INITIAL SUBMITTAL

SURRY SEPTEMBER/2000 EXAM
50-280/2000-301

SEPTEMBER 14 - 21, 2000

INITIAL SUBMITTAL
RO/SRO WRITTEN EXAMINATION

QUESTION 1: (1.0)

Given the following plant conditions:

- RHR is being warmed up.
- PRZR level is at 28%, with charging flow control in manual.
- The "A" RHR heat exchanger develops a 10 gpm tube leak.
- Heat exchanger bypass flow controller FCV-1605 is in Automatic.

Which ONE of the following is correct concerning the indications you would expect for this condition?

a. CC Head tank level decreases.

b. PRZR level decreases.

c. Indicated RHR flow increases.

d. RHR pump amps decrease.

ANSWER: b

[RO: Tier 2/Group 3]
[SRO: Tier 2/Group 3]

Answer correct: RHR pump discharge pressure is greater than the pressure of CC in the RHR Heat Exchanger. A tube leak will result in RHR flow into the CC system, PRZR Level will decrease and CC head tank level will increase.	Distractors plausible: a – CC pump discharge pressure is nearly as high as RHR pump discharge pressure. C – tube leak provides an additional flow path for RHR system flow. D – FCV-1605 will close to maintain a constant flow rate.	Distractors incorrect: a – RHR pump discharge pressure is greater than the pressure of CC in the RHR Heat Exchanger; a tube leak will result in RHR flow into the CC system. C – FCV-1605 will modulate to maintain indicated RHR flow constant. D – RHR pump amps will remain constant as FCV-1605 modulates to maintain RHR flow constant.
K/A: SYS005.K6.03	Objective: 1910	Source: New
Reference: ND-88.5-LP-1	Level: Comprehension	

COMPONENT COOLING SYSTEM LOADS

CARF/NST

17. Containment Instrument Air Compressor
18. Containment Air Recirc Fan Coolers
19. Neutron Shield Tank Coolers

CRDM Shroud Cooling/RCP

19. Shroud Cooling Coils
21. RCP Thermal Barrier Heat Exchangers †
22. RCP Motor Air Coolers
23. RCP Bearing Lube Oil Coolers

Hot Pipe Containment Penetration Cooling (>150^{0}F)

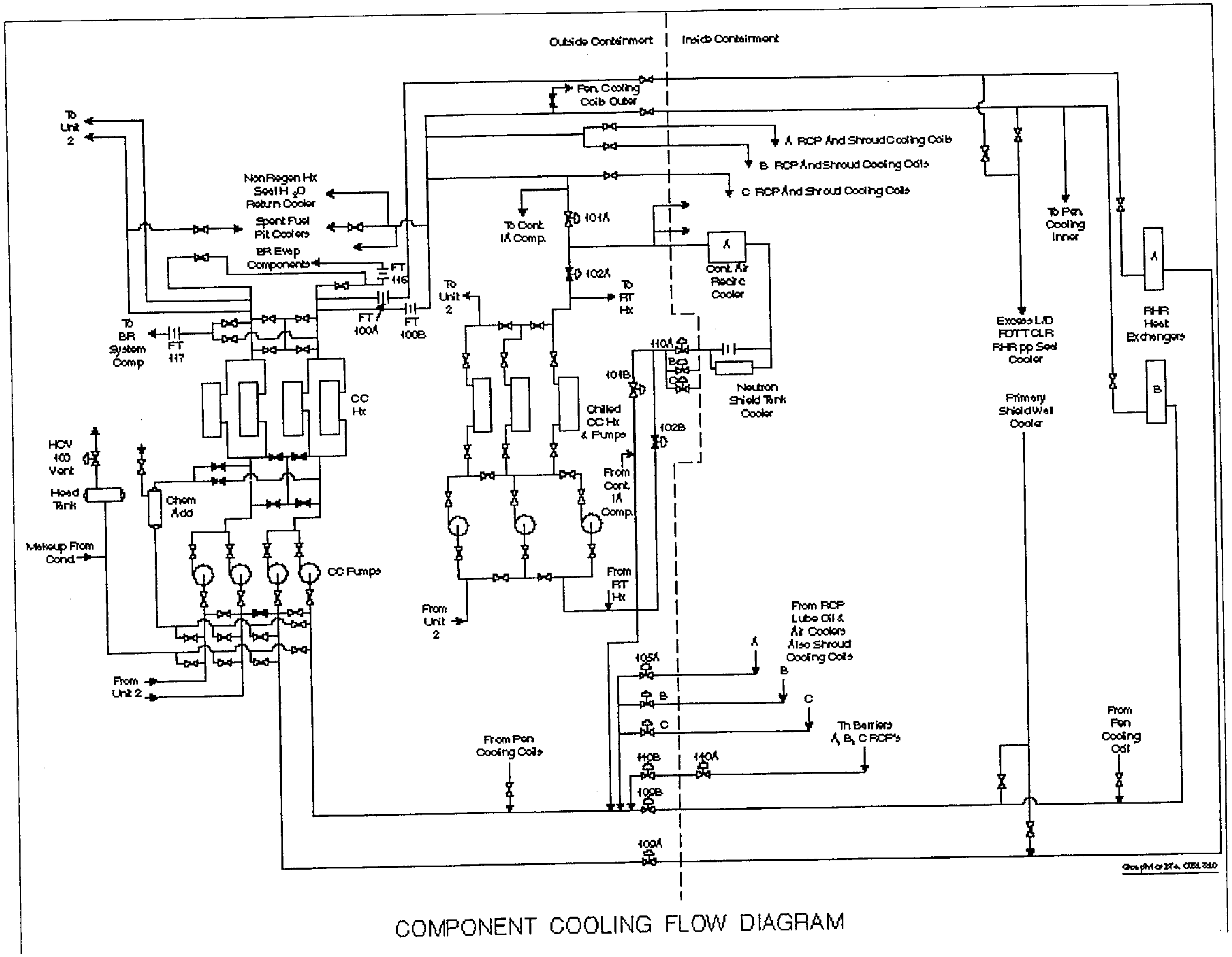
24. Containment Penetration Coolers
    a. Letdown
    b. Blowdown
    c. Main Steam
    d. Main Feed

Excess Letdown/RHR

25. Excess Letdown Heat Exchanger †
26. Primary Drains Cooler
27. RHR Heat Exchanger †
28. RHR Pump Seals †
29. Primary Shield Wall Coolers - for each loop penetration

Notes:

† Possible source of leakage into Component Cooling



COMPONENT COOLING FLOW DIAGRAM

QUESTION 2: (1.0)

Given the following plant conditions:

- Unit 1 core off-load is in progress.
- The Fuel Building supervisor informs the control room that an irradiated fuel assembly has become separated from the top nozzle and fallen on top of the fuel racks.
- Fuel building radiation monitors are in alarm.

In accordance with 0-AP-22.00, Fuel Handling Abnormal Conditions, the control room team should immediately__________.

a. actuate dumping of control room bottled air.

b. align fuel building exhaust through the iodine filters.

c. evacuate the fuel building **only** if radiation levels exceed 1 Rem/hour.

d. retrieve the fuel transfer cart and close the fuel transfer tube gate valve.

ANSWER: a

[RO: Tier 1/Group 3]
[SRO: Tier 1/Group 3]

Answer correct: per 0-AP-22, this is the only action of those listed that is required.	Distractors plausible: b – aligning fuel building exhaust through the filters would prevent release; c – this action is required when a malfunction of the SFP cooling system occurs, including loss of SFP level; d– this would prevent movement of radioactive material from the SFP to the cavity.	Distractors incorrect: b – fuel bldg exhaust doesn't need to be aligned to filters unless high radiation exists; c – fuel building should be evacuated regardless of radiation levels; d – this action is not required, per 0-AP-22.
K/A: APE036-AK1.01	Objective: 2505	Source: New
Reference: 0-AP-30, 0-AP-27	Level: Knowledge	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1]	__CHECK FUEL REPAIR - IN PROGRESS	GO TO Step 4.
[2]	__CHECK LOCAL RADIATION CONDITIONS - NORMAL	GO TO Step 4.
[3]	__GO TO STEP 18	
[4]	__STOP FUEL HANDLING OPERATIONS	
[5]	__EVACUATE THE AFFECTED AREA • Containment <u>OR</u> • Fuel Building	
[6]	__SECURE NORMAL MCR VENTILATION a) Close 1-VS-MOD-103C b) Close 1-VS-MOD-103D c) Verify stopped or stop 1-VS-F-15 d) Verify stopped or stop 1-VS-AC-4	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[7]	__DUMP MCR BOTTLED AIR:	
	a) Close 1-VS-MOD-103B (Dumps Unit 1 Cable Vault air bottles)	
	b) Set timer for 60 minutes	
	c) Check positive pressure of 0.05 inches - BEING MAINTAINED • PDI-VS-110 • PDI-VS-101 • PDI-VS-200 • PDI-VS-201	c) Close 1-VS-MOD-103A. (Dumps MER 3 air bottles)
	d) Check all Main Station Batteries - FRESHENING CHARGE IN PROGRESS	d) GO TO Step 8.
	e) Notify Electrical Department that Battery Room must be monitored for explosive concentration	
* 8.	__CHECK FUEL HANDLING ACCIDENT - IN PROGRESS FOR ONE HOUR (WHEN TIMER GOES OFF)	Do the following: a) <u>WHEN</u> Fuel Handling accident has been in progress for one hour (when timer goes off), <u>THEN</u> immediately perform Step 9. b) GO TO Step 10.

QUESTION 3: (1.0)

Given the following plant conditions:

- Reactor power is 20%, increasing to 100%.
- Annunciator G-B5, COMPU PRINTOUT ROD CONT SYS, is inoperable.
- Rod height is initially 140 steps on "D" bank.
- As rods are withdrawn, a blown fuse results in one immovable rod—F-6 in "D" bank (located near N-43).

If power ascension continues, quadrant power tilt ratio will __________ as indicated by __________ as rods are fully withdrawn.

a. decrease; N-43 Delta Flux meter

b. increase; N-43 Delta Flux meter

c. increase; annunciator G-C4, UPPER ION CHAMBER OR AUTO DEFEAT < 50% remaining lit as power exceeds 50%

d. decrease; annunciator G-D4, LOWER ION CHAMBER OR AUTO DEFEAT < 50% clearing as power exceeds 50%

ANSWER: c

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 1]

Answer correct: F-6 control rod will suppress flux in the N-43 quadrant; the upper detector will deviate from the average and the alarm will stay lit above 50% power.	Distractors plausible: a-misconception regarding axial flux vs. radial flux; b-QPTR will increase; d-annunciator normally clears above 50% power	Distractors incorrect: a-QPTR will increase and Delta Flux meters don't indicate QPTR; b-Delta Flux meters don't indicate QPTR; d-QPTR will increase and alarm will remain lit
K/A: APE005-AK1.02	Objective: 2559	Source: New
Reference: ARP G-C4, ND-93.2-LP-4	Level: Comprehension	

VIRGINIA POWER
SURRY POWER STATION

ANNUNCIATOR RESPONSE PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1G-C4	UPPER ION CHMBRS OR AUTO DEFEAT 50%	1
		PAGE 1 of 2

REFERENCES

1G-20

1. UFSAR - Section 7.0
2. Tech Spec 3.12
3. 11448-ESK-10G, 10AW

PROBABLE CAUSE

1. Alarm actuates when relay NIS-IV-K601 or NIS-IV-K602 senses core quadrant flux tilt greater than 2% OR power less than 50%.

2. Instrumentation failure has occurred.

APPROVAL RECOMMENDED	APPROVED	DATE
REVIEWED	CHAIRMAN STATION NUCLEAR SAFETY AND OPERATING COMMITTEE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	__CHECK REACTOR POWER - GREATER THAN OR EQUAL TO 50%	Exit procedure.
2.	__CALCULATE TILT USING UPPER DETECTOR NORMALIZED POWER RANGE CURRENTS - GREATER THAN 1.02	Do the following: a) Place UPPER SECTION switch on MISC CONT AND INDICATION PANEL to defeat alarming channel. b) Initiate calculation of core quadrant power tilt every 8 hours. c) Initiate a Work Request.
3.	__NOTIFY SHIFT SUPERVISOR	
4.	__REVIEW TECH SPEC 3.12	

- END -

Operations/Station Management, so that the situation can be considered without the demands of operating or supervising a unit continuously interrupting the decision process. Simply put, use the resources that are available and ask for assistance when it is necessary.

D. Delta Flux

Refer to/display H/T-4.6, Power Range NIS (Bottom Drawer).

1. Using the currents from the upper and lower detectors for each channel, subtract the bottom current from the top current, and this will provide an indication of <u>how</u> the power is distributed in the core.

Write on chalkboard: df = (Pt - Pb) x 100%

2. When the power is distributed evenly, delta flux is zero. When more power is being produced in the top than in the bottom of the core, delta flux is positive. With more power being produced in the bottom, delta flux is negative.

3. Target values for delta flux are determined periodically by station engineering. The reactor then must abide by restrictions placed on operating delta flux near the target value.

4. Tech Specs provides the following statements and limitations for delta flux:

   a. The reference delta flux target at power level Po is that delta flux with the core in equilibrium Xenon conditions (small or no oscillation) and the rods > 190 steps. The target flux difference shall be measured at least once per equivalent full power quarter. The target must be updated each effective full power month by either actual measurement or prediction calculations.

Refer to/display H/T-4.7, Delta Flux Limitation Graph (TS-3.12).

b. The actual delta flux shall be maintained within a + or - 5% band of the target.

c. If > 90% and go out of band, within 15 minutes either return within band or reduce power to < 90%.

d. At < 90%, delta flux may be out for a MAX of 1 hour in any 24 hour period, provided not in "unacceptable" area. A 1:1 penalty when > = 50% power.

e. If out of band for more than 1 hour in 24, reduce power to < 50% within 30 minutes and reduce the hi flux trip to ≤ 55% within the next 4 hours.

f. Power increases to > 90% depend on delta flux being in band.

g. At power ≤ 50%, delta flux may deviate from its target band. Power increases to > 50% depend on delta flux not being outside of band for more than 1 hour accumulated penalty during the preceding 24 hours. A 1 for 2 penalty is assessed for being outside band at powers of 15 - 50%.

h. No penalty minutes are assessed if outside of band when power is < 15%.

E. Quadrant Power Tilt

1. The symmetry of the radial power distribution is measured by the term Quadrant Power Tilt Ratio, QPTR. QPTR is defined in Technical Specifications in two ways:

Write on chalkboard the following definitions:

a. $$QPTR = \frac{maximum\ upper\ detector\ current}{average\ of\ upper\ detector\ currents}$$

OR

b. $$QPTR = \frac{maximum\ lower\ detector\ current}{average\ of\ lower\ detector\ currents}$$

2. In addition to being monitored by the Miscellaneous Indication and Control Drawer, the QPTR may be manually calculated by the operator.

Refer to QPTR Worksheet.

This form is used to calculate the QPTR using the outputs from the NI Drawers. The QPTR is normally performed using the computer program for QPTR. The method the computer uses and the worksheet method are the same.

Ensure the trainees are aware that they must be able to perform a manual QPTR without use of a computer.

Have trainees perform a QPTR calculation using the following data obtained from the PR NIs and the NI INFO Book

Reading on PR drawers

N-41		N-42		N-43		N-44	
Upper	Lower	Upper	Lower	Upper	Lower	Upper	Lower
110	121.6	119.5	118.2	96.2	111	129.0	119.4

PR Normalizing Values

N-41		N-42		N-43		N-44	
Upper	Lower	Upper	Lower	Upper	Lower	Upper	Lower
110.3	118.9	121.3	115.1	98.2	108.3	130.2	121.8

Using the above date the QPTR is:

Upper: .9973/.9882 = 1.009 Lower: 1.0269/1.0138 = 1.013

3. Technical Specifications section 3.12 imposes the following limitations on QPTR values:

   a. When reactor power is > 50%, QPTR is limited to a value of less than or equal to 1.02 (values of > 1.02 are indication of a dropped, stuck, or misaligned control rod).

b. If the quadrant power tilt is > 2% when reactor power is > 50%, then the HCF must be determined within 2 hours and the power level adjusted to meet the specs, -OR- if the HCF are not determined within 2 hours, the power level and high flux trip are reduced from rated power 2% per percent of quadrant power tilt. If the quadrant power tilt exceeds 10%, the power and high flux trip shall be reduced from rated power 2% per percent of quadrant tilt.

c. If after further period of 24 hours, the tilt is not corrected to less than 2%:

(1) If design HCF for rated power are NOT exceeded, an evaluation of the discrepancy is made and reported to the NRC.

(2) If design HCF for rated power ARE exceeded and power is > 10%, the NRC shall be notified and the OP-OT delta T trip settings shall be reduced 1% per percent the HCF exceeds the design values.

(3) If HCF are NOT determined, the NRC shall be notified and the OP-OT delta T trip settings shall be reduced by 2% per percent of quadrant power tilt.

d. When reactor power is ≤ 50%, there are no limits for QPTR.

F. Miscellaneous Indication and Control Drawer

Refer to/display H/T-4.8, Misc Control and Indication Drawer, and H/T-4.9, Misc Cont and Indication PNL.

1. This drawer, powered from VB-IV, receives an input signal from the four upper section PR detectors and the four lower section PR detectors. These inputs are processed to provide indication of flux deviation between the upper or lower PR detector sections.

2. All four upper PRNIs provide input to the averaging amplifier. The comparator compares the highest upper range detector to the average upper range detector. If the peak is 2% greater than the average, it will generate the "Upper Ion Chamber Deviation" annunciator.

3. The lower channel's deviation alarm is calculated the same as the upper channel.

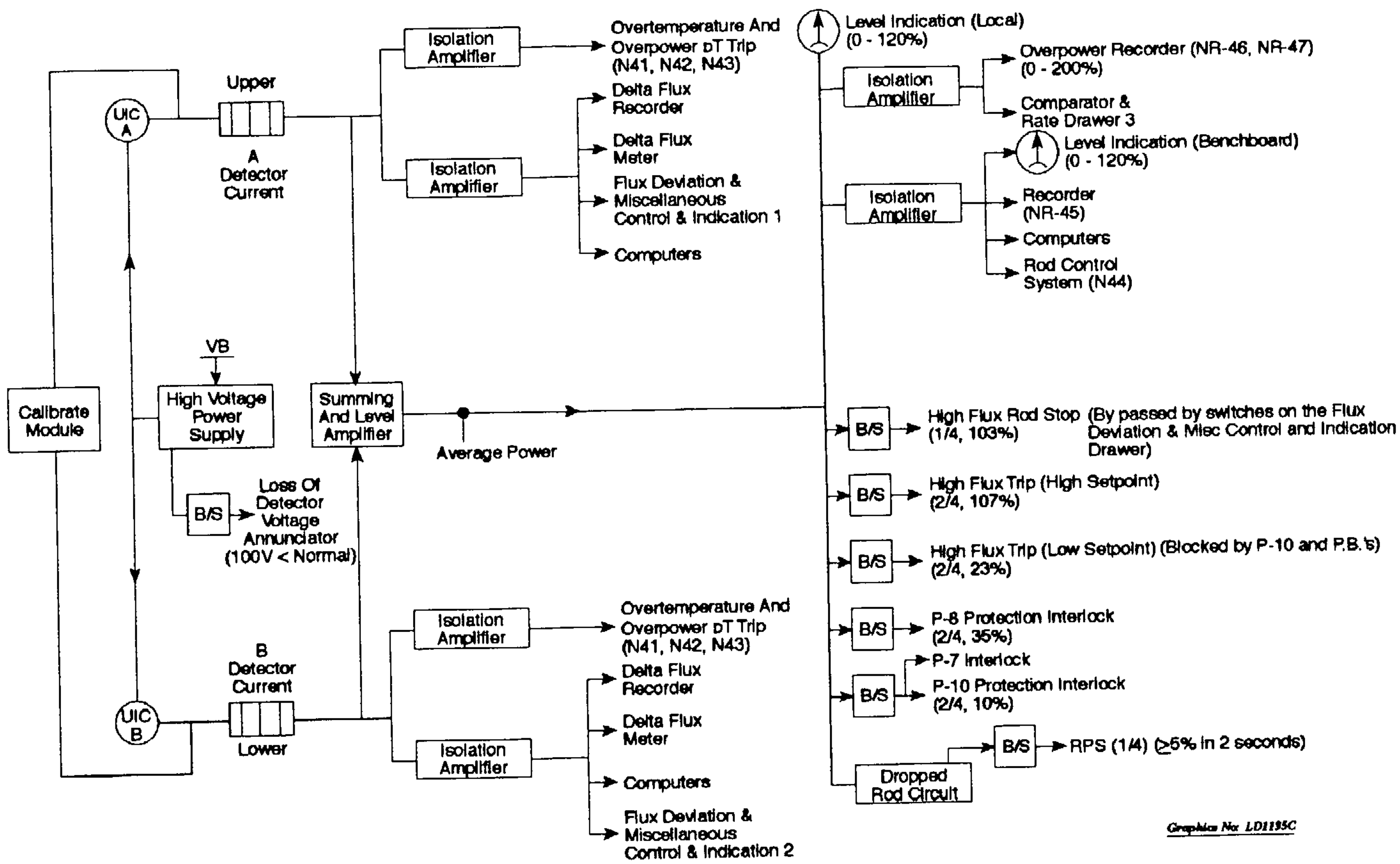
4. These deviation alarms are defeated when the averaging amplifier output is less than 50% power. The alarms can be/are defeated when less than 50% power because:

   a. When less than 50% power, the power generation is low enough so that significant power peaks which could result in exceeding Hot Channel Factors will not occur.

   b. It is defeated when <50% power to prevent nuisance alarms. For 100% power, the alarm setpoint would be 102% on the highest channel. For 50% power, the alarm setpoint would be 51% on the highest channel. For 20% power, the alarm setpoint would be 20.4% on the highest channel.

