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

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)

Step	Action/Expected Response	Response Not Obtained

NOTE: Steps 1 and 2 are IMMEDIATE OPERATOR actions.

__ **1 Verify NO load change in progress.**

YES

1 Check for cause of load change.

1.1 IF load rejection in progress or has occurred, THEN go to FNP-1-AOP-17.0, RAPID LOAD REDUCTION.

1.2 IF secondary leakage is indicated, THEN go to FNP-1-AOP-14.0, SECONDARY SYSTEM LEAKAGE.

__ **2 IF unexplained rod motion occurring, THEN stop rod motion.**

CORRRECT

2.1 IF rod control in AUTO, THEN place rod control in MANUAL.

2.1 IF rod control in MANUAL, THEN place rod control in AUTO

NOTE: In AUTO rod control, rods will step OUT if TAVG less than TREF by at least 1.5 degrees, and Rods will step IN if TAVG greater than TREF by at least 1.5 degrees.

2.1.1 IF AUTO rod motion due to TAVG/TREF mismatch, THEN verify rod motion stops when TAVG is within 1 degree of TREF

2.2 IF unexplained rod motion NOT stopped, THEN perform the following.

2.2.1 Trip the reactor

A & C INCORRRECT

2.2.2 Go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION

__Page Completed

2. 001K2.02 001/MOD/RO/MEM 3.6/3.7//N/2/CVR/VER 5 EDITORIAL

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.

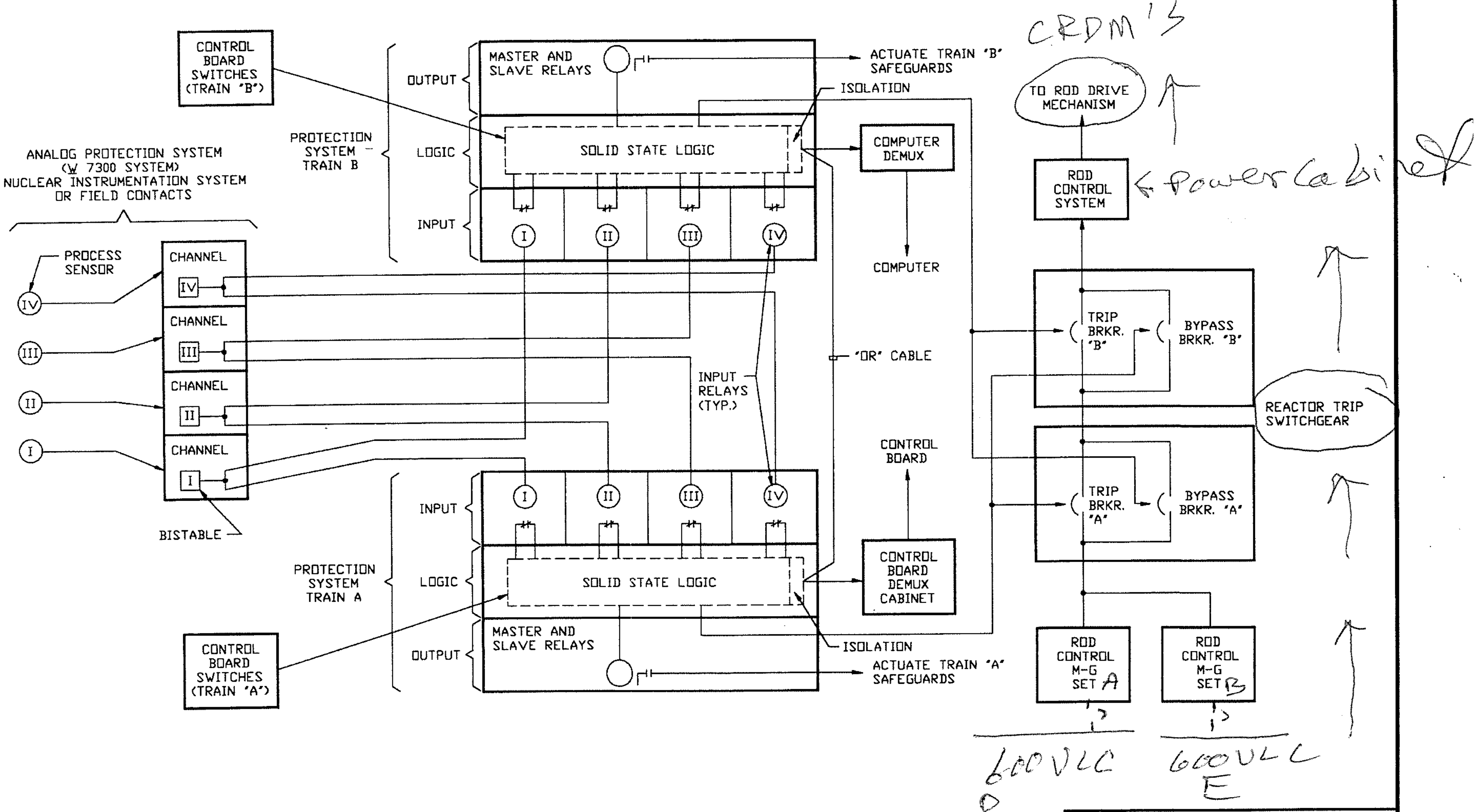
Previous NRC exam history if any: 2005 Vogtle NRC exam under 001K2.01 (power supplies to MG SETS)

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).



REACTOR PROTECTION SYSTEM BOUNDARIES

FIGURE F-1

DF03
1D 600V LOAD CENTER **AB - 139'** **D177010**

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q1R16B0006-A	**1D 600V LOAD CENTER**	
ED01	Q1R16BKRED01	PT COMPARTMENT	
ED02	**Q1R11B0004-A**	**1D 4160/600V SST (NORMAL) <<< DF03**	
ED03	N1C11M0001A-N	1A CRDM MG SET	
ED04	**Q1R42E0001A-A**	**1A BATTERY CHARGER >>> 1A 125V DC SWITCHGEAR (Q1R42B0001A-A)>>>**	F-3
ED05	**QSR17B0006-A**	**1F 600/208V MCC >>>**	F-79
ED06	Q1E22M0001A-A	1A REACTOR CAVITY DILUTION FAN	
ED08	**Q1R16B0002-A**	**1A 600V LOAD CENTER (ALTERNATE - EMERG) >>> EA09 >>>**	D-46
ED09	**Q1R42E0001C-AB**	**600V AC DISC SW Q1R18B0001A-A >>> 1C BATT CHARGER (A TRAIN SUPPLY) >>> 125V DC DISC SW Q1R18B0002A-A >>> 1A 125V DC SWGR >>>**	F-3
ED10	**Q1R17B0001-A**	**1A 600/208V MCC >>>**	F-92
ED11	N1T47M0001B-A	1B CRDM COOLER FAN	
ED12	**Q1R16B0008-AB**	**1F 600V LOAD CENTER (ALTERNATE) <<< EF06**	
ED13	**Q1R17B0509-A**	**1S 600/208V MCC (NORMAL) >>>**	F-98
ED14	**Q1R17B0008-A**	**1U 600/208V MCC >>>**	F-104
ED15	Q1E12M0001A-A	1A CONTAINMENT COOLER (EMERG./ LOW SPEED)	
ED16	Q1E12M0001B-A	1B CONTAINMENT COOLER (EMERG./ LOW SPEED)	

DG03
1E 600V LOAD CENTER **AB - 121'** **D177011**

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q1R16B0007-B	**1E 600V LOAD CENTER**	
EE01	Q1R16BKREE01	PT COMPARTMENT	
EE02	**Q1R11B0005-B**	**1E 4160/600V SST (NORMAL) <<< DG03**	
EE03	N1C11M0001B-N	1B CRDM MG SET	
EE05	**Q1R42B0001B-B**	**1B BATTERY CHARGER Q1R42E0001B-B >>> 1B 125VDC SWGR >>>**	G-3
EE06	**Q1R42B0001B-B**	**600V AC DISC SW Q1R18B0001B-B >>> 1C BATT CHARGER Q1R42E0001C-AB(B TRAIN SUPPLY) >>> 125V DC DISC SW Q1R18B0002B-B >>> 1B 125V DC SWGR >>>**	G-3
EE07	**Q1R16B0005-B**	**1C 600V LOAD CENTER (ALTERNATE-EMERG) >>> EC10 >>>**	E-5
EE08	Q1E12M0001C-B	1C CONTAINMENT COOLER (EMERG./ LOW SPEED)	
EE09	Q1E22M0001B-B	1B REACTOR CAVITY DILUTION FAN	
EE10	**Q1R17B0002-B**	**1B 600/208V MCC >>>**	G-76
EE11	**QSR17B0007-B**	**1G 600/208V MCC >>>**	G-84
EE12	**Q1R16B0008-AB**	**1F 600V LOAD CENTER (ALTERNATE) <<< EF08**	
EE13	N1T47M0001A-B	1A CRDM COOLER FAN	
EE14	**Q1R17B0510-B**	**1T 600/208V MCC >>>**	G-99
EE15	**Q1R17B0009-B**	**1V 600/208V MCC >>>**	G-105
EE16	Q1E12M0001D-B	1D CONTAINMENT COOLER (EMERG./ LOW SPEED)	

1. ROD CONT-40204I04 004/HLT//M (LEVEL 1) SYS/001K2.01//// Ban/BZ

001K2.02

Which ONE of the following correctly states the order of components through which power flows to the Control Rod Drive Mechanisms?

A. MG Supply Breakers, then 600v LCC D and E, then Motor Generator Sets, then Power Cabinets, then Reactor Trip Breakers.

B. MCC A and B, Power Cabinets, then MG Supply Breakers, then Motor Generator Sets, then Reactor Trip Breakers.

C✓ 600v LCC D and E, then MG Supply Breakers, then Motor Generator Sets, then Reactor Trip Breakers, then Power Cabinets.

D. MCC A and B, Motor Generator Sets, then MG Supply Breakers, then Reactor Trip Breakers, then Power Cabinets.

A. Incorrect. Plausible because all parts of the correct answer are listed, but in an incorrect order.
B. Incorrect. Plausible because all parts of the correct answer are listed, but in an incorrect order.
C. Correct. See Reference 1, Page 9.
D. Incorrect. Plausible because all parts of the correct answer are listed, but in an incorrect order. This choice would be correct if "Motor Breakers" were replaced with "Generator Breakers".

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.

2005 VNP nrc exam

K/A

001 Control Rod Drive

K2.01 Knowledge of bus power supplies to the following: One-line diagram of power supply to M/G sets.

K/A MATCH ANALYSIS

Question tests knowledge of the power supplies to the M/G Sets at the memory level.

3. 003A2.03 001/NEW/RO/C/A 2.7/3.1/N/N/4/CVR/Y

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?

	<u>**CAUSE**</u>	<u>**ACTION**</u>
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. This would be SRO only knowledge if it were not also a < 1 hour TS action to be in mode 3 prior to being able to conduct a normal shutdown. ?

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

Step	Action/Expected Response	Response Not Obtained

NOTE: Step 1 and 2 are IMMEDIATE OPERATOR actions.

Step	Action/Expected Response	Response Not Obtained
__ 1	**Maintain SG narrow range level stable at approximately 65% using:** [ ] Main Feedwater Regulating Valves [ ] Main Feedwater Bypass Regulating Valves. [ ] Auxiliary Feedwater Control Valves.	1 IF SG level rise cannot be controlled, THEN close the affected SG Main Feedwater Stop Valve(s) OR Auxiliary Feedwater Stop valve(s). [ ] 1A SG Q1N21MOV3232A [ ] 1B SG Q1N21MOV3232B [ ] 1C SG Q1N21MOV3232C OR [ ] 1A SG Q1N23MOV3350A [ ] 1B SG Q1N23MOV3350B [ ] 1C SG Q1N23MOV3350C
__ 2	**Check 1A and 1B RCPs - RUNNING.**	2 Manually close pressurizer spray valve for affected RCP. [ ] 1A RCS loop spray valve PK-444C [ ] 1B RCS loop spray valve PK-444D
__ 3	**Monitor Tavg for all three RCS loops ≥ 541°F. (TS 3.4.2)** A & B 2 nd Part No Correct	3 Perform the following.. 3.1 IF the main generator is ON LINE, THEN trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION Yes 3.2 IF the main generator is OFF LINE, THEN raise Tavg ≥ 541°F within 30 minutes 3.2.1 Adjust steam dumps to reduce secondary power demand as necessary 3.2.2 Verify rod control in MANUAL

Step 3 continued on next page

__Page Completed

Step	Action/Expected Response	Response Not Obtained
		3.2.3 Stabilize Tavg in the idle loop(s) > 541°F while maintaining the running loop(s) < 554°F by adjusting rod position and/or boron concentration
		3.2.4 IF unable to restore Tavg ≥ 541°F, THEN trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION
__ 4	**Maintain PRZR pressure 2200-2300 psig.**	
4.1	Control PRZR heaters as required.	
4.2	IF 1A and 1B RCPs running, THEN, control pressurizer pressure with both normal spray valves.	4.2 Perform the following 4.2.1 IF 1B RCP running, THEN control pressurizer pressure with PK-444D.

NOTE: Running 1A and 1C RCPs will be required to provide adequate spray flow through the 1A RCS loop spray valve.

Step	Action/Expected Response	Response Not Obtained
		4.2.2 IF 1A & 1C RCPs are running, THEN control pressurizer pressure with PK-444C.
		4.2.3 IF spray flow is adequate, THEN proceed to step 5.
		4.2.4 IF no spray valves are available, THEN proceed to step 4.4.
4.3	Proceed to step 5.	

Step 4 continued on next page

__Page Completed

Step	Action/Expected Response	Response Not Obtained
		5.2 IF letdown has isolated due to a plant transient, THEN establish normal letdown using ATTACHMENT 1, RESTORING LETDOWN.
		5.3 IF a letdown isolated due to a system malfunction, THEN perform the following: [ ] Attempt to restore any letdown flow using FNP-1-AOP-16.0, CVCS MALFUNCTION. [ ] Continue with applicable steps of this procedure.
		5.4 WHEN normal letdown restored AND IF required, THEN return to step 4.4 to establish auxiliary spray.
__ 6	**Maintain PRZR level at approximately 22%.**	
__ 7	**Within six hours of the loss of RCS flow complete the following:** 7.1 IF the unit is in Mode 1 or 2, THEN place unit in Mode 3 using the following procedures: [ ] FNP-1-UOP-3.1, POWER OPERATION [ ] FNP-1-UOP-2.1, SHUTDOWN OF UNIT FROM MINIMUM LOAD TO HOT STANDBY	C+D 2nd part incorrect

Step 7 continued on next page

__Page Completed

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ 541°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. T_{avg} in one or more RCS loops not within limit.	A.1	Be in MODE 3.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify RCS T_{avg} in each loop ≥ 541°F.	--------NOTE-------- Only required if low low T_{avg} alarm not reset and any RCS loop T_{avg} < 547°F ------------------------ 30 minutes thereafter

4. 004G2.4.21 001/NEW/RO/C/A 4.0/4.6/N/N/2/CVR/Y

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 have 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 will be started or not?

A. Enter FRP-C.1 RCPs will be started

B. Enter FRP-C.1 RCPs will **NOT** be started

C. Remain in FRP-C.2 RCPs will be started

D✓ Remain in FRP-C.2 RCPs will **NOT** 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

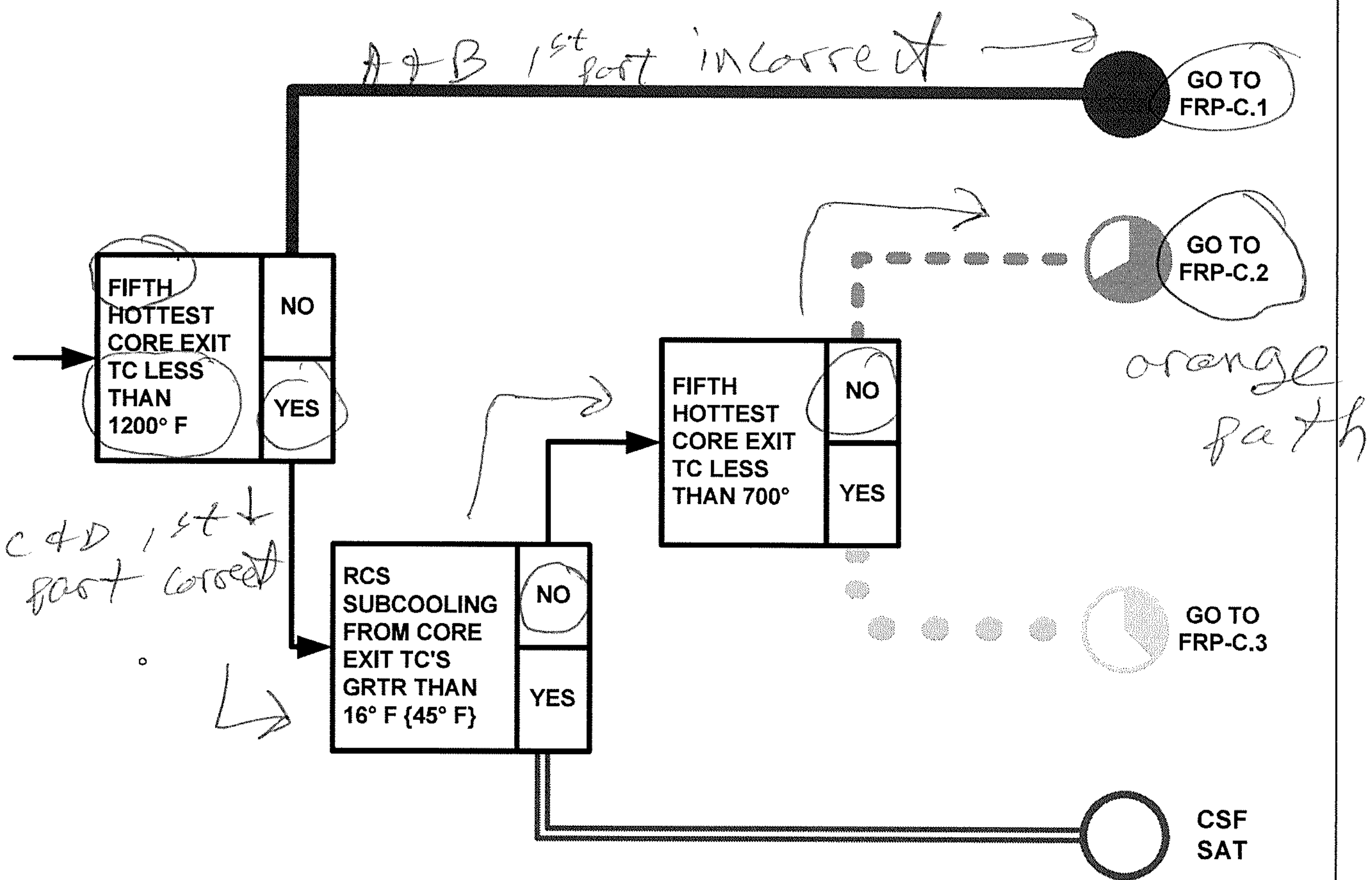
Previous NRC exam history if any:

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)


A+B 1st part incorrect
GO TO FRP-C.1
GO TO FRP-C.2
orange path
FIFTH HOTTEST CORE EXIT TC LESS THAN 1200° F
NO
YES
C+D 1st part correct
FIFTH HOTTEST CORE EXIT TC LESS THAN 700°
NO
YES
RCS SUBCOOLING FROM CORE EXIT TC'S GRTR THAN 16° F {45° F}
NO
YES
GO TO FRP-C.3
CSF SAT

Step	Action/Expected Response	Response NOT Obtained
19	**Check core cooling.**	
19.1	Check core exit T/Cs - LESS THAN 1200°F.	19.1 Proceed to Step 21. OBSERVE NOTE PRIOR TO STEP 21.
19.2	Check at least two RCS hot leg temperatures - LESS THAN 350°F. RCS HOT LEG TEMP [] TR 413	19.2 Return to step 17.
19.3	Check REACTOR VESSEL LEVEL indication - GREATER THAN 0% UPPER PLENUM.	19.3 Return to step 17.
20	**Go to FNP-1-EEP-1, LOSS OF REACTOR OR SECONDARY COOLANT, step 14.**	

NOTE: Normal conditions are desired but not required for starting RCPs.

21	**Check if RCPs should be started.**	incorect for C.2 A + C 2nd parts
21.1	Check core exit T/Cs - GREATER THAN 1200°F.	21.1 Proceed to step 22.

Step 21 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: Further degradation of core cooling can occur if any running RCP is stopped before being directed by this procedure even if normal support conditions are lost.

****

1 Monitor RWST level.

RWST
LVL
[] **LI 4075A**
[] **LI 4075B**

1.1 [CA] WHEN RWST level less than 12.5 ft, THEN go to FNP-1-ESP-1.3, TRANSFER TO COLD LEG RECIRCULATION.

2 Verify proper SI valve alignment using ATTACHMENT 2, SI VALVE ALIGNMENT FOR COLD LEG INJECTION.

Pgs 2-21 provided to show that no RCP is started in this procedure

D 2nd part Correct

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
3	**Check any HHSI flow - GREATER THAN 0 gpm.** A TRN HHSI FLOW [] FI 943 HHSI B TRN RECIRC FLOW [] FI 940	3 Perform the following. 3.1 Verify all available charging pumps started. CHG PUMP [] 1A amps > 0 [] 1B amps > 0 [] 1C amps > 0 3.2 Verify charging pump MOV disconnects closed using ATTACHMENT 3, CHARGING PUMP MOV DISCONNECTS. 3.3 Verify proper SI alignment. CHG PUMPS TO REGENERATIVE HX [] Q1E21MOV8107 closed [] Q1E21MOV8108 closed RWST TO CHG PUMP [] Q1E21LCV115B open [] Q1E21LCV115D open VCT OUTLET ISO [] Q1E21LCV115C closed [] Q1E21LCV115E closed HHSI TO RCS CL ISO [] Q1E21MOV8803A open [] Q1E21MOV8803B open CHG PUMP SUCTION HDR ISO [] Q1E21MOV8130A open [] Q1E21MOV8130B open [] Q1E21MOV8131A open [] Q1E21MOV8131B open CHG PUMP DISCH HDR ISO [] Q1E21MOV8132A open [] Q1E21MOV8132B open [] Q1E21MOV8133A open [] Q1E21MOV8133B open

Step 3 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

3.4 IF HHSI flow now established, THEN proceed to step 4, IF NOT, establish HHSI bypass SI flow.

CHG PMP RECIRC
TO RCS COLD LEGS
[] Q1E21MOV8885 open

HHSI TO
RCS CL ISO
[] Q1E21MOV8803A closed
[] Q1E21MOV8803B closed

3.5 IF HHSI flow now established, THEN proceed to step 4, IF NOT, perform the following.

3.5.1 Open HHSI isolation valves.

HHSI TO
RCS CL ISO
[] Q1E21MOV8803A
[] Q1E21MOV8803B

3.5.2 Align charging pump suction header isolation valves based on 1B charging pump status.

1B Charging Pump Status	Aligned As A Train pump	Aligned As B Train pump
CHG PUMP SUCTION HDR ISO Q1E21MOV	[] 8130A open [] 8130B open [] 8131A closed [] 8131B closed	[] 8130A closed [] 8130B closed [] 8131A open [] 8131B open

Step 3 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

3.5.3 Align charging pump discharge header isolation valves based on 1B charging pump status.

1B Charging Pump Status	Aligned As A Train pump	Aligned As B Train pump
CHG PUMP DISCH HDR ISO Q1E21MOV	[] 8132A open [] 8132B open [] 8133A closed [] 8133B closed	[] 8132A closed [] 8132B closed [] 8133A open [] 8133B open

NOTE: Continuing efforts to establish SI flow should not interfere with performance of the remainder of this procedure.

3.6 If HHSI flow NOT established, THEN Continue efforts to establish SI flow.

- HHSI flow
- LHSI flow
- Any form of RCS injection.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: Pump damage may occur if RHR pumps are operated on miniflow for longer than three hours with no CCW supplied to the RHR heat exchangers.

****

4 Check LHSI status.

4.1 Verify CCW flow to RHR heat exchangers - ESTABLISHED.

CCW TO
1A(1B) RHR HX
[] Q1P17MOV3185A open
[] Q1P17MOV3185B open

4.2 Check RCS pressure - LESS THAN 275 psig{435 psig}.

RNO: 4.2 Proceed to step 5.

1C(1A) LP
RCS NR PRESS
[] PI 402B
[] PI 403B

Step 4 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
4.3	Check both RHR flows - GREATER THAN 1.5×10^3 gpm. 1A(1B) RHR HDR FLOW [] FI 605A [] FI 605B	4.3 Verify LHSI properly aligned. RHR PMP [] 1A amps > 0 [] 1B amps > 0 1A(1B) RHR HX TO RCS COLD LEGS ISO [] Q1E11MOV8888A open [] Q1E11MOV8888B open RWST TO 1A(1B) RHR PUMP [] Q1E11MOV8809A open [] Q1E11MOV8809B open 1A(1B) RHR HX DISCH VLV [] HIK 603A open [] HIK 603B open 1A(1B) RHR HX BYP FLOW VLV [] FK 605A closed [] FK 605B closed 1A(1B) RHR TO RCS HOT LEGS XCON [] Q1E11MOV8887A open [] Q1E11MOV8887B open

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
5	**Check RCS vent paths.**	
5.1	Check any PRZR PORV ISO - POWER AVAILABLE.	5.1 Restore power to PRZR PORV ISO valves unless de-energized for inoperable PORVs not capable of being manually cycled.
5.2	Verify both PRZR PORVs - CLOSED.	5.2 Perform the following. 5.2.1 Close PRZR PORVs. 5.2.2 IF any valve can NOT be closed, THEN close its PORV ISO valve.

NOTE: The purpose of the following step is to establish an available PORV flowpath for mitigation of overpressure conditions, without relying on the PRZR code safety valves. A failed open PORV must not be unisolated. A leaking PORV which is isolated with power available to the isolation valve should remain isolated until needed to reduce RCS pressure or mitigate an RCS overpressure condition. Any leaking PORV should be re-isolated when not in use.

Step	Action/Expected Response	Response NOT Obtained
5.3	Check at least one PRZR PORV ISO - OPEN.	5.3 Open any PRZR PORV ISO not required to isolate an open or leaking PORV.
5.4	Verify reactor vessel head vent valves - CLOSED. RX VESSEL HEAD VENT OUTER ISO [] Q1B13SV2213A [] Q1B13SV2213B RX VESSEL HEAD VENT INNER ISO [] Q1B13SV2214A [] Q1B13SV2214B	

____Page Completed

Step	Action/Expected Response	Response NOT Obtained
6	**Check RCP status.**	
	6.1 Check at least one RCP - STARTED.	6.1 Proceed to Step 8.

****

CAUTION: To prevent potential seal damage, neither seal injection nor CCW cooling should be restored to a RCP which has lost both seal injection and CCW cooling.

****

NOTE: Normal support conditions for running RCPs are desired, however, RCP operation must continue even if support conditions cannot be maintained.

6.2 Verify No. 1 seal support conditions established.

6.2.1 [CA] Maintain seal injection flow - GREATER THAN 6 gpm.

6.2.2 Verify No. 1 seal leakoff flow - WITHIN FIGURE 1 LIMITS.

6.2.3 Verify No. 1 seal differential pressure - GREATER THAN 200 psid.

6.3 Verify CCW - ALIGNED.

CCW FROM
RCP THRM BARR
[] Q1P17HV3045 open
[] Q1P17HV3184 open

Step 6 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
6.4	Check RCP thermal barrier - INTACT. RCP THRM BARR CCW FLOW HI [] Annunciator DD2 clear	6.4 Verify CCW flow isolated. CCW FROM RCP THRM BARR [] Q1P17HV3045 closed [] Q1P17HV3184 closed
6.5	Check CCW to RCP oil coolers - SUFFICIENT. CCW FLOW FROM RCP OIL CLRS LO [] Annunciator DD3 clear	6.5 Verify CCW - ALIGNED. CCW TO RCP CLRS [] Q1P17MOV3052 open CCW FROM RCP OIL CLRS [] Q1P17MOV3046 open [] Q1P17MOV3182 open
6.6	Check RCP oil level - SUFFICIENT. RCP 1A(1B,1C) BRG UPPER/LOWER OIL RES LO LVL [] Annunciator HH1 clear [] Annunciator HH2 clear [] Annunciator HH3 clear	

NOTE: Since RCP damage may occur when operating RCPs without normal support conditions established or under highly voided RCS conditions, the intent of the following step is to save one RCP (which provides the best pressurizer spray capability) for future use, if all three RCPs are running.

Step	Action/Expected Response	Response NOT Obtained
7	**Check if one RCP should be stopped.**	
7.1	Check ALL RCPs - STARTED	7.1 Proceed to Step 9.
7.2	Stop RCP 1B.	
7.3	Proceed to Step 9.	

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
8	**Check core cooling.**	
8.1	Check REACTOR VESSEL LEVEL indication - GREATER THAN 0% UPPER PLENUM.	8.1 IF SI established, THEN return to step 2, IF NOT, proceed to step 9.
8.2	Check core exit T/Cs - LESS THAN 700°F.	8.2 IF core exit T/Cs falling, THEN return to step 2, IF NOT, proceed to step 9.
8.3	Go to procedure and step in effect.	
9	**Check SI accumulator discharge valve status.**	
9.1	Check power to discharge valves - AVAILABLE. 1A(1B,1C) ACCUM DISCH ISO [] Q1E21MOV8808A [] Q1E21MOV8808B [] Q1E21MOV8808C	9.1 Close accumulator discharge valve disconnects using ATTACHMENT 1.
9.2	Check discharge valves - OPEN. 1A(1B,1C) ACCUM DISCH ISO [] Q1E21MOV8808A [] Q1E21MOV8808B [] Q1E21MOV8808C	9.2 IF accumulators have NOT discharged, THEN open discharge valves. 1A(1B,1C) ACCUM DISCH ISO [] Q1E21MOV8808A [] Q1E21MOV8808B [] Q1E21MOV8808C
10	**Monitor CST level.**	
10.1	[CA] Check CST level greater than 5.3 ft. CST LVL [] LI 4132A [] LI 4132B	10.1 Align AFW pumps suction to SW using FNP-1-SOP-22.0, AUXILIARY FEEDWATER SYSTEM.
10.2	Align makeup to the CST from water treatment plant OR demin water system using FNP-1-SOP-5.0, DEMINERALIZED MAKEUP WATER SYSTEM, as necessary.	

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: To prevent potential release of radioactive material to the atmosphere, a faulted or ruptured SG should only be used if no intact SG is available.

****

11 Check intact SG levels.

Step	Action/Expected Response	Step	Response NOT Obtained
11.1	Check narrow range levels - GREATER THAN 31%{48%}.	11.1	Verify total AFW flow to intact SGs greater than 395 gpm. AFW FLOW TO 1A(1B,1C) SG [] FI 3229A [] FI 3229B [] FI 3229C AFW TOTAL FLOW [] FI 3229

Step 11 continued on next page.

___Page Completed

Step **Action/Expected Response** **Response NOT Obtained**

11.2 [CA] WHEN intact SG narrow range level 31%-65%{48%-65%}, THEN maintain intact SG narrow range level 31%-65%{48%-65%}.

11.2.1 Control MDAFWP flow.

MDAFWP FCV 3227
RESET
[] A TRN reset
[] B TRN reset

MDAFWP TO
1A/1B/1C SG
B TRN
[] FCV 3227 in MOD

Intact SG	1A	1B	1C
MDAFWP TO 1A(1B,1C) SG Q1N23HV	[] 3227A in MOD	[] 3227B in MOD	[] 3227C in MOD
MDAFWP TO 1A(1B,1C) SG FLOW CONT HIC	[]3227AA adjusted	[]3227BA adjusted	[]3227CA adjusted

Step 11 continued on next page.

___Page Completed

Step **Action/Expected Response** **Response NOT Obtained**

11.2.2 Control TDAFWP flow.

TDAFWP FCV 3228
[] RESET reset

TDAFWP
SPEED CONT
[] SIC 3405 adjusted

Intact SG	1A	1B	1C
TDAFWP TO 1A(1B,1C) SG Q1N23HV	[] 3228A in MOD	[] 3228B in MOD	[] 3228C in MOD
TDAFWP TO 1A(1B,1C) SG FLOW CONT HIC	[]3228AA adjusted	[]3228BA adjusted	[]3228CA adjusted

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: Performance of step 12 will cause accumulator injection which may result in a red path on the INTEGRITY status tree. This procedure should be completed before transition to FNP-1-FRP-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITIONS.

****

NOTE: After the low steam line pressure SI is blocked, excessive opening of steam dumps can cause a high steam flow LO-LO TAVG main steam isolation signal.

12 Reduce pressure in all intact SGs to 100 psig.

12.1 WHEN P-12 light lit (543°F), THEN perform the following.

12.1.1 Block low steam line pressure SI.

STM LINE PRESS SI
BLOCK - RESET
[] A TRN to BLOCK
[] B TRN to BLOCK

12.1.2 Verify blocked indication.

BYP & PERMISSIVE
STM LINE ISOL.
SAFETY INJ.
[] TRAIN A BLOCKED light lit
[] TRAIN B BLOCKED light lit

12.1.3 Bypass the steam dump interlock.

STM DUMP
INTERLOCK
[] A TRN to BYP INTLK
[] B TRN to BYP INTLK

Step 12 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
12.1.4	Adjust steam header pressure controller to control cooldown rate. STM HDR PRESS [] PK 464 adjusted	
12.2	[CA] Maintain RCS cold leg cooldown rate - LESS THAN 100°F IN ANY 60 MINUTE PERIOD.	
12.3	IF condenser available, THEN dump steam to condenser from intact SGs. BYP & PERMISSIVE COND AVAIL [] C-9 status light lit STM DUMP [] MODE SEL A-B TRN in STM PRESS STM DUMP INTERLOCK [] A TRN in ON [] B TRN in ON STM HDR PRESS [] PK 464 adjusted	12.3 Dump steam to atmosphere. 12.3.1 Direct counting room to perform FNP-0-CCP-645, MAIN STEAM ABNORMAL ENVIRONMENTAL RELEASE. 12.3.2 IF normal air available, THEN control atmospheric relief valves to dump steam from intact SGs, IF NOT, dump steam using FNP-1-SOP-62.0, EMERGENCY AIR SYSTEM. 1A(1B,1C) MS ATMOS REL VLV [] PC 3371A adjusted [] PC 3371B adjusted [] PC 3371C adjusted 12.3.3 IF no source of air available, THEN locally control SG atmospheric relief valves with handwheel to dump steam from intact SGs. (127 ft, AUX BLDG main steam valve room)

Intact SG	1A	1B	1C
Q1N11PCV	[] 3371A	[] 3371B	[] 3371C

Step	Action/Expected Response	Response NOT Obtained
12.4	Check all intact SG pressures - LESS THAN 100 psig.	12.4 Return to Step 11. OBSERVE CAUTION PRIOR TO STEP 11.

Step 12 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
12.5	Check at least two RCS hot leg temperatures - LESS THAN 350°F. RCS HOT LEG TEMP [] TR 413	12.5 Return to Step 11. OBSERVE CAUTION PRIOR TO STEP 11.
12.6	Stop SG pressure reduction. STM HDR PRESS [] PK 464 adjusted 1A(1B,1C) MS ATMOS REL VLV [] PC 3371A adjusted [] PC 3371B adjusted [] PC 3371C adjusted <u>OR</u> 1A(1B,1C) MS ATMOS REL VLV [] Q1N11PCV3371A closed [] Q1N11PCV3371B closed [] Q1N11PCV3371C closed	

****

<u>CAUTION</u>: Pump damage may occur if RHR pumps are operated on miniflow for longer than 3 hours with no CCW supplied to the RHR heat exchangers.

****

Step	Action/Expected Response	Response NOT Obtained
13	**Verify RHR pumps - STARTED.** RHR PUMP [] 1A amps > 0 [] 1B amps > 0	

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
14	**[CA] Check if SI accumulators should be isolated.**	

NOTE: Step 14.1 is a continuing action.

14.1 [CA] Check at least two RCS hot leg temperatures - LESS THAN 350°F.

RCS HOT LEG TEMP
[] TR 413

14.1 Perform the following.

14.1.1 <u>WHEN</u> at least two RCS hot leg temperatures are less than 350°F, <u>THEN</u> perform steps 14.2 and 14.3 to isolate accumulators.

14.1.2 Proceed to step 15. OBSERVE CAUTION PRIOR TO STEP 15.

14.2 Reset SI.

[] MLB-1 1-1 not lit (A TRN)
[] MLB-1 11-1 not lit (B TRN)

14.2 <u>IF</u> any train will <u>NOT</u> reset using the MCB SI RESET pushbuttons, <u>THEN</u> place the affected train S821 RESET switch to RESET. (SSPS TEST CAB.)

14.3 Close all SI accumulator discharge valves.

1A(1B,1C) ACCUM
DISCH ISO
[] Q1E21MOV8808A
[] Q1E21MOV8808B
[] Q1E21MOV8808C

14.3 Perform the following.

14.3.1 Vent any SI accumulator that cannot be isolated.

ACCUM
N2 VENT
[] HIK 936 open

SI ACCUM	1A	1B	1C
1A(1B,1C) ACCUM N2 SUPP/VT ISO Q1E21HV	[] 8875A open	[] 8875B open	[] 8875C open

14.3.2 <u>IF</u> an accumulator can <u>NOT</u> be isolated or vented, <u>THEN</u> consult the TSC staff to determine contingency actions.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: Core cooling may degrade during subsequent steps. FNP-1-CSF-0.2, CORE COOLING status tree should be closely monitored.

****

15 Stop all RCPs.

RCP
[] 1A
[] 1B
[] 1C

16 Reduce pressure in all intact SGs to atmospheric pressure.

16.1 Maintain RCS cold leg cooldown rate - LESS THAN 100°F IN ANY 60 MINUTE PERIOD.

16.2 IF condenser available, THEN dump steam to condenser from intact SGs.

BYP & PERMISSIVE
COND
AVAIL
[] C-9 status light lit

STM DUMP
[] MODE SEL A-B TRN in STM PRESS

STM DUMP
INTERLOCK
[] A TRN in ON
[] B TRN in ON

STM HDR
PRESS
[] PK 464

Response NOT Obtained:

16.2 Dump steam to atmosphere.

16.2.1 Direct counting room to perform FNP-0-CCP-645, MAIN STEAM ABNORMAL ENVIRONMENTAL RELEASE.

16.2.2 IF normal air available, THEN control atmospheric relief valves to dump steam from intact SGs, IF NOT, dump steam using FNP-1-SOP-62.0, EMERGENCY AIR SYSTEM.

1A(1B,1C) MS ATMOS
REL VLV
[] PC 3371A adjusted
[] PC 3371B adjusted
[] PC 3371C adjusted

Step 16 continued on next page.

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Step	Action/Expected Response	Response NOT Obtained

16.2.3 IF no source of air available,
THEN locally control SG atmospheric relief valves with handwheel to dump steam from intact SGs.
(127 ft, AUX BLDG main steam valve room)

Intact SG	1A	1B	1C
Q1N11PCV	[] 3371A	[] 3371B	[] 3371C

17 Verify any SI flow established.

- Verify any HHSI flow - GREATER THAN 0 gpm.

  A TRN
  HHSI FLOW
  [] FI 943

  HHSI
  B TRN RECIRC
  FLOW
  [] FI 940

  OR

- Verify any LHSI flow - GREATER THAN $1.5x10^3$ gpm.

  1A(1B)
  RHR HDR
  FLOW
  [] FI 605A
  [] FI 605B

17 Perform the following.

17.1 Continue efforts to establish SI flow.

- HHSI flow
- LHSI flow
- Any form of RCS injection.

17.2 Return to Step 16.

___Page Completed

Step	Action/Expected Response		Response NOT Obtained
18	**Check core cooling.** • REACTOR VESSEL LEVEL indication - GREATER THAN 0% UPPER PLENUM. • At least two RCS hot leg temperatures - LESS THAN 350°F. RCS HOT LEG TEMP [] TR 413	18	Return to step 16.
19	**Go to FNP-1-EEP-1, LOSS OF REACTOR OR SECONDARY COOLANT, step 14.**		

-END-

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

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.

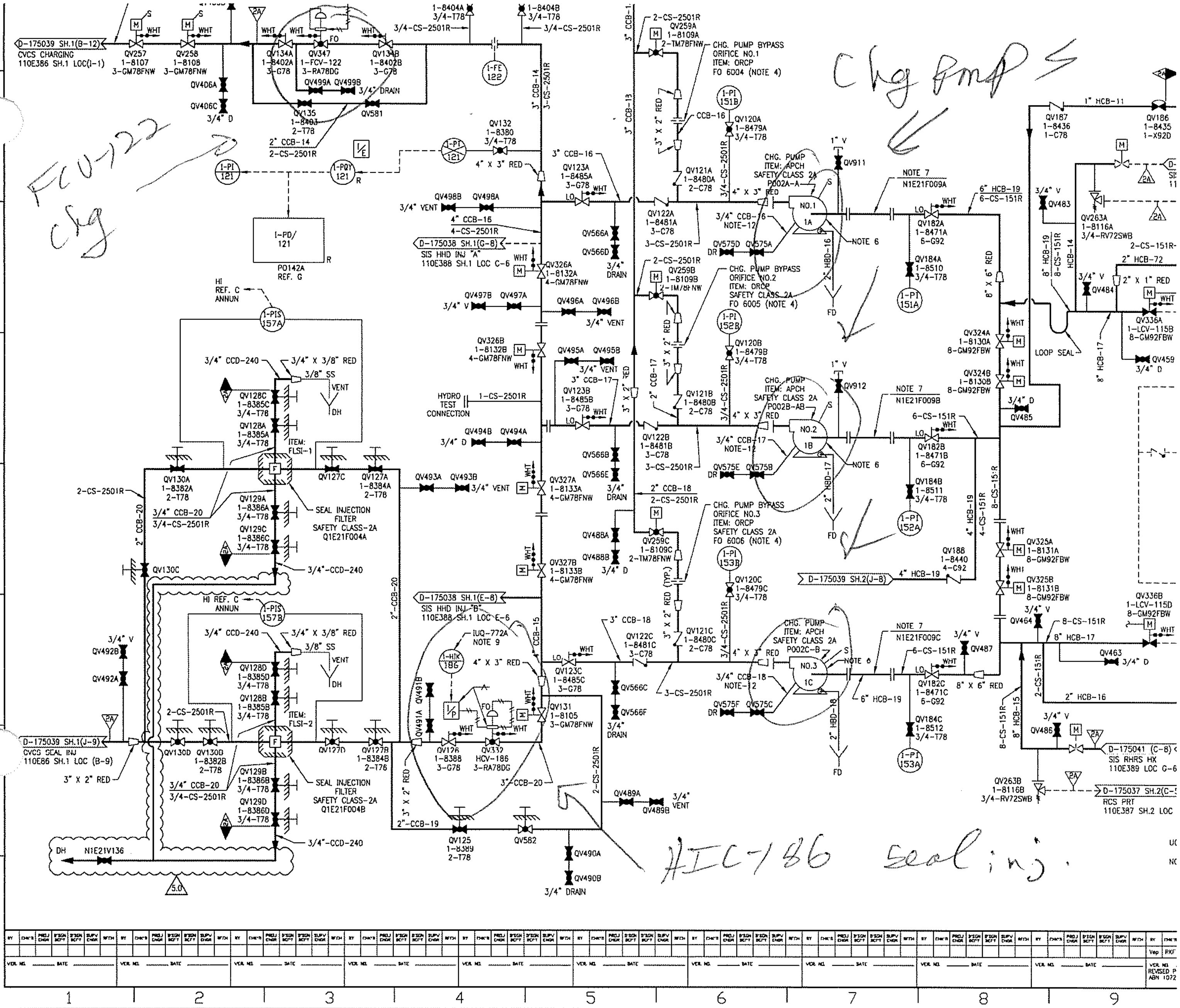
Objective:

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).

Date: 10/7/2009

Time : 11:08:41 AM



chg flow ↑ seal inj flow ↓

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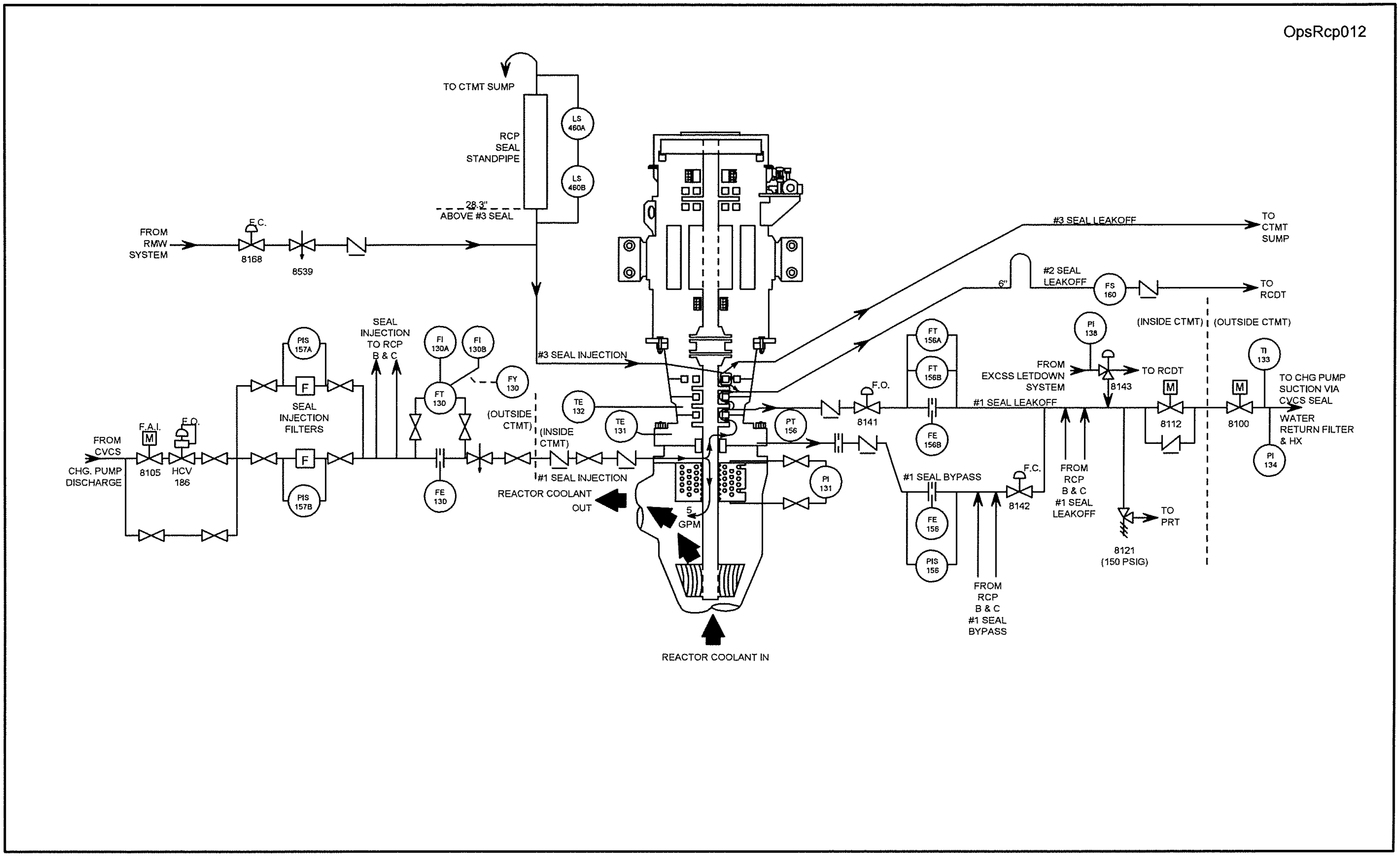


Figure 10 - RCP - Seal Injection System

6. 005A1.07 001/NEW/RO/C/A 2.5/3.1/N/N/3/CVR/Y

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 what the CR will **require** 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.
- Declare MOV-8889 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.
- Declare MOV-8889 INOPERABLE
- Repair or replacement of MOV8889.

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)

5.3 IF the RHR TO RCS HOT LEGS ISO, Q1E11MOV8889 Exercise Test is required, THEN perform the following:

5.3.1 Open the following valves:

_____ 5.3.1.1 RHR TO RCS HOT LEGS HDR TEST CONN ROOT, Q1E11V048A.

_____ 5.3.1.2 RHR TO RCS HOT LEGS HDR TEST CONN ROOT, Q1E11V048E.

_____ 5.3.2 Record the pressure on N1E11PI2262.

PRESSURE _______________ PSI.

_____ 5.3.3 IF the pressure indicated on N1E11PI2262 is less than OR equal to 550 psig, THEN perform the following:

_____ 5.3.3.1 IF required to isolate RHR TO RCS HOT LEGS ISO, Q1E11MOV8889 from an operating RHR train, THEN close RHR TO RCS HOT LEGS XCONN, Q1E11MOV8887A.

_____ 5.3.3.2 IF required to isolate RHR TO RCS HOT LEGS ISO, Q1E11MOV8889 from an operating RHR train, THEN close RHR TO RCS HOT LEGS XCONN, Q1E11MOV8887B.

_____ 5.3.3.3 Unlock and close disconnect switch Q1R18B036-B.

_____ A & C
Correct, 1st part

*5.3.3.4 Open RHR TO RCS HOT LEGS ISO Q1E11MOV8889 and record time required for valve opening in the ACTUAL STROKE TIME column of Table 1.

ACCEPTANCE CRITERIA:	**Stroke times are less than or equal to Maximum Allowable Times listed in Table 1.**

_____ *5.3.3.5 Close RHR TO RCS HOT LEGS ISO Q1E11MOV8889 and record time required for valve closing in the ACTUAL STROKE TIME column of Table 1.

ACCEPTANCE CRITERIA:	**Stroke times are less than or equal to the Maximum Allowable Times listed in Table 1.**

_____ 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:

incorrect B+D 1st part

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:

correct

1. Immediately retest the valve.
2. IF it is NOT possible to retest the valve, THEN declare the valve inoperable.

incorrect B+D 1st part

3. IF the valve is retested AND the second set of data is also outside the Acceptable Stroke Time Range, THEN perform the following:

correct A+B 2nd part

   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:

incorrect C+D second part

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.

7. 005K4.03 001/NEW/RO/C/A 2.9/3.2/N/N/4/CVR/Y

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%.
- RHR flow on FI-605A, 1A RHR HDR FLOW, is 3100 gpm.

At 1000:

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

At 1005:

- RHR system flow is stable.

At 1010:

- Instrument Air is lost to FCV-605A due to an air supply line break.

Which one of the following describes, **with no operator actions**:

1) RHR flow indicated on FI-605A at **1005**,

and

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

	At 1005 FI-605A, RHR HDR FLOW	At 1010 FCV-605A, 1A RHR HX BYP FLOW
A.	> 3100 gpm	Closed
B.	3100 gpm	Open
C.	> 3100 gpm	Open
D.✓	3100 gpm	Closed

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

5.0 NON-CRITICAL COMPONENT FUNCTIONAL DESIGN REQUIREMENTS

This section presents the functional design requirements for those components which are not critical to system function as defined in Section 3.0. These components enhance system performance, facilitate system maintenance or provide additional redundancy to components discussed in Section 3.0.

5.1 RHR HEAT EXCHANGER BYPASS FLOW CONTROL

Valves:

1-RHR-FCV-605A (Q1E11V033A)
1-RHR-FCV-605B (Q1E11V033B)

2-RHR-FCV-605A (Q2E11V033A)
2-RHR-FCV-605B (Q2E11V033B)

Flow Transmitters:

Q1E11FT605A	Q2E11FT605A
Q1E11FT605B	Q2E11FT605B

B&D 1st part correct

5.1.1 Basic Functions

These air-operated butterfly valves are provided to maintain a constant return flow to the RCS. The valves are normally closed and fail closed to ensure correct positioning during safety injection. An orifice type flow transmitter, FT-605A/B, is provided in each of the LHSI/RHR discharge headers, downstream of the RHR heat exchangers, to provide the necessary flow input for maintaining a constant RCS return flow during normal cooldown. (Reference 6.7.16)

A&D 2nd part correc

5.1.2 Functional Requirements

5.1.2.1 The valves must be designed for pressure and temperature conditions of 600 psig and 400°F. (References 6.4.12, 6.5.5, 6.5.6 and 6.5.13)

5.1.2.2 Valve thermal design transient requirements are summarized in Table T-14. (References 6.5.5 and 6.5.6)

5.1.2.3 Maximum allowable valve Cv equals 1050 at 600 full open. (Reference 6.4.12)

5.1.2.4 The design stroke time for opening or closing this air-operated valve is less than or equal to 10 seconds.

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

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

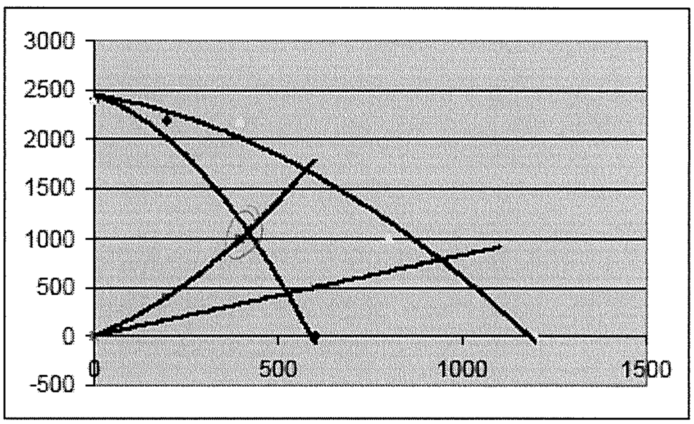
C✓ 450 gpm

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/CVR/Y

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 and vent the PRT to #7 WGDT, if necessary.

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 and vent the PRT to #7 WGDT, if 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

4.4 Reducing PRT Pressure

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. H2 < 4% and O2 < 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).

3.0 Precautions and Limitations

3.1 PRT temperature should not exceed 120°F during normal plant operation.

3.2 PRT nitrogen overpressure of 0.5 to 3 psig should be maintained to prevent formation of an explosive hydrogen - oxygen mixture.

3.3 PRT pressure should not exceed 6 psig during normal plant operation.

Correct D c 1st Parts

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.

3.6 At least one of the two reactor vessel head vent system paths, consisting of two valves in series powered from the Auxiliary Building DC Distribution System, shall be OPERABLE and closed at all times when in Modes 1-4. (TR 13.4.3)

3.7 While stroking the upstream valve (Q2B13SV2214A or Q2B13SV2214B), MCB closed indication could be momentarily lost on the downstream valve (Q2B13SV2213A or Q2B13SV2213B) due to minor water hammer. This phenomenon is common and documented for Plant Farley and for other plants, and has been evaluated to have no detrimental impact. {CR 2007103114}

3.8 While stroking the downstream valve (Q2B13SV2213A or Q2B13SV2213B), MCB closed indication could be momentarily lost on the upstream valve (Q2B13SV2214A or Q2B13SV2214B) due to rapid depressurization across the upstream valve. This phenomenon is documented for Plant Farley and has been evaluated to have no detrimental impact. {CR 2007103114}

4.3.2.18 IF RCDT was aligned to WHT, THEN perform the following:

1. Close RCDT DISCH TO WHT, Q2G21V009 (2-LWP-V-7137).
2. Open RCDT PUMP DISCH TO RHT ISO, Q2E21V315 (2 CVC V 8551).

4.3.2.19 Verify closed the following:

- PRT N2 SUPPLY ISO Q2B31HV8047
- PRT N2 SUPPLY ISO Q2B31HV8033
- Nitrogen supply from bulk storage to PRT valve 2-GWD-V-7920 (Q2G22V215) (121' PPR)

4.3.2.20 Align RCDT system as desired per FNP-2-SOP-50.0, LIQUID WASTE PROCESSING SYSTEM.

4.3.3 Gravity Draining PRT to WHT

NOTES:

- **This method of draining the PRT should only be used if RCDT pumps are inoperable.**
- **The bottom of the PRT sparger is 12" = ~500 gallons = ~5%. The sparger is a 12" perforated pipe that sits 12" off the bottom of the PRT. The top of the sparger is at 24" = ~1400 gallons = ~16%. The level doesn't have to be below the bottom of the sparger because the pipe is perforated on all sides, but it may be desirable.**

incorrect
B & C
2nd part S

4.3.3.1 Verify closed the following:

- PRT vent to GDT 2-RC-V-8025 (Q2B13V064), 121'.
- PRT vent to S/D Gas Decay Tanks 2-GWD-V-7935 (Q2G22V237), 83'

4.3.3.2 Verify closed nitrogen/hydrogen supply to S/D GDT's isolation valve 2-GWD-V-7849 (Q2G22V040).

4.3.3.3 Open nitrogen supply from bulk storage to PRT valve 2-GWD-V-7920 (Q2G22V215).

4.3.3.4 Verify PRT regulator 2-RC-PCV-8034 (Q2B13V042) adjusted to 3 psig.

4.3.3.5 Open the following PRT N2 SUPPLY ISO valves (MCB):

- Q2B31HV8047
- Q2B31HV8033

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

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.
- 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.

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

APPENDIX G
OPERATIONAL PERMISSIVES AND CONTROL INTERLOCKS

PERMISSIVES

Permissive	Source	Setpoint	Coincidence & Light Status	Function
1. P-4 Reactor Trip Interlock	Reactor Trip and Bypass Breakers	Breakers Open	RTA & BYA Open or RTB & BYB Open No Light	Prevents a rapid cooldown of primary system after a reactor trip. 1. Trips Turbine 2. Trips F.W. Reg Valves on Low T_{avg} 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.
2. P-6 I/R Power Escalation Permissive	NIS 35 and 36	10^{-10} amps	1/2 > Setpoint Lit > Setpoint Permission to Block Source Range	Allows power escalation into the IR by turning <u>Both</u> Train A & B Source Range Block switches to Block. Above setpoint 1. Blocks SR Hi ∅ Reactor Trip 2. Turns off Hi volt to SR Instr. Below setpoint Auto reinstates Hi volt to SR Instr.

incorrect A & C 2nd part

incorrect A & B 1st part

correct 4th parts
D Both
C 1st part
B 2nd part

11. 008AA2.26 001/NEW/RO/C/A 3.1/3.4/N/N/2/CVR/Y

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.
- R-2, 7, 11 and 12 are in alarm.

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 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

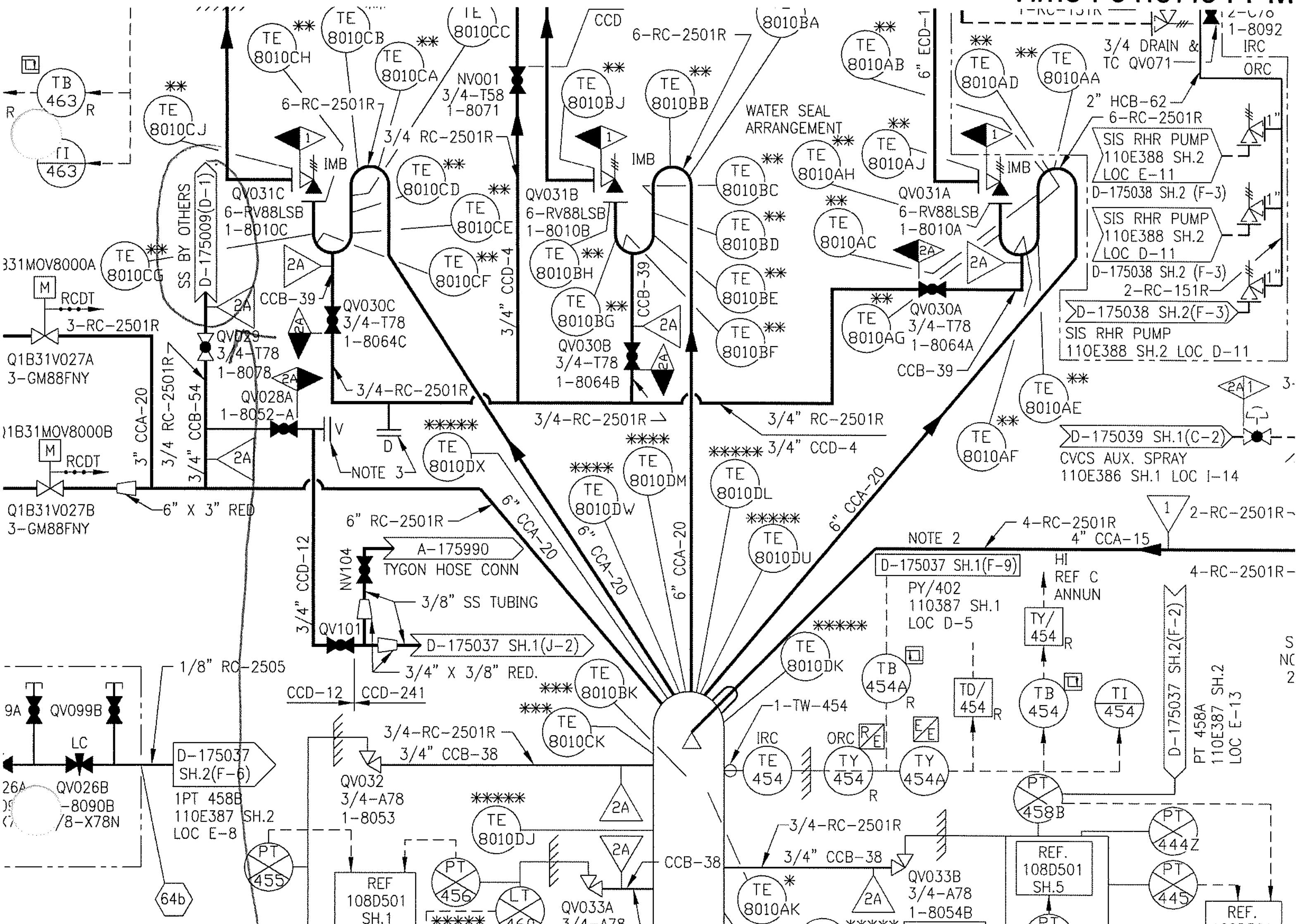
Previous NRC exam history if any:

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

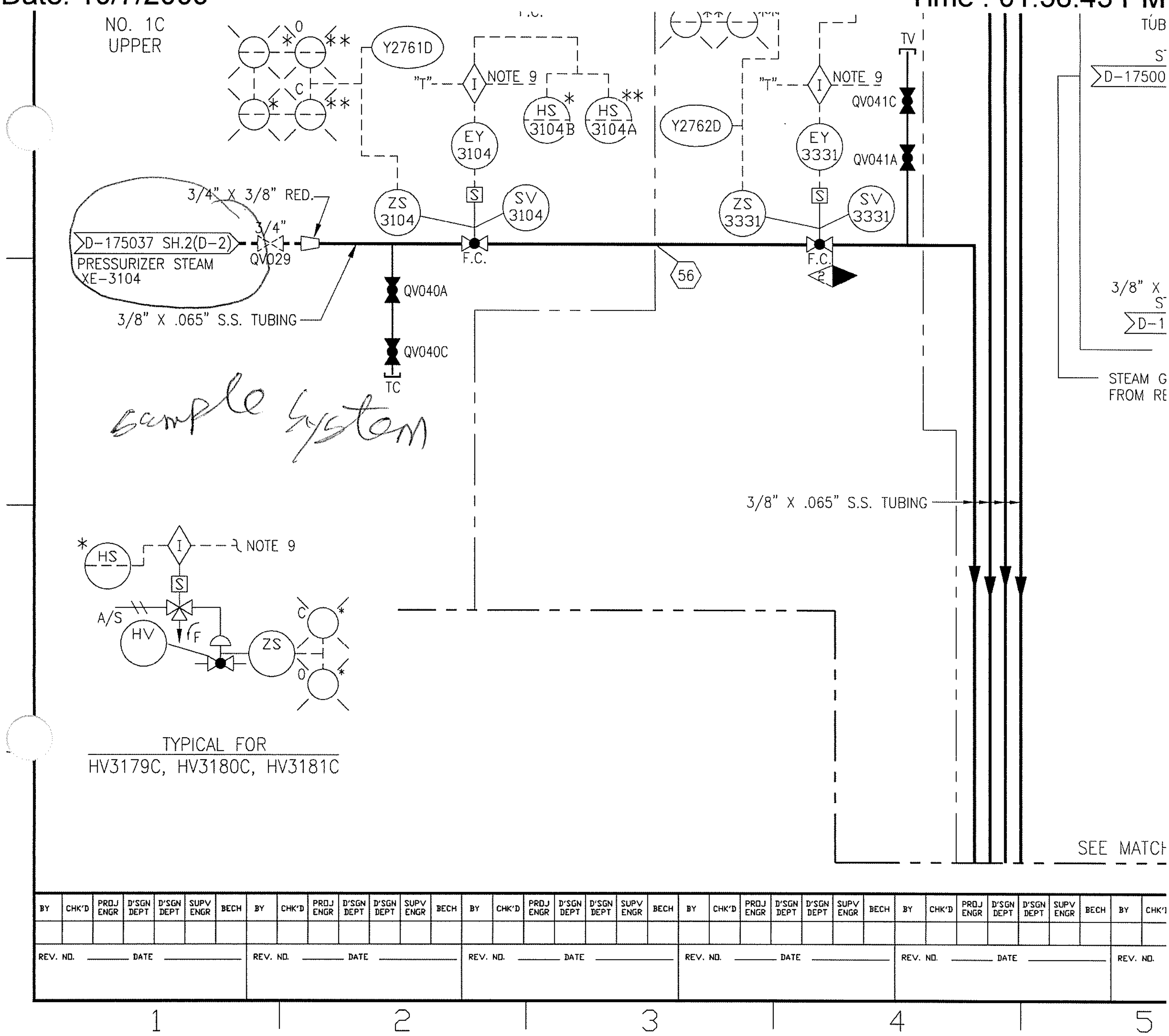
Match justification: This question presents a scenario with symptoms given of a steam space break. The indications have similarities to either a PORV or code safety leaking by OR another leakage path other than the PORV or code safeties, and some differences. In this case the applicant must correctly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



{przr
{steam space to sample system


NO. 1C
UPPER
Y2761D
"T"
NOTE 9
HS 3104B
HS 3104A
EY 3104
Y2762D
EY 3331
TV
QV041C
QV041A
D-17500
3/4" X 3/8" RED.
3/4"
D-175037 SH.2(D-2)
PRESSURIZER STEAM
XE-3104
QV029
ZS 3104
SV 3104
F.C.
ZS 3331
SV 3331
56
QV040A
QV040C
TC
3/8" X .065" S.S. TUBING
sample system
STEAM G
FROM RE
HS
HV
ZS
A/S
TYPICAL FOR
HV3179C, HV3180C, HV3181C
SEE MATCH
BY
CHK'D
PROJ ENGR
D'SGN DEPT
SUPV ENGR
BECH
REV. NO.
DATE
1
2
3
4
5

12. 008K3.01 001/FNP BANK/RO/C/A 3.4/3.5/N/N/3/CVR/Y

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 the required action?

A. • Higher Letdown temperature.

• Isolate Letdown and place Excess Letdown in service.

B✓ • Higher Letdown temperature.

• Place TK-144 in manual and adjust flow.

C. • Lower Letdown temperature.

• Isolate Letdown and place Excess Letdown in service.

D. • Lower Letdown temperature.

• Place TK-144 in manual and adjust flow.

A - Incorrect. The first part is correct (see B). The second part is incorrect (see B). Plausible, since "IF letdown flow cannot be reduced [in manual control], THEN this would be correct, but it is not the first strategy prior to manual control of temperature.

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. The first alarm to come in would be: ARP-1.4, DF1, LTDN TO DEMIN DIVERTED TEMP HI [at 135°F] , and that ARP states to:

"Take manual control of LTDN HX Outlet Temp TK-144 and attempt to increase CCW flow to the Letdown Heat Exchanger."

The OPS "Skill of the craft" policy also states that this is appropriate prior to attempting anything else. The ARP also states:

"6. Adjust charging or letdown flow as required to reduce the letdown flow temperature. AND: 5. IF letdown temperature can NOT be reduced, THEN close LTDN ORIF ISO 45 (60) GPM Q1E21HV8149A, B, and C."

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%. Also, if the valve did go open, it would cause boron absorption in the Mixed Bed demineralizer due to the cooler Letdown temperature. The second part is incorrect, but plausible. If the TK-144 valve could not be controlled in Manual, this would correct.

D - Incorrect. The first part is incorrect (see C). The second part is correct. Per OPS "skill of the craft" policy, placing a controller in MAN from AUTO when necessary to control parameters is always appropriate. One skill of the craft item, which may be performed as necessary without procedure guidance, is: "?? Adjusting pots and controllers, including transfer between AUTO and MANUAL, to maintain parameters within log spec or procedural specs." The other procedure guidance for a Low Letdown temperature is the requirement to maintain Reactor power <100% at all times.

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

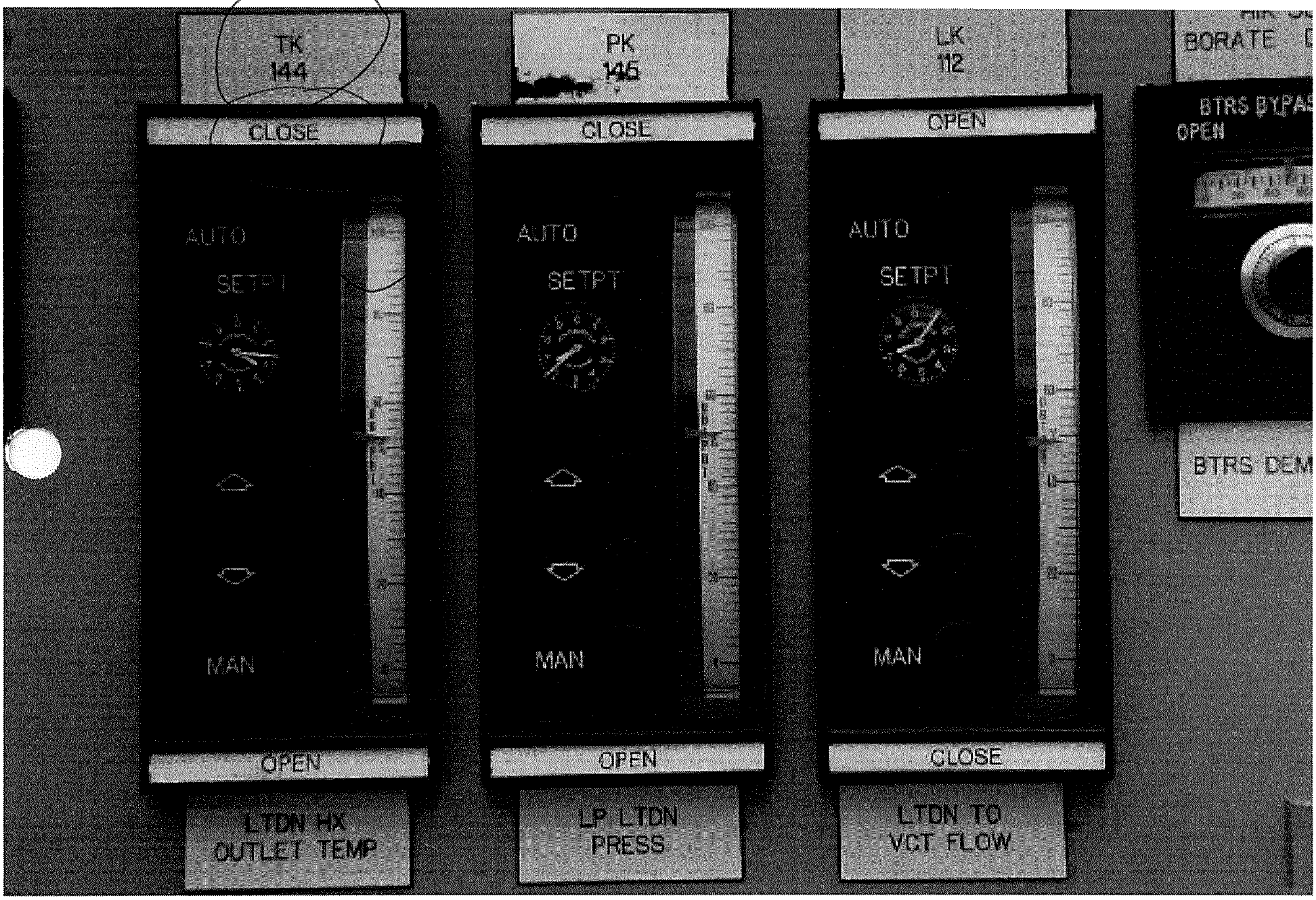
Match justification: This question presents a specific type of malfunction of the CCW system (Failure of the CCW to the Letdown HX control valve controller). To answer this question correctly, knowledge of the effect of this malfunction of the CCW system on the Load (Letdown) cooled by CCW is required. 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:

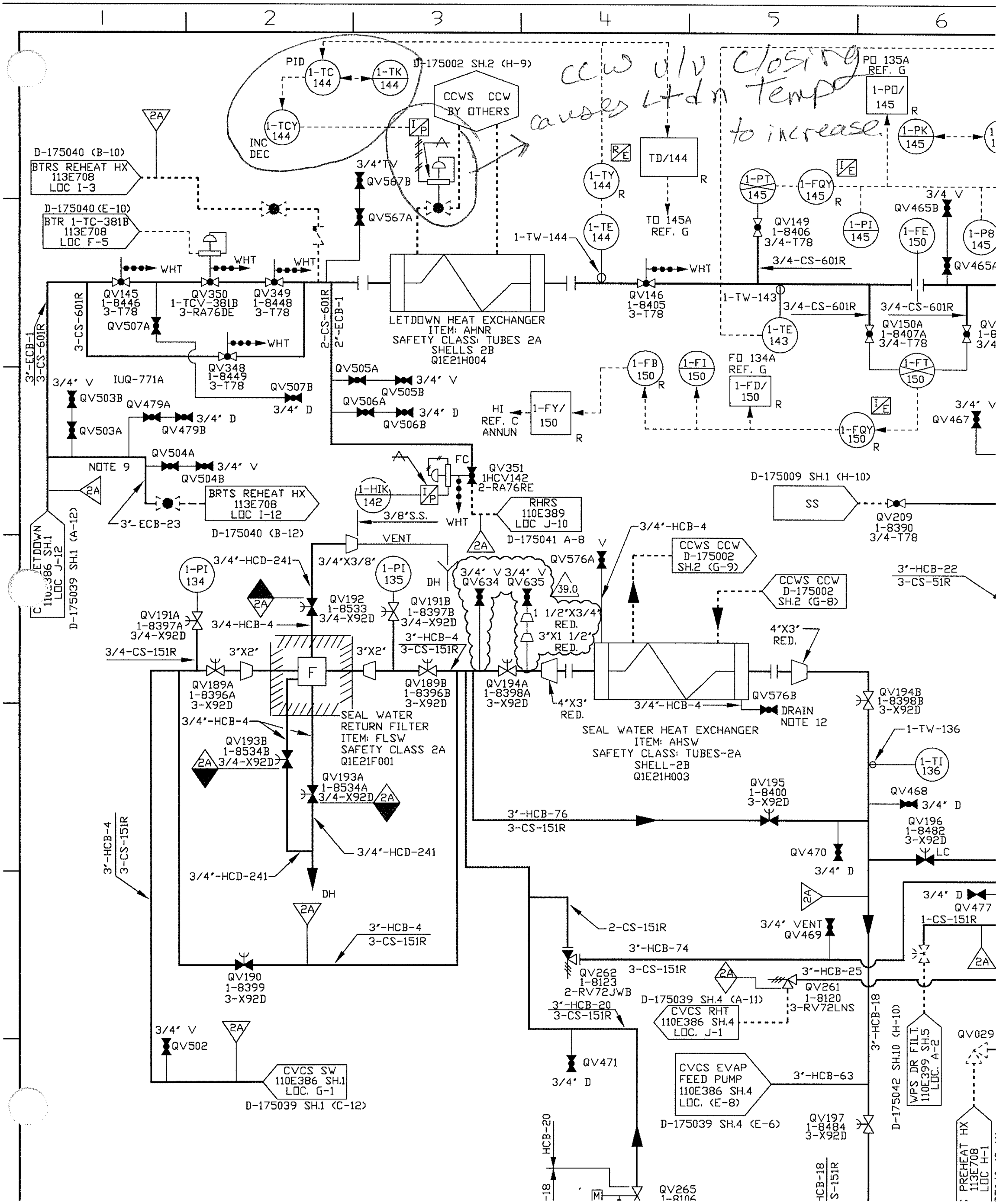
7. **DEFINE AND EVALUATE** the operational implications of normal / abnormal plant or equipment conditions associated with the safe operation of the CCW System components and equipment, to include the following (OPS-40204A07):
   - Normal control methods
   - Abnormal and Emergency Control Methods
   - 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

CLW
v/v closed at
100% Demand

correct A&B 1st parts
incorrect C&D 2nd parts



MCB TK-144
picture


ccw v/v closing causes Ltdn temp to increase.
PID
1-TC 144
1-TK 144
1-TCY 144
INC DEC
D-175002 SH.2 (H-9)
CCWS CCW BY OTHERS
D-175040 (B-10)
BTRS REHEAT HX 113E708 LOC I-3
D-175040 (E-10)
BTR 1-TC-381B 113E708 LOC F-5
QV567B
QV567A
QV145 1-8446 3-T78
QV350 1-TCV-381B 3-RA76DE
QV349 1-8448 3-T78
QV507A
QV348 1-8449 3-T78
LETDOWN HEAT EXCHANGER
ITEM: AHNR
SAFETY CLASS: TUBES 2A
SHELLS 2B
Q1E21H004
1-TY 144
1-TE 144
1-TW-144
TD/144
TO 145A REF. G
PD 135A REF. G
1-PD/ 145
1-PK 145
1-PT 145
1-FQY 145
1-PI 145
1-FE 150
QV149 1-8406 3/4-T78
QV146 1-8405 3-T78
3/4-CS-601R
1-TW-143
1-TE 143
QV150A 1-8407A 3/4-T78
1-FT 150
1-FB 150
1-FI 150
FD 134A REF. G
1-FD/ 150
1-FY/ 150
HI REF. C ANNUN
1-FQY 150
QV465B
QV467
QV505A
QV505B
QV506A
QV506B
QV507B
QV503B
QV503A
QV479A
QV479B
QV504A
QV504B
IUQ-771A
NOTE 9
BRTS REHEAT HX 113E708 LOC I-12
D-175040 (B-12)
3'-ECB-23
3'-ECB-1
3-CS-601R
2-CS-601R
2'-ECB-1
QV351 1HCV142 2-RA76RE
1-HIK 142
3/8'S.S.
WHT
RHRS 110E389 LOC J-10
D-175041 A-8
D-175009 SH.1 (H-10)
SS
QV209 1-8390 3/4-T78
LETDOWN 110E386 SH.1 LOC J-12
D-175039 SH.1 (A-12)
1-PI 134
1-PI 135
VENT
DH
QV191A 1-8397A 3/4-X92D
QV192 1-8533 3/4-X92D
QV191B 1-8397B 3/4-X92D
3/4-HCB-4
3/4'-HCD-241
QV634
QV635
QV576A
39.0
1 1/2'X3/4' RED.
3'X1 1/2' RED.
3/4'-HCB-4
CCWS CCW D-175002 SH.2 (G-9)
CCWS CCW D-175002 SH.2 (G-8)
3'-HCB-22
3-CS-51R
4'X3' RED.
3/4-CS-151R
QV189A 1-8396A 3-X92D
QV189B 1-8396B 3-X92D
QV194A 1-8398A 3-X92D
QV194B 1-8398B 3-X92D
3'-HCB-4
3'-CS-151R
SEAL WATER RETURN FILTER
ITEM: FLSW
SAFETY CLASS 2A
Q1E21F001
QV193B 1-8534B 3/4-X92D
QV193A 1-8534A 3/4-X92D
QV576B
DRAIN NOTE 12
SEAL WATER HEAT EXCHANGER
ITEM: AHSW
SAFETY CLASS: TUBES-2A
SHELL-2B
Q1E21H003
1-TW-136
1-TI 136
QV195 1-8400 3-X92D
QV468
QV196 1-8482 3-X92D
3'-HCB-76
3-CS-151R
QV470
2-CS-151R
QV190 1-8399 3-X92D
QV502
CVCS SW 110E386 SH.1 LOC. G-1
D-175039 SH.1 (C-12)
QV262 1-8123 2-RV72JWB
3'-HCB-74
3'-HCB-20
3-CS-151R
QV261 1-8120 3-RV72LNS
D-175039 SH.4 (A-11)
CVCS RHT 110E386 SH.4 LOC. J-1
QV469
QV477
1-CS-151R
3'-HCB-25
3'-HCB-18
D-175042 SH.10 (H-10)
WPS DR FILT. 110E399 SH.5 LOC. A-2
QV029
QV471
CVCS EVAP FEED PUMP 110E386 SH.4 LOC. (E-8)
D-175039 SH.4 (E-6)
3'-HCB-63
QV197 1-8484 3-X92D
PREHEAT HX 113E708 LOC H-1
QV265

LOCATION DF1

SETPOINT: 135°F

F1	LTDN TO DEMIN DIVERTED-TEMP HI

ORIGIN: 1-TY-143X Auxiliary Relay actuated by Temperature Bistable (N1E21TB143)

PROBABLE CAUSE

1. Low or Loss of CCW Flow to the Letdown Heat Exchanger.
2. Letdown Flow greater than Charging Flow.

AUTOMATIC ACTION

1. Letdown High Temperature Divert Valve Q1E21TCV143 diverts Letdown Flow to the VCT. {CMT 0008644}

OPERATOR ACTION

1. Verify Q1E21TCV143 has diverted letdown flow to VCT to bypass demins
2. Monitor charging and letdown flows and temperatures.

