **NOT** prevent a PTS condition).

D. Main Steam Line Isolation on High Flow (Main Steam Line Isolation on Low Pressure will **NOT** prevent a PTS condition).

Sequence of events:
With 2/3 TAVG failed above 543, 547, and 554 with a RX Trip, then the following occurs:
1) cooldown initiates on Rx Trip controller of steam dumps (ARMED by P-4) and Median Tavg is failed at 561°F = dumps STAY open trying to try to lower median Tavg to 547°F.
2) Main Feedwater Isolation on P-4 w/ 2/3 Protection Tavg channels ≤554°F does NOT occur to secure feed from SGFPs, so more cooldown due to ever feeding occures until P-14, SG Hi Hi level is reached.
3) P-12 will not actuate on 2/3 TAVG ≤ 543°F resulting in all steam dumps staying open.
4) even though there is excessive steam flow (Dumps + SGFP + other steam loads), there will be no high steam flow MSIV isolation since there is no P-12 actuation on LO LO TAVG.
5) Stm pressure falls to 585 psig (rate compensated)
   a) SI is actuated
   b) MSLIS is actuated

A - Incorrect. The FW isolation due to P-4 and LO TAVG does not occur due to the failures, but if it did it would limit the severity of this event by isolating the Feed Water flow;

Plausible: Main Feedwater Isolation is initiated to prevent excessive cooldown of the reactor or to lessen the severity of the transient overall.

B - Incorrect. MSLIS on High Flow will NOT actuate (see D). Low Main Steam pressure MSLIS will actuate (see C).

C -Correct. MS pressure will drop due to the TRIP controller of the Steam dumps causing an open signal to the steam dumps, and P-12 will not actuate to close the dumps at 543°F. By isolating the MSIVs, the steam release is stopped and the cooldown would be stopped at approximately 487 °F Tcold which is <100 °F from the 100% TCold value of approximately 547°F. Therefore PTS condition would be averted by this actuation signal.

D - Incorrect. The failures prevent the LO LO TAVG signal from occurring. Therefore only 1 (High steam flow) of the 2 parts (HIGH flow concurrent with LO-LO TAVG) of this signal will actuate.

Plausible: the purpose of this protective function is to back up the Low Steam line pressure MSLIS, for conditions when MSLIS has been blocked. This signal would, if it could actuate, limit the effects of the uncontrolled steam release from the SG and thereby, limit the cooldown.

WOG FRG-P.1
An event or series of events which leads to a relatively rapid and severe reactor vessel downcomer cooldown can result in a thermal shock to the vessel wall that may lead to a small flaw, which may already exist in the vessel wall, growing into a larger crack. The growth or extension of such a flaw may lead, in some cases [...], to a loss of vessel integrity.

NOTE TO EXAMINER:
Previous NRC exam history if any:

WE08EK1.1
E08 Pressurized Thermal Shock
EK1. Knowledge of the operational implications of the following concepts as they apply to the (Pressurized Thermal Shock) (CFR: 41.8 / 41.10, 45.3)
EK1.1 Components, capacity, and function of emergency systems. RO 3.5 SRO 3.8

Match justification:
The operational implication of the failure of the Protection Tavg signal on preventing PTS must be understood to select the correct answer. In the conditions given, the PTS is still prevented, but not by the same method as would occur if Tavg was operable. In this case, PTS is prevented only by the MSLIAS which closes MSIVs on Low Steam line pressure. Three other signals which would normally prevent or mitigate a PTS event are disabled by the failure (Hi Steam Flow/Lo Lo Tavg, FWIS/Lo Tavg, Steam Dump auto closure on Lo Lo Tavg), but not the MSLIAS on low steam line pressure.

• The given failure of the Median Tavg circuit (component), results in failure of several Reactor protection functions (function of emergency systems) which are either directly or indirectly involved with preventing, mitigating, or terminating a PTS challenge.

P-4 coincident with lo tavg-- overfeed= overcooling= PTS challenge
MSLIS-- Low pressure --- main function is for CNMT protection but also functions to isolate a break when down stream of the MSIV-- eliminating or terminating the C/D. (indirectly protecting from an oversteam event = overcooling=PTS challenge).
MSLIS--HIGH FlOW with LO-LO TAVG--- backup to the MSLIS low pressure for low power or shutdown modes of operation when MSLIAS-LP is blocked.
SI-- provided as valid distractor since SI actually can complicate a PTS condition by allowing rapid repressurization of the RCS, following a large cooldown. (PTS and/or Cold REPRESSURIZATION concern) but is required for core cooling.