5. The input from each PRNI to the deviation alarm section can be defeated by a switch on the Misc Indication and Control Drawer.

6. The input from each PRNI to the overpower rod stop section can be defeated by another switch on the Misc Indication and Control Drawer.



Upper/Lower Ion Chamber Current to ΔT Protection System for Δφ Penalty

Overpower Rod Stop 1/4 at 103%

Overpower Rx Trip 2/4 at 107%

Overpower Rx Trip (Low) 2/4 at 23%, manually blocked by P-10

P-8 3/4 PR < 35% blocks Low Flow Trip in 1/3 Loops

P-10 2/4 PR 10%, allows manual block of overpower trip (low stpt), IR High Flux Trip and Rod Stop, Blocks reset of SR High Voltage

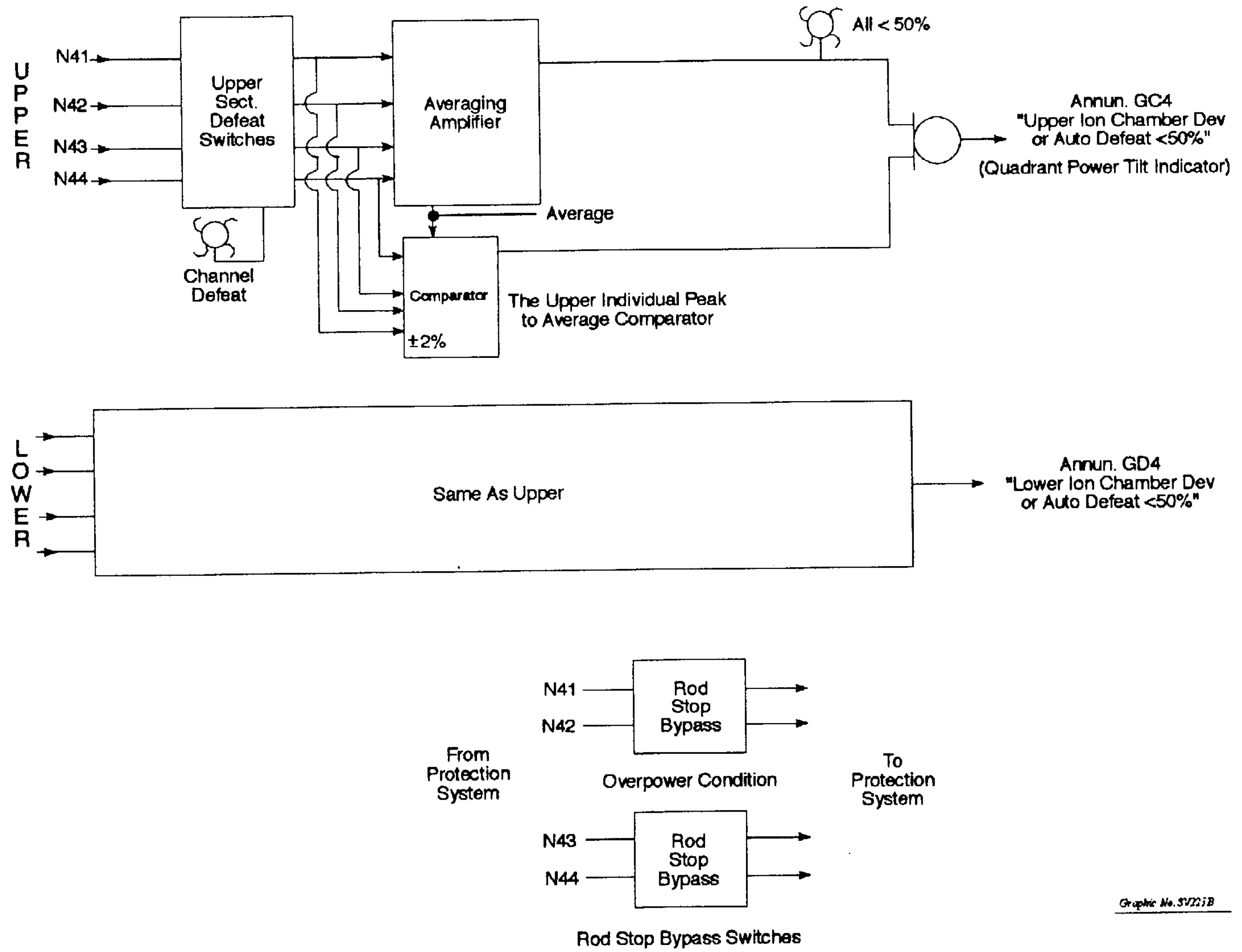
Dropped Rod (1/4) 5% reduction in 2 secs

1 - Compares Upper Detectors to the Avg of the Upper Detectors.

2 - Compares Lower Detectors to the Avg of the Lower Detectors.

3 - Compares Channel Avgs to the lowest channel.

POWER RANGE CHANNEL



MISC. CONTROL AND INDICATION DRAWER

QUESTION 4: (1.0)

Given the following plant conditions:

- Refueling is complete and RCS loops have been filled.
- The team has started "A" reactor coolant pump (RCP).
- Immediately after pump start, annunciator C-H4, RCP FRAME DANGER alarms.

In response, the team should __________.

a. immediately stop the RCP and request engineering to perform a structural integrity inspection

b. re-scale the vibration monitor and determine if readings are accurate

c. determine if vibrations are high; if so then immediately stop the RCP

d. request predictive analysis to take readings using hand-held instrumentation

ANSWER: c

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 1]

Answer correct: per C-H4, tripping the RCP should be considered upon receipt of a vibration alarm.	Distractors plausible: a – disregarding the CAUTION (and good Ops practice) this is what the ARP implies; b – rescaling is only done during initial alignment; determining if readings are accurate is correct; d – requesting PAG to take readings with a hand-held instrument is a normal evolution during startup to verify acceptable criteria.	Distractors incorrect: a – RCP vibration monitors are known to fail frequently, Ops always uses multiple indications when available prior to taking action; b – re-scaling the vibration meter requires support thus taking too long, which is how Calvert Cliffs got into trouble; d – PAG would only take data to support satisfactory operation.
K/A: APE015/017-AA1.23	Objective: 4006	Source: New
Reference: ARP C-H4	Level: Knowledge	

VIRGINIA POWER
SURRY POWER STATION

ANNUNCIATOR RESPONSE PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1C-H4	RCP FRAME DANGER	2
		PAGE 1 of 3

1C-60

REFERENCES

1. UFSAR 4.2.2.4
2. RCP Technical Manual
3. 11448-ESK-10C, 10AY
4. DCP 91-05, RTD Bypass Elimination
5. DCP 93-054-3, Reassessed CRDR - Outage Related (Relocation)
6. SER 2-97, RCP Damage from a Separated Component 2

PROBABLE CAUSE

1. Alarm actuates when RCPS VIB DET senses RCP frame vibration greater than or equal to 5 MILS.

   High vibration may be caused by one or more of the following:

   a. Misalignment of pump support framing.

   b. Excessive bearing wear.

   c. High or low level in lube oil reservoirs.

   d. Damage to pump seals.

   e. Damage to thermal barrier.

   f. Excessive wear on pump internals.

   g. Motor and pump misalignment.

2. Instrumentation failure has occurred.

APPROVAL RECOMMENDED	APPROVED	DATE
REVIEWED	CHAIRMAN STATION NUCLEAR SAFETY AND OPERATING COMMITTEE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION:
- Extremely high RCP vibrations may over-range the vibration sensor input causing the indications to fail high or low. Tripping the affected RCP should be considered upon receipt of a vibration alarm followed by indications that are off-scale high or low.
- The Reactor shall not remain critical with less than three RCPs running.
- If an RCP needs to be tripped with the Reactor critical, a Reactor trip must be performed before securing the RCP.

* *

1. __VERIFY ALARM - RCP FRAME VIBRATION GREATER THAN OR EQUAL TO 5 MILS
   - RCPS VIB DET

   Response Not Obtained: Do the following:

   a) Initiate a work request.

   b) Increase surveillance of RCP parameters:
      - Vibrations
      - Bearing temperatures
      - Motor amperage
      - CC flow
      - Seal injection and leakoff flows

   c) GO TO Step 5.

2. __MANUALLY TRIP REACTOR

3. __STOP AFFECTED RCP

4. __GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

5. __PROVIDE NOTIFICATIONS AS NECESSARY:

- OMOC
- STA
- Shift Supervisor
- Predictive Analysis
- System Engineer

- END -

QUESTION 5: (1.0)

RCS cooldown is in progress per 1-ES-0.2, Natural Circulation Cooldown.

During RCS depressurization, the team notes that subcooling has suddenly decreased from 45°F to 5°F. Per the ES-0.2 continuous action page, the team should ________.

a. initiate safety injection and transition to 1-ES-0.0, Re-diagnosis

b. initiate safety injection and transition to 1-E-0, Reactor Trip or Safety Injection

c. stop RCS depressurization and continue cooldown to restore subcooling

d. continue RCS depressurization and increase the cooldown rate to restore subcooling

ANSWER: b

[RO: Tier 1/Group 1]
[SRO: Tier 1/Group 1]

Answer correct: per 1-ES-0.2 continuous action page initiation of SI is required and transition to E-0..	Distractors plausible: a – initiating SI is correct and 20 degrees subcooling is not expected during a natural circulation cooldown, ES-0.0 is used when SI is in service and the team thinks they may be in the wrong procedure; c – stopping depressurization and continuing the cooldown will restore subcooling if nothing else is going on, this strategy is used in other EOPs; d – increasing the cooldown rate for a given depressurization rate will increase subcooling if nothing else is going on.	Distractors incorrect: b&c - per 1-ES-0.2 continuous action page, the only correct action is to initiate SI and go to 1-E-0. A – incorrect because 1-ES-0.0 will not take you to E-0.
K/A: EPE09-EA1.3	Objective: 3043	Source: New
Reference: 1-ES-0.2 continuous action page	Level: Knowledge	

CONTINUOUS ACTIONS PAGE FOR 1-ES-0.2

1. SI INITIATION CRITERIA

   Initiate SI and GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION, if EITHER condition listed below occurs:

   - RCS subcooling based on CETCs - LESS THAN 30°F [85°F]
   - PRZR level - CANNOT BE MAINTAINED GREATER THAN 11% [43%]

2. AFW SUPPLY SWITCHOVER CRITERIA

   Transfer to one of the following alternate AFW water supplies if ECST level decreases to less than 20%.

   a. 1-CN-TK-2, using 1-CN-150.

   b. 1-CN-TK-3, using AFW Booster Pumps.

   c. AFW Crosstie.

   d. Firemain.

3. RCP START CRITERIA

   - WHEN conditions in 1-OP-RC-001, STARTING AND RUNNING ANY RCP, are established THEN start one RCP and GO TO 1-GOP-2.4, UNIT COOLDOWN, HSD TO 351°F.

   - Following a loss of all seal cooling, affected RCP(s) should NOT be started without prior status evaluation.

   - RCPs should be run in the following order of priority to provide PRZR spray: C, A.

QUESTION 6: (1.0)

A reactor trip occurred and the team noted that all IRPIs indicated zero except the following:

- L-5 – 22 steps
- K-10 – 16 steps
- E-14 – 9 steps
- B-6 – 11 steps
- D-10 – 20 steps
- C-7 – 17 steps
- H-2 – 13 steps
- J-3 – 19 steps
- B-8 – 6 steps

Emergency boration via 1-CH-MOV-1350 commenced with "A" BAST level initially at 89% AND 8.1 % Boric Acid. Current RCS Concentration of Boron is 892 ppm.

Using the references provided, determine which ONE of the following is correct concerning the level in "A" BAST at which the team should stop emergency boration.

a. 28%

b. 52%

c. 58%

d. 73%

ANSWER: b

[RO: Tier 1/Group 1]
[SRO: Tier 1/Group 1]

Answer correct: per attached calculation—one rod (L-5) is >20 steps, six rods (K-10, B-6, D-10, C-7, H-2, J-3) indicate from 11 – 20 steps, total of 2 equivalent stuck rods requires 1110 gallons for each ESR, 2 X 1110 gallons = 2220 gallons. 89% or 6600 gallons – 2220 gallons = 4380 gallons or 52%.	Distractors plausible: Misconception concerning the meaning of "inclusive" could lead to incorrect determination; incorrect conversion from # of IRPIs to # of ESRs, multiple math errors possible.	Distractors incorrect: per attached calculation, the only correct answer is 52%.
K/A: APE024-AK3.02	Objective: 3249	Source: New
Reference: 1-ES-0.1, 1-DRP-003 Attachment 58	Level: Comprehension	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

10. __VERIFY ALL IRPIs - 10 STEPS OR LESS

IF two or more IRPIs indicate greater than 10 Steps, THEN do the following:

a) Verify or raise CHG flow - GREATER THAN 75 GPM

b) Align boration path:

1) Put BATP in FAST.

2) Open 1-CH-MOV-1350.

3) Verify emergency borate flow.

c) IF emergency borate flow path NOT available, THEN align alternate boration path:

• Borate using the Blender.

OR

• Manually align CHG pump suction to the RWST.

d) Record start time of emergency boration.

• ________________

e) Record in-service BAST level.

• ________________

f) Direct STA to initiate Attachment 1.

g) Initiate Shutdown Margin IAW 1-OP-RX-002, SHUTDOWN MARGIN (CALCULATED AT ZERO POWER).

NOTE: The amount of boration listed in the table in Step 4c is only an estimate of the actual boration required. The actual value is determined by calculations performed in 1-OP-RX-002. The amount of boration should be adjusted based on the SDM calculation.

___ 1. Using IRPI indication, determine the number of control rods greater than 10 Steps and complete the following table.

Actual RPI Indication	Record RPI IDs for RPIs indicating NOT fully inserted	Equivalent Stuck Rods (EQSR)	Record EQSR Subtotals:
Any Rod>20 steps		1 rod = 1 EQSR	
Rods indicating 11 - 20 (inclusive) steps withdrawn		1 to 6 rods = 1 EQSR 7 to 12 rods = 2 EQSR 13 to 31 rods = 3 EQSR 32 or more = 4 EQSR	
		Total Equivalent Stuck Rods:	

___ 2. IF only one Total Equivalent Stuck Rod was indicated in the above table, THEN stop emergency boration.

___ 3. IF two or more Equivalent Stuck Rods were indicated in the above table, THEN continue with emergency boration IAW Step 4.

___ 4. Use the table below to borate 200 ppm for each equivalent stuck rod as determined in Step 1.

___ a. Check current RCS boron concentration. Use the range within which the current ppm reading falls.

___ b. Check the BAST boron concentration for the in-service tank. Use the range within which the current concentration falls.

___ c. Multiply the value listed (gallons) by the number of equivalent stuck rods determined in Step 1 and borate that amount.

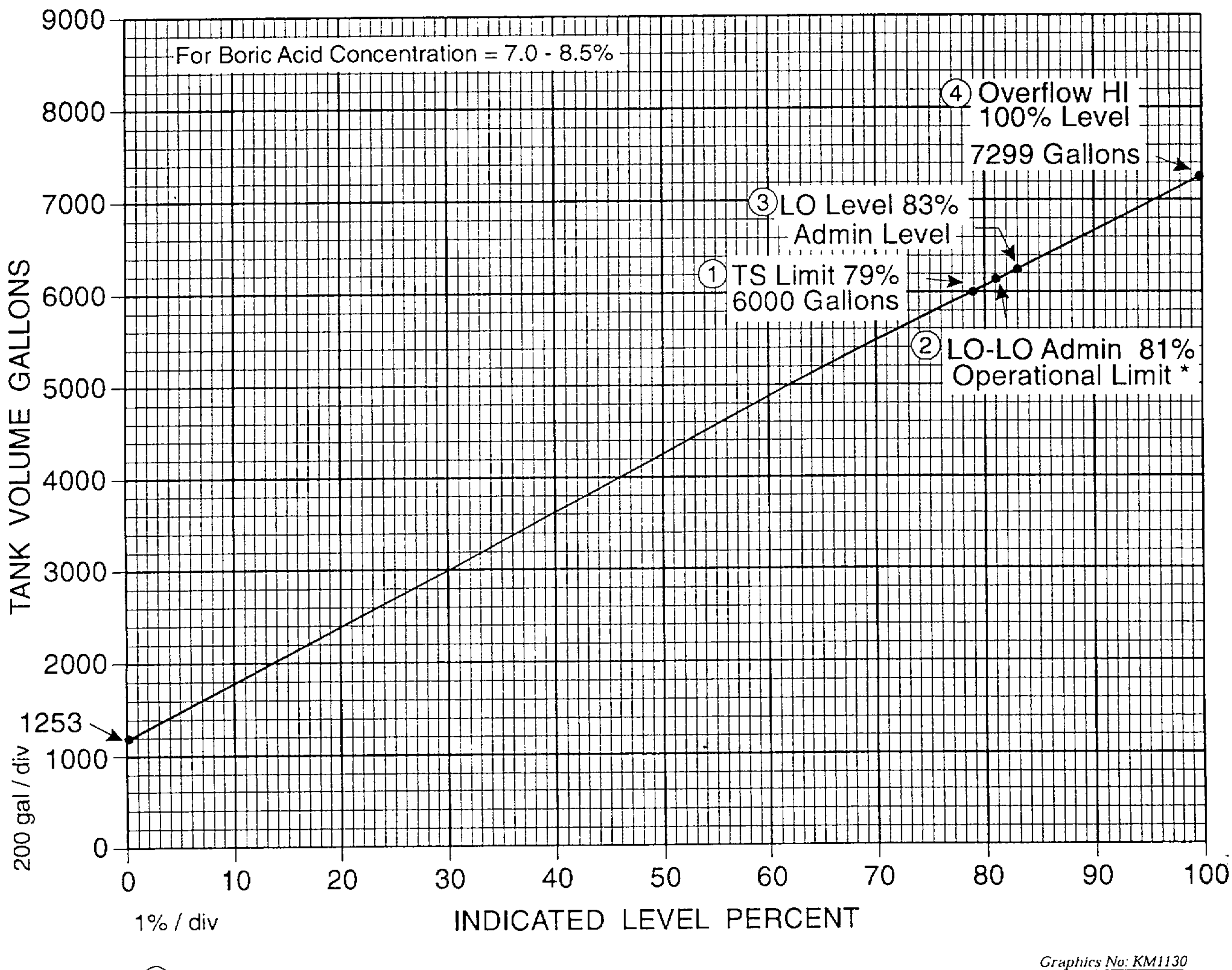
CURRENT RCS BORON (PPM)	BAST 7.0-7.24	BAST 7.25-7.49	BAST 7.50-7.7.4	BAST 7.75-7.99	BAST 8.0-8.24	BAST 8.25-8.5
0-100	1198	1156	1117	1080	1046	1014
101-200	1208	1165	1126	1088	1054	1021
201-300	1218	1175	1134	1097	1061	1028
301-400	1229	1184	1143	1105	1069	1036
401-500	1239	1194	1153	1114	1077	1043
501-600	1250	1204	1162	1122	1085	1051
601-700	1261	1214	1171	1131	1094	1059
701-800	1272	1225	1181	1140	1102	1066
801-900	1283	1235	1191	1149	1110	1074
901-1000	1295	1246	1201	1158	1119	1082
1001-1100	1307	1257	1211	1168	1128	1091
1101-1200	1318	1268	1221	1177	1137	1099
1201-1300	1331	1279	1231	1187	1146	1108
1301-1400	1343	1290	1242	1197	1155	1116
1401-1500	1356	1302	1253	1207	1164	1125

___ d. Adjust boration amount based on the value calculated in 1-OP-RX-002.

ATTACHMENT 58

(Page 1 of 1)

BORIC ACID STORAGE TANKS



1 TS Limit (6000 gallons)	79%
2 LO-LO Admin Operational Limit	81%
3 LO Level Admin Limit	83%
4 Overflow High Level	100%

* 81% ~ TS Limit with Gauge Inaccuracy Minimum Operational Limit

BORIC ACID STORAGE TANKS
1-CH-TK-1A, B, C

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.	___VERIFY ALL IRPIs - 10 STEPS OR LESS	<u>IF</u> two or more IRPIs indicate greater than 10 Steps, <u>THEN</u> do the following: a) Verify or raise CHG flow - GREATER THAN 75 GPM b) Align boration path: 1) Put BATP in FAST. 2) Open 1-CH-MOV-1350. 3) Verify emergency borate flow. c) <u>IF</u> emergency borate flow path <u>NOT</u> available, <u>THEN</u> align alternate boration path: • Borate using the Blender. <u>OR</u> • Manually align CHG pump suction to the RWST. d) Record start time of emergency boration. • ______________ e) Record in-service BAST level. • ______________ f) Direct ~~STA~~ to initiate Attachment 1. g) Initiate Shutdown Margin IAW 1-OP-RX-002, SHUTDOWN MARGIN (CALCULATED AT ZERO POWER).