Correct B & D 2nd part

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.

incorrect A & C second part

5. 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.**

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

References: A-177100, Sh. 206; D-175039, Sh.2; D-177091; D-177375; U-175997; PLS Document

13. 008K4.09 001/MOD/RO/C/A - 2.7/2.9//N/2/CVR/Y

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)

FSD A-181000
Appendix A
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

APPENDIX A

CCW PUMP ALIGNMENTS

1.0 **PURPOSE**

The following provides the functional requirements for the various CCW pump alignments. There are no new requirements in this Appendix in addition to those in the FSD text. The purpose of this Appendix is to extract the requirements from the FSD which pertain to pump alignments.

2.0 **INTRODUCTION**

CCW pumps C and A shall be train dedicated and aligned to the 4 kV buses F (train A) and G (train B), respectively. CCW motor pump B, when available, shall be aligned to either of the vital 4 kV buses, F (train A) or G (Train B), corresponding to whichever train the B pump has been valved into. The B pump shall be physically valved into the train from which it is set to receive its power.

3.0 **CCW PUMP ALIGNMENTS**

3.1 **ALL THREE CCW PUMPS OPERABLE**

3.1.1 **One Train-Dedicated Pump On-Service, Swing Pump B in Standby (Aligned to the On-Service Train), One Train-Dedicated Pump Off-Service (Normal System Alignment)**

3.1.1.1 **On-Service Pump Trips**

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

b Once the (swing) pump B becomes the on-service pump, the breaker of the pump with the fault is required to be racked out immediately, or the lock-out relay shall not be reset by the operator, in order to allow the proper operation of swing pump B in the event of an LOSP (Reference 6.7.11).

c Deleted (Reference 6.7.057)

3.1.1.2 **LOSP**

In the event of a LOSP with or without a SIAS, the two train-dedicated pumps, C and A, shall be automatically sequenced onto the diesel generators. Swing pump B shall continue to provide backup in the event of a trip of the dedicated pump in the train to which swing pump B is aligned (Reference 6.7.11).

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).

b. Deleted (Reference 6.7.057)

3.1.2.2 LOSP

Correct A →

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

B incorrect w/ mixup on train relationship of 1C CCW pmp

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).

C incorrect w/ mixup between LOSP & SI seq sequencer operation are required between pumps 1B and 1C and between pumps 1C and 1D to separate trains for fire protection purposes. (Reference 6.1.010)

3.1.4.4 Each pair of train oriented Service Water pumps along with the swing SW pump shall be provided with a minimum flow bypass valve (See Section 3.4 for the required flow rates) to recirculate service water to the service water intake structure wet pit. (References 6.4.014 and 6.4.018)

3.1.4.5 Each Service Water pump motor shall be equipped with bearing temperature monitoring devices. (Reference 6.5.003)

3.1.5 I & C Requirements

3.1.5.1 The Service Water pumps shall be automatically started by a signal from the LOSP or ESS sequencer. The Service Water swing pump shall be automatically started by a signal from the LOSP or ESS sequencer when in service replacing one of the train oriented pumps. (References 6.7.039 and 6.1.009)

D ⇒ incorrect SW swing pmp starts when selected even if all others are running.

3.1.5.2 Key interlocking of power supply breakers, disconnect switches, and SW header cross-connect valves shall be used to ensure alignment of the Service Water swing pump to one train only. (Reference 6.1.006)

3.1.5.3 Annunciation shall be provided in the Control Room to alert the operator when a Service Water pump breaker trips. (References 6.4.104 through 6.4.115, 6.1.007)

3.1.5.4 Monitor lights shall be provided in the control room to allow quick verification of the status of Service Water Pumps A, B, D, and E following a safety injection signal. (Reference 6.7.124)

3.1.5.5 The Service Water (SW) System shall have redundant level instrumentation to monitor and control the Storage Pond/Service Water Pump Wet Pit level. (Reference 6.4.075)

Service	Train	TPNS Nos.
SW Pump Wet Pit	A	N1(2)P25LI4066A
SW Pump Wet Pit	B	N1(2)P25LI4066B

1. CCW-40204A07 013/HLT//MEM 2.7/2.9/008K4.09////008A3.08

Bonk

008K4.09

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

- "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.

A Safety Injection occurs at this time.

Which one of the following combinations of CCW pumps will be running following the operation of the ESF sequencers?
(Assume no operator action is taken)

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

008K4.09
008 Component Cooling Water System (CCWS)
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

A. Incorrect, 1B CCW pump is running on A train therefore, 1C CCW pump will not start. This is plausible since on a load shed and LOSP sequencer operation this would occur.

B. Incorrect, 1B CCW pump is running on A train therefore, 1C CCW pump will not start, 1A pump will start on B train. This is plausible since this is the opposite train and CCW has backward logic.

C. Correct, 1B CCW pump is running on A train therefore, 1C CCW pump will not start. 1A CCW pump is running and does not recieve trip signal and it remains running.

D. Incorrect, 1B CCW pump is running on A train and 1A CCW pump is running and would recieve a start signal from B train sequencer. The 1C CCW pump will not start since the 1B CCW pump is running. This is plausible since the SW pumps would have all pumps running in this situation.

Lesson plan ops-52102G
The three CCW pumps (Figures 6 & 7) can be operated from the MCB or locally at the HSP by a three-position handswitch (STOP/AUTO/START, spring return to AUTO). A two-position selector switch (LOCAL/REMOTE) at the HSPs determines which station has control of the pumps. The dedicated pumps (A and C) will automatically start on receiving an "S"-signal or a loss of offsite power (LOSP) signal, provided that the local remote selector switch is in the REMOTE position and the MCB handswitch is in the AUTO position. The swing pump (B) acts as a backup for the dedicated pumps by being mechanically and electrically aligned to the same train as one or the other of the dedicated pumps. The swing pump will automatically start when: (1) the dedicated pump it is backing up trips on overload; (2) the selector switch is in the REMOTE position; (3) the MCB hand switch is in the AUTO position.

The swing pump also receives start signals from the safety injection (SI) sequencer and the **LOSP sequencer.** However, it will only start if the selector switch is in **REMOTE**, the MCB switch is in AUTO, **the dedicated pump A or C (depending on which train it is lined up to) has tripped on overload**, or its supply breaker has been racked out.

FSD A-181000
3.1.5.2 During normal plant operation, with all pumps operational, if the operating pump power supply breaker trips, the standby pump shall automatically start and supply CCW to the CCW heat exchanger in operation. The breaker of the pump with the fault shall be racked out immediately, or the lockout relay shall not be reset by the operator in order to allow the proper operation of the standby pump in the event of a loss of offsite power (LOSP). The CCW pump overload trip shall be alarmed in the MCR to alert the operator
(References 6.1.01, 6.4.15, 6.4.16, 6.4.17).

2008 NRC exam
Technical Reference: FSD A-181000

Learning Objective: 40204A07
List the automatic actions associated with the Component Cooling Water System components and equipment during normal and abnormal operations including (OPS40204A07):

- Normal 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

also 52102G02

Comments: This meets the KA since it tests the standby feature of the standby pump for the train it is aligned to. This is the standby feature for the main pump (ie., 1C or 1A CCW pump).

Our SW pumps do not have a feature where the standby pump looks to see if the other pump is running before starting or not starting the other pump in that train for an SI signal. In that case there would be 5 SW pumps running. For CCW, if the swing pump is running, then the other pump in that train will not start on the SI signal.

All distracters are plausible since our trains are not set up in a logical way and C CCW pump is A train and A CCW pump is B Train. Most other components are configured correctly and differently.

Had to change the stem to take into account the new CCW and charging pump line up.

FNP BANK: CCW-52102G02 05

14. 009EG2.1.23 001/NEW/RO/C/A 4.3/4.4/N/N/4/CVR/VER 5 EDITORIAL

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 must be adjusted to raise Przr level.

B. Place the SI ACTUATION switch to ACTUATE.

C. FK-122 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)

Step	Action/Expected Response	Response NOT Obtained

NOTE: During a LOCA, a full or rising pressurizer level may indicate a steam space LOCA exists. In that event, step 20.2, RNO provides guidance if the RCS must be operated water solid, and charging used to maintain subcooling instead of pressurizer level.

20.2 [CA] Maintain pressurizer level greater than 25%{50%}.

20.2.1 IF charging flow path aligned, THEN control charging flow.

CHG FLOW
[] FK 122 adjusted

A incorrect

B incorrect

20.2 IF solid plant operation required, THEN perform the following.

a) Maintain SUBCOOLED MARGIN MONITOR indication greater than 26°F{55°F}.

b) Control charging flow to stabilize subcooling at existing value.

CHG FLOW
[] FK 122 adjusted

****

CAUTION: To prevent potential seal damage, neither seal injection nor CCW cooling should be restored to a RCP which has lost both seal injection and CCW cooling.

****

___21 **[CA] Check if RCP(s) should be reconfigured to optimize RCS flow and pressurizer spray performance.**

21.1 Check RCP 1B - STOPPED.

21.1 Perform the following.

21.1.1 Verify RCPs 1A AND 1C - STOPPED.

RCP
[] 1A
[] 1C

21.1.2 Proceed to step 22. OBSERVE CAUTION AND NOTE PRIOR TO STEP 22.

Step 21 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

correct C

1 Monitor SI reinitiation criteria following HHSI isolation.

1.1	Greater than 16°F {45°F} ≥ 24% subcooled in ~~CETC mode~~ and PRZR level above 13% {43%}. 28%	1.1 Establish HHSI flow, and start additional CHG PUMPs as required using ATTACHMENT 5, RE-ESTABLISHING HHSI FLOW. ; incorrect D

2 Monitor FNP-1-EEP-2 and FNP-1-EEP-3 branch criteria.

2.1	No SG pressure falling in an uncontrolled manner or less than 50 psig.	2.1 <u>IF</u> affected SG <u>NOT</u> previously isolated, <u>THEN</u> go to FNP-1-EEP-2.
2.2	No high secondary radiation or SG level rising uncontrolled.	2.2 Establish HHSI flow, and start additional CHG PUMPs as required using ATTACHMENT 5, RE-ESTABLISHING HHSI FLOW <u>THEN</u> go to FNP-1-EEP-3.

3 Monitor switchover criteria.

3.1	RWST level greater than 12.5 ft.	3.1 Go to FNP-1-ESP-1.3.
3.2	CST level greater than 5.3 ft.	3.2 Align AFW pumps suction to SW using FNP-1-SOP-22.0.

4 Monitor charging miniflow criteria (during SI).

4.1	RCS pressure less than 1900 psig.	4.1 Verify miniflow valves open.
4.2	RCS pressure greater than 1300 psig.	4.2 Verify miniflow valves closed.

5 Monitor adverse containment criteria.

5.1	CTMT pressure less than 4 psig and radiation less than 10^5 R/hr.	5.1 Utilize bracketed adverse CTMT condition numbers.

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

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 valves **with no operator actions**?

All PRZR Backup Heaters will be ____(1)____

and

BOTH PRZR Spray Valves will be ____(2)____.

	(1) Backup Heaters	(2) Spray Valves
A.✓	ON	Opening
B.	ON	Closing
C.	OFF	Opening
D.	OFF	Closing

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).

LOCATION HA2

SETPOINT: 5% of Span above Level Program

ORIGIN: Level Bistable LB-459D from Level Transmitter LT-459 or LT-461 and TY-408 median TAVG.

A2
PRZR LVL DEV HI B/U HTRS ON

PROBABLE CAUSE

1. Pressurizer Level Instrument or Control System malfunction.
2. Plant Transient while in manual rod control.
3. Rod Control System malfunction.
4. Charging or Letdown System malfunction.

D is correct

AUTOMATIC ACTION

C 1st part correct

1. Pressurizer Backup Heaters energize.

OPERATOR ACTION

1. Place turbine load on HOLD.
2. Check pressurizer level indications and determine the actual level deviation.
3. <u>IF</u> an instrument failure has occurred, <u>THEN</u> go to FNP-1-AOP-100, INSTRUMENT MALFUNCTION.
4. Ensure that the pressurizer backup heaters are energized.
5. <u>IF</u> required, <u>THEN</u> take manual control of CHG FLOW FK 122 and decrease charging flow to return pressurizer level to the program band.
6. Determine the cause of the level deviation by checking:
   - 5.1 Charging flow
   - 5.2 Letdown flow
   - 5.3 BTRS flow
   - 5.4 Charging pump status
7. <u>IF</u> the alarm was caused by a plant transient, <u>THEN</u> control the transient and return Pressurizer Level to normal.
8. <u>IF</u> a charging <u>OR</u> letdown system malfunction exists, <u>THEN</u> go to FNP-1-AOP-16.0, CVCS MALFUNCTION.

References: A-177100, Sh. 357; U-260610; D-177109; D-177111; D-177112; D-177113; U-266647 PLS Document; Technical Specifications

16. 011K 5.05 001/NEW/RO/C/A 2.8/3.1/N/N/2/CVR/Y

Unit 1 is at 100% power, and has experienced a Pressurizer Level Control Malfunction due to the controlling pressurizer level transmitter failing.

The following conditions exist:

- PRZR LVL CONT CH, LS/459Z, is in the "I/II LT459/60" position.
- Tavg is 573.0°F.
- AOP-100, Instrumentation Malfunction, is in progress.
- Przr level is 40% and rising.
- Przr level control is in Manual.
- 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

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).

(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).

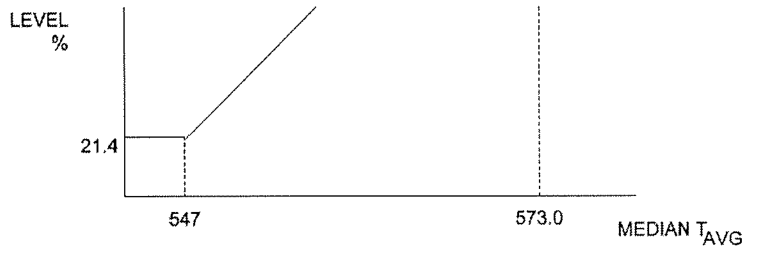
At 100°F, the specific volume of water is 0.016130 ft3/lbm per the steam tables, and it would expand to 0.02657 ft3/lbm @ pressurizer temperature of 648°F. The charging water of 56 gals/min will expand to 93.5 gals/min at PRZR temperature (which is 1% in the Pressurizer per Tank Curve 42). In approximately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

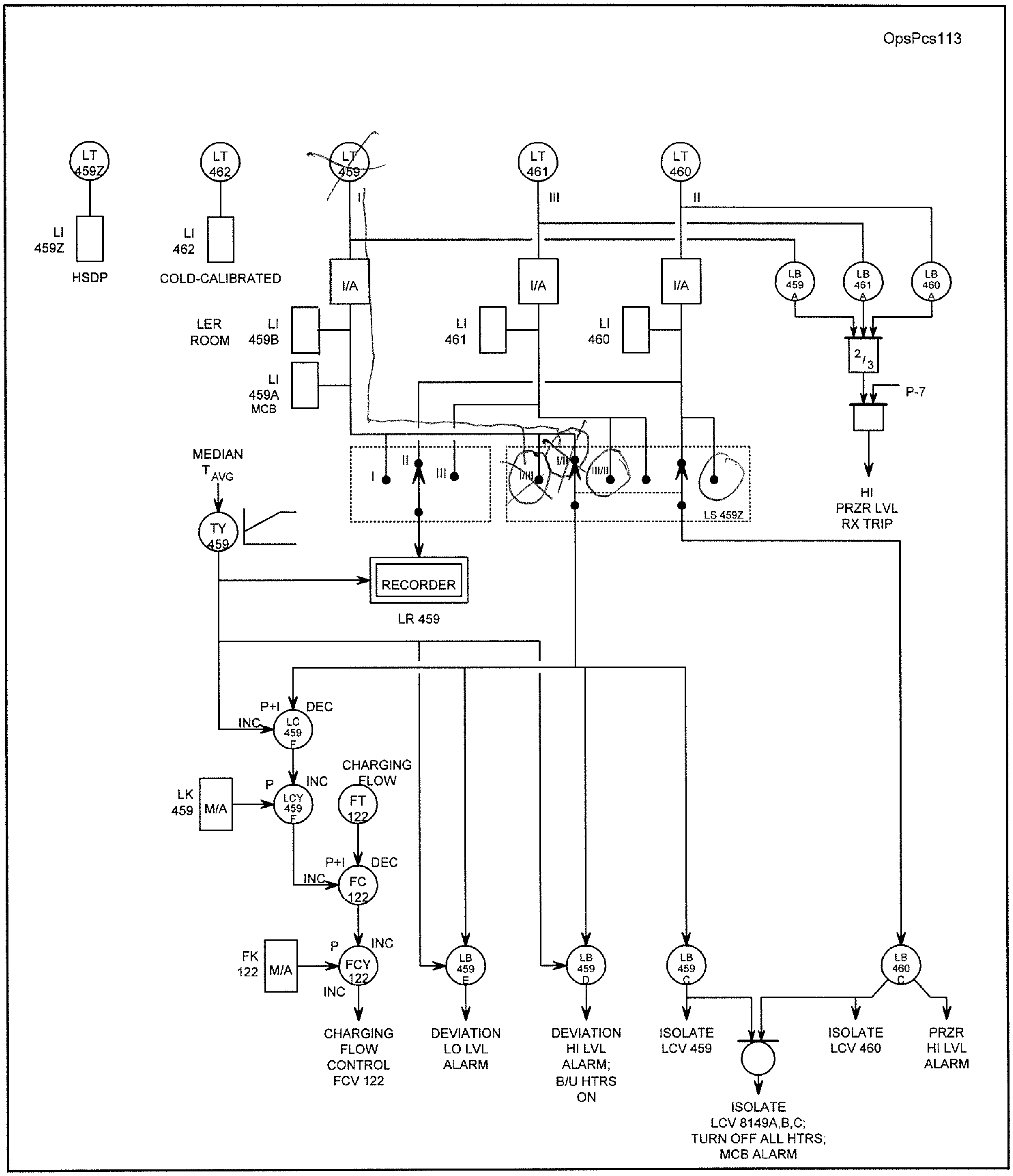
Objective:

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

SECTION 1.2

Figure 1



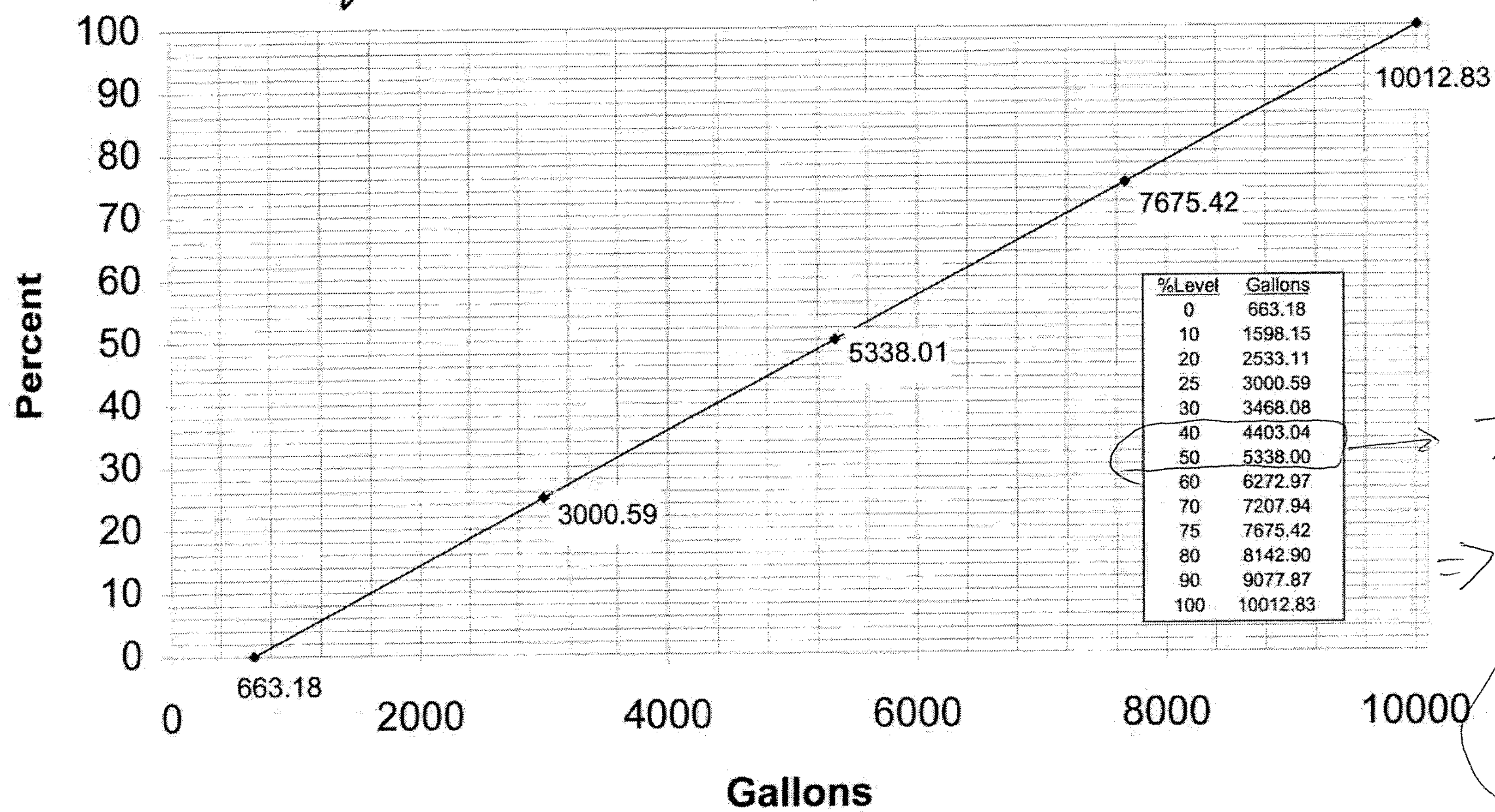
PRESSURIZER LEVEL PROTECTION AND CONTROL

Pressurizer (Q1B31K001) Capacity vs. % Level

Unit 1 Volume II Curve 42

Hot Calibrated*

Approved [signature] ES Manager   Date 1/29/03   Rev. No. 3.0



%Level	Gallons
0	663.18
10	1598.15
20	2533.11
25	3000.59
30	3468.08
40	4403.04
50	5338.00
60	6272.97
70	7207.94
75	7675.42
80	8142.90
90	9077.87
100	10012.83

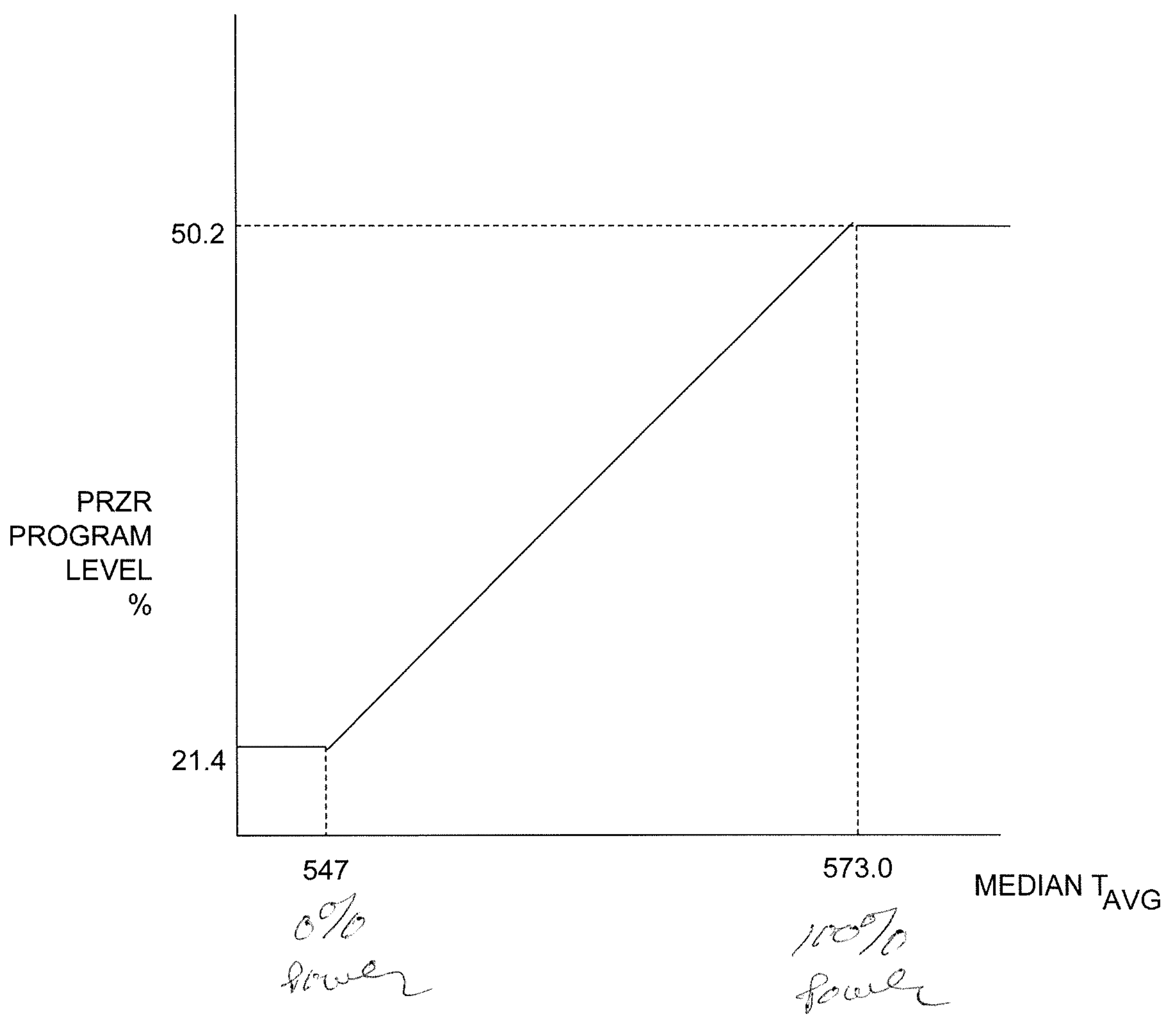
5338 gals
− 4403.04 gal
934.96 ga

=> 935 gals / 10%

93.5 gals / 1%

*Based on saturated liquid temperature at 2235 psig

Calculation Ref. No. SJ-00-2290-003



Pressurizer Program Level

Figure 8

17. 012K2.01 001/FNP BANK/RO/C/A 3.3/3.7/N/N/3/CVR/Y

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.

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

3.2.7 Interface Requirements

The STC shall interface with the SSPS and shall be supplied by qualified Class 1E power from the 120 Vac vital power cabinets. (References 6.7.014, 6.4.059, 6.4.060, 6.4.084)

3.3 REACTOR TRIP SWITCHGEAR

TPNS Nos.	Service
QC11E004A-AB	(RTA, BYB)
QC11E004B-AB	(RTB, BYA)

3.3.1 Basic Functions

The reactor trip switchgear functions to switch power to or remove power from the control rod positioning equipment. The switchgear opens the reactor trip and bypass breakers A and B on reactor trip causing the control rods to fall by gravity into the reactor core.

3.3.2 Functional Requirements

The switchgear assembly shall consist of two low voltage metal enclosed switchgear sections. One section will contain two series connected reactor trip circuit breakers. The second will contain two bypass circuit breakers connected so that a bypass breaker parallels each reactor trip breaker. The bypass circuit breaker is used to bypass the reactor trip breaker for on-line testing of the latter with the reactor in operation.

The system also includes two 260 volt line to line identical three phase Motor-Generator sets rated at 400 KVA, reverse current relay, generator output circuit breaker, a synchronizer, and a common ground relay.

Each circuit breaker shall have provisions for locking it in the "Test" and "Disconnected" draw-out positions.

The circuit breaker also includes positions for "Connected" and "Remove." (Reference 6.4.077)

Interposing relays shall be used to isolate Train A from Train B wiring where it is necessary to parallel these circuits into a single output.

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.

A correct

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. Aux Bldg 125V DC

D incorrect

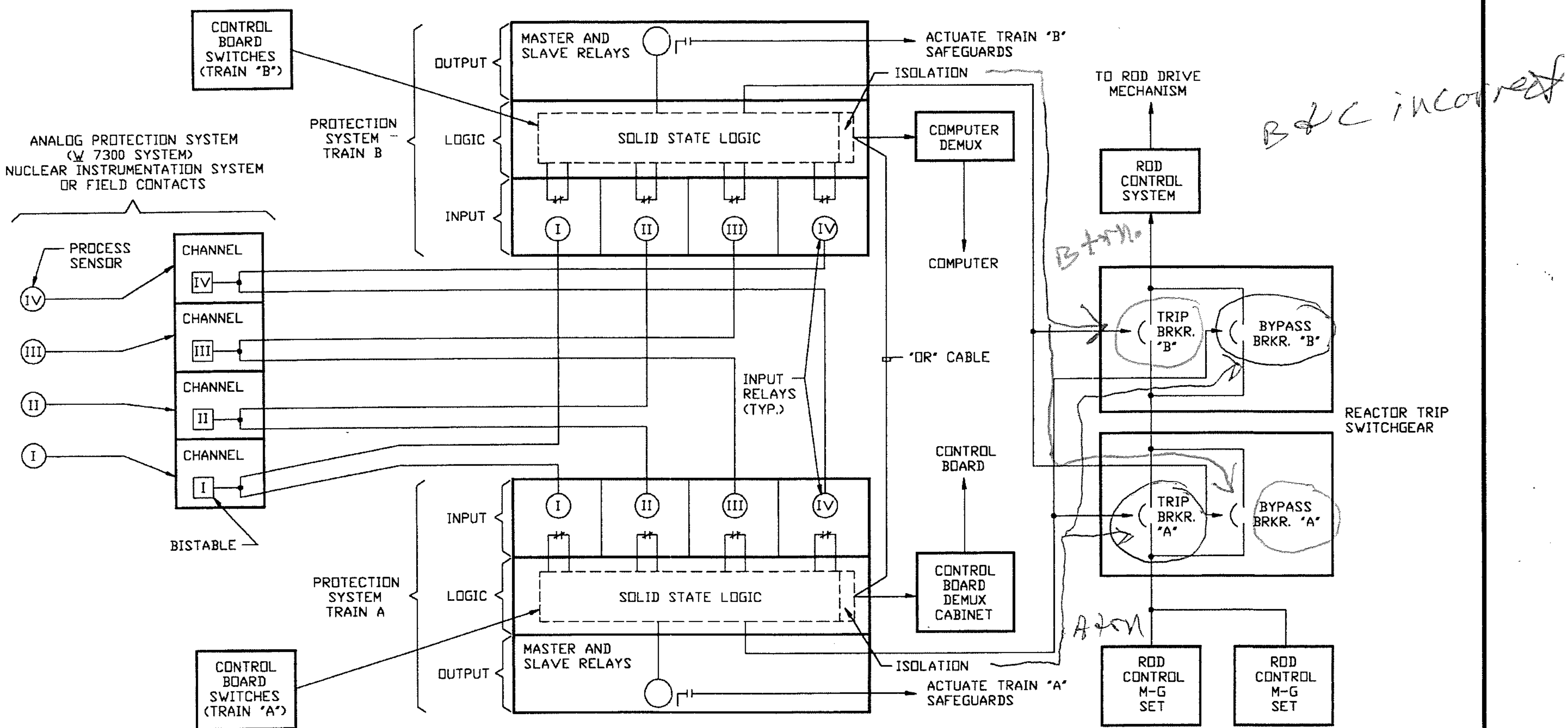
For the reactor trip bypass breaker, the SHTR relay is energized only by a manual pushbutton. After the reactor trip bypass breaker is opened, then a contact in series with the SHTR relay opens to de-energize the coil. Thus, the SHTR relay is only momentarily energized.

For the reactor trip breaker, the SHTR relay is energized by the closing of a contact associated with a shunt trip attachment relay (STA for 52/RTA and STB for 52/RTB). STA (STB) is energized from the RPS voltage to the UV trip coil of the 52/RTA (52/RTB). When the voltage is removed by an automatic reactor trip signal, the relay will de-energize, closing its contact to energize the shunt trip coil of 52/RTA (52/RTB). After the reactor trip 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.

3.3.3 Design Transients

The ambient design conditions are: 95% relative humidity and 40 deg. F to 120 deg. F temperature. (Reference 6.4.090)

Also see Protection Features 3.3.7.



REACTOR PROTECTION SYSTEM BOUNDARIES

FIGURE F-1

DF03 ← Attn 4160V vital Bus
ED04
LA13
1B 125V DC DIST PNL **AB-139'** **D177082**
(CONT'D)

BKR	TPNS	DESCRIPTION	SEE PAGE
1B-11	N1R15G0001C-N	1F 4160V BUS BREAKER TEST CABINET	
1B-12	**Q1N21L0001A-A**	**1A HOT SHUTDOWN PANEL AUX RELAY CABINET >>>**	**F-56**
1B-13	Q1H21E0004-A	1F 4160V BUS LOCAL CONTROL PANEL DIFFERENTIAL LOCKOUT RELAY CONTROL CIRCUIT	
1B-14	Q1R43E0001A-A	1F BUS LOADING SEQUENCER CONTROL POWER TO: LOSP SEQ, LOAD SHEDDING, BKR CLOSE FAILURE & SEQ LOCAL ANNUN	
1B-15	Q1H21E0504-A	1H 4160V BUS LOCAL CONTROL PANEL DIFFERENTIAL PROTECTION CONTROL CIRCUIT	
1B-16	Q1C11E0004B-AB	"A" REACTOR TRIP SWITCHGEAR CONTROL POWER TO BYPASS BREAKER & REACTOR TRIP BREAKER	
1B-17	**Q1H21NBAFP2605A-A**	**1A LOCAL HOT SHUTDOWN PANEL >>>**	**F-57**
	Q1H21NBAFP2605G-A	**1G LOCAL HOT SHUTDOWN PANEL >>>**	**F-58**
1B-18	Q1R43E0501A-A	1H BUS LOADING SEQUENCER CONTROL POWER TO LOAD SHEDDING CONTROL CIRCUIT	
1B-19	**Q1H25L0004-A**	**4A TERMINATION CABINET PANEL 4 REAR >>>**	**F-59**
1B-20	N1R15A0003-N	1C 4160V BUS UNDERVOLTAGE AND UNDERFREQUENCY PROTECTIVE RELAYING	
1B-21	**Q1H25L0006-A**	**6A TERMINATION CABINET PANEL 1 FRONT >>>**	**F-60**
1B-22	------------	SPARE	

18. 012K6.03 001/FNP BANK/RO/MEM 3.1/3.5/N/N/3/CVR/Y

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

- PT-455, PRZR PRESS, has failed off-scale HIGH.
- **NO Operator action** has been taken.

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

	Reactor Trip	Safety Injection
A.	1	1
B.✓	1	2
C.	2	1
D.	2	2

A - Incorrect. The first part is correct. The second part is incorrect, but plausible since if the failed instrument tripped all bistables in the fail safe condition it would be correct. It would also be correct after the applicable TS and procedure directed actions were complete (tripping all bistables), but the question specifies "assume no operator actions". Pressure SI is only on low pressure, and the instrument failing high does not automatically trip the low pressure bistable. The Reactor trip is on low or high pressure at this power level, and the high pressure condition would need only one more bistable in to cause a reactor trip.

B - Correct. One more bistable on high pressure would cause a reactor trip, but the SI is actuated on low pressure only, so two more in the low pressure condition are required for an SI.

C - Incorrect. First part is incorrect, but correct for a low pressure reactor trip. 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 incorrect (see A). Both parts together are also plausible since confusion could cause choosing the exact opposite of the correct answer.

D - Incorrect. The first part is incorrect, but plausible since for a low pressure reactor trip it is correct (see C). The second part is correct since the only SI PRZR pressure actuation is low pressure (see B).

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

B. **Symptoms**

I. The following are symptoms that require a reactor trip, if one has not occurred:

Reactor Trip	Instrumentation (TSLB)	Setpoint	Coincidence
1. Source Range High Flux (If not blocked)	NI-31,32 (TSLB3 1-1,1-2)	10^5 cps	1/2
2. Intermediate Range High Flux (If not blocked)	NI-35,36 (TSLB3 2-1,2-2)	Reference Surveillance Test Data Book for current S.P.	1/2
3. Power Range High Flux, Low Setpoint (If not blocked)	NI-41,42,43,44 (TSLB3 6-1,6-2,6-3,6-4)	25% Rx Pwr	2/4
4. Power Range High Flux, High Setpoint	NI-41,42,43,44 (TSLB2 11-1,11-2,11-3, 11-4)	109% Rx Pwr	2/4
5. Power Range High Positive Flux Rate	NI Cabinets (TSLB2 12-1,12-2,12-3, 12-4)	+5%/2 sec.	2/4
6. OTΔT	TI-412C,422C,432C (TSLB2 7-1,7-2,7-3)	117% +credits -penalties	2/3
7. OPΔT	TI-412B,422B,432B (TSLB2 8-1,8-2,8-3)	110%-penalties	2/3
① 8. Pressurizer ~~Low~~ Pressure	→channel I PI-455,456,457 (TSLB2 19-1,19-2,19-3)	1865 psig (rate compensated)	2/3 (Rx Pwr > 10%)
② 9. Pressurizer High Pressure	→ channel I PI-455,456,457 (TSLB2 20-1,20-2,20-3)	2385 psig	2/3

① one channel will Not actuate
② one channel already actuated (one more = 2/3)
A&B 1st parts correct
B&D 1st parts incorrect (but correct for low pressure R/trip)

II. The following are symptoms of a reactor trip:

a. Any reactor trip annunciator lit.

b. Rapid decrease in neutron level indicated by nuclear instrumentation.

c. All shutdown and control rods are fully inserted. Rod bottom lights are lit.

III. The following are symptoms that require safety injection, if one has not occurred:

SI Signal	Instrumentation (TSLB) Channel I	Setpoint	Coincidence
① 1. Pressurizer pressure ~~low~~ (If not blocked)	PI-455,456,457 (TSLB2 17-1,17-2,17-3)	1850 psig	2/3
2. Steam Line Differential pressure	PI-474,484,494, PI-475,485,495, PI-476,486,496 (TSLB4 10-2,10-3,10-4, 11-2,11-3,11-4, 12-2,12-3,12-4, 13-2,13-3,13-4, 14-2,14-3,14-4, 15-2,15-3,15-4)	100 psid	1 steam line 100 psig less than other two on 2/3 protection sets
3. Low Steam Line pressure (If not blocked)	PI-474,485,496 (TSLB4 19-2,19-3,19-4)	585 psig (rate compensated)	2/3
4. Containment pressure high	PI-951,952,953 (TSLB1 1-2,1-3,1-4)	4 psig	2/3
5. Manual	N/A	N/A	1/2

IV. The following are symptoms of a safety injection:

a. Any SI annunciator lit.

b. BYP & PERMISSIVE SAFETY INJECTION ACTUATED status light lit

c. MLB-1 1-1 or MLB-1 11-1 lit

d. HHSI flow greater than 0 gpm.

① One channel will Not actuate - 2 more low will be required to actuate an SI.

19. 013K2.01 002/NEW/RO/MEM 3.6/3.8/N/N/3/CVR/VER 5 EDITORIAL

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).

CAUTION:	**Failure to deenergize all sources of DC control power while engaged in breaker racking could result in equipment damage AND severe personal injury OR death.**

Warning that some ESF BKRS have more than one source of DC control power.

4.6 Refer to Table 1 to determine IF breaker has alternate DC control power, AND IF additional action is required.

NOTE:	**A breaker in an ESF 4160V bus, 1/2F, 1/2G, 1/2K, 1/2L, 1/2J and 1/2H, cannot be left in the TEST position if the switchgear is required to be operable in Modes 1-4, unless the seismic modification has been implemented on both the breaker and the cubicle. In Modes 5, 6, and defueled, the breaker may be left in the TEST position and the switchgear will still be operable. (See Precautions and Limitations.)**

4.7 PRIOR to racking 4160V Circuit Breaker, perform a visual inspection of visible control wires on the cubicle door to ensure:
- no significant insulation damage is present.
- visible wires are landed at terminations.

4.8 Removing Breaker to TEST Position

4.8.1 Insert racking lever on left side of fulcrum plate and attach to mating hole on breaker. (Refer to Fig. 2.)

4.8.2 Depress breaker release lever, this releases the breaker from the interlock bar.

A. At the same time push down on the racking lever to disengage the main line contacts.

B. When the plunger is out of the guide rail notch, the breaker release lever can be released.

NOTE:	**It is possible to use the racking lever to pull the breaker to the TEST position, but caution should be used to ensure that the operator's hands do not slip off the lever, or that the handle does not slip off the lever.**

4.8.3 Remove the racking lever from the breaker compartment.

Pmp 5

TABLE 1

BREAKERS WITH ALTERNATE DC CONTROL POWER

CAUTION: **Failure to deenergize all sources of DC control power while engaged in breaker racking could result in equipment damage AND severe personal injury OR death.**

1. Breakers in this table have two separate DC control power supplies. One DC supply feeds the breaker through the DC supply in the breaker cubicle. An alternate DC supply has been installed to provide a separate supply for components capable of being operated from the Hot Shutdown Panel (HSP).

2. Alternate DC power supplies are only energized when the handswitch on the HSP is place in LOCAL.

3. Remote DC control power fuses will be removed (replaced) from (in) breakers in the table in conjunction with securing (restoring) local DC control power OR the appropriate HSP control switch will be verified in the REMOTE position.

COMPONENT	BREAKER	REMOTE CONTROL FUSE LOCATION
Charging/HHSI Pump 1B/2B (B Train)	DG07	HSP-C
Charging/HHSI Pump 1C/2C	DG06	HSP-C
CCW Pump 1A/2A	DG04	HSP-C
CCW Pump 1B/2B (B Train)	DG05	HSP-C
MDAFW Pump 1B/2B	DG10	HSP-C
Pressurizer Heater Group 1B/2B	EC11	HSP-C

Pmp Z

A C

1G 4160V BUS AB - 121' D177006

BKR	**TPNS**	**DESCRIPTION**	**SEE PAGE**
	Q1R15A0007-B	**1G 4160V BUS**	
DG01	**N1R11A0501-N**	**1A STARTUP TRANSFORMER (ALTERNATE) <<<**	
DG02	**Q1R15A0506-B**	**1L 4160V BUS >>>**	L-1
DG03	**Q1R11B0005-B**	**1E 4160/600V SST (NORMAL) >>> EE02 >>>**	G-2
DG04	Q1P17M0001A-B	1A CCW PUMP	
DG05	Q1P17M0001B-AB	1B CCW PUMP DISC SWITCH Q1R18A00004B-B >>> 1B CCW PUMP (B TRAIN SUPPLY)	
DG06	Q1E21M0001C-B	1C CHARGING/HHSI PUMP	
DG07	Q1E21M0001B-AB	1B CHARGING/HHSI PUMP DISC SWITCH Q1R18A0001B-B >>> 1B CHARGING/HHSI PUMP (B TRAIN SUPPLY)	
DG08	**Q1R43A0502-B**	**1B DIESEL GENERATOR (EMERG) <<<**	
DG09	Q1E11M0001B-B	1B RHR/LHSI PUMP	
DG10	Q1N23M0001B-B	1B AFW PUMP	
DG11	Q1E13M0001B-B	1B CTMT SPRAY PUMP	
DG12	**Q1R11B0006-AB**	**1F 4160/600V SST DISC SWITCH Q1R18A0003B-B >>> 1F 4160/600V SST >>> 1F LOAD CENTER (B TRAIN SUPPLY) >>>**	F-113
DG13	**Q1R15A0504-B**	**1J 4160V BUS >>>**	J-1
DG14	Q1R15BKRDG14	PT COMPARTMENT	
DG15	**N1R11A0502-N**	**1B STARTUP TRANSFORMER (NORMAL) <<<**	

ESF Pumps 4/600 BKRS (AC)

D C

DG03
EE05
LB07
1E 125V DC DIST PNL Normal DC Control Power supply. **AB-121'** **D177083**

BKR	**TPNS**	**DESCRIPTION**	**SEE PAGE**
	Q1R41L0001E-B	**1E 125V DC DISTRIBUTION PANEL <<< LB07**	
1E-01	Q1R15A0007-B	1G 4160V BUS DC CONTROL POWER FOR INC BREAKERS DG01, DG08 & DG15	
1E-02	Q1R16B0005-B	1C 600V LOAD CENTER DC CONTROL POWER FOR INC BREAKERS EC02, EC07, EC08 & EC10	
1E-03	Q1R15A0007-B	1G 4160V BUS DC CONTROL POWER FOR FEEDER BREAKERS DG02, DG03, DG04, DG05, DG06, DG07, DG09, DG10, DG11, DG12 & DG13	
		1G 4160V BUS U/F TRIP AUX RELAYS (TRIP DG BKR DG08)	
1E-04	Q1R16B0005-B	1C 600V LOAD CENTER DC CONTROL POWER FOR FDR BREAKERS EC03, EC04, EC05, EC06, EC09, EC11, EC12, EC13 & EC14	
1E-05	**Q1H21NBL2702B-B**	**"B" TRAIN PENETRATION ROOM ISOLATION PANEL >>>**	G-51
1E-06	Q1R16B0007-B	1E 600V LOAD CENTER DC CONTROL POWER FOR INC BREAKERS EE02, EE07 & EE12	
1E-07	Q1R15A0504-B	1J 4160V BUS DC CONTROL POWER FOR BREAKERS DJ01, DJ02, DJ03, DJ04, DJ06 & DJ07	
		1J 4160V BUS U/F TRIP AUX RELAYS (TRIP DG BKR DJ06)	
	N1R15A0509-N	1J 4160V BUS BREAKER TEST CABINET	
1E-08	Q1R16B0007-B	1E 600V LOAD CENTER DC CONTROL POWER FOR FEEDER BREAKERS EE03, EE05, EE06, EE08, EE09, EE10, EE11, EE13, EE14, EE15	

DG04, DG06: CCW + Chg (HHSI) Pmps
DG09, DG10: RHR (LHSI) & AFW Pmps

PMP 4

DC

DG03
EE05
LB14
1F 125V DC DISTR PNL Alt. DC control/power supply **AB-121'** **D177083**

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q1R41L0001F-B	**1F 125V DC DISTRIBUTION PANEL <<< LB14**	
1F-01	**N1G22NBWPP2603C-N**	**WASTE PROCESSING BORON RECYCLE PANEL >>>**	G-70
1F-02	----------	SPARE	
1F-03	----------	SPARE	
1F-04	----------	SPARE	
1F-05	N1G22NAHR2610B-N	1B CATALYTIC H2 RECOMBINER DC CONTROL PANEL - ANNUNCIATORS & ECV-1119, TCV-1114, ECV-1112 SOLENOIDS	
1F-06	**N1G21NGWEP2602-N**	**WASTE ENCAPSULATION SYSTEM CONTROL PANEL >>>**	G-71
1F-07	B1TB0004	JUNCTION BOX FOR MSVR FLOODING SENSOR RELAYS 49-1, 49-2, 49-3 & LSX (2-3A FUSES)	
1F-08	----------	SPARE	
1F-09	**Q1H21NBAFP2605B-B**	**LOCAL HOT SHUTDOWN PANEL "1B" >>>**	G-72
1F-10	**Q1P15NFSS2607B-B**	**"B" TRAIN SAMPLE ISOLATION VALVE CONTROL PANEL >>>**	G-73
1F-11	----------	SPARE	
1F-12	**Q1N21L0001B-B**	**1B HOT SHUTDOWN PANEL AUX RELAY CABINET >>>**	G-74
1F-14	**Q1H21NBAFP2605C-B**	**1C LOCAL HOT SHUTDOWN PANEL SELECTOR SWITCH BOX >>>**	G-75

Alt control power to CHHSI & B AFW.

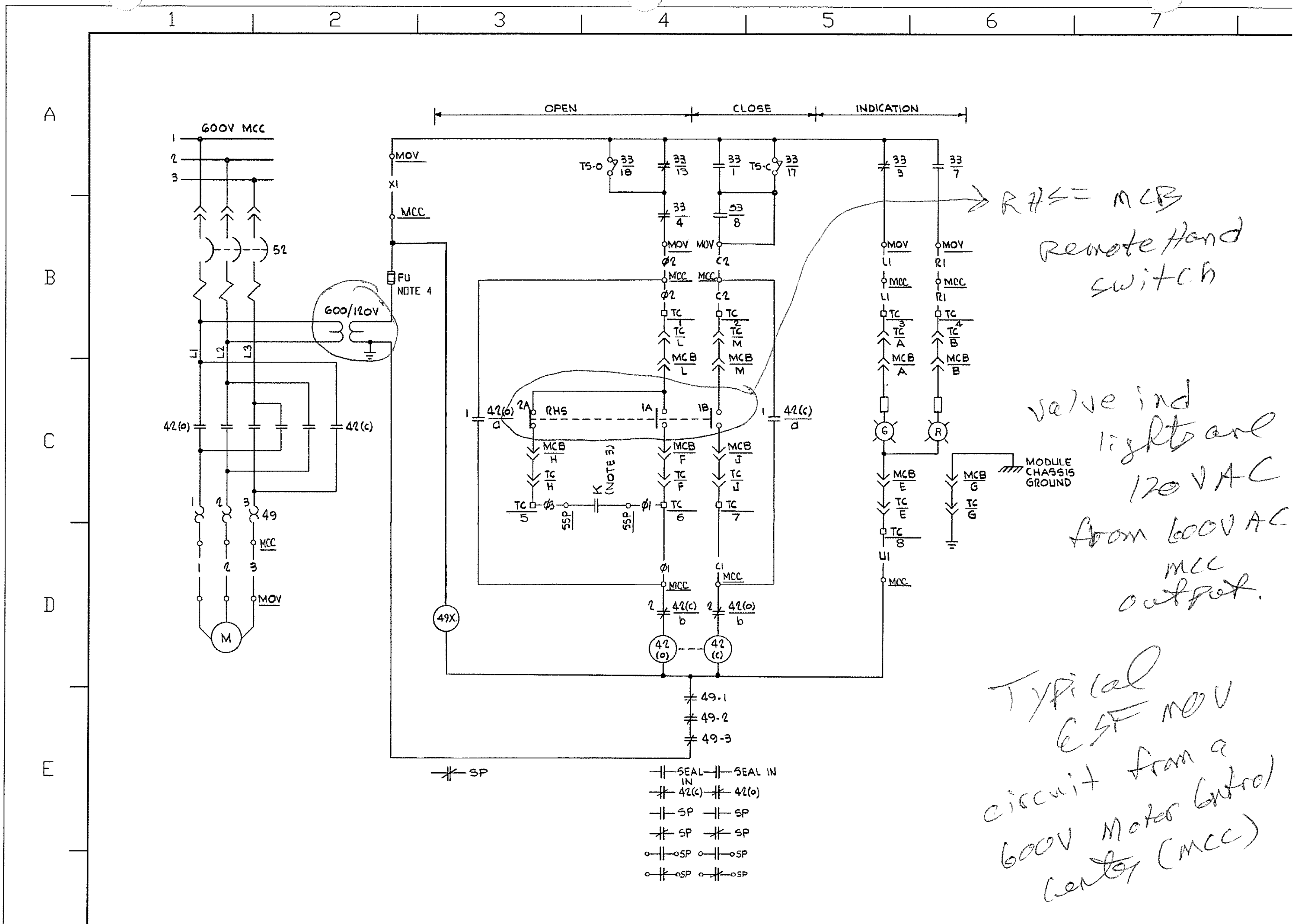
8 MP 4

DG03
EE05
LB14
1F-14
1C LOCAL HOT SHUTDOWN PNL
SEL SWITCH BOX **AB-121'** **D181664**

FUSE	TPNS	DESCRIPTION
	Q1H21NBAFP2605C-B	**1C LOCAL HOT SHUTDOWN PANEL SELECTOR SWITCH BOX <<< 1F-14**
F1	Q1H22L0004-B	RELAYS TR1-TR6 ON TRC-3 FOR: RX VSL HEAD VENT SVs 2213B-B & 2214B-B; PRZR PWR REL SVs 0444BA-B & BB-B AND ISO MOV 8000B-B; RWST MOV 0115D-B; CCW HX MOV 3047-B; "HSP SEL SW IN LOCAL" ALARM
F8	52-DG06	LOCAL MODE DC CONTROL PWR FOR FEEDER BKR TO CHG/HI HEAD SAFETY INJECTION PMP 1C ← Hot S/D Pnl
F9	52-DG07	LOCAL MODE DC CONTROL PWR FOR FEEDER BKR TO CHG/HI HEAD SAFETY INJECTION PMP 1B
F10	52-DG04	LOCAL MODE DC CONTROL PWR FOR FEEDER BKR TO CCW PMP 1A
F11	52-DG05	LOCAL MODE DC CONTROL PWR FOR FEEDER BKR TO CCW PMP 1B
F12	52-DG10	LOCAL MODE DC CONTROL PWR FOR FEEDER BKR TO AFW PMP 1B
F13	52-EC11	LOCAL MODE DC CONTROL PWR FOR FEEDER BKR TO PRZR HTR BACKUP GROUP 1B

Local Control

Typical "Alternate" Control Power


OPEN
CLOSE
INDICATION
600V MCC
600/120V
MOV
MCC
FU NOTE 4
RHS
MODULE CHASSIS GROUND
49X
M
RHS = MCB Remote Hand switch
valve ind lights are 120 VAC from 600 VAC MCC output.
Typical CSF MOV circuit from a 600V Motor Control Center (MCC)

6 7 8 9 10 11 12 13

DEVICE	DESCRIPTION	MFR/TYPE	REMARKS
49X	AUXILIARY RELAY	ROWAN TYPE E	1 NC CONTACT 120 V AC

A

33 (NOTE 6) 13.0

LIMIT SWITCH CONTACT DEVELOPMENT					
CONT	VALVE POSITION FULLY OPEN – FULLY CLOSED				FUNCTION
1					BYPASS CKT
2					(MON. LGT. GR. I)
3					GIL
4					OPEN LIMIT
5					SPARE
6					SPARE
7					RIL
8					CLOSED LIMIT
9					MCB ALARM
10					SPARE
11					SPARE
12					SPARE
13					BY PASS CKT.
14					SPARE
15					SPARE
16					SPARE

D-177414 SH. 1

NOT USED

A-177100 SH. 134

CONTACT CLOSED

17 CLOSING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECHANICAL OVERLOAD OCCURS DURING CLOSING CYCLE

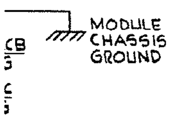
18 OPENING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECHANICAL OVERLOAD OCCURS DURING OPENING CYCLE

B

NOTES

1. VALVE IS SHOWN IN FULLY CLOSED POSITION
2. FOR MOTOR & STARTER SIZE, SEE DWG B-177556 SHT 19
3. CONTACT CLOSES ON SAFETY INJECTION ACTUATION SIGNAL, FOR RELAY AND CONTACT N°S SEE EQIPMENT TABLE
4. REFER TO FUSE MANUAL A-181987 FOR FUSE SIZE AND TYPE.
5. COMPONENTS MARKED WITH AN (*) ARE WITHIN E.Q. SCOPE. FOR INSTALLATION DETAILS SEE E.Q. REFERENCE DRAWING LISTED BELOW.
6. SEE PDMS FOR LIMIT SWITCH SETTING GUIDANCE. 13.0

C

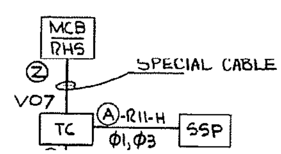


D

RHS AUTO CLOSE OPEN	BLK	CONT	CLOSE	AUTO	OPEN
	1 (FRONT)	A			X
		B	X		
	2 (REAR)	A		X	

REMOTE HANDSWITCH
GEMCO CAT N° 40453 2221-Y-AA3,A4
SPRING RETURN TO AUTO

E

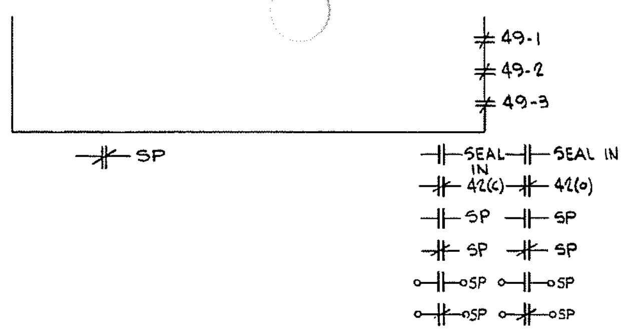


REFERENCE DRAWINGS

A-177538 — ELECTRICAL GENERAL DETAILS AND NOTES

B-181612/36 CONN. DIAG HHSI TO RCS CL. ISO. MDV Q1E21MDV8803A-A

F



E

F

G

SERVICE	FAC	SCHEME	MOV VALVE	MCC MCC CUBICLE	MCB MAIN CONTROL BOARD	TC TERMINATION CABINET	SSP SOLID STATE PROT. PNL.	33 LIMIT SWITCH	RHS REMOTE HANDSWITCH	START UP NUMBER		K RELAY Nº	SSP TERM NOS. & RELAY CONTACT NO.
HHSI TO RCS CL ISO MOV	IV	AFU-AG	Q1E21 MOV8803A-A *	FU-AG	N1H11NG MCB2500A-AB	Q1H25L008-A	Q1H11NG SSP2506M-A	N1E21 ZS8803A-A	N1E21 HS8803A-A	52		1-K604	SSP-A / TB603-11 9 10 SSP / TB60

H

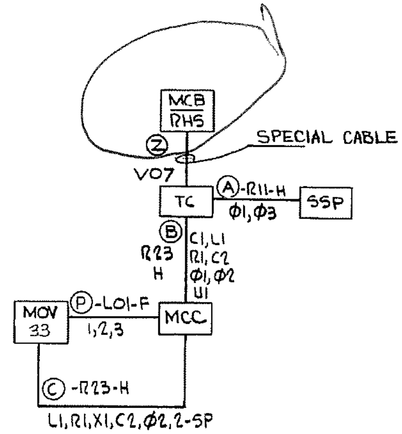
J

BY	CHK'D	PROJ ENGR	D'SGN DEPT	D'SGN DEPT	SUPV ENGR	BECH	BY	CHK'D	PROJ ENGR	D'SGN DEPT	D'SGN DEPT	SUPV ENGR	BECH	BY	CHK'D	PROJ ENGR	D'SGN DEPT	D'SGN DEPT	SUPV ENGR	BECH	BY	CHK'D	PROJ ENGR	D'SGN DEPT	D'SGN DEPT	SUPV ENGR	BECH	BY	CHK'D	PROJ ENGR	D'SGN DEPT	D'SGN DEPT	SUPV ENGR	BECH	BY	CHK'D	PROJ ENGR	D'SGN DEPT	D'SGN DEPT	SUPV ENGR	BECH	BY	CHK'D	PROJ ENGR	D'SGN DEPT	D'SGN DEPT	SUPV ENGR	BECH

VER. NO. ______ DATE ______ VER. NO. ______ DATE ______ VER. NO. ______ DATE ______ VER. NO. ______ DATE ______ VER. NO. ______ DATE ______ VER. NO. ______ DATE ______ VER. NO. ______ DATE ______

1 2 3 4 5 6 7

REMOTE HANDSWITCH
GEMCO CAT Nº 404532221-Y-AA3,A4
SPRING RETURN TO AUTO



BLOCK DIAGRAM

REFERENCE DRAWINGS

A-177538 — ELECTRICAL GENERAL DETAILS AND NOTES

B-181612/36 CONN. DIAG HHSI TO RCS CL. ISO. MOV Q1E21MOV8803A-A

D-181900/14 INSTALLATION DETAILS FOR E.Q. LIMITORQUE MOV'S.

K

RELAY Nº	SSP TERM NOS. & RELAY CONTACT NO.
1-K604	SSP-A TB603-11 —9 ‖ 10— SSP-A TB603-12

TC TERMINAL BLOCK NUMBERS							
1	2	3	4	5	6	7	8
4TB1-13	4TB1-10	4TB1-15	4TB1-16	4TB1-11	4TB1-12	4TB1-9	4TB1-14

CAD D1776141
OVY2005 JLO - 03

Southern Company Services, Inc. for

ALABAMA POWER COMPANY

PROJECT J.M. FARLEY NUCLEAR PLANT - UNIT NO.1

SUBJECT ELEMENTARY DIAGRAM
575V MOTOR OPERATED VALVE MOV8803A-A

DRAWN ASB CHECKED HMJ DESIGNED

APPROVED M. MALCOM DATE 8-14-72

APPROVED DATE

APPROVED DATE

SCALE NONE B/M

SHEET 1 OF — SHEETS

SUPERSEDES

D-177614

BY JLO CHK'D RLP PROJ ENGR JMR
VER. NO. 13.0 DATE 11-30-07
REVISED PER ABN 1072404801E002, VER. 1.0

DF03
ED14
1U 600/208V MCC **AB - 139'** **B177556-19**
(CONT'D)

BKR	TPNS	DESCRIPTION	SEE PAGE
FUN4	------------	SPARE	
FUN5	Q1E15MOV3362A-A	PENETRATION ROOM TO PENETRATION ROOM FILTER MOV	
FUO2	Q1N23MOV3764D-A	AUX FEEDWATER TO STEAM GENERATOR MOV	
FUO4	Q1N23MOV3764F-A	AUX FEEDWATER TO STEAM GENERATOR MOV	
FUO5	Q1E21LCV115B-A	RWST TO CHG PMP SUCTION ISOLATION	
FUP2L	Q1N12MOV3406-AB	TURBINE DRIVEN AUX FEED PUMP MOV	
FUP4L	------------	**1U 600/208V MCC XFMR >>> 1U MCC 208V SECTION >>>**	**F-109**
FUR2	Q1E21MOV8886-A	DISC SWITCH Q1R18B029-A >>> HHSI TO RCS HOT LEG MOV	
FUR3	Q1E21MOV8803B-AB	HHSI TO RCS CL ISO MOV	
FUR4	Q1E11MOV8811A-A	CTMT SUMP OUTLET MOV	
FUR5	Q1E14MOV3660-A	CTMT AIR SAMPLE MOV	
FUS2	Q1P16MOV3019A-A	CTMT COOLER SERVICE WATER INLET MOV	
FUS3	Q1P16MOV3019B-A	CTMT COOLER SERVICE WATER INLET MOV	
FUS4	Q1P17MOV3094B-A	SPENT FUEL POOL HX INLET MOV	
FUS5	------------	SPARE	
FUT2	Q1N23MOV3209A-A	AUX FEEDWATER PUMP SERVICE (MD) INTAKE MOV	
FUT3	Q1N23MOV3210A-A	AUX FEEDWATER PUMP SERVICE (MD) INTAKE MOV	

Typical ESF MOV, HHSI Cold leg inj. Valve

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

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 trains (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 channel or train 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 channels 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 **channels** 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

TABLE T-4 - ENGINEERED SAFEGUARDS ACTUATION SIGNALS

CONTAINMENT SPRAY ACTUATION

CONTAINMENT SPRAY	SETPOINT	COINCIDENCE	INTERLOCKS & BLOCKS	PROTECTION PROVIDED FOR	MODES OF OPERATION	FSD SECTION
High-3 containment pressure Deenergized Bypassed PB950A PB951A energize to actuate PB952A PB953A Deenergized by 1D 120V vital panel	27 psig	2/4 High-3 containment pressure signals ≥ setpoint	None	Protects containment for a loss of coolant or steam line break inside containment Prevents over pressurization of containment structure and subsequent building rupture	1, 2, 3	2.4 2.7.1 2.7.2 Fig. 2 Sht. 8
Manual	N/A	-2/4 switches -Consist of four momentary switches in two groups	-Containment spray actuation is manually reset by depressing both train A and train B reset push buttons on MCB	Operator discretion	1, 2, 3, 4	2.7.1 Fig. 2 Sht. 8

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Containment Spray — Automatic Actuation Logic and Actuation Relays (continued)

the use of the Manual Initiation Switches. Automatic Actuation Logic and Actuation Relays must be OPERABLE in MODE 4 to support system level Manual Initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

c. Containment Spray — Containment Pressure – High 3

This signal provides protection against a LOCA or an SLB inside containment. The transmitters (d/p cells) and electronics are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. Thus, the transmitters will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

This Function requires the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious. Note that this Function also has the inoperable channel placed in bypass (disabled) rather than trip to decrease the probability of an inadvertent actuation.

The Containment Pressure High 3 instrument Function consists of a two-out-of-four logic configuration. Since containment pressure is not used for control, this arrangement exceeds the minimum redundancy requirements. Additional redundancy is warranted because this Function is energize to trip. Containment Pressure — High 3 must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In

(continued)

injection is required and containment spray may be necessary to ensure the integrity of the containment building and limit offsite dosage. Actuation signals for these protective functions are provided by containment pressure transmitters. If 2/3 high-1 containment pressure channels sense a pressure greater than 4 psig, then safety injection will be initiated. (References 6.1.002, 6.4.007, 6.4.015, 6.4.036)

2. Containment High-2 Pressure

If containment pressure increases to 16.2 psig on 2/3 containment pressure channels, a main steam line isolation signal is generated. This signal is installed to isolate a leaking steam line and prevent further pressure increases in containment. (References 6.1.002, 6.4.007, 6.4.015, 6.4.036)

3. Containment High-3 Pressure

The high-3 containment building pressure detection circuitry functions to initiate a Phase B containment isolation and actuate the containment spray system. Spraying is necessary to prevent overpressurization of the containment structure and maintain structural integrity to limit the offsite dosage.

The high-3 pressure setpoint is 27 psig. An actuation signal occurs when 2/4 containment pressure channels exceed their setpoints. These are the only circuits which are energized to actuate, because inadvertent spray actuation is not desirable.

The containment spray actuation shall utilize all four pressure channels in 2/4 coincidence logic. Only one channel should be bypassed at a time. An additional protection channel degraded by single failure criteria should not render the spray actuation inoperable. The 2/4 coincidence logic which is utilized for containment spray ensures the availability of the minimum protection channels for actuation. It meets the requirement of IEEE-279. Moreover, the reliability of the circuit is established by periodic testing. (References 6.1.002, 6.4.007, 6.4.015, 6.4.036)

4. Low Pressurizer Pressure

A steam break accident or a LOCA will cause a decrease in pressurizer pressure due to the outsurge of water from the pressurizer. The outsurge will be caused by either the water loss from the LOCA or the rapid cooldown produced by the steam break. Safety injection will be initiated by low pressurizer pressure of 1850 psig, as sensed by 2/3 pressure detectors. This meets the

TABLE T-4 - ENGINEERED SAFEGUARDS ACTUATION SIGNALS

SAFETY INJECTION ACTUATION (CONTINUED)

ACTUATION SIGNAL	SETPOINT	COINCIDENCE	INTERLOCKS & BLOCKS	PROTECTION PROVIDED FOR	MODES OF OPERATION	FSD SECTION
High-1 containment pressure PB951B PB952B PB953B Deenergize to Actuate for SI Channel I, II, III	4 psig	2/3 High-1 pressure signals ≥ setpoint	None	Loss of coolant or steam line break within containment	1, 2, 3	2.4 2.7.1 2.7.2 Fig. 2 Sht. 8
Manual	N/A	1/2 Momentary switches	None	Operator discretion	1, 2, 3, 4	2.7.1 Fig. 2 Sht. 8

II. The following are symptoms of a reactor trip:

a. Any reactor trip annunciator lit.

b. Rapid decrease in neutron level indicated by nuclear instrumentation.

c. All shutdown and control rods are fully inserted. Rod bottom lights are lit.

III. The following are symptoms that require safety injection, if one has not occurred:

SI Signal	Instrumentation (TSLB)	Setpoint	Coincidence
1. Pressurizer pressure low (If not blocked)	PI-455,456,457 (TSLB2 17-1,17-2,17-3)	1850 psig	2/3
2. Steam Line Differential pressure	PI-474,484,494, PI-475,485,495, PI-476,486,496 (TSLB4 10-2,10-3,10-4, 11-2,11-3,11-4, 12-2,12-3,12-4, 13-2,13-3,13-4, 14-2,14-3,14-4, 15-2,15-3,15-4)	100 psid	1 steam line 100 psig less than other two on 2/3 protection sets
3. Low Steam Line pressure (If not blocked)	PI-474,485,496 (TSLB4 19-2,19-3,19-4)	585 psig (rate compensated)	2/3
* 4. Containment pressure high	PI-951,952,953 (TSLB1 1-2,1-3,1-4)	4 psig	2/3
5. Manual	N/A	N/A	1/2

IV. The following are symptoms of a safety injection:

a. Any SI annunciator lit.

b. BYP & PERMISSIVE
SAFETY
INJECTION
ACTUATED status light lit

c. MLB-1 1-1 or MLB-1 11-1 lit

d. HHSI flow greater than 0 gpm.

* High-1 Channels I, II, III deenergize to actuate SI

Trip/Actuation Accuracy - This definition includes comparator accuracy, channel accuracy for each input, and rack environmental effects. This is the tolerance expressed in process terms (or percent of span) within which the complete channel shall perform its intended trip/actuation function. This includes all instrument errors but no process effects such as streaming.

2.2 GENERAL FUNCTIONAL REQUIREMENTS-REACTOR TRIP SYSTEM, ENGINEERED SAFETY FEATURES ACTUATION SYSTEM.

2.2.1 The Reactor Protection System acts to limit the consequences of ANSI Condition II events (e.g. loss of feedwater, etc.). Minimum departure from nucleate boiling ratio (DNBR) shall not be less than the designed limit DNBR as a result of any anticipated transient or malfunction for Condition II events and as such shall meet 95/95 criterion. That is, departure from nucleate boiling will not occur on at least 95 percent of the limiting fuel rods at 95 percent confidence.

(DNBR design limit for vantage 5 fuel is 1.24/1.23 for the typical and thimble cells and 1.25/1.24 for LOPAR fuel for the typical and thimble cells respectively).

Rod linear Power density shall not exceed the rated design value (22.4 KW/ft.) and the stress limits of the reactor coolant system as specified for Condition II events (2735 psig) shall not be violated. Release of radioactive material for any Condition III fault shall not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius and shall not exceed the guidelines of 10 CFR 20. For any Condition IV fault, release of radioactive material shall not result in an undue risk to public health and safety nor shall it exceed the guidelines of 10 CFR 100, "Reactor Site Criteria."

The Engineered Safety Feature Actuation System, in addition to Reactor Trip, limits the consequences of ANSI Condition III events and mitigates ANSI Condition IV events. Table T-1 summarizes the ANS classification of faults and safety analysis approach as outlined in FNP-FSAR-Chapter 15. Most of the analyzed events/transients can be detected by one or more protection functions. The various Safety Analyses take credit for most of the protection functions. Those functions for which no credit is taken in the analyses are still required to be operable to enhance the overall reliability and diversity of the protection system. This design shall meet the requirements of IEEE-279-1971.