Objective:

1. **RECALL AND 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 (RPS) and Engineered Safeguards Features (ESF) to include setpoint, coincidence, rate functions (if any), reset features, and the potential consequences for improper conditions to include those items in the following tables (OPS-52201I07):

   - Table 1, Reactor Trip Signals
   - Table 2, Engineered Safeguards Features Actuation Signals
   - Table 5, Permissives
   - Table 6, Control interlocks

5. **DEFINE AND EVALUATE** the operational implications of abnormal plant or equipment conditions associated with the operation of the Reactor Protection System (RPS) components and equipment to include the following (OPS-52201I09).
   - Normal Control Methods
   - Abnormal and Emergency Control Methods
   - Automatic actuation including setpoint (example SI, Phase A, Phase B, MSLIAS, LOSP, SG level)
   - Actions needed to mitigate the consequence of the abnormality

73. WE11EK1.3 001/NEW/RO/C/A 3.6/4.0/N/N/3//SAT

A Dual unit LOSP with a **Unit 2** Large Break LOCA has occurred and the following conditions occurred:

- CTMT pressure is 6 psig.
- EEP-1.0, Loss of Reactor or Secondary Coolant, is in progress.

At 1000: WA2, 1-2A DG GEN FAULT TRIP, comes into alarm.

At 1020: the following alarms **have just come in**:

- CF3, 2A OR 2B RHR PUMP OVERLOAD TRIP
- CH2, RWST LVL A TRN LO
- CH3, RWST LVL B TRN LO

Which one of the following is:

1) the correct status of Unit 2 emergency recirculation capability,

and

2) the action(s) that the applicable procedure(s) direct?

A. 1) One train ONLY of emergency recirc capability has been lost.

2) Transfer to Cold Leg recirc **AND** Containment Spray recirc at this time.

B. 1) One train ONLY of emergency recirc capability has been lost.

2) Transfer to Cold Leg recirc, but do **NOT** transfer to Containment Spray recirc at this time.

C✓ 1) Both trains of emergency recirc capability have been lost.

2) Verify both Containment spray pumps secured, **AND** minimize HHSI flow to the minimum required to remove decay heat.

D. 1) Both trains of emergency recirc capability have been lost.

2) Verify both Containment spray pumps **AND** HHSI pumps are secured while attempting to restore at least one train of emergency recirc.

A - Incorrect. The first part is incorrect (see A). The second part is incorrect, but plausible. If the fist part was correct, the second part would be correct except for the "at this time". The CS recirc line up is not begun at the RWST LOW level alarm (12.5 ft) even thought the ECCS recirc alignment is. The CS recirc alignment would be commences at the RWST LOW LOW level alarm (4.5 ft), and not "at this time".

B - Incorrect. Both trains of emergency recirc are lost. A Train is indicated lost due to the only available A train DG tripped alarm WA2 (A train RHR, HHSI, and CS pumps are deenergized). CF4 has been stated as "in alarm" for consistency with a loss of A train RHR flow due to A train losing power. The B train recirc capability has been lost due to CF3 indicating that the B Train RHR pump has tripped.

Plausible, since improper diagnosing either train with the indications given would lead applicant to believe one train was still available. The second part is incorrect, since no trains of recirc are available, but plausible. If the first part was correct, the second part would also be correct (i.e. transfer to one train of Cold leg recirc and leave the Containment spray system in the injection mode until the LO LO RWST level at 4.5 ft, then transfer the CS system to recirc).

C - Correct. Both trains of Emergency recirc capability have been lost (see A). The high level actions of this procedure that a RO is required to know is that flow from the RWST is minimized and makeup to the RWST is maximized. For this scenario, one Containment spray pump is pumping the RWST water to the containment where it is unavailable to cool the core. ECP-1.1 will direct securing the Containment Spray pump and throttle the HHSI flow to the minimum required to cool the core. Commence makeup to the RWST is also required, but was not included in the correct answer for brevity. The answer is still correct without every action that will be completed.

D - Incorrect. The first part is correct (see A). The second part is incorrect, since some minimum HHSI flow will be maintained. Plausible, since all pumps would be secured at the RWST LOW LOW level alarm which comes in at 4.5 ft. Confusion may exist as to the difference between the action and setpoint for the RWST Lo alarm and the Lo Lo alarm. Also, attempting to restore at least one train of emergency recirc is directed by the procedure.

OPS-52531G, ESP-1.3, TRANSFER TO COLD LEG RECIRCULATION, ESP-1.4, TRANSFER TO SIMULTANEOUS COLD AND HOT LEG RECIRCULATION, lesson plan:

Major Action Categories in ESP-1.3

The major action categories are discussed below in more detail.

1. Align ECCS for recirculation.
2. Align CTMT spray for recirculation.

OPS-52532D, ECP-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, ECP-1.3, LOSS OF EMERGENCY COOLANT RECIRCULATION CAUSED BY SUMP BLOCKAGE

Major Action Categories in ECP-1.1

A high level summary of the actions performed in ECP-1.1 is given in the form of major action categories.