NUMBER 1-ES-0.1	ATTACHMENT TITLE	REVISION 24
ATTACHMENT 1	GALLONS OF BORIC ACID NEEDED TO INCREASE RCS BORON BY 200 PPM	PAGE 1 of 2

NOTE: The amount of boration listed in the table in Step 4c is only an estimate of the actual boration required. The actual value is determined by calculations performed in 1-OP-RX-002. The amount of boration should be adjusted based on the SDM calculation.

✓ 1. Using IRPI indication, determine the number of control rods greater than 10 Steps and complete the following table.

Actual RPI Indication	Record RPI IDs for RPIs indicating NOT fully inserted	Equivalent Stuck Rods (EQSR)	Record EQSR Subtotals:
Any Rod>20 steps	1	1 rod = 1 EQSR	1
Rods indicating 11 - 20 (inclusive) steps withdrawn	6	1 to 6 rods = 1 EQSR 7 to 12 rods = 2 EQSR 13 to 31 rods = 3 EQSR 32 or more = 4 EQSR	1
		Total Equivalent Stuck Rods:	2

N/A 2. IF only one Total Equivalent Stuck Rod was indicated in the above table, THEN stop emergency boration.

✓ 3. IF two or more Equivalent Stuck Rods were indicated in the above table, THEN continue with emergency boration IAW Step 4.

___ 4. Use the table below to borate 200 ppm for each equivalent stuck rod as determined in Step 1. =400ppm

✓ a. Check current RCS boron concentration. Use the range within which the current ppm reading falls. 892

√ b. Check the BAST boron concentration for the in-service tank. Use the range within which the current concentration falls. 8.1%

___ c. Multiply the value listed (gallons) by the number of equivalent stuck rods determined in Step 1 and borate that amount. 1110 x 2 = 2220

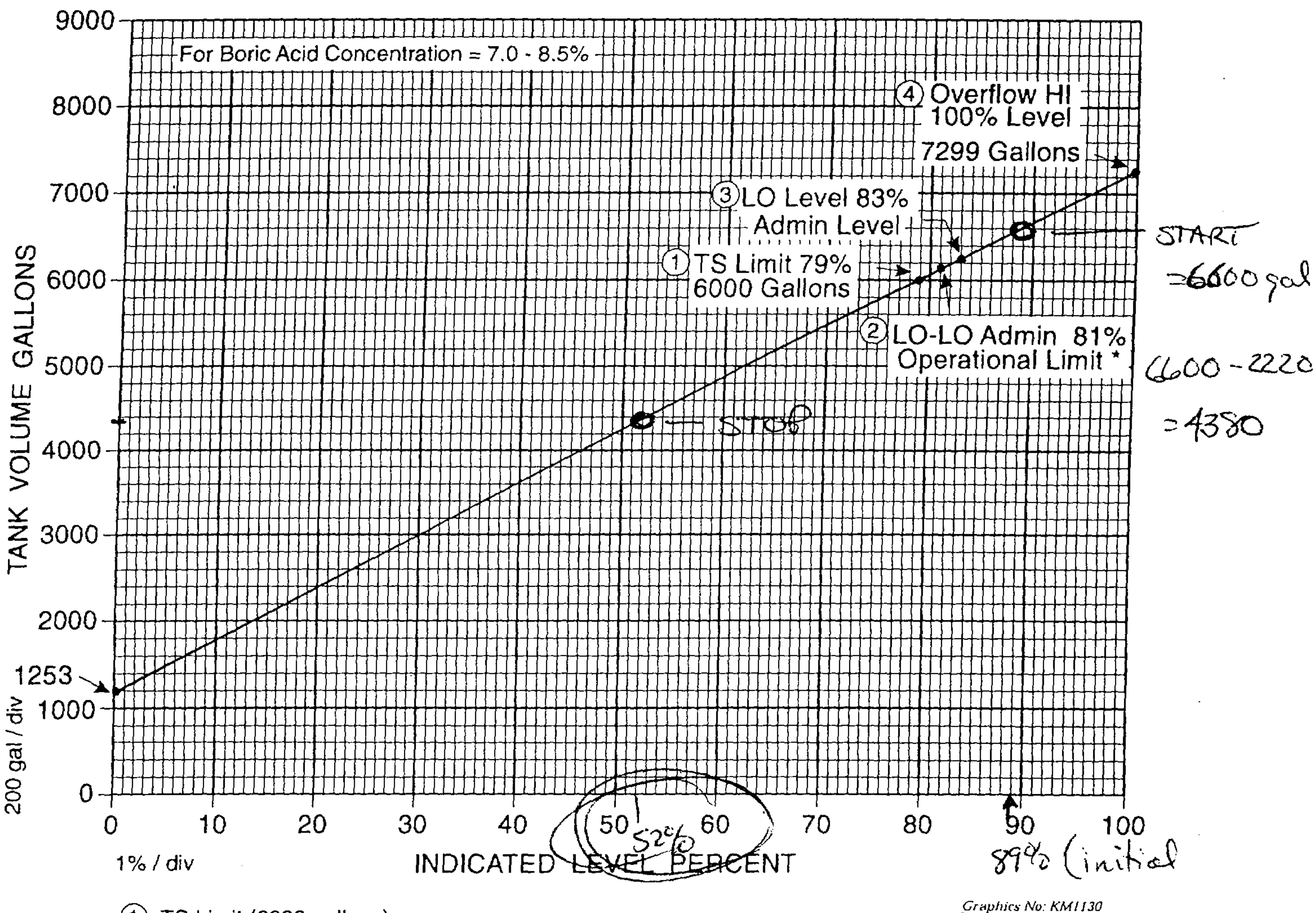
CURRENT RCS BORON (PPM)	BAST 7.0-7.24	BAST 7.25-7.49	BAST 7.50-7.7.4	BAST 7.75-7.99	BAST 8.0-8.24	BAST 8.25-8.5
0-100	1198	1156	1117	1080	1046	1014
101-200	1208	1165	1126	1088	1054	1021
201-300	1218	1175	1134	1097	1061	1028
301-400	1229	1184	1143	1105	1069	1036
401-500	1239	1194	1153	1114	1077	1043
501-600	1250	1204	1162	1122	1085	1051
601-700	1261	1214	1171	1131	1094	1059
701-800	1272	1225	1181	1140	1102	1066
801-900	1283	1235	1191	1149	1110	1074
901-1000	1295	1246	1201	1158	1119	1082
1001-1100	1307	1257	1211	1168	1128	1091
1101-1200	1318	1268	1221	1177	1137	1099
1201-1300	1331	1279	1231	1187	1146	1108
1301-1400	1343	1290	1242	1197	1155	1116
1401-1500	1356	1302	1253	1207	1164	1125

___ d. Adjust boration amount based on the value calculated in 1-OP-RX-002.

ATTACHMENT 58

(Page 1 of 1)

BORIC ACID STORAGE TANKS



①	TS Limit (6000 gallons)	79%
②	LO-LO Admin Operational Limit	81%
③	LO Level Admin Limit	83%
④	Overflow High Level	100%

* 81% ~ TS Limit with Gauge Inaccuracy Minimum Operational Limit

BORIC ACID STORAGE TANKS
1-CH-TK-1A, B, C

QUESTION 7: (1.0)

Given the following plant conditions:

- Reactor startup is in progress.
- Shutdown banks have been withdrawn.
- Control bank "D" is at 150 steps.
- The reactor is at 5% power controlling on the steam dumps.
- Multiple failures result in RCS pressure increasing to 2750 psig.

What is the required team response required by Technical Specifications?

a. Be in hot shutdown within 6 hours.

b. Be in cold shutdown in 30 hours.

c. Be in hot shutdown within 1 hour.

d. Suspend the startup until the pressure is below 2235 psig.

ANSWER: c

[RO: Tier 1/Group 1]
[SRO: Tier 1/Group 2]

Answer correct: per TS-2.2 definition of operational modes. Per T.S. section 6.3 Actions to be taken if a safety limit is exceeded.	Distractors plausible: a / b– This is the generic T.S. response for violation of limits. d – this is correct if we do not exceed the safety limit.	Distractors incorrect: a – The requirement is one hour to HSD. b – the requirement is hot shutdown no cold shutdown which cold appy more stess to the reactor vessel. D – the unit must be place in HSD so the startup must be terminated.
K/A: 000027/GEN-2.1.11	Objective: 2610	Source: New
Reference: TS-2.2 and TS 6.3 ND-93.3-LP-15	Level: Comprehension	

a. This Table includes instrumentation that provides input to Safety Injection Logics, MD and TD AFW pumps, Loss of power (DV and UV), Intake Canal Level, 2000 psig Pzr Press Interlock (P-11), Hi Stm flow SI input from T_{avg} (P-12), and 1/2 Rx Trip Bkrs open interlock (P-4).

b. Actions to be taken for instrument failures are located in statements following the Table.

c. Interlock functions (i.e., P-6, P-10, P-7, etc.) are described in TS Table 4.1a.

> NOTE: Point out to the trainees, using the following examples, that they must be thorough when using TS Tables.
>
> Items affected by a Pressurizer Pressure Channel failure are listed in both table 3.7-1 and 2, and would affect a permissive listed in Table 3.7-1.
>
> Items affected by a Steam Channel failure are listed in Table 3.7-1 and 2.

C. Technical Specifications Section 6.3

1. This specification states the required action to be performed if a T.S. Safety Limit is exceeded.

2. If the limit is exceeded, the following must be performed:

   a. Place the Reactor in HSD within 1 hour.

   b. SLV reported to V.P.-Nucl Ops, NRC and MSRC within 24 hours

   c. SLV report written and submitted within 14 days to the groups above.

2.2 SAFETY LIMIT, REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the maximum limit on Reactor Coolant System pressure.

Objective

To maintain the integrity of the Reactor Coolant System.

Specification

The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

Basis

The Reactor Coolant System[1] serves as a barrier which prevents radionuclides contained in the reactor coolant from reaching the environment. In the event of a fuel cladding failure the Reactor Coolant System is the primary barrier against the release of fission products. The maximum transient pressure allowable in the Reactor Coolant System pressure vessel under the ASME Code, Section III is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USAS Section B31.1 is 120% of design pressure. Thus, the safety limit of 2735 psig (110% of design pressure) has been established.[2]

The nominal settings of the power-operated relief valves at 2335 psig, the reactor high pressure trip at 2385 psig and the safety valves at 2485 psig are established to assure never reaching the Reactor Coolant System pressure safety limit. The initial hydrostatic test has been conducted at 3107 psig to assure the integrity of the Reactor Coolant System.

1) UFSAR Section 4

2) UFSAR Section 4.3

6.3 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

Specification

A. The following actions shall be taken in the event a Safety Limit is violated:

1. The facility shall be placed in at least hot shutdown within 1 hour.

2. The Safety Limit violation shall be reported to the Commission, the Vice President - Nuclear Operations, and the MSRC within 24 hours.

3. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSOC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

4. The Safety Limit Violation Report shall be submitted to the Commission, the Vice President - Nuclear Operations, and the MSRC within 14 days of the violation.

QUESTION 8: (1.0)

Given the following plant conditions:

- A main steam break occurred with the unit at 100% power.
- Safety injection actuated and all equipment functioned correctly.
- The RO noted that control rod H-14 IRPI indicates 226 and all other rod bottom lights are illuminated.

Which ONE of the following identifies the required team response?

a. Initiate Emergency boration.

b. No additional actions are required.

c. Rack out both rod drive MG set supply breakers.

d. Attempt to manually insert the rod.

ANSWER: b

[RO: Tier 1/Group 1]
[SRO: Tier 1/Group 1]

Answer correct: accident analyses accounts for the most reactive rod being stuck fully withdrawn.	Distractors plausible: a – with no SI flow, emergency boration is required for a stuck rod; c – This action is performed for a reactivity issue concerning loss of monitoring (AP-4.00, Loss of Source Range Detectors); d – it would be logical to attempt manual insertion of a stuck rod.	Distractors incorrect: a – emergency boration flow is ineffective with SI in service and is not necessary for the conditions stated; c – Not required by E-0, and performing a shutdown margin calculation would not **mitigate** the stuck rod; d – no guidance exists for manually inserting a stuck rod.
K/A: APE040-AA1.18	Objective: 2863	Source: New
Reference: ND-95.2-LP-3	Level: Comprehension	

10. UFSAR, part 14B, provides an analysis of high energy line breaks in various areas throughout the plant. As a part of this analysis, a normal feedwater line break in the Main Steam Valve house resulting in the loss of all normal feed and, due to the release of high energy steam/water mixture, a loss of all Auxiliary feedwater, was found to be a limiting event. This event is assumed to occur at 100% power, with a reactor trip occurring on low-low steam generator level (11%). Based on the results of the analysis, **Operator Action** must be taken within 10 minutes of the start of the event to secure the RCPS and align cross-tie from the opposite unit (only 1 MDAFW pump is assumed as available, providing 300-350 gpm flow), to prevent a loss of heat sink event leading to the following:

- RCS and Main Steam overpressure
- Steam generator dryout
- Possible adverse effects on Fuel Clad Integrity.

By performing these actions within the required time, the steam generators are maintained as a heat sink and the limitations for feeding a hot/dry SG and entering a condition requiring bleed and feed are avoided.

11. The UFSAR was revised in 1985 to reflect the reanalysis of the MS line break due the addition of the steam flow restricting orifices and the removal of the Boron Injection Tank (BIT) from the SI system. In addition, the analysis was veriified in 1991 with a change to Tech Spec Section 4.7, which delineates MSTV closure testing.

Instructor Note: High Steam flow SI signal closes the MSTVs and opens the RWST cross-tie valves. The RWST cross-tie valves are included in this signal because it has been postulated that a break in SFGDs could endanger pump suction piping from the RWST that runs under the SFGDs floor (the floor would have to be modified to ensure line protection).

12. The safety analysis of the main steam pipe rupture is designed to demonstrate that:

a. Assuming a stuck control rod with or without off-site power, and assuming a single failure in the ESF systems, there is no consequential damage to the primary system and the core remains in place and intact.

b. With no stuck rod assembly, and all equipment operating at design capacity, insignificant (or no) cladding rupture occurs.

c. There will be no DNB or clad perforation resulting from any single active failure in the main steam system. The single, active failure is the opening, with failure to close, of the largest of any single steam bypass, relief, or safety valve.

13. UFSAR states that although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the analysis shows that no DNB occurs for any rupture event, even if the most reactive control rod assembly is stuck in its fully withdrawn position.

14. It should be noted that, following a main steam line break, only one SG blows down completely. Thus, two SGs are still available for the dissipation of decay heat after the initial transient is over.

15. No radioactivity is released to the environment because of a steam line break unless there is or has been primary to secondary leakage in a steam generator. UFSAR analyzed the release potential for a MS line break with 1% failed fuel and a 10 gpm primary to secondary leak. The resulting dose value is 12.3 Rem to the thyroid at the site boundary. This thyroid dose is considerably less than the 10 CFR 100 guidelines.

Distribute AIA-3.1, Plant Response - Secondary Breaks, instruct trainees to review.

Distribute AIA-3.2, INPO Case Study 87-011. Direct the trainees to read the case study.

QUESTION 9: (1.0)

Given the following plant conditions:

- The unit has been placed on line.
- The turbine is at 15% power.
- Annunciator F-B6, TURB LO VAC, alarmed ten minutes ago.
- The reactor operator observes main condenser vacuum at 24.8" Hg and degrading.

In accordance with 1-AP-14.00, Loss of Main Condenser Vacuum, if vacuum can not be recovered the team should __________.

a. reduce turbine load until vacuum is stable

b. initiate 1-AP-23.00, Rapid Load Reduction

c. trip the turbine while continuing with 1-AP-14.00

d. trip the reactor and go to 1-E-0 while continuing with 1-AP-14.00

ANSWER: d

[RO: Tier 1/Group 1]
[SRO: Tier 1/Group 1]

Answer correct: per 1-AP-14, step 4, the reactor must be tripped if condenser pressure is < 26.5" Hg abs and power is < 30%.	Distractors plausible: a – this action is required if vacuum hasn't already degraded beyond the setpoint that requires tripping the reactor and turbine b – this is an option if condenser vacuum hasn't already degraded beyond the setpoint that requires tripping the reactor and turbine; c – tripping the turbine without tripping the reactor is possible with power below 10% (P-7).	Distractors incorrect: a & b – the turbine stall flutter limit was violated almost 30 minutes ago; therefore, a turbine trip is required. c – tripping the turbine is will cause a reactor trip which is unnecessarily challenging the unit.
K/A: APE051-AA2.02	Objective: 2830	Source: New
Reference: 1-AP-14.00	Level: Comprehension	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

<u>CAUTION</u>:
- To prevent Turbine damage from turbine stall flutter, Main Condenser vacuum must be:
  - maintained greater than 26.5 in-Hg when Turbine power is less than or equal to 30%.
  - maintained greater than 24.5 in-Hg when Turbine power is greater than 30%.
- If vacuum can <u>NOT</u> be recovered within five minutes based on the above parameters, the Turbine <u>must</u> be taken off line. The five minute limitation may be exceeded with Shift Supervisor approval if vacuum is recovering.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 1.	__CHECK TURBINE POWER - GREATER THAN 30%	GO TO Step 13.
* 2.	__CHECK MAIN CONDENSER VACUUM - GREATER THAN 24.5 IN-HG • CN-PR-101A • CN-PR-101B	Do the following: a) Place the Condenser Hoggers in service IAW Attachment 1. b) Initiate a Turbine ramp IAW Attachment 2. c) <u>IF</u> vacuum can <u>NOT</u> be recovered, <u>THEN</u> do the following: 1) Trip the Reactor. 2) Initiate 1-E-0, REACTOR TRIP OR SAFETY INJECTION.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14.	__CHECK MAIN CONDENSER VACUUM - GREATER THAN 26.5 IN-HG	Do the following: a) Place the Condenser Hoggers in service IAW Attachment 1. b) Initiate Turbine ramp IAW Attachment 2, if required. c) IF Output breakers closed AND vacuum can NOT be recovered, THEN do the following: 1) Trip the Reactor. 2) Initiate 1-E-0, REACTOR TRIP OR SAFETY INJECTION. d) IF Output breakers open, THEN do the following: 1) Verify Reactor power is less than 10%. 2) Trip the Turbine.
15.	__RETURN TO STEP 3	
16.	__VERIFY CONDENSER VACUUM - STABLE OR INCREASING	Do the following: a) Continue efforts to restore condenser vacuum to normal. b) WHEN cause of vacuum decrease is corrected, THEN GO TO Step 17.
17.	__VERIFY HOGGERS REMOVED FROM SERVICE IAW ATTACHMENT 7	

QUESTION 10: (1.0)

Given the following plant conditions:

- A complete loss of vital bus 1-III occurred with the unit at 100% power.
- Repairs will require at least one hour before the bus can be re-energized.
- CC flow to the "A" RCP could not be restored and the team tripped the unit.
- Five minutes after the trip, RCS average temperature decreased to 538°F in the idle RCS loop.

Which ONE of the following actuations will occur, if any?

a. No actuation will occur.

b. High steam flow SI **only** will occur.

c. Steam line isolation **only** will occur.

d. High steam flow SI and steam line isolation will **both** occur.

ANSWER: d

[RO: Tier 1/Group 1]
[SRO: Tier 1/Group 1]

Answer correct: channel III high steam flow bistables go to tripped condition on loss of vital bus 1-III; high steam flow SI and steam line isolation both occur with 2/3 high steam flow and P-12 lo-lo Tave interlock.	Distractors plausible: a – this would be correct for loss of vital bus 1-I or 1-II; b – misconception concerning P-12 interlock; c – this would be correct if high steam flow SI were blocked (also, misconception concerning P-12 interlock)	Distractors incorrect: unless action is taken to bypass the safety function, high steam flow SI and main steam line isolation will both occur when RCS average temperature decreases below 543°F.
K/A: APE057-AA2.18	Objective: 2266	Source: New
Reference: AP-10.03, ND-90.3–LP-5	Level: Comprehension	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: A de-energized AC Vital Bus shall be re-energized within 2 hours OR the unit must be placed in Hot Shutdown within the next 6 hours.