(References 6.1.008, 6.1.031, 6.7.013, 6.7.039)

(over)

2.2.2 The RPS system is housed in two physically and electrically independent equipment trains (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 channel or train 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,)

2.2.4 Actuation shall be automatic when the limits of the monitored parameters are exceeded. While one train is in test, the redundant train shall be capable of performing Reactor Trip/ESFAS actuation. (References 6.1.002, 6.7.014, 6.7.015)

2.2.5 The Reactor Protection System shall have provisions in the control room for manually initiating the reactor trip or actuating engineered safety features systems. (References 6.1.002, 6.7.014)

2.2.6 Instrument channels 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)

2.2.7 Auxiliary devices that are required to operate on an ESFAS actuation to support train-related functions, shall be supplied from the same distribution panel to prevent the loss of electric power in one protection set from causing the loss of equipment in the redundant protection set. (References 6.1.023, 6.4.081, 6.4.091)

2.2.8 Each distribution panel shall have access to its respective inverter and standby power supply. (References 6.1.023, 6.4.081, 6.4.091)

2.2.9 A protective action at the system level, once initiated, shall go to completion. Actuation is sealed-in until manually removed from operation. (Reference 6.7.014)

2.2.10 Each RPS actuation shall be alarmed and annunciated in the control room. (Reference 6.7.014)

2.2.11 Protection interlocks and bypasses shall be designed to meet the requirements of IEEE 279-1971 Sections 4.12 through 4.14. (Reference 6.7.014)

2.2.12 The RPS shall be capable of testing at power and shall follow the guide lines of Regulatory Guide 1.22, IEEE 338-1971, and IEEE 279-1971 Section 4.10. (References 6.7.014, 6.7.015, 6.7.032)

2.2.13 ESF shall be tested either in "GO TEST" mode or in "BLOCK" test mode. In "GO TEST" mode the ESF device is operated, or equipment alignments for special operation are performed. In the "BLOCK" test mode, where the end device test would cause plant upset, the end device actuation is blocked while the "ACTUATION" signal is verified by continuity check. Typical examples are Feedwater control valves, steam line isolation valves, RCP breaker test, etc. (References 6.7.003, 6.7.011, 6.7.032)

2.2.14 Electrical or mechanical interlocks and bypasses on safety-related equipment, when initiated manually or automatically, shall be continuously indicated in the main control room. No more than one train or channel may be bypassed at one time. (Reference 6.7.014)

2.2.15 Except as noted below, all reactor trip and safeguards actuation channels 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 may 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

(References 6.1.004, 6.1.011, 6.1.022, 6.4.007, 6.7.014, 6.7.045, 6.7.046)

2.2.16 Channel independence shall be required throughout the system, extending from the sensor through the devices actuating the protective function. Redundant logic system cabinets shall be maintained and separated from the analog channels. Reactor trip and ESFAS analog circuits may be routed in the same raceway if the circuits have the same power supply and sensor protection channel set. Cabinet separation criteria shall be verified by tests. (References 6.1.011, 6.1.042, 6.1.046, 6.4.081, 6.4.091, 6.7.001, 6.7.002, 6.7.003, 6.7.014, 6.7.054)

2.2.17 The system shall have functional diversity. As an example, for a loss-of-coolant accident, a safety injection signal can be obtained manually or by automatic initiation from two diverse parameter measurements. These are:

1. Low pressurizer pressure

DG03
EE05
LB06
1D 120V VITAL AC DIST PNL **AB-121'** **D177025**

BKR	TPNS	DESCRIPTION	SEE PAGE
	Q1R21L0001D-4	**1D 120V VITAL AC INST DISTRIBUTION PANEL CH 4**	
1D-MAIN	**Q1R21E0009D-4**	**MAIN BREAKER <<< INVERTER 1D <<< LB06 (PREFERRED) OR BKR 1H-04 (ALT) ON 1H 208/120V REG AC DIST PANEL**	
1D-01	Q1H11NGNIS2503D-4	1D NUCLEAR INSTRUMENTATION SYSTEM CABINET CHANNEL 4	
1D-02	Q1H11NGNIS2503D-4	1D NUCLEAR INSTRUMENTATION SYSTEM CABINET CHANNEL 4	
1D-03	**Q1H11NGPIC2505D-4**	**PROCESS I&C PROTECTION CAB #4 CHANNEL 4 >>>**	G-40
1D-04	**Q1H11NGPIC2505H-4**	**PROCESS I&C CONTROL CAB #8 CHANNEL 4 >>>**	G-41
1D-05	N1C56L0001B-N	1B INCORE T/C COLD REF CONN BOX	
1D-06	**N1G22NBWPP2603C-N**	**WASTE GAS PROCESSING PANELS >>>**	G-43
1D-07	Q1H11NGSSP2506K-A	TRAIN A SOLID STATE PROT SYSTEM INPUT CAB CH 4	
1D-08	Q1H11NGSSP2506G-B	TRAIN B SOLID STATE PROT SYSTEM INPUT CAB CH 4	
1D-09	------------	SPARE	
1D-10	**N1H11NGAR2506E-B**	**AUX RELAY RACK TRAIN B >>>**	G-44
1D-11	**N1H11NGMCB2500A-AB**	**MAIN CONTROL BOARD SECTION A >>>**	G-45
1D-12	N1H25L0041A-N	1A CONTROL BOARD DEMULTIPLEXER CABINET	
1D-13	**Q1H11NGASC2506C-B**	**TRAIN B AUX SAFEGUARD CAB >>>**	G-46
1D-14	**N1G12NBWPP2603A-N**	**BORON WASTE GAS PROCESSING PANEL >>>**	G-47
	N1G21NBWPP2603B-N	**LIQUID WASTE PROCESSING PANEL >>>**	G-48
1D-15	------------	SPARE	

DG03
EE05
LB06
1D 120V VITAL AC DIST PNL **AB-121'** **D177025**
(CONT'D)

BKR	**TPNS**	**DESCRIPTION**
1D-16	Q1H11NGSSP2506I -B	TRAIN B SOLID STATE PROTECTION SYS OUTPUT CAB
1D-17	-------------	SPARE
1D-18	Q1H11NGSSP2506J -B	TRAIN B SOLID STATE PROTECTION SYS SAFEGUARDS TEST CAB

slave relays don't start
B trn ESF components without
1D 120V vital AC power

21. 015AK2.10 001/NEW/RO/C/A - 2.8/2.8/N/N/2/CVR/Y

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 flow rates at 1010?

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

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 flow in the 1A & 1B loops is greater than 100%, but this is plausible since the amps in the 1A & 1B pumps are unchanged. The increased flow is due to decreased resistance to flow downstream of these pumps which is why flow increases with no increased amps to the pump motors. The second part is correct (see B).

D - Incorrect. Both parts incorrect but plausible. This would seem to be indicated by the amps and the fact that the 1C pump is not pumping any flow due to being tripped by annunciator indication. However the piping system of the RCS allows back flow into the C loop in this condition and it 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).

will effectively increase the unaffected SG's steaming rate and power output to compensate for the reduced steaming rate from the affected SG.

During the transient, the affected SG main feed regulating or bypass feed regulating valve controller(s) is in automatic. Initially, the regulating valve will go shut in response to the reduced steam flow from the affected SG. This causes SG level in the affected SG to begin to fall. Contributing to the drop in SG level is the phenomenon of SG shrink. Shrink is caused by the density change in the tube section of the SG, which occurs as a result of the drop in temperature on the primary side of the affected SG. The drop in indicated level causes the feed valves to reopen fully in an attempt to bring SG level back up. Overfeeding of the affected SG can occur, which could lead to a turbine trip and SG feed pump (SGFP) trips followed by a reactor trip. The turbine trip and SGFP trip occur at 82% SG level (P-14). To minimize the effects a loss of coolant flow has on the affected SG level, the operator is instructed to take manual control of the affected SGs feed regulating valves.

Another concern on any loss of coolant flow situation is pressurizer pressure control. Automatic control of pressurizer pressure will be affected due to the loss of spray flow if the loss of coolant flow occurs in loops A and/or B. If only one loop is involved, the affected loop's spray valve controller should be placed in MANUAL and the valve closed to prevent spray flow from the unaffected loop bypassing the pressurizer. If both A and B loops are affected, auxiliary spray flow should be utilized if normal letdown is available.

The loop flow indications observed by the operators would be as follows: For the affected loop, flow would slowly decrease to 0 and then return to approximately 10%; for the unaffected loops, the flow should increase to approximately 105% (each loop). The flow indication in the idle loop occurs as flow stops and then begins again in the reverse direction. Since flow rates in the RCS loops are derived from the differential pressure felt in an elbow in each loop, any flow at all will be indicated, regardless of the direction. The indication observed in the two loops with the running pumps is due simply to the pumps in those loops picking up a small portion of the flow lost in the idle loop.

22. 022A1.01 001/NEW/RO/MEM 3.6/3.7/N/N/3/CVR/Y

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 IAW SOP-12.1.

Which one of the following identifies the:

1) **MINIMUM** temperature at which a Technical Specification action statement must be entered for Tech Spec 3.6.5, Containment Temperature,

and

2) the speed of the Containment Cooling Fans IAW SOP-12.1?

	(1)	(2)
A.	110 °F	FAST
B.	110 °F	SLOW
C.✓	120 °F	FAST
D.	120 °F	SLOW

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
<u>OPERATOR ACTION</u>
5. <u>IF</u> containment average air temperature is greater than 110°F, <u>THEN</u> 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

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 120°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	24 hours

LOCATION BB3

OPERATOR ACTION (continued)

4. IF possible, THEN start additional fan coolers to insure that the containment average air temperature does not exceed 120°F.
5. IF containment average air temperature is greater than 110°F, THEN verify Containment Dome Recirc Fans in service on fast speed.
6. Refer to FNP-2-SOP-12.1, CONTAINMENT AIR COOLING SYSTEM, for guidance on operating Q2P16MOV3024A (B, C, D), EMERG SW FROM 2A (B, C, D) CTMT CLR.
7. Refer to the Technical Specifications, LCO 3.6.5, for checking Containment average air temperature and LCO Requirements.

References: A-207100, Sh. 98; B-205968; D-205010, Sh. 1; Technical Specifications; PCN B-86-2-3923; MDFD 88-1907; PCN B91-2-7619; PCN B91-2-7574, U-280362 (Vendor manual for N2T12TRSH3188)

NOTE: 2A CTMT CLR fan fast speed breaker EA-10-2 is interlocked with tie breaker EA08-2 to prevent starting if the emergency section of 2A 600V LC is aligned to 2D 600V LC (EA08-2 open). DCP B87-2-4592

4.1.6 Start 2A, 2B, 2C, and 2D containment coolers in FAST (SLOW) speed.

- 2A containment cooler
- 2B containment cooler
- 2C containment cooler
- 2D containment cooler

4.1.7 Check CTMT CLR 2A(2B,2C,2D) DISCH 3186A(B,C,D) OPEN light illuminated.

- CTMT CLR 2A DISCH 3186A
- CTMT CLR 2B DISCH 3186B
- CTMT CLR 2C DISCH 3186C
- CTMT CLR 2D DISCH 3186D

4.1.8 Place 2A, 2B, 2C, and 2D CTMT DOME RECIRC FANS in HIGH (LOW) speed.

- 2A CTMT DOME RECIRC FAN
- 2B CTMT DOME RECIRC FAN
- 2C CTMT DOME RECIRC FAN
- 2D CTMT DOME RECIRC FAN

4.1.9 Operate the containment dome recirculation fans and containment coolers as necessary to maintain containment temperature below 120°F. (See section 4.7 for shifting containment cooler fan speeds.)

4.1.10 Open 2A and 2B RX CAV CLG DMPRS

- 2A RX CAV CLG DMPR Q2E12HV3999A
- 2B RX CAV CLG DMPR Q2E12HV3999B

CAUTION: EMERG SW FROM 2A(2B,2C,2D) CTMT CLR valves Q2P16MOV3024A(B,C,D) [Q2P16V043A(B,C,D)] are normally maintained closed, BUT may be opened for temperature control. However, operation with these valves open should be minimized to reduce the potential for long term degradation to the containment coolers from the higher flow rates. Therefore, the Operations Shift Manager should be consulted prior to opening these valves.

4.1.11 IF necessary to maintain containment temperature below 120°F, THEN open and caution tag EMERG SW FROM 2A(2B,2C,2D) CTMT CLR valves, as desired.

- EMERG SW FROM 2A CTMT CLR VALVES Q2P16MOV3024A
- EMERG SW FROM 2B CTMT CLR VALVES Q2P16MOV3024B
- EMERG SW FROM 2C CTMT CLR VALVES Q2P16MOV3024C
- EMERG SW FROM 2D CTMT CLR VALVES Q2P16MOV3024D

4.1.12 WHEN additional service water flow through the containment coolers is no longer necessary for temperature control, THEN remove caution tag(s) and close EMERG SW FROM 2A(2B,2C,2D) CTMT CLR valves:

- EMERG SW FROM 2A CTMT CLR valves Q2P16MOV3024A
- EMERG SW FROM 2B CTMT CLR valves Q2P16MOV3024B
- EMERG SW FROM 2C CTMT CLR valves Q2P16MOV3024C
- EMERG SW FROM 2D CTMT CLR valves Q2P16MOV3024D

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

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. I 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. I 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. I 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).

Section: Procedure

Unit 1 ERP Step: 8 **Unit 2 ERP Step:** 8 ***ERG Step No:*** *8*

ERP StepText: Isolate RCP seals using ATTACHMENT 3.

ERG StepText: *Dispatch Personnel To Locally Close Valves To Isolate RCP Seals And Place Valve Switches In CLOSED Position*

Purpose: To isolate the RCP seals

Basis: This step groups three actions, with different purposes, aimed at isolating the RCP seals. The actions are grouped since all require an auxiliary operator, dispatched from the control room, to locally close containment isolation valves (the reference plant utilizes motor operated valves for the RCP seal return, RCP thermal barrier CCW return lines and RCP seal injection lines). This grouping assumes that the subject valves are located in the same penetration room area and that they are accessible. Concurrent with dispatching the auxiliary operator, the control room operator should place the valve switches for the motor operated valves in the closed position so that the valves remain closed when ac power is restored. Isolating the seal return line prevents seal leakage from filling the volume control tank (VCT) (via seal return relief valve outside containment) and subsequent transfer to other auxiliary building holdup tanks (via VCT relief valve) with the potential for radioactive release within the auxiliary building. Such a release, without auxiliary building ventilation available, could limit personnel access for local operations. Isolating the seal return line also enables pressure in the number 1 seal leakoff line to increase up to the relief valve setpoint of 150 psig. Maintaining a backpressure in the seal leakoff line of at least 150 psig enables development of high pressure in the number 1 seal leakoff cavity with a steady-state seal leakage rate established due to the self-limiting leakage characteristic of the number 1 seal. Under these conditions, with the number 1 seal functioning as expected and the number 2 seal remaining closed, the expected leakage flow rate is 21.1 gpm/pump. This is consistent with the steady-state pressure distribution and seal leakage determined in the WCAP-10541 analysis and used in the latest RCP seal leakage PRA model in WCAP-15603. Isolating the RCP seal injection lines prepares the plant for recovery while protecting the RCPs from seal and shaft damage that may occur when a charging/SI pump is started as part of the recovery. With the RCP seal STEP DESCRIPTION TABLE FOR ECA-0.0Step8 injection lines isolated, a charging/SI pump can be started in the normal charging mode without concern for cold seal injection flow thermally shocking the RCPs. Seal injection can subsequently by ~~established to the RCP consistent with appropriate plant specific procedures.~~ 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

Section: Procedure

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.

A & B 1st
part correct.

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." 3. Time-critical actions of this step (for example, local seal return isolation) may be located earlier in the guideline if necessary to meet individual plant capabilities and requirements. Note that the Step Sequence Requirements allow interchangeability between Steps 6, 7 and 8 of this guideline.

B & D
2nd part
correct

References:

Justification of Differences:

1 Placed actions in an Attachment to allow an extra operator to perform required actions outside of the control room without interfering with the flow of the procedure.

3.6 DO NOT attempt to start a RCP unless its oil lift pump has been delivering oil to the upper thrust shoes for at least two minutes. Observe the oil lift pumps indicating lights to verify correct oil pump motor operation and oil pressure. The oil lift pumps should run at least 1 minute after the RCP's are started. An interlock will prevent starting a RCP until 600 psig oil pressure is established.

3.7 Shift Supervisor's approval must be obtained prior to removing any seal wires or changing the position of any throttle valves.

3.8 RCP seal water injection flow of 6 gpm or CCW to the RCP thermal barrier must be continuously supplied when RCS temperature exceeds 150°F.

A 4C 2nd Part incorrect

3.9 Maintain RCP CCW and seal injection water supply temperature less than 105°F and 130°F respectively.

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).

3.11 For RCP operations, a minimum pressure differential of 200 psid must be maintained across RCP No. 1 seals.

3.12 The following precautions apply in the case of a RCP #1 seal failure.

3.12.1 DO NOT restart an RCP with an indicated No. 1 seal failure.

3.12.2 Refer to FNP-1-ARP-1.4, MAIN CONTROL BOARD ANNUNCIATOR PANEL "D", for guidance if No. 1 seal leakoff flow is abnormally low (Ann. DC1) or abnormally high (Ann. DC2).

3.13 The No. 1 seal bypass valve should NOT be opened unless either the pump bearing temperature (seal inlet temperature) or the No. 1 seal leakoff temperature approaches its alarm level. The No. 1 seal bypass valve should then be opened only if all of the following conditions are met:

3.13.1 Reactor coolant system pressure is greater than 100 PSIG AND less than 1000 PSIG.

3.13.2 No. 1 seal leakoff valve is open.

3.13.3 No. 1 seal leakoff flowrate is less than one GPM.

3.13.4 Seal injection water flow rate to each pump is greater than six GPM.

3.14 For RCP operations, the required minimum back pressure of 15 psig on the RCP No. 1 seals is ensured by maintaining a pressure of at least 18 psig in the VCT.

These requirements are to be satisfied assuming the nominal excess letdown flow rate (12,400 lbm/hr), normal RCP No. 1 seal leakage (3 gpm) from each RCP, 60 gpm recirculation flow from one charging pump, and normal VCT pressure. (Reference 6.2.1)

2.2.2.15 The CVCS is required to makeup for shrinkage during a 100 degree F/hr cooldown of the RCS from hot zero power to 350°F. This is considered to be an original design basis function of the CVCS.

2.2.3 Seal Injection and Leakoff

2.2.3.1 The CVCS is required to cool excess letdown, RCP No. 1 seal leakage, and recirculation flow from at least one charging pump to 115°F prior to returning the flow to the charging pump suction.

The cooling function is performed by the excess letdown heat exchanger and the seal water return heat exchanger. The 115°F temperature is based upon the RCP seal injection temperature limit of 130°F. (References 6.2.44, 6.2.7 and 6.2.8)

2.2.3.2 The CVCS is required to provide a seal water injection flow rate adjustable over a normal range of 8 to 13 gpm to each RCP No. 1 seal. It is required that suspended solid particles larger than 5 microns in size be removed from the injection stream.

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)

2.2.3.3 The CVCS is required to provide a means for cooling the RCP lower bearing under low RCS pressure conditions when the RCP No. 1 seal leakoff flow is less than 1.0 gpm.

24. 026AA1.01 001/NEW/RO/C/A 3.1/3.1/N/N/2/CVR/Y

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

At 1000:
- B Train CCW is on service, and aligned for split train operation.
- A loss of 1L 4160V Bus has occurred.

At 1015:
- RCP motor bearing temperatures are:
  - 1A: 172°F and rising
  - 1B: 165°F and rising
  - 1C: 197°F and rising

At 1020:
- RCP bearing temperatures are:
  - 1A: 230°F and rising
  - 1B: 221°F and rising
  - 1C: 240°F and rising

Which one of the following states:

1) the procedure(s) that must be entered,

and

2) the **EARLIEST** time that a reactor trip is required based on RCP Bearing Temperatures?

	(1)	(2)
A.	AOP-10.0 **ONLY**	1015
B.	AOP-10.0 **ONLY**	1020
C✓	Both AOP-9.0 **AND** 10.0	1015
D.	Both AOP-9.0 **AND** 10.0	1020

A - Incorrect. First part is incorrect since entry conditions are met for both AOP-10 & AOP-9.0. They would be done in parallel. AOP-10 would direct AOP-9 to be entered if it was not entered directly. AOP-10 ONLY is plausible, since the loss of SW is the initiating event and AOP-10 would deal with it. Second part is correct since the temperature for requiring a reactor trip on high RCP Motor bearing temperatures (195°F) per step 2 of AOP-9.0 have been exceeded.

B - Incorrect. The first part is incorrect (see A). The second part is incorrect (see A). Plausible, since per AOP-4.1, Abnormal Reactor Coolant Pump Seal Leakage, there is a RCP trip criteria for "CHECK RCP lower seal water bearing and seal water outlet temperatures stabilize less than 225°F". Confusion may exist between the two temperatures at which to trip the Reactor and RCP: 195 & 225.

C - Correct. The AOP-10 entry is required per the loss of the B train SW Buss, and loss of B train SW, per the entry conditions. AOP-10 will also direct entry into AOP-9 further into the procedure, and the entry conditions for AOP-9, Loss of CCW, directs entry into AOP-9.0 for a loss of SW supplying an operating CCW train. Second part is correct since 1015 is the earliest that 195°F RCP bearing temperature is exceeded, and this is the requirement for a reactor trip per the continuing action STEP 2 and associated note in AOP-9.0. Note in AOP-9 also states that if flow is not adequate to maintain temperatures, trip the reactor.

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

AOP-9.0 Ver. 22
AOP-10.0 Ver. 15

FNP-1-AOP-4.1, Abnormal Reactor Coolant Pump Seal Leakage, Version 5.0
__ 7 CHECK RCP lower seal water bearing
and seal water outlet temperatures
stabilizes less than 225°F.

FNP-1-AOP-9.0, Loss Of Component Cooling Water, Version 22.0
• Check RCP motor bearing temperatures less than 195°F.

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 temperatures and monitor the temperatures (i.e which procedure(s) is(are) entered for this condition, and at which temperatures a reactor trip is required).

Objective:

2. **EVALUATE** plant conditions and **DETERMINE** if entry into AOP-10.0, Loss of Service Water is required. (OPS-52520J02)
4. **LIST AND DESCRIBE** the sequence of major actions associated with AOP-10.0, Loss of Service Water. (OPS-52520J04).
5. **EVALUATE** plant conditions and **DETERMINE** if any system components need to be operated while performing AOP-10.0, Loss of Service Water. (OPS-52520J06).

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).

A. Purpose

This procedure provides actions for response to a loss of one or both trains of service water.

This procedure is applicable at all times.

B. Symptoms or Entry Conditions

I. This procedure is entered when a loss of either train of service water is indicated by any of the following:

a. Actuation of SW PRESS A TRN LO annunciator AD4 or SW PRESS B TRN LO annunciator AD5 (60 psig)

b. Actuation of SW PUMP TRIPPED annunciator AE4

c. Actuation of SW TO AUX BLDG HDR PRESS A OR B TRN LO annunciator AE5 (50 psig)

d. Trip of any operating SW PUMP

e. Rising temperatures on components supplied by service water

f. Loss of power to one or both SW 4160 V busses 1K or 1L

C+D
Correct

A. Purpose

This procedure provides actions for response to a loss of an operating component cooling water train.

This procedure is applicable at all times.

B. Symptoms or Entry Conditions

I. This procedure is entered when a loss of component cooling water is indicated by any of the following:

a. Trip of any operating CCW PUMP

b. Loss of SW supply to an operating CCW train

P

C & D correct

Step	Action/Expected Response	Response Not Obtained

NOTE:
- Step 2 is a continuing action step.
- IF RCP motor bearing temperatures exceed 195°F, THEN the ON SERVICE train is affected.
- Adequate CCW flow means sufficient cooling is available to maintain acceptable temperatures.(i.e. charging pumps, RHR cooling, SFP cooling, RCP's etc.)
- Indications of pump cavitation are: Abnormal CCW flow oscillations or cavitation noise reported at the pump.

Step	Action/Expected Response	Response Not Obtained
__ 2	**Check CCW system adequate for continued plant support.**	2 Perform the following:
	• Check CCW flow adequate in affected train. • Check RCP motor bearing temperatures less than 195°F. • Check CCW pump not cavitating. Stop any cavitating CCW pump. • CCW Surge tank level being maintained at or above 13 inches.	2.1 IF the ON SERVICE train is affected, THEN perform the following: 2.1.1 IF the reactor is critical, THEN trip the reactor and perform, FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure. 2.1.2 Verify all Reactor Coolant pumps stopped. 2.1.3 IF in Mode 3 or 4, THEN perform FNP-1-AOP-4.0, LOSS OF REACTOR COOLANT FLOW while continuing with procedure.

A & B 2nd parts correct

NOTE: Indications of CCW pump cavitation will be abnormal CCW flow oscillations or cavitation noise reported at the pump.

2.2 IF evidence of CCW pump cavitation exists, THEN stop affected CCW pump.

Step 2 continued on next page

__Page Completed

Step	Action/Expected Response	Response Not Obtained

****

CAUTION: To prevent potential seal damage, neither seal injection nor CCW cooling should be restored to a RCP which has lost both seal injection and CCW cooling.

****

NOTE: Refer to the Integrated Plant Computer page 1RCP, RCP Temperature Summary, for RCP seal water temperatures.

__ 7 **CHECK RCP lower seal water bearing and seal water outlet temperatures stabilizes less than 225°F.**

Monitor the following computer points for the affected pump.

- [ ] TE 132 RCP A SEAL WATER OUTLET TEMP
- [ ] TE0131 RCP A LOWER SEAL WATER BRG TEMP
- [ ] TE0129 RCP B SEAL WATER OUTLET TEMP
- [ ] TE 128 RCP B LOWER SEAL WATER BRG TEMP
- [ ] TE0126 RCP C SEAL WATER OUTLET TEMP
- [ ] TE0125 RCP C LOWER SEAL WATER BRG TEMP

7 Perform the following:

7.1 Shutdown the affected reactor coolant pump as follows:

7.1.1 Manually trip the reactor, AND go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION. {CMT 0003908}

7.1.2 WHEN the reactor is shutdown, THEN stop the affected RCP(s). {CMT 0003908}

7.2 IF 1A OR 1B RCP is secured, THEN close the pressurizer spray valve for the affected RCP.

- [ ] PK444C for 1A RCP
- [ ] PK444D for 1B RCP

Step 7 continued on next page

__Page Completed

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

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

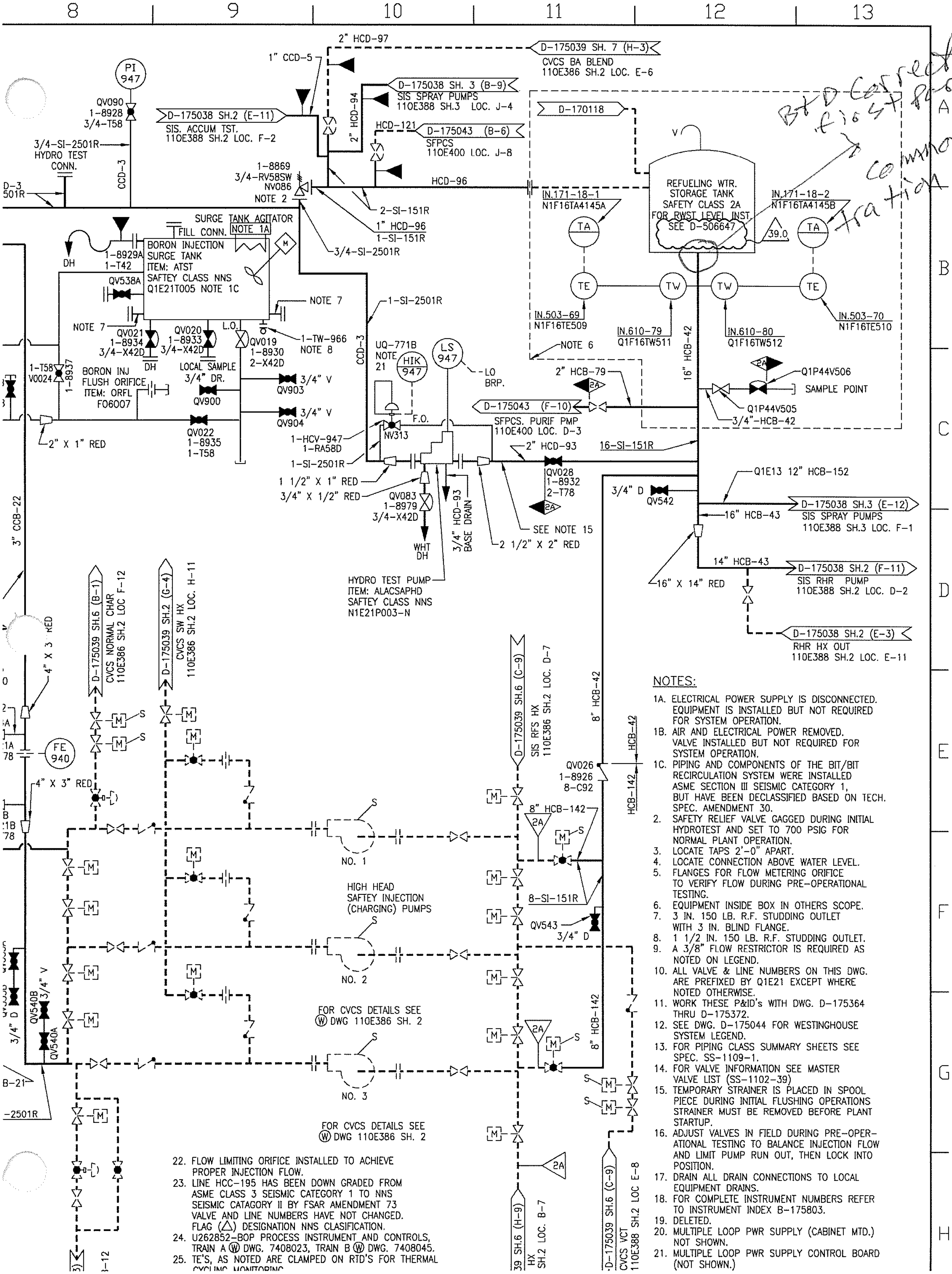
026K1.01
026 Containment Spray System
K1 Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
K1.01 ECCS 4.2 4.2

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.



NOTES:

1A. ELECTRICAL POWER SUPPLY IS DISCONNECTED. EQUIPMENT IS INSTALLED BUT NOT REQUIRED FOR SYSTEM OPERATION.
1B. AIR AND ELECTRICAL POWER REMOVED. VALVE INSTALLED BUT NOT REQUIRED FOR SYSTEM OPERATION.
1C. PIPING AND COMPONENTS OF THE BIT/BIT RECIRCULATION SYSTEM WERE INSTALLED ASME SECTION III SEISMIC CATEGORY 1, BUT HAVE BEEN DECLASSIFIED BASED ON TECH. SPEC. AMENDMENT 30.
2. SAFETY RELIEF VALVE GAGGED DURING INITIAL HYDROTEST AND SET TO 700 PSIG FOR NORMAL PLANT OPERATION.
3. LOCATE TAPS 2'-0" APART.
4. LOCATE CONNECTION ABOVE WATER LEVEL.
5. FLANGES FOR FLOW METERING ORIFICE TO VERIFY FLOW DURING PRE-OPERATIONAL TESTING.
6. EQUIPMENT INSIDE BOX IN OTHERS SCOPE.
7. 3 IN. 150 LB. R.F. STUDDING OUTLET WITH 3 IN. BLIND FLANGE.
8. 1 1/2 IN. 150 LB. R.F. STUDDING OUTLET.
9. A 3/8" FLOW RESTRICTOR IS REQUIRED AS NOTED ON LEGEND.
10. ALL VALVE & LINE NUMBERS ON THIS DWG. ARE PREFIXED BY Q1E21 EXCEPT WHERE NOTED OTHERWISE.
11. WORK THESE P&ID's WITH DWG. D-175364 THRU D-175372.
12. SEE DWG. D-175044 FOR WESTINGHOUSE SYSTEM LEGEND.
13. FOR PIPING CLASS SUMMARY SHEETS SEE SPEC. SS-1109-1.
14. FOR VALVE INFORMATION SEE MASTER VALVE LIST (SS-1102-39)
15. TEMPORARY STRAINER IS PLACED IN SPOOL PIECE DURING INITIAL FLUSHING OPERATIONS STRAINER MUST BE REMOVED BEFORE PLANT STARTUP.
16. ADJUST VALVES IN FIELD DURING PRE-OPERATIONAL TESTING TO BALANCE INJECTION FLOW AND LIMIT PUMP RUN OUT, THEN LOCK INTO POSITION.
17. DRAIN ALL DRAIN CONNECTIONS TO LOCAL EQUIPMENT DRAINS.
18. FOR COMPLETE INSTRUMENT NUMBERS REFER TO INSTRUMENT INDEX B-175803.
19. DELETED.
20. MULTIPLE LOOP PWR SUPPLY (CABINET MTD.) NOT SHOWN.
21. MULTIPLE LOOP PWR SUPPLY CONTROL BOARD (NOT SHOWN.)
22. FLOW LIMITING ORIFICE INSTALLED TO ACHIEVE PROPER INJECTION FLOW.
23. LINE HCC-195 HAS BEEN DOWN GRADED FROM ASME CLASS 3 SEISMIC CATEGORY 1 TO NNS SEISMIC CATAGORY II BY FSAR AMENDMENT 73 VALVE AND LINE NUMBERS HAVE NOT CHANGED. FLAG (△) DESIGNATION NNS CLASIFICATION.
24. U262852-BOP PROCESS INSTRUMENT AND CONTROLS, TRAIN A Ⓦ DWG. 7408023, TRAIN B Ⓦ DWG. 7408045.
25. TE'S, AS NOTED ARE CLAMPED ON RTD'S FOR THERMAL CYCLING MONITORING

a-r

3.3.3.2 The tank is designed according to the requirements of the ASME Boiler and Pressure Vessel Code Section III, Class 2. (Reference 6.1.29)

3.3.3.3 Level Indication complies with the requirements of IEEE-279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations. (Reference 6.1.33)

3.3.3.4 Level Indication complies with the requirements of IEEE-308-1969, Criteria for Class 1E Power Systems for Nuclear Power Generating Stations. (Reference 6.1.33)

3.3.4 I&C Requirements

3.3.4.1 The RWST must be provided with 2 level transmitters. One channel provides high, Technical Specification minimum, low, and low-low level alarms. The other channel provides low and low-low alarms. These channels provide the following alarm functions (References 6.2.14, 6.4.2, 6.7.79 and 6.7.80):

a. High Level Alarm - This alarm alerts the operator to a high level which could lead to tank overflow.

b. Technical Specification Minimum Level Alarm - Indicates that the RWST level is less than that required by the Technical Specifications during normal operations.

c. Lo Level Alarm - Alerts the operator that the ECCS pumps should be manually aligned for recirculation phase operation.

d. Lo-Lo Level Alarm - Alerts the operator that the CS pumps should be manually aligned for the recirculation phase. If an SI signal is present, this alarm will automatically open RHR/LHSI pump sump suction valves 8811A & B and 8812A & B. (References 6.7.43 [Section 6.3.2] and 6.7.42 [Section 6.3.2.4])

B & C incorrect for CS suctions

3.3.4.2 One of the two redundant level indicators in the control room must be operable for Post Accident Monitoring. (References 6.1.18, 6.7.15 and 6.7.16)

26. 027AK1.02 001/NEW/RO/MEM 2.8/3.1/N/N/3/CVR/VER 5 EDITORIAL

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:

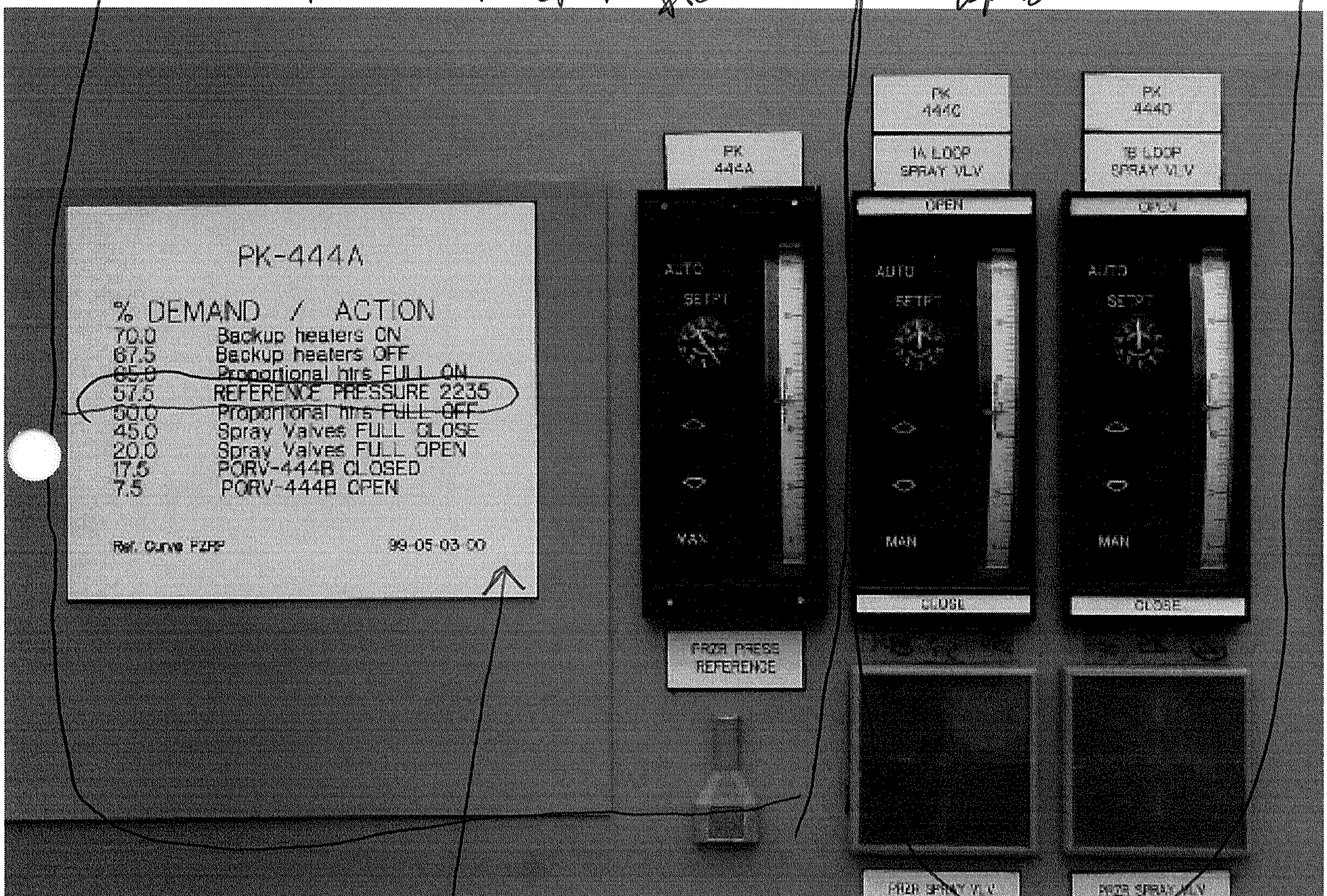
2. **RELATE AND IDENTIFY** the operational characteristics including design features, capacities and protective interlocks for the following components associated with the Pressurizer Pressure and Level Control System to include those items found on Figure 2, Pressurizer and Pressure Relief Tank, Figure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

Master controller: Demand goes down to lower press & up to raise press

Slave (spray Valve) controllers: Demand goes down to raise press & up to lower press



PT-444 failed to 2230 < ref. press 2235 =>
PK-444A demand goes up to raise press

C&D 1st parts correct

Table 2: Saturated Steam: Pressure Table

Abs Press. Lb/Sq In. p	Temp Fahr t	Specific Volume Sat. Liquid v_f	Evap v_{fg}	Sat. Vapor v_g	Enthalpy Sat. Liquid h_f	Evap h_{fg}	Sat. Vapor h_g	Entropy Sat. Liquid s_f	Evap s_{fg}	Sat. Vapor s_g	Abs Press. Lb/Sq In. p
0.08865	32.018	0.016022	3302.4	3302.4	0.0003	1075.5	1075.5	0.0000	2.1872	2.1872	0.08865
0.25	59.323	0.016032	1235.5	1235.5	27.382	1060.1	1087.4	0.0542	2.0425	2.0967	0.25
0.50	79.586	0.016071	641.5	641.5	47.623	1048.6	1096.3	0.0925	1.9446	2.0370	0.50
1.0	101.74	0.016136	333.59	333.60	69.73	1036.1	1105.8	0.1326	1.8455	1.9781	1.0
5.0	162.24	0.016407	73.515	73.532	130.20	1000.9	1131.1	0.2349	1.6094	1.8443	5.0
10.0	193.21	0.016592	38.404	38.420	161.26	982.1	1143.3	0.2836	1.5043	1.7879	10.0
14.696	212.00	0.016719	26.782	26.799	180.17	970.3	1150.5	0.3121	1.4447	1.7568	14.696
15.0	213.03	0.016726	26.274	26.290	181.21	969.7	1150.9	0.3137	1.4415	1.7552	15.0
20.0	227.96	0.016834	20.070	20.087	196.27	960.1	1156.3	0.3358	1.3962	1.7320	20.0
30.0	250.34	0.017009	13.7266	13.7436	218.9	945.2	1164.1	0.3682	1.3313	1.6995	30.0
40.0	267.25	0.017151	10.4794	10.4965	236.1	933.6	1169.8	0.3921	1.2844	1.6765	40.0
50.0	281.02	0.017274	8.4967	8.5140	250.2	923.9	1174.1	0.4112	1.2474	1.6586	50.0
60.0	292.71	0.017383	7.1562	7.1736	262.2	915.4	1177.6	0.4273	1.2167	1.6440	60.0
70.0	302.93	0.017482	6.1875	6.2050	272.7	907.8	1180.6	0.4411	1.1905	1.6316	70.0
80.0	312.04	0.017573	5.4536	5.4711	282.1	900.9	1183.1	0.4534	1.1675	1.6208	80.0
90.0	320.28	0.017659	4.8779	4.8953	290.7	894.6	1185.3	0.4643	1.1470	1.6113	90.0
100.0	327.82	0.017740	4.4133	4.4310	298.5	888.6	1187.2	0.4743	1.1284	1.6027	100.0
110.0	334.79	0.01782	4.0306	4.0484	305.8	883.1	1188.9	0.4834	1.1115	1.5950	110.0
120.0	341.27	0.01789	3.7097	3.7275	312.6	877.8	1190.4	0.4919	1.0960	1.5879	120.0
130.0	347.33	0.01796	3.4364	3.4544	319.0	872.8	1191.7	0.4998	1.0815	1.5813	130.0
140.0	353.04	0.01803	3.2010	3.2190	325.0	868.0	1193.0	0.5071	1.0681	1.5752	140.0
150.0	358.43	0.01809	2.9958	3.0139	330.6	863.4	1194.1	0.5141	1.0554	1.5695	150.0
160.0	363.55	0.01815	2.8155	2.8336	336.1	859.0	1195.1	0.5206	1.0435	1.5641	160.0
170.0	368.42	0.01821	2.6556	2.6738	341.2	854.8	1196.0	0.5269	1.0322	1.5591	170.0
180.0	373.08	0.01827	2.5129	2.5312	346.2	850.7	1196.9	0.5328	1.0215	1.5543	180.0
190.0	377.53	0.01833	2.3847	2.4030	350.9	846.7	1197.6	0.5384	1.0113	1.5498	190.0
200.0	381.80	0.01839	2.2689	2.2873	355.5	842.8	1198.3	0.5438	1.0016	1.5454	200.0
210.0	385.91	0.01844	2.16373	2.18217	359.9	839.1	1199.0	0.5490	0.9923	1.5413	210.0
220.0	389.88	0.01850	2.06779	2.08629	364.2	835.4	1199.6	0.5540	0.9834	1.5374	220.0
230.0	393.70	0.01855	1.97991	1.99846	368.3	831.8	1200.1	0.5588	0.9748	1.5336	230.0
240.0	397.39	0.01860	1.89909	1.91769	372.3	828.4	1200.6	0.5634	0.9665	1.5299	240.0
250.0	400.97	0.01865	1.82452	1.84317	376.1	825.0	1201.1	0.5679	0.9585	1.5264	250.0
260.0	404.44	0.01870	1.75548	1.77418	379.9	821.6	1201.5	0.5722	0.9508	1.5230	260.0
270.0	407.80	0.01875	1.69137	1.71013	383.6	818.3	1201.9	0.5764	0.9433	1.5197	270.0
280.0	411.07	0.01880	1.63169	1.65049	387.1	815.1	1202.3	0.5805	0.9361	1.5166	280.0
290.0	414.25	0.01885	1.57597	1.59482	390.6	812.0	1202.6	0.5844	0.9291	1.5135	290.0
300.0	417.35	0.01889	1.52384	1.54274	394.0	808.9	1202.9	0.5882	0.9223	1.5105	300.0
350.0	431.73	0.01912	1.30642	1.32554	409.8	794.2	1204.0	0.6059	0.8909	1.4968	350.0
400.0	444.60	0.01934	1.14162	1.16095	424.2	780.4	1204.6	0.6217	0.8630	1.4847	400.0
450.0	456.28	0.01954	1.01224	1.03179	437.3	767.5	1204.8	0.6360	0.8378	1.4738	450.0
500.0	467.01	0.01975	0.90787	0.92762	449.5	755.1	1204.7	0.6490	0.8148	1.4639	500.0
550.0	476.94	0.01994	0.82183	0.84177	460.9	743.3	1204.3	0.6611	0.7936	1.4547	550.0
600.0	486.20	0.02013	0.74962	0.76975	471.7	732.0	1203.7	0.6723	0.7738	1.4461	600.0
650.0	494.89	0.02032	0.68811	0.70843	481.9	720.9	1202.8	0.6828	0.7552	1.4381	650.0
700.0	503.08	0.02050	0.63505	0.65556	491.6	710.2	1201.8	0.6928	0.7377	1.4304	700.0
750.0	510.84	0.02069	0.58880	0.60949	500.9	699.8	1200.7	0.7022	0.7210	1.4232	750.0
800.0	518.21	0.02087	0.54809	0.56896	509.8	689.6	1199.4	0.7111	0.7051	1.4163	800.0
850.0	525.24	0.02105	0.51197	0.53302	518.4	679.5	1198.0	0.7197	0.6899	1.4096	850.0
900.0	531.95	0.02123	0.47968	0.50091	526.7	669.7	1196.4	0.7279	0.6753	1.4032	900.0
950.0	538.39	0.02141	0.45064	0.47205	534.7	660.0	1194.7	0.7358	0.6612	1.3970	950.0
1000.0	544.58	0.02159	0.42436	0.44596	542.6	650.4	1192.9	0.7434	0.6476	1.3910	1000.0
1050.0	550.53	0.02177	0.40047	0.42224	550.1	640.9	1191.0	0.7507	0.6344	1.3851	1050.0
1100.0	556.28	0.02195	0.37863	0.40058	557.5	631.5	1189.1	0.7578	0.6216	1.3794	1100.0
1150.0	561.82	0.02214	0.35859	0.38073	564.8	622.2	1187.0	0.7647	0.6091	1.3738	1150.0
1200.0	567.19	0.02232	0.34013	0.36245	571.9	613.0	1184.8	0.7714	0.5969	1.3683	1200.0
1250.0	572.38	0.02250	0.32306	0.34556	578.8	603.8	1182.6	0.7780	0.5850	1.3630	1250.0
1300.0	577.42	0.02269	0.30722	0.32991	585.6	594.6	1180.2	0.7843	0.5733	1.3577	1300.0
1350.0	582.32	0.02288	0.29250	0.31537	592.3	585.4	1177.8	0.7906	0.5620	1.3525	1350.0
1400.0	587.07	0.02307	0.27871	0.30178	598.8	576.5	1175.3	0.7966	0.5507	1.3474	1400.0
1450.0	591.70	0.02327	0.26584	0.28911	605.3	567.4	1172.8	0.8026	0.5397	1.3423	1450.0
1500.0	596.20	0.02346	0.25372	0.27719	611.7	558.4	1170.1	0.8085	0.5288	1.3373	1500.0
1550.0	600.59	0.02366	0.24235	0.26601	618.0	549.4	1167.4	0.8142	0.5182	1.3324	1550.0
1600.0	604.87	0.02387	0.23159	0.25545	624.2	540.3	1164.5	0.8199	0.5076	1.3274	1600.0
1650.0	609.05	0.02407	0.22143	0.24551	630.4	531.3	1161.6	0.8254	0.4971	1.3225	1650.0
1700.0	613.13	0.02428	0.21178	0.23607	636.5	522.2	1158.6	0.8309	0.4867	1.3176	1700.0
1750.0	617.12	0.02450	0.20263	0.22713	642.5	513.1	1155.6	0.8363	0.4765	1.3128	1750.0
1800.0	621.02	0.02472	0.19390	0.21861	648.5	503.8	1152.3	0.8417	0.4662	1.3079	1800.0
1850.0	624.83	0.02495	0.18558	0.21052	654.5	494.6	1149.0	0.8470	0.4561	1.3030	1850.0
1900.0	628.56	0.02517	0.17761	0.20278	660.4	485.2	1145.6	0.8522	0.4459	1.2981	1900.0
1950.0	632.22	0.02541	0.16999	0.19540	666.3	475.8	1142.0	0.8574	0.4358	1.2931	1950.0
2000.0	635.80	0.02565	0.16266	0.18831	672.1	466.2	1138.3	0.8625	0.4256	1.2881	2000.0
2100.0	642.76	0.02615	0.14885	0.17501	683.8	446.7	1130.5	0.8727	0.4053	1.2780	2100.0
2200.0	649.45	0.02669	0.13603	0.16272	695.5	426.7	1122.2	0.8828	0.3848	1.2676	2200.0
* 2300.0	655.89	0.02727	0.12406	0.15133	707.2	406.0	1113.2	0.8929	0.3640	1.2569	2300.0
2400.0	662.11	0.02790	0.11287	0.14076	719.0	384.8	1103.7	0.9031	0.3430	1.2460	2400.0
2500.0	668.11	0.02859	0.10209	0.13068	731.7	361.6	1093.3	0.9139	0.3206	1.2345	2500.0
2600.0	673.91	0.02938	0.09172	0.12110	744.5	337.6	1082.0	0.9247	0.2977	1.2225	2600.0
2700.0	679.53	0.03029	0.08165	0.11194	757.3	312.3	1069.7	0.9356	0.2741	1.2097	2700.0
2800.0	684.96	0.03134	0.07171	0.10305	770.7	285.1	1055.8	0.9468	0.2491	1.1958	2800.0
2900.0	690.22	0.03262	0.06158	0.09420	785.1	254.7	1039.8	0.9588	0.2215	1.1803	2900.0
3000.0	695.33	0.03428	0.05073	0.08500	801.8	218.4	1020.3	0.9728	0.1891	1.1619	3000.0
3100.0	700.28	0.03681	0.03771	0.07452	824.0	169.3	993.3	0.9914	0.1460	1.1373	3100.0
3200.0	705.08	0.04472	0.01191	0.05663	875.5	56.1	931.6	1.0351	0.0482	1.0832	3200.0
3208.2*	705.47	0.05078	0.00000	0.05078	906.0	0.0	906.0	1.0612	0.0000	1.0612	3208.2*

*Critical pressure

* specific volume ↑ => density ↓ (see footer on Table 3)

Table 3. Superheated Steam

Abs Press. Lb/Sq In. (Sat. Temp)		Sat. Water	Sat. Steam	Temperature — Degrees Fahrenheit 200	250	300	350	400	450	500	600	700	800	900	1000	1100	1200
1 (101.74)	Sh			98.26	148.26	198.26	248.26	298.26	348.26	398.26	498.26	598.26	698.26	798.26	898.26	998.26	1098.26
	v	0.01614	333.6	392.5	422.4	452.3	482.1	511.9	541.7	571.5	631.1	690.7	750.3	809.8	869.4	929.0	988.6
	h	69.73	1105.8	1150.2	1172.9	1195.7	1218.7	1241.8	1265.1	1288.6	1336.1	1384.5	1433.7	1483.8	1534.9	1586.8	1639.7
	s	0.1326	1.9781	2.0509	2.0841	2.1152	2.1445	2.1722	2.1985	2.2237	2.2708	2.3144	2.3551	2.3934	2.4296	2.4640	2.4969
5 (162.24)	Sh			37.76	87.76	137.76	187.76	237.76	287.76	337.76	437.76	537.76	637.76	737.76	837.76	937.76	1037.76
	v	0.01641	73.53	78.14	84.21	90.24	96.25	102.24	108.23	114.21	126.15	138.08	150.01	161.94	173.86	185.78	197.70
	h	130.20	1131.1	1148.6	1171.7	1194.8	1218.0	1241.3	1264.7	1288.2	1335.9	1384.3	1433.6	1483.7	1534.7	1586.7	1639.6
	s	0.2349	1.8443	1.8716	1.9054	1.9369	1.9664	1.9943	2.0208	2.0460	2.0932	2.1369	2.1776	2.2159	2.2521	2.2866	2.3194
10 (193.21)	Sh			6.79	56.79	106.79	156.79	206.79	256.79	306.79	406.79	506.79	606.79	706.79	806.79	906.79	1006.79
	v	0.01659	38.42	38.84	41.93	44.98	48.02	51.03	54.04	57.04	63.03	69.00	74.98	80.94	86.91	92.87	98.84
	h	161.26	1143.3	1146.6	1170.2	1193.7	1217.1	1240.6	1264.1	1287.8	1335.5	1384.0	1433.4	1483.5	1534.6	1586.6	1639.5
	s	0.2836	1.7879	1.7928	1.8273	1.8593	1.8892	1.9173	1.9439	1.9692	2.0166	2.0603	2.1011	2.1394	2.1757	2.2101	2.2430
14.696 (212.00)	Sh				38.00	88.00	138.00	188.00	238.00	288.00	388.00	488.00	588.00	688.00	788.00	888.00	988.00
	v	.0167	26.799		28.42	30.52	32.60	34.67	36.72	38.77	42.86	46.93	51.00	55.06	59.13	63.19	67.25
	h	180.17	1150.5		1168.8	1192.6	1216.3	1239.9	1263.6	1287.4	1335.2	1383.8	1433.2	1483.4	1534.5	1586.5	1639.4
	s	.3121	1.7568		1.7833	1.8158	1.8459	1.8743	1.9010	1.9265	1.9739	2.0177	2.0585	2.0969	2.1332	2.1676	2.2005
15 (213.03)	Sh				36.97	86.97	136.97	186.97	236.97	286.97	386.97	486.97	586.97	686.97	786.97	886.97	986.97
	v	0.01673	26.290		27.837	29.899	31.939	33.963	35.977	37.985	41.986	45.978	49.964	53.946	57.926	61.905	65.882
	h	181.21	1150.9		1168.7	1192.5	1216.2	1239.9	1263.6	1287.3	1335.2	1383.8	1433.2	1483.4	1534.5	1586.5	1639.4
	s	0.3137	1.7552		1.7809	1.8134	1.8437	1.8720	1.8988	1.9242	1.9717	2.0155	2.0563	2.0946	2.1309	2.1653	2.1982
20 (227.96)	Sh				22.04	72.04	122.04	172.04	222.04	272.04	372.04	472.04	572.04	672.04	772.04	872.04	972.04
	v	0.01683	20.087		20.788	22.356	23.900	25.428	26.946	28.457	31.466	34.465	37.458	40.447	43.435	46.420	49.405
	h	196.27	1156.3		1167.1	1191.4	1215.4	1239.2	1263.0	1286.9	1334.9	1383.5	1432.9	1483.2	1534.3	1586.3	1639.3
	s	0.3358	1.7320		1.7475	1.7805	1.8111	1.8397	1.8666	1.8921	1.9397	1.9836	2.0244	2.0628	2.0991	2.1336	2.1665
25 (240.07)	Sh				9.93	59.93	109.93	159.93	209.93	259.93	359.93	459.93	559.93	659.93	759.93	859.93	959.93
	v	0.01693	16.301		16.558	17.829	19.076	20.307	21.527	22.740	25.153	27.557	29.954	32.348	34.740	37.130	39.518
	h	208.52	1160.6		1165.6	1190.2	1214.5	1238.5	1262.5	1286.4	1334.6	1383.3	1432.7	1483.0	1534.2	1586.2	1639.2
	s	0.3535	1.7141		1.7212	1.7547	1.7856	1.8145	1.8415	1.8672	1.9149	1.9588	1.9997	2.0381	2.0744	2.1089	2.1418
30 (250.34)	Sh					49.66	99.66	149.66	199.66	249.66	349.66	449.66	549.66	649.66	749.66	849.66	949.66
	v	0.01701	13.744			14.810	15.859	16.892	17.914	18.929	20.945	22.951	24.952	26.949	28.943	30.936	32.927
	h	218.93	1164.1			1189.0	1213.6	1237.8	1261.9	1286.0	1334.2	1383.0	1432.5	1482.8	1534.0	1586.1	1639.0
	s	0.3682	1.6995			1.7334	1.7647	1.7937	1.8210	1.8467	1.8946	1.9386	1.9795	2.0179	2.0543	2.0888	2.1217
35 (259.29)	Sh					40.71	90.71	140.71	190.71	240.71	340.71	440.71	540.71	640.71	740.71	840.71	940.71
	v	0.01708	11.896			12.654	13.562	14.453	15.334	16.207	17.939	19.662	21.379	23.092	24.803	26.512	28.220
	h	228.03	1167.1			1187.8	1212.7	1237.1	1261.3	1285.5	1333.9	1382.8	1432.3	1482.7	1533.9	1586.0	1638.9
	s	0.3809	1.6872			1.7152	1.7468	1.7761	1.8035	1.8294	1.8774	1.9214	1.9624	2.0009	2.0372	2.0717	2.1046
40 (267.25)	Sh					32.75	82.75	132.75	182.75	232.75	332.75	432.75	532.75	632.75	732.75	832.75	932.75
	v	0.01715	10.497			11.036	11.838	12.624	13.398	14.165	15.685	17.195	18.699	20.199	21.697	23.194	24.689
	h	236.14	1169.8			1186.6	1211.7	1236.4	1260.8	1285.0	1333.6	1382.5	1432.1	1482.5	1533.7	1585.8	1638.8
	s	0.3921	1.6765			1.6992	1.7312	1.7608	1.7883	1.8143	1.8624	1.9065	1.9476	1.9860	2.0224	2.0569	2.0899
45 (274.44)	Sh					25.56	75.56	125.56	175.56	225.56	325.56	425.56	525.56	625.56	725.56	825.56	925.56
	v	0.01721	9.399			9.777	10.497	11.201	11.892	12.577	13.932	15.276	16.614	17.950	19.282	20.613	21.943
	h	243.49	1172.1			1185.4	1210.4	1235.7	1260.2	1284.6	1333.3	1382.3	1431.9	1482.3	1533.6	1585.7	1638.7
	s	0.4021	1.6671			1.6849	1.7173	1.7471	1.7748	1.8010	1.8492	1.8934	1.9345	1.9730	2.0093	2.0439	2.0768
50 (281.02)	Sh					18.98	68.98	118.98	168.98	218.98	318.98	418.98	518.98	618.98	718.98	818.98	918.98
	v	0.01727	8.514			8.769	9.424	10.062	10.688	11.306	12.529	13.741	14.947	16.150	17.350	18.549	19.746
	h	250.21	1174.1			1184.1	1209.9	1234.9	1259.6	1284.1	1332.9	1382.0	1431.7	1482.2	1533.4	1585.6	1638.6
	s	0.4112	1.6586			1.6720	1.7048	1.7349	1.7628	1.7890	1.8374	1.8816	1.9227	1.9613	1.9977	2.0322	2.0652
55 (287.07)	Sh					12.93	62.93	112.93	162.93	212.93	312.93	412.93	512.93	612.93	712.93	812.93	912.93
	v	0.01733	7.787			7.945	8.546	9.130	9.702	10.267	11.381	12.485	13.583	14.677	15.769	16.859	17.948
	h	256.43	1176.0			1182.9	1208.9	1234.2	1259.1	1283.6	1332.6	1381.8	1431.5	1482.0	1533.3	1585.5	1638.5
	s	0.4196	1.6510			1.6601	1.6933	1.7237	1.7518	1.7781	1.8266	1.8710	1.9121	1.9507	1.987	2.022	2.055
60 (292.71)	Sh					7.29	57.29	107.29	157.29	207.29	307.29	407.29	507.29	607.29	707.29	807.29	907.29
	v	0.01738	7.174			7.257	7.815	8.354	8.881	9.400	10.425	11.438	12.446	13.450	14.452	15.452	16.450
	h	262.21	1177.6			1181.6	1208.0	1233.5	1258.5	1283.2	1332.3	1381.5	1431.3	1481.8	1533.2	1585.3	1638.4
	s	0.4273	1.6440			1.6492	1.6829	1.7134	1.7417	1.7681	1.8168	1.8612	1.9024	1.9410	1.9774	2.0120	2.0450
65 (297.98)	Sh					2.02	52.02	102.02	152.02	202.02	302.02	402.02	502.02	602.02	702.02	802.02	902.02
	v	0.01743	6.653			6.675	7.195	7.697	8.186	8.667	9.615	10.552	11.484	12.412	13.337	14.261	15.183
	h	267.63	1179.1			1180.3	1207.0	1232.7	1257.9	1282.7	1331.9	1381.3	1431.1	1481.6	1533.0	1585.2	1638.3
	s	0.4344	1.6375			1.6390	1.6731	1.7040	1.7324	1.7590	1.8077	1.8522	1.8935	1.9321	1.9685	2.0031	2.0361
70 (302.93)	Sh						47.07	97.07	147.07	197.07	297.07	397.07	497.07	597.07	697.07	797.07	897.07
	v	0.01748	6.205				6.664	7.133	7.590	8.039	8.922	9.793	10.659	11.522	12.382	13.240	14.097
	h	272.74	1180.6				1206.0	1232.0	1257.3	1282.2	1331.6	1381.0	1430.9	1481.5	1532.9	1585.1	1638.2
	s	0.4411	1.6316				1.6640	1.6951	1.7237	1.7504	1.7993	1.8439	1.8852	1.9238	1.9603	1.9949	2.0279
75 (307.61)	Sh						42.39	92.39	142.39	192.39	292.39	392.39	492.39	592.39	692.39	792.39	892.39
	v	0.01753	5.814				6.204	6.645	7.074	7.494	8.320	9.135	9.945	10.750	11.553	12.355	13.155
	h	277.56	1181.9				1205.0	1231.2	1256.7	1281.7	1331.3	1380.7	1430.7	1481.3	1532.7	1585.0	1638.1
	s	0.4474	1.6260				1.6554	1.6868	1.7156	1.7424	1.7915	1.8361	1.8774	1.9161	1.9526	1.9872	2.0202

Sh = superheat, F
v = specific volume, cu ft per lb
h = enthalpy, Btu per lb
s = entropy, Btu per R per lb

$\rho = \frac{1}{v}$ = Density, lb per cu ft

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

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

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.6 ECCS Recirculation Fluid pH Control System

BASES

BACKGROUND

The Recirculation Fluid pH Control System is a passive system designed to raise the long term pH of the solution in the containment sump following a Design Basis Accident (DBA). The Recirculation Fluid pH Control System consists of three storage baskets containing trisodium phosphate (TSP) as $Na_3PO_4 \cdot 12H_2O \cdot \frac{1}{4}NaOH$. An equivalent amount of trisodium phosphate compound with a different chemical formula may be used. When equivalent compounds are used, the allowable weights/volumes may be different; however, the equivalent amount of trisodium phosphate compound must raise the pH of the recirculating solution into the range of 7.5 to 10.5. In the event of a loss of coolant accident (LOCA), the TSP contained in the storage baskets will be dissolved in the Reactor Coolant System (RCS) and Refueling Water Storage Tank (RWST) inventories lost through the pipe break. The resulting increase in the recirculation solution pH into the range of 7.5 to 10.5 assures that iodine is retained in solution and that chloride induced stress corrosion on mechanical systems and components is minimized (Ref. 1). The Recirculation Fluid pH Control System performs no function during normal plant operation.

A&C 1st Part Correct

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. Fuel damage following a DBA will cause iodine to be released into the reactor coolant and containment atmosphere. Iodine released to the containment atmosphere is absorbed by the containment spray and washed into the containment sump. Since the ECCS water is borated for reactivity control, the recirculation solution in the containment sump will initially be acidic with a pH of approximately 4.5. In a low pH (acidic) solution, some of the dissolved iodine will be converted to a volatile form and evolve out of solution into the containment atmosphere. In order to reduce the potential for elemental iodine evolution, the ECCS recirculation solution is adjusted (buffered) to achieve a long term alkaline pH of no less than 7.5. An alkaline pH promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. In addition to ensuring iodine is retained in solution, an alkaline recirculation solution will minimize chloride induced stress corrosion cracking of austenitic stainless steel

B&D first Part incorrect RWST water pH=4.5

A&D 2nd Part incorrect

B&C 2nd Part Correct

(continued)

28. 029EA1.06 001/MOD/RO/C/A 3.2/3.1/N/N/2/CVR/Y

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 **MINIMUM** required action 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 **ONLY**.

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 **ONLY**.

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)

Step	Action/Expected Response	Response NOT Obtained

___ 3 **Verify AFW pumps - RUNNING.**

3.1 MDAFWPs - RUNNING

[] 1A amps > 0
[] 1B amps > 0

3.2 TDAFWP - RUNNING IF NECESSARY

- TDAFWP STM SUPP FROM 1B(1C) SG

[] MLB-4 1-3 lit
[] MLB-4 2-3 lit
[] MLB-4 3-3 lit

- TDAFWP SPEED

[] SI 3411A > 3900 rpm

- TDAFWP SPEED CONT

[] SIC 3405 at 100%

NOTE:
- 2500 gallons of emergency boration is required for each control rod not fully inserted, up to a maximum of 17,309 gallons.
- [CA] Emergency boration should continue until an adequate shutdown margin is established.

___ 4 **Initiate Emergency Boration of the RCS.**

4.1 Verify at least one CHG PUMP - RUNNING.

Step 4 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
4.2	Start a boric acid transfer pump. BATP [] 1A → T/O [] 1B → Tripped → N →	4.2 Perform the following. 4.2.1 Align charging pump suction to RWST. RWST TO CHG PUMP [] Q1E21LCV115B open [] Q1E21LCV115D open VCT OUTLET ISO [] Q1E21LCV115C closed [] Q1E21LCV115E closed B & D 1st part correct → 4.2.2 Proceed to step 4.4.
4.3	Align normal emergency boration. ← SKIP to 4.4 EMERG BORATE TO CHG PUMP SUCT [] Q1E21MOV8104 open	4.3 Perform the following. • Align charging pump suction to RWST. RWST TO CHG PUMP [] Q1E21LCV115B open [] Q1E21LCV115D open VCT OUTLET ISO [] Q1E21LCV115C closed [] Q1E21LCV115E closed OR Adc 1st Part incorrect → • Align manual emergency boration flow path. BORIC ACID TO BLENDER [] Q1E21FCV113A open MAN EMERG BORATION [] Q1E21V185 open (100 ft, AUX BLDG rad-side chemical mixing tank area)

Step 4 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
4.4	Establish adequate letdown.	
4.4.1	Verify 45 GPM letdown orifice - IN SERVICE. LTDN ORIF ISO 45 GPM [] Q1E21HV8149A open	
4.4.2	Verify one 60 GPM letdown orifice - IN SERVICE. LTDN ORIF ISO 60 GPM [] Q1E21HV8149B open [] Q1E21HV8149C open	
4.5	Check pressurizer pressure LESS THAN 2335 psig.	4.5 Verify PRZR PORVs and PRZR PORV ISOs - OPEN. IF NOT, THEN open PRZR PORVs and PORV ISOs as necessary until pressurizer pressure less than 2135 psig.
4.6	Establish adequate charging flow. • IF boration is from boric acid storage tank, THEN verify charging flow - GREATER THAN 40 GPM. N/A ← A + D 2nd parts incorrect OR • IF boration is from the RWST, THEN verify charging flow - GREATER THAN 92 GPM. ← B + C 2nd parts correct	

Step 4 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

4.7 Verify emergency boration flow adequate.

4.7.1 IF normal emergency boration flow path aligned, THEN check emergency boration flow greater than 30 GPM.

BORIC ACID
EMERG BORATE
[] FI 110

A & D 2nd parts incorrect

incorrect

4.7.2 IF manual emergency boration flow path aligned, THEN check boric acid flow greater than 30 GPM.

MAKEUP FLOW
TO CHG/VCT
[] BA
FI 113

4.7.3 IF boration is from the RWST, THEN verify charging flow - GREATER THAN 92 GPM.

B & C
2nd parts
correct

___Page Completed

1. 029EA1.05 001/1/1/ATWT - BIT OUTLET SW/C/A - 3.7/MODIFIED/R/NRC RO/TNT / RLM Bar/2

029EA1.06

The following plant conditions exist:

- Unit 2 has experienced an Anticipated Transient Without Trip (ATWT) and has implemented FRP-19211, Response to Nuclear Power Generation ATWT.
- A Charging Pump is running.
- Boric Acid Transfer Pump # 1 is tagged out.
- Boric Acid Transfer Pump # 2 trips on start.
- SI has **NOT** actuated at this time.
- The SS has directed the RO to establish Emergency Boration in accordance with SOP-13009, "CVCS Reactor Makeup Control System".

Which **ONE** of the following actions would establish a **CORRECT** emergency boration flow path in accordance with the SOP? (Assume 12 gpm seal return flow)

A. 1) Open HV-8104 EMERGENCY BORATE Valve.

   2) Adjust charging flow controller FIC-0121 to obtain > 42 gpm flow through the Normal Charging flow path.

B. 1) Open LV-0112D and LV-0112E RWST TO CHARGING PUMP SUCT valves.

   2) Adjust charging flow controller FIC-0121 to obtain > 42 gpm flow through the Normal Charging flow path.

C. 1) Open FV-110A BA to Blender and FV-110B BLENDER OUTLET TO CHARGING PUMPS SUCT.

   2) Adjust charging flow controller FIC-0121 to obtain > 100 gpm flow through the Normal Charging Path.

D✓ 1) Open LV-0112D and LV-0112E RWST TO CHARGING PUMP SUCT valves and HV-8801A and HV-8801B BIT DISCHARGE ISOLATION valves.

   2) Verify BIT flow (FI-0917A), plus total seal injection flow, minus total seal return flow is > 100 gpm.

K/A

029 Anticipated Transient Without Scram (ATWS):

EA1.05 Ability to operate and monitor the following as they apply to an ATWS.

BIT outlet valve switches.

K/A MATCH ANALYSIS

Question gives a plausible scenario with an ATWT in progress. Neither Boric Acid Transfer Pump is available. Candidate must choose a correct emergency boration flow path that would achieve Emergency Boration Flow.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Without BA Transfer Pumps available there would be no flow through HV-8104. Plausible the candidate may not realize BA Transfer Pump impact on flow path. Flow rates given would satisfy the flow path if BA Transfer Pumps available.

B. Incorrect. LV-112D and LV-112E would satisfy the flow path requirements but the minimum flow requirement via this path would be 100 gpm. Plausible candidate could recognize a correct flow path but confuse the flow rate requirements.

C. Incorrect. FV-0110A and FV-0110B would not have flow through this path without the Boric Acid Transfer Pumps available. Plausible candidate may not realize BA Transfer Pump impact on the flow path and confuse the flow rate requirements.

D. Correct. Opening LV-112D and LV-112E would establish boration flow from RWST and flow requirements would be satisfied with 100 gpm to BIT. 100 gpm used to sound more like choice B to make question symetrical with choices and NOT be a NOT question.

REFERENCES

19211-C, Nuclear Power Generation ATWT page 4

13009-1/2, CVCS Makeup Control System section 4.9 for Emergency Boration pages 38 through 41.

LO-PP-09300-06-001, 003, and 004 from Vogtle LO Active Exam Bank

VEGP learning objectives:

LO-PP-09300-06, Describe all emergency flow path

a. borated water source and discharge flow path

b. minimum flow requirements

29. 032AK3.02 001/NEW/RO/C/A 3.7/4.1/N/N/3/CVR/Y

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 ~~to prevent damage to the detectors~~

B. (1) automatically energize prematurely

(2) de-energized until approximately 15 minutes post-trip ~~to prevent damage to the detectors~~

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

(2) energized approximately 5 minutes post-trip ~~to prevent a loss of reactor power indication~~

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

(2) energized approximately 15 minutes post-trip ~~to prevent a loss of reactor power indication~~

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.

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.

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)

5.21 WHEN the Source Range Permissive P-6 light goes off (2/2 intermediate ranges less than 1 x 10^{-10} amps), THEN check the following:

_____ 5.21.1 Check SOURCE RANGE TRAIN A TRIP BLOCED/HIGH VOLT OFF status light is OFF.

_____ 5.21.1 Check SOURCE RANGE TRAIN B TRIP BLOCED/HIGH VOLT OFF status light is OFF.

Ed D 1st parts correct.

_____ 5.21.3 Check SR LOSS OF DETECTOR VOLAGE annunciator FA3 is clear.

5.21.4 IF a source range channel high voltage is NOT automatically energized due to an under compensated or malfunctioning intermediate range channel, THEN manually RESET the affected source range high voltage when the operable intermediate range channel indicates less than 5 x 10-11 amps.

under compensation IR causes ind. to be higher than actual.

_____ 5.21.4.1 IF required, THEN take SOURCE RANGE BLOCK-RESET A TRAIN to RESET.

_____ 5.21.4.2 IF required, THEN take SOURCE RANGE BLOCK-RESET B TRAIN to RESET.

Ad B 1st parts incorrect.

5.22 WHEN the Source Range Nuclear Instruments are energized, THEN perform the following:

_____ 5.22.1 Verify the scaler timer aligned and operating properly.

_____ 5.22.2 Verify audio count rate amplifier aligned and operating properly.

5.23 Within one hour after P-6 is reached perform the following:

_____ / _____ 5.23.1 Perform SR 3.3.1.1(channel check) for the Source Range Nuclear Instruments.

_____ 5.23.2 Document the channel check in FNP-1-STP-1.0, OPERATIONS DAILY AND SHIFT SURVEILLANCE REQUIREMENTS.

_____ / _____ 5.23.3 IF the channel check cannot be performed, THEN verify adequate shutdown margin using FNP-1-STP-29.1, SHUTDOWN MARGIN CALCULATION (TAVG 547°F), or FNP-1-STP-29.2, (TAVG < 547°F OR BEFORE THE INITIAL CRITICALITY FOLLOWING REFUELING)

NOTE: The rod position corresponding to ECC 0.5% ΔK/K is determined in FNP-1-STP-29.6, Appendix 1, CALCULATION OF ESTIMATED CRITICAL CONDITION.

____ / ____ 5.16 IF criticality has NOT been achieved with the rods withdrawn to 0.5% ΔK/K (500 pcm) past the estimated critical position, THEN perform the following:

____ / ____ 5.16.1 Insert all control banks to the bottom of the core.

____ 5.16.2 Direct Chemistry to sample the RCS for boron concentration.

____ 5.16.3 Re-calculate the ECC.

____ 5.16.4 Determine and correct any discrepancy in the ECC.

____ 5.16.5 IF no error can be found in the ECC or the RCS boron concentration, THEN contact the Reactor Engineering for assistance with the ECC.

____ 5.16.6 IF required, THEN establish the correct critical boron concentration.

____ / ____ 5.16.7 WHEN all errors have been corrected, THEN using the Inverse Count Rate Ratio Plot procedure per Appendix 1, withdraw the control rods in MANUAL to establish reactor criticality.

____ / ____ 5.17 Establish a startup rate of approximately 3/4 decade per minute.

A&B 1st parts - incorrect on power decrease

5.18 WHEN the Source Range Permissive P-6 light is on (1/2 intermediate ranges greater than 10^{-10} amps), THEN perform the following:

____ / ____ 5.18.1 Block the Source Range High Flux Reactor Trip.

- [] Source Range BLOCK-RESET A TRN taken to BLOCK
- [] Source Range BLOCK-RESET B TRN taken to BLOCK

____ / ____ 5.18.2 On the Bypass and Permissive Panel, verify that the following windows are illuminated.

- [] SOURCE RANGE TRAIN A BLOCKED HI VOLTS OFF
- [] SOURCE RANGE TRAIN B BLOCKED HI VOLTS OFF

____ / ____ 5.18.3 Verify SR LOSS OF DET VOLTAGE annunciator FA3 is illuminated.

____ 5.18.4 Verify the Source Range NI drawers indicate zero voltage.

____ 5.18.5 Ensure the Scaler-Timer is shutdown per FNP-1-SOP-39.0, NUCLEAR INSTRUMENTATION SYSTEM.

General Physics Corporation GFES
Lesson Plan

REACTOR SHUTDOWN AND RCS COOLDOWN

A normal plant shutdown and cooldown are performed periodically for refueling or maintenance. Power reduction is performed by decreasing the external load on the turbine generator in conjunction with a boration of the RCS. This maintains control rod position and satisfies axial flux difference requirements and rod insertion limits. As power is decreased below 15%, the rods are put in manual control and the reactor operator manipulates the rods as necessary to control RCS temperature. When the turbine generator load has been decreased to approximately 50 MW the turbine is tripped and the control rods are positioned to maintain approximately 2% reactor power.

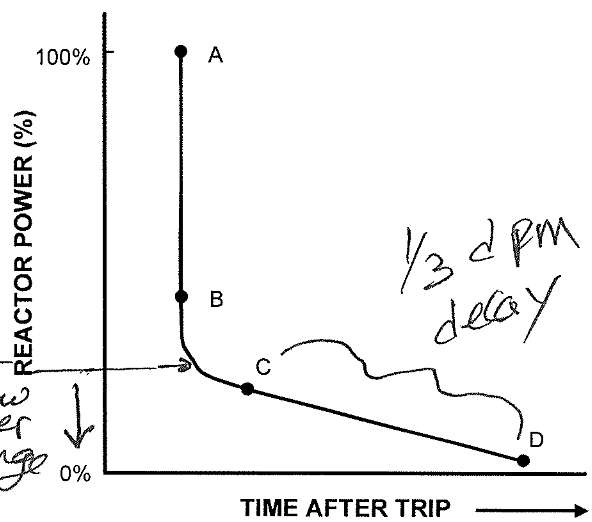
After power has been stabilized at approximately 2%, the reactor operator records the information required for Reference Reactivity Data (RRD): power level, rod position, and actual boron concentration. This data will be used for calculation of shutdown margins and for subsequent ECPs. Once the RRD data has been recorded, the reactor is shutdown by fully inserting all control banks.

Before starting the reactor cooldown, the RCS is borated to achieve the xenon-free shutdown margin required by technical specifications for RCS temperature below 200°F (typically -1,000 Δk/k). Once this boration is completed and the RCS boron concentration has been verified by chemical analysis, the cooldown is performed. When cold shutdown conditions are reached, the shutdown margin is re-verified. If adequate, the shutdown banks are fully inserted and the reactor trip breakers are opened.