1. Continue attempts to restore ECR.
2. Increase/conserve RWST level.
3. Initiate cool down to cold shutdown.
4. Depressurize RCS to minimize RCS subcooling.
5. Try to add makeup to RCS from alternate source.
6. Depressurize SGs to cool down and depressurize RCS.
7. Maintain RCS heat removal.

Previous NRC exam history if any:

WE11EKI.3
E11 Loss of Emergency Coolant Recirculation
EK1. Knowledge of the operational implications of the following concepts as they apply to the (Loss of Emergency Coolant Recirculation) (CFR: 41.8 / 41.10 / 45.3)
EK1.3 Annunciators and conditions indicating signals, and remedial actions associated with the (Loss of Emergency Coolant Recirculation). RO 3.6 SRO 4.0

Match justification: The first part of the question is written to present various Alarms and require the applicant to determine that the status of emergency recirc capability in that both trains of emergency recirc is lost. The second part of the question requires the applicant to know what the remedial actions for the loss of recirc are (high level RO required knowledge).

Objective:

1. **EVALUATE** plant conditions and **DETERMINE** if entry into (1) ECP-1.1, Loss of Emergency Coolant Recirculation; and/or (2) ECP-1.3, Loss of Emergency Coolant Recirculation, Caused by Sump Blockage is required. (OPS-52532D02)
2. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing (1) ECP-1.1, Loss of Emergency Coolant Recirculation; (2) ECP-1.3, Loss of Emergency Coolant Recirculation, Caused by Sump Blockage. (OPS-52532D06)

74. WE12EK2.1 001/NEW/RO/C/A 3.4/3.7/N/N/3/REVISED/FIX

Unit 2 is in Mode 3, preparing to open the MSIVs after warm-up of the Main Steam lines is complete using FNP-2-SOP-17.0, Main and Reheat Steam.

- RCS Temp is 547°F.
- All required manual MSIV Bypass Warmup and Air Operated MSIV Bypass valves are open IAW FNP-2-SOP-17.0.
- Steam header pressure and each SG pressures are approximately equal.

The UO opens the MSIVs, and immediately after opening all MSIVs:
- A large Steam Break occurs in the Turbine Building.
- The MSIVs would NOT close EITHER automatically OR when the MCB handswitches were placed in the CLOSE position.

Which one of the following describes the number of **manual** MSIV Bypass Warmup valve(s) installed, and the final position of the Air Operated MSIV Bypass Valves?

_(1)_ **manual** Main Steam Isolation Bypass Warmup Valve(s) is(are) installed to warm up main steam lines per FNP-2-SOP-17.0,

and

the final positions of the Air Operated MSIV Bypass Valves is _(2)_.

	(1)	(2)
A.	3	open
B.	3	closed
C.	1	open
D.✓	1	closed

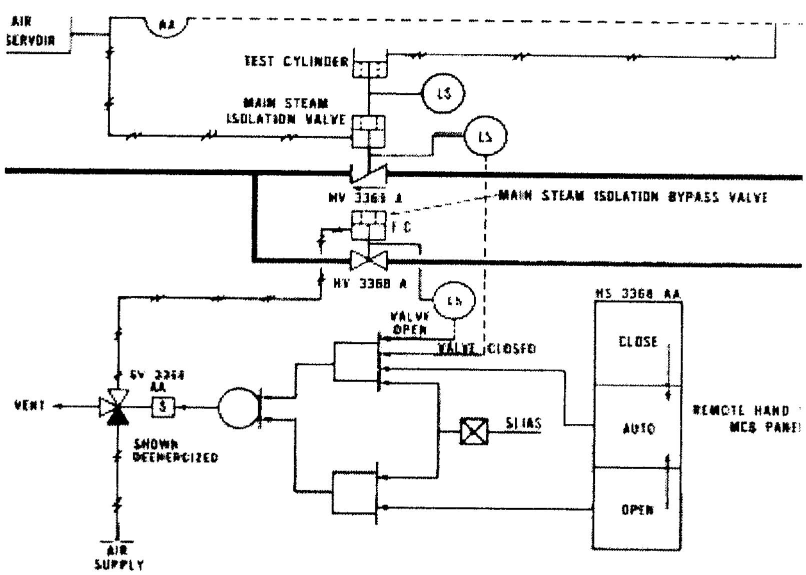
Changed one of the two parts of the question and each choice per CE suggestion.