[1] __EVALUATE FAILURE OF VITAL BUS 1-III:

- Check Vital Bus 1-III voltage on MCR voltmeter - LESS THAN 117 VOLTS

AND

- Check the following TVs - CLOSED
  - 1-CC-TV-105A
  - 1-CC-TV-110A

RESPONSE NOT OBTAINED: GO TO Step 6.

NOTE:
- Safety Injection is imminent if Loop A or B Tave drops below 543°F. 5
- If Safety Injection occurs, RCS temperature should be controlled below 543°F to prevent recurring SI signals. 5

[2] __CHECK UNIT - AT POWER

RESPONSE NOT OBTAINED: IF unit on RHR, THEN do the following:

a) Monitor RHR flow and pump amps.

b) Locally throttle RHR HX outlet valves as necessary:

- HX A, 1-RH-19
- HX B, 1-RH-24

GO TO Step 4.

[3] __TRIP THE REACTOR AND INITIATE 1-E-0, REACTOR TRIP OR SAFETY INJECTION

c. Actions necessary to stabilize the plant for a loss of VB 1-II are listed in Attachment 1.

d. An attempt is made to re-energize the vital bus by pushing the alternate source to load button on the UPS or using the manual bypass switch.

e. The team must stop at this point until the vital bus is re-energized. After the bus is energized, the remainder of the procedure restores affected systems to pre-event conditions.

f. The principle plant effects, should vital bus 1-II be lost, are the following:

   (1) Loss of CC to "B" RCP oil coolers and motor coolers

   (2) Loss of letdown

   (3) Loss of pressurizer pressure auto control

   (4) Loss of power to blender control and indication

   (5) Loss of Channel 2 NIs (SR, IR, and PR).

   (6) No arming signal to steam dumps

   (7) Loss of feed reg valve "A" control

   (8) Loss of S/G blowdown

5. AP-10.03, Loss of Vital Bus III

Ensure trainees have the latest revision of AP-10.03 to follow for this presentation. Perform a step-by-step discussion of this procedure highlighting applicable areas.

a. Initially a determination is made to see if VB 1-III or VB 1-IIIA is lost.

b. If VB 1-III is lost, the reactor is tripped and "A" RCP is secured due to loss of CC to the RCP lube oil coolers. The team should initiate E-0 and continue with AP-10.03.

c. Actions necessary to stabilize the plant for a loss of VB 1-III are listed in Attachment 1.

d. An attempt is made to re-energize the vital bus by pushing the alternate source to load button on the UPS or using the manual bypass switch.

e. The team must stop at this point until the vital bus is re-energized. After the bus is energized, the remainder of the procedure restores affected systems to pre-event conditions.

f. The principle plant effects, should vital bus 1-III be lost, are the following:

(1) Loss of CC to "A" RCP lube oil and stator coolers

(2) Loss of PR channel III (N-43)

(3) Loss of all main feed control bypass valve control

(4) Failure of steam dumps to control T_{ave} to required T_{ref}

(5) Loss of all steam generator feed regulating valve control (controllers are affected).

6. AP-10.04, Loss of Vital Bus IV

Ensure trainees have the latest revision of AP-10.04 to follow for this presentation. Perform a step-by-step discussion of this procedure highlighting applicable areas.

a. Initially a determination is made to see if VB 1-IV or VB 1-IVA is lost.

b. If VB 1-IV is lost, the reactor is tripped and "C" RCP is secured due to loss of CC to the RCP lube oil coolers. The team should initiate E-0 and continue with AP-10.04.

c. Actions necessary to stabilize the plant for a loss of VB 1-IV are listed in Attachment 1.

d. An attempt is made to re-energize the vital bus by pushing the alternate source to load button on the UPS or using the manual bypass switch.

e. The team must stop at this point until the vital bus is re-energized. After the bus is energized, the remainder of the procedure restores affected systems to pre-event conditions.

f. The principle plant effects, should vital bus 1-IV be lost, are the following:

(1) Loss of CC flow to "C" RCP stator and oil coolers

(2) Loss of PR channel IV (N-44)

(3) Loss of automatic pressure control of RCS

QUESTION 11: (1.0)

While responding to a Hi Hi CLS actuation on Unit 2 the Reactor Operator receives annunciator 2B-H2, BCHX OUTLET Hi Temp. Unit 1 is shutdown with RCP's running.

Which ONE of the following events caused the annunciator to alarm?

a. Phase III Containment Isolation.

b. Load Shed.

c. "E" and "F" transfer bus undervoltage.

d. Load Sequencing.

ANSWER: c

[RO: Tier 1/Group 1]
[SRO: Tier 1/Group 1]

Answer correct: Hi Hi CLS with a RSS undervolage will cause Service Water to be isolated to the BC heat exchangers resulting in the high temperature alarms.	Distractors plausible: a – Phase III isolation occurs on Hi Hi CIS. B – Load shed will actuate on the Unit 1 Trip with U 2 already shutdown. D – Load Sequencing will occur in this situation.	Distractors incorrect: a – Phase III isolation just affects containment loads. B – The BC pumps are not load shed pumps. D – The BC pumps are not load sequenced pumps.
K/A: APE062-AK3.02	Objective: 2119	Source: New
Reference: ND-89.5-LP-2	Level: Comprehension	

b. CC H/X's have local temperature, pressures, differential pressure and flow readings.

The flow sensors use a portable differential pressure gage.

c. RS H/X's have flow and temperature sensors with read outs in the MCR.

2. Controls

a. MOV-SW-101A, B (BC H/X Supply)

(1) Operated from benchboard 1-1

(2) Valves receive an automatic closure signal on either of the following:

(a) Intake canal level 23 feet, 6 inches

(b) High-high CLS with a blackout

b. MOV-SW-102A, B (CC H/X and SW-P-4 Supply)

(1) Operated from benchboard 1-1

(2) Valves receive an automatic closure signal on either of the following:

(a) Intake canal level 23 feet, 6 inches

(b) High-high CLS with a blackout

(3) A time delay allows these valves to be reopened 5 minutes after

QUESTION 12: (1.0)

Given the following plant conditions:

- Unit 1 was at 100% power when noxious fumes forced evacuation of the control room.
- 1-AP-20.00, Main Control Room Inaccessibility, was entered.

Which ONE of the following identifies the control manipulation required to transfer charging flow control to the ASDP?

a. Place the switch for half station 1-CH-FCV-1122 to "LOCAL".

b. Place the "H" Group Transfer Switch to "LOCAL".

c. Place the "J" Group Transfer Switch to "LOCAL".

d. Place the switch for Manual/Auto controller FCV-FC-1122 to "LOCAL".

ANSWER: a

[RO: Tier 1/Group 1]
[SRO: Tier 1/Group 1]

Answer correct: This is the method of transferring control stated in 1-AP-20.00 to the ASDP.	Distractors plausible: b/c – this the method of transferring the majority of controls on the ASDP. D – The MCR has a Manual/Auto Controller so assumption could be that the ASDP has one as well.	Distractors incorrect: b/c Charging flow control has a separate switch for transfer control. D – Charging flow control has a Half station station not a controller at the ASDP.
K/A: APE068-AK2.03	Objective: 3136	Source: New
Reference: 1-AP-20.00	Level: Knowledge	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: • Malfunctioning controls can be isolated from the MCR by using 0-FCA-1.00, LIMITING MCR FIRE, Attachment 1.

• Transfer of H and J group controls to the Auxiliary Shutdown Panel depends on the availability of the associated Emergency Bus.

9. ___TRANSFER CONTROL TO THE AUXILIARY SHUTDOWN PANEL:

a) Put the following switches to the same position as the corresponding MCR switch

- MD AFW Pumps
- Turbine Driven AFW Pump SOVs
- Charging Pumps
- Boric Acid Transfer Pumps

- PRZR BACKUP HEATERS Group A
- PRZR BACKUP HEATERS Group E

b) Put the following switches in the LOCAL position

- PRZR BACKUP HEATERS GROUP E MODE SELECT

- H GROUP TRANSFER SWITCH
- J GROUP TRANSFER SWITCH

- PRZR BACKUP HEATERS GROUP A MODE SELECT

- CHG FLOW CONTROL, ( )-CH-FCV-( )122

Half station only

QUESTION 13: (1.0)

Which ONE of the following conditions describes a loss of containment integrity?

a. The seals on the Personnel airlock (inner and outer) are tested 18 days after use while at power.

b. The inner containment airlock door is left open while performing maintenance on the outer door O-rings at Cold Shutdown.

c. One containment airlock door is opened for maintenance at Intermediate Shutdown.

d. A containment penetration exceeds Tech Spec leakage rate limits while at Cold Shutdown.

ANSWER: a

[RO: Tier 1/Group 1]
[SRO: Tier 1/Group 1]

Answer correct: per TS-3.8.B, both doors must normally be closed; one door may be open for normal transit entry and exit, and entry to repair the inner air lock door is allowed.	Distractors plausible: c – candidate misconception regarding action "A" of Technical Specifications; b/d – In cold shutdown requirements for the containment integrity are not required.	Distractors incorrect: c – outer door open would **not** render CTMT inoperable, operation may continue per TS-3.8B; b – with inner door inoperable, the **outer** door is the operable CTMT boundary; d – leakage rate spec only applies in modes above Cold Shutdown.
K/A: APE069-AA2.01	Objective: 1871	Source: NEW
Reference: TS-3.8B, ND-88.4-LP-2	Level: Knowledge	

3.8 CONTAINMENT

Applicability

Applies to the integrity and operating pressure of the reactor containment.

Objective

To define the limiting operating conditions of the reactor containment.

Specification

A. CONTAINMENT INTEGRITY

1. CONTAINMENT INTEGRITY, as defined in TS Section 1.0, shall be maintained whenever the Reactor Coolant System temperature exceeds 200°F.

    a. Without CONTAINMENT INTEGRITY, re-establish CONTAINMENT INTEGRITY in accordance with the definition within 1 hour.

    b. Otherwise, be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. The inside and outside isolation valves in the Containment Ventilation Purge System shall be locked, sealed, or otherwise secured closed whenever the Reactor Coolant System temperature exceeds 200°F.

3. The inside and outside isolation valves in the containment vacuum ejector suction line shall be locked, sealed, or otherwise secured closed whenever the Reactor Coolant System temperature exceeds 200°F.

B. Containment Airlocks

1. Each containment airlock shall be OPERABLE with both doors of the personnel airlock closed except when the airlock is being used for normal transit entry and exit through the containment, then at least one airlock door shall be closed.

    a. With one airlock or associated interlock inoperable, maintain the OPERABLE door closed and either restore the inoperable door to OPERABLE status or lock closed the OPERABLE door within 24 hours.

    b. If the personnel airlock inner door or interlock is inoperable, the outer personnel airlock door may be opened for repair and retest of the inner door. If the inoperability is due to the personnel airlock inner door seal exceeding the leakage test acceptance criteria, the outer personnel airlock door may be opened for a period of time not to exceed fifteen minutes with an annual cumulative time not to exceed one hour per year for repair and retest of the inner door seal.

    c. Otherwise, be in HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

C. Containment Isolation Valves

1. Containment isolation valves shall be OPERABLE.† With one or more isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE† in each affected penetration that is open and either:

    a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

    b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

---

† Non-automatic or deactivated automatic containment isolation valves may be opened on an intermittent basis under administrative control. The valves identified in TS 3.8.A.2 and TS 3.8.A.3 are excluded from this provision.

b. Otherwise, be in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

3. The inside and outside isolation valves in the Containment Ventilation Purge System shall be locked, sealed, or otherwise secured closed whenever the Reactor Coolant System temperature exceeds 200°F.

4. The inside and outside isolation valves in the containment vacuum ejector suction line shall be locked, sealed, or otherwise secured closed whenever the Reactor Coolant System temperature exceeds 200°F.

5. Each containment airlock shall be OPERABLE with both doors of the personnel airlock closed except when the airlock is being used for normal transit entry and exit through the containment, then at least one airlock door shall be closed.

   a. With one airlock or associated interlock inoperable, maintain the OPERABLE door closed and either restore the inoperable door to OPERABLE status or lock closed the OPERABLE door within 24 hours.

   b. If the personnel airlock inner door or interlock in inoperable, the outer personnel airlock door may be opened for repair and retest of the inner door. If the inoperability is due to the personnel airlock inner door seal exceeding the leakage test acceptance criteria, the outer personnel airlock door may be opened for a period of time not to exceed fifteen minutes with an annual cumulative time not to exceed one hour per year for repair and retest of the inner door seal.

   c. T.S.4.4. requires that the seals on the personnel airlock be tested within 7 days of use. The actual requirement to test the hatch seal is part of the 10CFR requirement listed in this section.

QUESTION 14: (1.0)

A letdown radiation monitor alarm prompts the team to verify a mixed bed IX is in service.

Which ONE of the following identifies the basis for this action?

a. A mixed bed IX must be in service because the letdown radiation monitor is located in series with the IX.

b. A mixed bed IX will remove excess Lithium that could potentially damage the fuel.

c. A mixed bed IX will help reduce radioactive activity in the RCS by trapping the radioactive particles.

d. A mixed bed IX will raise primary pH which will minimize fuel element damage.

ANSWER: c

[RO: Tier 1/Group 1]
[SRO: Tier 1/Group 1]

Answer correct: c – a mixed bed IX as a filter for the RCS.	Distractors plausible: a – The pressure drop across the mixed bed IX does increase the flow through the letdown radiation monitors. B – The Cation IX removes Lithium. D – A mixed bed IX affects pH .	Distractors incorrect: a - Flow through the Letdown radiation monitors will occur without a mixed bed IX in service. B – A mixed bed IX does not remove Lithium. D – A mixed bed IX acts to maintain pH.
K/A: APE076-AK3.06	Objective: 1792	Source: New
Reference: ND-88.3-LP-2	Level: Knowledge	

c. Charging pump suction valves and headers

d. Charging pumps

e. Charging pump discharge valves and headers and cross-connect

f. MOV 1289 A and B

g. FCV-1122

h. Regenerative heat exchanger

i. Normal charging line and aux spray line

3. System interconnections

Highlight the following.

a. RHR inlet to letdown HCV-1142

b. Reactor cavity purification inlet and return lines

c. Letdown radiation monitoring

d. Blender makeup system

e. Alternate sources of suction for the charging pumps

f. Loop fill

(4) Pressure is sensed by PT-1145 just upstream of the valve. This pressure signal is sent to the control station and indicated on vertical board (VB) 1-1.

j. Low pressure L/D line relief valve RV-1209 - relieves to the VCT at 200 psig.

k. Radiation Monitors

(1) Two radiation monitors are installed in a line running between the inlet to TCV-1143 and the VCT. The Delta P across the IXs provide the flow through the rad monitors.

(2) These rad monitors provide indication of RCS activity.

l. TCV-1143

(1) TCV-1143 protects the IX resin from damage due to high L/D line temperature.

(2) The TCV will divert L/D flow to bypass the IXs if L/D temperature exceeds 145°F.

(3) Controlled by a Normal-Auto-Divert switch on BB 1-1.

(4) In Divert, the IXs are bypassed.

(5) In Auto, the TCV will divert at 145°F and automatically realign to the IXs when temperature goes below the reset point.

(6) In Norm, the IXs are in service. The switch is spring return from Norm to Auto.

(7) TE-1143 provides input for TCV-1143 positioning. It also provides indication on VB 1-1. This TE is located next to TE-1144.

m. Reactor Cavity Purification (RL)

(1) RL ties into letdown at the inlet of the letdown filter and the return to the cavity taps off between the reactor coolant filter and LCV-1115A.

(2) The letdown system is used to reduce the activity of the reactor cavity during refueling operations.

(3) It is used to remove crud shaken loose from assemblies during fuel movement.

n. Letdown filter

(1) Removes suspended solids from the letdown flowpath.

(2) The micron size of the filter is reduced if possible each time the filter is replaced to clean up the RCS as much as possible. Presently, a 5 micron filter is installed.

o. Mixed Bed IXs

(1) Two lithium hydroxide resin beds for each unit.

(2) Li OH is used to maintain pH in the proper band.

(3) Design flowrate through each bed is 120 gpm. This is the reason letdown flow is maintained at 105 gpm.

(4) Normally one bed is in service and one in standby.

p. Cation IX

(1) One per unit located on IX alley.

(2) Loaded with H^+ type resin. Used to increase pH by removing excess free lithium. Too much lithium removes OH ions, pushing the pH in the acidic direction. Excess lithium is formed by the neutron-boron reaction.

(3) The cation IX is placed in service at the request of the chemists.

(4) It can also be used to remove cesium from the RCS.

(5) Design flowrate through the cation IX is 60 gpm. This flow is achieved by partially bypassing the IX when it is in service.

q. Deborating IXs

(1) Two per unit located on IX alley. Loaded with OH type resin.

(2) Used near the end of life for removing boron from the RCS. Generally used when Boron concentration is less than 50 ppm.

(3) After the demin has been lined up manually, the operator can place it in service from the MCR using HCV-1244.

(4) HCV-1244 is a two position switch, norm and divert on BB 1-1. In Norm, the IXs are bypassed. In Divert, the IXs are in service.

r. Reactor coolant filter

(1) One filter per unit located on IX alley.

(2) Removes any resin fines or crud that are released by the IXs.

(3) The micron size of the filter is reduced if possible each time the filter is replaced to clean up the RCS as much as possible. Presently, a 10 micron filter is installed.

s. LCV-1115A

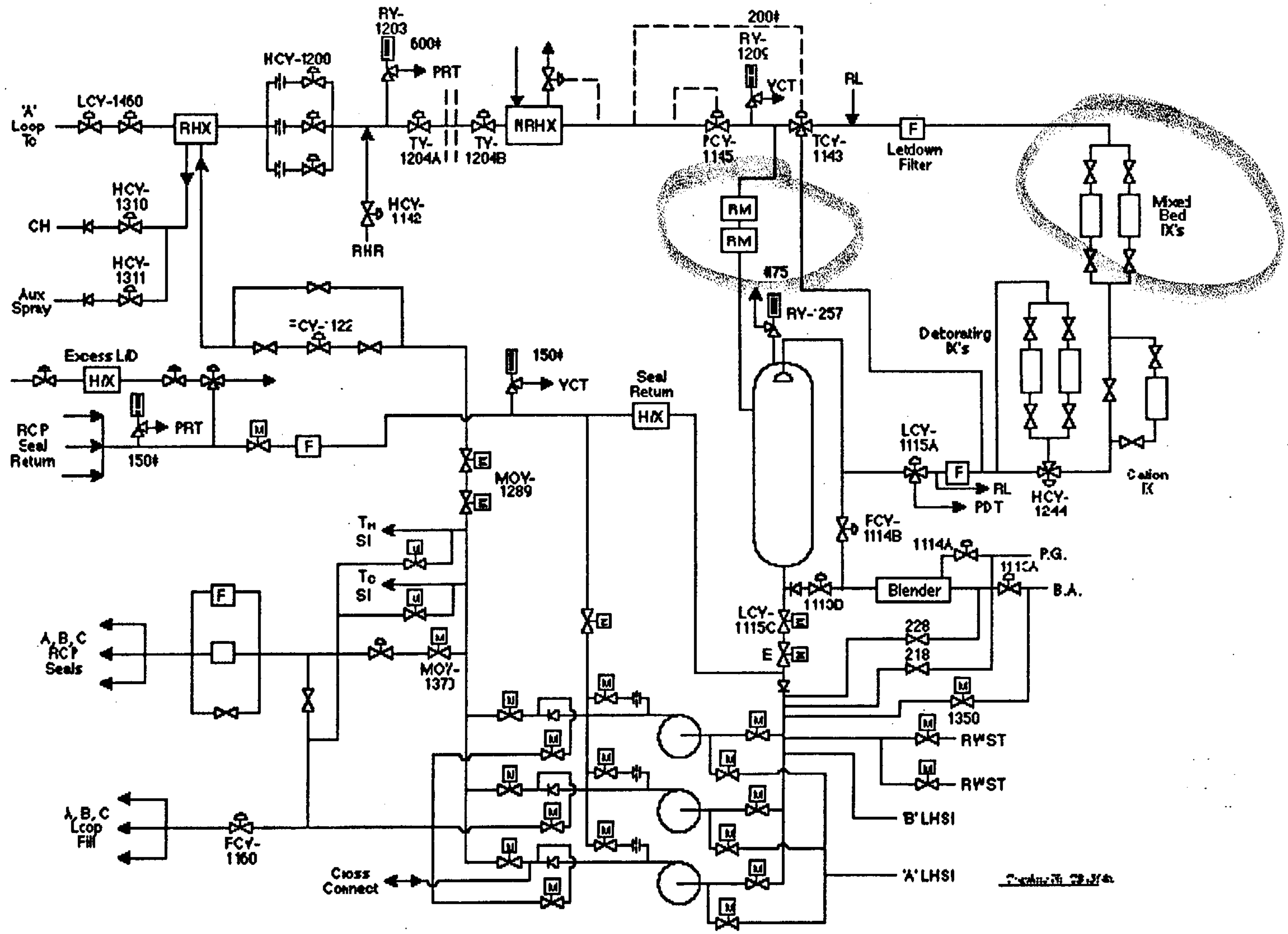
(1) LCV-1115A normally aligns letdown flow to the VCT. On a high level signal the LCV will divert letdown flow to the PDT.