RESPONSE TO A REACTOR TRIP

The actions taken by reactor operators following a reactor trip are dictated by approved station procedures. These procedures ensure that the reactor is shut down, the turbine is tripped, normal and/or emergency power sources are available, and the plant response is as expected. If needed, compensatory actions are taken in accordance with the procedures.

Figure 8-24 shows the behavior of a reactor power drop following a reactor trip.



Below power Range

Figure 8-24 Reactor Power Drop Following a Reactor Trip

The indicated fission rate decreases to below the power range immediately upon insertion of the control and shutdown rods. This rapid reactivity insertion is denoted by the neutron flux trace (power drop) from A to B in the figure. This is referred to as prompt drop following the reactor trip.

During the period from B to C, the neutron population is dominated by the appearance of delayed neutrons from shorter- and intermediate-lived delayed neutron precursors.

These precursors, which were formed when the reactor was at 100% power, decay within a few minutes. Once the shorter-lived precursors have effectively all decayed, neutron population is controlled by the appearance of delayed neutrons from the longest-lived precursors.

From C to D, power falls at a constant –80 second period based on the mean life of the longest lived delayed neutron precursor, bromine-87 (half-life of about 56 seconds). The –80 second period is equivalent to about a -1/3 decade per minute (DPM) startup rate (SUR). This continues until neutron population is low enough for the effect of source neutrons to be seen and a subcritical equilibrium is reached.

Core thermal power remains high for several seconds after the trip (as shown by points B to C). There is a time lag of a few seconds for the heat generated in the fuel to be conducted into the coolant, and the decay heat immediately following the prompt drop is approximately 7% of rated thermal power (RTP), assuming a trip from equilibrium full power operation. (This occurs at about point C.)

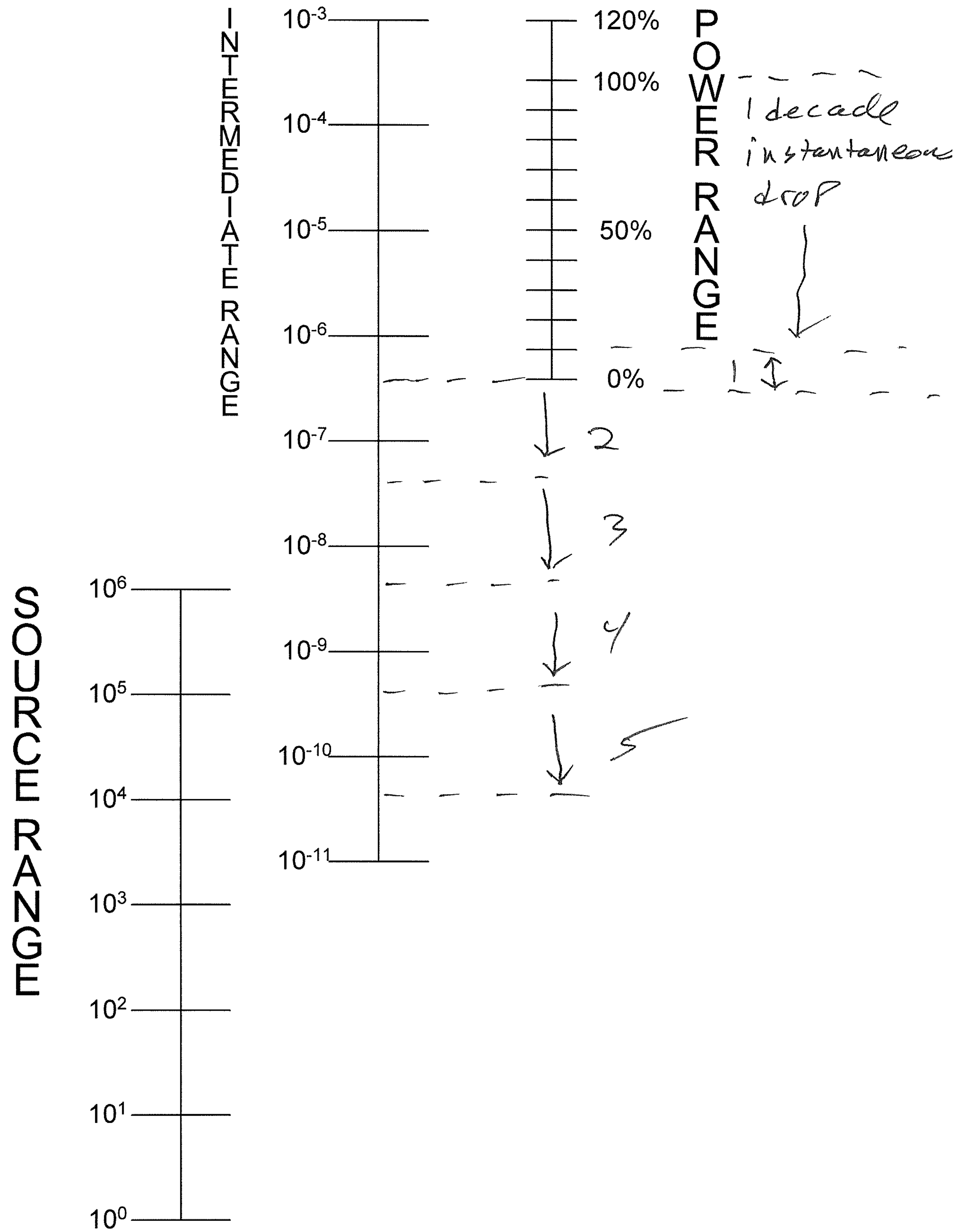
- RCS temperature is reduced by the steam dump system and stabilizes at no-load T_{ave}.
- Ten seconds after the trip, decay heat is still approximately 5% RTP, and it decreases to about 1% RTP in a little less than three hours (between points C to D).

A reactor that has been operating at steady-state 100% power trips, dropping rods worth 10 %Δk/k (10,000 pcm) into the core. This causes an immediate prompt drop in reactor power to approximately ______ %, followed by a slower decrease.

1/3 dpm decay for
≈ 5 decades
= ≈ 15 minutes

Example 8-25

Figure 1


INTERMEDIATE RANGE
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
10^{-8}
10^{-9}
10^{-10}
10^{-11}
120%
100%
50%
0%
POWER RANGE
1 decade instantaneous drop
1
2
3
4
5
SOURCE RANGE
10^{6}
10^{5}
10^{4}
10^{3}
10^{2}
10^{1}
10^{0}

5.11 WHEN the reactor is critical, THEN perform one of the following:

_____
__/__ 5.11.1 Verify the low low Tavg alarm reset (RX COOLANT LOOPS 1A, 1B or 1C TAVG LO-LO annunciator HF4) AND all RCS Loop Tavg greater than or equal to 547°F.

OR

_____
__/__ 5.11.2 Verify each reactor coolant loop Tavg greater than or equal to 541°F at least once ever 30 minutes per FNP-1-STP-35.l, UNIT STARTUP TECHNICAL SPECIFICATION VERIFICATION. (Technical Specification 3.4.2)

CAUTIONS:	• **During all rod withdrawals, monitor nuclear instrumentation. Criticality shall be anticipated any time the control rods are being withdrawn. Additionally, during any approach to criticality monitor all pertinent instrumentation to allow errors in the ECC or problems with other instrumentation to be detected early. Consider the use of audio count rate speakers as an aid to determine increasing flux rate. (SOER 88-02)**
	• **Do not exceed a sustained startup rate of one decade per minute.**

6 decade at 1dpm => 6 minute

_____ 5.12 Using the Inverse Count Rate Ratio procedure per Appendix 1 withdraw the control rod banks in MANUAL to establish reactor criticality. **{CMT-0008411}**

_____
__/__ 5.13 Verify proper overlap of I.R. (See Fig 1) Note S.R. count when I.R. starts to come on scale. (CR 1-2000-148)

_____ 5.13.1 Record Source Range Counts. ________ (N31)
________ (N32)

A&C 2nd parts in correct

_____ 5.13.2 Record SR counts in Reactor Operators Log.

_____ 5.13.3 Record SR counts in Surveillance Test Data Book.

2.8.1 Reactor Permissives

1. Power Escalation

The overpower protection provided by the excore nuclear instrumentation shall consist of three discrete, but overlapping levels. Continuation of startup operation of power increase shall require a permissive signal from the higher range instrumentation channels before the lower range level trips can be manually blocked by the operator.

A one out of two intermediate range permissive signal P-6 (set at 1×10^{-10} amp) is required prior to source range level trip blocking and source range detector high voltage cutoff. Source range level trips are automatically reactivated and high voltage restored when both intermediate range channels are below the permissive (P-6) level. There shall be a manual reset switch for administratively reactivating the source range level trip and detector high voltage when between the permissive P-6 and P-10 (set at 10% rated thermal power) level, if required.

2nd ~~1st~~ part of D correct ("energized" ~~1st~~ 2nd part correct)

Source range level trip block and high voltage cutoff shall always be active when above the permissive P-10 level.

The intermediate range reactor trip and power-range (low setpoint) reactor trip shall only be blocked after satisfactory operation and permissive information are obtained from two of four power range channels which indicates P-10. Individual blocking switches shall be provided so that the low setpoint power range trip and intermediate range trip can be independently blocked. Moreover P-10 allows the operator to manually block the intermediate range C-1 rod stop. These trips are automatically reactivated when any three of the four power range channels are below the permissive (P-10) level, thus ensuring automatic activation to more restrictive trip protection. See Table T-3 for a comprehensive list of Reactor Protection System permissives. (References 6.4.007, 6.4.011, 6.7.012)

2. Blocks of Reactor Trips at Low Power

Permissive P-7 shall prevent unnecessary at power reactor trips during low power by auto blocking the following reactor trips:

- Low reactor coolant flow in any two loops
- RCP breaker trip
- Undervoltage condition on RCP electrical buses
- Underfrequency condition on RCP electrical buses

30. 035K3.02 005/NEW/RO/C/A 4.0/4.3/N/N/4/HOWARDS/VER 5 EDITORIAL

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 TCOLD 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 [...].

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: With all RCPs secured RCS cooldown may cause a false FNP-1-CSF-0.4 Integrity Status Tree indication for the ruptured loop. Disregard ruptured loop cold leg temperature until completion of step 30.

****

NOTE:
- The steam dumps will be interlocked closed when RCS TAVG reaches P-12 (543°F). This interlock may be bypassed for A and E steam dumps with the STM DUMP INTERLOCK switches.
- Excessive opening of steam dumps can cause a high steam flow LO-LO TAVG main steam line isolation signal.

6.4 IF condenser available, THEN dump steam to condenser from intact SGs at maximum attainable rate.

BYP & PERMISSIVE
COND
AVAIL
[] C-9 light lit

STM DUMP
[] MODE SEL A-B TRN in STM PRESS

STM DUMP
INTERLOCK
[] A TRN in ON
[] B TRN in ON

STM HDR
PRESS
[] PK 464 adjusted

6.4 Dump steam to atmosphere.

6.4.1 Direct counting room to perform FNP-0-CCP-645, MAIN STEAM ABNORMAL ENVIRONMENTAL RELEASE.

6.4.2 IF normal air available, THEN control atmospheric relief valves to dump steam from intact SGs at maximum attainable rate. IF NOT, dump steam using FNP-1-SOP-62.0, EMERGENCY AIR SYSTEM.

1A(1B,1C) MS ATMOS
REL VLV
[] PC 3371A adjusted
[] PC 3371B adjusted
[] PC 3371C adjusted

Step 6 continued on next page.

___Page Completed

Section: Procedure

Unit 1 ERP Step: 6.4 CAUTION-1 **Unit 2 ERP Step:** 6.4 CAUTION-1 ***ERG Step No:*** *6 CAUTION-1*

ERP StepText: With all RCPs secured RCS cooldown may cause a false FNP-2-CSF-0.4 Integrity Status Tree indication for the ruptured loop. Disregard ruptured loop cold leg temperature until completion of step 30.

ERG StepText: *If RCPs are not running, the following steps may cause a false F-0.4, Integrity Status Tree indication for the ruptured loop. Disregard the ruptured loop T-cold indication until after performing Step 29.*

Purpose: To alert the operator that during a natural circulation cooldown a false symptom of a red or orange path condition in F-0.4, Integrity Status Tree is possible due to the redirection of SI flow in the ruptured loop.

Basis: 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 of 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 the other loops are circulating 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. After the cooldown and depressurization is completed and SI is terminated, the operator should monitor the F-0.4, Integrity Status Tree to determine if a red or orange path still exists and FR-P.1 should be implemented. His decision should be based on the symptoms existing after SI is terminated. If a multiple or subsequent accident occurs, the operator could transfer out of E-3 prior to terminating SI. For that case he should monitor the F-0.4, Integrity Status Tree when he makes the transition out of E-3 to determine if at that time a red or orange path exists and FR-P.1 should be implemented. STEP DESCRIPTION TABLE FOR E-3Step 6 - CAUTION

Knowledge: If a multiple or subsequent accident occurs, the operator could transfer out of E-3 prior to terminating SI. For that case he should monitor the F-0.4, Integrity Status Tree when he makes the transition out of E-3 to determine if at that time a red or orange path exists and FR-P.1 should be implemented.

References: DW-96-028

31. 037AA2.08 001/NEW/RO/C/A 2.8/3.3/N/N/2/CVR/Y

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

- R-15A, SJAE EXH, has failed.
- 1A SG has developed a 10 gpm tube leak.
- One of the 1A SG safeties is leaking by.

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

A.✓ R-19, SGBD SAMPLE, alarm.

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

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

D. R-60A, 1A STEAM GENERATOR, alarm.

A - 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.

B - 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).

C - 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.

D - Incorrect. R-60 is a high range monitor that does not upscale in the event of SGTL with no fuel failure, even with an open safety or SG Atmospheric relief. Plausible, since if it was a lower scale, it would alarm prior to R-19 since it is monitoring the steam coming from the 1A SG with the tube leak. Also, it would alarm if the dose from the SG was high enough (such as due to a SGTR and fuel element failure). The SGTL combined with the safety leakby allows it to monitor the actual 1A SG contaminated steam as it escapes through the safety.

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.

FNP-1-ARP-1.6, FH1
AUTOMATIC ACTIONS (cont)
2. ARDA will automatically start for the following conditions:
2.1 ARDA will automatically start when any of the following monitors go into alarm for two consecutive system polls one minute apart on either unit and use the latest 15 minute average monitor value to perform the calculations:
Plant Vent Stack Monitors R29 (SPING)
Noble Gas 4.44e-4 c/ml
Iodine 1.20e-6 c/ml
Particulate 4.00e-5 c/ml
Steam Jet air Ejector R15C 27 mr/hr
TDAFW Exhaust R60D 38 mr/hr
Steam Generator A R60A 38 mr/hr [listed here to show the nomenclature in the procedure]
Steam Generator B R60B 38 mr/hr
Steam Generator C R60C 38 mr/hr

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 how the failed R-15 applies to the SGTL is required. I. E., since 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

FARLEY NUCLEAR PLANT
UNIT 1
SYSTEM OPERATING PROCEDURE SOP-69.0

N-16 PRIMARY TO SECONDARY LEAK DETECTION SYSTEM

1.0 Purpose

To provide guidance for operation of the Primary to Secondary Leak Detection System.

2.0 Initial Conditions

2.1 120V Regulated Instrumentation Panel 1B is energized per FNP-1-SOP-36.4, 120V A.C. DISTRIBUTION SYSTEMS.

3.0 Precautions and Limitations

C incorrect
<20%
power

3.1 The system receives a reactor power input from power range channel N-43. IF N-43 fails OR is in Test OR is less than 20% power, THEN the system cannot accurately estimate a leak rate in the AV mode, and the indicators will display "PN <20%". If desired, the Counting Room can configure the N-16 system in the ME counts per second (C/S) mode using FNP-0-CCP-31, LEAK RATE DETERMINATION. While not able to provide a leak rate determination, this mode can be used to indicate if leakage is increasing based on the indication trending up. The AV mode is the preferred mode of operation above 20% reactor power. The ME mode should only be utilized below 20% reactor power.

3.2 The N-16 Leak Detection System cannot determine the location of a leak within a specific Steam Generator. The system can however provide a more accurate leak rate determination if the location of the leak is known to be in one of the following locations:

Cold Leg - CB, Hot Leg - HB or U-Bend region - BE

WHEN a leak location is selected (CB, HB or BE), THEN the processor displays a leak rate that assumes the leak is at the location you have selected. The AV mode is essentially the average of the three leak rates at the specific locations.

3.3 The N-16 system is limited to an upward range of 1,000 gallons per day.

increasing radiation to initiate the RMS High Radiation annunciator on the main control board. In addition, each function shall actuate an indicating light on the ratemeter front panel defining the alarm condition (References 6.4.034, 6.4.214, and 6.4.249). The ratemeter operation selector switch shall actuate the RMS CH Test annunciator on the main control board when placed in any position other than OPERATE (References 6.4.034 and 6.4.249).

3.2.5.3.4 Normally, no contaminated leakage is expected into the Service Water system. Accordingly, the monitor setpoint should be set approximately one half decade above the detector's normal response (Reference 6.7.062 and 6.7.080).

3.2.5.4 Interface Requirements

The instrument power supply for the RMS system panel N1H11NGRM 2502A, B, and C is 120 VAC distribution panel 1B, breaker number 2 (Reference 6.4.219). The control power supply for the RMS system panel N1H11NGRM 2502A, B, and C is 208/120 VAC control power panel 1N, breaker number 6 (Reference 6.4.107). The instrument power supply for the RMS system panel N2H11NGRM 2502A, B, and C is 120 VAC distribution panel 2B, breaker number 2 (Reference 6.4.345). The control power supply for the RMS system panel N2H11NGRM 2502A, B, and C is 208/120 VAC control power panel 2N, breaker number 6 (Reference 6.4.106).

3.2.5.5 Shielding Design

In addition to the shielding provided by the monitor housing, additional shielding was added surrounding the high voltage electronics housing above the detector. The purpose of the added shielding is to reduce the influence of background radiation, that caused spiking of the monitors and isolation of the process system and to improve monitor sensitivity (References 6.7.033, 6.7.035, and 6.7.080).

3.2.6 Steam Generator Blowdown

Service	TPNS Nos.
SG Blowdown to Processing System	ND11RE 0023A
SG Blowdown Discharge	ND11RE 0023B

3.2.6.1 Basic Function

3.2.6.1.1 Radiation detector RE 0023A, located in the steam generator blowdown discharge line upstream of the steam generator blowdown surge tank, monitors for an increase in radioactivity in the secondary system. To minimize contamination of the processing system and potential inadvertent release of radioactive gases through the surge tank vent, steam generator blowdown is automatically terminated by closing valve NB21FCV 1152 when the activity exceeds the setpoint. An increase in radioactivity in the secondary system would be indicative of a steam generator tube rupture accident (References 6.7.084 and 6.4.366).

3.2.6.1.2 Radiation monitor RE 0023B is located downstream of the steam generator blowdown discharge pumps prior to discharging to the dilution discharge on the service water system. This detector monitors the discharge stream to comply with GDCs 60 and 64. On an increase in radioactivity, the discharge is isolated by automatically closing NB21RCV023B (References 6.7.084 and 6.4.366).

3.2.6.2 Functional Requirements

3.2.6.2.1 An in-line liquid monitor shall be provided to directly monitor the process medium. The use of this type of monitor provides the fastest response time and easiest decontamination (References 6.4.366 and 6.7.080).

3.2.6.2.2 RE 0023B shall alarm and isolate the effluent discharge prior to exceeding the limits of ten times the concentrations stated in 10 CFR Part 20, Appendix B, Table 2, Column 2 for radionuclides other than dissolved or entrained noble gases. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in unrestricted areas will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 50, for a member of the public, and (2) the limits of 10 CFR 20.1301 for the population. The limiting concentration of dissolved and entrained noble gases is 1×10^{-4} µCi/ml (Reference 6.7.078, 6.7.081).

Setpoints are based on ensuring the discharge limits presented in Section 2.1.2 of the ODCM are not exceeded.

3.3.16.5 Interface Requirements

3.3.16.5.1 The 120 VAC power supply for RMS panel QSD11RE 0035A is 208/120 VAC control power panel 1R, breaker number 15 (Reference 6.4.108). The 208 VAC pump power supply for RMS panel QSD11RE 0035A is 208/120 VAC control power panel 1R, breaker number 12 (Reference 6.4.108). The 120 VAC power supply for RMS panel QSD11RE 0035B is 208/120 VAC control power panel 1S, breaker number 15 (Reference 6.4.108). The 208 VAC pump power supply for RMS panel QSD11RE 0035B is 208/120 VAC control power panel 1S, breaker number 12 (Reference 6.4.108).

3.3.16.5.2 The instrument air system shall provide a dry, filtered air source for monitor purge air (Reference 6.7.080).

3.3.17 Main Steam Safety and Atmospheric Relief and TDAFW Pump Exhaust Noble Gas Monitors

Service	TPNS Nos.
Steam Generator A Safety/Atmospheric Relief Valve Exhaust	ND11RE 0060A
Steam Generator B Safety/Atmospheric Relief Valve Exhaust	ND11RE 0060B
Steam Generator C Safety/Atmospheric Relief Valve Exhaust	ND11RE 0060C
Turbine Driven AFW Pump Exhaust	ND11RE 0060D

3.3.17.1 Basic Function

3.3.17.1.1 The monitors provide post-accident effluent monitoring for the Steam Generator Safety Valve and Turbine Driven Auxiliary Feedwater Pump turbine exhaust points in compliance with RG 1.97 (References 6.4.051 and 6.4.350).

3.3.17.1.2 These monitors may be used for RG 1.21 effluent activity tracking (References 6.7.078 and 6.7.060).

3.3.17.2 Functional Requirements

3.3.17.2.1 The monitors shall provide continuous indication over a range of 10^{-1} to 10^{3} microcuries per cubic centimeter (μCi/cc). The monitors shall provide an approximately linear response for gamma energies between 0.5 and 3 MeV (References 6.4.051 and 6.4.350, and 6.7.003). NUREG 0737, Clarification Item II.F.1, Attachment 1, requires that noble gas radiation monitors be provided for effluent points which monitor from normal operating levels to a maximum of 10^5 μCi/cc (Xenon-133 calibration) for undiluted containment effluents and 10^{-1} to 10^3 μCi/cc for buildings with systems containing primary coolant such as the auxiliary building. The present plant configuration provides area monitors for these parameters used in a process monitoring application. The meters for these monitors read in R/hr and conversion charts have been provided to convert the meter readings to μCi/cc. The present monitors indicate a range of 10^{-2} to 10^6 mR/hr, which corresponds to 10^{-5} to 1.4. x 10^4 μCi/cc (References 6.3.001 and 6.7.005).

3.3.17.2.2 Each radiation monitor is located to view its respective steam generator safety valve plumes and atmospheric relief valve plume. Each monitor is located on the auxiliary building roof and oriented to minimize the effects of radiation shine from the containment following a design basis LOCA. The monitors are located as far from the safety valve and main steam atmospheric vent valve vent stacks as permitted by the containment wall while keeping all monitors the same distance from the centerline of their respective main steam atmospheric vent valve. The monitor for the "B" steam generator limits this distance to 17.17 feet. Therefore, the monitors are located on a 17.17-foot circle around the main steam atmospheric vent valve stack and in a configuration not facing the containment, but have the steam generator safety valve plumes and the main steam atmospheric vent valve plume for that particular steam generator in full view.

The viewing angle in the horizontal plane was determined based on the location of the monitors and was determined to be 55°. A viewing angle of 55° allows monitoring of all plumes of a particular steam generator while excluding the field of view from most of the other steam generator plumes, and not facing containment.

For the vertical plane, the monitor viewing angle must be small enough to prevent the monitor from being influenced from radiation shine from the containment and large

32. 038EA2.07 001/NEW/RO/C/A 4.4/4.8/N/N/3/CVR/Y

Unit 1 has manually Tripped and Safety Injected from 14% power IAW AOP-2, Steam Generator Tube Leakage. The following conditions exist:

- DA-07, 1A 4160V BUS SUPP FROM 1A S/U XFMR, breaker tripped open.
- AFW Flows were maintained matched to all 3 SGs until securing AFW Flow.
- AFW Flow has been secured to all SGs.

- SG Pressures:
  - 1A 980 psig and stable
  - 1B 980 psig and stable
  - 1C 980 psig and stable
- SG NR levels:
  - 1A 61% and stable
  - 1B 61% and rising
  - 1C 50% and lowering

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

A.✓ SGTR on 1B SG **ONLY**.

B. SGTR on 1A **AND** 1B SGs **ONLY**.

C. SGTR on 1B SG **AND** a Steam Leak on 1C SG **ONLY**.

D. SGTR on 1A **AND** 1B SG **AND** a Steam Leak on 1C SG.

A - Correct. The level rising in 1B SG with no AFW flow, Pressurizer level dropping with max chg and no letdown and all SG pressures stable indicates this answer to be correct. C SG lvl is decreasing due to being the only SG steaming with no AFW flow. This is normal indication for this condition, and will require AFW flow to maintain C SG lvl.

B - Incorrect. The SGTR on 1B SG is correct, but the SGTR on 1A is incorrect. Plausible, since with the AFW flows matched to all SGs, and 1A SG level higher than the other intact SG by 10% NR Lvl, a SG tube leak would be indicated if not for the tripped RCP in that loop, and the lvl being stable instead of rising with no AFW flow AND no steaming. If the RCP was not tripped, this would be a correct answer, since with no AFW flow and decay heat removal level staying constant would indicate a SGTR.

C - Incorrect. The SGTR on 1B SG is correct, but the Steam Leak on 1C SG is incorrect due to the pressure being stable at 980 psig on all three SGs. Plausible, since 1C Level is dropping with the 1A SG level stable and the 1C SG dropping and no AFW flow to any of the SGs.

D - Incorrect. SGTR on 1A & 1B incorrect (see B). Steam leak on 1C incorrect (see C).

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/Y

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

5. High-1 containment pressure
(PB-951B, PB-952B, PB-953B) — 4.0 psig

6. Time delay on SI manual reset — 1 minute

B. Steam line isolation

1a. High steam line flow in coincidence with lo-lo T_{avg}
(FB-474A, FB-475A, FB-484A, FB-485A, FB-494A, FB-495A)

d/p corresponding to 110% of full steam flow at full load
40% of full steam flow between 0% and 20% load

d/p setpoint linear with turbine first stage pressure between 20% load and full load

1b. Lo-lo T_{avg} (P-12)
(TB-412E, TB-422E, TB-432E) — 543°F

1c. Filter Lag Time Constant
(FY-474B, FY-475B, FY-484B, FY-485B, FY-494B, FY-495B) — 0 Seconds*
*May be set up to 1.5 seconds

2. Low steam line pressure
(PB-474A, PB-485A, PB-496A) — 585 psig
Lead time constant
(PY-474B, PY-485B, PY-496B) — 50 seconds
Lag time constant
(PY-474B, PY-485B, PY-496B) — 5 seconds

3. High-2 containment pressure
(PB-951B, PB-952B, PB-953B) — 16.2 psig

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

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:

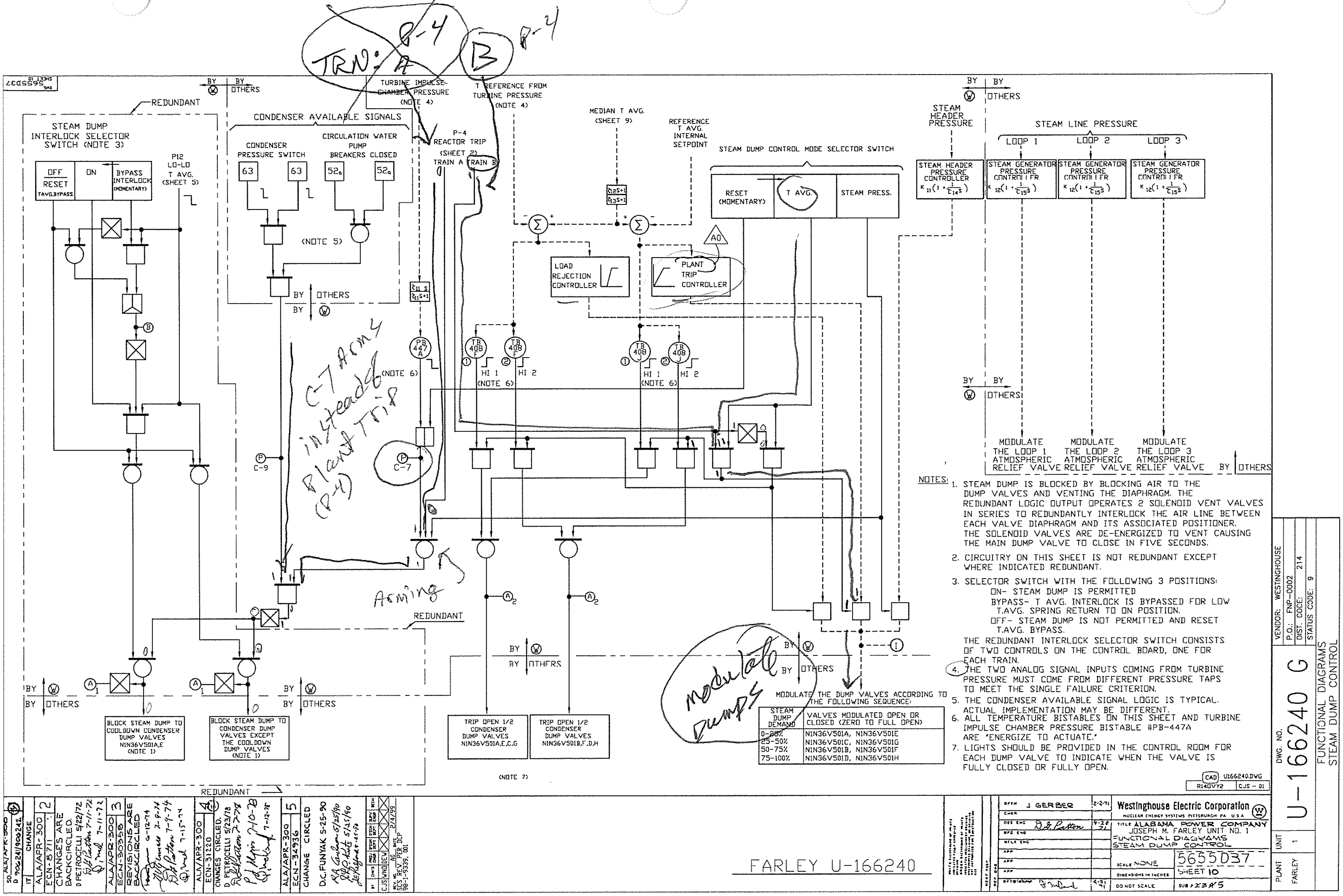
5. **DEFINE AND EVALUATE** the operational implications of abnormal plant or equipment conditions associated with the operation of the Steam Dump System components and equipment to include the following (OPS-52201G07):

- 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


TRN: A P-4
B P-4
STEAM DUMP INTERLOCK SELECTOR SWITCH (NOTE 3)
OFF RESET TAVG.BYPASS
ON
BYPASS INTERLOCK (MOMENTARY)
P12 LO-LO T AVG. (SHEET 5)
REDUNDANT
CONDENSER AVAILABLE SIGNALS
CONDENSER PRESSURE SWITCH
CIRCULATION WATER PUMP BREAKERS CLOSED
(NOTE 5)
TURBINE IMPULSE CHAMBER PRESSURE (NOTE 4)
T REFERENCE FROM TURBINE PRESSURE (NOTE 4)
P-4 REACTOR TRIP (SHEET 2) TRAIN A TRAIN B
MEDIAN T AVG. (SHEET 9)
REFERENCE T AVG. INTERNAL SETPOINT
STEAM DUMP CONTROL MODE SELECTOR SWITCH
RESET (MOMENTARY)
T AVG.
STEAM PRESS.
LOAD REJECTION CONTROLLER
PLANT TRIP CONTROLLER
STEAM HEADER PRESSURE
STEAM HEADER PRESSURE CONTROLLER
STEAM LINE PRESSURE
LOOP 1 LOOP 2 LOOP 3
STEAM GENERATOR PRESSURE CONTROLLER
MODULATE THE LOOP 1 ATMOSPHERIC RELIEF VALVE
MODULATE THE LOOP 2 ATMOSPHERIC RELIEF VALVE
MODULATE THE LOOP 3 ATMOSPHERIC RELIEF VALVE
BY W BY OTHERS
PB 447 A
TB 408
HI 1 HI 2 (NOTE 6)
C-9
C-7
C-7 Arming instead of Plant Trip (P-4)
Arming
Modulate Dumps
BLOCK STEAM DUMP TO COOLDOWN CONDENSER DUMP VALVES N1N36V501A,E (NOTE 1)
BLOCK STEAM DUMP TO CONDENSER DUMP VALVES EXCEPT THE COOLDOWN DUMP VALVES (NOTE 1)
TRIP OPEN 1/2 CONDENSER DUMP VALVES N1N36V501A,E,C,G
TRIP OPEN 1/2 CONDENSER DUMP VALVES N1N36V501B,F,D,H
(NOTE 7)
MODULATE THE DUMP VALVES ACCORDING TO THE FOLLOWING SEQUENCE:
STEAM DUMP DEMAND | VALVES MODULATED OPEN OR CLOSED (ZERO TO FULL OPEN)
0-25% | N1N36V501A, N1N36V501E
25-50% | N1N36V501C, N1N36V501G
50-75% | N1N36V501B, N1N36V501F
75-100% | N1N36V501D, N1N36V501H
NOTES:
1. STEAM DUMP IS BLOCKED BY BLOCKING AIR TO THE DUMP VALVES AND VENTING THE DIAPHRAGM. THE REDUNDANT LOGIC OUTPUT OPERATES 2 SOLENOID VENT VALVES IN SERIES TO REDUNDANTLY INTERLOCK THE AIR LINE BETWEEN EACH VALVE DIAPHRAGM AND ITS ASSOCIATED POSITIONER. THE SOLENOID VALVES ARE DE-ENERGIZED TO VENT CAUSING THE MAIN DUMP VALVE TO CLOSE IN FIVE SECONDS.
2. CIRCUITRY ON THIS SHEET IS NOT REDUNDANT EXCEPT WHERE INDICATED REDUNDANT.
3. SELECTOR SWITCH WITH THE FOLLOWING 3 POSITIONS:
ON- STEAM DUMP IS PERMITTED
BYPASS- T AVG. INTERLOCK IS BYPASSED FOR LOW T.AVG. SPRING RETURN TO ON POSITION.
OFF- STEAM DUMP IS NOT PERMITTED AND RESET T.AVG. BYPASS.
THE REDUNDANT INTERLOCK SELECTOR SWITCH CONSISTS OF TWO CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
4. THE TWO ANALOG SIGNAL INPUTS COMING FROM TURBINE PRESSURE MUST COME FROM DIFFERENT PRESSURE TAPS TO MEET THE SINGLE FAILURE CRITERION.
5. THE CONDENSER AVAILABLE SIGNAL LOGIC IS TYPICAL. ACTUAL IMPLEMENTATION MAY BE DIFFERENT.
6. ALL TEMPERATURE BISTABLES ON THIS SHEET AND TURBINE IMPULSE CHAMBER PRESSURE BISTABLE #PB-447A ARE 'ENERGIZE TO ACTUATE.'
7. LIGHTS SHOULD BE PROVIDED IN THE CONTROL ROOM FOR EACH DUMP VALVE TO INDICATE WHEN THE VALVE IS FULLY CLOSED OR FULLY OPEN.
CAD U166240.DWG
FARLEY U-166240
Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS PITTSBURGH PA U.S.A.
TITLE ALABAMA POWER COMPANY JOSEPH M. FARLEY UNIT NO. 1 FUNCTIONAL DIAGRAMS STEAM DUMP CONTROL
J GERBER
5655D37
SHEET 10
SCALE NONE
DO NOT SCALE
DWG. NO. U-166240 G
FUNCTIONAL DIAGRAMS STEAM DUMP CONTROL
VENDOR: WESTINGHOUSE
P.O.: FNP-0002
DIST. CODE: 214
STATUS CODE: 9
UNIT 1
PLANT FARLEY
CHANGE ECN-8711 ALA/APR-300
CHANGES ARE BACKCIRCLED.
ECN-30358 ALA/APR-300
REVISIONS ARE BACKCIRCLED
ECN-31220 ALA/APR-300
CHANGES CIRCLED.
ECN-34936 ALA/APR-300
CHANGE CIRCLED
D.C.FUNYAK 5-25-90

range channels exceeds a current equivalent to 20 percent reactor power. The rod stop may be manually blocked when above the P-10 setpoint, but is automatically reinstated below P-10 (3/4). This can be manually bypassed at NIS racks. (References 6.4.007, 6.4.011, 6.4.016)

C-2 Interlock. The C-2 overpower rod stop blocks automatic and manual control rod withdrawal. The block action occurs when 1/4 power range channels exceeds 103 percent reactor power. Each power range channel may be manually bypassed at the NIS racks. All power range channels cannot be bypassed at the same time. Only two power range channels may be blocked at one time using two switches located on the NIS Miscellaneous Control and Indication Drawer. (References 6.4.007, 6.4.011, 6.4.016)

C-3 Interlock. The C-3 control interlock is generated by the OTDT circuitry. The setpoint is 3 percent below the variable OTDT reactor trip setpoint. C-3 generates a block of automatic and manual rod withdrawal, when 2/3 loop delta Ts exceed their setpoint. The function of the rod block is to eliminate the cause of the impending trip, thereby preventing it. Since relatively slow transients are typical of those requiring OTDT protection, there is sufficient time for a load reduction to correct the situation. (References 6.2.003, 6.4.007, 6.4.012, 6.4.016)

C-4 Interlock. The C-4 control interlock is generated by the OPDT circuitry. The setpoint is 3 percent below the variable OPDT reactor trip setpoint. C-4 generates a block of automatic and manual rod withdrawal, when 2/3 loop delta Ts exceed their setpoint. The function of the rod block is to eliminate the cause of the impending trip, thereby preventing it. (References 6.2.003, 6.4.007, 6.4.012, 6.4.016)

C-5 Interlock. The C-5 interlock ensures that automatic rod withdrawal system is prevented when less than 15 percent power. It also prevents automatic rod withdrawal when power falls below 15 percent. The setpoint is 15 percent power as detected by turbine first stage impulse pressure. (References 6.4.007, 6.4.016, 6.4.022)

C-7 Interlock. This control interlock arms the steam dumps upon a load rejection (when in coincidence with C-9). The steam dump demand interlock (C-7) is actuated when turbine load is reduced by greater than 15 percent with a 120 second time constant. Rate differentiation of the first stage turbine impulse chamber pressure signal provides the equivalent turbine load signal. It must be manually reset. (Only PT-447 provides input to C-7.) (References 6.4.007, 6.4.017)

C-9 Interlock. C-9 is the condenser-available interlock. This interlock allows the steam dump valves to be armed if the condenser is available. It also prevents an overpressure condition which could damage the

- Pressurizer low pressure
- Pressurizer high level

Permissive P-7 shall block the above listed reactor trips below 10 percent of full power. The low power signal is derived from the power range neutron flux channels (P-10) and the turbine impulse chamber pressure channels (P-13). The blocking feature occurs during the absence of P-7, meaning that P-7 is not active. This occurs when 3/4 power range neutron flux channels are below setpoint and 2/2 turbine impulse pressure channels are below setpoint. The P-7 Permissive is active when 2/4 power range neutron flux channels or 1/2 turbine impulse pressure channels are above setpoint. When P-7 is active all previous blocked reactor trips are reinstated.

The P-8 Permissive blocks a reactor trip when the plant is below 30 percent of full power on a low reactor coolant flow or a RCP breaker open signal in any one loop (1/3 coincidence). The block action (absence of the P-8 Permissive signal) occurs when three out of four neutron flux power range signals are below the 30% setpoint. Thus, below the P-8 setpoint, the reactor will be allowed to operate with one inactive reactor coolant loop and trip will not occur until two loops are indicating low flow.

Permissive P-9 shall block a reactor trip following a turbine trip below 35 percent power which is based on the ability of the rod control system and steam dump system to adequately control T_{avg} on a 50% load rejection (FNP Tech Spec Section B2-8). If 2/4 power range nuclear instruments are above 35 percent power, a turbine trip will cause a reactor trip. If 3/4 power range nuclear instruments are below the P-9 setpoint, a turbine trip will not cause a reactor trip.

See Table T-3 for a comprehensive list of the protection system blocks. (References 6.4.007, 6.4.011, 6.4.022, 6.7.012)

2.8.2 ESF Permissives

1. P-4 Permissive

   The P-4 permissive is generated when both the reactor trip breaker and the bypass breaker, which physically bypasses it, are open. Train A of the reactor protection system uses RTA and BYB, and train B uses RTB and BYA. The following are functions of P-4:

   a. Causes a turbine trip

b. Main Feedwater Isolation - Closes main feedwater regulating valves and feedwater bypass valves if low T_{avg} (554°F) is also present and requires a manual reset.

c. Seals in feedwater isolation signal from safety injection or steam generator high-high water level

d. Resets high steam flow setpoint to 40 percent

e. Allows operator reset of the safety injection signal after a 60 second time delay

This feature ensures that the reactor is tripped and that all emergency core coolant system (ECCS) loads are started before the operator overrides what could be a spurious actuation signal. The block does not prevent the operator from reinitiating safety injection through use of either manual safety injection actuation switch

f. Arms steam dumps on a plant trip, defeats the output of the load rejection controller, and places the plant trip controller into control.

(References 6.4.007, 6.4.009, 6.7.012 6.7.059, 6.7.060)

2. P-11 Permissive

The permissive P-11 allows manual block of low pressurizer pressure safety injection actuation. This permits normal plant cooldown and depressurization.

When 2/3 pressurizer pressure bistables sense less than 2000 psig, the low pressurizer pressure safety injection signal may be manually blocked. Placing the train A and train B pressurizer safety injection block switches to BLOCK will now prevent low pressurizer pressure (1850 psig) from initiating safety injection. Each switch will initiate the block function in its respective protective train. If pressure rises above the P-11 setpoint on 2/3 channels, the block automatically clears. If 2/3 channels exceed the P-11 setpoint, and power is available, any shut ECCS accumulator isolation valves will automatically open. In addition, 2/3 pressurizer bistables below the P-11 setpoint blocks the automatic opening of the pressurizer power operated relief valves (PORVs). (References 6.4.007, 6.4.13, 6.7.012, 6.4.017)

6. speed gain — 32 steps/minute/°F[4]

7. manual rod speed

control rods	48 steps/minute[1]
shutdown rods	62 steps/minute[1]

2. <u>Steam Dump Control</u>

A. Impulse unit time constant of loss of load interlock channel

(PY-447C) — 120 sec.

B. Sudden load loss setpoint (C-7)

(PB-447A) — 15% of full load

C. Proportional gain in percent of total dump capacity per °F

	* Setpoint for full load	
	T_{avg} = 577.2°F	T_{avg} = 567.2°F
Loss of load controller (TY-408J)	9.0%/°F[1]	16.3%/°F[1]
Plant trip controller (TY-408L)	3.3%/°F[1]	5.0%/°F[1]

* For other full load T_{avg} between 567.2°F and 577.2°F, the proportional gain setpoints should be calculated as follows.

(a) Loss of load controller (TY-408J), %/°F

$$= \frac{100}{[(\text{Full Load } T_{avg} - T_{no\text{-}load})/2] - \text{Deadband (TY-408 J)}}$$

(b) Plant trip controller (TY-408L), %/°F

$$= \frac{100}{(\text{Full Load } T_{avg} - T_{no\text{-}load}) - \text{Deadband (TY-408 L)}}$$

There are 8 condenser dump valves. The controllers should be adjusted such that the dump capacity is linear with the output of TY-408J, TY-408L, or PC-464 (below). That is, the second bank does not begin to modulate open until the first bank has

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received a signal to modulate full open; etc. The sequence for modulating the valves closed is the reverse of the opening sequence; i.e., the fourth bank to open is the first bank to close, and the third bank starts to close after the fourth bank has received a signal to close; etc. The first four valves to modulate open are also the first four valves to be tripped open. The last four valves to modulate open are the last four valves to trip open. The two valves in the first bank are designated as the cooldown dump valves. The input ranges for modulation (full closed to full open) and the order for modulating the dump valves open are:

BANK	IN / OUT (Fully closed to fully open)	IN / OUT	VALVES
First bank (Cooldown valves)	4.0 - 8.0 ma ⇒ 0.0 - 2.5 v ⇒ 4 - 20 ma (N1C24TY-408S)	(N1C24TY-408N)	N1N36V501A[1] N1N36V501E[1]
Second bank	8.0 - 12.0 ma ⇒ 2.5 - 5.0 v ⇒ 4 - 20 ma (N1C24TY-408T)	(N1C24TY-408P)	N1N36V501C[1] N1N36V501G[1]
Third bank	12.0 - 16.0 ma ⇒ 5.0 - 7.5 v ⇒ 4 - 20 ma (N1C24TY-408U)	(N1C24TY-408Q)	N1N36V501B[1] N1N36V501F[1]
Fourth bank	16.0 - 20.0 ma ⇒ 7.5 - 10.0 v ⇒ 4 - 20 ma (N1C24TY-408V)	(N1C24TY-408R)	N1N36V501D[1] N1N36V501H[1]

D. Lead time constant (TY-408D) — 5 seconds[1]

E. Lag time constant (TY-408D) — 5 seconds[1]

F. Deadband steam dump controller for loss of load (TY-408J) — 4°F[1]

G. Deadband steam dump controller for plant trip (TY-408L) — 0°F

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H. High (T_{avg} - T_{ref}) (TB-408F)

	*Setpoint for full load T_{avg} = 577.2°F	T_{avg} = 567.2°F
First and second bank trip open (High 1)	9.5°F(1)	7.0°F(1)
Third and fourth bank trip open (High 2)	15.1°F(1)	10.1°F(1)

I. High (T_{avg} - $T_{no\text{-}load}$) (TB-408J)

	*Setpoint for full load T_{avg} = 577.2°F	T_{avg} = 567.2°F
First and second bank trip open (High 1)	15.1°F(1)	10.1°F(1)
Third and fourth bank trip open (High 2)	30.2°F(1)	20.2°F(1)

* For other full load T_{avg} between 577.2°F and 567.2°F, the setpoints should be calculated as follows:

High (T_{avg} - T_{ref}) Setpoints (TB-408F)

(a) High 1 (T_{avg} - T_{ref}) valve trip open (TB-408F), °F
(First and second bank trip open)
= [b below - deadband (TY-408J)]/2 + deadband (TY-408J)

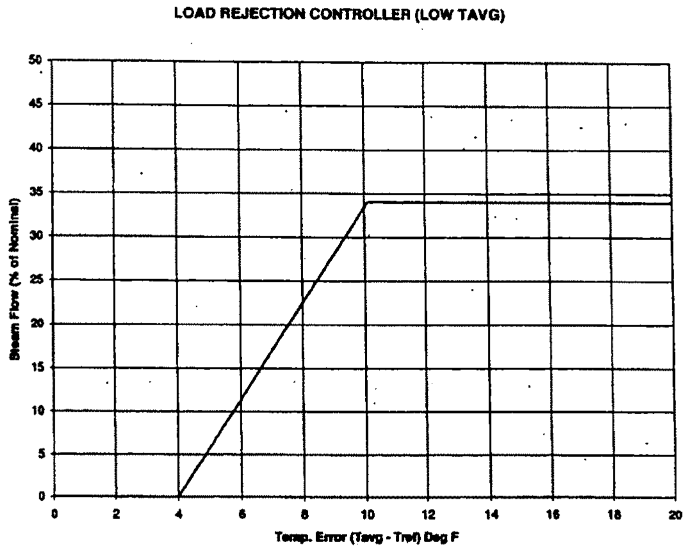
(b) High 2 (T_{avg} - T_{ref}) valve trip open (TB-408F), °F
(Third and fourth bank trip open)
= (Full Load T_{avg} - $T_{no\text{-}load}$)/2

High (T_{avg} - $T_{no\text{-}load}$) Setpoints (TB-408J)

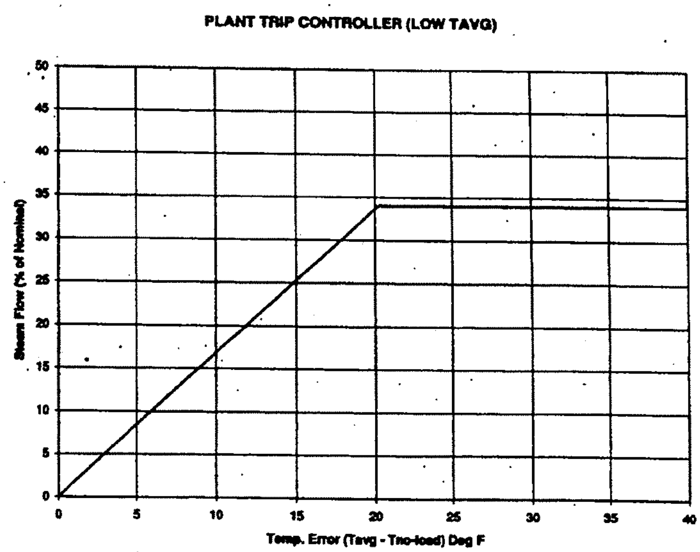
(a) High 1 (T_{avg} - $T_{no\text{-}load}$) valve trip open (TB-408J), °F
(First and second bank trip open)
= [b below - deadband (TY-408L)]/2 + deadband (TY-408L)

(b) High 2 (T_{avg} - $T_{no\text{-}load}$) valve trip open (TB-408J), °F
(Third and fourth bank trip open)
= (Full Load T_{avg} - $T_{no\text{-}load}$)

U-266647 P&L Document



4°F dead Band
547° + 4° = 551°F



No dead band
547°F

Steam Dump Control System (Low T_{avg})

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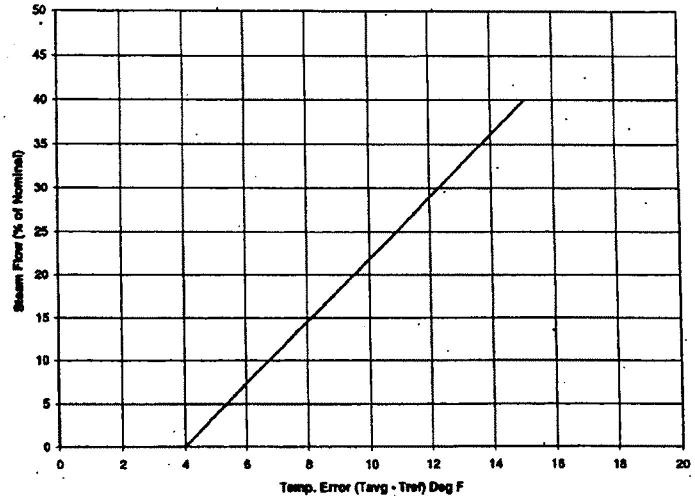
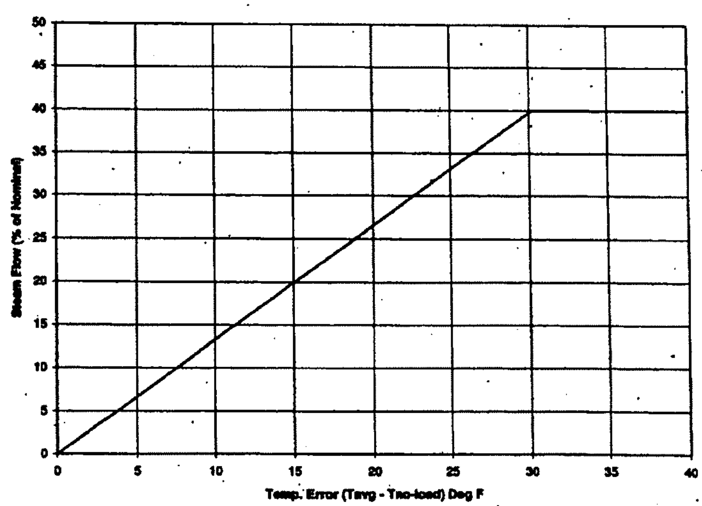
LOAD REJECTION CONTROLLER (HIGH TAVG)



4°F dead band

547° + 4°F = 551°F

PLANT TRIP CONTROLLER (HIGH TAVG)



No dead Band

547°F

Steam Dump Control System (High T_{avg})

J. Header pressure controller

(PC-464)

set pressure	1005 psig(1)
proportional band (based on total condenser dump capacity)	200 psi(1)
reset time constant	300 sec.(1)

K. Steam generator relief valve controllers

proportional band (valve full stroke)	250 psi(1)
reset time constant	8.5 sec.(1)
set pressure	1035 psig

3. Pressurizer Pressure Control (Refer to following figure)

A. Pressurizer pressure controller

(PC-444A)

proportional gain	0.5% controller output/psi
reset time constant	600 sec.(1)
rate time constant	0 sec.(1)
pressure setpoint, Pref	2235 psig (42.5% controller output)(4)

B. Spray valve controllers

(PC-444C, PC-444D)

proportional gain in percent spray valve lift per psi	4%/% controller output(1)
setpoint where spray is initiated on compensated pressure signal from PC-444A	55% controller output(1)

C. Variable heat controller proportional gain in percent

heating power per psi	-6.67%/% controller output(1)
setpoint where proportional heating is full on, on signal from PC-444A	35% controller output(1)

TABLE T-3 - REACTOR PERMISSIVES AND ESF PERMISSIVES

PERMISSIVE	MEASURED PARAMETER(S)	SETPOINT	COINCIDENCE & *LIGHT STATUS	FUNCTION	MODES OF OPERATION	FSD SECTION
P-4 Reactor trip interlock	Reactor trip and bypass breakers RTA and BYA or RTB and BYB	Breakers Open	- Either reactor trip breaker and its associated bypass breaker are open - No control board indication other than reactor trip and bypass breaker position indication on the reactor control panel	Prevents a rapid cooldown of the primary system after a reactor trip - Trips turbine - In coincidence with low T_{avg}, initiates closure of main feedwater reg valves and bypass valves - Prevents opening of main feedwater isolation valves if closed on safety injection or SG high-high water level signal - Prevents reactuation of automatic safety injection after safety injection manual reset - Resets high steam flow setpoint to 40% - Allows reset of safety injection signal after a time delay - Defeats the output of the Load Rejection Controller - Arms condenser steam dump valves	1, 2, 3 and 4	2.8.2 Fig. 2, Sht. 8, 10, 13, 15
P-6 Intermediate range neutron flux power escalation permissive	NIS intermediate range neutron flux channels NC35D and NC36D	10^{-10} amps	- Allows manual block of source range reactor trip on 1/2 intermediate range neutron flux channels > setpoint - Auto reinstates source range reactor trip on 2/2 intermediate range setpoint neutron flux below setpoint - Permissive status light is lit when 1/2 channels > setpoint	Allows power escalation into the Intermediate Power Range by turning both MCB Train A and B source range block switches to BLOCK above setpoint - Allows manual blocking of source range high neutron flux reactor trip - Actuating manual block handswitches deenergizes source range instruments - 2/2 intermediate range channels below setpoint auto reinstates high voltage to source range detectors and reinstates source range high neutron flux reactor trip	2 below the P-6 setpoint	2.8.1 2.6.1 Fig. 2 Sht. 4

* Designates Bypass and Permissive Light Box Status

35. 045A1.05 001/NEW/RO/C/A 3.8/4.1/N/N/2/CVR/Y

Unit 1 was at 30%, and the following conditions occurred:

- Control Rods are in Manual.
- The Main Turbine was manually tripped.

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

Tavg __(1)__ and RCS Pressure __(2)__.

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

A - Correct. Steam pressure goes up, causing Tcold to go up. This causes Tavg to go up. This causes a przr insurge which compresses the steam space and RCS pressure goes up.

B - Incorrect. The first part is correct (see A). The second part is incorrect (see A). Plausible, since the insurge is subcooled water from the RCS, but the steam space is compressed which causes pressure to go up. Once the Steam Dumps and/or SG Atmospherics open to reduce SG pressure, the RCS Tavg goes down and the outsurge causes pressure to go down to less than it was initially due to the subcooling of the przr liquid and steam space.

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 after the initial response.

Ran on simulator Laptop (IC 47)
Initial values:
Rods in Manual
29% Reactor power
2243 psig
554°F

Tripped turbine:
Initially: temp, press, went up, power went down.

Values 2 minutes later:
16.5% Reactor power stable NI (18% Delta T)
2209 psig
562.5°F (lower than peak of approx. 569°F)

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 Temperature and pressure following the Turbine Trip. Written to preclude steam dump operation due to testing steam dump operation elsewhere in exam.

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/CVR/Y

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.

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) the type of Boration required IAW the ARP for FE1?

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

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).

LOCATION FE1

E1
CONT ROD BANK POSITION LO

SETPOINT: Variable; 10 Steps Greater than LO-LO Alarm Setpoint.
$Z_{LO} = Z_{LO\text{-}LO} + K_4$
Where K_4 = 10 Steps (6.25 inches)

ORIGIN: Rod Insertion Limit Computer

PROBABLE CAUSE

NOTE:
- **Zinc Addition System injection will result in a continuous RCS dilution of as much as 1.7 gph, which may result in a reduction in shutdown margin if compensated for by inward rod motion instead of boration.**
- **This annunciator has REFLASH capability.**

Reactor Coolant System Boric Acid Concentration too low for Reactor Power Level due to:
A. Plant Transient
B. Xenon Transient
C. Dilution of RCS

AUTOMATIC ACTION

NONE

OPERATOR ACTION

1. Check indications and determine that actual control bank rod position is at low insertion limit.
   1.1 Click on Rod Supervision button on Applications Menu.
   1.2 Click on Rod Insertion Limits button.
   1.3 Determine if low insertion limit exceeded.
2. IF reactor coolant system dilution is in progress, THEN stop dilution.
3. IF a plant transient is in progress, THEN place the turbine load on "HOLD".
4. Refer to FNP-1-UOP-3.1, POWER OPERATIONS.
5. Borate the Control Bank "OUT" as necessary using the Boron Addition Nomographs. {**CMT 0008900**}
6. Refer to the Technical Specifications section on Reactivity Control.

B + D 2nd parts correct
[No specification of type of Boration required]

A + C 2nd parts incorrect

References: A-177100, Sh. 291; U-260610; U266647 PLS Document; Technical Specifications DCP 93-1-8587; {**CMTs 0008554, 0008887**}

LOCATION FE2

SETPOINT: Variable with Reactor Power as measured by ΔT and TAVG.

ORIGIN: Rod Insertion Limit Computer

E2
CONT ROD BANK POSITION LO-LO

PROBABLE CAUSE

NOTE:
- **Zinc Addition System injection will result in a continuous RCS dilution of as much as 1.7 gph, which may result in a reduction in shutdown margin if compensated for by inward rod motion instead of boration.**
- **This annunciator has REFLASH capability.**

1. Reactor Coolant System Boric Acid Concentration too low to ensure Reactor Protection under Accident conditions due to;
   A. Plant Transient
   B. Xenon Transient
   C. Dilution of RCS

AUTOMATIC ACTION

NONE

OPERATOR ACTION

A+C incorrect but plausible (would be correct if rods were 10 steps further inserted)

1. Check indications and determine that actual control bank rod position is at the low-low insertion limit.
   1.1 Click on Rod Supervision button on Applications Menu.
   1.2 Click on Rod Insertion Limits button.
   1.3 Determine if low-low insertion limit exceeded.
2. Emergency borate the reactor coolant system in accordance with FNP-1-AOP-27.0, EMERGENCY BORATION.
   {CMTs 0008555, 0008900}
3. <u>IF</u> a plant transient is in progress,
   <u>THEN</u> place turbine load on "HOLD".
4. Refer to FNP-1-UOP-3.1, POWER OPERATIONS.
5. Refer to the Technical Specifications section on Reactivity Control.

References: A-177100, Sh. 292; U-260610; U266647 PLS Document; Technical Specifications; DCP 93-1-8587; **{CMT 0008887}**

6.3.2 As a minimum, the Specific Reactivity Management Practices contained in Attachment 2 will be followed.

6.3.3 The SS shall maintain direct supervisory oversight of reactivity manipulations. This means the SS will approve each reactivity manipulation, with the exception of conditions described in step 6.3.8. During times where frequent reactivity manipulations are necessary, the SS can assign a reactivity management SRO to perform this function while the SS maintains oversight.

C&D 1st Part incorrect due to 6.3.8 ONLY

6.3.4 A reactivity brief shall take place at the beginning of each shift in modes 1 and 2. The reactivity brief should include expected reactivity manipulations during the shift needed to maintain current plant conditions or in the case of planned startups, shutdowns or power maneuvers the brief should include a discussion of reactivity changes that would be required to execute these power changes. In addition to this, the reactivity brief should include a discussion of pertinent current core reactivity parameters and any planned work activities that could potentially affect reactivity. The reactivity briefing sheet or OATC turnover sheet shall contain a list of degraded or out of service reactivity manipulation equipment.

6.3.5 When power reduction is necessary, only steam flow adjustments will be effective in reducing and maintaining reactor power below limits. While control rod insertion may appear to provide some immediate relief from high power conditions, the effects are temporary without reducing total steam flow and will only reduce nuclear instrument accuracy due to the resultant cooldown. Turbine load adjustments must be made to reduce and control reactor power, with control rods used primarily to maintain Tave on program during the power reduction. (PWR Only)

6.3.6 Peer checks will be used for reactivity changes, with the exception of conditions described in step 6.3.8.

6.3.7 During some plant operations, one or more of the various indications of reactor power may not be accurate. Therefore, control room operators should always monitor all indications of reactor power and maintain it within licensed limits.

6.3.8 Transient Conditions

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. The requirement to have another licensed operator peer check the reactivity manipulations under these conditions is also not required since it is unlikely that other licensed operators would be available during the manipulation. The SS shall be briefed as soon as possible on the amount of negative reactivity added (number of steps of rod insertion, amount of boron added (PWR Only), Recirculation Pump speed adjustments (BWR Only), etc.

A&B 1st Part correct

C&D 1st Part Not Correct

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

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).

C. Automatic Actions

1 Both MDAFWPs will automatically start on a trip of both SGFPs.

2 The TDAFWP will automatically start at 28% narrow range level in two steam generators.

3 A reactor trip will occur if any SG level decreases to 28% narrow range.

4 A turbine trip and feedwater isolation will occur if any SG level increases to 82% narrow range.

5 A reactor trip will occur if either intermediate range high flux trip bistable (TSLB3-2.1 or TSLB 3-2.2) does not reset before the reactor power is reduced below 10%.

6 Rapidly reducing turbine load may cause the steam dump system to operate in automatic. This will prevent further reduction in total steam flow. Dump operation in steam pressure mode at pot settings other than specified for 1005 psig will affect RCS cooldown rate.

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.

A & B 1st parts correct
A & C 2nd parts correct

C & D 1st parts incorrect
B & D 2nd parts incorrect.

Step	Action/Expected Response	Response Not Obtained

NOTE:
- Steps 1 through 1.3 AND 2 through 2.1 are IMMEDIATE ACTION steps.
- This procedure steps through probable condensate and feedwater system malfunctions in a systematic diagnostic manner. If the cause of the condensate and feed malfunction is known, THEN the associated procedure section (step) may be implemented immediately.
  1. Single SGFP trip - step 1
  2. Both SGFPs tripped - step 2
  3. SGFP malfunction - step 3 OBSERVE CAUTION prior to step 3
  4. Main feedwater regulating valve malfunction - step 4
  5. Loss of feedwater heater - step 5
  6. SGFP low suction pressure - step 6

Page included for continuity only

Step	Action/Expected Response	Response Not Obtained
__ 1	**Check only one SGFP - RUNNING**	1 Proceed to step 2.
1.1	Check generator load GREATER THAN 540 MW	1.1 Proceed to step 3 OBSERVE CAUTION prior to step 3.
1.2	Check rapid turbine load reduction required.	1.2 IF rapid turbine load reduction NOT required, THEN reduce turbine load using normal DEH controls as required AND proceed to step 3.
1.3	Check DEHC in OPERATOR AUTO	1.3 Perform the following
1.3.1	Depress the SGFP SETPOINT button on the DEHC keypad	a) IF required by generator load
1.3.2	Press the "P8" key	• THEN press FAST ACTION AND GV CLOSE pushbuttons
1.3.3	On the SGFP SETPOINT screen verify the following appear: [ ] A TARGET of "540" MW [ ] A RATE of "1200" MW/Min	• Release the FAST ACTION and GV CLOSE pushbuttons at ≈ 730 MWe as indicated on the digital DEHC display
1.3.4	Depress the "GO" pushbutton	b) IF not required by generator load, THEN press the GV CLOSE pushbutton as required to reduce load

Step 1 continued on next page

__Page Completed

Step	Action/Expected Response	Response Not Obtained
1.4	Monitor for correct DEHC system response	

NOTE: A boration of 1 GAL per reduced MW will limit rod insertion and assist in maintaining Delta I.

Step	Action/Expected Response	Response Not Obtained
1.5	Reduce reactor power to match turbine power using control rods and boron.	Page included for continuity only
1.5.1	Verify rod control in MANUAL.	
1.5.2	Adjust control rods in MANUAL to reduce reactor power and control RCS TAVG. • Manual Rod Control • Manual boration per FNP-1-SOP-2.3, CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM. • Emergency boration per FNP-1-SOP-2.3, CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM Figure 6.	
1.6	Check proper operation of the steam dumps.	
1.7	Verify automatic operation of the feedwater regulating valves adequate. [ ] 1A SG FW FLOW FK 478 [ ] 1B SG FW FLOW FK 478 [ ] 1C SG FW FLOW FK 498	1.7 Take manual control of the feedwater regulating valves to control SG level.

Step 1 continued on next page

__Page Completed

Step	Action/Expected Response	Response Not Obtained
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. ← 1 [ ] Match feed flow with steam flow. ← 2 [ ] Return feedwater regulating valves to AUTO. ← 3	1.8 IF SG narrow range levels NOT maintained greater than 28%, THEN trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.
1.9	Monitor feedwater flow and steam flow.	
1.10	Verify that feedwater and steam flow trend to approximately equal values for the target, turbine load.	
1.11	Maintain SG narrow range level approximately 65%.	

A & B 1st parts correct
C & D 1st parts incorrect
A & ~~C~~ 2nd parts correct
B & D 2nd parts incorrect

****

CAUTION: The LOSS OF LOAD INTERLOCK C 7A should not be reset in the event of a failure of PT-447 which actuates C-7A without consultation with the Operations manager.

****

Step	Action/Expected Response	Response Not Obtained
1.12	Check LOSS OF LOAD INTERLOCK C-7A on the BYP & PERMISSIVES panel NOT illuminated.	1.12 IF C-7A is to be reset, THEN perform the following 1.12.1 Verify that all steam dump valves indicate closed. 1.12.2 Verify 0 demand on STM HDR PRESS controller PK 464 and STM DUMP DEMAND TI408 1.12.3 Place STM DUMP INTLK TRAIN A and TRAIN B to OFF RESET 1.12.4 Place STM DUMP MODE SEL TRAINS A-B to RESET and then release to spring return to TAVG.

Step 1 continued on next page

__Page Completed

38. 055EK3.02 001/MOD/RO/MEM 4.3/4.6/N/N/2/CVR/Y

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 describes the **primary reason** for the step which directs dumping steam from intact SGs at maximum controllable rate?

A. To minimize potential for SGTR.

B.✓ To minimize RCS inventory loss.

C. To maximize TDAFW pump flow.

D. To prevent steam voiding in the reactor vessel upper head.

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.

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…**

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 lower SG pressure would allow more water to be pumped to the SG's in the initial 2 hour period. However, the background document requires only SG level of >31% narrow range on one SG for an adequate heat sink.

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.

ECP-0.0, Loss Of All Ac Power, Revision 22

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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)

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 2.3.

B Correct

Knowledge: SG depressurization should proceed as quickly as possible and should not be limited by the Technical Specification RCS cooldown limit of 100°F/hr.

References:

Justification of Differences:

1 Since the ERG Note contains a directed action it was incorporated into the ERP step for clarification and to enhance procedure flow.

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. This step is written for plants which utilize air operated valves (AOVs) in the itemized locations. The step structure assumes that the AOVs fail closed on loss of all ac power (i.e., loss of air supply). The operator, therefore, checks that the valves are closed. If any valve is open, the operator should attempt to close the valve. Reasons for a valve remaining open are plant specific, for example the valves may have legitimate or spurious open signals and air pressure could be available due to air receivers or air bottles located in the air supply system. Plants with air receivers may take up to 30 minutes to lose air pressure. If nitrogen bottles are provided for specific valves, such as PORVs, pneumatic pressure may be available for more than 30 minutes. The sequence for checking valves is based on capacity of the outflow lines and potential for RCS inventory loss: 1) The pressurizer PORVs are checked first. Since the turbine_driven AFW pump should be running, the secondary side is removing decay heat and RCS pressure should be under the pressurizer PORV setpoint. 2) The letdown line isolation valves adjacent to the RCS loop are checked second. These valves are normally open and receive a low pressurizer level isolation signal. If these valves, in conjunction with the letdown orifice isolation valves, remain open, a leak path to the pressurizer relief tank (PRT) via the letdown line relief valve may exist. These valves, including the letdown orifice isolation valves, if necessary, should be manually closed as soon as possible to isolate the letdown line and minimize RCS inventory loss prior to automatic isolation on low pressurizer level. Note that isolating the letdown line at the containment penetration will not isolate the letdown relief valve leak path to the PRT. STEP DESCRIPTION TABLE FOR ECA-0.0Step3 3) The excess letdown line isolation valves adjacent to the RCS loop are checked third. These valves are normally closed and do not receive a low pressurizer level isolation signal. If these valves are open a leak path to the PRT via the RCP seal return relief valve may exist. These valves should be closed to isolate the excess letdown line. Note that isolating the seal return line at the containment penetration will not isolate excess letdown inventory loss to the PRT via the seal return relief valve. 4) Any additional plant specific RCS outflow lines should be included. Following completion of this step, the only RCS inventory leakage path should be the RCP controlled leakage seals. Plants which utilize motor operated valves (MOVs) for letdown or excess letdown isolation will not be able to remotely close these valves to isolate these RCS outflow paths. These plants should isolate these lines at containment. 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. An alternative which can be evaluated based on plant specific considerations is to dispatch personnel inside containment to manually close the subject isolation valves.

B
Correct

Section: **Procedure**

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

References:

Justification of Differences:

1 Changed to make plant specific.

2 Used action verb "verify" vice "check" since the intent of the step is to ensure the RCS is isolated.

Step	Action/Expected Response	Response NOT Obtained
3.4	Verify all reactor vessel head vent valves - CLOSED. RX VESSEL HEAD VENT OUTER ISO [] Q1B13SV2213A [] Q1B13SV2213B RX VESSEL HEAD VENT INNER ISO [] Q1B13SV2214A [] Q1B13SV2214B	

****

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.

C incorrect

****

Step	Action/Expected Response	Response NOT Obtained
___ 4	**Verify total AFW flow GREATER THAN 395 gpm.** AFW FLOW TO 1A(1B,1C) SG [] FI 3229A [] FI 3229B [] FI 3229C AFW TOTAL FLOW [] FI 3229	4 Verify proper AFW alignment. 4.1 Verify TDAFWP running. TDAFWP STM SUPP FROM 1B(1C) SG [] MLB-4 1-3 lit [] MLB-4 2-3 lit [] MLB-4 3-3 lit TDAFWP SPEED [] SI 3411A > 3900 rpm TDAFWP SPEED CONT [] SIC 3405 adjusted to 100% 4.2 <u>IF</u> TDAFWP <u>NOT</u> running, <u>THEN</u> locally verify TDAFWP TRIP THROTTLE VLV Q1N12MOV3406 open. (100 ft, AUX BLDG TDAFWP room)

Step 4 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: Accumulator nitrogen injection into the RCS may result from reduction of SG pressure to less than 100 psig.

****

NOTE: Reduction of intact SGs pressure should continue even if pressurizer level is lost or reactor vessel head voiding occurs. D incorrect

___16 **Reduce intact SGs pressure to 200 psig.**

16.1 Check at least one intact SG narrow range level - GREATER THAN 31%{48%}.

16.1 Perform the following:

16.1.1 Maintain maximum AFW flow to intact SGs until narrow range SG level greater than 31%{48%} in at least one SG.

TDAFWP
SPEED CONT
[] SIC 3405 adjusted to 100%

16.1.2 WHEN narrow range level in at least one intact SG is greater than 31%{48%}, THEN perform steps 16.2 through 16.7.

16.1.3 Proceed to step 17.

Step 16 continued on next page.

___Page Completed

1. EC-0.0/.1.2-52532A03 003/HLT//MEM 4.3/4.6/EPE055EK3.02////

Bonl2

055EK3.02

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

- "Reduce intact SGs to 200 psig:"

Which ONE of the following correctly describes the reason for stopping the SG pressure reduction at 200 psig?

A. To prevent losing Pressurizer level.

B. To minimize RCS inventory loss out of the RCP seals.

C. To prevent steam voiding in the reactor vessel upper head.

D✓ To prevent injection of SI Accumulator nitrogen into the RCS.

EPE055EK3.02
055 Loss of Offsite and Onsite Power (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

A INCORRECT Plausible, since PRZR level may be lost as stated in ECP-0.0, and is an undesirable condition. Also the note below the caution says the SG pressure reduction should continue if voiding occurs or if Przr level was lost. For this event, depressurization of SGs should be continued even if this does occur so this answer is incorrect.

B INCORRECT This is given as the reason why the pressure reduction is being done, not why the pressure reduction is only to 200 psig. This distractor is plausible in that it is closely related to the FINAL PRESSURE to which the SGs are to be depressurized and describes the reason why the depressurization is done, not why it is stopped at 200 pisg. In other procedures, the accumulator MOVs are isolated and and pressure reduction is continued to 100 psig, but with no power, this procedure has pressure maintained at 200 psig. Seal Injection & CCW to thermal Barriers are lost in a LOSP with Loss of all AC power. Seals are a major concern in this procedure.

C INCORRECT Plausible, since voiding may occur as stated in ECP-0.0. For this event, depressurization of SGs should be continued even if this does occur so this answer is incorrect. Also the note below the caution in ECP-0.0 says the SG pressure reduction should continue if voiding occurs or if Przr level was lost.

D CORRECT This is given as the basis.

REFERENCES:
1. ECP-0.0, Loss of All AC Power, Step 28 and NOTES and CAUTION on page 28.
CAUTION: Accumulator nitrogen injection into the RCS may result from reduction of SG pressure to less than 100 psig.

Finally, the operator should be aware of the limiting low pressure necessary to prevent introduction of noncondensibles from the accumulators. Understanding these considerations, the operator will be able to depressurize and control secondary pressure to minimize RCS inventory loss while minimizing the possibility of introducing nitrogen into the RCS and returning the reactor core to a critical condition.

Reduce Intact SG Pressure to 200 psig
SGs are to be depressurized in the this step to maximize delivery (into the RCS) of the water contained in the SI accumulators while minimizing introducing nitrogen (into the RCS). Introducing nitrogen into the RCS could impede natural circulation by collecting in high points of the RCS piping (e.g., SG U-tubes). An ideal gas expansion calculation was used to determine the pressure in the RCS following a complete discharge of the contents in the accumulators. This RCS pressure is then correlated to the SG pressure (100 psig), which would be indicated when the accumulators have fully discharged.

2008 NRC exam

Technical Reference: ECP-0 rev 22 and and the background documents for ECP-0, FNP-0-ECB-0.0 Loss of ALL AC

Learning Objective:
State the basis for all cautions, notes, and actions associated with EEP-3 (OPS52530A03)

Comments:
- This question matches the K/A in that it asks the applicant to describe the reason for a particular action contained in the EOP for this event.

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

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).

LOCATION WD1

D1	
	1A INV FAULT

SETPOINT:
1. Battery near exhaustion 107V DC.
2. Inverter output undervoltage 108V AC

ORIGIN:
1. Battery near exhaustion X7 Voltage sense board
2. Inverter current limit A3 Ammerter Relay
3. Inverter output undervoltage K3 Relay via X8 voltage sense board
4. Inverter overheating X10 relay board
5. Out of sync X12 relay board
6. Fan failure
7. Bypass source supplying load

PROBABLE CAUSE

1. Bypass source carrying load.
2. Inverter out of sync with bypass supply.

AUTOMATIC ACTION

1. IF DC input voltage drops to 103 V DC, THEN inverter transfers to bypass source.
2. IF inverter fails, THEN the bypass source should carry the load.
3. An inverter fault when the bypass source is not available resulting in a loss of power to 1A 120 VAC Vital Instrument Panel will be indicated by the following:
   A. Source Range Channel 31 will be de-energized.
   B. Intermediate Range Channel 35 will be de-energized.

CAUTION: Outward rod motion is blocked by the High Power Rod Stop Bistable being tripped.

   C. NI-41 will be de-energized with associated alarms and indications.
   D. Annunciators FD3, FD4, DF1 and DK3 will alarm.
   E. No amperage indication on 1A Inverter ammeter.
   F. IF RCP breaker indication is lost > 35% power, the reactor will trip.