A - Incorrect. 1) Incorrect. Only one manual warmup valve is installed on Unit 2. Plausible, since 3 steam lines must ultimately be warmed up and on Unit 1 3 manual warmup valves exist. This would be a correct choice if it were on Unit 1. 2) Incorrect. The MSIV Bypass valves are interlocked with the MSIVs such that they close immediately when the MSIV is not Closed. Further, if MSLIAS is satisfied (585 psig rate sensitive) the BYPASS valves will have shut.
Plausible: The MSIV Bypasses are normally shut AND are not operated by the MCB MSIV "TRIP" handswitch directly. These valves are equipped with their own handswitches.

B - Incorrect. 1) Incorrect (see A). 2) this is correct. The MSIV bypass valves will be closed immediately upon the MSIV Closed limit switch not being satisfied.

C - Incorrect. 1) Correct (see D #1). 2) Incorrect (See A #2).

D -Correct. 1) both parts are correct. on Unit 2, only one manual valve is installed to warmup the main steam lines. 2) The MSIVs are interlocked such that when the MSIV is no longer closed, the bypass valves close.



FNP-1-SOP-17.0,Version 59.0

4.2.5 Slowly open main steam isolation bypass warmup valves:

☐ 1A MS BYP WARMUP VLV, N1N11V019A

☐ 1B MS BYP WARMUP VLV, N1N11V019B

☐ 1C MS BYP WARMUP VLV, N1N11V019C

FNP-2-SOP-17.0, Version 51.0

4.2.5 Slowly open Main Steam Isolation Bypass Warmup Valve N2N11V019

Previous NRC exam history if any:

WE12EK2.1
E12 Uncontrolled Depressurization of all Steam Generators
EK2. Knowledge of the interrelations between the (Uncontrolled Depressurization of all Steam Generators) and the following: (CFR: 41.7 / 45.7)
EK2.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
RO 3.4 SRO 3.7

Match justification:
the Uncontrolled depressurization -- STEAM break with NO successful MSIV closure. Manual features, and interlocks--- MSIV Bypass valves are interlocked with MSIV position to ensure that the MSLIS can complete its function. These actions are those required by ECP-2.1 in an attempt to terminate the uncontrolled depressurization.

Objective:

6. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the Main and Reheat Steam System components and equipment, to include the following (OPS-40201A07):
   - Normal control methods
   - Abnormal and Emergency Control Methods
   - Automatic actuation including setpoint (example SI, Phase A, Phase B, MSLIAS, LOSP, SG level)
   - Protective isolations such as high flow, low pressure, low level including setpoint
   - Protective interlocks
   - Actions needed to mitigate the consequence of the abnormality

75. WE16EK2.1 001/BANK/RO/C/A 3.0/3.3/N/N/2/HBF/SAT

A Large Break LOCA has occurred on Unit 2, and the following conditions exist:

- R-27A and B, CTMT HI RANGE, indicates 3 Rem/hr.
- RE-11, CTMT PART, and RE-12, CTMT GAS, on the Integrated Plant Computer (IPC) shows an initial upscale followed by a slow trend towards background levels.

Which ONE of the following describes the reason for the observed trend on RE-11 and RE-12 towards a background count rate?

RE-11 and RE-12 are isolated from containment **directly** from a _______ signal.

A.✓ Phase A

B. Phase B

C. Safety Injection

D. Containment Ventilation Isolation

A. Correct, CTMT ATMOS TO R-11/12 Q1E14MOV3660,3657 &3658 are closed by a containment isolation - 'T' signal.

B. Incorrect, Containment phase B occurs at a higher pressure and does not affect the R-11/R-12 valves.

C. Incorrect, CTMT ATMOS TO R-11/12 Q1E14MOV3660,3657 &3658 close on a 'T' signal not an 'S' signal.

D. Incorrect, Containment ventilation isolation signal is generated by a MANUAL Phase A or B signal, any signal that generates an SI, HI-HI rad on RE-24A/B and does not affect the isolation valves for R-11/12.

EEP-0, ATTACHMENT 2, Revision 38

Previous NRC exam history if any: N/A

WE16EK2.1
E16 High Containment Radiation
EK2. Knowledge of the interrelations between the (High Containment Radiation) and the following: (CFR: 41.7 / 45.7)
EK2.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. RO 3.0 SRO 3.3

Match justification: The interrelation between the given high radiation condition in in containment during an accident and the function of the safety system and the automatic features which isolate the high radiation from the public must be understood to answer this question.

High containment radiation is indicated on R-27A & B as well as R-11 & 12 due to the LOCA, and then R-11 & 12 are isolated due to the Automatic ISOLATION SIGNALS. This meets entry criteria for the FNP-1-FRP-Z.3, Response To High Containment Radiation Level (YELLOW PATH: Both CTMT RAD **NOT** LESS THAN 2 R/hr.) per FNP-1-CSF-0.5 CONTAINMENT Revision 17

Objective:
RMS-40305A07