(2) The valve is controlled by a three position switch on BB 1-1, Norm - Auto - Divert.

(3) Norm position aligns the LCV to the VCT.

(4) Divert position aligns the LCV to the PDT.

(5) In Auto, there are two modes of control:



CHARGING AND LETDOWN SYSTEM DRAWING

QUESTION 15: (1.0)

Given the following plant conditions:

- Unit 1 is stable at 50% power.
- Rod control is in AUTOMATIC with "D" bank at 190 steps.
- T_{ave} and T_{ref} are initially matched at 560°F.
- A failure in the median/high select T_{ave} control unit results in T_{ave} indicating 556°F.

As a result, control rods will __________.

a. insert at 40 steps per minute

b. withdraw at 40 steps per minute

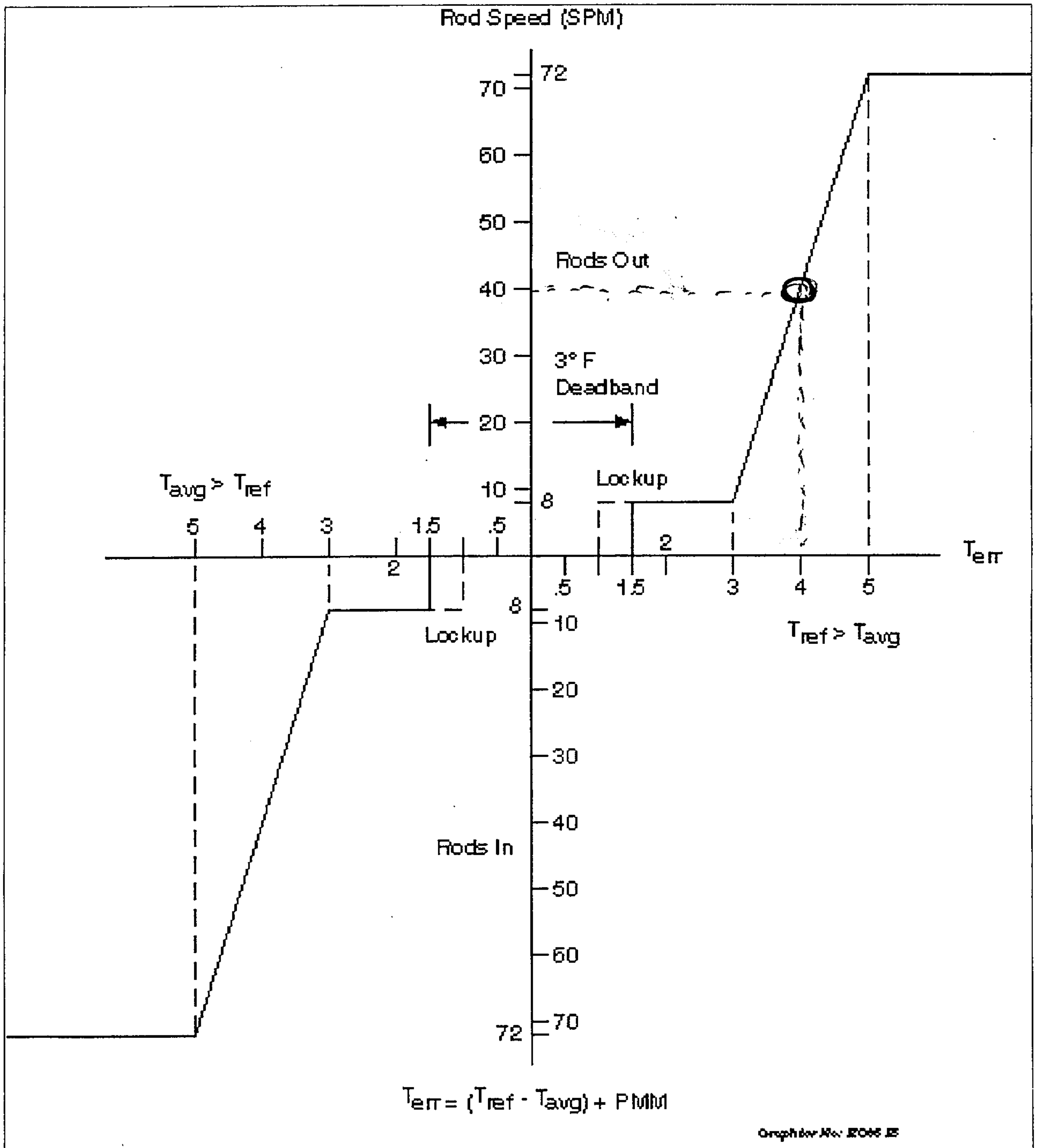
c. insert at 32 steps per minute

d. withdraw at 32 steps per minute

ANSWER: b

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 1]

Answer correct: rod speed changes linearly from 8 spm at 3°F mismatch to 72 spm at 5°F mismatch; rod speed with a 4°F mismatch =	Distractors plausible: a & c – candidate misconception regarding direction of rod motion with $T_{ave} < T_{ref}$, a – rod speed **is** 40 spm; c & d – candidate calculates the rod speed based on half the **difference** between 8 spm and 72 spm, but forgets to add the 8 spm; d – rods **are** withdrawn.	Distractors wrong: a & c – rods are withdrawn; c & d – rod speed is 40 spm.
K/A: APE001-AK2.06	Objective: 2636	Source: NEW
Reference: ND-93.3-LP-3	Level: Comprehension	



CONTROL ROD SPEED VS ERROR

QUESTION 16: (1.0)

Which ONE of the following will directly result in a Unit 1 Reactor Trip from 100% power?

a. With MG 1-1 tagged out for maintenance, "C" RSST locks out.

b. Loss of the 1C1-2 transformer supplying Rod control.

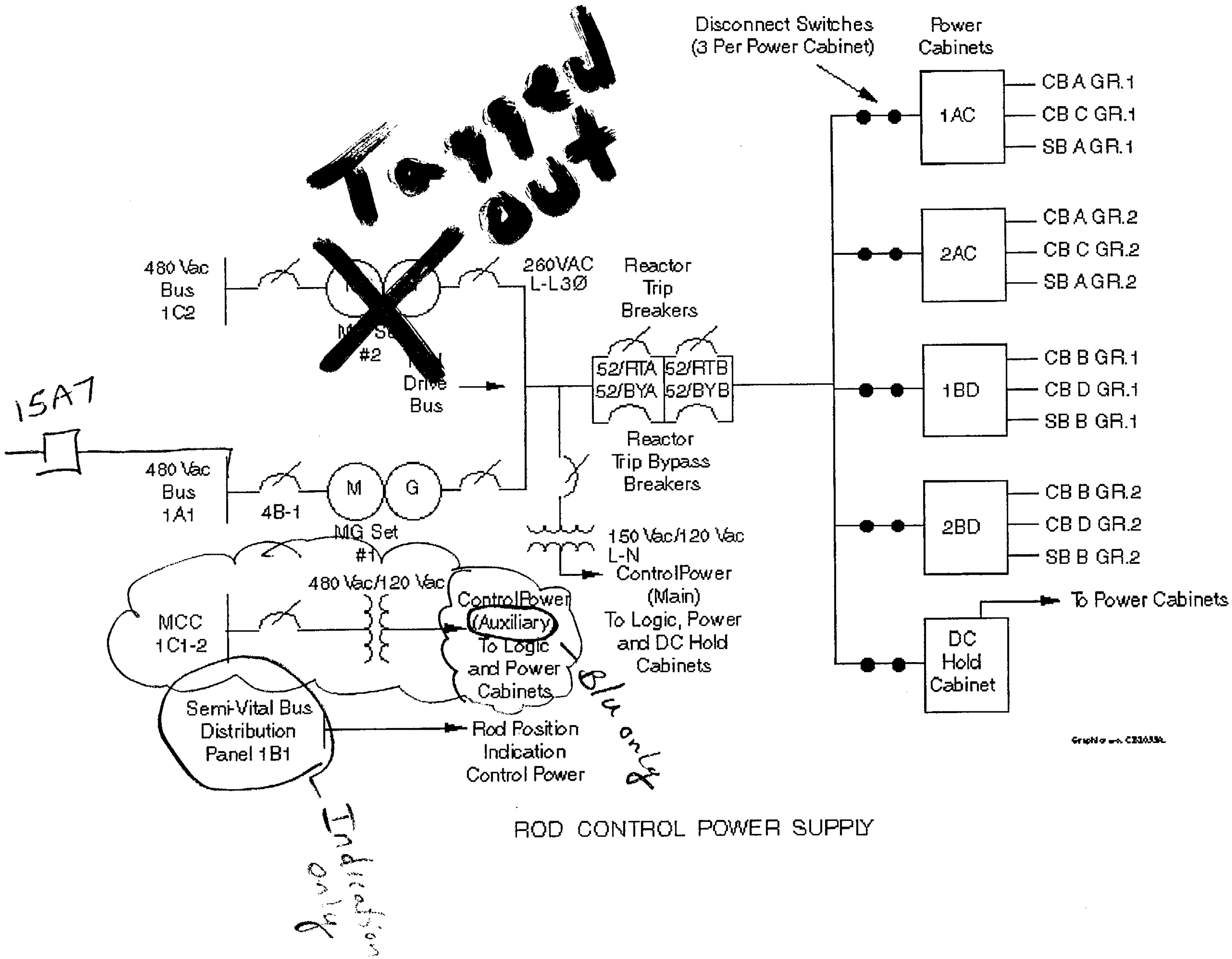
c. With MG 1-2 tagged out for maintenance, breaker 15A7, 480 supply breaker, trips open.

d. Loss of semi-vital bus power to the IRPI cabinets.

ANSWER: c

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 1]

Answer correct: C MG 1-1 is supplied from the 15A7 breaker. With both MG sets de-energized the control rods drop into the core after 1 second (coast down time of the MG flywheel).	Distractors plausible: A – While the other MG is powered from "C" SS, the "C" bus is supplied from Station service transformers at power. (would be correct if power were less than 30%) B - This is an alternate power supply and would cause a trip if MG supply were unavailable) D- IRPI used to cause a turbine runback but now only provides indication and annuciator.	Distractors incorrect: A – MG 1-2 receives power from the 'B' RSST; therefore, it is still maintaining control rods held. B – This provides the redundant backup power to the cabinet electronics which would provide an Non-urgent failure but would not drop the control rods. D – This does not affect the rod control system.
K/A: APE003.AK2.05	Objective: 2633	Source: NEW
Reference: ND-93.3-LP-3	Level: Comprehension	



ROD CONTROL POWER SUPPLY

QUESTION 17: (1.0)

The following conditions exist:

- Unit 1 tripped from 30% power due to a loss of the “A” RSST.
- The Unit 1 Terry Turbine tripped on overspeed.
- The #3 EDG failed to auto start.
- The Outside Service Building Operator has isolated the “J” train of AFW.

Which ONE of the following system relationships exists?

a. “B” Motor Driven AFW pump recirculation line is isolated.

b. With Unit 1 AFW cross-tie established from Unit 2, Unit 1 “J” AFW will be re-established.

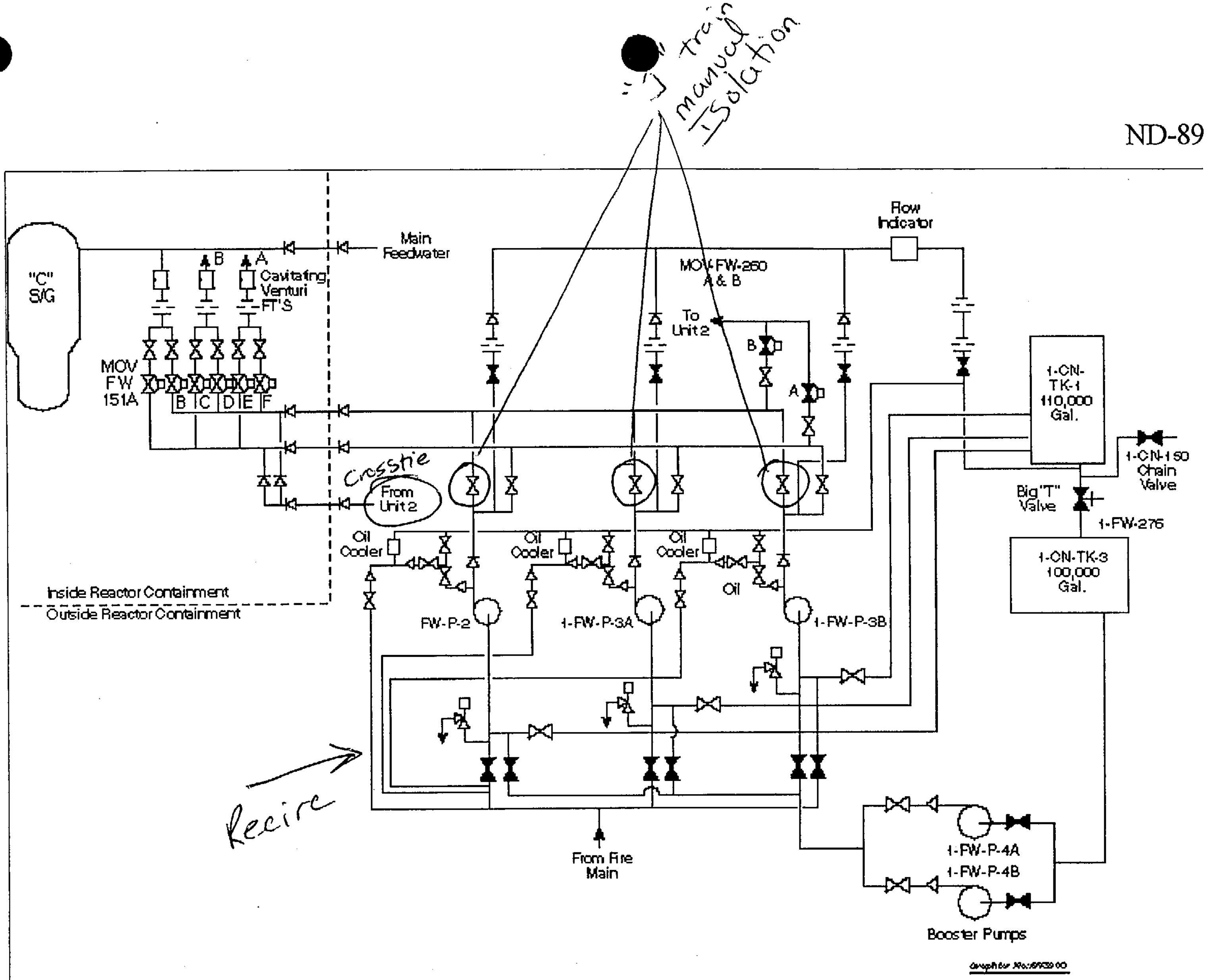
c. If Unit 1 “J” bus is restored, Unit 1 “J” header AFW flow will initiate unmitigated.

d. The Unit 1 “J” train must be un-isolated to initiate Unit 2 AFW cross-tie.

ANSWER: b

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 2]

Answer correct: b The cross-tie connection is downstream of the manual isolation valves. Therefore, opening of the cross-tie will restore flow to locally isolated Unit 1 ‘J’ header.	Distractors plausible: A – Since the “B” pump is powered from the “J” bus, recirculation isolation would have no implications in this scenario C – Systems are normally fail safe, the operator action to isolate the header is manual valve isolation. D – Unit 2 cross-tie taps into the Unit 1 lines.	Distractors incorrect: A – Recirculation is NOT isolated. C – The manual isolation prevents all “J” header flow. D- Unit 2 cross-tie taps in downstream of the pump isolation valves.
K/A: EPE007G2.4.35	Objective: 2049	Source: NEW
Reference: ND-89.3-LP-4 handout 4.2	Level: Comprehension	



AUXILIARY FEEDWATER SYSTEM

QUESTION 18: (1.0)

You are monitoring 100% power operation using the SPDS in the TSC. During this monitoring the following Top Level Display Functions are shown:

1-Reactivity Control	Green
2-Core Heat Removal	Red
3-Secondary Heat Removal	Green
4-RCS Integrity	Red
5-Radioactivity Control	Red
6-Containment Conditions	Red

Which ONE of the following identifies the accident in progress?

a. Steam Generator Tube Rupture.

b. Steam Line Break Inside Containment.

c. Steam Line Break Outside Containment

d. Small Break LOCA.

ANSWER: d

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 2]

Answer correct:	Distractors plausible:	Distractors incorrect:
A small break LOCA will cause thel six of the Top Level Display Functions to indicate as shown.	a. A steam generator tube rupture would cause #'s 4 and 5 to be red. b. A steam line break in containment would cause #'s 3, 4, and 6 to be red. c. A steam line break outside of containment would cause # 3 to be red.	a. Containment conditions remain normal b. Radiation conditions remain normal c. Containment and radiation conditions remain normal.
K/A: EPE009.EA1.10	Objective: 1978	Source: NEW
Reference: ND-93.4-LP-1	Level: Comprehension	

(2) The Operator Main Menu is mainly for the STA or ERF Computer Operator, not the Control Room Operator.

(3) The Programmer Menu is for use by authorized programming personnel. In most cases, its functions are security restricted to only a few select CRTs.

c. SPDS - Safety Parameter Display System

(1) This display is designated to give rapid, but detailed assessment status for 6 areas:

- Reactivity - RTV
- Core Heat Removal - CHR
- Secondary Heat Removal - SHR
- RCS Integrity - RCS
- Radioactivity Control - RAD
- Containment Conditions - CNT

(2) The TOP level is a bar chart summary of the six monitored parameters.

(3) The MID level displays are more detailed reviews of subsystems that make up the high level parameters.

Use actual displays or copies from the Surry SPDS User's Guide.

(4) The LOW level displays are parameter vs. time trend displays. They will display the last 25 minutes of data (if available).

d. ERG - Emergency Response Guidelines.

This status color means that the point is suspect. This is from being off-scan with an inserted value.

(3) Red

This status color means that the point is critical. A HI-HI or LO-LO alarm, pump running or valve open are examples of items which will value red.

(4) Yellow

This status color means that the point is in caution. An example would be when a HI or LO alarm is active for the parameter.

(5) Green

This status color means that the item is normal, i.e., the pump is not running or the valve is shut.

d. Parameter status code

Any time a parameter is out of its normal or expected range, its value will have two characters (or status codes) following it, in order to assist the user in interpreting its condition or quality.

Examples:

QUESTION 19: (1.0)

The following conditions exist:

- The Unit Reactor Operator noticed pressurizer level decreasing rapidly and performed AP-16.00, Excessive RCS leakage.
- The team initiated a manual SI based on excessive RCS leakage.
- A Health Physics technician reported steam from the Safeguards valve pit area.
- With RCS pressure at 1650 psig, the Unit Reactor Operator reported LHSI flow at 4500 gpm per pump.
- The team has entered ECA-1.2, LOCA outside Containment.
- Immediately after closing 1-SI-MOV-1890C, all LHSI flow reduced to zero.

Which ONE of the following identifies the status of the leak (assuming there is only one leak location)?

a. RCS leak is still active.

b. RWST leak is still active.