OPERATOR ACTION

NOTE: **The following controls are affected if 1A 120 VAC Vital Instrument Panel is De-energized:**

C correct

- **A TRN SSPS output relay power is lost.**
- **VCT Hi Lvl Divert Valve - Q1E21LCV115A diverts to the RHT if in auto.**
- **LTDN Hi Temp Divert Valve - Q1E21TCV143 bypasses the demineralizers.**
- **1A & 1B Reactor makeup water pumps start.**
- **1A & 1B BAT pumps start.**
- **RMW to Blender - Q1E21FCV114B and Boric Acid to blender - Q1E21FCV113A opens if Rx M/U Control System is in auto.**
- **If LT 112 VCT level is out of service, RWST to Chg Pump Suction Valves – Q1E21LCV115B & D open.**
- **Q1E21LCV460 will not close on PZR low level.**
- **Annunciator KG4, TURB TV closed alert, will be in alarm and bistable TSLB2, 14-1 will be lit.**
- **Annunciator KH5, TURB Auto/Stop oil press low, will be in alarm and bistable TSLB2, 13-1 will be lit.**

if in auto as verified by the simulator

Add B incorrect

D incorrect

1. IF 1A 120 VAC VITAL INSTRUMENT PANEL is de-energized, THEN immediately perform the following:
   A. IF a reactor trip occurs, THEN refer to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.
   B. Attempt to restore power from the bypass source by performing the following:
      1. IF the "BYPASS SOURCE AVAILABLE" lamp is illuminated on the inverter, THEN transfer 1A INVERTER MANUAL BYPASS SWITCH to the "BYPASS SOURCE TO LOAD" position.
      2. IF the "BYPASS SOURCE AVAILABLE" lamp is NOT illuminated on the inverter, THEN perform the following:
         - Verify 1A MCC Energized
         - Verify Closed Q1R17BKRFAF5L Supply to 1G 208V/120V REGULATED AC DISTRIBUTION PANEL
         - Verify closed Breaker #8 IN 1G REGULATED AC DISTRIBUTION PANEL
         - Transfer 1A INVERTER MANUAL BYPASS SWITCH to the BYPASS SOURCE TO LOAD position.

OPERATOR ACTION cont'd

2. Notify appropriate personnel to determine the cause and correct.

NOTE:	**Per Table 3 of FNP-0-ACP-52.1, Guidelines for Scheduling of On-Line Maintenance, A, B, C, D or F Inverters on bypass source are considered to be unavailable due to being status A1 for the Maintenance Rule. This unavailability should be logged for tracking purposes.**

2. Refer to Technical Specification 3.8.9 and 3.8.10.

3. IF 1A 120 VAC VITAL INSTRUMENT PANEL was de-energized, THEN perform the following when it is re-energized:

   A. Verify proper operation of Pressurizer level control and heaters.

   B. Reset hi flux positive rate trip signal on NI-41 and verify proper operation of NI-41.

   C. Verify the following CVCS components are correctly aligned for current plant conditions:

      - Q1E21LCV115A
      - Q1E21LCV115C & E
      - Q1E21LCV115B & D
      - Q1E21TCV143

   D. Verify the following Reactor Makeup Control System components are correctly aligned for current plant conditions:

      - Q1E21FCV113A
      - Q1E21FCV114B

   E. Stop unnecessary Reactor Makeup Water and BAT pumps.

   F. Verify all other MCB controls and indications have returned to normal.

4. Verify control systems outside the control room have returned to normal.

5. WHEN the cause of the fault has been determined AND corrected, THEN return 1A Inverter to service.

{CMT 0009705} {CMT 0005094} applies to entire annunciator

References: D-177024; U-279610; PCN B87-1-2899; D-177218, Sh. 2; D-177214; {CMT 0009705} {CMT 0005094}

40. 059A2.04 001/MOD/RO/C/A 2.9/3.4/N/N/3/CVR/Y

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 Feedwater flow rate required

and

2) the Main Feed System flowpath required for feeding the SGs IAW FRP-H.1?

A. 1) Feed ALL SGs at a minimum total flow of 395 gpm.

2) Use the Main Feedwater Regulating Valves.

B. 1) Feed ALL SGs at a minimum total flow of 395 gpm.

2) Use the Main Feedwater Regulating Bypass Valves.

C. 1) Feed ONE SG at a time at a flow limited to between 20-100 gpm.

2) Use the Main Feedwater Regulating Valves.

D✓ 1) Feed ONE SG at a time at a flow limited to between 20-100 gpm.

2) Use the Main Feedwater Regulating Bypass Valves.

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 plausible, since the MFRVs are the valves used most often with the SGFPs operating, and are used exclusively on a shutdown until AFW is supplying the SGs. Also, this would be correct per FRP-H.1 Step 7.21 RNO if any of the MFRB valves would not open.

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**. IF bleed and feed is imminent OR bleed and feed is in progress and RCS temperatures are rising, THEN there is no limit on the feed flow rate." FRP-H.1 STEP 7.21 states: "**Control feedwater regulating bypass valves** to supply main feedwater to intact SGs."

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 in the MFW system (impacts of feeding a dry S/G on the MFW system), and the required valve to use and flowrates required 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)

Step	Action/Expected Response	Response NOT Obtained
4	**Monitor CST level.**	
4.1	[CA] Check CST level greater than 5.3 ft. CST LVL [] LI 4132A [] LI 4132B	4.1 Align AFW pumps suction to SW using FNP-1-SOP-22.0, AUXILIARY FEEDWATER SYSTEM.
4.2	Align makeup to the CST from water treatment plant OR demin water system using FNP-1-SOP-5.0, DEMINERALIZED MAKEUP WATER SYSTEM, as necessary.	

NOTE:

- IF some form of secondary feed flow becomes imminent and normal charging is in service, THEN raising charging flow will reduce the potential subsequent loss of pressurizer level due to cooldown shrinkage.
- IF it is necessary to feed a hot, dry SG(s) [RCS hot leg temperature > 550°F AND SG wide range level < 12%{31%}], THEN it (they) should be fed one at a time at a flow rate of 20 gpm to 100 gpm until RCS hot leg temperature falls to less than 550°F. IF bleed and feed is imminent OR bleed and feed is in progress and RCS temperatures are rising, THEN there is no limit on the feed flow rate.

C+D 1st parts correct.

A+B 1st parts incorrect [temps are NOT rising]

Step	Action/Expected Response	Response NOT Obtained
5	**[CA] Try to establish AFW flow to at least one SG.**	
5.1	Verify blowdown from all SGs - ISOLATED. 1A(1B,1C) SGBD ISO [] Q1G24HV7614A closed [] Q1G24HV7614B closed [] Q1G24HV7614C closed	

Step 5 continued on next page.

___Page Completed

Step | **Action/Expected Response** | **Response NOT Obtained**

7.20 Adjust master speed controller to raise feedwater discharge header pressure to 50 psi greater than steam header pressure.

FW
HDR
PRESS
[] PI 508

STM
HDR
PRESS
[] PI 464A

B & D 2nd parts correct

A & C 2nd parts incorrect since the Bypass v/v's will work

7.21 Control feedwater regulating bypass valves to supply main feedwater to intact SGs.

Intact SG	1A	1B	1C
1A(1B,1C) SG FW BYP FLOW FK	[] 479 adjusted	[] 489 adjusted	[] 499 adjusted

7.21 Locally unlock and control main feedwater regulating valves with handwheels. (127 ft, AUX BLDG main steam valve room)

Intact SG	1A	1B	1C
1A(1B,1C) SG FW FLOW Q1C22FCV	[] 478	[] 488	[] 498
Key	Z-121	Z-120	Z-119

7.22 <u>WHEN</u> P-12 light lit, <u>THEN</u> perform the following.

7.22.1 Block low steam line pressure SI.

STM LINE PRESS SI
BLOCK - RESET
[] A TRN to BLOCK
[] B TRN to BLOCK

7.22.2 Verify blocked indication.

BYP & PERMISSIVE
STM LINE ISOL.
SAFETY INJ.
[] TRAIN A BLOCKED light lit
[] TRAIN B BLOCKED light lit

___Page Completed

1. FRP-H-52533F03 019/HLT//C/A 3.7/4.3/W/E05EA2.2////

Barr/R

059A2.04

Given the following:

- The reactor was tripped on simultaneous loss of both Steam Generator Feed Pumps.
- 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 595 °F and are still rising.
- SGFP "A" has been started.
- SG Wide Range Levels are:
  - A= 8%
  - B= 8%
  - C= 14%

Which ONE of the following describes the method and rate of establishing feedwater flow at this time?

A. Feed rate to A and B SGs is limited to between 20-100 gpm. No limit on the feed rate to C SG. Use the respective Feedwater Control Bypass Valves.

B.✓ There is no limit on the feed rate to any of the SGs. Use the respective Feedwater Control Bypass Valve.

C. Feed rate to A and B SGs is limited to between 20-100 gpm. No limit on the feed rate to C SG. Use the Feedwater Control Valves.

D. There is no limit on the feed rate to any of the SGs. Use the respective Feedwater Control Valves.

See note in FRP-H.1 regarding feeding of hot dry SGs.
2008 Harris NRC Exam

Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink) Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Technical Reference: FRP-H.1, Attachment 1

41. 059G2.1.19 001/NEW/RO/C/A 3.9/3.8/N/N/2/CVR/Y

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

- A Computer Alarm for 6A FW HTR Dump valve position is in.
- 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

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**

<u>AUTOMATIC ACTION</u>

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

LOCATION KC4

C4
FW HTR OR DRN TK LVL HI

SETPOINT: 1. Heater Drain Tanks: 4.25 inches above center of tank.
2. L.P. Heaters: 3.0 inches above normal liquid level.
3. H.P. Heaters: 3.0 inches above normal liquid level.

APPROXIMATE MCB LEVEL

HEATER:			HTR DRN TK:		
	1A&B	13"		A	51.75"
	2A&B	9"		B	51.75"
	3A&B	13"			
	4A&B	8"			
	6A&B	21"			

ORIGIN: The following Level Switches:

a) 1A HDT N1N26LI502
b) 1A Heater N1N21LI526
c) 2A Heater N1N21LI530
d) 3A Heater N1N21LI534
e) 4A Heater N1N21LI538
f) 6A Heater N1N21LI542
g) 1B HDT N1N21LI504
h) 1B Heater N1N21LI528
I) 2B Heater N1N21LI532
j) 3B Heater N1N21LI536
k) 4B Heater N1N21LI540
l) 6B Heater N1N21LI544

PROBABLE CAUSE

1. Heater tube leak or rupture.
2. Malfunction of a Level Controller on a Low Pressure or High Pressure Heater or Heater Drain Tank.

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.

B+D 2nd parts incorrect

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 ability to drain to the HDT. On high level, the #5 FW HTR extraction steam will close. (AI 2008202332)

OPERATOR ACTION

A+B 1st parts incorrect

C+D 1st parts correct

1. Determine which heater or drain tank level is high.
2. Verify that the dump to condenser valve of the affected heater or drain tank is open.
4. IF a heater tube leak OR rupture is indicated, THEN isolate the affected heater.
5. IF a feedwater heater malfunction is indicated, THEN go to FNP-1-AOP-13.0, CONDENSATE AND FEEDWATER MALFUNCTION
6. Refer to FNP-1-SOP-21.0 for limitations on Turbine operation with one or more Feedwater Heaters isolated.
6. Monitor Feedwater Heater and Drain Tank Levels to ensure that they are returning to normal.

7. Notify appropriate personnel to determine and correct the cause of the alarm.

TABLE 1

Heater	Setpoint for Extraction Stm MOV Isolation	Level Switch	Extraction Steam MOV	Supply Breaker
1A	18" above HLL	N1N39LS502A	N1N35V517A-N	N1R17BKRHBBB3
	18" above HLL	N1N39LS502B	N1N35V518A-N	N1R17BKRHBBB4
1B	18" above HLL	N1N39LS502A	N1N35V517B-N	N1R17BKRHBBB3
	18" above HLL	N1N39LS502B	N1N35V518B-N	N1R17BKRHBBB4
2A	18" above HLL	N1N39LS505A	N1N35V519A-N	N1R17BKRHBBC4
2B	18" above HLL	N1N39LS505B	N1N35V519B-N	N1R17BKRHBBC5
3A	18 ¼" above HLL	N1N39LS508A	N1N35V506A-N	N1R17BKRHAAA4
3B	18 ¼" above HLL	N1N39LS508B	N1N35V506B-N	N1R17BKRHAAA5
4A	14" above HLL	N1N39LS510A	N1N35V507A-N	N1R17BKRHBBA6
4B	14" above HLL	N1N39LS510B	N1N35V507B-N	N1R17BKRHBBB2
5A	18" above HLL	N1N39LS512A	N1N35V502A-N	N1R17BKRHAAB2
5B	18" above HLL	N1N39LS512B	N1N35V502B-N	N1R17BRKHAAB3
6A	18" above HLL <u>OR</u>	N1N39LS515A <u>OR</u>	N1N35V503A-N	N1R17BRKHAAA6
	closes @ 0-2/+0 opens@ 6 ± 2 ΔP	N1N35PDS547A <u>OR</u> B		
6B	18" above HLL <u>OR</u>	N1N39LS515B <u>OR</u>	N1N35V503B-N	N1R17BKRHAAB5
	closes @ 0-2/+0 opens@ 6 ± 2 ΔP	N1N35PDS547A <u>OR</u> B		

A&C 2nd parts correct

References: A-177100/459; A-170750/88&89; D-172570 thru D-172577; D-172578/1&2; D-172579/1&2; B-175968; B-170058/97&98; A-170256; A-170257

B&C 2nd parts incorrect

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

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.

Which one of the following describes the purpose of the actions directed by SOP-22.0, Auxiliary Feedwater System?

To establish availability of ________________

A. 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**.

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

4.7 Aligning Service Water to the AFW System.

CAUTION:	**Service water does not meet secondary makeup specifications and should only be used when required by emergency conditions.**

NOTES:	• **Refer to Tech. Specs. section 3.7.6 Condition A.** • **2 BOP keys from the Shift Support Supervisor's office will be required in the following steps.**

4.7.1 Notify Shift Chemist that SW will be added to the SG's.

NOTE:	**FNP-1-SOP-24.0, SERVICE WATER SYSTEM, requires starting/stopping SW pumps as necessary to maintain pressure in each header greater than 70 psig but less than 130 psig as indicated on PI-3001AA and PI-3001BA or by adjusting CCW HX DISCH FCV HIC 3009A(B, C).**

4.7.2 Verify service water is in operation per FNP-1-SOP-24.0, SERVICE WATER SYSTEM maintaining proper SW pressure.

4.7.3 Open MDAFWP SW SUPP: (BOP key operated switches)

- Q1N23MOV3209A A trn
- Q1N23MOV3209B B trn

4.7.4 Open: (BOP)

- MDAFWP SW SUPP Q1N23MOV3210A A trn
- MDAFWP SW SUPP Q1N23MOV3210B B trn
- TDAFWP SW SUPP Q1N23MOV3216. Both TRNS

4.7.4.1 IF necessary to align TDAFWP suction from B Train service water, THEN perform the following.

NOTE:	**During the following two steps service water trains will be temporarily cross-connected.**

4.7.4.1.1 Unlock and open B train service water to TDAFWP suction:

- Q1N23V015D (in 1B MDAFWP Room) *
- Q1N23V015C (above MDAFWP Room) *

4.7.4.1.2 Unlock and close A Train service water to TDAFWP suction:

- Q1N23V015B (above MDAFWP Rooms) *
- Q1N23V015A (in 1A MDAFWP Room) *

4.7.5 IF required, THEN place AFW system in operation per section 4.1 or 4.3 of this SOP.

*In-Plant (manual) valves

4.8 Condensate Storage Tank Feed and Bleed.

4.8.1 Verify CST LVL LI 4005B ≥ 12.5 ft and CST LVL LI 4132B is reading maximum indication.

4.8.2 Verify with Chemistry that water is within specifications to be drained to the yard drains.

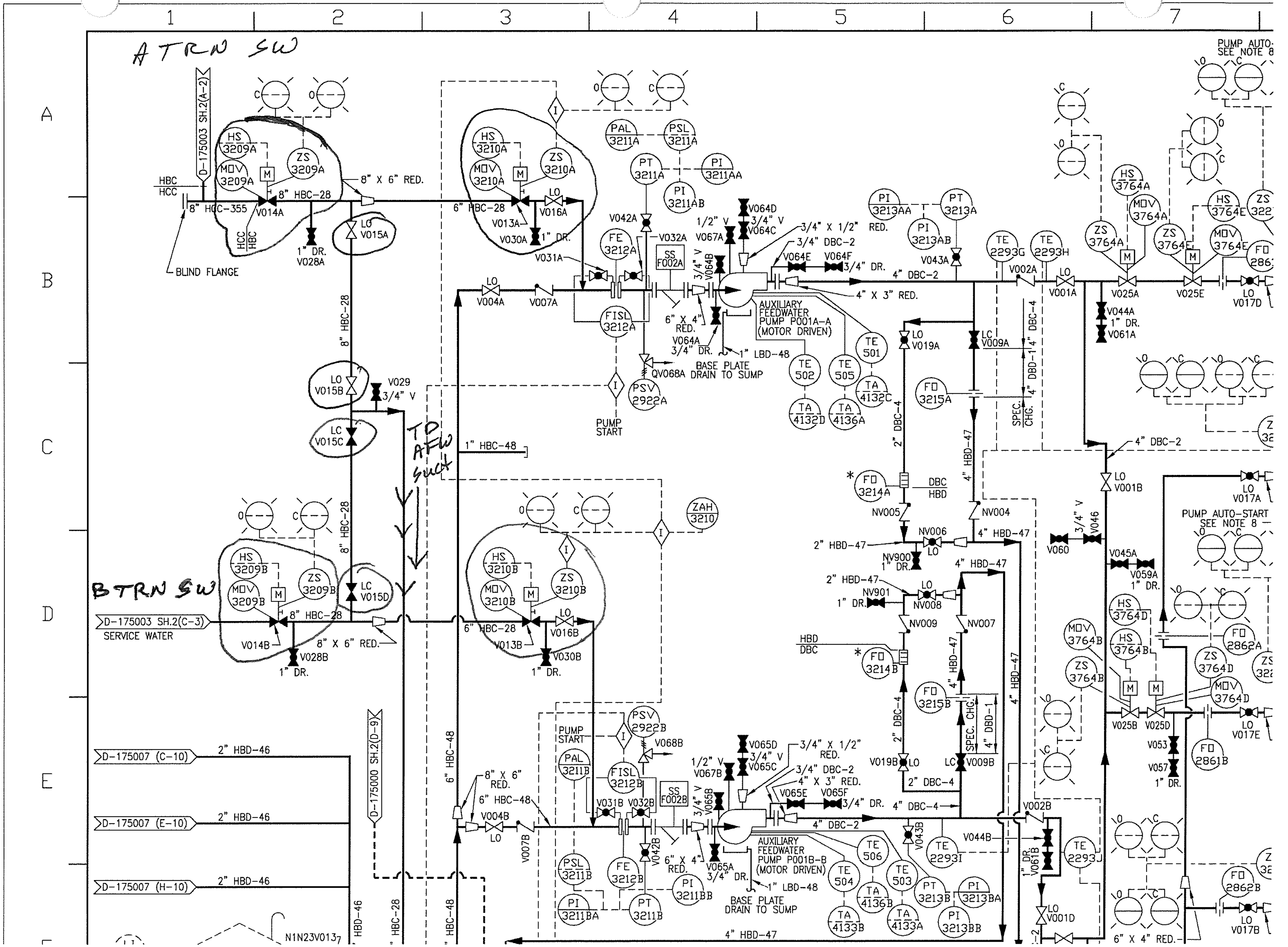
4.8.3 Remove blind flange and throttle open CST drain valve Q1P11V508.

4.8.4 Verify CST LVL LI 4005B ≥ 12.5 ft and CST LVL LI 4132B is reading its maximum indication at least once every 30 minutes while feed and bleed operation is in progress.

4.8.5 IF either level indication criteria is NOT met, THEN close CST drain valve (Q1P11V508) and restore CST level per FNP-1-SOP-21.0, CONDENSATE AND FEEDWATER SYSTEM.

4.8.6 WHEN feed and bleed operation is complete, THEN close CST drain valve (Q1P11V508) and replace blind flange.

4.8.7 Verify CST LVL LI 4005B ≥ 12.5 FT and CST LVL LI 4132B is reading its maximum indication.


ATRN SW
BTRN SW
TD AFW suct
BLIND FLANGE
SERVICE WATER
D-175003 SH.2(A-2)
D-175003 SH.2(C-3)
D-175007 (C-10)
D-175007 (E-10)
D-175007 (H-10)
D-175000 SH.2(D-9)
8" X 6" RED.
8" HBC-28
6" HBC-28
1" HBC-48
6" HBC-48
8" HCC-355
2" HBD-46
4" HBD-47
4" DBC-2
2" DBC-4
AUXILIARY FEEDWATER PUMP P001A-A (MOTOR DRIVEN)
AUXILIARY FEEDWATER PUMP P001B-B (MOTOR DRIVEN)
BASE PLATE DRAIN TO SUMP
PUMP START
PUMP AUTO-START SEE NOTE 8
SPEC. CHG.
4" X 3" RED.
6" X 4" RED.
3/4" X 1/2" RED.
1" LBD-48
N1N23V013

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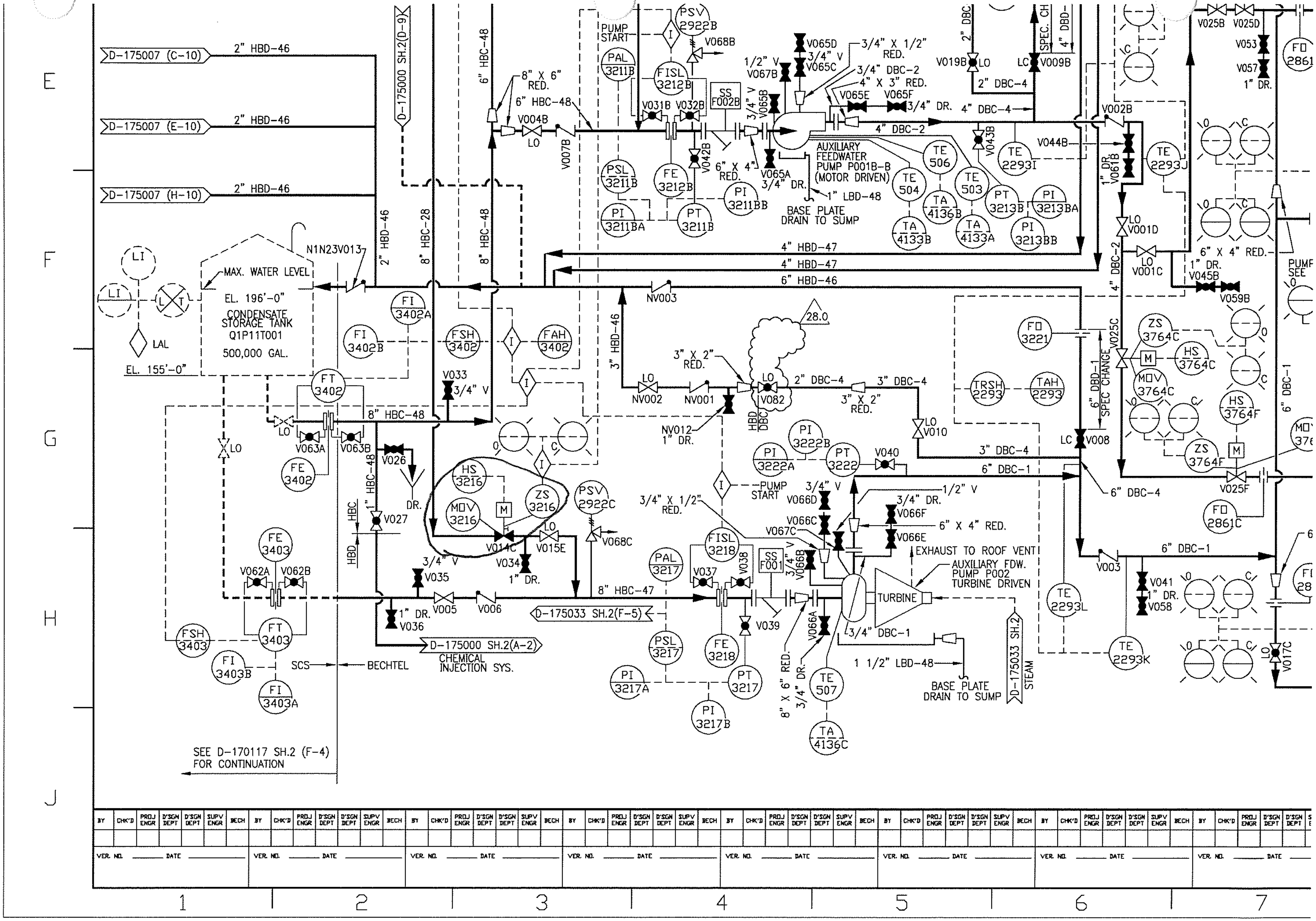

Unit 1 has had a LOSP followed by a SBLOCA, and the following conditions exist:

- 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

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 4

Step	Action/Expected Response	Response NOT Obtained
1.5	Verify BKR DG02 (1G 4160 V bus tie to 1L 4160 V bus) - CLOSED.	1.5 IF diesel generator cooling NOT supplied, THEN secure 1B diesel generator using ATTACHMENT 5, SECURING A DIESEL GENERATOR WITH A SAFETY INJECTION SIGNAL PRESENT.
1.6	Verify all RCP busses - ENERGIZED. [] 1A 4160 V bus [] 1B 4160 V bus [] 1C 4160 V bus	
1.7	Check 1E 4160 V bus - ENERGIZED.	1.7 Establish power to 1C 600 V LC emergency section loads. 1.7.1 Place handswitch for pressurizer heater group 1B in OFF. 1.7.2 Open BKR EC08-1. 1.7.3 Close BKRs EE07-1 and EC10-1.

****

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.

2.860+0.3= 3.160 MW

****

Step	Action/Expected Response	Response NOT Obtained
		1.7.4 Energize pressurizer heater group 1B as required.
1.8	Check 1D 4160 V bus - ENERGIZED.	1.8 Proceed to step 1.10.
1.9	IF 1D 4160 V bus energized, THEN return to PROCEDURE STEPS, step 13.	

Step 1 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 4

VERIFYING 4160 V BUSSES ENERGIZED

1 **Verify 4160 V busses energized.**

A&C 2nd parts incorrect
B&D 2nd parts correct

****

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.

→ 3.160 MW > 3.1 MW

****

A+B 1st parts correct
C&D 1st parts incorrect

NOTE: Plant conditions may dictate establishment of contingency electrical lineups. FNP-1-AOP-5.1, CONTINGENCY ELECTRICAL ALIGNMENTS provides guidance for establishing those lineups.

Step	Action/Expected Response	Response NOT Obtained
1.1	Check offsite power - AVAILABLE.	1.1 Request Shift Manager coordinate efforts to restore offsite power.
1.2	Check BKR DF01 (1A startup transformer to 1F 4160 V bus) - CLOSED.	1.2 Verify 1F 4160 V bus energized by 1-2A or 1C diesel generator.
1.3	Verify BKR DF02 (1F 4160 V bus tie to 1K 4160 V bus) - CLOSED.	1.3 IF diesel generator cooling NOT supplied from Unit 2, THEN secure 1-2A and/or 1C diesel generator using ATTACHMENT 5, SECURING A DIESEL GENERATOR WITH A SAFETY INJECTION SIGNAL PRESENT.
1.4	Check BKR DG15 (1B startup transformer to 1G 4160 V bus) - CLOSED.	1.4 Verify 1G 4160 V bus energized by 1B diesel generator.

Step 1 continued on next page.

___Page Completed

44. 063A3.01 001/FNP BANK/RO/C/A 2.7/3.1/Y 2007/N/4/CVR/VER 5 EDITORIAL

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

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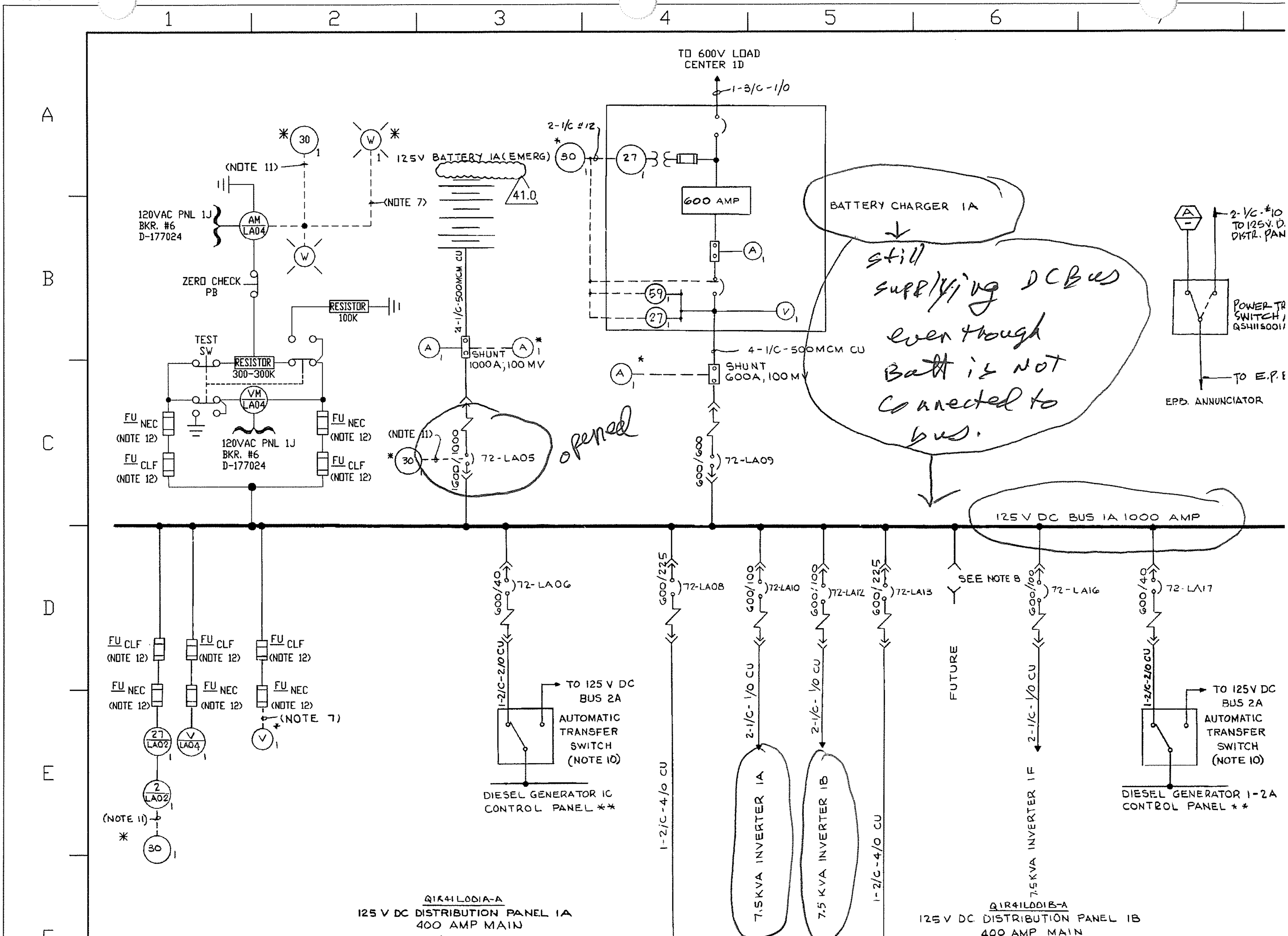

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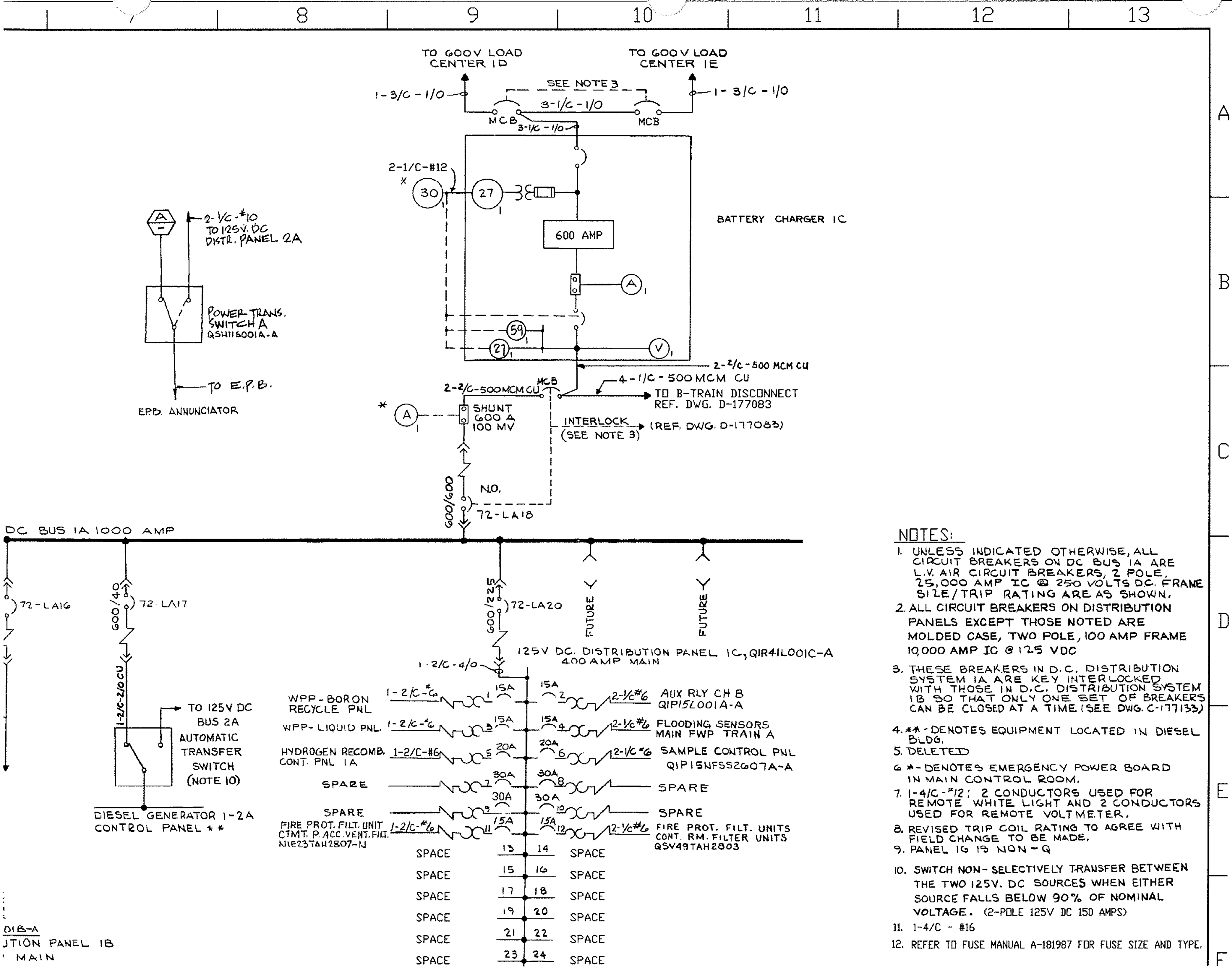
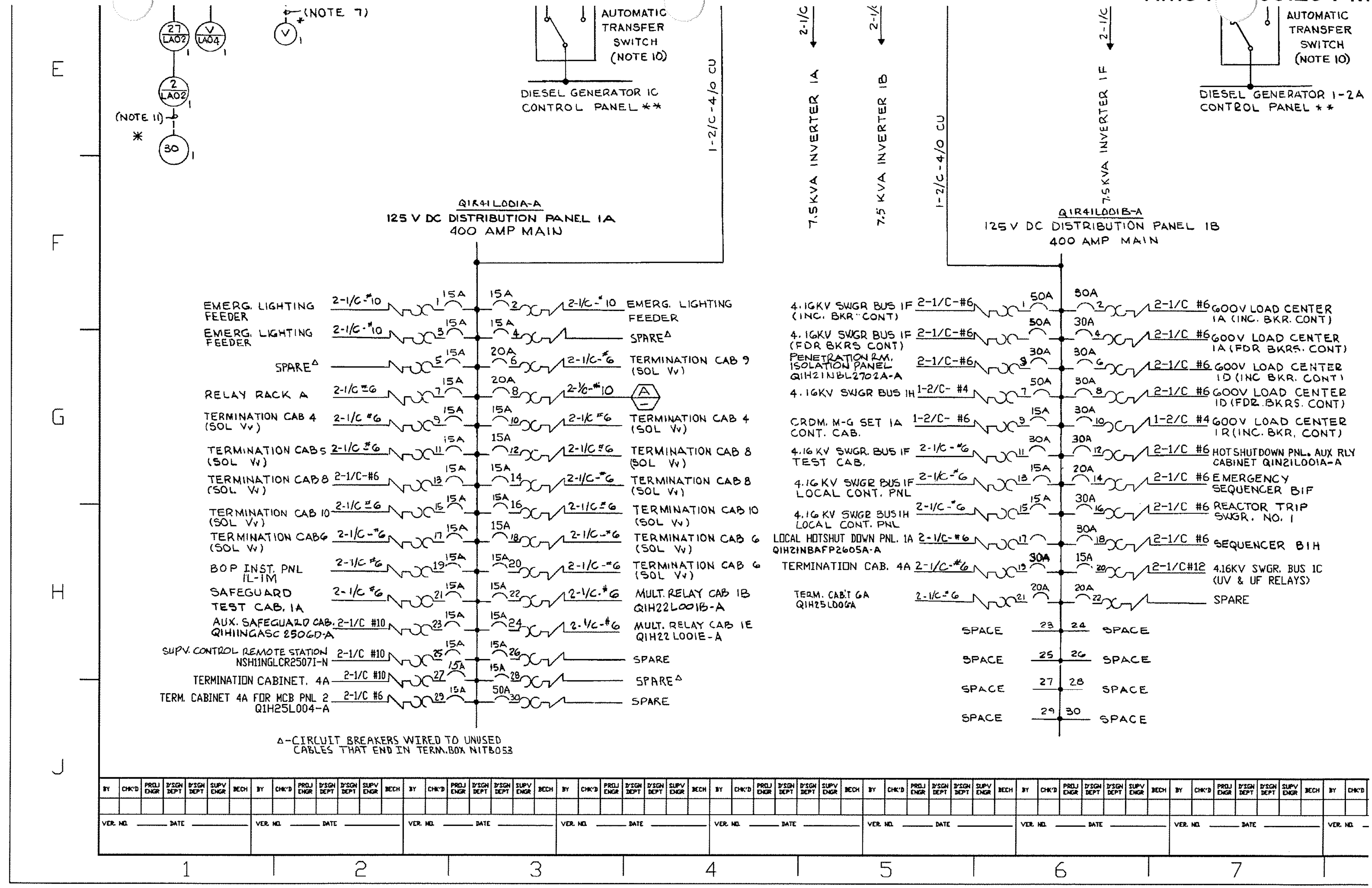
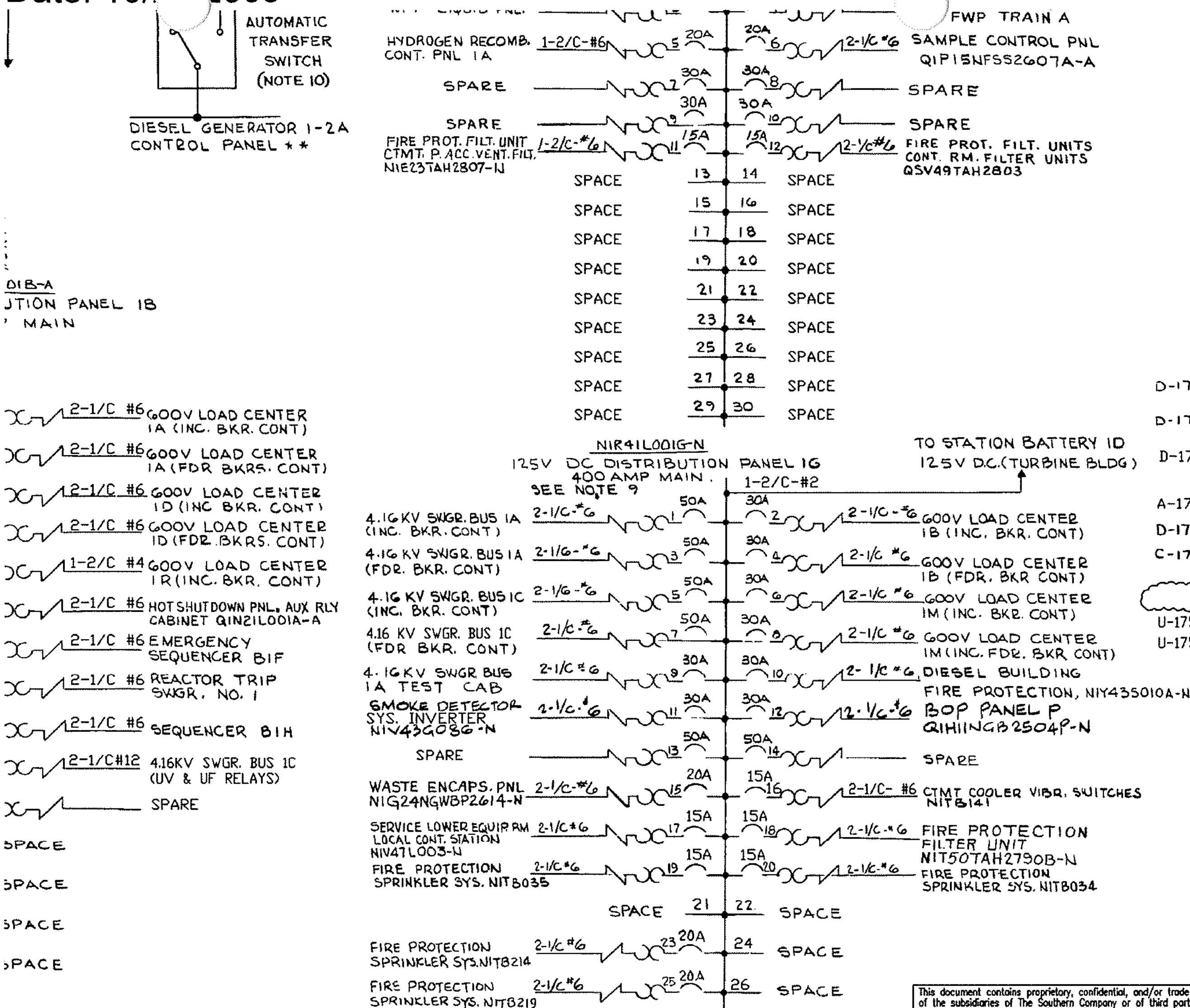


NOTES:

1. UNLESS INDICATED OTHERWISE, ALL CIRCUIT BREAKERS ON DC BUS 1A ARE L.V. AIR CIRCUIT BREAKERS, 2 POLE, 25,000 AMP IC @ 250 VOLTS D.C. FRAME SIZE/TRIP RATING ARE AS SHOWN.
2. ALL CIRCUIT BREAKERS ON DISTRIBUTION PANELS EXCEPT THOSE NOTED ARE MOLDED CASE, TWO POLE, 100 AMP FRAME 10,000 AMP IC @ 125 VDC
3. THESE BREAKERS IN D.C. DISTRIBUTION SYSTEM 1A ARE KEY INTERLOCKED WITH THOSE IN D.C. DISTRIBUTION SYSTEM 1B SO THAT ONLY ONE SET OF BREAKERS CAN BE CLOSED AT A TIME (SEE DWG. C-177133)
4. ** - DENOTES EQUIPMENT LOCATED IN DIESEL BLDG.
5. DELETED
6. * - DENOTES EMERGENCY POWER BOARD IN MAIN CONTROL ROOM.
7. 1-4/C-#12; 2 CONDUCTORS USED FOR REMOTE WHITE LIGHT AND 2 CONDUCTORS USED FOR REMOTE VOLTMETER.
8. REVISED TRIP COIL RATING TO AGREE WITH FIELD CHANGE TO BE MADE.
9. PANEL 1G IS NON-Q
10. SWITCH NON-SELECTIVELY TRANSFER BETWEEN THE TWO 125V. DC SOURCES WHEN EITHER SOURCE FALLS BELOW 90% OF NOMINAL VOLTAGE. (2-POLE 125V DC 150 AMPS)
11. 1-4/C - #16
12. REFER TO FUSE MANUAL A-181987 FOR FUSE SIZE AND TYPE.



E
F
G
H
J
27 LA02
V LA04
(NOTE 7)
V
2 LA02
(NOTE 11)
30
AUTOMATIC TRANSFER SWITCH (NOTE 10)
DIESEL GENERATOR 1C CONTROL PANEL **
1-2/C-4/0 CU
2-1/C
7.5KVA INVERTER 1A
7.5 KVA INVERTER 1B
7.5KVA INVERTER 1F
1-2/C-4/0 CU
DIESEL GENERATOR 1-2A CONTROL PANEL **
Q1R41L001A-A
125 V DC DISTRIBUTION PANEL 1A
400 AMP MAIN
Q1R41L001B-A
125 V DC DISTRIBUTION PANEL 1B
400 AMP MAIN
EMERG. LIGHTING FEEDER 2-1/C-#10 15A 1 | 2 15A 2-1/C-#10 EMERG. LIGHTING FEEDER
EMERG. LIGHTING FEEDER 2-1/C-#10 15A 3 | 4 15A SPARE△
SPARE△ 15A 5 | 6 20A 2-1/C-#6 TERMINATION CAB 9 (SOL Vv)
RELAY RACK A 2-1/C #6 15A 7 | 8 20A 2-1/C-#10 A
TERMINATION CAB 4 (SOL Vv) 2-1/C #6 15A 9 | 10 15A 2-1/C #6 TERMINATION CAB 4 (SOL Vv)
TERMINATION CAB 5 (SOL Vv) 2-1/C #6 15A 11 | 12 15A 2-1/C #6 TERMINATION CAB 8 (SOL Vv)
TERMINATION CAB 8 (SOL Vv) 2-1/C-#6 15A 13 | 14 15A 2-1/C-#6 TERMINATION CAB 8 (SOL Vv)
TERMINATION CAB 10 (SOL Vv) 2-1/C #6 15A 15 | 16 15A 2-1/C #6 TERMINATION CAB 10 (SOL Vv)
TERMINATION CAB 6 (SOL Vv) 2-1/C-#6 15A 17 | 18 15A 2-1/C-#6 TERMINATION CAB 6 (SOL Vv)
BOP INST. PNL 1L-1M 2-1/C #6 15A 19 | 20 15A 2-1/C-#6 TERMINATION CAB 6 (SOL Vv)
SAFEGUARD TEST CAB. 1A 2-1/C #6 15A 21 | 22 15A 2-1/C-#6 MULT. RELAY CAB 1B Q1H22L001B-A
AUX. SAFEGUARD CAB. Q1H1INGASC2506D-A 2-1/C #10 15A 23 | 24 15A 2-1/C-#6 MULT. RELAY CAB 1E Q1H22L001E-A
SUPV. CONTROL REMOTE STATION NSH11NGLCR2507I-N 2-1/C #10 15A 25 | 26 15A SPARE
TERMINATION CABINET. 4A 2-1/C #10 15A 27 | 28 15A SPARE△
TERM. CABINET 4A FOR MCB PNL 2 Q1H25L004-A 2-1/C #6 15A 29 | 30 50A SPARE
△-CIRCUIT BREAKERS WIRED TO UNUSED CABLES THAT END IN TERM.BOX N1TB053
4.16KV SWGR BUS 1F (INC. BKR CONT) 2-1/C-#6 50A 1 | 2 30A 2-1/C #6 600V LOAD CENTER 1A (INC. BKR. CONT)
4.16KV SWGR BUS 1F (FDR BKRS CONT) 2-1/C-#6 50A 3 | 4 30A 2-1/C #6 600V LOAD CENTER 1A (FDR BKRS. CONT)
PENETRATION RM. ISOLATION PANEL Q1H21NBL2702A-A 2-1/C-#6 30A 5 | 6 30A 2-1/C #6 600V LOAD CENTER 1D (INC BKR. CONT)
4.16KV SWGR BUS 1H 1-2/C- #4 50A 7 | 8 30A 2-1/C #6 600V LOAD CENTER 1D (FDR. BKRS. CONT)
CRDM. M-G SET 1A CONT. CAB. 1-2/C- #6 15A 9 | 10 30A 1-2/C #4 600V LOAD CENTER 1R (INC. BKR. CONT)
4.16 KV SWGR. BUS 1F TEST CAB. 2-1/C-#6 30A 11 | 12 30A 2-1/C #6 HOT SHUTDOWN PNL. AUX RLY CABINET Q1N21L001A-A
4.16 KV SWGR BUS 1F LOCAL CONT. PNL 2-1/C-#6 15A 13 | 14 20A 2-1/C #6 EMERGENCY SEQUENCER B1F
4.16 KV SWGR BUS 1H LOCAL CONT. PNL 2-1/C-#6 15A 15 | 16 30A 2-1/C #6 REACTOR TRIP SWGR. NO. 1
LOCAL HOTSHUT DOWN PNL. 1A Q1H21NBAFP2605A-A 2-1/C-#6 17 | 18 30A 2-1/C #6 SEQUENCER B1H
TERMINATION CAB. 4A 2-1/C-#6 30A 19 | 20 15A 2-1/C #12 4.16KV SWGR. BUS 1C (UV & UF RELAYS)
TERM. CAB'T 6A Q1H25L006A 2-1/C-#6 20A 21 | 22 20A SPARE
SPACE 23 | 24 SPACE
SPACE 25 | 26 SPACE
SPACE 27 | 28 SPACE
SPACE 29 | 30 SPACE
BY CHK'D PROJ ENGR D'SGN DEPT D'SGN DEPT SUPV ENGR BECH
VER. NO. DATE
1 2 3 4 5 6 7



4. ** - DENOTES EQUIPMENT LOCATED IN DIESEL BLDG.
5. DELETED
6. * - DENOTES EMERGENCY POWER BOARD IN MAIN CONTROL ROOM.
7. 1-4/C-#12; 2 CONDUCTORS USED FOR REMOTE WHITE LIGHT AND 2 CONDUCTORS USED FOR REMOTE VOLTMETER.
8. REVISED TRIP COIL RATING TO AGREE WITH FIELD CHANGE TO BE MADE.
9. PANEL 1G IS NON-Q
10. SWITCH NON-SELECTIVELY TRANSFER BETWEEN THE TWO 125V. DC SOURCES WHEN EITHER SOURCE FALLS BELOW 90% OF NOMINAL VOLTAGE. (2-POLE 125V DC 150 AMPS)
11. 1-4/C - #16
12. REFER TO FUSE MANUAL A-181987 FOR FUSE SIZE AND TYPE.

REFERENCE DRAWINGS:

D-177000	SINGLE LINE ELECTRICAL AUXILIARY SYSTEM (NORMAL 4160V & 600V)
D-177001	SINGLE LINE ELECTRICAL AUXILIARY SYSTEM (EMERGENCY 4160V & 600V)
D-177024	SINGLE LINE, 120V AC VITAL & REGULATED SYSTEM A
A-177538	ELECTRICAL GENERAL DETAILS & NOTES
D-177083	SINGLE LINE DC DISTRIBUTION SYSTEM 1B
C-177133	INTERLOCK SCHEMATIC BATTERY CHARGER 1C
U-175725	DC SWGR. LA05 DC GROUND DETECTOR
U-175727	DC SWGR. LA02 CONTROL CIRCUIT

41.0

CAD D1770821
0VY2002 DCP-01

Southern Company Services, Inc. for

ALABAMA POWER COMPANY
PROJECT J.M. FARLEY NUCLEAR PLANT - UNIT NO.1
SUBJECT SINGLE LINE D.C. DISTRIBUTION SYSTEM 1A
DRAWN M.A. CHECKED B.T. DESIGNED
APPROVED M. MALCOM DATE 12/4/70
SCALE NONE
SHEET 1 OF — SHEETS
D-177082

DCP MLH DEW
VER. NO. 41.0 DATE 06/07/06
REVISED PER ABN 1049005501E002, VER 1.0

7 8 9 10 11 12 13

OPERATIONS CRITICAL DRAWING

45. 064A4.03 001/NEW/RO/MEM 3.2/3.3/N/N/3/CVR/VER 5 EDITORIAL

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) the **direction** that the GOVERNOR MOTOR SPEED/MW switch must be turned to adjust frequency prior to closing the output breaker?

	(1)	(2)
A.	1G Bus Frequency	RAISE
B.	1G Bus Frequency	LOWER
C.	1B DG Frequency	RAISE
D.✓	1B DG Frequency	LOWER

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. Also, 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 D). Plausible, since it would be correct if the synchroscope was traveling in the other direction for this condition, or if the DG was powering the bus and off site power was being restored to the bus. The LOWER is correct for the given conditions.

C - Incorrect. The first part is correct, but the second part is incorrect. Plausible, since it would be correct if the synchroscope was either going too slow in the same direction, or traveling at any speed in the other direction.

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 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

NOTE:	**To prevent entering an LCO, Step 5.8 should be completed in its entirety. If the entire step can NOT be completed, the Shift Supervisor shall declare the D/G inoperable at the time of synchronization (Step 5.9) and enter the applicable LCO. (CR 2008105195)**

_____ 5.8 To ensure the DG load is within the 300 hour rating (4474kW) in the event an LOSP occurs with the DG connected to the bus, verify the following equipment not in service:

[ ] B SW Screen Wash Pump IF 1-2L LC aligned to Unit 1
[ ] B RW Screen Wash Pump IF 1-2J LC aligned to Unit 1
[ ] MDFP
[ ] #4 RW Pump
[ ] #5 RW Pump

NOTE:	**When running two DGs simultaneously, DGs should NOT be synchronized and tied to the same unit. For routine surveillance the diesel is normally aligned to the unit number corresponding to the air header number aligned as indicated in the surveillance schedule.**

*5.9 Synchronize 1B Diesel as follows:

5.9.1 Verify an operator is standing by at DG08-1 for breaker observation.

_____ 5.9.2 Place SYNCH SWITCH for 1B DG output breaker (DG08-1) in MAN position.

NOTE:	**Initialing for completion of Steps 5.9.3 and 5.9.4 may be done after Step 5.9.5.**

_____ 5.9.3 Establish and maintain the following conditions until Step 5.9.4 is completed:

- Adjust generator voltage to match running voltage.
- Adjust generator frequency to establish a slow synchroscope speed in the FAST direction.

CAUTION:	**It is important to expeditiously raise load to ≈ 50 kW after breaker closure to prevent a reverse power trip. If the switch for the output breaker and the governor motor switch are physically far apart, consideration should be given to utilizing two operators for the following two steps. (OR 1-2000-282)**

_____ 5.9.4 Just prior to the synchroscope reaching 12:00 position, close 1B DG output breaker.

46. 065AK3.04 001/FNP BANK/RO/C/A 3.0/3.2/N/N/2/CVR/Y

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 action(s) required for the TDAFW system that must be taken and the reason?

A. Use the jacking device to open HV-3228A, B, and C, TDAFWP TO 1A, 1B, AND 1C SG FLOW CONT valves, to ensure an adequate heat sink.

B. ✓ Start the emergency air compressors and align the EACs to supply air to HV-3235A and HV-3235B, TDAFWP STM SUPP valves, to ensure an adequate heat sink.

C. Use the jacking device to close HV-3235A and HV-3235B, TDAFWP STM SUPP valves, in the main steam valve room (MSVR) to prevent an uncontrolled cooldown.

D. Start the emergency air compressors and align the EACs to supply air to HV-3228A, B, and C, TDAFWP TO 1A, 1B, AND 1C SG FLOW CONT valves, to prevent an uncontrolled cooldown.

A – Incorrect. The AFW FCV's fail open, and do not need to be jacked open to provide an adequate heat sink. Plausible, since they do need to be jacked closed or throttled when necessary during a loss of air. Also, confusion may exist between the steam admission valves which do fail closed (after 2 hours) and the FCVs which fail open.

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 vavles 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.

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. With only one AFW pump available and SG levels below 31%, at least 395 gpm TDAFW flow is required. Closing the steam admission valves would not be allowed in this case. This is plausible, since the SG levels are trending up and close to the levels at which AFW flow can be secured. Also, jacking devices can be used to close the valves such as isolating a faulted or ruptured SG, 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 emergency air compressors must be started by for the Steam Admission valves which have air accumulators keeping them open for at least 2 hours, but cannot be assumed to last for 4 hours. Plausible, since confusion may exist as to which of the the TDAFW valves are supplied by the emergency air compressors and which valves fail open and which valves have 2 hour rated air accumulators to keep them open with a loss of air. The UPS does supply the HV3228 valve solenoids for up to two hours, but the emergency air does not supply HV3228s.

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)

Step	Action/Expected Response	Response Not Obtained
__ 7	**Maintain PRZR level between 20-50%.**	
7.1	Alternately cycle open and closed one of the following MOVs for charging control as required. CHG PUMPS TO REGENERATIVE HX [ ] Q1E21MOV8107 [ ] Q1E21MOV8108	7.1 Go to FNP-1-AOP-16.0, CVCS MALFUNCTION while continuing with this procedure.
__ 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.

****

Step	Action/Expected Response	Response Not Obtained
8.1	WHEN TDAFW Pump is started, THEN vary TDAFW Pump Speed to control AFW flow. TDAFWP SPEED CONT [ ] SIC 3405 adjusted	8.1 IF the TDAFW Pump steam admission valves fail closed, THEN align emergency air using FNP-1-SOP-62.0, EMERGENCY AIR SYSTEM.

B →

Correct

__ Page Completed

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION: To provide adequate heat sink total AFW flow must remain greater than 395 gpm until at least one SG narrow range level is greater than 31%.

****

C incorrect

1.1.4 IF cooldown continues, THEN minimize total AFW flow.

AFW FLOW TO
1A(1B,1C) SG
[] FI 3229A
[] FI 3229B
[] FI 3229C

AFW
TOTAL FLOW
[] FI 3229

- Control MDAFWP flow.

MDAFWP FCV 3227
RESET
[] A TRN reset
[] B TRN reset

MDAFWP TO
1A/1B/1C SG
B TRN
[] FCV 3227 in MOD

SG	1A	1B	1C
MDAFWP TO 1A(1B,1C) SG Q1N23HV	[]3227A in MOD	[]3227B in MOD	[]3227C in MOD
MDAFWP TO 1A(1B,1C) SG FLOW CONT HIC	[]3227AA adjusted	[]3227BA adjusted	[]3227CA adjusted

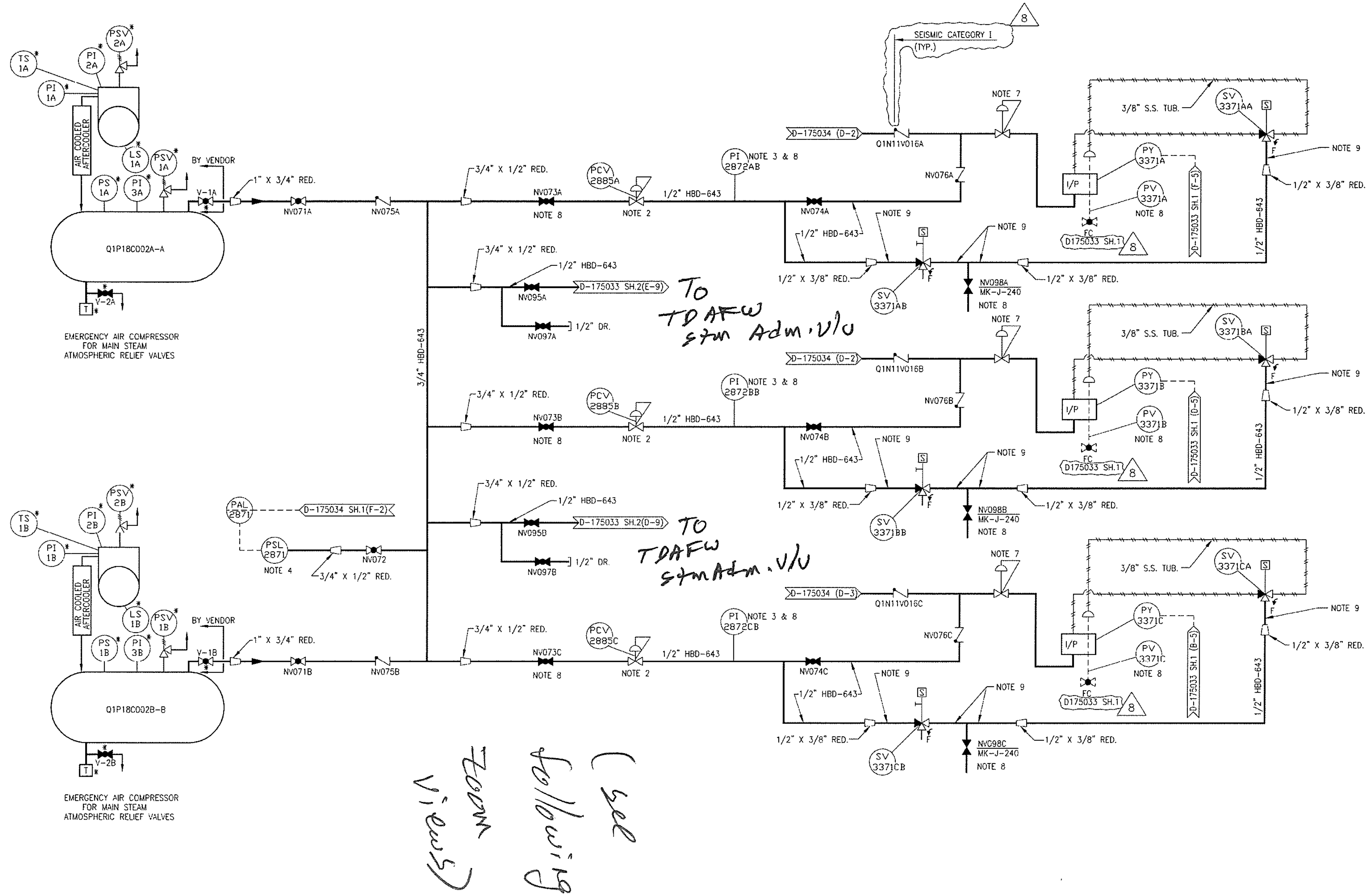
Step 1 continued on next page.

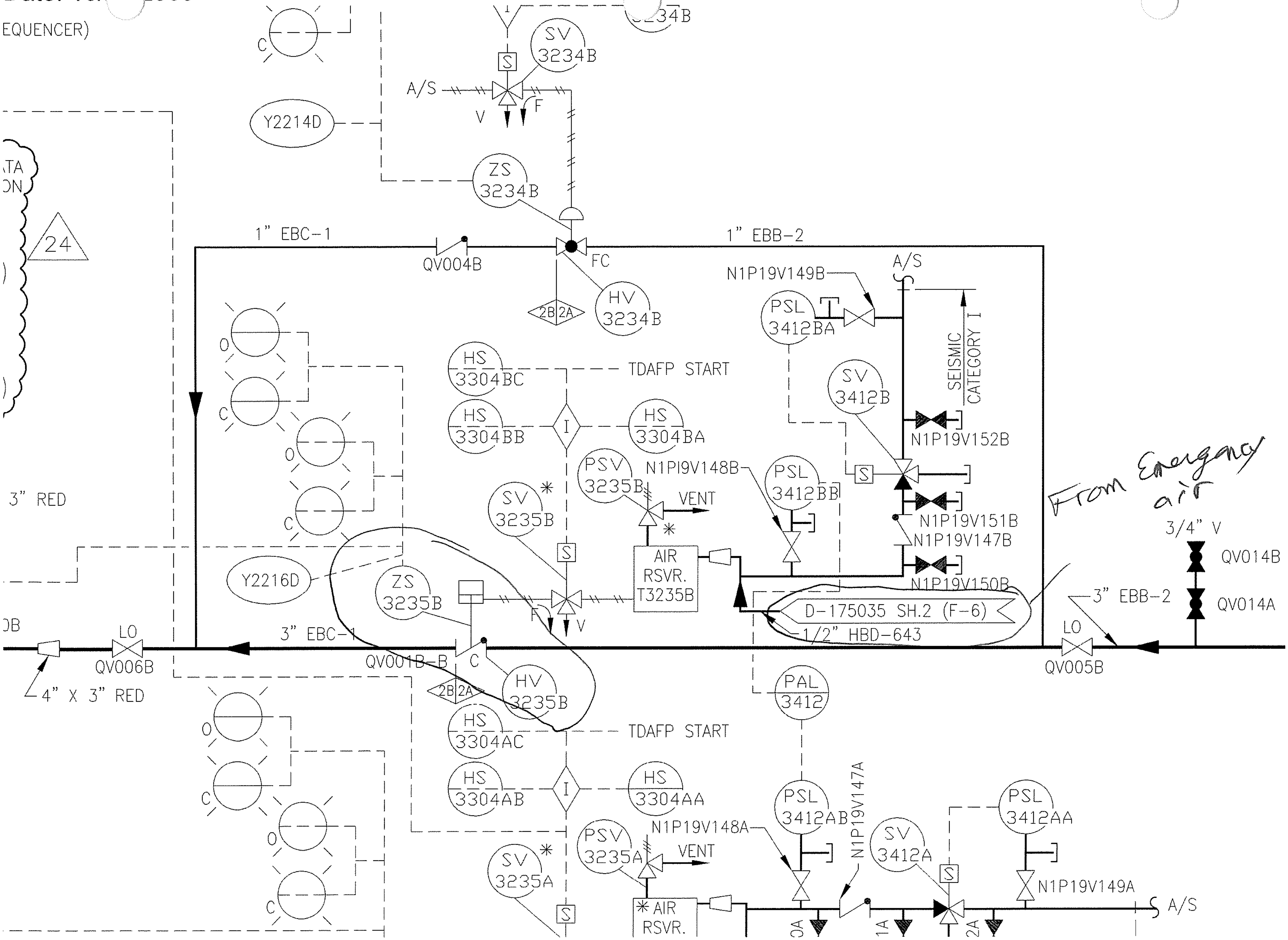
___Page Completed

TABLE 1

COMPONENT NUMBER	NAME	MANUAL OPERATOR	FAILED POSITON	OPERATOR DRAWING
N1N21V870	SGFP 1A INBOARD SEAL PRESS REG	NO	OPEN	
N1N21V901	SJAE BYP FCV	YES	OPEN	
N1N21V902	GS COND BYPASS FCV	YES	AS IS	
N1N21V908	CNDS MINIMUM FLOW FCV	YES	OPEN	
N1N21V909A	SFGP 1A RECIRC FCV	YES	OPEN	U-213892
N1N21V909B	SFGP 1B RECIRC FCV	YES	OPEN	U-161476
N1N21V916	CONDENSATE PUMPS BACK UP COOLING WATER PCV	YES	OPEN	
N1N22V725	SGFP SEAL DRAIN TANK AUTO DUMP TO THE CONDENSER	YES	CLOSED	
Q1N23FCV3227A (1-AFW-FCV-3227A)	MDAFW PUMP TO STM GEN 1A	YES	OPEN	U-176885
Q1N23FCV3227B (1-AFW-FCV-3227B)	MDAFW PUMP TO STM GEN 1B	YES	OPEN	U-176885
Q1N23FCV3227C (1-AFW-FCV-3227C)	MDAFW PUMP TO STM GEN 1C	YES	OPEN	U-176885
Q1N23FCV3228A (1-AFW-FCV-3228A)	TDAFW PUMP TO SG 1A	YES	OPEN	U-176884
Q1N23FCV3228B (1-AFW-FCV-3228B)	TDAFW PUMP TO SG 1B	YES	OPEN	U-176884
Q1N23FCV3228C (1-AFW-FCV-3228C)	TDAFW PUMP TO SG 1C	YES	OPEN	U-176884
Q1N25V001A (1-CI-HV-3772A)	CHEM ADD TO 1A SG ISO	YES	CLOSED	
Q1N25V001B (1-CI-HV-3772B)	CHEM ADD TO 1B SG ISO	YES	CLOSED	
Q1N25V001C (1-CI-HV-3772C)	CHEM ADD TO 1C SG ISO	YES	CLOSED	
N1N26V887A	HTR DRN PUMP 1A RECIRC	YES	OPEN	

A incorrect


SEISMIC CATEGORY I (TYP.)
EMERGENCY AIR COMPRESSOR FOR MAIN STEAM ATMOSPHERIC RELIEF VALVES
Q1P18C002A-A
Q1P18C002B-B
AIR COOLED AFTERCOOLER
BY VENDOR
1" X 3/4" RED.
NV071A
NV075A
NV071B
NV075B
3/4" HBD-643
3/4" X 1/2" RED.
PCV 2885A
PCV 2885B
PCV 2885C
NV073A
NV073B
NV073C
NOTE 8
NOTE 2
1/2" HBD-643
PI 2872AB
PI 2872BB
PI 2872CB
NOTE 3 & 8
NV074A
NV074B
NV074C
D-175034 (D-2)
D-175034 (D-3)
Q1N11V016A
Q1N11V016B
Q1N11V016C
NV076A
NV076B
NV076C
NOTE 7
NOTE 9
3/8" S.S. TUB.
SV 3371AA
SV 3371BA
SV 3371CA
PY 3371A
PY 3371B
PY 3371C
PV 3371A
PV 3371B
PV 3371C
I/P
FC
D175033 SH.1
D-175033 SH.1 (F-5)
D-175033 SH.1 (D-5)
D-175033 SH.1 (B-5)
1/2" X 3/8" RED.
SV 3371AB
SV 3371BB
SV 3371CB
NV098A MK-J-240
NV098B MK-J-240
NV098C MK-J-240
D-175033 SH.2(E-9)
D-175033 SH.2(D-9)
NV095A
NV095B
NV097A
NV097B
1/2" DR.
PAL 2871
PSL 2871
D-175034 SH.1(F-2)
NV072
NOTE 4
To TDAFW stm Adm. v/v
TO TDAFW stmAdm. v/v
(see following zoom views)



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makeup water from the demineralized water storage tank (References 6.4.049, 6.4.051, 6.4.063).

3.21.7.2 The CST shall interface with the condensate system so that condensate water may be used as a makeup water source (References 6.4.049, 6.4.051).

3.21.7.3 The recirculation line from the AFWS shall connect to the tank at 19 feet above the base of the tank. This location is above the portion of the tank dedicated to the AFW emergency supply (References 6.4.050, 6.5.008).

3.21.7.4 The freeze line protection for the CST piping and instrumentation lines shall be provided with nonsafety-related power from the 120-208 V ac, distribution cabinets 1CC and 2CC (References 6.4.064, 6.4.065).

3.22 TDAFW PUMP UPS SYSTEM

TPNS No. QN23L001-AB (UPS) and QN23E001-AB (Battery)

3.22.1 Basic Function

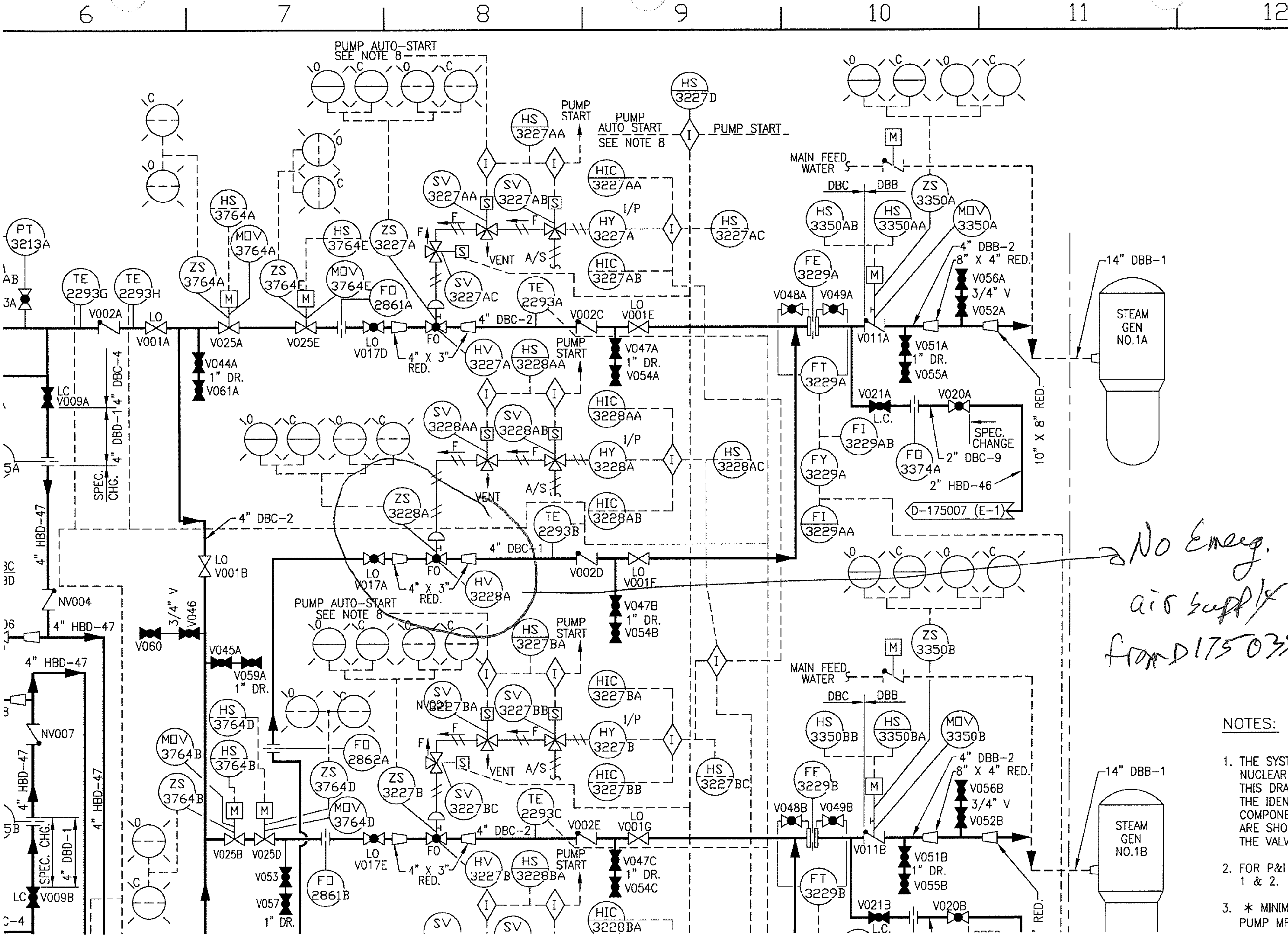
The UPS system shall be designed to provide an uninterruptible source of 120 V ac and 125 V dc power supply for control of the TDAFW pump turbine drive (QN23P003), steam admission valve (QN12HV3226), steam supply isolation valves (QN12HV3235A, B), instrument air valves (NN12SV3412A, B) and the TDAFW 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).

The UPS system is provided with additional component redundancy to enhance the reliability of the UPS system. This redundancy includes an alternate (backup) UPS system (Battery Charger, Inverter, and Rectifier) located inside the same cabinet (Reference 6.5.015).

3.22.2 Functional Requirements

3.22.2.1 The UPS shall normally be supplied from an emergency diesel generator-backed 208 V ac single phase 60 Hz source (References 6.4.002, 6.4.045).

3.22.2.2 In the event of inverter failure, the supply to all components shall automatically transfer within 50 msecs to a step-down transformer (within UPS) powered from the ac source (Reference 6.7.075).



No Emerg. air supply from D175035

47. 068K6.10 001/MOD/RO/C/A 2.5/2.9/N/N/3/CVR/Y

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.
- The control power fuse is found to be illuminated on R-18.

1) What effect, if any, does this condition have on RCV-18,

and

2) what action(s) is(are) required IAW SOP-50.1, Liquid Waste Processing System Liquid Waste Release From Waste Monitor Tank?

A. 1) RCV-18 will **NOT** automatically close.

2) Close RCV-18 with the Handswitch on the Liquid Waste Panel.

B.✓ 1) RCV-18 automatically closes.

2) Verify RCV-18 closed at the Handswitch on the Liquid Waste Panel.

C. 1) RCV-18 will **NOT** automatically close.

2) Close the manual discharge valve to the environment.

D. 1) RCV-18 automatically closes.

2) Override RCV-18 in the open position with the manual jacking device, implement ODCM actions for an inoperable R-18, and continue the release.

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. If this condition was present, the second part would be correct due to the inoperability of R-18 per SOP-50.1 Step 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."

B - Correct. Control power being lost causes a fail safe automatic function of closing RCV-18. The second part is correct per SOP-50.1 Step 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."

C - Incorrect. The first part is incorrect (see A). The second part is incorrect since the release must be stopped immediately per SOP-50.1 Step 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." Plausible, since after ODCM actions have been implemented, the release could be continued with R-18 inoperable, but not prior to them being implemented.

D - Incorrect. The first part is correct (see B). The second part is incorrect, since there is no manual operator on RCV-18. Plausible, since many AOVs have manual operators, and after the ODCM actions have been taken, the release may be continued with an inoperable R-18. The valve would be overridden open, but most likely the handswitch would be taken to open with jumpers installed to defeat the close signal.