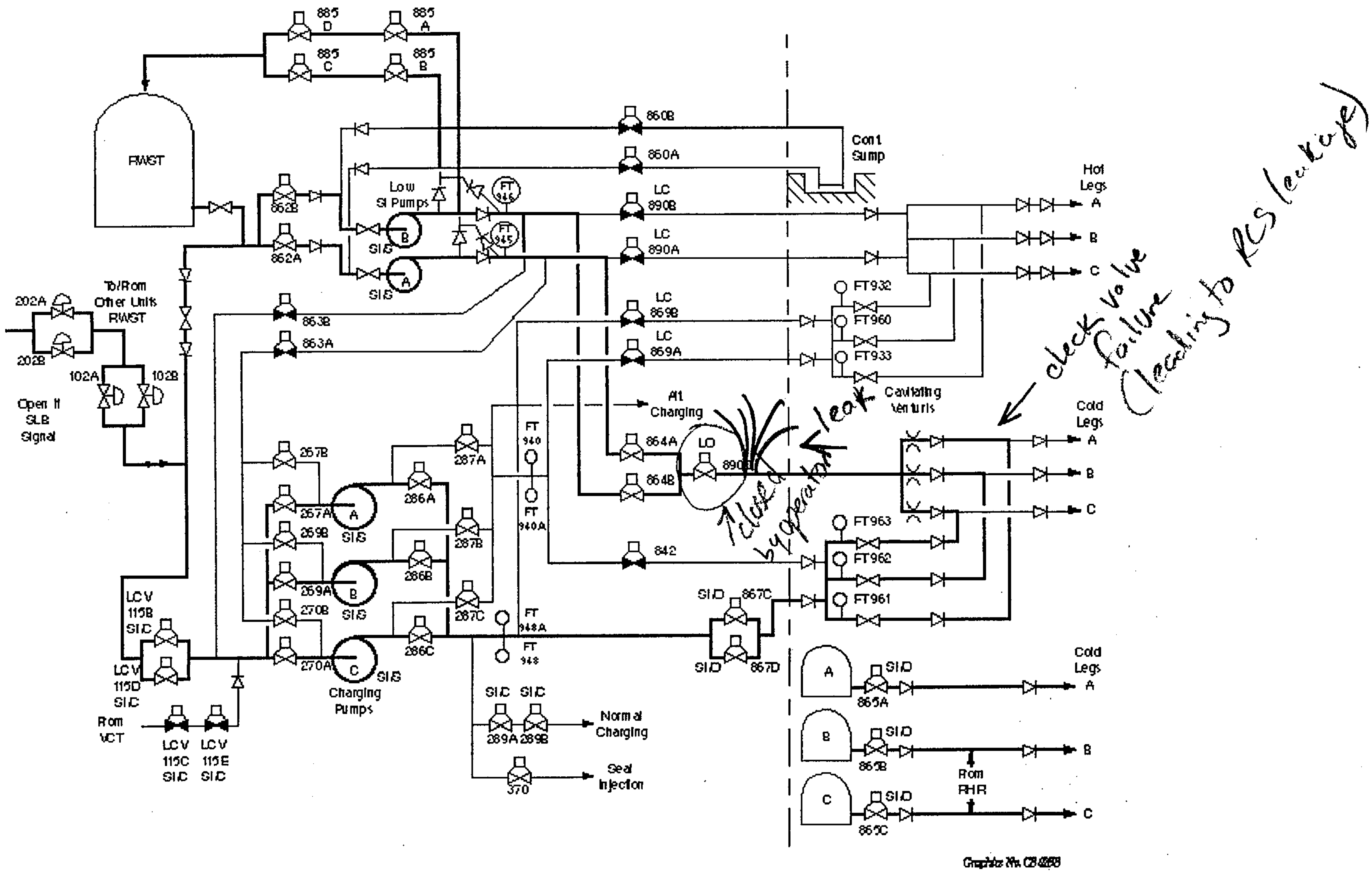
c. The leak is isolated.

d. The leak is NOT on the LHSI piping.

ANSWER: a

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 1]

Answer correct:	Distractors plausible:	Distractors incorrect:
The flow of water from the RWST through the leak has been stopped; however, the RCS leak through the break point has not been.	B – The LHSI pumps were discharging RWST water through the safety injection system leak until isolated. C – The RWST discharge is isolated D – The flow of RWST water has stopped which was one source of water leaking.	B – The RWST leak was isolated by closing 1-SI-MOV-1890C. C – The RWST leak is isolated by the RCS leak has not been. D – The leak is outside of containment and was discharging RWST water, via the LHSI pumps until isolated. Therefore, it is definitely on the LHSI piping.
K/A: W/E04.EK3.3	Objective: 2954	Source: New
Reference: ECA-1.2,	Level: Comprehension	



SI To INJECTION

QUESTION 20: (1.0)

All running RCP's must be stopped after Hi Hi CLS actuation.

Which ONE of the following is the reason for the RCP's being stopped?

a. to reduce RCS inventory loss via the break.

b. to prevent a deeper core uncovery due to two phase flow within the RCS.

c. to minimize electrical load on the Reserve Station Service busses during Hi Hi CLS.

d. to prevent damage to the RCP motor from a loss of Component Cooling.

ANSWER: d

[RO: Tier 1/Group 1]
[SRO: Tier 1/Group 1]

Answer correct: Hi Hi CLS isolates Component Cooling Water to the Motor Stator and lube oil.	Distractors plausible: a/b – this is the reason you stop the RCP's during a small break LOCA. C – Load on the RSS busses is a concern during Safeguards actuation which is why Auto Start Inhibit, Load Shed, and Load Sequencing were installed to reduce this load.	Distractors incorrect: a/b – This is the basis for RCP trip criteria which is a separate issue during a large break LOCA and not during a small break LOCA. d – load reduction does not include tripping of the RCP's
K/A: APE026-AK3.02	Objective: 2816	Source: New
Reference: 1-E-0 step 12	Level: Knowledge	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*12.	__CHECK IF CS REQUIRED:	
	a) CTMT pressure - HAS EXCEEDED 23 PSIA	a) Do the following: 1) IF CTMT pressure has exceeded 17.7 psia, THEN verify or align the following valves: • 1-RM-TV-100A - CLOSED • 1-RM-TV-100B - CLOSED • 1-RM-TV-100C - CLOSED • 1-SV-TV-102 - CLOSED • 1-IA-TV-101A - CLOSED • 1-IA-TV-101B - CLOSED • 1-IA-AOV-103 - OPEN 2) GO TO Step 13.
	b) Manually initiate HI HI CLS	
	c) Trip all RCPs	Due to phase III CTMT Isolation (Cooling water to motor/oil coolers)
	d) Verify CS pumps - RUNNING	d) Manually start pump(s). IF any pump will NOT start, THEN monitor OSRS pumps for cavitation. IF cavitation is indicated, THEN put affected OSRS pump in PTL.
	e) Check ISRS pumps - RUNNING (Time Delayed)	e) Manually start pumps.
	f) Check OSRS pumps - RUNNING (Time Delayed)	f) Manually start pumps.
	g) Initiate Attachment 1	

(1) To prevent loss of suction to the HHSI pumps, crosstie valves to the RWST of the unaffected unit are automatically opened on high steam flow isolation signals. Substep (d) manually verifies swapover.

(2) Substep (e) defeats automatic transfer of the SI pumps to sump recirculation on low RWST level. This is done to prevent pump suction transfer to an empty sump.

13. **<u>STEP 12</u>: CHECK IF CS REQUIRED.**

a. The purpose of this step is to ensure automatic actuation of Containment Spray and Containment Isolation Phase II and III if containment pressure exceeded the HI-HI CLS setpoint.

b. The asterisk (*) in front of the step number points out that this step is a **CONTINUOUS ACTION STEP**. (rk)

c. The basis behind this step is composed of several parts:

(1) The check to see if pressure "HAS EXCEEDED" the HI-HI CLS setpoint is worded this way in the event pressure may have temporarily exceeded 23 psia and then decreased due to spray actuation. In this case, the system operation should still be verified. Plant admin policy is to "back-up" auto signals with manual action, thus the operator is directed to initiate Hi-Hi CLS manually.

(2) The RNO has the operator check if containment pressure has exceeded the Hi CLS setpoint. If the setpoint was exceeded, the operator is directed to verify Phase II containment isolation.

(3) Since Component Cooling to the RCP seals and motors has been isolated, the RCPs are tripped to preclude overheating of the seals and motors.

(4) If containment pressure exceeds the HI-HI CLS setpoint, CS is automatically initiated to mitigate the containment pressure transient. In the event a CS pump fails, the ORS pump associated with a failed spray pump is monitored for cavitation since flow from the spray pump is necessary to ensure adequate pump NPSH during the injection phase. If cavitation is apparent, as indicated by fluctuations in pump flow or current, the affected pump is stopped.

(5) The Phase II and III valves are closed to isolate additional potential release paths from containment. Verification of valve closures and other automatic actions associated with HI-HI CLS is accomplished by initiating Attachment 1.

Ask: What is the purpose of time delays on the IRS and ORS pumps? Answer: The time delays ensure adequate sump inventory has accumulated before the pumps begin running (for NPSH requirements)

d. The time delay reminder notation under the IRS and ORS pumps is to alert the team that manual start of these pumps should be delayed just as automatic actuation would be.

e. If containment pressure has remained below 23 psia but has increased above atmospheric, the Critical Safety Function Status Trees will recommend performance of a yellow-path FR to address this condition.

QUESTION 21: (1.0)

The Operating team has just swapped charging pumps and the following conditions exist:

Charging flow control is in automatic.
Charging flow indicates 40 gpm.
"B" Charging pump indicates 82 amps (12 amps above normal).
Program Pressurizer level indicates 53.7%.
Actual Pressurizer level indicates 49%.
1-CH-FCV-1122 indicates full demand.

Which ONE of the following has initiated this response?

a. 1-CH-FCV-1122 air supply line severed.

b. "A" charging pump discharge check valve failed.

c. 1-CH-HCV-1311, Auxiliary Spray valve failed open.

d. Regenerative heat exchanger tube leak.

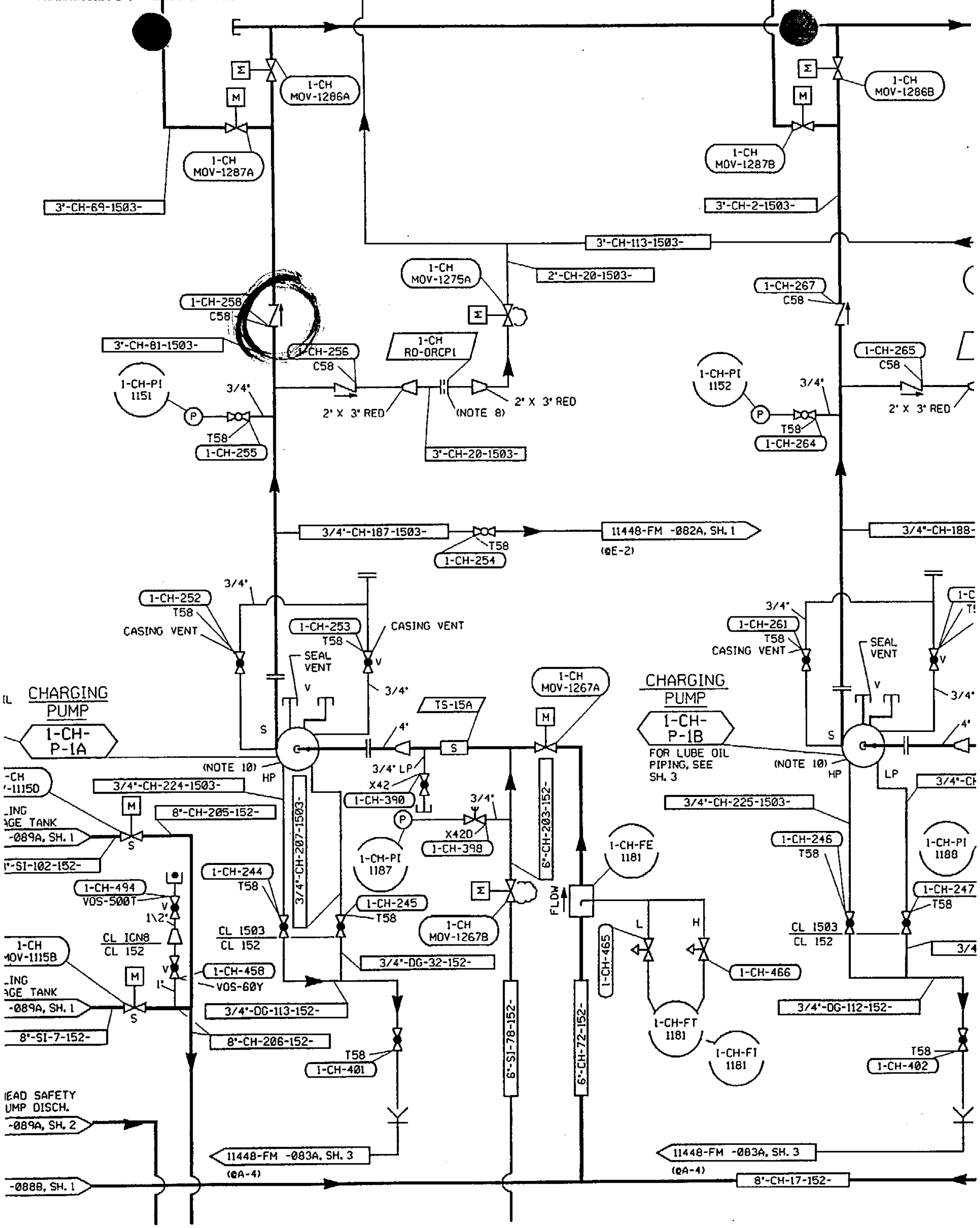
ANSWER: b

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 2]

Answer correct: The stuck open check valve is causing high flow for the "B" charging pump, yet RCS inventory is decreasing.	Distractors plausible: a. Full demand on the valve and high amps support this. c. full demand on FCV-1122 and unknown conditions support this. d. decreasing pressurizer level and high pump amps support this.	Distractors incorrect: a. Charging flow would be higher and pressurizer level would be higher (program and actual swapped) c. Charging pump amps would be unaffected, pressurizer level would be normal. d. Charging flow would be higher to maintain pressurizer level.
K/A: APE022.AA1.03	Objective: 1821	Source: NEW
Reference: AP-8.00	Level: Comprehension	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	__CHECK FOR CHG PUMP CHECK VALVE FAILURE:	
	a) Close normal and alternate discharge MOVs on <u>one</u> standby CHG pump • 1-CH-MOV-1286A, B, or C • 1-CH-MOV-1287A, B, or C	
	b) Verify running or start lead CHG pump	
	c) Check Charging flow and pressure - RETURNED TO NORMAL	c) Do the following: 1) Open MOVs closed in Step 4a. 2) Close normal and alternate discharge MOVs on standby CHG pump not closed in Step 4a. 3) <u>IF</u> CHG flow and pressure return to normal, <u>THEN</u> initiate a Work Request to repair faulty check valve <u>AND</u> GO TO Step 34. 4) <u>IF</u> CHG flow and pressure do <u>NOT</u> return to normal, <u>THEN</u> do the following: a. Place CHG pump control switches in PTL. b. Open the MOVs closed in Step 4c2 RNO. c. GO TO Step 5.
	d) Initiate a Work Request on failed check valve	
	e) GO TO Step 34	

** WARNING: THIS IS ONLY A PARTIAL SECTION OF ENTIRE DRAWING. **



QUESTION 22: (1.0)

The following conditions exist:

- The RCS is at 256°F, 300 psig.
- The RCS is solid.
- RHR is in-service, with 1-CH-PCV-1145 controlling RCS pressure.
- Outside air is valved into Containment.
- A loss of ALL instrument air occurs.

Assuming all systems respond as expected, which ONE of the following identifies the effect on the RCS?

a. Temperature will decrease.

b. Pressure will increase to a maximum of 2235 psig.

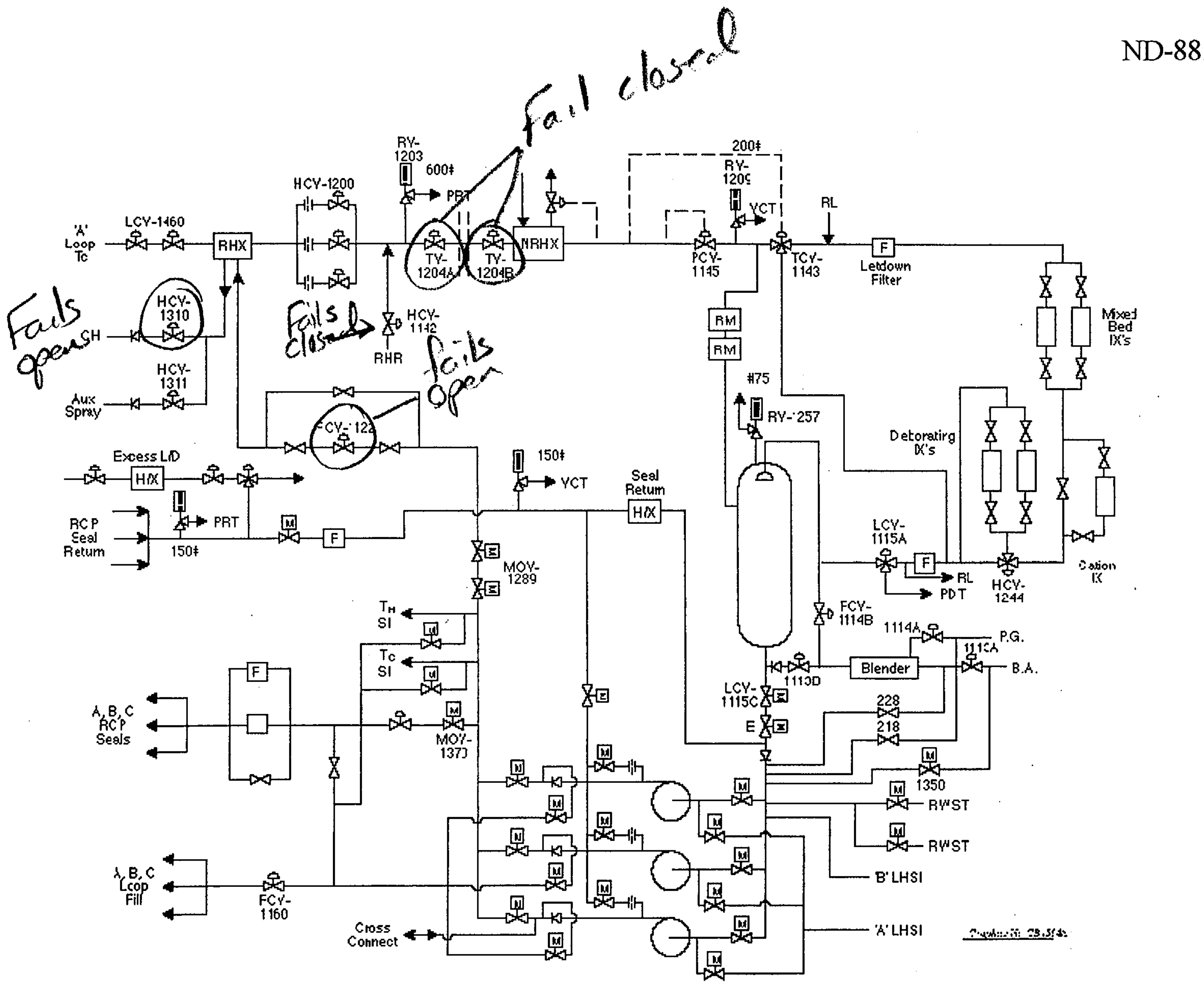
c. Pressure will increase to a maximum of 365 psig.

d. Inventory will decrease.