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 required actions 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

RADIATION MONITOR REFERENCE TABLE (cont)

RE	LOCATION	TYPE	DETECTOR	FUNCTION	ACTIONS
R-12*	Containment Atmosphere (AB 121')	Gas	G-M (W)		Perform Step 4.12
R-13	Waste Gas Compressor Suction (AB 100' WGC Valve Room)	Gas	G-M (W)		Perform Step 4.13
R-14 ODCM	Plant Vent Stack (AB Roof)	Gas	G-M (W)	Closes HCV-14	Perform Step 4.14
R-15A ODCM	Condenser Air Ejector Discharge Header (TB 155')	Gas	G-M		Perform Step 4.15A
R-15B*	Condenser Air Ejector (Intermediate Range) (TB 189')	Gas	G-M (Eberline)		Perform Step 4.15B
R-15C*	Condenser Air Ejector (High Range) (TB 189')	Gas	Ion Chamber (Eberline)		Perform Step 4.15B
R-17A	Component Cooling Water (CCW Hx Room)	Liquid	Scint. (W)	Closes CCW surge tank vent (RCV-3028)	Perform Step 4.17
R-17B	Component Cooling Water (CCW Hx Room)	Liquid	Scint.	Closes CCW surge tank vent (RCV-3028)	Perform Step 4.17 D, C, & A
R-18 ODCM	Waste Monitor Tank Pump Discharge (AB 121' at the Batching Funnel)	Liquid	Scint. (W)	Closes RCV-18	Perform Step 4.18
R-19	Steam Generator Blowdown/Sample (AB 139')	Liquid	Scint. (W)	Isolates sample lines 3328, 3329, 3330	Perform Step 4.19
R-20A	Service Water from Containment Coolers A and B (AB 121' BTRS Chiller Room)	Liquid	Scint. (W)		Perform Step 4.20
R-20B	Service Water from Containment Coolers C and D (AB 121')	Liquid	Scint. (W)		Perform Step 4.20

2nd part incorre

B & [illegible] 1st part corre

A & C 1st part incorre

*Technical Specification related

LOCATION FH1

OPERATOR ACTION (cont) D, C, & A 2nd part incorrect

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.

4.19 IF **R-19** alarms AND remains above the alarm setpoint (not a momentary spike), THEN notify the Counting Room to immediately sample the SGs per FNP-0-CCP-31, LEAK RATE DETERMINATION, to determine the leak rate. Refer to FNP-1-SOP-45.0, RADIATION MONITORING SYSTEM for guidance in sampling steam generators with R-19 in alarm.

4.20 **R-20A** AND **R-20B** would not normally be expected to indicate high radioactivity since SW pressure is higher than the opposite side of the 'coolers' upstream of R-20A and R-20B. Request Counting Room to sample the SW effluent. IF high activity is confirmed, THEN investigate for a possible cross system connection to the SW system. **{CMT 0005153}**

4.21 IF **R-21** alarms, THEN implement FNP-0-EIP-9.0, EMERGENCY CLASSIFICATION AND ACTIONS. **{CMTs 0008751, 0008755}.**

4.22 IF **R-22** alarms, THEN implement FNP-0-EIP-9.0, EMERGENCY CLASSIFICATION AND ACTIONS. **{CMTs 0008751, 0008755}.**

4.23 IF **R-23A** OR **R-23B** alarms AND remains above the alarm setpoint (not a momentary spike), THEN perform the following:

4.23.1 Notify the Counting Room to immediately sample the SGs per FNP-0-CCP-31, LEAK RATE DETERMINATION, to determine the leak rate

4.23.2 Contact the RAD man to verify blowdown secured.

4.24 IF **R-26A** OR **R-26B** alarms, THEN refer to FNP-1-SOP-50, LIQUID WASTE PROCESSING SYSTEM for potential problems with the liquid waste system.

References: A-177100, Sh. 306; U-260841; D-181751; D-181752; D-181753; FSAR, Section 11.4

LOCATION FH2

H2
RMS CH FAILURE

SETPOINT: Not Applicable

ORIGIN: Any of the below listed Area, Process or Gaseous and Particulate Monitors: R01, R02, R03, R04, R05, R06, R07, R08, R10, R11, R12, R13, R14, R15, R17A R17B, R18, R19, R20A, R20B, R21, R22, R23A, R23B, R26A, or R26B.

PROBABLE CAUSE

1. Loss of input signal to any of the above listed Radiation Detection Channels.
2. Loss of Power to a Channel.
3. Radiation Monitoring System testing in progress.

AUTOMATIC ACTION

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.

OPERATOR ACTION

NOTE: Low Alarm Light "on" indicates failure.

1. Check indications on radiation monitoring system console and determine which radiation monitor channel indicates a failure.
2. Notify chemistry and health physics personnel.
3. Notify Instrument Service Personnel to:
   A. Investigate the failure.
   B. Make repairs as necessary.
4. Return the Radiation Monitor System Channel to service, in accordance with FNP-1-SOP-45.0, RADIATION MONITORING SYSTEM, as soon as possible.
5. Refer to the Technical Requirements Manual section on Radiation Monitoring Instrumentation.

References: A-177100, Sh. 307; U-260841; D-181751; D-181752; D-181753; FSAR, Section 11.4

Boll

1. LIQ SD WAST-52106A05 003/HLT//C/A 4.6/4.4/APE059AG2.4.49//YES//

068K6.10

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

- Annunciator FH2, RMS CH FAILURE, alarms.
- 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.
- The control power fuse is found to be illuminated on R-18.

When called, the Radside SO reports that Waste Monitor Tank Pump #2 discharge flow transmitter, FT-1085A, indicates 35 gpm.

Which one of the following describes the actions **required** IAW SOP-50.1, Liquid Waste Processing System Liquid Waste Release From Waste Monitor Tank?

Direct the Radside SO to ______________________

A. fail air to RCV-18, WMT Disch to Environment
OR
close the manual discharge valve to the environment.

B. fail air to RCV-18, WMT Disch to Environment
OR
using the manual handwheel, close RCV-18, WMT Disch to Environment.

C✓ close WMT Disch To Environment, RCV-18 at the LWP
OR
close the manual discharge valve to the environment.

D. close WMT Disch To Environment, RCV-18 at the LWP
OR
using the manual handwheel, close RCV-18, WMT Disch to Environment.

APE059G2.4.49
059 Accidental Liquid Radwaste Release
2.4 Emergency Procedures / Plan
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) IMPORTANCE RO 4.6 SRO 4.4

A. Incorrect – Plausible since failing air to RCV-18 could stop the release if the problem was other than mechanical binding, but this is not the procedurally correct way to secure the release IAW SOP-50.1. RCV-18 fails closed on a loss of air. Also, if RCV 18 did not close on the auto close signal, it may be mechanically bound and not shut when air is failed. The second part is correct.

B. Incorrect – First part is incorrect (see A). Second part is incorrect due to not being in the procedure and the valve does not have a manual handwheel. Plausible since many AOVs have manual handwheels and would be used to close their respective valve if necessary.

C. Correct – SOP-50.1 states that if the discharge is in progress and R-18 becomes inoperable the discharge is to be immediately stopped and the Shift Support Supervisor notified. The manual discharge valve will stop the release since RCV-18 did not close and is the procedurally correct method to secure the release at the end of SOP-50.1. RCV-18 Handswitch at the Liquid Waste Panel (LWP) is also directed to be taken to the closed position in SOP-50.1 and may also isolate the release.

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

2008 NRC exam

Technical Reference:SOP-50.1 Ver. 60.0, AOP-6 Ver. 31, LIQUID WASTE PERMIT, ARP-1.6 Ver. 58, FH1 AND FH2

Comments: This question tests the Immediate Actions of a procedure that is used by both the systems operator and CRO to perform a release. Since this deals with the INOPERABLE side of R-18 from the Control room and the required actions should this occur at an RO level for a liquid release, it meets the KA.

48. 069AA2.01 001/FNP BANK/RO/MEM 3.7/4.3/Y 2007/N/2/CVR/Y

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

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Structural integrity of the containment not conforming to the requirements of SR 3.6.1.2.	A.1 Restore the structural integrity to within limits.	24 hours
B. Containment inoperable for reasons other than Condition A.	B.1 Restore containment to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

B Correct.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program.
SR 3.6.1.2	Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.	In accordance with the Containment Tendon Surveillance Program

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

A incorrect (wrong TS)

LCO 3.6.2 Two containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTES---

1. Entry and exit is permissible to perform repairs on the affected air lock components.
2. Separate Condition entry is allowed for each air lock.
3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	---NOTES--- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. ---	(continued)

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

C incorrect for Mode 5

LCO 3.6.3 Each containment isolation valve shall be OPERABLE. The 8-inch containment mini-purge supply and exhaust isolation valves may be open for safety-related reasons.

APPLICABILITY: MODES 1, 2, 3, and 4. ~~5~~

ACTIONS

-----NOTES-----

1. Penetration flow path(s) except for 48-inch purge valve flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

▷ incorrect

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed and held in place by four bolts;

b. One door in each air lock is capable of being closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

49. 071A4.01 001/NEW/RO/MEM 2.7/2.2/N/N/3/GTO/CVR/Y

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 **ONLY** 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 **ONLY** 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).

temperature, range, orifice I.D and readout location. (References 6.2.9, 6.5.29.v, 6.5.29.x, 6.5.29.aa and 6.5.73)

3.13.3 Equipment Qualification Requirements

3.13.3.1 The electronic transmitters shall be environmentally qualified as detailed in references 6.7.26, 6.7.27, 6.7.29.h and 6.7.29.i. Maintenance specified in the EQ package shall be performed to maintain qualified life. (References 6.7.23, 6.7.26, 6.7.27, 6.7.29.h, 6.7.29.i and 6.7.28)

3.13.4 Interface Requirements

3.13.4.1 The channel must be powered from a 120 Volt-AC regulated instrument power system. This regulated AC power shall be 118 volts $\pm$ 2%, 60 cps $\pm$ 2% nominal, 3% maximum harmonic distortion (normal), 5% maximum. (Reference 6.5.21)

3.13.5 Equipment Protection Features

3.13.5.1 The differential pressure device shall be capable of withstanding a sustained input pressure of 1.5 times system design pressure. This applied pressure shall not damage the instrument so that it can perform its function. (Reference 6.5.21)

3.14 VCT LEVEL

LICA-112
LIC-115

3.14.1 Basic Functions

3.14.1.1 These level instruments shall provide measurement of VCT level and shall supply signals for control and actuation of valves LCV-115A, B, C, D and E.

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.

The instruments shall alert the control room operator of high- and low-level conditions in the VCT and when the Reactor Makeup Control System controls are not positioned properly to support auto-makeup to the VCT. (References 6.2.1, 6.2.56, 6.2.57, 6.2.9 and 6.4.1)

3.14.1.2 Control Room indication of VCT level is required by RG 1.97 for post accident monitoring. (References 6.7.24, 6.7.25 and 6.7.23)

3.14.2 Functional Requirements

3.14.2.1 Level indication shall be provided on the main control board for LI-115. This provides the operators with indication for normal operations and is also used to satisfy post accident monitoring (RG 1.97) requirements. The signal from LI-112 shall be indicated locally. (References 6.2.1, 6.2.9, 6.4.l, 6.7.24, 6.7.25 and 6.7.23)

3.14.2.2 The instruments shall support the following level setpoint requirements. (Refer to References 6.2.7 and 6.2.8 for the specific alarm and control setpoint):

<u>Emergency Makeup Start RWST</u> -- Upon coincident (2/2 logic) signals from both level instruments, valves LCV-115B and D (QV336A/B) from the RWST shall open, and valves LCV-115C and E (QV376A/B) from the VCT shall close to maintain suction to the charging pumps. Should one of these signals fail to actuate, the operator still has at least 2.5 - minutes time period before the potential suction loss to the charging pumps. The 2.5-minute time period assumes 120-gpm letdown being diverted to the holdup tank and no makeup supplied by the Reactor Makeup Control System. (References 6.2.1, 6.2.9, 6.2.58, 6.3.12, 6.3.13 and 6.2.56)

<u>Emergency Makeup Stop RWST</u> -- This setpoint provides for the clear signal of the emergency makeup start and allows the operator to reposition the RWST and VCT isolation valves to their normal position. (References 6.3.12, 6.3.13 and 6.2.58)

<u>Low Level Alarm (LT-112 and LT-115)</u> -- A visual and audible alarm shall be provided on the main control board, warning the operator of a low-level condition. This setpoint

is below the setpoint for automatic initiation of makeup to the VCT. This warns the operator that the automatic makeup function has failed to actuate or that the system is not causing the VCT level to rise. (References 6.2.1, 6.2.9, 6.2.58, 6.3.12 and 6.3.13)

Auto Makeup Start (LT-115) -- This setpoint shall start auto makeup to the VCT from the Reactor Makeup Control System and activate and alarm if the Reactor Makeup Control System is not set in its auto makeup mode. This setpoint should be activated when the VCT level is approaching the bottom of the tank but above the three level setpoints discussed above. The setpoint was arbitrarily set 3-inches above the low level alarm. (References 6.2.1, 6.2.9, 6.2.58, 6.3.12 and 6.3.13)

NOT Stated in FSD, But verified on simulator:

White light indicates an Auto open signal (modulate OR full divert), and is to the right of the handswitch light array in the position of the green light of other valves.

The red light indicates the the valve is to the VCT position

Red/White indicates differently than other valves (there is no green):

Red = at least partially to the VCT position
White = at least partially to the RHT position

Auto Makeup Stop LT-115) -- This setpoint shall stop auto makeup to the VCT from the Reactor Makeup Control System. This setpoint is set at 20% above the auto makeup start setpoint since this is the maximum available deadband of LB-115C. (References 6.2.1, 6.2.9, 6.2.58, 6.2.59 and 6.2.60)

B & A 1st parts correct (40% level per stem of question)

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)

High Level Alarm (LT-115) -- A visual and audible alarm shall be provided on the main control board, warning the operator of a high-level condition. This setpoint is above the setpoint for modulate divert of makeup to the VCT. This warns the operator that the modulate divert function has failed to actuate or that the system is not causing the VCT level to drop. The high alarm also warns the operator that the level is approaching the maximum level recommended for degassing operations. (References 6.2.1, 6.2.9, 6.2.58, 6.3.12 and 6.3.13)

Fully Divert (LT-115) -- This setpoint shall cause valve LCV-115A to fully divert letdown to the RHTs. The full divert function is provided as a back-up in the event that the modulate divert function (performed by LT-112) should fail. The primary restricting consideration here is to prevent pressurizing the VCT over its maximum normal operating pressure of 65 psig and risk lifting the safety valve. (References 6.2.1, 6.2.9, 6.2.58, 6.3.12, and 6.3.13)

3.14.2.3 Refer to Table T-11 for instrumentation design requirements including instrumentation type, design pressure, temperature, range and readout location. (References 6.2.9 and 6.5.29.e)

3.14.3 Equipment Qualification Requirements

3.14.3.1 The electronic transmitter (LT-115) shall be environmentally qualified as detailed in references 6.7.26, 6.7.27, 6.7.29.h and 6.7.29.i. Maintenance specified in the EQ package shall be performed to maintain qualified life. (References 6.7.23, 6.7.26, 6.7.27, 6.7.29.h, 6.7.29.i and 6.7.28)

3.14.4 Interface Requirements

3.14.4.1 The channel must be powered from a 120 Volt-AC regulated instrument power system. This regulated AC power shall be 118 volts ± 2%, 60 cps ± 2% nominal, 3% maximum harmonic distortion (normal), 5% maximum. (Reference 6.5.21)

3.14.5 Equipment Protection Features

3.14.5.1 The differential pressure device shall be capable of withstanding a sustained input pressure of 1.5 times system design pressure. This applied pressure shall not damage the instrument so that it can perform its function. (Reference 6.5.21)

3.15 ACCUMULATOR TANK PRESSURE

PIA-921 (Tank 1)	PIA-923 (Tank 1)
PIA-925 (Tank 2)	PIA-927 (Tank 2)
PIA-929 (Tank 3)	PIA-931 (Tank 3)

3.15.1 Basic Functions

3.15.1.1 These pressure instruments shall be utilized by the operator to help set and maintain the gas overpressure in the accumulators within the Technical Specification limits

FARLEY NUCLEAR PLANT
UNIT 1
SYSTEM OPERATING PROCEDURE SOP-2.4

CHEMICAL AND VOLUME CONTROL SYSTEM
BORON RECYCLE SYSTEM

1.0 Purpose

This procedure provides Initial Conditions, Precautions and Limitations, and Instructions necessary for the operation of the Boron Recycle System. Instructions are included in the following sections:

4.1 #1 (#2, #3) RHT operation.

4.1.1 Swapping In Service RHT's.

4.2 Recycle evaporator feed pump operation.

4.3 Placing RHT on Big Recirc or Transferring between RHTs via the recycle evaporator feed demineralizers and filters.

4.4 Placing RHT on Big Recirc or Transferring between RHTs with the recycle evaporator feed demineralizers and filters <u>BYPASSED</u>.

4.5 #1 (#2, #3) RHT discharge to charging pump suctions.

4.6 #1 (#2, #3) RHT discharge to SFPCS transfer canal.

4.7 #1 (#2, #3) RHT discharge to waste evaporator. (Deleted by Version 41.0) step 4.8.5

4.8 #1 (#2, #3) RHT venting with WASTE GAS SYSTEM. → PCV-25/ opened + will auto close @ setpoint

4.9 #1 (#2 #3) RHT venting with RHT GAS SAMPLE PANEL. → PCV-25/ NOT opened

4.10 Draining RHTs to WHT or FDT.

4.11 Transferring between RHTs using recycle evaporator feed pump miniflow line.

4.12 #1 (#2, #3) RHT pressure check for gas buildup under bladder using temporary gage or installed instrumentation.

Appendix 1 Transfer of Unit-1 RHTs to Unit-1 RWST

Appendix 2 Transfer of Unit 1 RHTs to Unit 2

4.8 #1(#2, #3) RHT venting with WASTE GAS SYSTEM. **(CMT 0005260)**

NOTE:	**Indicate completion of applicable (*) steps by initialing on procedure sign-off list FNP-1-SOP-2.4B.**

*4.8.1 Transfer the water from the RHT which is to be vented per section 4.4 until level is between 50% and 5%. Level may be left higher than 50% with Shift Supervisor's permission.

4.8.1.1 IF steps 4.8.1.2 and 4.8.1.3 are required by the Shift Supervisor, THEN notify Maintenance to remove manway on #1 (#2, #3) RHT.

4.8.1.2 Have Health Physics check above the bladder for combustible gas concentration to detect bladder leakage.

4.8.1.3 Operations personnel determine the size of the gas bubble.

NOTE:	**The waste gas system must be aligned to the low pressure mode for operations with the eductor.**

*4.8.2 Align off service waste gas compressor to eductor as follows:

4.8.2.1 Notify Health Physics Foreman prior to educting RHTs due to dose rate changes on eduction piping.

4.8.2.2 Verify RMW and CCW aligned to off service compressor.

4.8.2.3 Verify open compressor suction valve 1-GWD-V-7907A(B) (Q1G22V06lA [B]).

4.8.2.4 Close compressor discharge valve to recombiners 1-GWD-V-7910A (B) (Q1G22V064A [B]).

4.8.2.5 Open compressor discharge to eductor 1-GWD-V-7911A(B) (Q1G22Vl97A [B]).

4.8.2.6 Open eductor return isolation 1-GWD-V-7807 (Q1G22V008).

4.8.2.7 Start 1A (1B) waste gas compressor.

(Step 4.8 continued on next page)

CAUTION:	**In event of a significant diaphragm leak, reactor coolant system draining, or spent fuel pit draining, align the waste gas compressor to a shutdown tank per FNP-1-SOP-51.0, WASTE GAS SYSTEM, instead of the normal path to a gas decay tank.**

*4.8.3 Close or verify closed the following valves:

- #1 (#2, #3) RHT MAIN INLET ISO 1-CVC-V-8554A (B,C) (Q1E21V284A [B,C]).
- #1(#2, #3) RHT MINIFLOW ISO valve 1-CVC-V-8556A (B,C) (Q1E21V309A[B,C]).
- #1RHT EDUCTOR SUCT Q1E21V314A.
- #2 RHT EDUCTOR SUCT Q1E21V314B.
- #3 RHT EDUCTOR SUCT Q1E21V314C.

*4.8.4 Verify locked closed RHT equipment drains and valve leakoffs inlet line isolation 1-CVC-V-8557A(B,C) (Q1E21V311A[B,C]). (Master Z Key)

*4.8.5 Open or verify open the following valves:

- #1(#2, #3) RHT EDUCTOR SUCT Q1E21V314A(B,C) for the RHT to be vented.
- RHT eductor suction line pressure control 1-CVC-PCV-251 (Q1E21V366).

B&D second parts incorrect

- RHT EDUCTOR SUCT ISO valve Q1E21V319.
- RHT eductor suct sample bypass 1-CVC-V-8638 (Q1E21V321).

(Step 4.8 continued on next page)

NOTE:	**Contact HP when venting RHTs to take periodic air samples of the AUX building due to the potential of creating airborne areas when venting hot RHT.**

4.9 #1 (#2, #3) RHT venting with RHT GAS SAMPLE PANEL.

NOTE:	**Indicate completion of applicable (*) steps by initialing on procedure sign-off list FNP-1-SOP-2.4B.**

*4.9.1 Transfer the water from the RHT which is to be vented per section 4.4 until level is between 50% and 5%. Level may be left higher than 50% with Shift Supervisor's permission.

4.9.1.1 IF steps 4.9.1.2 and 4.9.1.3 are required by the Shift Supervisor, THEN have Maintenance remove the manway on #1 (#2, #3) RHT.

*4.9.1.2 Have Health Physics check above the bladder for combustible gas concentration to detect bladder leakage.

4.9.1.3 Operations personnel determine the size of the gas bubble.

4.9.2 Collect the required equipment.

4.9.2.1 Vacuum pump with flow meter suitable for monitoring flow in the range of 0-3 cfm (85 liters/min). (Health Physics air sampler)

4.9.2.2 50' electrical extension cord with 3 prong plug or adapter.

4.9.2.3 50' tygon hose.

4.9.2.4 Suction hose, approximately 10' poly flow or thick walled tygon hose.

*4.9.3 Verify closed #1, #2, and #3 RHT sample line isolation valves (Aux Bldg, 121' in hallway outside RHT room inside SAMPLE PANEL):

- N1E21V324A
- N1E21V324B
- N1E21V324C

*4.9.4 Verify closed RHT SAMPLE LINE ISO valve N1E21V325B. (Aux Bldg, 121' in hallway outside RHT room inside SAMPLE PANEL)

*4.9.5 Connect vacuum pump suction hose to RHT SAMPLE LINE ISO valve N1E21V325B. (Aux Bldg, 121' in hallway outside RHT room inside SAMPLE PANEL)

(Step 4.9 continued on next page)

NOTE: After hose is routed to the HVAC exhaust duct, it should be blue tagged and left hanging for future venting. Ensure hose is securely routed.

*4.9.6 Connect a hose to the discharge of the vacuum pump and route to a HVAC exhaust duct.

*4.9.7 Verify hose is placed to discharge gases directly into exhaust duct.

*4.9.8 Verify the radwaste ventilation system is in operation.

*4.9.9 Check hose connections and verify all connections are suitable to prevent leakage of H_2 gas.

*4.9.10 Slowly open #1 (#2, #3) RHT SAMPLE LINE ISO valve for the RHT to be vented (Aux Bldg, 121' in hallway outside RHT room inside SAMPLE PANEL):

- N1E21V324A
- N1E21V324B
- N1E21V324C

*4.9.11 Slowly open RHT SAMPLE LINE ISO valve N1E21V325B. (Aux Bldg, 121' in hallway outside RHT room inside SAMPLE PANEL)

*4.9.12 Start vacuum pump.

*4.9.13 Adjust flow rate to maintain less than 2.0 cfm (55 liters/min).

(Step 4.9 continued on next page)

CAUTION: **Eduction should not continue past 20" H_2O vacuum to prevent loss of overflow water seal.**

NOTES: **The intent of the following step is to lock in vacuum under the bladder prior to securing the vacuum pump to prevent reintroduction of oxygen. Boron Recycle Panel annunciator #23 (RHT EDUCTOR LO PRESS) comes in at 7" H_2O vacuum and should be used as a prompt that eduction is almost complete. If required, eduction may be secured prior to reaching the desired vacuum.**

AdC 2nd part to Correct.

Indicated RHT level may rise significantly due to a slight vacuum being drawn, even if no vacuum is indicated on PIS-251. A SMALL addition of N_2 may be necessary to return indicated level to normal. Monitor RHT level during N_2 addition and secure N_2 addition when indicated level returns to normal.

*4.9.14 WHEN a vacuum is indicated under the bladder OR flow rate drops to zero, THEN perform the following:

4.9.14.1 Close #1(#2, #3) RHT SAMPLE LINE ISO valve for the RHT that was vented (Aux. Bldg. 121' in hallway outside RHT room inside panel):

- N1E21V324A
- N1E21V324B
- N1E21V324C

4.9.14.2 Secure the vacuum pump.

*4.9.15 Close N1E21V325B RHT SAMPLE LINE ISO valve. (Aux Bldg, 121' in hallway outside RHT room inside SAMPLE PANEL)

*4.9.16 Disconnect vacuum pump, leaving discharge hose to plant ventilation exhaust duct in place and properly secured and blue tagged.

*4.9.17 Ensure ends of tygon vent hose are properly covered, taped, and secured after use for contamination control, and radiological protection.

(Step 4.9 continued on next page)

*4.9.18 IF vacuum is indicated OR it is otherwise desired to add N_2, THEN notify Chemistry to introduce N_2 under the RHT bladder per FNP-1-CCP-667. Unless further eductions are planned, N_2 should only be added to the point of breaking vacuum and should not cause a positive pressure under the bladder.

NOTE: If vacuum is indicated, do not continue until the vacuum has been broken by addition of N_2 under the bladder.

*4.9.19 Ensure all equipment is properly stored.

4.9.20 Notify the Health Physics Foreman that the RHT venting process is complete.

*4.9.21 IF required, THEN have Mechanical Maintenance reinstall manway on #1 (#2, #3) RHT.

4.9.22 WHEN RHT eduction is complete, THEN ensure an Autolog entry is made for documentation.

50. 072G2.1.27 001/NEW/RO/MEM 3.9/4.0/N/N/2/CVR/Y

Which one of the following states **only** correct purposes and/or functions of the Area Radiation Monitoring System (ARMS)?

A✓ • 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.

B. • Provide warning of any radiation hazard that could develop.

• 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 advance warning of a plant malfunction that could lead to a health hazard or plant damage.

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.

A - Correct. Per FSD: Radiation Monitoring System, A-181015. The FSD gives the function for 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.

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, but 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)

Read pgs 3-1 through pg 3-18 to verify correct answer: No ARMs shifts ventilation to recirc even though the R-60's do secure ventilation

Plausibility for B & D 2nd part incorrect

ALSO No ARMS secured an effluent release

iso's liquid releases C & D 1st parts incorrect

B&D 2nd ports incorrect, shifts ventilation to recirc

The B Train power supply for the RMS system panel Q2H25NGR2504I is 120 VAC distribution panel 2K, breaker number 5 (References 6.4.353 and 6.4.354).

3.1.11.7 Shielding Design

The detectors shall be located at widely separated locations (References 6.4.020, 6.7.005, and 6.7.034). The NUREG 0737 guidance for the location of these detectors states that they should be located to view a large segment of the containment atmosphere. The monitors are calibrated to respond to noble gas energies. The noble gases would be displaced to the upper regions of the containment and the monitors should be situated to obtain the best view of those areas, however, they should not be located such that maintenance would be difficult (i.e., not in the containment dome). Additionally, they should be placed such that radioactive shine from the reactor is minimized. The locations selected for these monitors are on the containment wall approximately 5 feet above floor elevation 155'-0". These locations provide functional shielding by placing the detectors below the 'shine' horizon from the reactor cavity (References 6.4.020 and 6.4.039).

3.1.12 Low Level Radwaste Building Area Monitor

TPNS No.

NSD21RE 0066A through F

3.1.12.1 Basic Function

These detectors monitor the low level radwaste buildings to alert operators to an increase in radiation levels as required by GDC 63 (Reference 6.7.084).

3.1.12.2 Functional Requirements

The monitors shall provide continuous indication over a range of 0.1 to 10^4 millirads per hour (mR/hr). (Reference 6.4.135).

3.1.12.3 I&C Requirements

3.1.12.3.1 The monitors shall provide a flat (±15 percent) response for gamma energies between 40 keV and 1.25 MeV (Reference 6.7.052).

3.1.12.3.2 The monitor ratemeter shall provide a minimum of two single pole double throw alarm relays for external use. Each relay shall be fully adjustable over the entire indicated range of the monitor (Reference 6.7.052).

B & P 2nd Parts incorrect

3.1.12.3.3 Upon detection of gross activity in excess of the trip setpoint, the monitors shall stop the ventilation system fans (Reference 6.4.129).

3.1.12.3.4 The setpoint for this monitor is based on being high enough to prevent spurious alarms, but low enough to alert the personnel in the low level radioactive storage building to an increase in radiation levels (Reference 6.7.080).

3.1.12.4 Interface Requirements

The power supply for the Radiation Level Indicator Panel NSD21G502 is 120/208 VAC distribution cabinet 1TT, breaker number 14 (Reference 6.4.130).

3.2 PROCESS LIQUID MONITORS

3.2.1 Recycle Evaporator Condensate Discharge

TPNS No. ND11RE0016

Unit 1 detector ND11RE 0016 has been spared and is abandoned in place. Control and alarm functions of the detector have been disconnected and disabled. The detector remains physically installed in the discharge line from the recycle evaporator.

3.2.1.1 Basic Function

This detector monitors the effluent from the recycle evaporator and directs it to either the Reactor Makeup Water (RMW) System or the Recycle Holdup tanks to minimize contamination of the RMW system (References 6.4.358 and 6.7.080).

3.2.1.2 Functional Requirements

3.2.1.2.1 An in-line liquid monitor shall be provided to directly monitor the process medium. The use of this type of monitor provides the fastest response time and easiest decontamination (References 6.4.358 and 6.7.080).

3.2.2.5.2 The A train DC control power supply for the main control board panel NH11NGMCB 2500A-AB is 125 VDC distribution panel 1A, breaker number 15 (Reference 6.4.105). The B train DC control power supply for the main control board panel NH11NGMCB 2500A-AB is 125 VDC distribution panel 1D, breaker number 12 (Reference 6.4.115).

3.2.3 Waste Monitor Tank Discharge Radiation Monitor

TPNS No.

ND11RE 0018

C&P 1st part incorrect

3.2.3.1 Basic Function

This detector monitors the radioactive waste processing system common liquid effluent line while waste is being discharged to the environment and initiates automatic isolation of the discharge if setpoints are exceeded to comply with GDC 60 and GDC 64 (References 6.4.038 and 6.7.084).

3.2.3.2 Functional Requirements

3.2.3.2.1 An in-line liquid monitor shall be provided to directly monitor the process medium. The use of this type of monitor provides the fastest response time and easiest decontamination (References 6.4.038, and 6.7.080).

3.2.3.2.2 This monitor shall alarm and isolate the effluent discharge prior to exceeding the limits of ten times the concentrations stated in 10 CFR Part 20, Appendix B, Table 2, Column 2 for radionuclides other than dissolved or entrained noble gases. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in unrestricted areas will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 50, for a member of the public, and (2) the limits of 10 CFR 20.1301 for the population. The limiting concentration of dissolved and entrained noble gases is 1×10^{-4} μCi/ml (References 6.7.078 and 6.7.081).

Setpoints are based on ensuring the discharge limits presented in Section 2.1.2 of the ODCM are not exceeded.

inputting to a recorder (References 6.4.109, 6.4.318, 6.7.079, and Open Item Observation RMS-FSD-014).

3.3.15.3.3 The monitor ratemeter shall provide a minimum of two single pole double throw alarm relays for external use. Each relay shall be fully adjustable over the entire indicated range of the monitor. One relay shall actuate on a high signal to actuate the local annunciator. In addition, each function shall actuate an indicating light on the monitor front panel defining the alarm condition. A power on/off switch shall be provided for the ratemeter (Reference 6.4.318).

3.3.15.3.4 The sampler control panel is mounted in the monitor skid and shall provide switches for detector power on/off, pump stop/start, and an alarm light indicating low flow conditions in the monitor. A local control panel shall be provided at the monitor skid that contains switches for pump operation and indicating lights providing alarm and fail status (Reference 6.4.318).

3.3.15.3.5 The setpoints for this monitor are calculated to ensure that the limits specified by 10 CFR 20, Appendix B, Note 2, Table 1, Column 3 are not exceeded (References 6.7.062 and 6.7.083).

3.3.15.4 Interface Requirements

The 120 VAC power supply for RMS panel NSD11RE 0034 is 208/120 VAC control power panel T, breaker number 12 (Reference 6.4.108). The 208 VAC pump power supply for RMS panel NSD11RE 0034 is 208/120 VAC control power panel 1DD, breaker number 14 (Reference 6.4.108).

3.3.16 Control Room Ventilation Inlet Noble Gas Monitor

TPNS Nos.

QSD11RE 0035A, B

3.3.16.1 Basic Function

These detectors provide redundant, safety related, monitoring of the outside air entering the control room through the computer room air intake and to initiate automatic isolation of the control room ventilation

B+2 2nd Parts incorrect

system dampers (HV-3622, 3624, 3626, and 3628 for train A and HV-3623, 3625, 3627, and 3629 for train B) if setpoints are exceeded. This air inlet is common to the main control room and is isolated to comply with the criteria in GDC 19. In addition, the TSC ventilation system is automatically shifted to recirculation mode (References 6.4.377, 6.4.378, 6.4.379, 6.4.380, 6.4.385, 6.7.001, and 6.7.084).

For a complete description of the operation of the Control Room Ventilation Systems, see the Control Room Ventilation System FSD, Drawing No. A-181006. For a complete description of the operation of the Technical Support Center Ventilation Systems, see the Auxiliary Building Ventilation System FSD, Drawing No. A-181016.

3.3.16.2 Functional Requirements

3.3.16.2.1 The monitor shall consist of an off-line gas monitor located in an area of low background radiation (References 6.4.290, 6.4.349, and 6.7.080).

3.3.16.2.2 The monitor shall provide a pumping system with a sample flow rate of 8.5 scfm. The sample shall pass through a pre-filter to remove particulates larger than 5 microns prior to entering the volume chamber for counting (Reference 6.4.317). The flow rate of 8.5 scfm is the manufacturer's standard design. In the present configuration, this flow rate provides a monitor response and control room intake isolation time of approximately 7.4 seconds. The fuel handling accident analysis inside containment credits RE-035A and B for isolation of the control room due to high radiation within 60 seconds following accident initiation (References 6.3.004 and 6.3.006).

3.3.16.2.3 This monitor's function is to detect radioactive gases, however, certain elements of isokinetic design have been maintained in the transport system design. The tubing leading up to the monitor inlet contains radius bends in excess of five times the tube diameter. As described in FSD Section 4.3, this is to minimize plateout of particulates in the sample lines. A sample probe is included on the inlet tube at the duct interface. The probe is not an isokinetic design. The probe provides a beveled side which is placed to face into the upstream air flow. This prevents a sheer across the face of the probe tip which could create a vacuum and decrease the sample flow rate. The system is operated in an anisokinetic mode with sample velocity

51. 073K3.01 001/NEW/RO/C/A 3.6/4.2/N/N/2/HBF/Y

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.

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 $\triangle$P 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 [...]

Question #	51
K/A	073K3.01

REFERENCE Docs

2.0 Initial Conditions

2.1 The electrical distribution system is energized and aligned for normal operation per FNP-1-SOP-36.0, PLANT ELECTRICAL DISTRIBUTION LINE-UP, with exceptions noted.

2.2 120V AC electrical distribution system is energized and aligned for operation per FNP-1-SOP-36.4, 120V A.C. DISTRIBUTION SYSTEMS.

2.3 Power fuses are installed in all radiation monitoring system instrument drawers.

2.4 The radiation monitoring system is aligned per System Checklist FNP-1-SOP-45.0A.

2.5 Containment radiation monitors R-11 and R-12 are aligned for operation per FNP-1-SOP-12.2, CONTAINMENT PURGE AND PRE-ACCESS FILTRATION SYSTEM (applies only to R-11 and R-12).

2.6 Have I&C verify proper monitor alignment per FNP-1-IMP-227.47, VALVE ALIGNMENT FOR OPERATION OF UNIT 1 RE0011/12 AND RE0021/22.

3.0 Precautions and Limitations

3.1 Due to slow filter paper speed of the APD a five hour time period is required for the detector indication to reach equilibrium value after changing filter paper or filter paper speed.

3.2 Alarms on Radiation Monitoring System Panel must be acknowledged to provide main control board annunciator reflash capability.

3.3 A common annunciator on the main control board is actuated on high radiation from any channel. Individual drawers shall be checked to determine the alarming channel(s).

3.4 A common annunciator on the main control board is actuated when any channel is in the test mode.

3.5 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. Prior to removing a channel from service, ensure any automatic functions are either disabled or acceptable with respect to the affect (including reportability consideration) on associated system operation. (Refer to Tables A, B and C for monitors with automatic functions.)

RADIATION MONITOR REFERENCE TABLE (cont)

RE	LOCATION	TYPE	DETECTOR	FUNCTION	ACTIONS
R-21	Plant Vent Stack (AB 155')	APD	Scint. (Victoreen)		Perform Step 4.21
R-22 ODCM	Plant Vent Stack (AB 155')	Gas	G-M (W)		Perform Step 4.22
R-23A	SG Blowdown Surge Tank Inlet (AB 130')	Liquid	Scint. (W)	Closes FCV-1152	Perform Step 4.23
R-23B ODCM	SG Blowdown Surge Tank Discharge (AB 130')	Liquid	Scint. (W)	Closes RCV-23B	Perform Step 4.23
R-24A*	Containment Purge (AB 155')	Gas	Scint.	Closes containment purge supply & exhaust dampers 2866C & 2867C and 3198A & D	No input to this alarm
R-24B*	Containment Purge (AB 155')	Gas	Scint. (Victoreen)	Closes valves: 2866D &2867D, 3196, 3197, 3198B & C	No input to this alarm
R-25A/ R-25B*	Spent Fuel Pool Ventilation (AB 184')	Gas	Scint. (Victoreen)	Trip fuel bldg supply and exhaust fans; closes SFP HVAC supply and exhaust dampers; starts associated trains of penetration room filtration.	No input to this alarm
R-26A	Recycle Evap. Cond. Recovery Unit (AB 100')	Liquid	Scint. (W)		Perform Step 4.24
R-26B	Waste Evap. Cond. Recovery Unit (AB 100')	Liquid	Scint. (W)		Perform Step 4.24
R-27A*	Containment (High Range)	Area	Ion Chamber (Victoreen)		No input to this alarm

Technical Specification related

LOCATION FH5

SETPOINT: Variable, as per FNP-1-RCP-252

ORIGIN: Radiation Monitor Cabinet Channels R-25A or R-25B, Spent Fuel Pool Vent

H5
SFP AREA RE25 A OR B HI RAD

PROBABLE CAUSE

1. High Radiation Level in the discharged air from the Spent Fuel Pool Area Ventilation Fans.
2. 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.

AUTOMATIC ACTION

NOTE: The unaffected train penetration room filtration system may also start, due to low ΔP in the spent fuel pool.

Trips the Fuel Handling Area Supply and Exhaust Fans, closes the Fuel Handling Area Supply and Exhaust Dampers AND starts the Penetration Room 1A OR 1B Filtration Units.

OPERATOR ACTION

1. Determine which radiation monitor indicates high activity.
2. Verify that the required automatic actions listed above have occurred. IF any automatic actions have not occurred, THEN go to FNP-1-SOP-58.0. (The section for Fuel Handling Area Heating and Ventilation Operation for guidance)
3. Announce receipt of the alarm and the affected area on the public address system.
4. Have all personnel evacuate the affected area.
5. Implement FNP-0-EIP-9, EMERGENCY CLASSIFICATION AND ACTIONS.
6. Determine the validity of the high activity indication as follows:
   - 6.1 Verify that the instrument is aligned for normal operation and is functioning properly.
   - 6.2 Sample or survey the affected system or area as required.
7. Determine the source or cause of the high activity and correct or isolate as required.
8. DO NOT allow personnel to enter the affected area without the approval of the Health Physics Department.
9. IF high activity indication is due to instrument failure, THEN refer to Technical Specifications, section 3.3.8.

{CMT 0008659} applies to entire annunciator.}

References: A-177100, Sh. 310; U-258400; D-181658; D-181671; D-177394 Sh. 1 & 2; FSAR, Section 11.4; D-175045

3.0 Precautions and Limitations

3.1 The 600V Load Center Dampers are controlled by outside air temperature (N1V47TSHL3633) and this feature should not be defeated:

- Dampers close when temperature $\geq 60°$ F
- Dampers open when temperature $\leq 56°$ F

3.2 Placing an engineered safety features pump in local control places its respective pump room cooler in automatic only.

3.3 PRF System auto start from R-25A or R-25B requires operator action to verify open SFP TO 1A PRF SUPPLY DMPR, Q1V48HV3538A or SFP TO 1B PRF SUPPLY DMPR, Q1V48HV3538B.

3.4 Auxiliary Building Battery Charger Room Coolers are no longer considered attendant equipment. TS 3.7.19, ESF Room Coolers, is now applicable, and Auxiliary Building Battery Charger Room Coolers are considered support equipment. A battery charger room cooler can only support one operating charger. For this situation, operating means in service, load testing, in general when producing heat.

3.5 When Service Water is unavailable to these room coolers, their associated fan motor must remain tagged out to prevent the fan from running and adding a heat load to the associated room.

- CCW Pump Room
- Battery Charger Room
- 1A/1B MCC Room

3.6 600V Load Center 1D & 1E Room Coolers are no longer considered attendant equipment. TS 3.7.19, ESF Room Coolers, is now applicable, and 600V Load Center 1D & 1E Room Coolers are considered support equipment.

4.8 Fuel Handling Area Heating and Ventilation Operation.

CAUTION:	**One train of PRF must be in operation per step 4.8.2.1 if SFP HVAC is not in service. (SFP HVAC shuts down or the SFP EXH FAN SUCT DMPR Q1V48HV3990A(B) for the running fan is found Closed.)**

NOTE:	**BOTH trains of PRF should be started and Spent Fuel Pool Ventilation secured prior to sipping known leaking fuel assemblies in the Spent Fuel Pool to prevent radioactive gas release causing R-25A or B alarm and subsequent PRF auto start. (CMT 10518)**

4.8.1 To place the fuel handling area heating and ventilation in service, perform the following:

CAUTION:	**Both Auxiliary Building Main Exhaust fans must be in operation if Containment Purge and SFP Exhaust fans are running simultaneously.**

NOTE:	**Exhaust fan will shutdown after 20 seconds if dampers are not open in step 4.8.1.2.**

4.8.1.1 Start one of the following fans.

- 1A SFP EXH FAN, N1V48M001A
- 1B SFP EXH FAN, N1V48M001B

4.8.1.2 Open the following:

- SFP EXH FAN SUCT DMPR, Q1V48HV3990A
- SFP EXH FAN SUCT DMPR, Q1V48HV3990B
- SFP AHU DISCH TO SFP Q1V48HV3991A
- SFP AHU DISCH TO SFP Q1V48HV3991B

4.8.1.3 Start SFP AHU SUPP FAN, N1V48M002.

4.8.1.4 Verify open the following:

- SFP TO PRF FLTR UNIT Q1V48HV3538A
- SFP TO PRF FLTR UNIT Q1V48HV3538B

4.8.1.5 IF required, THEN secure any running train(s) of PRF per FNP-1-SOP-60.0, PENETRATION ROOM FILTRATION SYSTEM.

NOTE:	**BOTH trains of PRF should be started and Spent Fuel Pool Ventilation secured prior to sipping known leaking fuel assemblies in the Spent Fuel Pool to prevent radioactive gas release from causing the R-25A or R-25B alarm and subsequent PRF auto start. (CMT 10518)**

4.8.2 To remove fuel handling area heating and ventilation from service, perform the following:

4.8.2.1 IF 'A' train PRF is to be run AND aligned to SFP Room, THEN perform the following:

1. Verify requirements of Step 3.16 and 3.19 are met.

NOTE:	**Technical Specification SR 3.7.12.1 requires both trains of PRF aligned to the spent fuel pool room during movement of irradiated fuel assemblies in the spent fuel pool room. The following step is N/A during fuel movement.**

2. Close SFP TO 1B PRF SUPPLY DMPR, Q1V48HV3538B.
3. Verify open SFP TO 1A PRF SUPPLY DMPR, Q1V48HV3538A
4. Start 1A PRF RECIRC FAN, Q1E15M002A.
5. Start 1A PRF EXH FAN, Q1E15M001A
6. IF Q1V48HV3538B NOT closed THEN, start 'B' train PRF as follows:
    a. Start 1B PRF RECIRC FAN, Q1E15M002B
    b. Start 1B PRF EXH FAN, Q1E15M001B

3.7 PLANT SYSTEMS

3.7.12 Penetration Room Filtration (PRF) System

LCO 3.7.12 Two PRF trains shall be OPERABLE.

------------------------------------ NOTE ------------------------------------
The PRF and Spent Fuel Pool Room (SFPR) boundaries may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4 for post LOCA mode of operation,
During movement of irradiated fuel assemblies in the SFPR for the fuel handling accident mode of operation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PRF train inoperable.	A.1 Restore PRF train to OPERABLE status.	7 days
B. Two PRF trains inoperable in MODE 1, 2, 3, or 4 due to inoperable PRF boundary.	B.1 Restore PRF boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4. <u>OR</u> Two PRF trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the SFPR.	D.1 Place OPERABLE PRF train in operation. <u>OR</u> D.2 Suspend movement of irradiated fuel assemblies in the SFPR.	Immediately Immediately

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Two PRF trains inoperable during movement of irradiated fuel assemblies in the SFPR.	E.1	Suspend movement of irradiated fuel assemblies in the SFPR.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	---------------------------NOTE--------------------------------- Only required to be performed during movement of irradiated fuel assemblies in the SFPR. -- Verify two PRF trains aligned to the SFPR.	24 hours
SR 3.7.12.2	Operate each PRF train for ≥ 15 minutes in the applicable mode of operation (post LOCA and/or refueling accident).	31 days
SR 3.7.12.3	Perform required PRF filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.4	Verify each PRF train actuates and the normal spent fuel pool room ventilation system isolates on an actual or simulated actuation signal.	18 months
SR 3.7.12.5	Verify one PRF train can maintain a pressure ≤ -0.125 inches water gauge with respect to adjacent areas during the post LOCA mode of operation at a flow rate ≤ 5500 cfm.	18 months on a STAGGERED TEST BASIS
SR 3.7.12.6	Verify one PRF train can maintain a slightly negative pressure with respect to adjacent areas during the fuel handling accident mode of operation at a flow rate ≤ 5500 cfm.	18 months on a STAGGERED TEST BASIS

1. AUX BLDG VT-40304B07 015

Given the following plant conditions:

- Unit 1 is Mode 5.
- Containment purge is in operation.
- STP-50, Radiation Monitor Monthly Source Check, is being performed.
- R-25A, SFP VENT, radiation monitor starts increasing.

Which one of the following describes the system response if R-25A exceeds the alarm setpoint?

A. The automatic actions of R-25A will be blocked while performing this STP.

B. The 1A fuel handling area supply and exhaust fans trip and the 1A fuel handling area supply and exhaust dampers close. The 1B fuel handling area supply and exhaust fans start.

C✓ The fuel handling area supply and exhaust fans trip, the fuel handling area supply and exhaust dampers close and the penetration room 1A and 1B filtration units start.

D. The fuel handling area supply and exhaust fans trip, the fuel handling area supply and exhaust dampers close and the containment purge supply and exhaust valves close.

Feedback

FH5 SFP AREA RE25 A OR B HI RAD

A. Incorrect- If the alarm circuit actuates during the STP the automatic functions provided by the instrument will occur.

B. Incorrect- The A fuel handling area fans will trip, the fuel handling area dampers will close, but penetration room filtration units will start, not the other train of SFP HVAC.

C. Correct - These are the correct automatic actions that will occur when R-25A alarms. FH5 SFP AREA RE25 A OR B HI RAD Automatic action: Trips the Fuel Handling Area Supply and Exhaust Fans, closes the Fuel Handling Area Supply and Exhaust Dampers AND starts the Penetration Room1A OR 1B Filtration Units.

When R25A alarms, 1A PRF starts. When the Fuel handling area supply and exhaust dampers close, the other train of PRF starts.

Lesson plan auto starts:
During normal operation, the PRF system is aligned to automatically process the exhaust air from the spent fuel pool upon receipt of an actuation signal initiated by either (1) high radiation or (2) low flow in the spent fuel pool exhaust system.

D. Incorrect- The fuel handling area fans will trip, the fuel handling area dampers will close, but the containment purge valves will not close.

Notes

2006 NRC exam

K/A: Area Radiation Monitoring (ARM) System - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including: Radiation levels.

2. AUX BLDG VT-62107B01 004

Given the following plant conditions:

- Unit 1 is Mode 6 and refueling operations are in progress.
- Containment purge is in operation.

During a Channel Operational Test (COT) on R-25B, SFP VENT, the trip setpoint was found to be out of tolerance high and this was reported to the Shift Manager. After this report was made R-25A, SFP VENT, failed high and the red alarm light is illuminated.

Which one of the following describes the system response following the failure and the actions that **fully** meet the required actions IAW Tech Specs?

The Spent Fuel handling area supply and exhaust fans __(A)___.
The Spent Fuel handling area supply and exhaust dampers __(B)___.
Immediately ____________(C)____________________.

A. (A) trip

(B) remain open

(C) ensure BOTH PRF trains are operating

B. (A) remain running

(B) close

(C) secure fuel handling in the SFP room

C✓ (A) trip

(B) close

(C) ensure BOTH PRF trains are operating

D. (A) remain running

(B) remain open

(C) secure fuel handling in the SFP room

Feedback

TS 3.3.8 immediate ACTION completion times

SRO level due to 55.43 b 2

A. Incorrect- The fuel handling area supply and exhaust fans trip, the fuel handling area supply and exhaust dampers close, and 1A and 1B PRF units start due to the high alarm on R-25A. This is due to the following, when R25A alarms, 1A PRF starts. When the Fuel handling area supply and exhaust dampers close, the other train of PRF starts.

B. Incorrect- Both trains of PRF start. The proper TS action is to ensure both trains of

PRF start. If this can not be done then condition C will have the fuel movement stopped.

C. Correct - These are the correct automatic actions that will occur when R-25A alarms. FH5, SFP AREA RE25 A OR B HI RAD Automatic action: Trips the Fuel Handling Area Supply and Exhaust Fans, closes the Fuel Handling Area Supply and Exhaust Dampers AND starts the Penetration Room 1A OR 1B Filtration Units.

This is also the TS action for both rad monitors OOC. The candidate has to know that a failed COT will render the rad monitor inoperable and the one failed high will cause the automatic actions. The response to ensure Both trains of PRF are running is IAW TS 3.3.8.

When R25A alarms, 1A PRF starts. When the Fuel handling area supply and exhaust dampers close, the other train of PRF starts. Normally the supply fan and one exhaust fan, A or B, are in operation.

Lesson plan auto starts:
During normal operation, the PRF system is aligned to automatically process the exhaust air from the spent fuel pool upon receipt of an actuation signal initiated by either (1) high radiation or (2) low flow in the spent fuel pool exhaust system.

D. Incorrect- The fuel handling area fans will trip, the fuel handling area dampers will close, and both the Penetration Room 1A and 1B Filtration Units will be operating. This is the incorrect TS action. Proper use of Tech specs would require first attempting condition B before attempting C. Also the automatic actions have placed both PRF trains in service, so Condition B is met. There would be no reason to enter condition C.

Bases
3. Spent Fuel Pool Room Radiation
The LCO specifies two required Gaseous Radiation Monitor channels to ensure that the radiation monitoring instrumentation necessary to initiate the PRF remains OPERABLE. Each monitor will initiate the associated train of PRF and isolate the normal Spent Fuel Pool Room ventilation.
For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY requires correct valve lineups, sample pump operation, and detector OPERABILITY.

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

K/A
072 Area Radiation Monitoring
A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the ARM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure.

K/A MATCH ANALYSIS
R-25A, is a ventilation monitor for the area of the SFP and is inoperable due to a detector failure. Tech Specs are impacted by this failure and must be complied with to correct, control, or mitigate the consequences. The question is SRO-only level because it requires basis knowledge to arrive at the answer, in which the candidate needs to know what the affects are of a failed COT which is defined in TS bases.

3. RMS-40305A07 003

Annunciator FH5, SFP AREA RE25 A OR B HI RAD, is in alarm on Unit 1.

It has been determined that Spent Fuel Pool (SFP) Exhaust Flow Gas monitors R-25A and R-25B indicate high activity.

Which ONE of the following describes the automatic action(s) that occur as a result of this alarm?

The SFP supply and exhaust fans ___(A)___, the SFP HVAC supply and exhaust dampers close, and ___(B)___.

A. (A) remain running

(B) penetration room filtration units do **NOT** start

B. (A) remain running

(B) BOTH penetration room filtration units 1A and 1B start

C. (A) trip

(B) the penetration room filtration units do **NOT** start

D✓ (A) trip

(B) BOTH penetration room filtration units 1A and 1B start

Feedback

A - Incorrect, These fans trip and will not remain running as ctmt purge and minipurge fans do. This is plausible since on an alarm from R-24A and B the mini purge fans and ctmt purge fans will remain running and the valves will close and PRF does not start.

B - Incorrect, These fans trip and with both alarms R-25 A & B in alarm, both filtration units will start.

C - Incorrect, see B above

D - Correct- These fans trip and BOTH trains of PRF starts

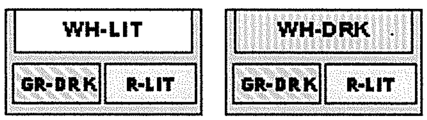
Notes

2001 nrc exam

52. 074EA1.28 005/NEW/RO/C/A 3.7/3.9/N/N/3/HBF/Y

The following plant conditions exist on Unit 1:

- FRP-C.2, Response to Degraded Core Cooling, is in progress.
- All attempts to establish HHSI flow were unsuccessful.
- SG depressurization to 100 psig has been completed.
- CETCs are 725°F and steady.
- All ECCS disconnects have been closed.
- B Train SI could **NOT** be reset, MLB 1 11-1 remains lit.
- The OATC has placed all SI Accumulator Discharge Isolation valves in the closed position.
- The following are the indications available on the MCB:



1B ACCUM DISCH ISO Q1E21MOV8808B

1C ACCUM DISCH ISO Q1E21MOV8808C

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	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.

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 **<u>also</u>** 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 $\geq$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 conditoin assumed then pressure in the RCS could be presumed to be very high-- not considering the SG cooldown to 100 psig.

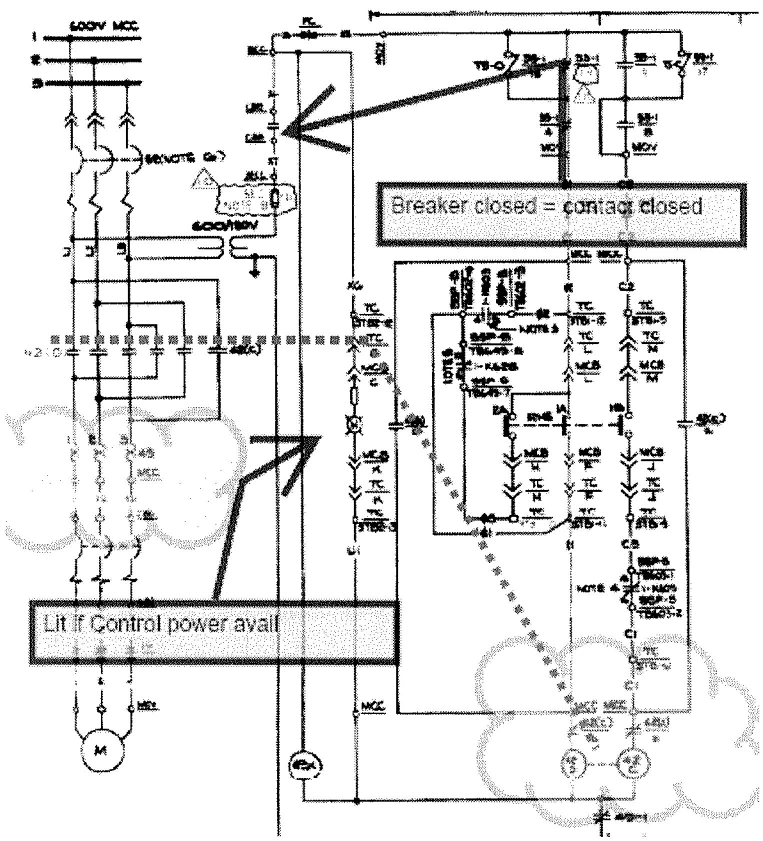
2) this part is correct see B#2.

D - Incorrect. 1) See A.

2) equivalent to discussion in C for the wrong train component.

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 successful 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

Question #	52
K/A	074EA1.28

REFERENCE Docs

Step	Action/Expected Response	Response NOT Obtained
8	**Check core cooling.**	
8.1	Check REACTOR VESSEL LEVEL indication - GREATER THAN 0% UPPER PLENUM.	8.1 IF SI established, THEN return to step 2. IF NOT, proceed to step 9.
8.2	Check core exit T/Cs - LESS THAN 700°F.	8.2 IF core exit T/Cs falling, THEN return to step 2. IF NOT, proceed to step 9.
8.3	Go to procedure and step in effect.	
9	**Check SI accumulator discharge valve status.**	
9.1	Check power to discharge valves - AVAILABLE. 1A(1B,1C) ACCUM DISCH ISO [] Q1E21MOV8808A [] Q1E21MOV8808B [] Q1E21MOV8808C	9.1 Close accumulator discharge valve disconnects using ATTACHMENT 1.
9.2	Check discharge valves - OPEN. 1A(1B,1C) ACCUM DISCH ISO [] Q1E21MOV8808A [] Q1E21MOV8808B [] Q1E21MOV8808C	9.2 IF accumulators have NOT discharged, THEN open discharge valves. 1A(1B,1C) ACCUM DISCH ISO [] Q1E21MOV8808A [] Q1E21MOV8808B [] Q1E21MOV8808C
10	**Monitor CST level.**	
10.1	[CA] Check CST le [illegible] than 5.3 ft. CST LVL [] LI 4132A [] LI 4132B	10.1 Align AFW pumps suction to SW using FNP-1-SOP-22.0, AUXILIARY FEEDWATER SYSTEM.
10.2	Align makeup to the CST from water treatment plant OR demin water system using FNP-1-SOP-5.0, DEMINERALIZED MAKEUP WATER SYSTEM, as necessary.	

RO "checks power available' and OPEN ; should be as found condition since EEP-1 step 6 performs this action and likely first transition from E-0 for conditions.

IF NOT LOCAL actions directed step 9.1 RNO... operation here is not likely. this is a check step.

KA match;

Degraded or inadequate Core Cooling + ECCS Accumulators (Core Flood tanks) operation and controls and indications.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
14	**[CA] Check if SI accumulators should be isolated.**	

NOTE: Step 14.1 is a continuing action.

Action/Expected Response	Response NOT Obtained
14.1 [CA] Check at least two RCS hot leg temperatures - LESS THAN 350°F. RCS HOT LEG TEMP [] TR 413	14.1 Perform the following. 14.1.1 <u>WHEN</u> at least two RCS hot leg temperatures are less than 350°F, <u>THEN</u> perform steps 14.2 and 14.3 to isolate accumulators. 14.1.2 Proceed to step 15. OBSERVE CAUTION PRIOR TO STEP 15.
14.2 Reset SI. [] MLB-1 1-1 not lit (A TRN) [] MLB-1 11-1 not lit (B TRN)	14.2 <u>IF</u> any train will <u>NOT</u> reset using the MCB SI RESET pushbuttons, <u>THEN</u> place the affected train S821 RESET switch to RESET. (SSPS TEST CAB.)
14.3 Close all SI accumulator discharge valves. 1A(1B,1C) ACCUM DISCH ISO [] Q1E21MOV8808A [] Q1E21MOV8808B [] Q1E21MOV8808C	14.3 Perform the following. 14.3.1 Vent any SI accumulator that cannot be isolated. ACCUM N2 VENT [] HIK 936 open

CAUTION: Do not mention S821 since part of SRO WE11EG2.1.28

add for plausibility of distractor.

KA match;

Degraded or inadequate Core Cooling + ECCS Accumulators (Core Flood tanks) operation and controls and indications.

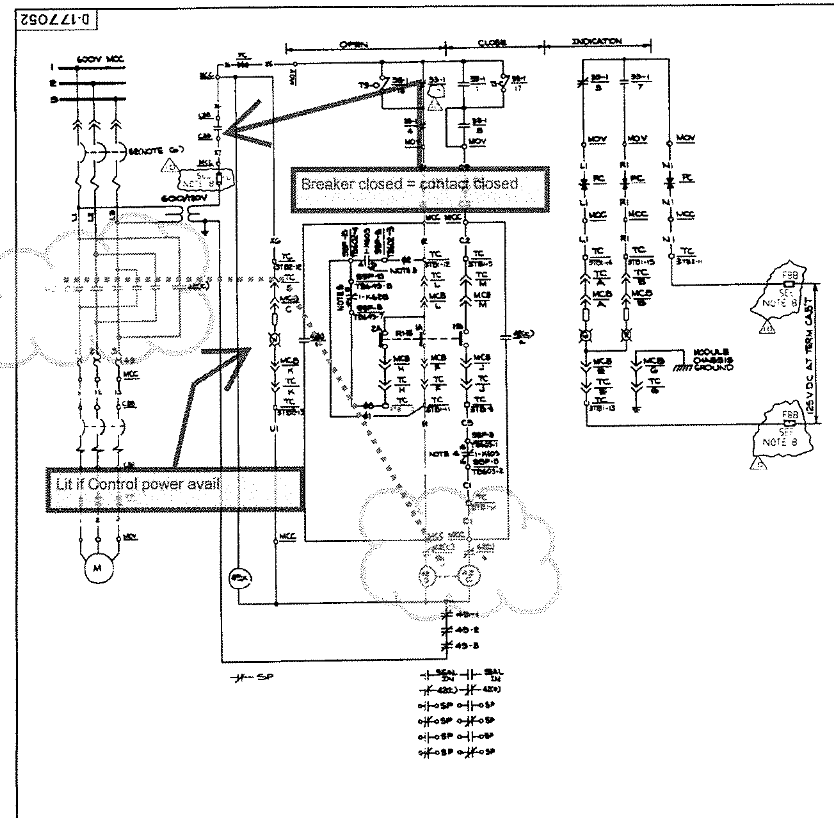
SI ACCUM	1A	1B	1C
1A(1B,1C) ACCUM N2 SUPP/VT ISO Q1E21HV	[] 8875A open	[] 8875B open	[] 8875C open

if disconnect not closed locally, then RNO action would be required.

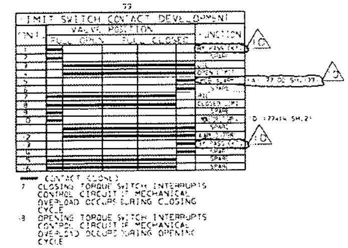
14.3.2 <u>IF</u> an accumulator can <u>NOT</u> be isolated or vented, <u>THEN</u> consult the TSC staff to determine contingency actions.

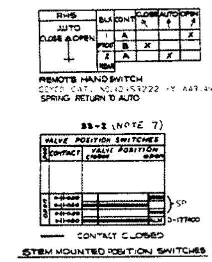
____Page Completed

D-177052



SERVICE	PAC	SCHEME	MCC CUBICLE	VALVE	MAIN CONTROL BOARD	TERMINATION CABINET	SOLID STATE PROT SYSTEM	REMOTE HANDSWITCH	POWER PENETRATION	CONTROL PENETRATION	LIMIT SWITCH	STEM MOUNTED POSITION SW.	DRAWING 95% ACCESS No	START UP NUMBER	CIRCUIT BREAKER BOX
ACCUMULATOR IB DISCHARGE VLVE	IV	BFY-62	FV-62	Q1E21 MOV8808B-B	N1H11NG MCB2500A-4B	Q1H25L029-B	Q1H11NG SSP2501-B	N1E21 HS0808-B	Q1T52B016-B	Q1T52B088-B	N1E21 25880808A-B	Q1E21 25808088-B	U-170075	52	Q1E18B088-B



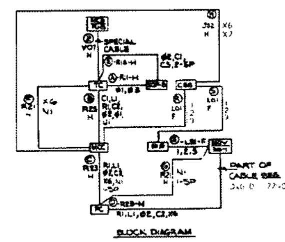


NOTES

1. VALVE IS SHOWN IN FULLY CLOSED POSITION.
2. FOR MOTOR AND STARTER SIZES SEE DWG B-177556 SH. 19 & 20.
3. CONTACT CLOSES ON SAFETY INJECTION SIGNAL.
4. CONTACT OPENS ON SAFETY INJECTION SIGNAL.
5. CONTACT CLOSES AT AND ABOVE P11 LOW PRESSURIZER PRESSURE SET POINT.
6. POWER LOCKED OUT FROM VALVE DURING NORMAL OPERATION.
7. LIMIT SWITCHES ARE PROVIDED WITH NAMCO EC-210 CONNECTORS (SEE ... FOR PIN DETAILS).
8. SEE A- B 997 FOR FUSE TYPE AND SIZE.

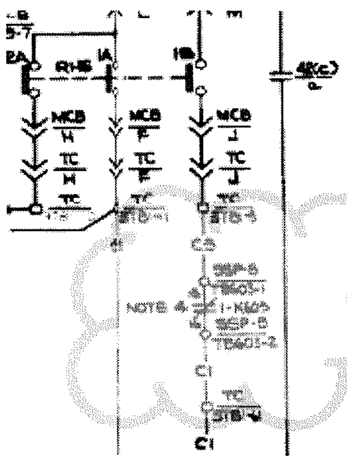
REFERENCE DRAWINGS

A-177538 ELECTRICAL GENERAL DETAILS AND NOTES.
B-177556 MCC SCHEDULES



SOUTHERN SERVICES INC.
FOR
ALABAMA POWER COMPANY

D-177052

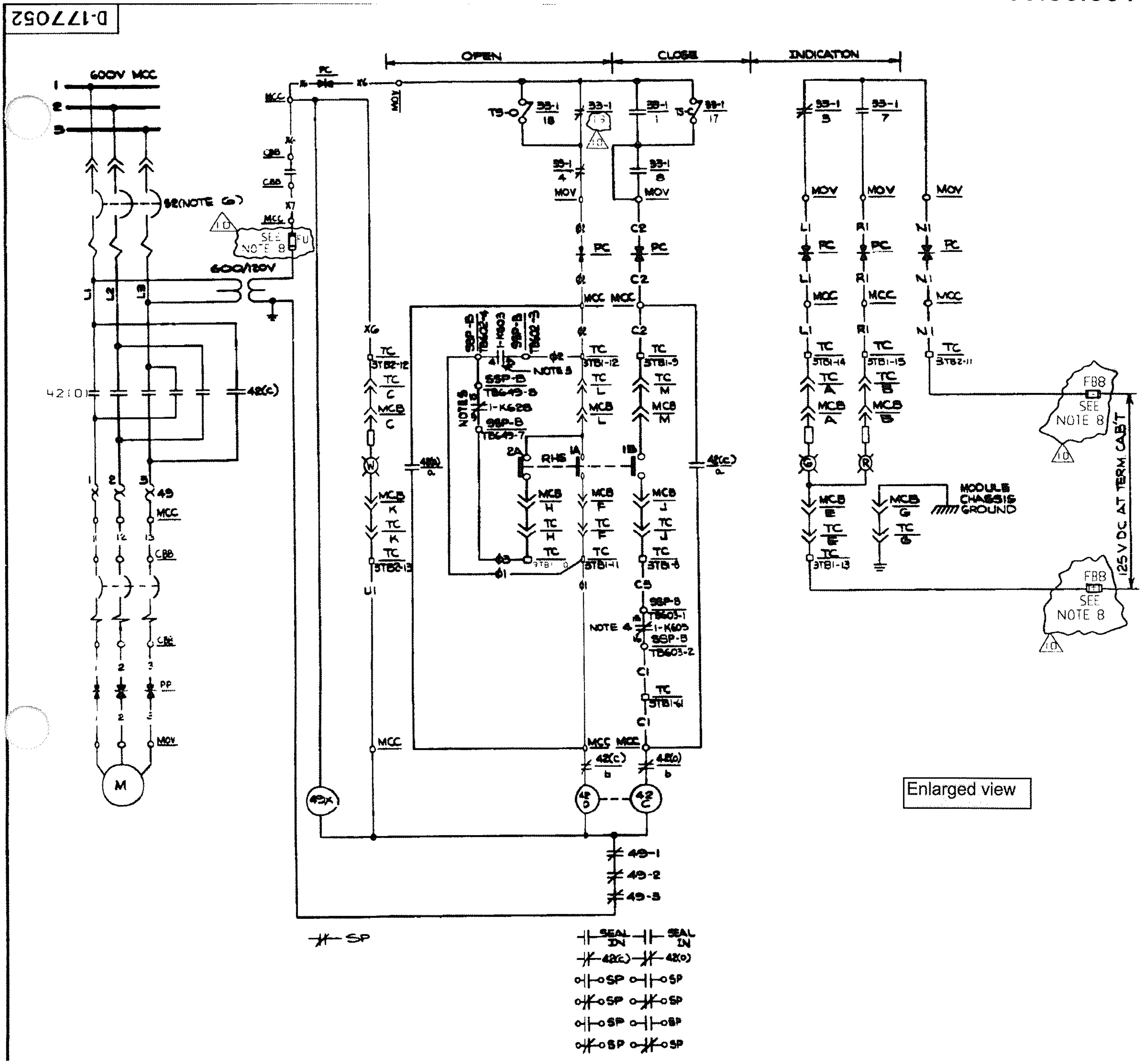


NOTES

1. VALVE IS SHOWN IN FULLY CLOSED POSITION.
2. FOR MOTOR AND STARTER SIZE SEE DWG B-177556 SH 19 & 20.
3. CONTACT CLOSES ON SAFETY INJECTION SIGNAL.
4. CONTACT OPENS ON SAFETY INJECTION SIGNAL.
5. CONTACT CLOSES AT AND ABOVE P11 LOW PRESSURIZER PRESSURE SET POINT.

RHS	BLK	CONT	CLOSE	AUTO	OPEN
AUTO CLOSE OPEN	1 FRONT	A			X
		B	X		
	2 REAR	A		X	

REMOTE HANDSWITCH
GEMCO CAT. NO. 0459222-X-AA9.AW
SPRING RETURN TO AUTO


D-177052
600V MCC
OPEN
CLOSE
INDICATION
600/120V
SEE NOTE 8
FU
42(O)
42(C)
49
M
49X
MOV
PC
MCC
TC
MCB
RHS
42 O
42 C
49-1
49-2
49-3
SP
SEAL IN
NOTE 4
NOTE 5
NOTE 6
SSP-B
MODULE CHASSIS GROUND
FBB
125 V DC AT TERM CAB'T
Enlarged view

53. 075A2.03 001/NEW/RO/C/A 2.5/2.7/N/N/4/CVR/Y

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).

Question # 53

K/A 075A2.03

REFERENCE Docs

Step | **Action/Expected Response** | **Response Not Obtained**

NOTE:
- DEHC point CNDP1 displays condenser pressure in Hga absolute on the point detail page. On all other pages CNDP1 displays condenser pressure in psia.
- IPC points PT0214 and PT0215 display condenser pressure in psia.
- MCB recorder PR 4029 displays condenser pressure in psia.
- Main turbine trip will occur at 4.41 psia (9 in Hga)
- SGFP trip will occur at 5.9 psia (12 in Hga)

< p-9 therefore No Rx Trip on turbine trip.

LOSS of All MFW pumps--> requires a trip per AOP-13 when > 5%.

__ **1 Monitor Condenser pressure**

1.1 IF condenser pressure is at 2.3 psia, THEN increased monitoring of condenser pressure is required.

__ **2 Monitor turbine trip criteria.**

2.1 WHEN turbine power greater than or equal to 30%, THEN verify condenser pressure less than 2.7 psia (5.5 in Hga).

RNO 2.1 Perform the following.

2.1.1 IF reactor power greater than 35%, THEN trip the reactor.

2.1.1.1 Go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.

2.1.1.2 Perform the remainder of this procedure in conjunction with FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.

2.1.2 IF reactor power less than 35%, THEN place MAIN TURB EMERG TRIP to TRIP for ≥ 5 seconds.

2.1.2.1 Perform FNP-1-AOP-3.0, TURBINE TRIP BELOW P-9 SETPOINT in parallel with this procedure.

2.1.3 Proceed to step 3.

Step 2 continued on next page

__Page Completed

Step	Action/Expected Response	Response Not Obtained
		1.12.5 Place STM DUMP INTLK TRAIN A and B to ON.
1.13	Check reactor power change < 15%	1.13 Notify Shift Radiochemist to sample the RCS per FNP-1-STP-746.
1.14	Check parameters within limits for continued at power operation. • Pressurizer level greater than 15% • Pressurizer pressure greater than 2100 psig • SG narrow range levels 35%-75% • TAVG 541°F - 580°F • Control rod bank position Lo-Lo Annunciator FE2 Clear • Delta I within limits specified in the COLR	1.14 IF the Team is NOT confident that a parameter is being restored, THEN trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.
		Correct part #1
__ 2	**Check Both SGFPs - TRIPPED**	2 Proceed to step 3 OBSERVE CAUTION prior to step 3.
2.1	Check Reactor Power - LESS THAN 5%.	2.1 Trip the reactor and go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.
2.2	Reduce reactor power to approximately 2%.	B & C distractors part #2
2.2.1	Verify rod control in MANUAL.	
2.2.2	Stabilize TAVG by adjusting rod position and/or boron concentration.	
2.2.3	Check for proper operation of steam dumps.	

Step 2 continued on next page

__ Page Completed

CONTROL INTERLOCKS

	Interlock	Source	Setpoint	Coincidence & Light Status	Function
6.	C-7 Sudden Loss of Load	Turb. Impulse Press Instr. 447 rate ckt.	15% Turb. Power Reduction 120 sec Time Const.	1/1 > setpoint lit > setpoint	Arms steam dump valves. Manually reset by placing the control mode selector switch to reset momentarily.
7.	C-9 Condenser Available	Cond. Press. Switch and Circ Water Pump Bkrs.	8" Hg Vac. and Closed	2/2 > setpoint and 1/2 > setpoint lit > setpoint	Allows steam dump valves to be armed. NO stm dumps
8.	C-11 Bank D Stop	P/A Converter Bank D Position	220 Steps	1/1 = setpoint lit after being at setpoint for 3 min.	Stops outward rod motion in auto.
9.	C-20 AMSAC Enabled	Turb. Impulse Pressure Inst. 2446 and 2447	> 40%	2/2 > setpoint; also sealed in for 260 seconds after Pimp lowers to < 40%; lit < setpoint	Allows AMSAC to be armed.

SETPOINT: 1. 1.3 psia when < 30% power
2. 2.3 psia when > 30% power

ORIGIN: DEH

K1
TURB COND VAC LO

PROBABLE CAUSE

1. Low Circulating Water Flow or I
2. Gland Seal Steam System fault.
3. Improper valve lineup resulting i

Turbine trip and no dumps results in significant reduction in Mass flow into the condenser

AUTOMATIC AC

NOTE: **<u>IF</u> condenser vacuum decreases to 21" Hg (4.41 psia), <u>THEN</u> a turbine trip occurs.**

$$q_{cond} = m \Delta h_{cond}$$
$$m_{stm} \downarrow \Rightarrow q_{cond} \downarrow$$
$$q_{cond} \downarrow = m_{circwater}\, c\,(T_{out} - T_{in}) \downarrow$$
$$m_{circwater}\quad c \leftrightarrow$$
$$(T_{in}) \leftrightarrow$$
$$(T_{out} - T_{in}) \downarrow \Rightarrow (T_{out}) \downarrow$$

NONE

OPERATOR ACTION

CAUTION: **<u>WHEN</u> the turbine is operating at ≥ 30% load, <u>THEN</u> the maximum permissible condenser pressure is 5.5 inches Hg (2.7 psia). <u>WHEN</u> the turbine is operating at < 30% load, <u>THEN</u> the maximum permissible condenser pressure is 3.5 inches Hg (1.7 psia).**

1. Perform the actions required by FNP-1-AOP-8.0, PARTIAL LOSS OF CONDENSER VACUUM.

References: A-177100, Sh. 491; D-172803; D-170812, Sh. 2; U-162213, Tab 5; Westinghouse Customer Advisory Letter 86-02; DCP P-95-1-8943

54. 076A3.02 001/NEW/RO/MEM 3.7/3.7//N/3/CVR/Y

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).

Question #	54
K/A	076A3.02

REFERENCE Docs

These pressure switches shall meet or exceed seismic qualification requirements contained in IEEE 344-1975. (References 6.7.023 and 6.7.024)

3.43.4 Interface Requirements

The Electrical Distribution System shall provide power through the Annunciator Cabinet 1C for audible and visual annunciation upon closing of the pressure switch contacts. (References 6.4.202 - 6.4.205)

3.43.5 Failure Modes and Effects Analysis

There are redundant pressure switches and low pressure alarms on each train of service water. Therefore, should SW pressure approach the low pressure setpoint, the failure of a single pressure switch will not prevent an alarm.

A spurious low pressure alarm from one pressure switch could be checked for validity against the Auxiliary Building header pressure indicator. If the alarm appears true, plant operators should check the SW pumps for proper operation or take other actions as appropriate.

3.44 TURBINE BUILDING SERVICE WATER SUPPLY ISOLATION VALVES

TPNS Nos.	Unit 1	Unit 2
Train A -	Q1P16V515 Q1P16V516	Q2P16V515 Q2P16V516
Train B -	Q1P16V514 Q1P16V517	Q2P16V514 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.

a,b,c plausibility

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 longterm operation of the diesels.

3.44.2 Functional Requirements

3.44.2.1 The valve design inlet pressure and temperature conditions must be 150 psig and 115°F. (References 6.5.007 and 6.7.153)

3.44.2.2 The valve operator shall be capable of opening and closing the valve at a maximum differential pressure of 150 psig. (Reference 6.5.007)

3.44.2.3 The Turbine Building Service Water Supply Isolation Valves shall be electric motor operated. A handwheel must be provided on each valve operator to stroke the valve in the event of motor failure. (Reference 6.5.007) The Unit 1 SW Supply Isolation Valves are not readily accessible for manual operation due to the welded security covers on Valve Box VB-1.

3.44.2.4 The Inservice Testing Plan has established a stroke time for closing these motor operated valves of less than or equal to 75 seconds (+7.5 seconds). However, there are no system functional requirements imposing a specific stroke time.

3.44.2.5 Motor operators shall be capable of starting and accelerating the load with 80% nameplate voltage at the motor terminals. (References 6.5.007, 6.7.123, 6.7.150, and 6.7.175)

3.44.2.6 Safety-related MOVs are subject to the requirements of NRC Generic Letter 89-10, dated 6/28/89, "Safety-Related Motor Operated Valve Testing and Surveillance." (Reference 6.7.043)

3.44.3 Code Requirements

3.44.3.1 The valves must be designed in accordance with requirements of ASME B&PV Code, Section III, Class 3, 1971 Edition along with additional design codes and

55. 077AA1.02 001/BANK/RO/MEM 3.8/3.7/N/N/3/HBF/Y

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.

Question # 55

K/A 077AA1.02

REFERENCE Docs

3.3 Generator and Switchyard

3.3.1 Do not adjust generator excitation with the Manual Voltage adjust switch when the generator is on the line without Operations Manager approval.

3.3.2 WHEN adjusting VARs, THEN do NOT exceed 22 kv $\pm$ 5% (20.9 - 23.1 kv). IF outside this band, THEN reduce generator output due to stator overheating.

3.3.3 The Admin. limit for main generator MVARs is –300 to prevent the auto adjuster from going to its mechanical stop.

3.3.4 IF plant conditions are met per APPENDIX 5, LOAD LIMITATIONS WITH A 500 Kv TRANSMISSION LINE OUT OF SERVICE, THEN the total plant MW limitation value has to be met within 30 minutes from the initiating condition(s). (AI 2004204794)

3.3.5 Alabama Control Center should be notified any time a Power System Stabilizer is removed from or returned to service.

3.3.6 The NERC standards in APPENDIX 6 will be met whenever the plant is on line. These standards were promulgated in accordance with the Energy Policy Act to ensure grid reliability.

3.4 Chemistry

3.4.1 Maintain the RCS and steam generator chemistry within limits required by FNP-0-CCP-202, CHEMICAL SPECIFICATIONS.

3.4.2 Chemistry should be notified of the following:

- To sample RCS per STP-746 if Rx power changes by ≥ 15% of rated thermal power within a 1 hr period. (SR 3.4.16.2)
- Any significant changes in plant load

Quality Guide, and FSAR voltages for the ESF components. Ensure an Auto Log entry is made similar to the following: "During the Unit 1/2 refueling/forced outage, voltage will not be able to be raised / lowered to meet the APC 500kV voltage schedule. Relief granted by ACC (System Operator name)."

4.8.1.6 Unit capability curves should be adhered to when attempting to maintain bus voltage schedule.

4.8.1.7 Station service voltage limits should be observed when attempting to maintain bus voltage.

4.8.1.9 The Voltage is subject to change at the request of the ACC System Operator. ACC System Operator requests shall be met in a timely manner. Any emergency request shall be met as soon as possible.

4.8.1.10 The System Operator would like to minimize switching the shunt reactor in/out of service as much as possible. With one Unit off for refueling, placing the shunt in service, and leaving it in service, appears to be successful in allowing Farley to maintain the Voltage Schedule within the NERC allowable, ***undirected*** deviation. This also keeps the shunt reactor switching to a minimum.

4.8.1.11 The ACC System Operator does not want to deviate too far from the scheduled voltage because a single contingency could put Farley in a situation for high / low on-site voltages, and possible damage to equipment.

4.8.1.12 The ACC System Operator is allowed some discretion for short durations from the allowable, undirected deviation from the Voltage Schedule.

4.8.2 Guidelines to Raise and Lower System Voltage
The Farley Operators shall adjust the main generator reactive load voltage to meet the system requirements as directed by the System Operator, while observing the following guidelines.

4.8.2.1 The reactive load adjustments cannot exceed 22kV +/- 5% (20.9 - 23.1kV).

4.8.2.2 The Farley administrative limit is -300 MVARs to prevent the auto adjuster from going to its mechanical stop.

4.8.2.3 The 230kV Shunt Reactor is placed in service when the 230kV bus voltage needs to be lowered, and the 230kV Capacitor Bank is placed in service when the 230kV bus voltage needs to be raised. Because the two devices perform opposite functions, they never should be in service at the same time, and an interlock scheme is provided on switches 955 and 957 to prevent this from happening.

4.23 Generator Voltage Regulator Balance Meter Null Adjustment When Operating In Auto Voltage Control

CAUTION:	**DO NOT use manual voltage adjust switch to adjust generator voltage when in auto voltage control. Use the auto voltage adjust switch to adjust generator voltage, and the manual voltage adjust switch to adjust voltage regulator balance (null) meter.**

NOTE:	**The voltage regulator balance meter should be monitored more frequently during load changes.**

4.23.1 Monitor the voltage regulator balance meter to ensure it remains as close to zero as practical (approximately +/-0.25 volts).

4.23.2 IF null meter reading less than zero, THEN raise voltage regulator balance meter indication by slowly taking Manual Voltage Adjust Switch to lower.

4.23.3 IF null meter reading greater than zero, THEN lower voltage regulator balance meter indication by slowly taking Manual Voltage Adjust Switch to raise.

4.7.9 Have Maintenance check that no trip signal is present on the line distance relay, N1N31RLYGEN21KD. (AI 2008207695)

4.7.9.1 IF no trip signal is present, THEN reconnect the output of the relay by closing knife switch labeled "GENERATOR OVERCURRENT TRIP (KD-41 LOCKOUT)" located on Meter & Relay Panel No. 6 (N1H11L506).

4.7.9.2 IF a trip signal is present, THEN write a condition report to investigate the reason for a trip signal before reconnecting the output from the relay.

4.7.10 Using the Manual Voltage Adjuster Switch, increase generator output voltage to 22 KV, ± 0.2 KV, as indicated on analog voltmeter 4099 or digital voltmeter 5122. (CR2009103418)

NOTE:	**In the following step, the amber bar will be lit while the Regulator Transfer Switch is in the TEST position. (CR2009103418)**

4.7.11 Place the Regulator Transfer Switch in the TEST position.

NOTE:	**Raising the voltage with the Auto Voltage Adjust Switch will cause the null meter (VM4098) to move in the positive direction.**

4.7.12 Using the Automatic Voltage Adjuster Switch, null (zero) the regulator balance voltmeter.

CAUTION:	**To prevent a possible turbine trip that could occur if the regulator does not null out properly, perform step 4.7.14 without delay after completing step 4.7.13.**

4.7.13 Place the Regulator Transfer Switch in the ON position.

CAUTION:	**All generator voltage adjustments must now be made with the Automatic Voltage Adjuster Switch.**

4.7.14 Verify generator output voltage is approximately 22KV as indicated on analog voltmeter 4099 or digital voltmeter 5122. Adjust voltage as necessary using the Automatic Voltage Adjuster Switch.

APPENDIX B

TRANSFERRING THE VOLTAGE REGULATOR TO MANUAL

1.0 Have I&C attach a digital voltmeter across Voltage Regulator Transfer Meter VM4098.

NOTE:	**IF the Regulator Transfer Meter VM4098 needle has slight oscillations, THEN the meter should be nulled based on the average or midpoint of the oscillations.**

2.0 Zero the Voltage Regulator Transfer Meter VM4098 using the Man Voltage Adjust Switch using the digital voltmeter for closer match.

2.1 For 0 to 150 MVARs IN or for 0 to 100 MVARs OUT, a slightly positive value (between 0 & +0.1V) should be obtained on the Voltage Regulator Transfer Meter VM4098 using the Manual Voltage Adj Switch.

2.2 For 100 to 300 MVARs OUT, a slightly negative value (between 0 & -0.1V) should be obtained on the Voltage Regulator Transfer Meter VM4098 using the Manual Voltage Adjust Switch.

2.3 For more than 150 MVARs IN or more than 300 MVARs OUT, transfer to Manual should not take place without further evaluation unless an emergency condition exists. In the event of an emergency, use the guidance in step 2.1 to null for > 150 MVARs IN, and use the guidance in step 2.2 to null for > 300 MVARs OUT.

3.0 Going to Manual on the Voltage Regulator

NOTE:	**WHEN the voltage regulator is turned to TEST or OFF, THEN the PSS inputs are disabled.**

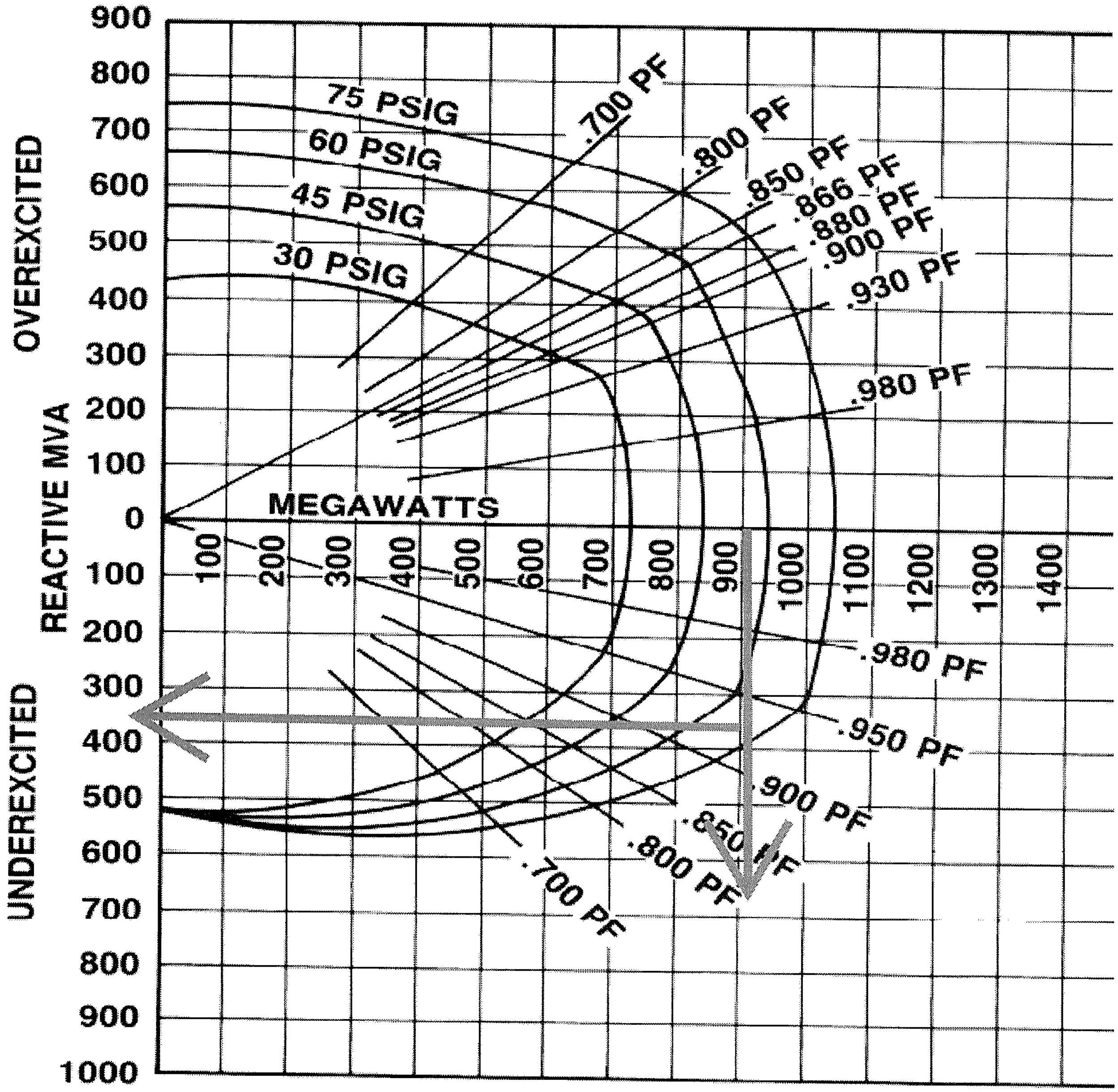
3.1 Turn Voltage Regulator Transfer Switch to TEST.

3.2 IF a large swing in generator voltage occurs, THEN turn Voltage Regulator Transfer Switch back to ON.

3.3 IF staying in manual on the Voltage Regulator, THEN record in the RO's Log the date, time, and reason for going to manual on the Voltage Regulator. (NERC Phase III)

NOTE:	**WHEN the voltage regulator is turned to TEST or OFF, THEN the PSS inputs are disabled.**

4.0 If required the Voltage Regulator Transfer Switch may be turned to OFF.



OVEREXCITED
UNDEREXCITED
REACTIVE MVA
MEGAWATTS
75 PSIG
60 PSIG
45 PSIG
30 PSIG
.700 PF
.800 PF
.850 PF
.866 PF
.880 PF
.900 PF
.930 PF
.980 PF
.950 PF
900
800
700
600
500
400
300
200
100
0
1000
1100
1200
1300
1400

17. Given the following:

- Unit 1 is at 100% RTP.
- All systems are aligned normally.
- Generator reactive load is currently at "0" MVARs.
- The Transmission Operator has notified the plant that system voltage problems require Unit 1 to establish the maximum allowable outgoing reactive load.

Which ONE of the following identifies the MAXIMUM outgoing reactive load in accordance with GOI-6, "Apparatus Operation", and the correct operation of the Exciter Voltage Adjuster?

	Maximum Outgoing Reactive Load	Exciter Voltage Adjuster
A.	240 MVARs	Lower
B.	240 MVARs	Raise
C.	300 MVARs	Lower
D.	300 MVARs	Raise

56. 078K2.02 001/NEW/RO/MEM 3.3/3.5/N/N/2/CVR/Y

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).

Question # 56

K/A 078K2.02

REFERENCE Docs

DG03
EE10
1B 600/208V MCC **AB - 121'** **B177556-2**
(CONT'D)

BKR	TPNS	DESCRIPTION	SEE PAGE
FBI6	------------	SPARE	
FBJ2L	------------	SPARE	
FBJ3L	------------	SPARE	
FBJ4R	------------	SPARE	
FBJ5L	**Q1R22L0006B-B**	**3-15 KVA SINGLE PHASE CONSTANT VOLTAGE TRANSFORMERS >>> 1H REG AC DIST PNL >>>**	G-80
FBJ5R	Q1P18M0002B-B	1B EMERG AIR COMP FOR MAIN STEAM ATMOS RELIEF VALVES	
FBJ7	N1T49M0001B-N	1B REACTOR CAVITY COOLING FAN	
FBK2	------------	SPARE	
FBK5L	**Q1R17B0002-B**	**1B 600/208V MCC XFMR >>> 1B MCC 208V SECTION**	G-82
FBM2	------------	SPARE	
FBM3	Q1P12M0001B-B	1B REACTOR MAKEUP WATER PUMP	
FBM4L	N1E15K0002A-N	PENETRATION FILTER MECH EQPT ROOM UNIT HEATER 1A	
FBM5	Q1E16M0001B-AB	DISC SWITCH Q1R18B003B-B >>> 1B CHARGING/HHSI PUMP ROOM COOLER FAN	
FBO2	N1T49MOV3310B-N	REACTOR CAVITY COOL FAN MOV	
FBO3	------------	SPARE	
FBO4	------------	SPARE	

DF03
ED10
1A 600/208V MCC **AB - 139'** **B177556-1**
(CONT'D)

BKR	TPNS	DESCRIPTION	SEE PAGE
FAJ5	Q1E19M0001C-A	1C CTMT POST LOCA AIR MIXING FAN	
FAJ6	---------------	SPARE	
FAJ7	---------------	SPARE	
FAK2L	Q1R37E0001A-A Q1R37L0004A-A	TRANSFORMER 1A (Q1R37E0001A-A) >>> 1B-A SIS DISTRIBUTION PANEL-TRAIN A Q1R37L0004A-A (EXCEPT FOR BREAKER 24, ALL BREAKERS IN PANEL 1B-A LEFT IN OPEN POSITION) >>> POST ACCIDENT HYDROGEN ANALYZER HEAT TRACING ALARM	
FAK2R	---------------	SPARE	
FAK4L	N1D11RE0010-N	PENETRATION RM AIR PARTICLE DET MONITOR	
FAK4R	---------------	SPARE	
FAK6	---------------	SPARE	
FAL2	---------------	SPARE	
FAL3L	Q1E17G0001A-A	1A CTMT H2 RECOMBINER HEATER	
FAL3R	---------------	SPARE	
FAL5L	Q1P18M0002A-A	1A EMERG. AIR COMPRESSOR FOR MAIN STEAM ATMOS. RELIEF VALVES	
FAM4	---------------	SPARE	
FAM5L	N1G21NDRE2608-N	RECYCLE EVAP CONTROL PANEL	
FAM5R	**---------------**	**600/208V MCC XFMR >>> 1A MCC 208V SECTION**	**F-97**

57. 103A4.03 001/MOD/RO/MEM 2.7/2.7/N/N/3/HBF/VER 5 EDITORIAL

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 [...].

Question # 57

K/A 103A4.03

REFERENCE Docs

3.3 INSTRUMENTATION

3.3.6 Containment Purge and Exhaust Isolation Instrumentation

LCO 3.3.6 The Containment Purge and Exhaust Isolation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Required radiation monitoring channel inoperable.	A.1 Restore the affected channel to OPERABLE status.	4 hours
B. ------------NOTE------------ Only applicable in MODE 1, 2, 3, or 4. ------------------------------- One or more Functions with one or more manual or automatic actuation trains inoperable. <u>OR</u> Required Action and associated Completion Time of Condition A not met.	B.1 Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves," for containment purge and exhaust isolation valves made inoperable by isolation instrumentation.	Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ------------NOTE------------ Only applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. -------------------------------- One or more manual channel(s) inoperable. OR Two radiation monitoring channels inoperable. OR Required Action and associated Completion Time for Condition A not met.	C.1 Place and maintain containment purge and exhaust valves in closed position. OR C.2 Enter applicable Conditions and Required Actions of LCO 3.9.3, "Containment Penetrations," for containment purge and exhaust isolation valves made inoperable by isolation instrumentation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

------------------------------NOTE------------------------------
Refer to Table 3.3.6-1 to determine which SRs apply for each Containment Purge and Exhaust Isolation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.2	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.6.3	Perform MASTER RELAY TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.6.4	Perform COT.	92 days
SR 3.3.6.5	Perform SLAVE RELAY TEST.	18 months
SR 3.3.6.6	---------------NOTE--------------- Verification of setpoint is not required. ---------------------------------- Perform TADOT.	18 months
SR 3.3.6.7	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.6.8	Verify ESF RESPONSE TIME within limit.	18 months on a STAGGERED TEST BASIS

Table 3.3.6-1 (page 1 of 1)
Containment Purge and Exhaust Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	1,2,3,4, (a), (b)	2	SR 3.3.6.6	NA
2. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5 SR 3.3.6.8	NA
3. Containment Radiation Gaseous (R-24A, B)	1,2,3,4 (a), (b)	1 2	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7	$\leq 2.27 \times 10^{-2}$ μCi/cc (c)(d) $\leq 4.54 \times 10^{-3}$ μCi/cc (c)(e) $\leq 2.27 \times 10^{-3}$ μCi/cc (c)(f)
4. Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a., for all initiation functions and requirements.			

(a) During CORE ALTERATIONS.
(b) During movement of irradiated fuel assemblies within containment.
(c) Above background with no flow.
(d) With mini-purge in operation.
(e) With slow speed main purge in operation.
(f) With fast speed main purge in operation.

Question #27

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103 A4.03	
	Importance Rating	2.7	

Ability to manually operate and/or monitor in the control room: ESF slave relays

Proposed Question: Common 27

Given the following conditions:

- A LOCA has occurred.
- Containment Pressure is 28 psig and rising.
- All required actuations have occurred.

Which ONE (1) of the following describes the conditions required and operation of relays to reset Containment Isolation Phase B?

A. The Phase B slave relays are de-energized when the Phase B control switch is placed in RESET. Components may be repositioned as required.

B. The Phase B master relay is de-energized when the control switch is placed in RESET. When the master relay is de-energized, components may be repositioned.

C. Containment Spray must be reset to allow resetting the master relay for Phase B. Placing the Phase B control switch to RESET will de-energize the slave relays to allow components to be repositioned.

D. Initiating condition must clear and control switch must be placed in RESET to de-energize the slave relays that allow components to be repositioned.

Proposed Answer: A

Explanation (Optional):

A. Correct.

B. Incorrect. Master relay only operates on the actuation signal. When resetting, slave relays reset

C. Incorrect. Spray and CIB have independent resets, even though initiating signal is the same

D. Incorrect. If switch is placed in RESET, initiating condition may still exist and the components will still reset

Technical Reference(s) OTO-SA-00001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 5

Comments:

WTSI 53122 developed but not yet used on NRC exam

1. CS&COOL-40302D07 011

Bortz

The containment spray system has actuated. The team wants to close MOV-8820A and B, CS pump to spray header isolation valves.

Which one of the following is the **minimum** action the operator must take to ensure MOV-8820A and B remain closed prior to closing the valves?

A. Reset the Phase B actuation signal ONLY by depressing a pair of reset pushbuttons on the MCB.

B. Wait one minute since the open signal is only present for one minute after the valves open.

C. Reset the containment spray actuation signal ONLY by depressing a pair of reset pushbuttons on the MCB.

D. Reset both the containment spray actuation signal and the Phase B actuation signal by depressing 2 sets of reset pushbuttons on the MCB.

Feedback

OPS-52102C

A. Incorrect – Phase B is a containment isolation signal and does not open any valves including MOV8820A/B

B. Incorrect – If the CS actuation signal is not reset then the valves will roll back open

C. Correct – **Reset the containment spray actuation signal ONLY by depressing a pair of reset pushbuttons on the MCB.**

P signal or CS actuation opens this valve and remains Sealed in with an R/L logic and is reset with 2 Pushbuttons on the MCB

D. Incorrect – this is not the min. since Phase B does not need to be reset

A containment isolation phase B and containment spray actuation can be manually initiated from the safeguards panel. There are four PHASE B CTMT ISO CS ACTUATION switches. These switches are grouped in two sets of two switches each. Simultaneously placing both switches in a set in the ACTUATE position will cause a manual containment phase B isolation and containment spray actuation signal. There are four reset push buttons on the safeguards panel--two for containment spray (train A and train B) and two for containment isolation phase B (train A and B). All four push buttons must be momentarily depressed to clear the containment spray actuation and containment isolation phase B signals.

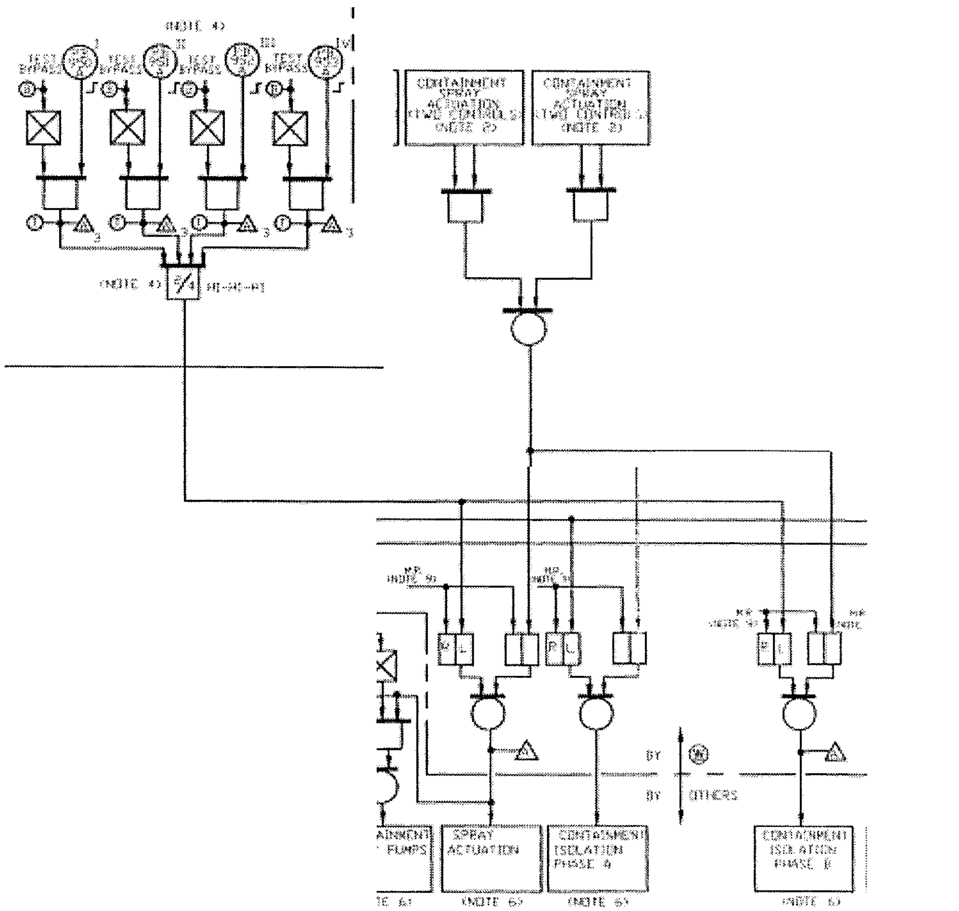
Notes

2005 NRC exam

026A4.05 Ability to manually operate and/or monitor in the control room:
Containment spray reset switches

List the automatic actions associated with the Containment Spray and Cooling System components and equipment during normal and abnormal operations including (OPS40302D07):

• Normal control methods
• Automatic actuation including setpoint (example SI, Phase-B, LOSP) and the effect of selecting the containment cooler control to local.



6. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.
7. FEEDWATER ISOLATION INCLUDES THE TRIPPING OF ALL MAIN FEEDWATER PUMPS.
8. SERVICE WATER SYSTEM ISOLATION IS USED ONLY IF REQUIRED.
9. THE REDUNDANT MANUAL RESET CONSISTS OF TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.

58. 103K3.03 001/NEW/RO/C/A 3.7/4.1/N/N/3/CVR/Y

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).

Question # 58

K/A 103K3.03

REFERENCE Docs

3.3 INSTRUMENTATION

3.3.6 Containment Purge and Exhaust Isolation Instrumentation

LCO 3.3.6 The Containment Purge and Exhaust Isolation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Required radiation monitoring channel inoperable.	A.1 Restore the affected channel to OPERABLE status.	4 hours
B. ------------NOTE------------ Only applicable in MODE 1, 2, 3, or 4. -------------------------------- One or more Functions with one or more manual or automatic actuation trains inoperable. <u>OR</u> Required Action and associated Completion Time of Condition A not met.	B.1 Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves," for containment purge and exhaust isolation valves made inoperable by isolation instrumentation.	Immediately

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ------------NOTE------------ Only applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. ------------------------------- One or more manual channel(s) inoperable. OR Two radiation monitoring channels inoperable. OR Required Action and associated Completion Time for Condition A not met.	C.1	Place and maintain containment purge and exhaust valves in closed position.	Immediately
	OR		
	C.2	Enter applicable Conditions and Required Actions of LCO 3.9.3, "Containment Penetrations," for containment purge and exhaust isolation valves made inoperable by isolation instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

------------------------------NOTE------------------------------
Refer to Table 3.3.6-1 to determine which SRs apply for each Containment Purge and Exhaust Isolation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.2	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.6.3	Perform MASTER RELAY TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.6.4	Perform COT.	92 days
SR 3.3.6.5	Perform SLAVE RELAY TEST.	18 months
SR 3.3.6.6	---------------NOTE--------------- Verification of setpoint is not required. ---------------------------------- Perform TADOT.	18 months
SR 3.3.6.7	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.6.8	Verify ESF RESPONSE TIME within limit.	18 months on a STAGGERED TEST BASIS

Table 3.3.6-1 (page 1 of 1)
Containment Purge and Exhaust Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	1,2,3,4, (a), (b)	2	SR 3.3.6.6	NA
2. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5 SR 3.3.6.8	NA
3. Containment Radiation Gaseous (R-24A, B)	1,2,3,4 (a), (b)	1 2	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7	$\leq 2.27 \times 10^{-2}$ μCi/cc (c)(d) $\leq 4.54 \times 10^{-3}$ μCi/cc (c)(e) $\leq 2.27 \times 10^{-3}$ μCi/cc (c)(f)
4. Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a., for all initiation functions and requirements.			

(a) During CORE ALTERATIONS.
(b) During movement of irradiated fuel assemblies within containment.
(c) Above background with no flow.
(d) With mini-purge in operation.
(e) With slow speed main purge in operation.
(f) With fast speed main purge in operation.

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

Choice C

a. The equipment hatch is capable of being closed and held in place by four bolts;

Choice B

b. One door in each air lock is capable of being closed; and

Choice D

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

Choice A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2	Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	18 months
SR 3.9.3.3	-------------------------------NOTE---------------------------- Only required for an open equipment hatch. --- Verify the capability to install the equipment hatch.	7 days

59. G2.1.17 001/NEW/RO/MEM 3.9/4.0/N/N/2/CVR/Y

Which one of the following demonstrates proper communication techniques that ensure information is transmitted and received effectively IAW ACP-1.0, Plant Communications?

A. The UO speaking to an extra SO: “Go check out the alpha charging pump and let me know if it is ready to start”.

SO: "Check out the alpha charging pump and let you know when it's ready to start".

UO: "That's correct".

B✓ The UO speaking to the Rover: “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. The SS speaking to both Control Room Operators: “Secure the one charlie reactor coolant pump".

OATC: "Secure the one charlie reactor coolant pump".

SS: "That's correct".

The UO obtains a peer check and secures the one charlie reactor coolant pump.

D. The SS speaking to the OATC: "Secure containment minipurge supply and exhaust fans".

OATC secures containment minipurge supply and exhaust fans, and then reports to the SS: "Containment minipurge supply and exhaust fans are secured".

SS: "Containment minipurge supply and exhaust fans are secured".

OATC: "That's correct".

A - Incorrect. The unit number is required by step 5.1.3 of ACP-1.0 to ensure the proper component is operated. This is especially important since an extra operator is being addressed, but would be required even for the Rad Side System Operator. Plausible, because the UO is assigned to a specific unit, and it may seem that it is obvious which unit is intended, but a unit is required to prevent wrong unit component manipulation events regardless of who is speaking or who is being addressed.

B - Correct. 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.

C - Incorrect. This communication must be "directed" using the person's name or title to ensure the appropriate person receives the instruction per step 5.1.4 "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." Plausible, since normally eye contact is made while speaking and is sufficient in other than instructional communication. It is a common misconception that if both people think they know who is being spoken to they don't need to direct the communication. However, whenever more than one person is in the area, to ensure information is transmitted and received effectively, ACP-1.0, step 5.1.4 requires a name or title to precede this communication. Also, the operator (OATC) who repeated the communication back must perform the action, or use another 3 way communication to the UO prior to the UO completing the action.

D - Incorrect. A repeat back must be obtained prior to the action being performed for instructional communications per step 5.1.5 & 5.1.6. Plausible, since no repeat back is obtained after informational communications per step 5.1.7. This sequence described is acceptable in the case of soliciting a meter reading when no action is required other than obtaining the reading and communicating the value; SS asks for a meter reading, prior to repeating back, the operator obtains and communicates the meter reading, SS acknowledges the meter reading, operator states: "that's correct".

ACP-1.0, Version 5.0

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

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).

Question # 59

K/A G2.1.17

REFERENCE Docs

5.0 General Communications Guidelines

5.1 This section addresses verbal communications that are instructional in nature. All plant personnel are encouraged to use these communication techniques at times other than required by this procedure so that these techniques become a regular habit.

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.

The following communication techniques are provided to ensure information is transmitted and received effectively.

5.1.1 Make verbal communications clear, complete and concise. Table 1 shows a suggested phonetic alphabet and is provided as an aid to enhance clear and concise communications.

5.1.2 Keep verbal instructions limited in scope. Break complicated instructions or multiple actions into simple steps containing single actions or communicate them in writing.

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.

Distractor A

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.

Distractor C

5.1.5 The sender of information is responsible for ensuring that the receiver of information gives a paraphrased repeat back. The sender is then to acknowledge the receiver by verifying the information is correct. This will result in the "three-way" communication loop being completed as expected.

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.

D distractor

plausibilty

The intent of this step is not to delay "Immediate Actions" until a repeat back has been made. Immediate Action steps should be performed when required, and then communicated to the appropriate individual.

5.1.7 Informational communication normally does not require acknowledgement. However, it is appropriate for the sender of information to solicit acknowledgement when it is deemed necessary.

5.1.8 The use of sign language is discouraged except in accepted industry practices such as crane operations, or in extremely high noise areas. Individuals using sign language should ensure understanding of the signs to be used prior to commencing the evolution.

5.1.9 Do not assume that other members of your team understand what you are doing or thinking. Proper communications among team members ensures that each person knows what the others are doing.

5.1.10 The use of profanity is not conducive to enhancing the quality of communication and is offensive to many people. Profanity degrades the professional atmosphere and therefore shall not be used.

5.2 Plant Public Address System (Gaitronics)

5.2.1 Minimize its use as the paging system to reduce background noise.

5.2.2 Channel 5 is designated for emergency use only.

5.2.3 In situations where Operations personnel deem it necessary that an immediate response be obtained, it is appropriate to announce over the public address system for an individual to pickup on line 5. This line's designation as emergency use only should imply to the person being paged that immediate response is expected.

5.2.4 The Public Address System may be used to inform or update plant personnel of the status of an abnormal or emergency condition, change in plant status, or a major plant event in progress or anticipated.

APPENDIX 1

These are examples of how the phonetic alaphabet should be used:

Stating a TPNS designation:

B, correct choice

Written: Q1E21V009C
Spoken: Q1E21V009 Charlie

Stating a breaker designation:

Written: EA08
Spoken: Echo Alpha 08

Stating a 7300 card designation:

Written: C-8 145
Spoken: Charlie 8 145

Stating a conduit designation:

Written: NHS 940
Spoken: NHS 940

Stating a work order status

Written: P1
Spoken: Papa 1

60. G2.1.45 001/NEW/RO/MEM 4.3/4.3/N/N/4/HBF/Y

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.

$\left(\frac{7\%}{1085psig}\right)(750psig) = 4.84\%$ resulting in a MW load of: $\left(\frac{901MW}{100\%}\right)(95.16\%) = 857MW$

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

$\left(\frac{3\%}{1035psig}\right)(750psig) = 2.17\%$ resulting in MW reduction of: $\left(\frac{901MW}{100\%}\right)(97.73\%) = 881MW$

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 [...].

Question # 60

K/A G2.1.45

REFERENCE Docs

outside the containment. These valves are Safety Class 2A and Category I Seismic.

The range of pressure settings of the safety valves on each line is in equal increments from 1075 psig to 1129 psig. The maximum actual capacity of a single valve fully open at 1085 psig is 890,000 lb/h.

10.3.2.2.4 Relief Valves

Installed on each main steam line upstream of the main steam isolation valves and downstream of the safety valve is one atmospheric relief valve. The valves are pneumatically actuated and sized to pass 405,500 lb/h of steam at 1025 psig. They are capable of going from fully closed to fully open within 35 seconds or less. The valves are also capable of being modulated over the pressure range of 1085 psig to 100 psig. Valve control is automatic by steam line pressure with remote manual control of the setpoint. A local manual operator is provided for valve operation in the event of complete loss of automatic control. An emergency source of control air is provided to enable remote manual operation.

10.3.3 EVALUATION

Following a sudden load rejection of up to 50 percent, the MSSS prevents a reactor trip by bypassing the steam directly to the condenser as described in subsection 10.4.4. Following a turbine trip or load rejection above 50 percent or when the turbine bypass system is not available, the MSSS effects a safe reactor trip by removing excessive heat from the reactor coolant through the exhausting of secondary steam through atmospheric power-operated relief valves and the spring-loaded safety valves. The power-operated relief valves and the spring-loaded safety valves also protect the steam generator and the main steam piping from overpressure.

In the unlikely event of a main steam line rupture, the isolation valves in the main steam lines provide steam line isolation, as described in subsection 5.5.5. The valving safety requirements are established to cover the following situations:

A. Break in the Steam Line From One Steam Generator Inside Containment

   The steam generator associated with the damaged line will discharge completely into the containment. Without reverse flow protection, the other steam downstream point in the steam line before the containment.

C. The portion of the MSSS up to and including the main steam isolation valves is necessary for the safe shutdown of the plant and is Safety Class 2A and Category I Seismic.

D. Uncontrolled steam release as a result of a steam line failure is limited to the contents of one steam generator in order to keep the related effect upon the reactor core within prescribed bounds.

E. The failure of any main steam line or malfunction of a valve installed therein will not:

   1. Render inoperable any engineered safety feature (ESF).

   2. Result in the containment pressure exceeding the design value or impairing its integrity.

Other safety-related design provisions include:

A. The steam generator safety valves.

B. The steam generator relief valves.

C. The steam supply to the turbine-driven auxiliary feedwater pump. The steam supply to this turbine has a safety classification because of the safety-related functions of the auxiliary feedwater system. The turbine steam supply lines are connected to two steam generator steam lines upstream of the steam line protective valving to provide both redundancy and dependability of supply. Isolation valves in each line maintain the separation of the main steam lines by preventing any interconnecting backflow.

D. Each steam generator includes an internal restriction which acts to limit the maximum flow and the resulting thrust forces created by a main steam line break.

10.3.1.3 Design Data

A.	Number of steam generators	3
B.	Total flow (lb/h)	12.26 E+06

Each valve fails closed on loss of electrical or air supply. They are capable of modulating over the pressure range of 100 to 1085 psig with a stroke time of 35 seconds or less. Each valve is sized to pass a total of approximately 405,500 lb/h of steam (10 percent of plant maximum calculated steam flow) at the no-load pressure of 1025 psig. Additionally, the maximum actual capacity of any single valve at an inlet steam pressure corresponding to the steam generator shell design pressure (1085 psig) shall not exceed 890,000 lb/h ste
to limit the steam release if any one v
open.

Discharge from the power-operated relie
atmosphere through penetrations in the
The discharge piping was analyzed for s
in addition to the normal operating con

During emergency conditions, the main s
relief valves provide a means to contro
generator pressures, and, or cool down
plant conditions these relief valves also give the plant flexibility of operation and the capability for a controlled cooldown. Isolation valves are provided upstream of each valve to allow maintenance. During a period when all other valves are out of service, the steam generator safety valves provide the necessary relieving capability.

10% of a single steam Generator's Max calculated flow.

3 SG max flow = 12.26 Mlb/hr
12.26/3 = 4.086 Mlbm/hr/SG

(.4055 Mlbm/hr) / (4.086 Mlbm/hr) =10%

therefore:

(.4055 Mlbm/hr) / (12.26 Mlbm/hr) =3%

Prior to shipment, each valve was hydrostatically tested in the manufacturer's facilities in accordance with the applicable code. Leakage was 40 cm^3/h in the valves that were tested. During plant operation, the valves are accessible for inspection. The operability of the alternate air supply system may be demonstrated during refueling shutdowns by using the alternate air supply to cycle open and closed each of the power-operated relief valves.

10.3.9 MAIN STEAM ISOLATION VALVES

The main steam isolation valves consist of two swing-disc check valves in each of the three main steam lines. These valves are located outside of the containment downstream of the main steam safety valves.

The main steam line isolation valves and their bypass valves are designed to stop forward flow and to isolate the steam generators and the main steam lines on signal initiated by engineered safety features actuation system under any of the following conditions:

NOTE:	**When the feedback loop is placed in/out of service Target and Demand are reset to equal Pimp. Thus any RAMP or HOLD in progress will be stopped. As a result the operator will have to reenter the desired Target and Rate. IMP PRESS LOOP should remain in service until >97% power to avoid control issues associated with #4 GV when initiating a ramp from near 95% (CR2007105465).**

5.20 WHEN >97% power AND the IMP PRESS LOOP is in service, THEN remove the feedback loop as follows:

5.20.1 Check Turbine on HOLD.

5.20.2 On the FEEDBACK STATUS DISPLAY, move the cursor to IMP PRESS LOOP IN.

5.20.3 Depress the SELECT key and verify IMP PRESS LOOP is highlighted in reverse video.

5.20.4 Depress the STOP key and verify the FEEDBACK STATUS indicates IMP PRESS OUT.

NOTE:

- **During steady state power operations the reactor core power level is limited to 2775 mega watts thermal. (NRC Regulatory Issue Summary 2007-21)**
- **Minor fluctuations of turbine/reactor power are expected due to grid fluctuations and normal tolerances in control systems. Unplanned increases in reactor power due to positive reactivity addition shall be promptly terminated with turbine load reduction, control rod insertion and/or boration. Reactor power shall be returned to at or below the previous steady state power level.**
- **DEH demand of 923 MW will cause #4 GV to go to full open position**

5.21 Increase power to ≤ 100%, ensuring TAVG does not exceed 577.2°F.

5.21.1 WHEN required, THEN take prompt action to compensate for excessive positive reactivity additions.

5.21.2 When available, the 15 minute average core thermal power (QC4621M15) should be monitored and maintained below 2775 MWth. If this value is exceeded, the OATC must reduce power to ensure the 1 hour average power remains below the limit of 2775 MWth.

5.21.3 In no case should 102% power be exceeded.

3.2 Turbine

3.2.1 Do not exceed the turbine generator loading rates specified by FNP-2-SOP-28.1, TURBINE GENERATOR OPERATION.

3.2.2 The following apply for condenser pressure conditions:

- WHEN the turbine is operating at ≥ 30% load, THEN the maximum permissible condenser pressure is 5.5 inches Hg. (2.7 psia).
- WHEN the turbine is operating at < 30% load, THEN the maximum permissible condenser pressure is 3.5 inches Hg. (1.7 psia).
- Refer to FNP-2-AOP-8.0, PARTIAL LOSS OF CONDENSER VACUUM, for remedial actions (Westinghouse Customer Advisory Letter 86-02.)

3.2.3 The following DEH valve position limits apply

- The DEH valve position limit shall be maintained at 115% while ramping or when at 100%.
- The DEH valve position limit should be maintained 8 to 10% above valve position demand when holding for more than 24 hours at some power level less than 100%.

3.2.4 The DEH system queries the valve LIMIT LOWER and RAISE keys in ~1 second intervals. Therefore when adjusting the valve position limit a wait of >~2 seconds should be allowed after the key is released before depressing the key again to ensure the DEH system recognizes the key had been released. {AI2002200059}.

3.2.5 Operate the MSR controls per FNP-2-SOP-28.1, TURBINE GENERATOR OPERATION, Precautions & Limitations.

3.2.6 When the Impulse Pressure feedback loop is placed in/out of service Target and Demand are reset to equal Pimp. Thus any RAMP or HOLD in progress will be stopped. As a result the operator will have to reenter the desired Target and Rate. IMP PRESS LOOP should remain in service until >97% power to avoid control issues associated with #4 GV when initiating a ramp from near 95% (CR2007105465)

A steam generator partition factor of 0.1 and a condenser air ejector partition factor of 10^{-4} have been used in the evaluation of environmental consequences of postulated accidents (e.g., Steam Generator Tube Rupture, subsection 15.4.3.4). These are conservative values for the range of water chemistry allowed by WCAP-7452 based on measurements made at operating Westinghouse plants.

10.3.6 INSTRUMENTATION APPLICATIONS

The steam flow restrictors installed in the steam generators are also used for steam flow measurements during normal operation. Two flow transmitters and two pressure transmitters are installed in the main steam line from each steam generator. The steam flow and pressure signals are fed into reactor protection and feedwater control system circuits to control the feedwater flow to each steam generator, to close the isolation valves in case of rupture in main steam lines, and to open the power-operated relief valves in case of overpressure.

10.3.7 MAIN STEAM SAFETY VALVES

Overpressure protection for the three steam generators is provided by the main steam safety valves. The design basis for the main steam safety valves is that they must have sufficient capacity so that main steam pressure does not exceed 110% of the steam generator shell-side design pressure. Based on this requirement, the valves are sized to relieve 105% of the maximum calculated steam flow at an accumulation pressure not exceeding 110% of the steam generator shell design pressure.

Design parameters for the main steam safety valves are given in table 10.3-1.

Due to the large mass flowrate, each steam generator is protected by a number of valves. The maximum actual capacity of a single valve fully open at 1085 psi gauge does not exceed 890,000 lb/h. This provision serves to limit steam release if any one valve inadvertently sticks open.

The main steam safety valves are located on the main steam lines outside the containment and upstream of the main steam isolation valves. Each of the three main steam lines is equipped with five safety valves. To prevent chattering during operation of the safety valves, each of the five valves on a steam line is set at a different set pressure. The first valve set pressure is 1075 psig, which corresponds to the steam generator shell design pressure minus the pressure loss from the steam generator to the valve. Each of the remaining valves is set at a higher pressure such that all valves are open and

61. G2.1.9 001/NEW/RO/MEM 2.9/4.5/N/N/2/CVR/Y

Which one of the following lists **ONLY** those personnel in which BOTH MUST 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).

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K/A G2.1.9

REFERENCE Docs

6.10.2.2 Control Room Supervision

The Shift Manager is the senior management representative on shift and has primary responsibility for the safe operation of the facility. The Shift Supervisor and Shift Manager maintain awareness by walking down control room boards on a frequency no less than once per shift.

Periodically, each Shift Supervisor should tour the plant areas outside of the Control Room, but should be able to respond to the control room promptly.

6.11 Main Control Room Access

6.11.1 Standard

Access to the main control room is managed so operators are not distracted from properly monitoring plant parameters.

6.11.2 Expectations

6.11.2.1 Access Protocol

Access to the control room area is limited to personnel requiring access for official business in order to avoid distractions to operators. The duty Shift Supervisor or Shift Manager is authorized to refuse entry, or direct individuals to leave, in order to maintain reactor control and minimize distractions. At the Controls Area (ATCA) and Control Room Boundaries are defined in local procedures.

Personnel who are exempt from requesting permission to enter the ATCA include:

Distractor A, B &C

- Site VP
- Plant Manager
- Operations Manager and Operations Superintendents
- On-duty shift operating crew, including the STA
- NRC Inspectors

D

All others wishing to enter the ATCA must obtain permission from a licensed operator on shift. Once an individual has obtained permission to enter the ATCA, that person may exit and enter without re-authorization during the shift as long as the purpose has not changed. The on shift control room crew compliment is granted access to the ATCA without permission.

The on-shift Shift Supervisor is responsible for maintaining the activities in the control room at an appropriate level. The Shift Supervisor, in conjunction with the control room staff, continually evaluates plant conditions, ongoing maintenance, testing, and the number of personnel in the control room to ensure the control room crew is able to maintain operational focus.

62. G2.2.3 001/FNP BANK/RO/MEM 3.8/3.9/N/N/2/CVR/Y

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

Question #	62
K/A	G2.2.3

REFERENCE Docs

1A 4160V BUS AB - 139' D177002

BKR	TPNS	DESCRIPTION	SEE PAGE
	N1R15A0001-N	**1A 4160V BUS**	
DA01	**N1R12A0502-N**	**1B UNIT AUX TRANSFORMER (NORMAL) <<<**	
DA02	N1R15BKRDA02	PT COMPARTMENT	
DA03	N1N21M0001A-N	1A CONDENSATE PUMP	
DA04	N1B41M0001A-N	1A REACTOR COOLANT PUMP	
DA05	N1P26M0001A-N	1A CIRC WATER PUMP	
DA06	**N1R11B0007-N**	**1I 4160/600V SST >>> EI02**	**A-2**
DA07	**N1R11A0501-N**	**1A STARTUP TRANSFORMER (ALTERNATE) <<<**	

condition of stem:
The RCP buses are
supplied from the SUT

1B SU AND 2A SU NO POWER.

1A BUS UNAFFECTED-- 1A RCP REMAINS RUNNING.

Correct Pt 1

1B 4160V BUS AB - 139' C177003

BKR	TPNS	DESCRIPTION
	N1R15A0002-N	**1B 4160V BUS**
DB01	**N1R12A0502-N**	**1B UNIT AUX TRANSFORMER (NORMAL) <<<**
DB02	N1R15BKRDB02	PT COMPARTMENT
DB03	N1B41M0001B-N	1B REACTOR COOLANT PUMP
DB04	N1P26M0001B-N	1B CIRC WATER PUMP
DB05	**N1R11A0502-N**	**1B START-UP TRANSFORMER (ALTERNATE) <<<**

condition of stem:
The RCP buses are
supplied from the SUT

1B SU AND 2A SU NO POWER.

1B BUS AFFECTED-- 1B RCP no power.

C&D distractors

1C 4160V BUS AB - 139' C177004

BKR	TPNS	DESCRIPTION	SEE PAGE
	N1R15A0003-N	1C 4160V BUS	
DC01	N1R11A0502-N	1B START-UP TRANSFORMER (NORMAL) <<<	
DC02	N1R15BKRDC02	PT COMPARTMENT	
DC03	N1B41M0001C-N	1C REACTOR COOLANT PUMP	
DC04	N1R12A0502-N	1B UNIT AUX TRANSFORMER (ALTERNATE) <<<	
DC05	NSR31G0501-N	1O 4160/600V SST >>> EO02	C-2
	N1R11G0510-N	1T 4160/600V SST >>> ET02	C-50

C&D distractors

2A 4160V BUS AB - 139' D-207002

BKR	TPNS	DESCRIPTION	SEE PAGE
	N2R15A0001-N	2A 4160V BUS	
DA01	N2R12A0502-N	2B UNIT AUX TRANSFORMER <<<	
DA02	N2R15BKRDA02	PT COMPARTMENT	
DA03	N2N21M0001A-N	2A CONDENSATE PUMP	
DA04	N2B41M0001A-N	2A REACTOR COOLANT PUMP	
DA05	N2P26M0001A-N	2A CIRC WATER PUMP	
DA06	N2R11B0007-N	2I 4160/600V SST >>> EI02 (NORMAL)	A-2
DA07	N2R11A0502-N	2B STARTUP TRANSFORMER <<<	

A&C choices

2B 4160V BUS **AB - 139'** **D-207003**

BKR	TPNS	DESCRIPTION
	N2R15A0002-N	**2B 4160V BUS**
DB01	**N2R12A0502-N**	**2B UNIT AUX TRANSFORMER <<<**
DB02	N2R15BKRDB02	PT COMPARTMENT
DB03	N2B41M0001B-N	2B REACTOR COOLANT PUMP
DB04	N2P26M0001B-N	2B CIRC WATER PUMP
DB05	**N2R11A0501-N**	**2A START-UP TRANSFORMER <<<**

B&D choices

2C 4160V BUS **AB - 139'** **D-207004**

BKR	TPNS	DESCRIPTION	SEE PAGE
	N2R15A0003-N	2C 4160V BUS	
DC01	N2R11A0501-N	2A START-UP TRANSFORMER <<<	
DC02	N2R15BKRDC02	PT COMPARTMENT	
DC03	N2B41M0001C-N	2C REACTOR COOLANT PUMP	
DC04	N2R12A0502-N	2B UNIT AUX TRANSFORMER <<<	
DC05	NSR11B0525-N	LOW LEVEL RADWASTE STORAGE FACILITY 4160-120/208V XFMR >>> FUSED DISC SW NSR18S0546-N >>> 1TT AC DIST CAB >>>	C-2

B&D choices

63. G2.2.36 001/NEW/RO/C/A 3.1/4.2/N/N/3/CVR/Y

Unit 1 is in Mode 6 with refueling 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)

Question # 63

K/A G2.2.36

REFERENCE Docs

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources — Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems — Shutdown"; and

b. One diesel generator (DG) capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	------------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. --	
	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
		(continued)

3.9 REFUELING OPERATIONS

3.9.4 Residual Heat Removal (RHR) and Coolant Circulation — High Water Level

LCO 3.9.4 One RHR loop shall be OPERABLE and in operation.

--NOTE--
The required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	AND	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	AND	

(continued)

64. G2.3.13 001/FNP BANK/RO/C/A 3.4/3.8/Y 2008/N/3/CVR/Y

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

Question # 64

K/A G2.3.13

REFERENCE Docs

OPERATOR ACTIONS

1. Check indications on radiation monitoring system console and determine which radiation monitor channel indicates high activity.
2. Insure that any automatic actions, associated with the alarmed channel, have occurred.
3. Perform the following general actions as appropriate.
   - 3.1. Determine the source or cause of the high activity and correct or isolate as required.
   - 3.2 Determine the validity of the high activity indication as follows:
     - 3.2.1 Verify that the instrument is aligned for normal operation and is functioning properly.
     - 3.2.2 IF a known problem exists such that the detector is saturated, THEN momentarily pull the affected detector's fuses (located on the front of the drawer) to clear the condition.
     - 3.2.3 If requested to disable a remote audible alarm, refer to FNP-1-SOP-45.0, P&L 3.6.
     - 3.2.4 Sample or survey the affected system or area as required. **{CMT 0008755}.**
   - 3.3 Do not allow personnel to enter the affected area without the approval of the Health Physics Department.
   - 3.4 IF high activity indication is due to instrument failure, THEN refer to Technical Specifications, section 3.3.3, 3.4.15 and TRM TR 13.3.4.
   - 3.5 IF high activity indication of RCS leakage is present AND accompanied by either decreasing pressurizer level, OR decreasing VCT level, THEN go to FNP-1-AOP-1.0, RCS LEAKAGE.
   - 3.6 IF high activity indication of Steam Generator Tube Leakage is present, THEN go to FNP-1-AOP-2.0, STEAM GENERATOR TUBE LEAKAGE.
   - 3.7. IF ARDA activated and not required, THEN have counting room stop the automated dose assessment per FNP-0-EIP-9.1, AUTOMATED DOSE ASSESSMENT METHOD.
   - 3.8 WHEN radiation levels have decreased below alarm setpoint, THEN reset the appropriate HI radiation alarm on the RAD monitor drawer.

LOCATION FH1

AUTOMATIC ACTIONS (cont)

2. ARDA will automatically start for the following conditions:

2.1 ARDA will automatically start when any of the following monitors go into alarm for two consecutive system polls one minute apart on either unit and use the latest 15 minute average monitor value to perform the calculations:

Plant Vent Stack Monitors R29 (SPING)	
Noble Gas	4.44e-4 □c/ml
Iodine	1.20e-6 □c/ml
Particulate	4.00e-5 □c/ml
Steam Jet air Ejector R15C	27 mr/hr
TDAFW Exhaust R60D	38 mr/hr
Steam Generator A R60A	38 mr/hr
Steam Generator B R60B	38 mr/hr
Steam Generator C R60C	38 mr/hr

2.2 ARDA will also automatically start when any of the following monitors go into alarm for two consecutive system polls one minute apart on either unit. The ARDA system will use the plant Vent stack SPING latest 15 minute average monitor value to perform the calculations when these monitors activate the system:

Plant Vent Stack Monitors	
Gas monitor R 14	13,000
Gas monitor R 21	1800
Particulate monitor R 22	156

R-11 not listed.. not in circuit to start ARDA.

OPERATOR ACTIONS (cont)

4. In addition to the general actions perform supplementary steps indicated in the "ACTIONS" column of the following table. (Some of the radiation monitors included in the table do not input into this alarm but are included for reference).

RADIATION MONITOR REFERENCE TABLE

RE	LOCATION	TYPE	DETECTOR	FUNCTION	ACTIONS
R-1A	Control Room (Unit I Panel)	Area	G-M (W)		Perform Step 4.1
R-1B	Technical Support Center (Unit II Panel) R-1B	Area	G-M (W)		No input to this alarm
R-2	Containment (155' elev)	Area	G-M (W)		Perform Steps 4.2
R-3	Radiochemistry Lab (AB 139')	Area	G-M (W)		Perform Step 4.3
R-4	#3 Charging Pump (AB 100')	Area	G-M (W)		Perform Step 4.4
R-5*	Spent Fuel Pool Room (AB 155')	Area	G-M (W)		Perform Steps 4.5
R-6	Sampling Room (AB 139')	Area	G-M (W)		Perform Step 4.6
R-7	In-core NIS Area (CTMT 129', near Seal Table)	Area	G-M (W)		Perform Steps 4.7
R-8	Drumming Station (AB 155')	Area	G-M (W)		Perform Step 4.8
R-9	SG Sample Panel (Unit II Panel) (AB 139')	Area	G-M (W)		No input to this alarm
R-10	Penetration Room Filtration Discharge (AB 155')	APD	Scint. (Victoreen)		Perform Step 4.10
R-11*	Containment Atmosphere (AB 121')	APD	Scint. (Victoreen)		Perform Step 4.11

*Technical Specification related

OPERATOR ACTION (cont)

4.8 IF **R-8** in alarm THEN perform the following:
 4.8.1 Announce the affected area on the public address system.
 4.8.2 Have all personnel evacuate the affected area.

4.9 Step not used

4.10 IF **R-10** alarms and high activity in the penetrations rooms is possible, THEN consider placing penetration room filtration in service using FNP-1-SOP-60 PENETRATION ROOM FILTRATION SYSTEM.

A&B part #1

A&C part #2

4.11 IF **R-11** alarms, THEN perform the following:
 4.11.1 IF personnel are in containment and unaware of the high activity, THEN announce the affected area on the public address system.
 4.11.2 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.
 4.11.3 IF RCS leakage is possible then perform actions of FNP-1-AOP-1.0, RCS LEAKAGE

4.12 IF **R-12** alarms, THEN perform the following:
 4.12.1 IF personnel are in containment and unaware of the high activity, THEN announce the affected area on the public address system.
 4.12.2 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.
 4.12.3 IF RCS leakage is possible then perform actions of FNP-1-AOP-1.0, RCS LEAKAGE

4.13 IF **R-13** alarms, THEN refer to FNP-1-SOP-51, WASTE GAS SYSTEM for potential problems with the waste gas system.

4.14 IF **R-14** alarms, THEN perform the following:
 4.14.1 IF dry storage operations are in progress, THEN perform the following, as appropriate:
  4.14.1.1 IF dry storage personnel have notified OPS that the R-14 alarm is possible due to dry storage operations, THEN regard as an expected alarm.
  4.14.1.2 Contact dry storage personnel AND determine if dry storage operations probably caused the R-14 alarm.

LOCATION FH4

SETPOINT: Variable, as per FNP-1-RCP-252

ORIGIN: Radiation Monitor Cabinet Channels R-24A or R-24B Containment Purge

H4
CP
RE24 A OR B
HI RAD

PROBABLE CAUSE

1. High Radiation Level in the Containment Purge Exhaust Line.
2. 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.

B&D Part #2

AUTOMATIC ACTION

1. Isolates Containment by closing Purge Supply and Exhaust Valves 1-CP-HV-3196, 1-CP-HV-3197, 1-CP-HV-3198A, B, C, & D, 1-CP-HV-2867C & D and 1-CP-HV-2866C & D.

OPERATOR

1. Determine which radiation monitor indicates high activity.
2. Verify that any required automatic actions have occurred and if required, secure any running containment purge or mini-purge fans.
3. Notify HP personnel of alarm.
4. Implement FNP-0-EIP-9, EMERGENCY CLASSIFICATION AND ACTIONS.
5. Determine the validity of the high activity indication as follows:
   - 5.1 Verify that the instrument is aligned for normal operation and is functioning properly.
   - 5.2 Sample or survey the affected system or area as required.
6. Determine the source or cause of the high activity and correct or isolate as required.
7. DO NOT allow personnel to enter the affected area without the approval of the Health Physics Department.
8. IF high activity indication is due to instrument failure, THEN refer to Technical Specifications, section 3.3.6.
9. IF high activity indication of RCS leakage is present AND accompanied by either decreasing pressurizer level OR decreasing VCT level, THEN go to FNP-1-AOP-1.0, RCS LEAKAGE.

References: A-177100, Sh. 309; U-258400; D-181658; D-181671; D-177199; D-177204; FSAR, Section 11.4; D-175010, Sh. 2.

A. Purpose

This procedure provides actions for response to a minor RCS leak which is within the capacity of the normal Charging and Makeup System to maintain pressurizer level, in order to permit a controlled Reactor shutdown and RCS cooldown to cold shutdown.

For RCS breaks 3/8 of an inch or less, A single charging pump with letdown isolated can maintain the pressurizer level at the reactor coolant system pressure, and an ECCS actuation is not required.(FSAR 6.3.3.3 and 9.3.4.1.1.8) A 3/8 inch liquid space break is approximately a 130 gpm leak at normal operating pressure (2235 psig).

This procedure is applicable in Modes 1, 2 and 3.

B. Symptoms or Entry Conditions

1. This procedure is entered when excessive RCS leakage is indicated by any of the following:

   a. Unexplained rise in charging flow

   b. Unexplained reduction in VCT level

   c. Results of FNP-1-STP-9.0, RCS LEAKAGE TEST

   d. As directed from the following annunciator response procedures.

      [ ] FNP-1-ARP-1.5 Annunciator EA2, "CHG HDR FLO HI LO"
      [ ] FNP-1-ARP-1.8 Annunciator HB2, "PRZR LVL DEV LO"
      [ ] FNP-1-ARP-1.8 Annunciator HD1, "PRZR PRESS REL VLV 445A OR B/U HTRS ON"
      [ ] FNP-1-ARP-1.8 Annunciator HC1 "PRZR PRESS HI/LO"

Step	Action/Expected Response	Response Not Obtained

NOTE:
- Charging flow is limited to approximately 130 gpm when FK 122 is operated in automatic.
- The intent of step 1 is to ensure that SI is actuated if PRZR level cannot be maintained stable above the low level heater interlock setpoint. If possible, pressurizer level should be restored to the normal program value. A stable pressurizer level permits a controlled orderly shutdown and cooldown to cold shutdown.

Step	Action/Expected Response	Response Not Obtained
__ 1	**Maintain pressurizer level stable at or near programmed level.**	1 Perform the following.
	[ ] Control charging flow	1.1 Verify reactor tripped.
	OR	
	[ ] Reduce letdown flow	1.2 IF reactor tripped, THEN actuate SI.
	OR	
	[ ] IF required to maintain PRZR level, THEN isolate letdown.	1.3 Go to FNP-1-EEP-0, REACTOR TRIP OR SAFETY INJECTION.

__Page Completed

Step	Action/Expected Response	Response Not Obtained

****

CAUTION: IF VCT level indication is lost due to a low level, OR IF it is suspected that the lower level tap has been uncovered, THEN the VCT level transmitters will need to be vented for reliable level indication.{2006203596}

****

__ 2 **Maintain VCT level greater than 20%.**

2.1 Verify reactor makeup system - IN AUTOMATIC.

OR

2.2 Control reactor makeup system in manual using FNP-1-SOP-2.3, CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM.

2 Perform the following.

a) Verify charging pump suction aligned to RWST.

RWST
TO CHG PUMP
[ ] Q1E21LCV115B open
[ ] Q1E21LCV115D open

VCT
OUTLET ISO
[ ] Q1E21LCV115C closed
[ ] Q1E21LCV115E closed

b) Begin unit shutdown using FNP-1-UOP-3.1, POWER OPERATION and FNP-1-UOP-2.1, SHUTDOWN OF UNIT FROM MINIMUM LOAD TO HOT STANDBY.

__Page Completed

Step	Action/Expected Response	Response Not Obtained
__ 3	**Determine RCS leak rate.**	
3.1	Determine RCS leak rate from CVCS flow balance. ___________(charging flow) +___________(seal injection flow) -___________(letdown flow) -___________(#1 seal leakoff flow) =___________(RCS leak rate)	
3.2	<u>IF</u> plant conditions are stable, <u>THEN</u> determine RCS leak rate using FNP-1-STP-9.0, RCS LEAKAGE TEST.	
__ 4	**<u>WHEN</u> RCS leak rate determined, <u>THEN</u> evaluate required actions using Technical Specifications.**	
__ 5	**<u>WHEN</u> RCS leak rate determined, <u>THEN</u> evaluate event classifiation and notification requirements using FNP-0-EIP-8, NON-EMERGENCY NOTIFICATIONS and FNP-0-EIP-9, EMERGENCY CLASSIFICATION <u>AND</u> ACTIONS.**	
__ 6	**<u>WHEN</u> RCS leak rate greater than 50 gpm, <u>THEN</u> align 1A and 1B post LOCA containment hydrogen analyzers for service using ATTACHMENT 1.**	

__Page Completed

Step | Action/Expected Response | Response Not Obtained

****

CAUTION: Since abnormal conditions may exist, nonessential personnel should not be permitted in containment until RCS leakage has been located.

****

NOTE:

- The intent of step 7 is to provide a systematic leakage search plan. Steps 7.2 through 7.12 may be done in any order.
- IF at anytime the location of an RCS leak is discovered or reported, THEN actions to isolate the leak should be taken immediately.
- WHEN all leakage sources have been identified, THEN continue with step 8 and further leakage identification actions may be terminated.

__ 7 **Identify RCS leakage source.**

7.1 Frequently monitor CVCS flow balance as the actions of steps 7.2 through 7.12 are taken.

___________(charging flow)

+___________(seal injection flow)

-___________(letdown flow)

-___________(#1 seal leakoff flow)

=___________(RCS leak rate)

7.2 Check containment radiation - NORMAL.

[ ] R-2 CTMT 155 ft
[ ] R-7 SEAL TABLE
[ ] R-11 CTMT PARTICULATE
[ ] R-12 CTMT GAS

7.2 Perform the following.

7.2.1 Consult Shift Manager to evaluate requirement for containment entry.

7.2.2 Evaluate placing CTMT sump pump handswitches in PULL-TO-LOCK to prevent overfilling the WHT.

Step 7 continued on next page

__Page Completed

Step	Action/Expected Response	Response Not Obtained
		7.2.3 Verify containment ventilation isolation.
		7.2.3.1 Stop MINI PURGE SUPP/EXH FAN.
		7.2.3.2 Verify containment mini purge dampers - CLOSED. CTMT PURGE DMPRS MINI-2866C & 2867C FULL-3198A & 3198D [ ] 2866C [ ] 2867C CTMT PURGE DMPRS MINI-2866D & 2867D FULL-3196 & 3197 BOTH-3198B & 3198C [ ] 2866D [ ] 2867D
7.3	Check auxiliary building radiation - NORMAL. [ ] R-4 1C CHG PUMP RM [ ] R-6 SAMPLE RM AREA [ ] R-10 PRF	7.3 Perform the following. 7.3.1 Announce the hazard area using the Gaitronics System. 7.3.2 Evacuate the hazard area of non-essential personnel. 7.3.3 Dispatch Health Physics and Operations personnel to visually inspect accessible portions of charging, letdown, BTRS and seal injection systems using ATTACHMENT 3, LEAK INSPECTION – 121' Elevation and Above and ATTACHMENT 4, LEAK INSPECTION - 100' Elevation and Below.
7.4	Check no SG tube leakage.	7.4 Go to FNP-1-AOP-2.0, STEAM GENERATOR TUBE LEAKAGE.

Step 7 continued on next page

__Page Completed

65. G2.3.15 001/NEW/RO/MEM 2.9/3.1/N/N/3/GTO/CVR/Y

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.

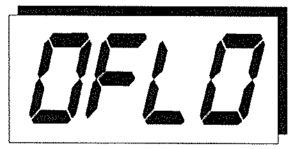
Objective: OPS-30401A G2.3.5 and 2.3.15 objectives

Question # 65

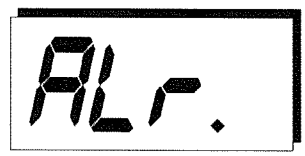
K/A G2.3.15

REFERENCE Docs

4.4.3 ***Overflow alarm:*** If the displayed count rate is over 999E (999,000 cpm), the **OFLO** LCD's blinks on the display.



4.4.4 ***Threshold alarm***: If the reading exceeds threshold value, the **ALr.** LCD's and the reading are displayed alternately, accompanied by an audible beep.



Pressing the SPEAKER/→ push-button mutes the audible alarm, but the **ALr.** LCD's and the reading are continuing to be displayed alternately, until the reading decreases to 0.75 from threshold value. In case the reading exceeds threshold value and then decreases below 0.75 of threshold value, the **ALr.** LCD's and the beep sound are automatically cancelled, even though the SPEAKER/→ push-button has not been pressed.

4.5 Instrument is operational from -10°C to +50°C. (15°F to +122°F)

5.0 Operation and Response Check

The instrument must be response checked daily or prior to use whichever is less frequent.

A&D part #2

5.1 Ensure the instrument calibration is current as indicated by the calibration sticker.

5.2 Turn the instrument on by depressing the ON/OFF button.

5.3 Battery voltage is checked internally if battery voltage is low, the LCD will blink and the display will show "bAt". "bAt" alarm must be cleared out prior to use.

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.

B & C part 2

Correct part 1

5.6 If the detector is set to alarm ensure the alarm sounds at the value indicated on the Set Point Sticker on the meter.

5.7 Clicks should be heard from the speaker during upscale readings.

NOTE: Meter fluctuation is normal and is caused by the random nature of radioactive decay.

NOTE: When using a low range GM tube in very high exposure fields (e.g. >2 R/hr), the instrument may saturate. The display will show OFLO.

5.8 Push the "RESET/Mode" button and release. The reading should drop rapidly then climb back to the source reading.

5.9 If the instrument responds properly, HP Form 224 will normally be completed IAW FNP-0-RCP-201.

5.10 If the instrument does not respond properly, then remove the instrument from service IAW FNP-0-RCP-201.

5.11 The meter reading must be multiplied by appropriate correction factors to obtain the proper field intensity.

NOTE: The batteries should be removed from the instrument if it is to be totally inactive for a long period of time (e.g. in storage >6 months).

5.12 After use, turn off the instrument prior to storage.

6.0 Calibration

6.1 Off-site calibration

The instrument may be sent off-site to a vendor for calibration in accordance with FNP-0-RCP-201.

6.2 On-site calibration

Calibration of the RAM-100 is in two basic units. The base unit which includes the rate meter and display, and the detector unit which includes the amplifier and high voltage assembly

CAUTION: Damage to the instrument and/or injury to personnel can occur by touching the electrical components.

6.2.1 Base Unit

6.2.1.1 Disconnect the detector from the base unit.

6.2.1.2 Connect the base unit to a pulser via the pulser adaptor.

6.2.1.3 Turn on the meter.

6.2.1.4 Turn on the pulser.

6.2.1.5 Set the input amplitude to 2.7 ± 10% volts.

6.2.1.6 Set the input Frequency to 400 cpm press the reset button on the base unit and obtain the meter reading.

6.2.1.7 Repeat for Frequency setting of 4,000 cpm and 40,000 cpm.

6.2.1.8 If the meter readings are within ± 10% of the input frequency record the data on HP Form 240 and proceed to step 6.2.2.

6.2.1.9 If instrument does not calibrate to within ± 10% of the reference points then remove the instrument from service in accordance with FNP-0-RCP-201.

6.2.2 Detector Calibration

6.2.2.1 Open the detector housing by removing the 3 phillips screws.

6.2.2.2 Connect the detector to the meter.

6.2.2.3 Connect a high voltage detector to the D1 cathode (See Figure 1).

6.2.2.4 Turn on the meter.

6.2.2.5 Measure the high voltage to the detector as the D1 cathode record "As Found" data on HP Form 240.

6.2.2.6 Expose detector to source 0515.00.00, record "As Found" data on HP Form 240.

6.2.2.7 If the meter reading is within ± 10% of the reference source record the "As Left" data on HP Form 240 and proceed to step 6.2.2.10. If the meter reading is NOT within ± 10% of the reference source, proceed to step 6.2.2.8.

6.2.2.8 Enter calibration mode by holding the RESET/MODE push button for 2 seconds. Then scroll using the RESET/Mode push button until the blinking display shows "CAL".

6.2.2.9 Adjust the meter reading to match the calibration source by turning R12 Calibration Factor Adjustment (see Figure 1).

6.2.2.10 If the high voltage is 875 volts ± 5v then record this voltage in the "As Left" section on HP Form 240 and proceed to 6.2.2.13.

6.2.2.11 If the high voltage is out of tolerance then adjust high voltage to 875 volts ± 5v by turning R14 (See Figure 1) and record "As Left" data on HP Form 240.

6.2.2.12 Expose detector to source 0515.00.00 and record "As Left" data on HP Form 240.

6.2.2.13 Turn the meter off.

6.2.2.14 Place the detector back into the housing and tighten the 3 phillips screws.

6.2.2.15 If instrument does not calibrate to within ± 10% of the calibration reference source then remove the instrument from service IAW FNP-0-RCP-201.

6.2.3 Affix a correction factor sticker with a correction factor of 10 to instruments using pancake type GM probes.

6.2.4 Affix a "Daily Response Check" sticker, HP Form 224 to the instrument.

6.2.5 Affix calibration sticker to instrument.

66. G2.3.4 001/NEW/RO/MEM 3.2/3.7/N/N/3/CVR/Y

Which one of the following correctly states the dose limits that Radiation Worker qualified personnel may be **required** to receive in an emergency IAW EIP-14.0, Personnel Movement, Relocation, Re-entry and Site Evacuation,

1) to protect valuable property

and

2) to save a life?

	(1)	(2)
A.	5 Rem	25 Rem
B.✓	10 Rem	25 Rem
C.	5 Rem	100 Rem
D.	10 Rem	100 Rem

A - Incorrect. The first part is incorrect, but plausible, since it is the 10 CFR 20 annual limit for TEDE, which is more than twice the 2 REM FNP Admin limit for normal annual dose. However, in an emergency, a radiation worker qualified individual may be required to receive up to 10 Rem to protect valuable property. The second part is correct.

B - Correct. Both parts are correct.

C - Incorrect. The first part is incorrect (see A). The second part is incorrect, but plausible, since it is the limit that a radiation worker can "voluntarily" receive, but cannot be required to receive over 25 Rem.

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

EIP-14 v23
7.12 Emergency situations may transcend the normal requirement of maintaining personnel exposures below 10CFR20 limits, as noted in step 7.10. Emergency exposures shall be minimized to every degree practicable. Farley Nuclear Plant personnel who have completed the onsite radiation protection training may be required to receive an exposure up to 25 rem TEDE for the activity and conditions described below. For those same personnel to receive in excess of 25 rem, they must voluntarily agree to receive an emergency dose in excess of 25 rem, but less than 100 rem. Emergency doses received do not have to take into account the annual dose to date. Persons volunteering to receive in excess of 25 rem must be made fully aware of the risks involved. Emergency exposure limits are as follows:

TEDE	ACTIVITY	CONDITION DOSE
10 REM	PROTECTING VALUABLE PROPERTY	LOWER DOSE NOT PRACTICAL
25 REM	LIFE SAVING OR PROTECTION OF LARGE POPULATIONS	LOWER DOSE NOT PRACTICAL
>25, <100 REM	LIFE SAVING OR PROTECTION OF LARGE POPULATIONS	VOLUNTEERS ONLY THAT ARE FULLY AWARE OF THE RISKS INVOLVED

Limit the dose to the lens of the eyes to 3 times the listed value. Limit the dose to other organs, including skin and extremities to 10 times the listed values.

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).

Question #	66
K/A	G2.3.4

REFERENCE Docs

7.10 Farley Nuclear Plant personnel who have completed the onsite radiation protection training may be required to receive an exposure up to the following 10CFR20 limits:

	10CFR20 limit	Administrative limit
Whole body (TEDE)	- 5 rem	- 2 rem
Lens of the eyes	- 15 rem	- 6 rem
Skin of the whole body	- 50 rem	- 20 rem
Extremities	- 50 rem	- 20 rem
Internal organs	- 50 rem	- 20 rem

7.11 Dosimetry records for potential re-entry team members are available in the Dosimetry Lab.

CAUTION: EMERGENCY EXPOSURE LIMITS SHALL ONLY BE AUTHORIZED BY THE E.D.