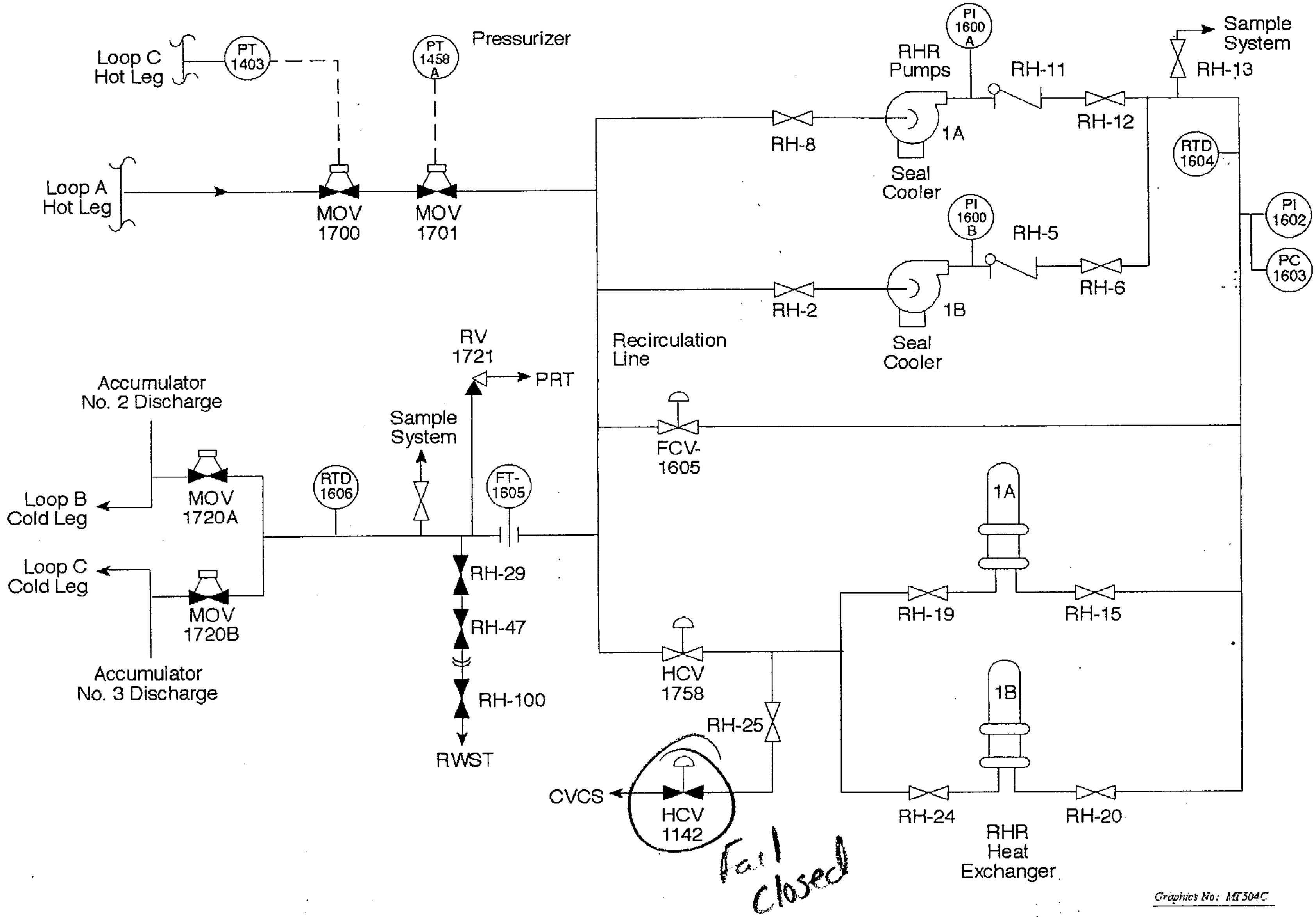
ANSWER: c

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 2]

Answer correct: A loss of air causes letdown to isolate and charging flow to maximize. Pressure will increase to the OPMS setpoint.	Distractors plausible: a. RHR bypass fails closed, RHR outlet control valve fails open. b. Normal PORV setpoint. d. misconception of fail position of valves (multiple required to be analyzed)	Distractors incorrect: a. CC supply to the RHR Heat exchangers fails closed b. OPMS must be in service. d. Level stays full, with charging failing to maximum.
K/A: APE025.AK1.01	Objective: 1792	Source: NEW
Reference: ND-88.3-LP-2, ND-93.3-LP-6	Level: Comprehension	



CHARGING AND LETDOWN SYSTEM DRAWING



RESIDUAL HEAT REMOVAL SYSTEM

a. Less than 365 psig on PT-403 and either MOV-1536 is not open or the associated keyswitch is in disable. (or)

b. Less than 345 psig on PT-458 and either MOV-1535 is not open or the associated keyswitch is in disable.

2. Pressurizer NDTT High Pressure

a. The Pressurizer NDTT High Pressure alarm is actuated to alert the operator that the RCS pressure is approaching the NDT overpressure protection setpoint. The operator should attempt to perform actions to mitigate the pressure excursion and return pressure to normal.

Refer to/display H/T-6.6, NDTT High Pressure Alarm Logics, and use with the following:

b. This alarm is actuated when either PT-403 is greater than 375 psig or PT-458 is greater than 355 psig and the associated keyswitch is in enable.

D. OPMS PORV Setpoints

Refer to/display H/T-6.7, OPMS Operations, and use with the following.

1. < 365 psig from PT-403 or < 345 psig from PT-458 provides the "Pzr OPMS Required" alarm (if disabled or associated block MOV closed).

2. > 375 psig from PT-403 or > 355 psig from PT-458 provides the "Pzr NDTT High Pressure" alarm (if enabled).

3. At 365 psig increasing from PT-458, PORV PCV-1456 is opened.

QUESTION 23: (1.0)

During the performance of 1-FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, the team is directed to stop both low-head safety injection (LHSI) pumps and all but one charging pump.

Which ONE of the following describes the potential consequences of failure to perform these actions?

a. Containment failure.

b. Brittle failure of the reactor vessel.

c. Loss of suction to containment spray pumps.

d. Loss of suction to LHSI and charging pumps.

ANSWER: b

[RO: Tier 1/Group 1]
[SRO: Tier 1/Group 1]

Answer correct: per WOG B/G document, SI flow may have contributed to the RCS cooldown and prevent RCS pressure reduction.	Distractors plausible: a & c – candidate misconception concerning depletion rate of RWST and design of Containment Spray Systems; d – candidate misconception concerning auto-swapover of SI pumps to CTMT sump (RMT).	Distractors incorrect: a & c – design of Containment Spray systems accounts for loss of RWST inventory to SI system; d – SI system suction auto-swaps to CTMT sump at low RWST level (RMT).
K/A: APE08-EK1.1	Objective: 3246	Source: New
Reference: WOG B/G document for FR-P.1, ND-93.5-LP-46	Level: Knowledge	

A. Integrity Status Tree

1. The Integrity status tree provides a systematic method to explicitly determine the status of the Integrity Critical Safety Function. It represents the fourth highest priority CSF and is entered directly after the Heat Sink tree.

Refer to/display H/T-46.2, Integrity Status Tree.

2. Operational Limits Curve Description

   a. This tree is unique among all the CSF Status Trees in that all the reference values against which current plant parameters are compared, do not appear explicitly at the branch points. Rather, two reference values are curves shown in figures to be included with the status tree.

   b. The main concern of the Integrity status tree is the reactor vessel wall and its ability to maintain integrity when subjected to a rapid cooling or rapid pressurization transient. As the thick walled vessel ages, it tends to lose its ductility due to radiation embrittlement, and its nil-ductility temperature, that temperature at which it begins to exhibit brittle behavior, increases.

      (1) Operators are trained to be aware of the brittle fracture concern, and are required by Technical Specifications to limit heatup and cooldown rates to conservatively limit thermal stresses below a critical yield stress to prevent a postulated vessel wall flaw from growing and possibly failing the vessel.

      (2) Operators are also trained to maintain RCS pressure and temperature within Technical Specification limits to address both thermal shock and cold overpressure concerns.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED	
10.	___VERIFY INSTRUMENT AIR AVAILABLE:		
	a) Check annunciator B-E-6 - NOT LIT	a) Initiate Attachment 1.	11
	b) Check at least one CTMT IA compressor - RUNNING	b) <u>IF</u> CC pump running, <u>THEN</u> start one CTMT IA compressor. <u>IF</u> <u>NOT</u>, <u>THEN</u> locally crosstie to turbine building IA (Zone 5 key required):	11
		• Use 1-OP-IA-005, Administrative Control of Unit 1 IA to Unit 1 CTMT Valves 1-IA-446 and 1-IA-447 <u>OR</u>	11
		• Use 1-OP-IA-006, Administrative Control of Unit 2 IA to Unit 1 CTMT Valves 2-IA-446 and 2-IA-447	11
	c) Verify 1-IA-TV-100 - OPEN	c) Open valve 1-IA-TV-100.	11
11.	___STOP SI PUMPS AND PUT IN AUTO: • All but one CHG pump • LHSI pumps		

QUESTION 24: (1.0)

During performance of FR-S.1, Response to Nuclear Power Generation/ATWS, manual alignment of the Charging pump suction to the RWST is performed if emergency boration is unavailable.

Which ONE of the following identifies why manual SI initiation is **NOT** performed?

a. Trips the running Main Feed Pumps.

b. Excessive RCS pressure spike due to an auto turbine trip.

c. Phase I Containment Isolation.

d. Initiates electrical transient due to a Main Generator Trip.

ANSWER: a

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 1]

Answer correct:	Distractors plausible:	Distractors incorrect:
WOG background identifies a loss of feed ATWS as worst case scenario	b. This is actually a negative result of tripping the turbine. c. Many systems are affected by a phase 1 isolation, assessment of a major effect is plausible. d. An electrical transient is inevitable during this event.	b. Turbine trip required to conserve inventory in the case of a loss of feed atws. c. No limiting conditions exist with a phase I islation d. The electrical transient is inevitable regardless of the SI (initiated by the turbine trip)
K/A: EPE029.G.2.4.18	Objective: 2989	Source: NEW
Reference: ND-95.3-LP-36	Level: Knowledge	

power, loss of offsite power (no RCS flow), and accidental depressurization of the RCS.

c. Loss of secondary side heat removal. Examples are loss of feedwater at power and loss of load.

4. The loss of load with loss of heat sink event is the most limiting ATWT overall.

5. The transient analyses performed for ATWT events evaluate both departure from nucleate boiling (DNB) ratio and RCS pressure. In the loss of feedwater and loss of load cases, the DNBR increases with time; therefore, peak RCS pressure is the parameter of concern. The accidental depressurization of the RCS creates the most limiting DNBR case.

6. ATWT events cause heat to be produced in the primary at a rate faster than it can be removed by the secondary. This results in a heatup of the primary and consequent expansion of water into the pressurizer. This, in turn, results in a primary pressure excursion transient. It is this increase in primary pressure that makes the loss of load ATWT the most limiting case. Therefore, one of the most important features in mitigating the consequences of ATWT events is the ability of the pressurizer PORVs and safety valves to limit the pressure rise.

Ask a trainee: From a nuclear point of view, what reactivity coefficients will primarily control this event? Answer: MTC and FTC.

QUESTION 25: (1.0)

20 minutes after a reactor trip from 66% power, the following NI indications exist:

- N-35 indicates 4.0x10-9 amps, SUR indicates 0 DPM.
- N-36 indicates 2.1x10-11amps, SUR indicates –0.1 DPM.
- Source range detectors N-31 and N-32 are de-energized.

Which ONE of the following describes the abnormality in the above condition?

a. N-35 is undercompensated.

b. N-36 is undercompensated.

c. N-35 is overcompensated.

d. N-36 is overcompensated.

ANSWER: a

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 2]

Answer correct: The undercompensated detector will indicate higher.	Distractors plausible: b/c/d – Misunderstanding on compensation of intermediate range.	Distractors incorrect: b – Undercompensated would make it indicate higher. c – overcompensated would make N-35 indicate lower. D – Is responding as expected following a reactor trip.
K/A: APE033.AK1.01	Objective: 2550	Source: New
Reference: ND-93.2-LP-3	Level: Comprehension	

C. Over and Under-Compensation

Refer to/display H/T-3.5, Under/Overcompensation Effects.

1. Reliable intermediate range indication in the overlap region between source and intermediate ranges is dependent upon proper adjustment of compensating voltage to the CIC.

2. If the compensating voltage for the detector is set too high, the detector is <u>Overcompensated</u>.

   a. The current generated in the inner can is greater than the current produced by the outer can.

   b. When the inner and outer can currents are summed, all of the outer can γ current and some of the Neutron current will be cancelled by the excessive inner can γ current.

   c. This effect causes the IR indication to be less than actual core power which is non-conservative.

3. When compensating voltage is set too low, the detector is <u>Undercompensated</u>.

   a. The inner can γ current will be too small to cancel out the outer can γ current.

   b. This results in the IR indication being greater than actual core power.

QUESTION 26: (1.0)

Which ONE of the following Steam Generator primary to secondary leakage conditions would permit continued power operation in accordance with Tech Specs?

a. "A" leakage at 1.2 gpm.

b. "A" leakage at .812 gpm and "C" SG at .25 gpm.

c. "A" leakage at .345 gpm, "B" leakage at .25 gpm, and "C" leakage at .3 gpm.

d. "A" leakage at .33 gpm, "B" leakage at .34 gpm, and "C" leakage at .335 gpm.

ANSWER: c

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 2]

Answer correct: Tech Specs allow 1 gpm total and 500 gpd through any 1 SG. c. does not violate either .	Distractors plausible: a. Only 1 SG affected, leakage minimal. b. Both SGs are less than 1 gpm each d. Each SG less than 500 gpd.	Distractors incorrect: a. violates 1 gpm total and 500 gpd. b. Violates 1 gpm total and 500 gpd on the "A" d. Violates the 1 gpm total.
K/A: APE037.AA2.10	Objective: 1726	Source: NEW
Reference: TECH SPEC section 3.1.C.6	Level: Comprehension	

6. If the primary-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System exceeds 1 gpm total and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System, reduce the leakage rate to within limits within 4 hours or be in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

7a. Prior to going critical all primary coolant system pressure isolation valves listed below shall be functional as a pressure isolation device, except as specified in 3.1.C.7.b. Valve leakage shall not exceed the amounts indicated.

	Unit 1	Unit 2	Max. Allowable Leakage (see note (a) below)
Loop A, Cold Leg	1-SI-79, 1-SI-241	2-SI-79, 2-SI-241	≦ 5.0 gpm for each valve
Loop B, Cold Leg	1-SI-82, 1-SI-242	2-SI-82, 2-SI-242	
Loop C, Cold Leg	1-SI-85, 1-SI-243	2-SI-85, 2-SI-243	

b. If Specification 3.1.C.7.a cannot be met, an orderly shutdown shall be initiated and the reactor shall be in hot shutdown within 6 hours and in the cold shutdown condition within the following 30 hours. 74

Notes

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

QUESTION 27: (1.0)

The following conditions exist during a Spent Fuel Cask load in the Fuel Building:

- Annunciator RM-C-3, Fuel Pit Bridge ALERT/FAILURE alarmed.
- Radiation Monitor indicates 7.32 mr/hr.
- High alarm test indicates 1.00E1 mr/hr.
- Alert alarm test indicates 5.00E0 mr/hr.
- Local reports from the Fuel Building indicate cloudiness in the water.

Which ONE of the following <u>immediate</u> <u>actions</u> is required for this situation?

a. Direct HP to establish alternate radiation monitoring.

b. Evacuate the fuel building.

c. Manually dump both banks of bottled air.

d. Immediately start one emergency MCR supply fan.

ANSWER: b

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 2]

Answer correct:	Distractors plausible:	Distractors incorrect:
Sufficient indications exist to validate a fuel failure. Immediate evacuation is warranted.	a. Only one RM is available to verify actual conditions. HP commonly verifies levels independently. b. Air bottles are dumped to maintain desirable conditions in the MCR. c. MCR fans are started to maintain desirable conditions in the MCR.	a. HP will be precluded from indepentatly verifying conditions due to the evacuation. c. Only one bank is dumped. d. Fans are started after air bottle banks are exhausted.
K/A: APE061.AA2.05	Objective: 2505	Source: NEW
Reference: ARP RM-C-3, AP-22.00	Level: Knowledge	

VIRGINIA POWER
SURRY POWER STATION

ANNUNCIATOR RESPONSE PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
0-RM-C3	FUEL PIT BRDG ALERT/FAILURE	3
		PAGE 1 of 3

REFERENCES

1. UFSAR 11.3, 9.12
2. 11448-ESK-10R
3. Tech Spec 3.10
4. DCP 89-21-3, INSTALLATION OF RM-SW-107A, B, C
5. DCP 92-028, RM Ratemeter and Recorder Replacement

PROBABLE CAUSE

1. Alarm actuates when 1-RM-RMS-153 detects activity greater than alert setpoint at Fuel Pit Bridge.

   High activity may be caused by one or more of the following:

   - Fuel failure
   - Spent Fuel Pit level decreasing

2. Instrumentation failure has occurred.

APPROVAL RECOMMENDED	APPROVED	DATE
REVIEWED	CHAIRMAN STATION NUCLEAR SAFETY AND OPERATING COMMITTEE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: An LCO will be entered during REFUELING OPERATIONS if the Fuel Pit Bridge monitor is inoperable IAW Tech Spec 3.10.

* *

NOTE: Upon failure of digital ratemeter with all EEEEEs indicated on the display, the digital ratemeter will need to be reset and a source check performed in accordance with 0-OPT-RM-001, Radiation Monitoring Equipment Check. 3

1. __VERIFY ALARM - READING ON MONITOR OR CHART RECORDER GREATER THAN OR EQUAL TO ALERT SETPOINT OR RADIATION LEVEL HAS TRENDED UP

- 1-RM-RI-153
- 1-RM-RR-175A, Pen 3

Do the following:

a) Check monitor failed.

- 1-RM-RI-153 FAIL light - LIT

b) IF monitor failed, THEN do the following:

1) Stop any Fuel Building activity in progress.

2) Review Tech Spec 3.10.

3) Notify HP to establish alternate methods for radiation monitoring.

4) WHEN alternate methods for radiation monitoring are established, AND Tech Spec 3.10 compliance has been verified, THEN resume Fuel Building activity.

5) Initiate a Work Request.

6) GO TO Step 7.

c) IF monitor NOT failed, THEN do the following:

1) Evaluate entry into 0-AP-10.13, LOSS OF MAIN CONTROL ROOM ANNUNCIATORS.

2) Initiate a Work Request.

3) GO TO Step 7.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	__EVACUATE AFFECTED AREA	
3.	__NOTIFY HEALTH PHYSICS TO DO THE FOLLOWING: • Verify area evacuated as necessary • Control access as necessary • Survey area as necessary • Investigate cause • Determine need for setpoint change	
4.	__CHECK FUEL PIT LEVEL - NORMAL	Initiate 0-AP-22.02, LOSS OF SPENT FUEL PIT LEVEL.
5.	__CHECK WITH FUEL HANDLING PERSONNEL - ABNORMAL CONDITION EXISTS	GO TO Step 7.
6.	__INITIATE 0-AP-22.00, FUEL HANDLING ABNORMAL CONDITIONS	
7.	__PROVIDE NOTIFICATIONS AS NECESSARY: • Shift Supervisor • OMOC • STA • Health Physics • Instrumentation Department	

- END -

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1]	__CHECK FUEL REPAIR - IN PROGRESS	GO TO Step 4.
[2]	__CHECK LOCAL RADIATION CONDITIONS - NORMAL	GO TO Step 4.
[3]	__GO TO STEP 18	
[4]	__STOP FUEL HANDLING OPERATIONS	
[5]	__EVACUATE THE AFFECTED AREA • Containment <u>OR</u> • Fuel Building	
[6]	__SECURE NORMAL MCR VENTILATION a) Close 1-VS-MOD-103C b) Close 1-VS-MOD-103D c) Verify stopped or stop 1-VS-F-15 d) Verify stopped or stop 1-VS-AC-4	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[7]	__DUMP MCR BOTTLED AIR:	
	a) Close 1-VS-MOD-103B (Dumps Unit 1 Cable Vault air bottles)	
	b) Set timer for 60 minutes	
	c) Check positive pressure of 0.05 inches - BEING MAINTAINED • PDI-VS-110 • PDI-VS-101 • PDI-VS-200 • PDI-VS-201	c) Close 1-VS-MOD-103A. (Dumps MER 3 air bottles)
	d) Check all Main Station Batteries - FRESHENING CHARGE IN PROGRESS	d) GO TO Step 8.
	e) Notify Electrical Department that Battery Room must be monitored for explosive concentration	
* 8.	__CHECK FUEL HANDLING ACCIDENT - IN PROGRESS FOR ONE HOUR (WHEN TIMER GOES OFF)	Do the following: a) <u>WHEN</u> Fuel Handling accident has been in progress for one hour (when timer goes off), <u>THEN</u> immediately perform Step 9. b) GO TO Step 10.

QUESTION 28: (1.0)

Which ONE of the following is performed in accordance with FR-Z.3, Response to Containment High Radiation?

a. Isolate SW to 2/4 inservice RSHXs.

b. If containment pressure has not increased to 23 psia, initiate CS flow to add NaOH.

c. Verify that non-essential containment penetrations are isolated.

d. Establish Containment Purge and Exhaust.

ANSWER: c

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 2]

Answer correct: FR-Z.3 provide guidance in isolating	Distractors plausible: a – This would be performed if RSHX radiation is high. B – Because NaOH removes Iodine. D – Cat 1 filters are designed to remove airborn.	Distractors incorrect: a – Because this will not affect containment radiation levels b – Because you do not initiate Containment Spray if less than 23 psia. D – To establish purge must be < 200 degrees mode of applicability for FR-Z.3 >350
K/A: EPEE16.EK1.3	Objective: 3314	Source: NEW
Reference: FR-Z.3	Level: Knowledge	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	__CHECK ALL CTMT ISOLATION VALVES CLOSED IAW ATTACHMENT 1	IF a flow path is NOT required for recovery action, THEN close manually OR isolate locally.
2.	__CHECK ACTIVITY REMOVAL SYSTEMS:	
	a) Check CS system - OPERATING	a) Do the following: 1) IF CTMT pressure has remained less than 23 psia, THEN try to start CTMT air recirc fans. 2) GO TO Step 3.
	b) Verify NAOH addition to spray established - CAT LEVEL DECREASING WITH RWST	b) Verify all required caustic supply valves open. IF required caustic supply valves NOT open, THEN manually or locally open valves.
3.	__NOTIFY TSC PERSONNEL OF CTMT RADIATION LEVEL TO GET RECOMMENDED ACTIONS	
4.	__RETURN TO PROCEDURE AND STEP IN EFFECT	

- END -

QUESTION 29: (1.0)

Given the following plant conditions:

- Unit 1 tripped from 100% power due to a loss of all main and auxiliary feedwater.
- All efforts to restore S/G feedwater flow have failed.
- RCS bleed and feed has now been initiated.