7.12 Emergency situations may transcend the normal requirement of maintaining personnel exposures below 10CFR20 limits, as noted in step 7.10. Emergency exposures shall be minimized to every degree practicable. Farley Nuclear Plant personnel who have completed the onsite radiation protection training may be required to receive an exposure up to 25 rem TEDE for the activity and conditions described below. For those same personnel to receive in excess of 25 rem, they must voluntarily agree to receive an emergency dose in excess of 25 rem, but less than 100 rem. Emergency doses received do not have to take into account the annual dose to date. Persons volunteering to receive in excess of 25 rem must be made fully aware of the risks involved. Emergency exposure limits are as follows:

TEDE DOSE	ACTIVITY	CONDITION
10 REM	PROTECTING VALUABLE PROPERTY	LOWER DOSE NOT PRACTICAL
25 REM	LIFE SAVING OR PROTECTION OF LARGE POPULATIONS	LOWER DOSE NOT PRACTICAL
>25, <100 REM	LIFE SAVING OR PROTECTION OF LARGE POPULATIONS	VOLUNTEERS ONLY THAT ARE FULLY AWARE OF THE RISKS INVOLVED
Limit the dose to the lens of the eyes to 3 times the listed value. Limit the dose to other organs, including skin and extremities to 10 times the listed values.		

67. G2.4.13 001/MOD/RO/MEM 4.4/4.6/N/N/3/CVR/Y

Which one of the following states the **MINIMUM** authority by position to **approve** a 6 REM exposure during a Site Area Emergency?

A. The HP Supervisor ONLY.

B✓ The Emergency Director ONLY.

C. Either the HP Supervisor OR Emergency Director.

D. Both the HP Supervisor AND the Emergency Director.

A - Incorrect. ED only must approve exceeding 10CFR20 radiation exposure limits listed in step 7.10 of EIP-14. Plausible, since the HP Supervisor may authorize exceeding FNP Admin radiation limits in an emergency.

B - Correct. This is required by EIP-14, Version 23.0, step 7.8.

C - Incorrect. ED only must approve exceeding 10CFR20 radiation exposure limits listed in step 7.10 of EIP-14. Plausible, since the HP Supervisor, OR Emergency Director in the HP Supervisor's absence, may authorize exceeding FNP Admin radiation limits in an emergency.

D - Incorrect. Plausible, since the HP Supervisor, OR Emergency Director in the HP Supervisor's absence, may authorize exceeding FNP Admin radiation limits in an emergency. It would be reasonable to assume that the HP Supervisor would be involved in and possibly required to approve the decision to exceed any radiation dose limit, and with the ED approve exceeding the 10CFR20 limit. However, the ED is soley responsible for authorizing exceeding this limit.

EIP-14.0 Version 23.0

7.8 Re-Entry is the responsibility of the Emergency Director, and requires verbal ED approval to execute a re-entry. Re-entries may be authorized and executed by the OSC Manager or Maintenance Supervisor, with ED approval. Approval to exceed 10CFR20 radiation exposure limits listed in step 7.10 must be approved by the Emergency Director. Approval to exceed plant administrative dose limits listed in step 7.10 must be approved by the HP Supervisor, or the Emergency Director in the HP Supervisor's absence.

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:

Objective:

6. **LIST AND IDENTIFY** the responsibilities of individual using a procedure (OPS-40504A07).

Question # 67

K/A G2.4.13

REFERENCE Docs

7.5 The re-entry guideline/log (Figures 3/4) will serve as a tracking mechanism for re-entries. One copy of the guideline will remain with the OSC and, if desired, another copy will be given to the re-entry team leader. The guideline may be photocopied, or a two-part form may be used. The re-entry guideline will be sequentially numbered.

7.6 Individuals listed on the re-entry guideline as responsible for completion of guideline items are not required to personally initial the guideline, but are responsible for ensuring that each requirement is performed and initialed by the person performing or ensuring performance of the task.

7.7 Radiological monitoring will be established for each re-entry. The following parameters will be considered when determining the degree of radiological monitoring:

- Releases in progress
- Dose rates, airborne and contamination levels
- Stability of plant radiological conditions

7.8 Re-Entry is the responsibility of the Emergency Director, and requires verbal ED approval to execute a re-entry. Re-entries may be authorized and executed by the OSC Manager or Maintenance Supervisor, with ED approval. Approval to exceed 10CFR20 radiation exposure limits listed in step 7.10 must be approved by the Emergency Director. Approval to exceed plant administrative dose limits listed in step 7.10 must be approved by the HP Supervisor, or the Emergency Director in the HP Supervisor's absence.

7.9 An Emergency Repair Party which functions as a re-entry team shall consist of at least two (2) persons.

7.10 Farley Nuclear Plant personnel who have completed the onsite radiation protection training may be required to receive an exposure up to the following 10CFR20 limits:

	10CFR20 limit	Administrative limit
Whole body (TEDE)	- 5 rem	- 2 rem
Lens of the eyes	- 15 rem	- 6 rem
Skin of the whole body	- 50 rem	- 20 rem
Extremities	- 50 rem	- 20 rem
Internal organs	- 50 rem	- 20 rem

7.11 Dosimetry records for potential re-entry team members are available in the Dosimetry Lab.

CAUTION: EMERGENCY EXPOSURE LIMITS SHALL ONLY BE AUTHORIZED BY THE E.D.

7.12 Emergency situations may transcend the normal requirement of maintaining personnel exposures below 10CFR20 limits, as noted in step 7.10. Emergency exposures shall be minimized to every degree practicable. Farley Nuclear Plant personnel who have completed the onsite radiation protection training may be required to receive an exposure up to 25 rem TEDE for the activity and conditions described below. For those same personnel to receive in excess of 25 rem, they must voluntarily agree to receive an emergency dose in excess of 25 rem, but less than 100 rem. Emergency doses received do not have to take into account the annual dose to date. Persons volunteering to receive in excess of 25 rem must be made fully aware of the risks involved. Emergency exposure limits are as follows:

TEDE DOSE	ACTIVITY	CONDITION
10 REM	PROTECTING VALUABLE PROPERTY	LOWER DOSE NOT PRACTICAL
25 REM	LIFE SAVING OR PROTECTION OF LARGE POPULATIONS	LOWER DOSE NOT PRACTICAL
>25, <100 REM	LIFE SAVING OR PROTECTION OF LARGE POPULATIONS	VOLUNTEERS ONLY THAT ARE FULLY AWARE OF THE RISKS INVOLVED
Limit the dose to the lens of the eyes to 3 times the listed value. Limit the dose to other organs, including skin and extremities to 10 times the listed values.		

1. EPIP PERS-40501B09 011/HLT/LOCT/RO/C/A (LEVEL 2/3) PROC/G2.4.13///LOCT/

Which of the following can authorize exposures exceeding 10CFR20 limits, during a Site Area Emergency?

BANK

A. HP Manager.

B.✓ Emergency Director.

C. TSC Manager AND HP Manager.

D. HP Manager AND Emergency Director.

EIP-14.0 Version 23.0

7.8 Re-Entry is the responsibility of the Emergency Director, and requires verbal ED approval to execute a re-entry. Re-entries may be authorized and executed by the OSC Manager or Maintenance Manager, with ED approval. **Approval to exceed 10CFR20 radiation exposure limits listed in step 7.10 must be approved by the Emergency Director**. Approval to exceed plant administrative dose limits listed in step 7.10 must be approved by the HP Manager, or the Emergency Director in the HP Manager's absence.

A. Incorrect - per the above

B. Correct - per the above

C. Incorrect - per the above

D. Incorrect - per the above

68. G2.4.16 001/MOD/RO/C/A 3.5/4.4/N/N/3/CVR/Y

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.**
- Core Exit Thermocouples (CETCs) read 1200°F and 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 directed to transition to SAMGs (Severe Accident Management Guidelines).

C. Immediately transition to SAMGs (Severe Accident Management Guidelines) 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.

A - Incorrect. Remaining in ECP-0.0 until power is restored is incorrect, since transition to SAMGs (at step 23) would be required prior to energizing electrical busses, if they were not energized by then. Plausible, since the FRPs are not entered with loss of all AC per SOP-0.8, Emergency Response Procedure User's Guide, and are monitored for information only until power is restored. However, the SAMGs are entered to protect the containment barrier after assuming that the clad barrier is lost (CETCs >1200°F) even with loss of all AC.

B - Correct. Step 23 of ECP-0.0 will direct entry into the SAMG network, but entry into SAMGs is not made until ECP-0.0 has first accomplished steps which minimize DC, isolate RCS seals, dump accumulators to help cool the core, and several other things.

C - Incorrect. Step 23 of ECP-0.0 will direct entry into the SAMG network, and entry is not made until directed. Plausible, since in the ERG network, in most cases, and entry condition for FRPs would direct immediate entry due to the hierarchy of procedure usage. Examinee may correctly realize that the FRPs are not entered with loss of power, and the SAMGs are entered with loss of power, but incorrectly apply the immediate entry criteria for the SAMGs.

D - Incorrect. The FRPs are not entered at all with a total loss of vital AC power, but this is plausible, since in the other ERPs (without loss of power), this choice would be correct. The entry condition for this procedure is met with CETCs >1200°F.

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)

Question # 68

K/A G2.4.16

REFERENCE Docs

4.0 Critical Safety Function Status Trees (CSFSTs)

4.1 General

The ERP network is designed to protect the health and safety of the public by maintaining the fission product barriers intact. In the initial stage of ERP performance, if AC power is available, the user ensures that the automatic plant systems are functioning properly to protect these barriers. Afterwards, the CSFSTs are monitored to detect challenges to the barriers due to worsening plant conditions or equipment failure and to direct the user to an appropriate procedure.

4.2 Applicability

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.

4.3 Proper Use

The CSFSTs follow a logic tree format. Each CSFST has a single entry point at the left side of the page. When manually monitoring the CSFSTs, the user must enter at this point and then proceed to the right until reaching an endpoint. The endpoint will either indicate that the particular CSF is satisfied or direct the user to an appropriate procedure. The Safety Parameter Display System (SPDS) provides real time monitoring of the CSFSTs. The user should perform CSF-0 to determine if SPDS is functioning properly. If so, it should be used. If not, manual monitoring is required.

Step	Action/Expected Response	Response NOT Obtained
___21	**Check PHASE B CTMT ISO not required.**	
21.1	Check containment pressure - HAS REMAINED LESS THAN 27 psig. [] MLB-3 1-1 not lit [] MLB-3 6-1 not lit	21.1 Verify PHASE B CTMT ISO. 21.1.1 Verify PHASE B CTMT ISO actuated. 21.1.2 Verify PHASE B CTMT ISO alignment. CCW FROM RCP THRM BARR [] Q1P17HV3045 closed [] Q1P17HV3184 closed CCW FROM RCP OIL CLRS [] Q1P17MOV3182 closed [] Q1P17MOV3046 closed CCW TO RCP CLRS [] Q1P17MOV3052 closed IA TO CTMT [] Q1P19HV3611 closed (BOP) 21.1.3 Reset containment spray signal. CS RESET [] A TRN [] B TRN
___22	**[CA] Locally monitor spent fuel pool level. (155 ft, AUX BLDG spent fuel room)**	
22.1	Check spent fuel pool level - GREATER THAN 153 ft.	22.1 Consult TSC staff to determine spent fuel pool makeup requirements.
___23	**Check core exit T/Cs - LESS THAN 1200°F.**	23 <u>IF</u> fifth hottest core exit T/C greater than 1200°F <u>AND</u> rising, <u>THEN</u> go to FNP-1-SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE.

___Page Completed

Step	Action/Expected Response		Response NOT Obtained

****

CAUTION: Bus failure could result from starting loads in excess of the capacity of the power source.

****

Step	Action/Expected Response		Response NOT Obtained
___24	**Check at least one train of 4160 V ESF busses - ENERGIZED.** [] A Train (F & K) power available lights lit [] B Train (G & L) power available lights lit	24	Return to Step 11. OBSERVE CAUTION PRIOR TO STEP 11. STAY in ECP-0.0 until >1200 or power is restored.
___25	**Verify SW system operating.**		
25.1	Verify at least one SW train - HAS TWO SW PUMPs RUNNING. [] A Train (1A,1B or 1C) [] B Train (1D,1E or 1C)		
25.2	Verify SW flow through at least one train of containment coolers - GREATER THAN 0 gpm. SW THROUGH CTMT CLRS INLET [] FI 3013A [] FI 3013B		
25.3	Verify SW to DG BLDG valves - OPEN. SW TO/FROM DG BLDG - A HDR [] Q1P16V519/537 SW TO/FROM DG BLDG - B HDR [] Q1P16V518/536		

___Page Completed

1. EC-0.0/.1.2-52532A08 015/HLT/LOCT/RO/C/A (LEVEL 2/3) PROC/G2.4.16///LOCT/ bank

Unit 1 was operating at 100% power when a total loss of offsite anc
electrical power occurred. Given the following events and conditions:

- The crew is performing the actions of ECP-0.0, LOSS OF ALL AC POWER.
- Power has not been restored.
- The operator reports core exit thermocouples read 1200°F and increasing.

Which one of the following statements correctly describes the actions the crew should take?

A. Immediately go to FRP-C.1, RESPONSE TO INADEQUATE CORE COOLING.

B. Remain in ECP-0.0 until after power is restored to at least one emergency bus then transition to FRP-C.1.

C. Complete ECP-0.0 and when directed to implement monitoring CSF status trees in the appropriate recovery procedure, verify a valid RED path exists and transition to FRP-C.1.

D✓ Remain in ECP-0.0 until guidance to transition to SAMGs (Severe Accident Management Guidelines) is met.

A. Incorrect, FRPs are not applicable when ECP-0.0 is in effect

B. Incorrect. Transition to SACRG-1 is required prior to checking at least on train of 4160V ESF busses energized.

C. Incorrect. SACRG-1 will be implemented at step 23 of ECP-0.0, after transition to SACRG no other procedures will implemented.

D. Correct. ECP-0.0 Step 23 with 5 CETs > 1200 °F then go to SACRG-1.
DO NOT USE WITH 52532A08 - 5

69. WE01EG2.2.2 001/MOD/RO/MEM 4.6/4.1/N/N/2/HBF/Y

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 16.0 "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.

Question # 69

K/A WE01EG2.2.2

REFERENCE Docs

Teamwork, like communication, is essential to effective plant operation at all times but especially during emergency operations. No individual can observe everything that is happening during a casualty. It is vital that the operating crew function as a team and maintain open communication between all team members. Each team member has a responsibility to ask questions when he does not understand something and to point out any situation in which he believes the team may be proceeding in the wrong direction. This helps to ensure that the team fully considers all aspects of the situation before reaching a decision.

During an emergency event, a large number of priority tasks must be correctly performed in a limited time. In this situation, the potential for error is increased. The team can minimize this potential if each member follows procedural guidance and strictly performs each task accordingly, while communicating the status of his efforts to the other members. Team members should back each other up when possible to provide additional assurance that tasks are properly completed.

3.0 Procedure Usage

3.1 Entry Conditions

There are two entry points to the ERP network. The first is if a reactor trip or safety injection occurs or is required. When this occurs the network is entered at step 1 of EEP-0, REACTOR TRIP OR SAFETY INJECTION. The second is if a complete loss of AC power to the safeguards busses occurs. For this condition the network is entered at step 1 of ECP-0.0, LOSS OF ALL AC POWER.

Once the ERP network has been entered, the user is directed to other ERPs by transition steps. ESP-0.0, REDIAGNOSIS, may be entered at any time after exiting EEP-0, REACTOR TRIP OR SAFETY INJECTION, when a safety injection is in progress or is required and no red or orange path FRP is being implemented. This procedure is entered based on the user's judgment and is designed to help him determine which ERP should be implemented if any confusion develops.

Each ERP has it's "Purpose" and "Symptoms or Entry Conditions" listed on the first page. This information is presented to help the user ensure that he has transitioned to the correct procedure.

3.2 Notes and Caution Statements

The ERPs have been written to provide concise directed action steps for the user. For this reason, there are many cases where information in addition to action steps is provided to assist the user in proper performance of a step. If the information is needed to prevent personnel injury, mitigate the accident, prevent loss of life or prevent damage to equipment, it is placed in a caution statement. Other information is placed in a note.

Step	Action/Expected Response	Response NOT Obtained

NOTE: This procedure should only be used if SI in progress or required.

___ 1	**Check if any SG is not faulted.**	
	1.1 Check pressures in all SGs - ANY STABLE OR RISING.	1.1 IF a controlled cooldown is in progress, THEN proceed to Step 2. IF NOT, THEN the following applies. • IF main steam lines have NOT been isolated, THEN go to FNP-1-EEP-2, FAULTED STEAM GENERATOR ISOLATION. OR • IF main steam lines are isolated, THEN go to FNP-1-ECP-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS.
___ 2	**Check if all SGs are not faulted.**	
	2.1 Check no SG pressure - FALLING IN AN UNCONTROLLED MANNER OR LESS THAN 50 psig.	2.1 IF affected SG(s) NOT previously isolated, THEN go to FNP-1-EEP-2, FAULTED STEAM GENERATOR ISOLATION.

___Page Completed

The immediate actions in EEP-0, FRP-S.1, and ECP-0.0 will be performed, in order, by the OATC. The UO will also perform the immediate actions in order unless directed otherwise by the Shift Supervisor. If the UO is not in the control room when an event occurs, performance of the immediate actions by the OATC alone is sufficient. When the operator(s) have finished their 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.

3.8 Manual Operator Actions and Early Operator Actions

3.8.1 If the condition is recognized in sufficient time, crews are expected to take manual actions prior to reaching the automatic setpoint for the following ESF actuations: Reactor Trip, Turbine Trip, SI and MSIV isolation. The determination of whether to manually initiate an anticipated automatic action would include consideration of parameter trends and applicable plant parameter values being near the setpoint.

3.8.2 Operators are expected to take manual action to address ESF components which fail to actuate when required (with the exception of starting a DG or closing the output breaker, which requires the procedure to be used to ensure load shed is verified). The Shift Supervisor should be informed as soon as possible after initiating the manual action.

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. The applicable procedure step(s) will be referenced.

4.4 Priority

The CSFSTs shall be continuously monitored in the following order of priority:

S - Subcriticality
C - Core Cooling
H - Heat Sink
P - Integrity
Z - Containment
I - Inventory

If a red path is identified, the user will, unless specifically directed otherwise, immediately suspend the procedure in effect and transition to the specified FRP. CSFST monitoring will continue so that if a higher priority red path occurs, it will be identified. 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. The STA should validate the need for the FRP entry while immediate actions are being performed.

If an orange path is identified, the user will monitor the remaining CSFSTs to ensure that no red path exists. Unless specifically directed otherwise, he will then suspend the procedure in effect and transition to the specified FRP. CSFST monitoring will continue so that if a higher priority orange path or any red path occurs, it will be identified.

If a yellow path is identified, the user is not required to transition to the specified FRP. This indicates an off normal condition that the user should be aware of, but which does not yet challenge a CSF. Implementation of a yellow path FRP is based upon operator judgement when it is determined that adequate time exists to implement it. Optimal recovery procedures (EEPs, ESPs, and ECPs) have priority over yellow path FRPs. While performing a yellow path FRP, continuous actions or foldout page items of the optimal recovery procedure in effect are still applicable and should be monitored. Concurrent procedure usage should not cause difficulties since yellow path FRPs are only performed when adequate time exists.

Once an FRP has been entered due to a red or orange path, the FRP must be performed to completion unless it is preempted by a higher priority FRP. It is expected that the FRP will correct the red or orange condition before all of the operator actions are performed but the user must continue until the FRP directs a transition. In general, the performance of the critical safety functions is based on the current plant parameters. IF a red or orange path condition comes in and clears, THEN the associated FRP does not need to be performed. IF conditions degrade, THEN the status of the safety function will become a continuous red or orange condition at which time the operator would be directed to the appropriate critical safety function.

1. E-0/ESP-0.0-52530A02 012

An inadvertent Safety injection has occurred on Unit 1. ESP-1.1, SI Termination, was in progress when the following conditions occurred:

- Pressurizer level is dropping rapidly.
- SG narrow range water levels are:
  - -1A SG 85% increasing
  - -1B SG 30% decreasing
  - -1C SG 32% decreasing
- All SG pressures are decreasing rapidly.
- MSIVs are open.

Which one of the following would be correct if ESP-0.0, Rediagnosis, were used to determine the correct procedural transition for the above conditions?

The crew could enter ESP-0.0, Rediagnosis, to ______________.

A. verify the SG pressures are decreasing independent of a controlled cooldown and transition to ECP-2.1, Uncontrolled Depressurization of All Steam Generators.

B.✓ verify the SG pressures are decreasing independent of a controlled cooldown and transition to EEP-2, Faulted Steam Generator Isolation.

C. close the MSIVs and verify the SG depressurization stops then transition to EEP-2, Faulted Steam Generator Isolation.

D. close the MSIVs and verify the SG depressurization stops then transition to EEP-3, Steam Generator Tube Rupture.

W/E01EK3.2 E01 Rediagnosis
EK3. Knowledge of the reasons for the following responses as they apply to the (Reactor Trip or Safety Injection/Rediagnosis) (CFR: 41.5, 41.10, 45.6, 45.13)
EK3.2 Normal, abnormal and emergency operating procedures associated with (Reactor Trip or Safety Injection/Rediagnosis).
IMPORTANCE RO 3.0 SRO 3.9

A. Incorrect. This would be the correct actions if MSIVs were already closed, however the crew would not transition to ECP-2.1 since the MS lines are not isolated.
B. Correct. Per step 1 of ESP-0.0 with SG pressures NOT stable or rising and no controlled cooldown in progress with the MS line not having been isolated.
C. Incorrect. ESP-0.0 does not contain the steps to isolate the MS lines in order to create the transition path to EEP-2., and if the Depressurization was stopped, transition to EEP-2 would no longer be required.
D. Incorrect. There is also a SGTR ongoing by the above conditions and the high A SG level lends credibility to this distractor. However, this transition occurs later in the steps of ESP-0.0, and would not be correct under these conditions. The priority of ESP-0.0 is to send to EEP-2 first if there is a SG fault, then to EEP-3 if there is SGTR with no SG Fault, then to EEP-1 if there is no SG Fault or SGTR.

FNP-1-ESP-0.0, Rediagnosis

HLT-32 audit exam
2006 NRC exam

K/A: Rediagnosis - Knowledge of the reasons for the following responses as they apply to the (Reactor Trip or Safety Injection/Rediagnosis): Normal, abnormal and emergency operating procedures associated with (Reactor Trip or Safety Injection/Rediagnosis).

70. WE04EA2.2 001/MOD/RO/MEM 3.6/4.2/N/N/3/CVR/Y

The Unit 1 crew has transitioned to ECP-1.2, LOCA Outside Containment.

Which one of the following correctly states:

1) a system which is isolated,

and

2) the parameter used to determine if the break is isolated IAW ECP-1.2?

A.✓ 1) RHR Cold Leg injection path.
2) RCS Pressure rising.

B. 1) RHR Cold Leg injection path.
2) RCS Subcooling rising.

C. 1) HHSI Cold Leg injection path.
2) RCS Pressure rising.

D. 1) HHSI Cold Leg injection path.
2) RCS Subcooling rising.

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)

Question # 70

K/A WE04EA2.2

REFERENCE Docs

Step	Action/Expected Response	Response NOT Obtained
___ 3	**Identify source of leak.**	
3.1	Isolate A train RHR cold leg injection path. 1A RHR HX TO RCS COLD LEGS ISO [] Q1E11MOV8888A closed RHR TO RCS HOT LEGS XCON [] Q1E11MOV8887A closed	A & B part 1
3.2	Check RCS pressure - RISING. 1C(1A) LOOP RCS WR PRESS [] PI 402A [] PI 403A	C & A part 2 3.2 Proceed to step 3.4.
3.3	Go to FNP-1-EEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.	
3.4	Restore A train RHR cold leg injection path. 1A RHR HX TO RCS COLD LEGS ISO [] Q1E11MOV8888A open RHR TO RCS HOT LEGS XCON [] Q1E11MOV8887A open	
3.5	Isolate B train RHR cold leg injection path. 1B RHR HX TO RCS COLD LEGS ISO [] Q1E11MOV8888B closed RHR TO RCS HOT LEGS XCON [] Q1E11MOV8887B closed	

Step 3 continued on next page.

___Page Completed

1. ECP-1.2-52532E07 003/HLT//M (LEVEL 1) PROC/W/E04EA2.2//~~2009 AUDIT//~~ Bank

WE04EA2.2

The crew has transitioned to ECP-1.2, LOCA Outside Containment.

Which ONE of the following parameters is used to determine if the break is isolated, in accordance with ECP-1.2?

A. Pressurizer level increasing.

B.✓ RCS pressure increasing.

C. Core exit thermocouple temperature decreasing.

D. RCS subcooling increasing.

DISTRACTOR ANALYSIS:

A - Incorrect; PZR level alone is not sufficient indication of break isolation.

B - Correct; Step 2 of ECP-1.2

C - Incorrect; Core exit thermocouple temperature decreasing is indication of heat removal not a sole indicator of break isolation.

D - Incorrect; RCS subcooling increasing is indication of heat removal and possible pressure increase but not a sole indicator of break isolation.
2006 NRC exam
Turkey Point 2002 NRC Exam

71. WE05EK1.1 001/NEW/RO/MEM 3.8/4.1/N/N/4/HBF/Y

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 all Reactor Vessel Head vents.

C✓ Open one PORV AND Open all Reactor Vessel Head vents.

D. Open one PORV AND Open all Reactor Vessel Head vents, AND place 1B Charging pump on B train and start 1B Charging pump.

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.

$$\left(\frac{1200 gal}{\text{min}}\right)\left(\frac{7.48 lbs}{gal}\right)\left(\frac{60\,\text{min}}{hr}\right) = 538560 lbs/hr$$

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).

Question # 71

K/A WE05EK1.1

REFERENCE Docs

Given the following conditions:

- A loss of heat sink has occurred.
- The operating crew is establishing RCS Bleed and Feed in accordance with EOP-FRHS-1, Loss Of Secondary Heat Sink.
- The RO opens one PORV. He reports that the second PORV will NOT open.

Which one of the following describes the consequences of the PORV failure?

a. Bleed and Feed cooling of the RCS must be terminated and secondary depressurization to inject condensate pump flow must be immediately initiated.

b. ALL SGs will require depressurization to inject the alternate source of feedwater.

c. The RCS will rapidly re-pressurize when the SGs empty, resulting in a violation of the RCS Safety Limit.

d. The RCS may not depressurize quickly enough to ensure sufficient SI flow to provide RCS heat removal, and other RCS openings may have to be established.

Answer	d	Exam Level	B	Cognitive Level	Comprehension	Facility	Salem 1 & 2	Exam Date:	6/11/2004
Tier	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				00WE05K101
E05	Loss of Secondary Heat Sink							Record Number	25

EK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Secondary Heat Sink:

EK1.1 Components, capacity, and function of emergency systems. 3.8 4.1

Explanation of Answer: Distractor c is incorrect because although a red condition on Core Cooling may eventually occur, there is available makeup with charging. Distractor b is incorrect because Bleed and Feed is preferable, because SI flow may NOT be adequate at the PORV setpoint. Distractor a is incorrect because action to align condensate pumps is already taken, and not as a contingency to Bleed and Feed. D is correct because FRHS Basis document describes on page 33 the consequences of not having both PORV's open, and it is D.

Reference Title

Response to Loss of Secondary Heat Sink

Learning Objectives

FRHS00E004 — Explain the effectiveness/ineffectiveness of Safety Injection flow in mitigating the consequences of a loss of heat sink event, with no other operator action

Material Required for Examination:

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Beaver Valley NRC Exam 12/1/2002

Section: Procedure

Unit 1 ERP Step: 17 **Unit 2 ERP Step:** 17 ***ERG Step No:*** *15*

ERP StepText: Establish RCS bleed path.

ERG StepText: *1. Establish RCS Bleed Path; 2. Verify Adequate RCS Bleed Path*

Purpose: 1. To open all PRZR PORVs to establish an RCS bleed path. 2. To verify that an adequate RCS bleed path is established and, if not, to establish alternative bleed path or cooling methods.

Basis: 1. The operator ensures that the pressurizer block valves are open and opens both pressurizer PORVs to establish an RCS bleed path. These valves must be maintained in the open position until secondary heat sink is restored. Once the pressurizer PORVs are open, the RCS will depressurize and the charging/SI pumps and/or high-head SI pumps will deliver subcooled flow to the RCS. This will provide adequate RCS heat removal until flow can be established to the steam generators to restore secondary heat sink. 2. After manually opening the pressurizer PORVs, the operator should check that both pressurizer PORVs are maintained in the open position. If both valves are maintained open, sufficient RCS bleed flow exists to permit RCS heat removal. 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. Although only one open PRZR PORV may not be sufficient to maintain adequate RCS bleed flow, the operator should maintain one PRZR PORV open, if possible, and open all RCS high point vents to provide additional bleed path capability. In addition, the operator should align any available low pressure water source to the SG(s). The operator should then attempt to open a steam generator PORV for at least one intact SG and depressurize that SG to atmospheric pressure to inject the low pressure water source to restore secondary heat removal. If a low pressure water source can not be aligned, a SG should not be depressurized in order to minimize the risk of tube creep rupture that can occur following a severe accident where the SG tubes are subjected to high RCS temperatures and large primary-to-secondary pressure differences. It should be noted that RCS inventory depletion will occur from the open single PRZR PORV, the PRZR safety valves, and high point vents as the steam generator(s) is being depressurized to atmospheric pressure.

Knowledge: o The operator should verify that the pressurizer PORVs do not automatically close following release of the control board switches. If the pressurizer PORVs do automatically close due to a spring return to auto switch, the operator should manually maintain the control board switches in the open position. o The operator may observe increasing pressurizer level after the pressurizer PORVs are opened. Eventually the pressurizer may become water solid with water relief occurring through the pressurizer PORVs.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 345,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures ≤ 325°F, and MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the ± 1% tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the

(continued)

72. WE08EK1.1 001/NEW/RO/C/A 3.5/3.8/N/N/2/HBF/VER 5 EDITORIAL

Given the following plant conditions for Unit 1:

- A failure has occurred on TI-412, Loop A T_{AVG}, resulting in a constant output equivalent to 561°F.
- A failure has occurred on TI-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 that will prevent a Pressurized Thermal Shock condition from developing, **with no operator actions**?

A. Low Main Steam Pressure SI.

B. P-4, Reactor Trip Interlock.

C. Main Steam Line Isolation on High Flow.

D✓ Main Steam Line Isolation on Low Pressure.

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. Low Main Steam pressure Si will actuate, however a Pressurized Thermal Shock (PTS) condition is initiated by cooldown. An SI actuation can cause pressure to be maintained high during a cooldown event which will actually complicate, NOT PREVENT, plant conditions creating a PTS condition, particularly if RCS pressure is rapidly restored after a significant cooldown. PTS is a HIGH pressure condition with a significant COOLDOWN.

Plausible: This failure will result in a Low MS pressure SI actuating on Low MS header pressure of 585 psig.

B - 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. (A181007 pg 2-26).

C - 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.

D -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.

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,

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

Question # 72

K/A WE08EK1.1

REFERENCE Docs

2. DESCRIPTION

An event or series of events which leads to a relatively rapid and severe reactor vessel downcomer cooldown can result in a thermal shock to the vessel wall that may lead to a small flaw, which may already exist in the vessel wall, growing into a larger crack. The growth or extension of such a flaw may lead, in some cases (where propagation is not stopped within the wall), to a loss of vessel integrity. The objective of Function Restoration Guideline FR-P.1 is to prevent the growth of a flaw and, in the event the limits set forth are exceeded, provide specific actions which appropriately restrict operation to prevent further challenges to vessel integrity.

Two separate types of events lead to entry into this guideline:

- Pressurized Thermal Shock Events

Several possible transients can be hypothesized which will produce rapid and extensive temperature decreases in the RCS cold leg(s) and, by inference, also the reactor vessel downcomer region. The rate and extent of cooldown determine whether entry into this guideline is on a RED or ORANGE priority. The actions in this guideline attempt to stop the cooldown, i.e., stabilize temperature, and also decrease RCS pressure to reduce the pressure stress component of total stress in the reactor vessel wall, partially offsetting the large thermal stress created by the rapid cooldown.

- Cold Overpressure Events

For this type of event, entry on an ORANGE priority is warranted if RCS pressure has exceeded the Cold Overpressure Protection Limit, and RCS cold leg temperature is sufficiently low that vessel ductility is reduced. There is little or no thermal stress associated with this event, so the benefit in using this guideline comes from the prompt RCS pressure reduction actions which supplement the Cold Overpressure Protection System.

for the required engineered safety features lines. Phase B isolation is initiated by containment pressure High-3 (27 psig) or by manual actuation (using 2/4 Containment Phase B Isolation/Containment Spray Actuation handswitches).

The Containment Ventilation Isolation isolates the containment atmosphere from the environment to limit the release of radioactive fission products in the event of an accident. This function is actuated on the completion of the SI logic, high radioactivity levels in the purge exhaust, or by manual initiation of either Phase A Containment Isolation or Phase B Isolation/Containment Spray Actuation. (References 6.1.022, 6.4.007, 6.4.015, 6.7.012, 6.4.080)

3. Main Steam Line Isolation

Isolation of the Main Steam lines limits the effects of an uncontrolled release of steam either inside or outside the containment. For a break upstream of the isolation valves (MSIV) in the steamlines, valves closure will limit the release to the blowdown of the one affected steam generator. A break downstream of the valves is limited to the depressurization of the pipe volume downstream of the valves. This results in a rapid termination of the event and significantly reduces the mass lost from the secondary.

The Main Steam Line Isolation is initiated by the following:

a. High steam line flow with low-low T_{avg} , 1/2 steam flow channels above setpoint (40% of full steam flow between 0-20% load and increasing linearly to 110% at full load) on 2/3 steam lines with $T_{avg} \leq$ P-12

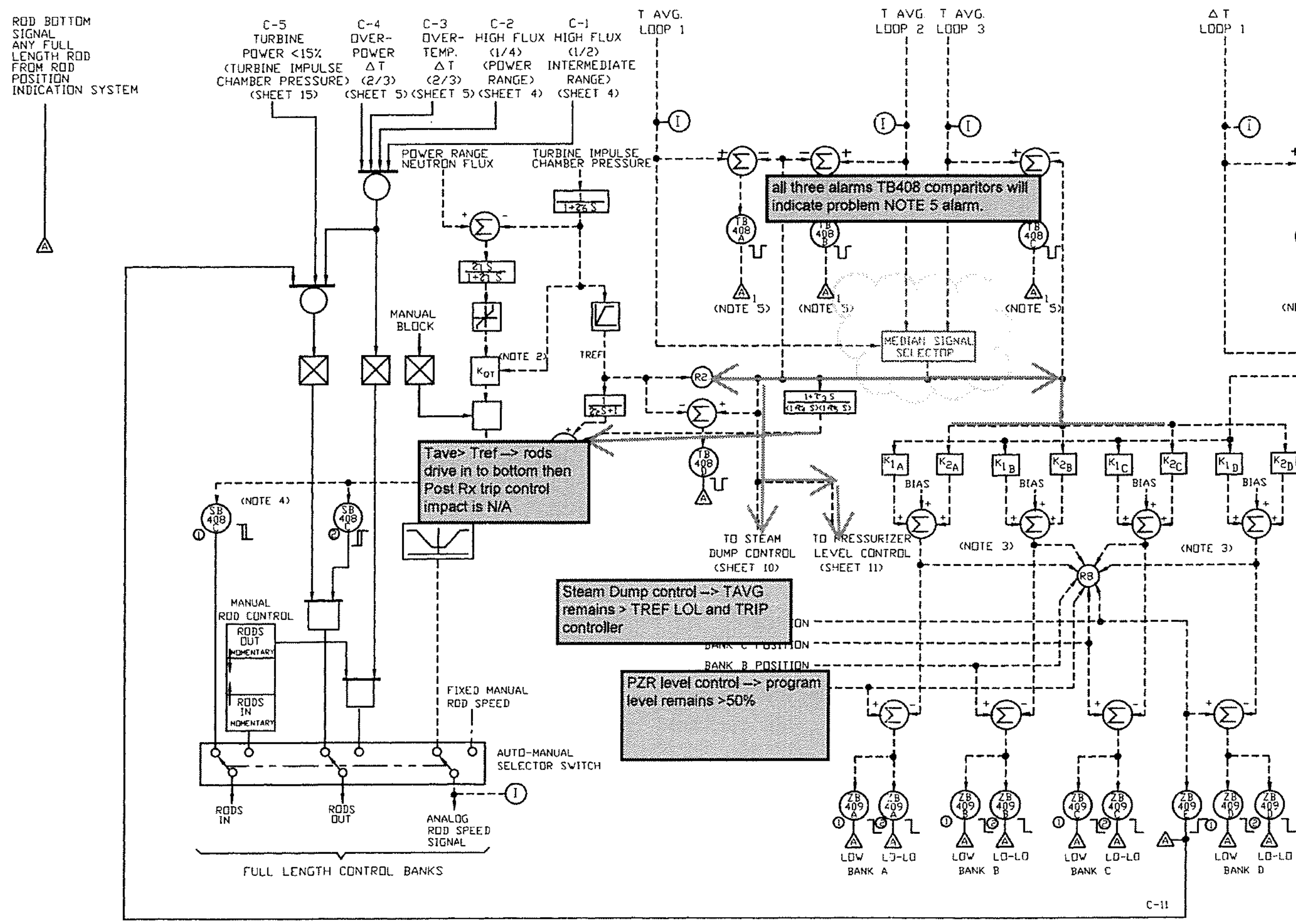
b. Low steam pressure; $\leq$ 585 psig on 2/3 S.G.

c. High-2 containment pressure; $\geq$ 16.2 psig on 2/3

d. Manual. By closing each MSIV by operating individual hand switches. (References 6.1.022, 6.4.007, 6.4.015, 6.7.012)

4. Main Feedwater Isolation and Turbine Trip

The Main Feed Line Isolation is initiated to prevent excessive cooldown of the reactor or to lessen the severity of the transient overall. The following signals are utilized to initiate the Main Feed line Isolation:



NOTES:

1. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT.
2. K_{OT} MAY VARY INVERSELY PROPORTIONAL TO LOAD WITH A FIXED LIMIT, OR MAY VARY IN TWO DISCRETE STEPS WITH BREAK POINTS AT 30-50% AND 60 TO 80% TURBINE LOAD.
3. THE SUMMER OUTPUTS HAVE FIXED MANUALLY ADJUSTABLE UPPER LIMITS.
4. THE ROD DIRECTION BISTABLES #SB408C ARE "ENERGIZE TO ACTUATE"
5. ALARM 1 AND ALARM 3 MUST HAVE REFLASH CAPABILITY.

ROD CONTROLS AND BLOCKS

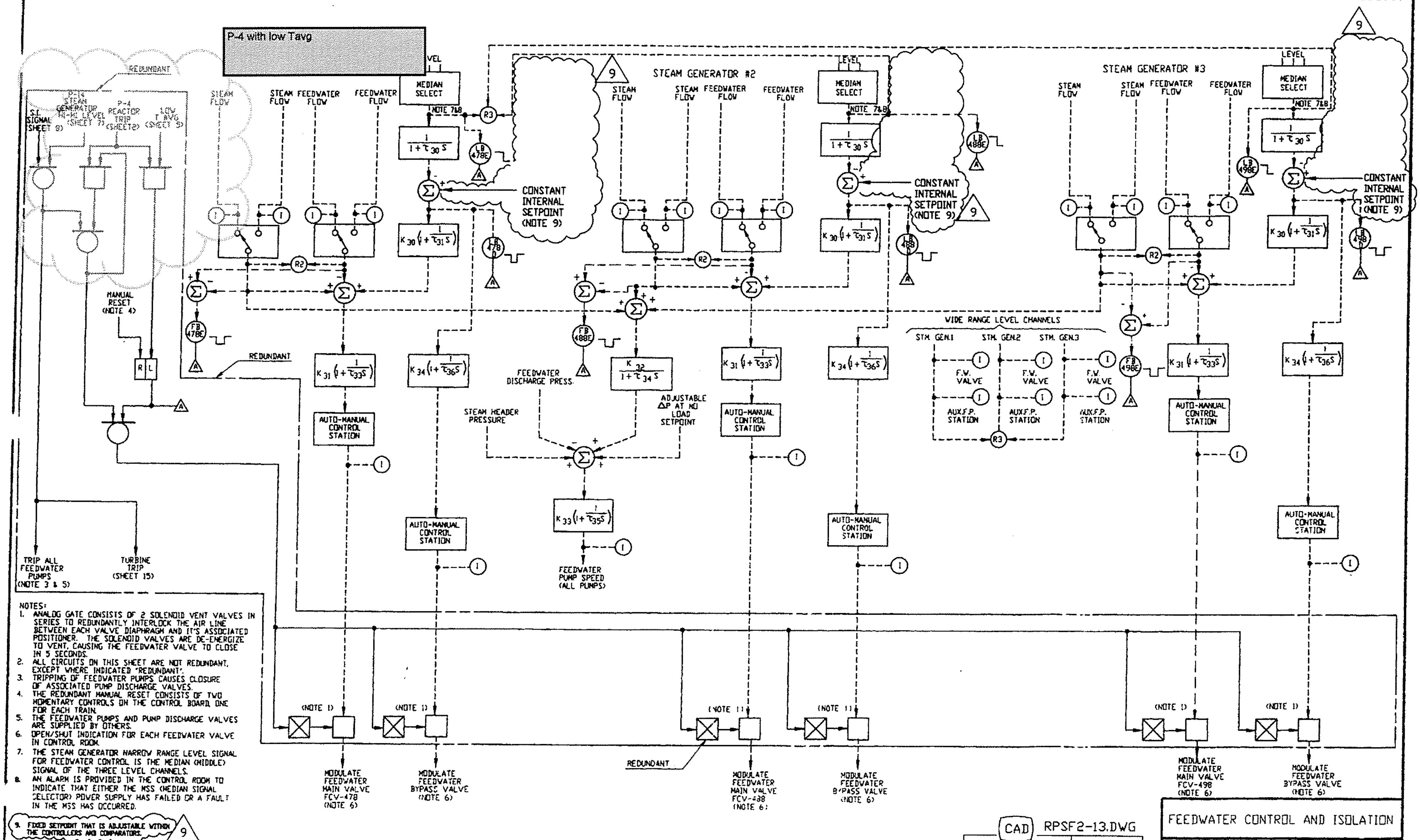
FIGURE F-2 SHEET 9

RPSFSD09.DWG

FNP UNITS 1 & 2

RPS-FSD

A-181007



FEEDWATER CONTROL AND ISOLATION

FIGURE F-2 SHEET 13

REV. 9

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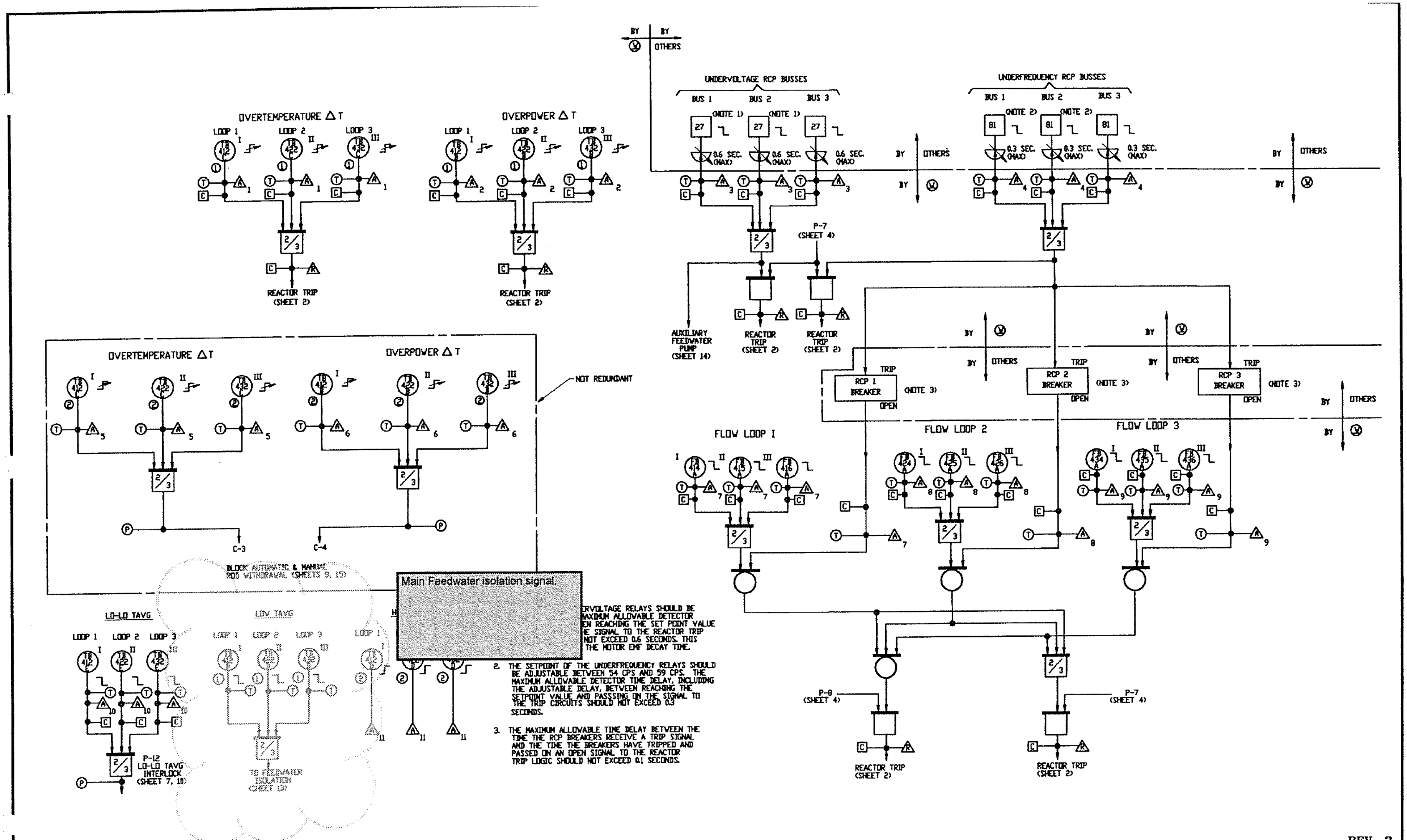
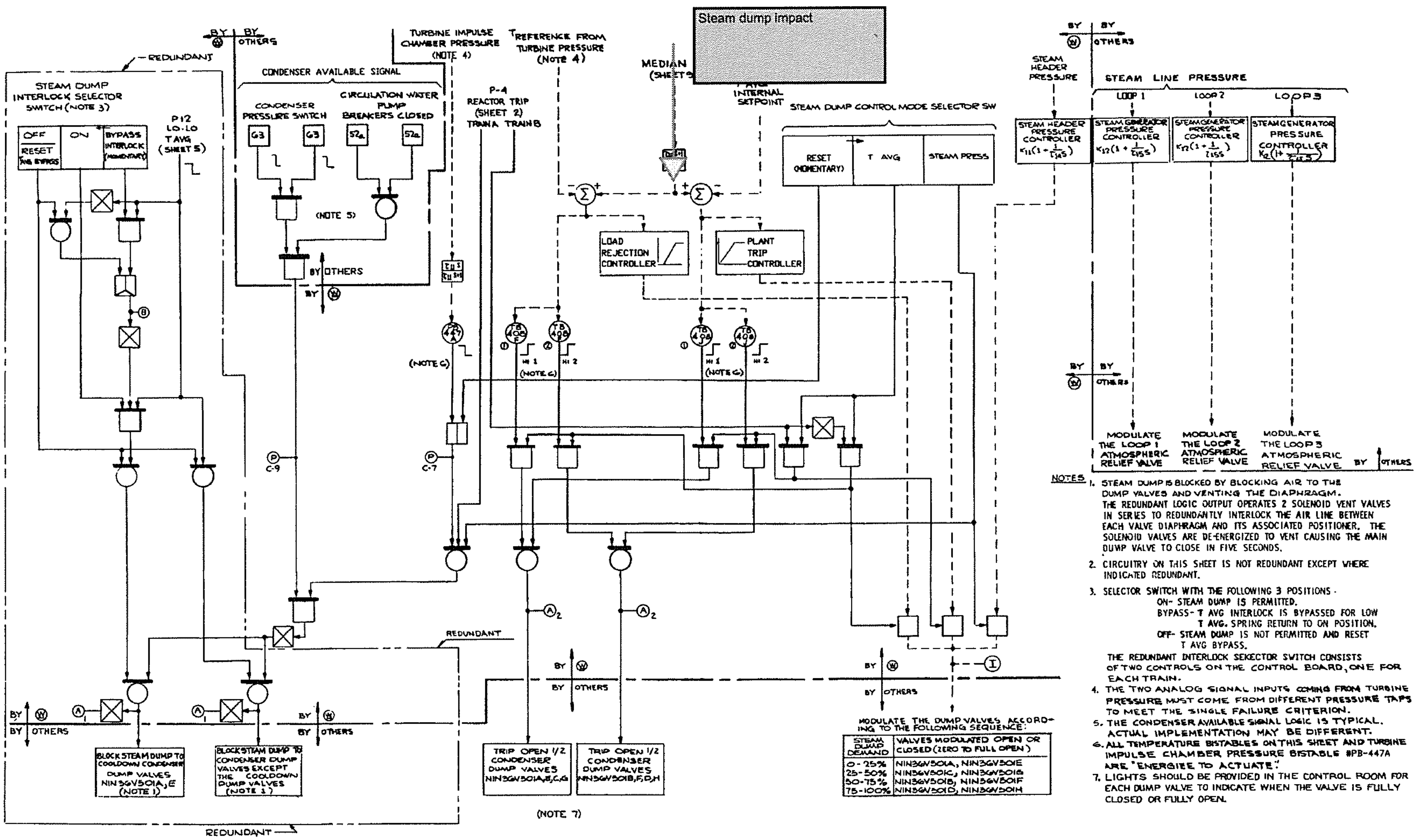


REV. 2

PRIMARY COOLANT SYSTEM
TRIP SIGNALS

FIGURE F-2 SHEET 5

REFERENCE U-166235 & U-207294



STEAM DUMP DEMAND	VALVES MODULATED OPEN OR CLOSED (ZERO TO FULL OPEN)
0-25%	N1N3GV501A, N1N3GV501E
25-50%	N1N3GV501C, N1N3GV501G
50-75%	N1N3GV501B, N1N3GV501F
75-100%	N1N3GV501D, N1N3GV501H

NOTES

1. STEAM DUMP IS BLOCKED BY BLOCKING AIR TO THE DUMP VALVES AND VENTING THE DIAPHRAGM. THE REDUNDANT LOGIC OUTPUT OPERATES 2 SOLENOID VENT VALVES IN SERIES TO REDUNDANTLY INTERLOCK THE AIR LINE BETWEEN EACH VALVE DIAPHRAGM AND ITS ASSOCIATED POSITIONER. THE SOLENOID VALVES ARE DE-ENERGIZED TO VENT CAUSING THE MAIN DUMP VALVE TO CLOSE IN FIVE SECONDS.
2. CIRCUITRY ON THIS SHEET IS NOT REDUNDANT EXCEPT WHERE INDICATED REDUNDANT.
3. SELECTOR SWITCH WITH THE FOLLOWING 3 POSITIONS -
   ON- STEAM DUMP IS PERMITTED.
   BYPASS- T AVG INTERLOCK IS BYPASSED FOR LOW T AVG. SPRING RETURN TO ON POSITION.
   OFF- STEAM DUMP IS NOT PERMITTED AND RESET T AVG BYPASS.
   THE REDUNDANT INTERLOCK SEKECTOR SWITCH CONSISTS OF TWO CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
4. THE TWO ANALOG SIGNAL INPUTS COMING FROM TURBINE PRESSURE MUST COME FROM DIFFERENT PRESSURE TAPS TO MEET THE SINGLE FAILURE CRITERION.
5. THE CONDENSER AVAILABLE SIGNAL LOGIC IS TYPICAL. ACTUAL IMPLEMENTATION MAY BE DIFFERENT.
6. ALL TEMPERATURE BISTABLES ON THIS SHEET AND TURBINE IMPULSE CHAMBER PRESSURE BISTABLE #PB-447A ARE "ENERGIZE TO ACTUATE".
7. LIGHTS SHOULD BE PROVIDED IN THE CONTROL ROOM FOR EACH DUMP VALVE TO INDICATE WHEN THE VALVE IS FULLY CLOSED OR FULLY OPEN.

REV. 4 7/98

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

STEAM DUMP CONTROL

FIGURE F-2, SHEET 10

73. WE11EK1.3 001/NEW/RO/C/A 3.6/4.0/N/N/3/CVR/Y

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 the two alarms together: CF3 & CF5 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)

Question # 73

K/A WE11EK1.3

REFERENCE Docs

LOCATION CF3

1A pump lost power 20 mins before

SETPOINT: Variable Current/Time

F3
1A OR 1B RHR PUMP OVERLOAD TRIP

ORIGIN: 86 Relay (DF-09) Control Circuit, 1A RHR Pump
or 86 Relay (DG-09) Control Circuit, 1B RHR Pump

PROBABLE CAUSE

1. 1A or 1B RHR Pump overloaded
2. 1A or 1B RHR Pump electrical or mechanical fault

AUTOMATIC ACTION

1A or 1B RHR Pump Trip.

OPERATOR ACTION

1. Check indications and determine which RHR pump has tripped.
2. IF the NON-AFFECTED pump is NOT running, THEN start the pump in accordance with FNP-1-AOP-12.0, RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION.
3. Refer to FNP-1-AOP-12.0, RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION.
4. Notify appropriate personnel to determine and correct the cause of the RHR Pump trip.
5. Return the RHR System to normal operation as soon as possible.
6. Refer to Technical Specifications, Sections 3.4.6, 3.4.7, 3.4.8, 3.5.3, 3.9.4 and 3.9.5, for LCO Requirements.

References: A-177100, Sh. 168; D-175041; D-177193; A-177048, Sh. 260 & 274; Technical Specifications.

LOCATION CH2

SETPOINT: 12′7″ ± 1″ above Tank Bottom
(150,000 Gallons)

ORIGIN: Level Transmitter Q1F16LT-501 through a comparator card bistable designated LSL503 in BOP Cabinet J.

H2
RWST LVL A TRN LO

PROBABLE CAUSE

1. RWST in use for Safety Injection purposes.
2. RWST in use for Refueling purposes.
3. Failed Level Transmitter.

AUTOMATIC ACTION

NONE

OPERATOR ACTION

1. IF an ECCS actuation signal is present, THEN refer to FNP-1-ESP-1.3, TRANSFER TO COLD LEG RECIRCULATION.
2. Determine actual tank level as indicated by LI-4075A & B, on the MCB OR the local level indicator on the side of the RWST.
3. IF an ECCS Actuation Signal is NOT present OR the tank is NOT being used for Refueling, THEN notify appropriate personnel to determine and correct the cause of the alarm.
4. IF required, THEN restore RWST level to normal per FNP-1-SOP-2.3, CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM, section 4.2.3.
5. Refer to Technical Specification 3.3.3 for LCO requirements.

References: A-177100, Sh. 177; A-170750, Pg. 95; D-173497; Technical Specifications

LOCATION CH3

H3	RWST LVL B TRN LO

SETPOINT: 12′7″ ± 1″ above Tank Bottom
(150,000 Gallons)

ORIGIN: Level Transmitter Q1F16LT502 through a comparator card bistable designated LSL504 in BOP Cabinet K.

PROBABLE CAUSE

1. RWST in use for Safety Injection purposes.
2. RWST in use for Refueling purposes.
3. Failed Level Transmitter.

AUTOMATIC ACTION

NONE

OPERATOR ACTION

1. If an ECCS actuation signal is present, THEN refer to FNP-1-ESP-1.3. TRANSFER TO COLD LEG RECIRCULATION.
2. Determine actual tank level as indicated by LI-4075A & B, on the MCB OR the local level indicator on the side of the RWST.
3. IF an ECCS Actuation signal is NOT present OR the tank is NOT being used for Refueling, THEN notify appropriate personnel to determine and correct the cause of the alarm.
4. IF required, THEN restore RWST level to normal per FNP-1-SOP-2.3, CHEMICAL AND VOLUME CONTROL SYSTEM REACTOR MAKEUP CONTROL SYSTEM, section 4.2.3.
5. Refer to Technical Specification 3.3.3 for LCO requirements.

References: A-177100, Sh. 178; A-170750, Pg. 95; B-170058, Sh. 72; D-173497; Technical Specifications

A. **Purpose**

This procedure provides actions to restore emergency coolant recirculation capability, to delay depletion of the RWST by adding makeup and reducing outflow, and to depressurize the RCS to minimize break flow.

B. **Symptoms or Entry Conditions**

I. This procedure is entered when emergency coolant recirculation capability is lost; from the following:

a. FNP-1-EEP-1, LOSS OF REACTOR OR SECONDARY COOLANT, step 14, when cold leg recirculation capability cannot be verified.

b. FNP-1-ESP-1.3, TRANSFER TO COLD LEG RECIRCULATION, step 7, when at least one flow path from the containment sump cannot be established or maintained.

c. FNP-1-ECP-1.2, LOCA OUTSIDE CONTAINMENT, step 3, when a LOCA outside containment cannot be isolated.

Step	Action/Expected Response	Response NOT Obtained
___10	**Evaluate containment spray requirements.**	
10.1	Check containment spray pumps - ALIGNED TO RWST. RWST TO 1A(1B) CS PUMP [] Q1E13MOV8817A open [] Q1E13MOV8817B open	10.1 IF containment spray pumps aligned to the the containment sump, THEN proceed to Step 12.
10.2	Determine number of containment spray pumps required based on the Table below.	

RWST LEVEL	CONTAINMENT PRESSURE	FAN COOLERS RUNNING IN EMERGENCY MODE	SPRAY PUMPS REQUIRED
GREATER THAN 12.5 FT	GREATER THAN 54 PSIG	—	2
	BETWEEN 27 PSIG AND 54 PSIG	0, 1	2
		2, 3	1
		4	0
	LESS THAN 27 PSIG	—	0
BETWEEN 4.5 FT and 12.5 FT	GREATER THAN 54 PSIG	—	2
	BETWEEN 27 PSIG and 54 PSIG	1, 2	1
		3, 4	0
	LESS THAN 27 PSIG	—	0
LESS THAN 4.5 FT	—	—	0

10.3 Establish required number of started containment spray pumps.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

NOTE: Step 21 is a continuing action.

___21 **[CA] Check SI termination criteria.**

21.1 Check REACTOR VESSEL LEVEL. — 21.1 Proceed to step 27.

- IF any RCP started, THEN check REACTOR VESSEL LEVEL greater than 72% UPPER PLENUM.

OR

- IF no RCP started, THEN check REACTOR VESSEL LEVEL greater than 0% UPPER PLENUM.

Step 21 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

NOTE: TABLE 1 provides minimum SI flow to remove decay heat vs. time elapsed after shutdown.

21.2 Check SUB COOLED MARGIN MONITOR - GREATER THAN 66°F {95°F} SUBCOOLED IN CETC MODE.

21.2 Establish minimum SI flow.

21.2.1 IF charging pump suction aligned to RHR, THEN stop all CHG PUMPs.

21.2.2 Verify both RHR PUMPs stopped.

21.2.3 Open miniflow valve for available charging pump.

1A(1B,1C) CHG PUMP MINIFLOW ISO
- [] Q1E21MOV8109A
- [] Q1E21MOV8109B
- [] Q1E21MOV8109C

21.2.4 Open common miniflow isolation valve.

CHG PUMP MINIFLOW ISO
- [] Q1E21MOV8106

21.2.5 Verify RWST to charging pump valves open.

RWST TO CHG PUMP
- [] Q1E21LCV115B
- [] Q1E21LCV115D

21.2.6 Close RHR supply to A AND B train charging pump suction.

1A(1B) RHR HX TO CHG PUMP SUCT
- [] Q1E11MOV8706A
- [] Q1E11MOV8706B

Step 21 continued on next page.

____Page Completed

Step	Action/Expected Response	Response NOT Obtained

21.2.7 Start CHG PUMP with miniflow valve open.

21.2.8 Maintain core exit T/C temperature stable or falling.

- IF core exit T/C temperatures rising AND started charging pump aligned to A train, THEN establish A train SI flow.

  HHSI TO
  RCS CL ISO
  [] Q1E21MOV8803A open
  [] Q1E21MOV8803B open

- IF core exit T/C temperatures rising AND started charging pump aligned to B train, THEN establish B train SI flow.

  CHG PUMP RECIRC
  TO RCS COLD LEGS
  [] Q1E21MOV8885 open

21.2.9 Open and close HHSI isolation valves to control SI flow to keep core exit T/C temperatures stable or falling.

HHSI TO
RCS CL ISO
[] Q1E21MOV8803A
[] Q1E21MOV8803B

CHG PUMP RECIRC
TO RCS COLD LEGS
[] Q1E21MOV8885

Step 21 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
		21.2.10 Consult TSC staff to determine if additional options are available or needed to further minimize SI flow and conserve RWST inventory.
		21.2.11 Proceed to step 27.
___22	**Reset safeguards signals.**	
	22.1 Verify PHASE A CTMT ISO RESET. [] MLB-2 1-1 not lit [] MLB-2 11-1 not lit	
	22.2 Verify PHASE B CTMT ISO RESET. [] MLB-3 1-1 not lit [] MLB-3 6-1 not lit	
___23	**Establish instrument air to containment.**	
	23.1 Check 1D 4160 V bus - ENERGIZED.	23.1 Proceed to step 23.3.
	23.2 <u>IF</u> 1D 4160 V bus energized, <u>THEN</u> proceed to step 23.4.	
	23.3 Establish power to 1A 600 V LC emergency section loads.	
	23.3.1 Verify open BKR EA08-1.	
	23.3.2 Verify closed BKRs ED08-1 and EA09-1.	

Step 23 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

***

CAUTION: To ensure adequate supply voltage to all class 1E loads and to meet short circuit analysis constraints, only one air compressor, 1C (preferred) or 1A, may be powered from the diesel generator. One air compressor will consume 0.16 MW of diesel generator load.

***

23.4 Verify 1C air compressor in service.

23.4.1 Verify 1C air compressor handswitch in AUTO after START/RUN.

23.4.2 Verify 1C air compressor started.

23.4 Align 1A air compressor for service.

a) Verify 1C air compressor handswitch in OFF.

b) Verify SI - RESET.

[] MLB-1 1-1 not lit
[] MLB-1 11-1 not lit

***

CAUTION: IF offsite power is lost after sequencer is reset, THEN manual actions may be required to restart safeguards equipment.

***

c) Reset B1F sequencer by depressing the ESS STOP RESET pushbutton on the sequencer panel. (139 ft, AUX BLDG A train SWGR room)

d) Place BKR DF13 SYNCH SWITCH in MAN.

e) Close BKR DF13 (1F 4160 V bus tie to 1H 4160 V bus).

f) IF 1H 4160 V bus energized, THEN energize 1G 600 V LC from normal supply.

[] BKR DH01 closed
[] BKR EG02-1 closed

g) Start 1A AIR COMPRESSOR.

Step 23 continued on next page.

____Page Completed

Step	Action/Expected Response	Response NOT Obtained
23.5	Check INST AIR PRESS PI 4004B - GREATER THAN 85 psig.	23.5 Perform the following.

NOTE: The intent of this step is to regain control of critical air operated components including PORVs and atmospherics. Based on plant conditions and availability of manpower, the applicability, priority and performance of the following actions is at the discretion of the Shift Supervisor.

23.5.1 Restore air pressure.

- Verify proper air compressor operation using FNP-1-SOP-31.0, COMPRESSED AIR SYSTEM.

<u>OR</u>

- <u>IF</u> 2C air compressor available, <u>THEN</u> align 2C air compressor to Unit 1 using FNP-1-SOP-31.0, COMPRESSED AIR SYSTEM.

23.5.2 <u>IF</u> instrument air <u>NOT</u> restored, <u>THEN</u> align nitrogen supply to PORVs using FNP-1-SOP-62.1, BACK-UP AIR OR NITROGEN SUPPLY TO THE PRESSURIZER POWER OPERATED RELIEF VALVES.

23.5.3 <u>IF</u> instrument air <u>NOT</u> restored, <u>THEN</u> align emergency air supply to atmospheric relief valves and/or TDAFWP using FNP-1-SOP-62.0, EMERGENCY AIR SYSTEM.

Step 23 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
		23.5.4 <u>IF</u> instrument air <u>NOT</u> restored because 1G 600 V LC is deenergized, <u>THEN</u> energize 1G 600 V LC from 1F 600 V LC using FNP-1-SOP-36.3, 600, 480 AND 208/120 VOLT AC ELECTRICAL DISTRIBUTION SYSTEM.
		23.5.5 <u>WHEN</u> instrument air pressure restored, <u>THEN</u> perform step 23.6.
		23.5.6 Proceed to step 24.
	23.6 Check instrument air to containment. IA TO CTMT [] MLB-3 1-2 <u>NOT</u> lit IA TO PENE RM PRESS LO [] Annunciator KD1 clear	23.6 Align instrument air to containment. (BOP) IA TO PENE RM [] N1P19HV3825 open [] N1P19HV3885 open IA TO CTMT [] Q1P19HV3611 open
___24	**Stop SI pumps.**	
	24.1 Verify both RHR PUMPs - STOPPED.	
	24.2 Verify only one CHG PUMP - STARTED.	

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

___25 **Isolate HHSI flow.**

25.1 Verify charging pump miniflow valves - OPEN.

1A(1B,1C) CHG PUMP
MINIFLOW ISO
[] Q1E21MOV8109A
[] Q1E21MOV8109B
[] Q1E21MOV8109C

CHG PUMP
MINIFLOW ISO
[] Q1E21MOV8106

25.2 Close HHSI isolation valves.

HHSI TO
RCS CL ISO
[] Q1E21MOV8803A
[] Q1E21MOV8803B

___26 **Establish normal charging.**

26.1 Manually close charging flow control valve.

CHG FLOW
[] FK 122

Step 26 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained

26.2 Verify charging flow path aligned.

26.2.1 Verify charging pump discharge flow path - ALIGNED.

CHG PUMP
DISCH HDR ISO
[] Q1E21MOV8132A open
[] Q1E21MOV8132B open
[] Q1E21MOV8133A open
[] Q1E21MOV8133B open

CHG PMPS TO
REGENERATIVE HX
[] Q1E21MOV8107 open
[] Q1E21MOV8108 open

26.2.2 Verify only one charging line valve - OPEN.

RCS NORMAL
CHG LINE
[] Q1E21HV8146

RCS ALT
CHG LINE
[] Q1E21HV8147

26.3 Establish desired charging flow using charging flow control valve.

CHG FLOW

[] FK 122

____Page Completed

Step	Action/Expected Response	Response NOT Obtained

NOTE: Step 27 is a continuing action.

Step	Action/Expected Response	Response NOT Obtained
___27	**[CA] Verify RCS makeup flow - ADEQUATE.**	
27.1	Check REACTOR VESSEL LEVEL. • IF any RCP started, THEN check REACTOR VESSEL LEVEL greater than 72% UPPER PLENUM. OR • IF no RCP started, THEN check REACTOR VESSEL LEVEL greater than 0% UPPER PLENUM.	27.1 Increase RCS makeup flow to maintain required REACTOR VESSEL LEVEL. • CHG FLOW [] FK 122 adjusted OR • HHSI TO RCS CL ISO [] Q1E21MOV8803A open [] Q1E21MOV8803B open OR • CHG PUMP RECIRC TO RCS COLD LEGS [] Q1E21MOV8885 open

Step 27 continued on next page.

___Page Completed

Step	Action/Expected Response	Response NOT Obtained
27.2	Check core exit T/Cs stable or falling.	27.2 Increase RCS makeup flow to maintain core exit T/Cs stable or falling. • CHG FLOW [] FK 122 adjusted OR • HHSI TO RCS CL ISO [] Q1E21MOV8803A open [] Q1E21MOV8803B open OR • CHG PUMP RECIRC TO RCS COLD LEGS [] Q1E21MOV8885 open

___Page Completed

74. WE12EK2.1 001/NEW/RO/C/A 3.4/3.7/N/N/3/HBF/Y

Unit 2 is in Mode 3, preparing to open the MSIVs after warm-up of the Main Steam lines is complete using SOP-17.0, Main and Reheat Steam.

- RCS Temp is 547°F.
- All SG MSIV **Bypass** valves are open.
- 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 when the MCB handswitches were placed in the CLOSE position.

Which one of the following describes:

1) the position of the MSIV **Bypass** valves,

and

2) the system response to placing each SG MSIV-TEST switches to TEST?

A. 1) The MSIV Bypass valves are open.

2) The MSIVs will be driven fully closed with air;

B. 1) The MSIV Bypass valves closed.

2) The MSIVs will be driven fully closed with air;

C. 1) The MSIV Bypass valves are open.

2) The MSIVs will be partially closed with air;

D✓ 1) The MSIV Bypass valves are closed.

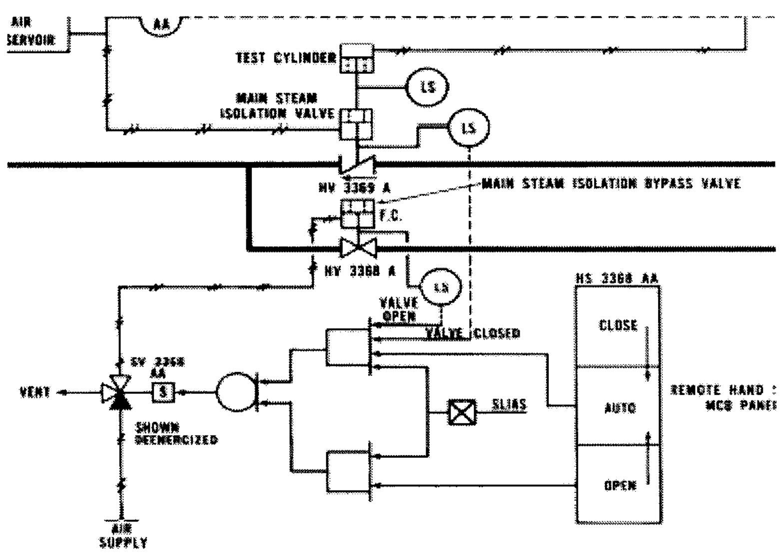
2) The MSIVs will be partially closed with air;

A - Incorrect. 1) 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 MSIVs are normally shut AND are not operated by the MCB MSIV "TRIP" handswitch directly. These valves are equipped with their own handswitches. 2) The test Cylinder is only capable of moving the MSIV disc off its backseat, it is incapable of driving the valve fully closed. System forward flow impacting the disc (swing-check) will force the valve shut.

B - Incorrect. 1) this is correct. The MSIV bypass valves will be closed immediately upon the MSIV Closed limit switch not being satisfied. 2) See A #2

C - Incorrect. 1) See A #1. 2) This portion is correct.

D -Correct. 1) The MSIVs are interlocked such that when the MSIV is no longer closed, the bypass valves close. 2) The TEST cylinder provides enough air force to move the MSIV disc off its backseat, in addition to the spring that should normally move the valve off the backseat, but should not cause the disc to enter the flow stream. It is the weight of the disc and the flow of steam which results in the MSIV closing.



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

Question #	74
K/A	WE12EK2.1

REFERENCE Docs

NOTE:	**When an MSIV is opened, its associated bypass valve will automatically close. The upstream and downstream MSIV for at least one loop must be opened before proceeding to the other loops to maintain the bypass flowpath.**

4.3.10 WHEN steam header pressure AND individual steam generator pressures are approximately equal, THEN open the MSIVs.

- Q2N11HV3369A
- Q2N11HV3370A
- Q2N11HV3369B
- Q2N11HV3370B
- Q2N11HV3369C
- Q2N11HV3370C

4.3.11 WHEN the MSIVs reach the fully open position, THEN verify that the associated bypass valves automatically close.

4.3.12 WHEN desired, THEN unisolate main steam drain pot level control valves by performing the following:

- Open N2N11V905A 2A MS LINE DRN POT TO COND ISO
- Open N2N11V905B 2B MS LINE DRN POT TO COND ISO
- Open N2N11V905C 2C MS LINE DRN POT TO COND ISO
- Open N2N11V905D 2D MS LINE DRN POT TO COND ISO

4.3.12.1 Clear the admin tracking item initiated in step 4.3.1.

75. WE16EK2.1 001/BANK/RO/C/A 3.0/3.3/N/N/2/HBF/Y

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

Question # 75

K/A WE16EK2.1

REFERENCE Docs

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 3

PHASE A CONTAINMENT ISOLATION

NOTE:
- ATTACHMENT 3, FIGURE 1 provides a listing of component names corresponding to each MLB-2 location.
- ATTACHMENT 10 provides a listing of sequenced loads.

1 **Check all the following MLB-2 indicating lights lit.**

1 Verify associated component status.

	1	2	3	4	5	6	7	8	9	10
1	CTMT ISO PHASE A	3657 CLOSED	3198A CLOSED	3772A CLOSED	8112 CLOSED	LCV1003 CLOSED	7126 CLOSED	CONT RM FILT FAN 1A ON	CONT RM PRZN FAN 1A ON	3622 CLOSED
2	3234A CLOSED	3660 CLOSED	3198D CLOSED	3772B CLOSED	8149A CLOSED	3377 CLOSED	3103 CLOSED	3104 CLOSED	3649A CLOSED	3624 CLOSED
3	P16V515 CLOSED	3318B CLOSED	2866C CLOSED	3772C CLOSED	8149B CLOSED	3380 CLOSED	8033 CLOSED	3765 CLOSED	3649B CLOSED	3626 CLOSED
4	P16V517 CLOSED	3999A CLOSED	2867C CLOSED	3443 CLOSED	8149C CLOSED	8871 CLOSED	8028 CLOSED	3766 CLOSED	3649C CLOSED	3628 CLOSED

	11	12	13	14	15	16	17	18	19	20
1	CTMT ISO PHASE A	3658 CLOSED	3198B CLOSED	3196 CLOSED	8100 CLOSED	7136 CLOSED	3331 CLOSED		CONT RM FILT FAN 1B ON	CONT RM PRZN FAN 1B ON
2	3234B CLOSED		3198C CLOSED	3197 CLOSED	8152 CLOSED	3376 CLOSED	3332 CLOSED		3623 CLOSED	3627 CLOSED
3	P16V514 CLOSED	3318A CLOSED	2866D CLOSED	3067 CLOSED	8880 CLOSED	7150 CLOSED	3333 CLOSED		3625 CLOSED	3629 CLOSED
4	P16V516 CLOSED	3999B CLOSED	2867D CLOSED	3095 CLOSED	8860 CLOSED	8961 CLOSED	3334 CLOSED		8047 CLOSED	3659 CLOSED

2 **Notify control room of phase A containment isolation status.**

-END-

ATTACHMENT 3

FIGURE 1

LOCATION	COMPONENT NUMBER	NAME
1-1	N/A	CTMT ISO PHASE A
1-2	Q1N12HV3234A	TDAFWP STM SUPP WARMUP ISO (BOP)
1-3	Q1P16V515	SW TO TURB BLDG ISO A TRN
1-4	Q1P16V517	SW TO TURB BLDG ISO B TRN
2-1	Q1E14HV3657	CTMT ATMOS TO R-11/12 ISO (BOP)
2-2	Q1E14MOV3660	CTMT ATMOS TO R-11/12 ISO (BOP)
2-3	Q1E14MOV3318B	CTMT ΔP ISO (BOP)
2-4	Q1E12HV3999A	RX CAV CLG DMPR (BOP)
3-1	Q1P13HV3198A (Q1P13V284)	CTMT PURGE DMPRS (HS-3198)
3-2	Q1P13HV3198D (Q1P13V281)	CTMT PURGE DMPRS (HS-3198)
3-3	Q1P13HV2866C (Q1P13V301)	MINI-PURGE SUPPLY DAMPER
3-4	Q1P13HV2867C (Q1P13V303)	MINI-PURGE EXHAUST DAMPER
4-1	Q1N25HV3772A	CHEM ADD TO 1A SG ISO (BOP)
4-2	Q1N25HV3772B	CHEM ADD TO 1B SG ISO (BOP)
4-3	Q1N25HV3772C	CHEM ADD TO 1C SG ISO (BOP)
4-4	Q1P17HV3443	CCW FROM EXC LTDN/RCDT HXS
5-1	Q1E21MOV8112	RCP SEAL WTR RTN ISO
5-2	Q1E21HV8149A	LTDN ORIF ISO 45 GPM
5-3	Q1E21HV8149B	LTDN ORIF ISO 60 GPM
5-4	Q1E21HV8149C	LTDN ORIF ISO 60 GPM
6-1	Q1G21LCV1003	RCDT LCV
6-2	Q1G21HV3377	CTMT SUMP DISCH (BOP)

1. RMS-40305A07 031

A LOCA has occurred on Unit 2. A review of the radiation monitor trends for RE-11 and RE-12 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 towards a background count rate?

A✓ RE-11 and RE-12 are isolated from containment by a 'T' signal.

B. RE-11 and RE-12 are isolated from containment by a 'P' signal.

C. RE-11 and RE-12 are isolated from containment by a 'S' signal.

D. RE-11 and RE-12 are isolated from containment by a Containment Ventilation Isolation signal.

EEP-0 ATTACHMENT 2

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.

Rewrote the question and modified the distractors - tgb 4/30/09

HLT-28 AUDIT EXAM