In all cases, RCS bleed and feed will __________.

a. prevent core uncovery

b. prevent an inadequate core cooling condition

c. provide temporary core cooling until a secondary heat sink can be restored

d. depressurize the RCS sufficiently to enable the team to place RHR in service

ANSWER: c

[RO: Tier 1/Group 1]
[SRO: Tier 1/Group 1]

Answer correct: bleed and feed is not intended to provide long-term core cooling, it is only an interim measure to "buy time" for the operators to restore a secondary heat sink or place RHR in service.	Distractors plausible: a & b – bleed and feed **does** delay or minimize the possibility of core uncovery and inadequate core cooling; d – bleed and feed will depressurize the RCS to a degree, depending on many variables.	Distractors incorrect: a & b – bleed and feed will not prevent core uncovery and inadequate core cooling for all cases; d – bleed and feed does not depressurize the RCS sufficiently to place RHR in service for all cases.
K/A: APE074-EK2.03	Objective: 3303	Source: New
Reference: ND-95.3-LP-41, FR-H.1	Level: Knowledge	

F & B is the process of manually initiating SI and permitting the automatic cycling of the PORVs at their setpoint to vent RCS inventory and provide decay heat removal and core cooling.

(1) If symptoms for loss of secondary heat sink are reached, RCS bleed and feed heat removal is initiated through SI actuation (feed path) and opening the pressurizer PORVs (bleed path).

(2) Bleed and feed heat removal is maintained until the secondary heat sink is re-established and verified.

c. **RESTORE AND VERIFY SECONDARY HEAT SINK.**

(1) After RCS bleed and feed heat removal is established, the team continues attempts to restore NR level in at least one SG.

(2) After level is established, the effectiveness of the secondary heat sink is verified by decreasing RCS temperatures.

d. **TERMINATE RCS BLEED AND FEED HEAT REMOVAL.**

(1) With a verified secondary heat sink, the team performs a coordinated sequence for SI flow reduction and closing of pressurizer PORVs.

(2) After the completion of the sequence, the team is transferred to ES-1.1, SI Termination, for plant recovery.

QUESTION 30: (1.0)

Unit 2 is at Hot Shutdown with the steam dump system in Tave Mode. Steam Dump Controller offset has allowed Tave to increase up to 554°F. The Unit 2 semi-vital bus lost power 5 minutes ago.

Which ONE of the following identifies the status of the Unit 2 PORV's?

a. All open and being automatically controlled in Local/Auto.

b. All open and being automatically controlled in Remote/Auto.

c. All shut and being manually controlled in Local/Auto.

d. All shut and being manually controlled in Remote/Auto.

ANSWER: b

[RO: Tier 1/Group 3]
[SRO: Tier 1/Group 3]

Answer correct: Upon a loss of the semi-vital bus the S/G PORV's will transfer their control to the relay room and continue to modulate at setpoint.	Distractors plausible:.a/c/d- mis-understanding concerning transfer of control upon loss of power and how long the UPS for this system lasts.	Distractors incorrect: a/c/d – the control will be transferred to the relay room and controlled at the last setpoint.
K/A: E13-EK2.1	Objective: 1930	Source: New
Reference: ND-89.1-LP-2	Level: Comprehension	

(1) The right Bargraph shows demand to the controller in the Relay Room.

(2) The center Bargraph shows the demand from the controller in the Relay Room to the valve.

(3) When the LED is lit above the right Bargraph, the display will show a 0.0-100 demand to the controller. When the LED is over the center Bargraph, the display will show 0-100% demand to the valve.

(4) Similar to the Auto mode, if the LED is not lit above any of the Bargraphs, the operator cannot adjust valve position.

c. If normal and backup power from the SVB is lost to MBR 8, which supplies the relay rack controllers for the SG PORVs, the indications and controls supplied from this MBR remain functioning for a period of 30 minutes due to a battery supplied UPS in the MBR. This means that the SG PORVs remain operational for this 30 minute period. If SVB breaker #26 (Power supply for the SG PORV controllers in the MCR) trips, the SG PORVs will continue to operate in automatic based on the last setpoint set by the operator. When the breaker is reset, the operator will have to return the controller to Local operation. On a loss of the SVB, control of the SG PORVs will shift to the control unit in the Relay Room, which is powered by the UPS for a period of 30 minutes. During this time the control system will automatically operate the PORV based on the last setpoint set into the benchboard controller (usually 1035). At the end of the 30 minute period, the SG PORV will fail closed.

d. If power is lost for greater than 30 minutes and the UPS expires, indications and control will be lost. Upon power restoration, first the computer must reboot which takes about ≈3.5 minutes. After the computer has rebooted, it will send the data needed to reboot the individual MBR control processors. A total of ≈4.5 minutes will elapse before the control board indications are regained and the SG PORVs are operational.

e. When SVB power is restored, the benchboard controller will have an "R" backlit signifying Remote (Rack) control. To return to Local (Benchboard) operation, the "AM" key is pressed and the display will change from "R" to "L" signifying a return to Local (Benchboard) control.

f. Control signal for the controller comes from the control channel for steam line pressure.

g. Valve position indication (red/green lights) is provided directly above the controller.

8. Decay Heat Release Valve Controller

   a. This valve is manually isolated due to EQ concerns.

   b. This valve will not be operated during any mode of plant operation.

9. Main Steam Safety Valve Position Indication

   a. Regulatory Guide 1.97 (Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Environs Conditions During and Following an Accident) requires that the main steam safety valve position indication be provided to the control room operator in order to assess plant environs conditions during and following an accident.

QUESTION 31: (1.0)

With unit 1 at 100% power, the RO takes the following data from the power range NIs.

	N-41	N-42	N-43	N-44
Upper Detector Current	159.7	139.5	157.0	141.7
100% Upper Detector Current	266.6	234.5	262.9	237.4
Lower Detector Current	166.0	145.1	160.7	144.4
100% Lower Detector Current	278.6	236.7	270.0	240.2

Using the references provided, determine which ONE of the following is correct.

a. QPTR is 1.0253.

b. QPTR is 1.0158.

c. QPTR is 1.0195.

d. QPTR is 1.0209.

ANSWER: c

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct: Calculation attached	Distractors plausible: a – This is the value for N-41 upper; various math and/or transposition errors could lead to this conclusion; b & d– This is the value for N-43 upper; various math and/or transposition errors could lead to this conclusion.	Distractors incorrect: a, b & d – QPTR is 1.0195.
K/A: SYS001-A3.04	Objective:2559	Source: NEW
Reference: ND-93.2-LP-4	Level: Comprehension	

Radial Tilt Calculation Worksheet
Unit ___

Upper Flux Channels

Time	Data	NI-41	NI-42	NI-43	NI-44	Average NI $\frac{41+42+43+44}{4}$	Peak NI-__	Radial Tilt $\frac{PEAK}{AVERAGE}$
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							

Lower Flux Channels

Time	Data	NI-41	NI-42	NI-43	NI-44	Average NI $\frac{41+42+43+44}{4}$	Peak NI-__	Radial Tilt $\frac{PEAK}{AVERAGE}$
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							

Approved by: ______ Supt. Operations

Approved By: ______ SNSOC

Completed By: ______

Date: 6/10/98

Date: 6/10/98

Date: ______

Radial Tilt Calculation Worksheet
Unit 1

Upper Flux Channels

Time	Data	NI-41	NI-42	NI-43	NI-44	Average NI $\frac{41+42+43+44}{4}$	Peak NI-41	Radial Tilt $\frac{PEAK}{AVERAGE}$
	Actual	159.7	139.5	157.0	141.7			
	Normalized	266.6	234.5	262.9	237.4			
	Actual÷Normalized	0.5990	0.5949	0.5972	0.5969	0.5970	0.5990	1.0034
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							

Lower Flux Channels

Time	Data	NI-41	NI-42	NI-43	NI-44	Average NI $\frac{41+42+43+44}{4}$	Peak NI-42	Radial Tilt $\frac{PEAK}{AVERAGE}$
	Actual	166.0	145.1	160.7	144.4			
	Normalized	278.6	236.7	270.0	240.2			
	Actual÷Normalized	0.5958	0.6130	0.5952	0.6012	0.6013	0.6130	1.0195
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							
	Actual							
	Normalized							
	Actual÷Normalized							

Approved by: [signature], EOT
Supt. Operations

Approved By: [signature]
SNSOC

Completed By: ____________________

Date: 6/10/98

Date: 6/10/98

Date: ____________________

E. Quadrant Power Tilt

1. The symmetry of the radial power distribution is measured by the term Quadrant Power Tilt Ratio, QPTR. QPTR is defined in Technical Specifications in two ways:

Write on chalkboard the following definitions:

a. $$QPTR = \frac{\textit{maximum upper detector current}}{\textit{average of upper detector currents}}$$

OR

b. $$QPTR = \frac{\textit{maximum lower detector current}}{\textit{average of lower detector currents}}$$

2. In addition to being monitored by the Miscellaneous Indication and Control Drawer, the QPTR may be manually calculated by the operator.

Refer to QPTR Worksheet.

This form is used to calculate the QPTR using the outputs from the NI Drawers. The QPTR is normally performed using the computer program for QPTR. The method the computer uses and the worksheet method are the same.

Ensure the trainees are aware that they must be able to perform a manual QPTR without use of a computer.

Have trainees perform a QPTR calculation using the following data obtained from the PR NIs and the NI INFO Book

Reading on PR drawers

N-41		N-42		N-43		N-44	
Upper	Lower	Upper	Lower	Upper	Lower	Upper	Lower
110	121.6	119.5	118.2	96.2	111	129.0	119.4

PR Normalizing Values

N-41		N-42		N-43		N-44	
Upper	Lower	Upper	Lower	Upper	Lower	Upper	Lower
110.3	118.9	121.3	115.1	98.2	108.3	130.2	121.8

Using the above date the QPTR is:

Upper: .9973/.9882 = 1.009 Lower: 1.0269/1.0138 = 1.013

3. Technical Specifications section 3.12 imposes the following limitations on QPTR values:

   a. When reactor power is > 50%, QPTR is limited to a value of less than or equal to 1.02 (values of > 1.02 are indication of a dropped, stuck, or misaligned control rod).

QUESTION 32: (1.0)

Given the following plant conditions:

- Unit 1 is at 28% power.
- Annunciator 1C-C4, RCP 1C SEAL LKOFF HI FLOW, has just alarmed.
- "C" RCP no. 1 seal leak-off flow is indicating >8 gpm.

What actions are required?

a. 1) Stop "C" RCP.
   2) Verify "C" loop flow decreases to zero.
   3) Close "C" RCP seal leak-off valve.

b. 1) Go to 1-E-0, Reactor trip or Safety Injection, while continuing with AP-9.00.
   2) When the reactor is tripped, then close "C" RCP seal leak-off valve.
   3) Stop "C" RCP and verify ≤5 minutes have passed since seal failure occurred.

c. 1) Monitor RCP pump radial temperature and seal water return temperature.
   2) Initiate unit shutdown to allow stopping "C" RCP within 1 hour.
   3) When the unit is in HSD, then stop "C" RCP.
   4) Close "C" RCP seal leak-off valve.

d. 1) Go to 1-E-0, Reactor Trip or Safety Injection, while continuing with AP-9.00.
   2) When the reactor is tripped, then stop "C" RCP.
   3) Verify "C" loop flow decreases to zero.
   4) Close "C" RCP seal leak-off valve.

ANSWER: d

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct: per 1-AP-9.00, this is the correct sequence of actions.	Distractors plausible: a – with Rx power less than 30% (P-8) reactor will not **automatically** trip on single loop loss of flow; b – actions are correct, just not in the correct sequence; c – candidate misconception regarding no. 1 seal failure indications, these actions would be correct for less catastrophic seal problems.	Distractors incorrect: a – the facility license does not allow reactor operation with less than 3 Rx coolant loops in service; b – the RCP must be stopped before closing the seal leak-off valve; c – these actions are inappropriate for the conditions stated (e.g. – catastrophic failure of no. 1 seal).
K/A:SYS003-A1.09	Objective: 4010	Source: New
Reference: 1-AP-9.00	Level: Comprehension	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION:
- Total No. 1 Seal leakoff is the total of the indicated leakoff from No. 1 Seal and calculated No. 2 Seal leakoff. 13
- Number 2 Seal leakoff rate should be determined by the difference between PDTT inleakage rate before (from previously calculated 1-OPT-RC-10.0) and after the increase in Number 2 Seal leakoff. 13

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 1.	__VERIFY SEAL LEAKOFF - WITHIN NORMAL OPERATING RANGE IAW ATTACHMENT 1 • 1-CH-FR-1190, RCP SEAL LEAK OFF FLOW	<u>IF</u> affected RCP is <u>NOT</u> running, <u>THEN</u> GO TO Step 7. 13 <u>IF</u> affected pump is running, <u>THEN</u> do the following: 13 a) <u>IF</u> Number 1 Seal leakoff is low, <u>AND</u> is caused by high Number 2 Seal leakage, <u>THEN</u> GO TO Step 13. • PDTT Level - INCREASING • Standpipe Level - HI ALARM IN b) <u>IF</u> seal leakoff is less than 0.8 gpm, <u>THEN</u> GO TO Step 11. c) <u>IF</u> seal leakoff is between 0.8 gpm and 1.0 gpm, <u>THEN</u> GO TO Step 13. d) <u>IF</u> Total No. 1 seal leakoff is greater than 6.0 gpm, <u>THEN</u> GO TO Step 34. e) <u>IF</u> seal leakoff is between 5.0 gpm and 6.0 gpm, <u>THEN</u> GO TO Step 3.
2.	__GO TO STEP 23	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION:
- An RCP should be secured for low seal leakoff (less than 0.8 gpm) within 8 hours if Attachment 2 parameters are stable. 13
- An RCP should be secured for high seal leakoff using the following time limits: 1) Stop the RCP immediately (within 5 minutes) after a manual Reactor trip if Total No. 1 seal leakoff flow has increased to greater than 8 gpm. (regardless of Attachment 2 parameters.) 2) Stop the RCP within 8 hours if Total No. 1 seal leakoff flow is greater than 6 gpm and Attachment 2 parameters are stable. 13
- An RCP with high or low seal leakoff should be secured immediately (within 5 minutes) after a manual Reactor trip if any Attachment 2 parameter is continuously increasing or at Action level. 13

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
34.	__CHECK UNIT STATUS - ON LINE	GO TO Step 36.
35.	__REMOVE UNIT FROM SERVICE IAW SS DIRECTION: • GOP-2 Series Operating Procedures OR • 1-E-0, REACTOR TRIP OR SAFETY INJECTION OR • 0-AP-23.00, RAPID LOAD REDUCTION	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: If an immediate (within 5 minutes) RCP trip was performed due to high or low seal leakoff, the RCP SEAL LKOFF ISOL VV should be closed within three to five minutes after pump trip. 13

* *

36. __TRIP AFFECTED RCP IAW SS DIRECTION

37. __CLOSE THE AFFECTED RCP SEAL LEAKOFF ISOLATION VALVE AS NECESSARY:

- PP A/HCV-1303A, RCP A
- PP B/HCV-1303B, RCP B
- PP C/HCV-1303C, RCP C

38. __CHECK THERMAL BARRIER CC FLOW ON AFFECTED RCP - IN SERVICE

Do the following:

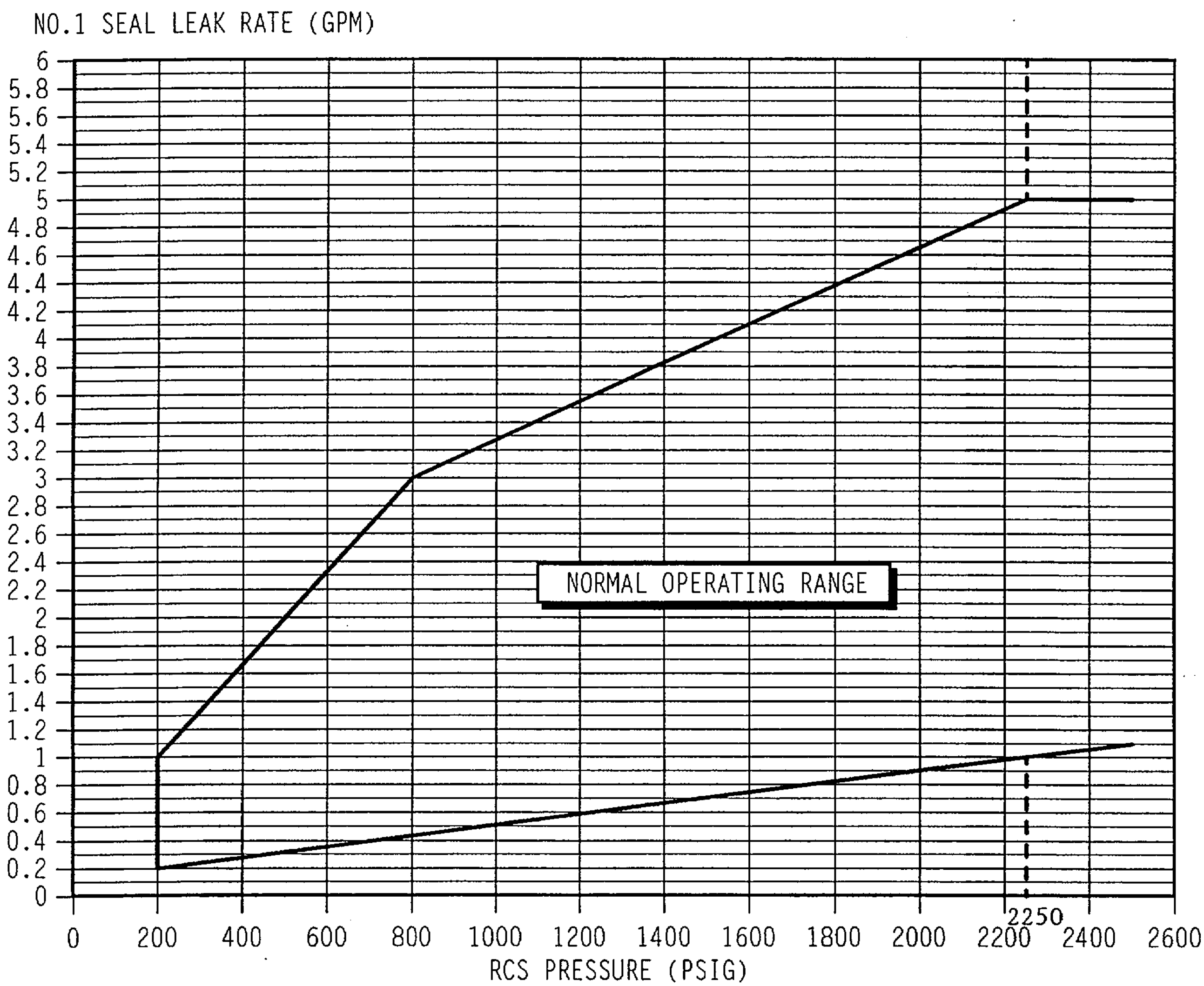
a) Verify open or open the following valves:

- TV-CC-120A, B, or C
- 1-CC-TV-140A
- 1-CC-TV-140B

b) Check for Thermal Barrier tube leakage:

- CC Surge Tank Level - INCREASING AT 1% PER MINUTE INDICATES APPROXIMATELY 35 GPM LEAKAGE
- Thermal Barrier CC temperature - INCREASING
- Thermal Barrier CC flow - HIGHER THAN NORMAL
- PRZR level - DECREASING
- PRZR pressure - DECREASING

(STEP 38 CONTINUED ON NEXT PAGE)



NO.1 SEAL LEAK RATE (GPM)
6
5.8
5.6
5.4
5.2
5
4.8
4.6
4.4
4.2
4
3.8
3.6
3.4
3.2
3
2.8
2.6
2.4
2.2
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
NORMAL OPERATING RANGE
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2250
2400
2600
RCS PRESSURE (PSIG)

QUESTION 33: (1.0)

Unit 1 was tripped from 100% power due to a Main Steam Trip valve spurious closure. Unit 2 remains at 100% power. During electrical swapover, breaker 15E1 tripped on fault.

Which ONE of the following describes the Unit 1 and Unit 2 Reactor Coolant Pump (RCP) status?

a. All RCPs running.

b. All RCPs running except 1B.

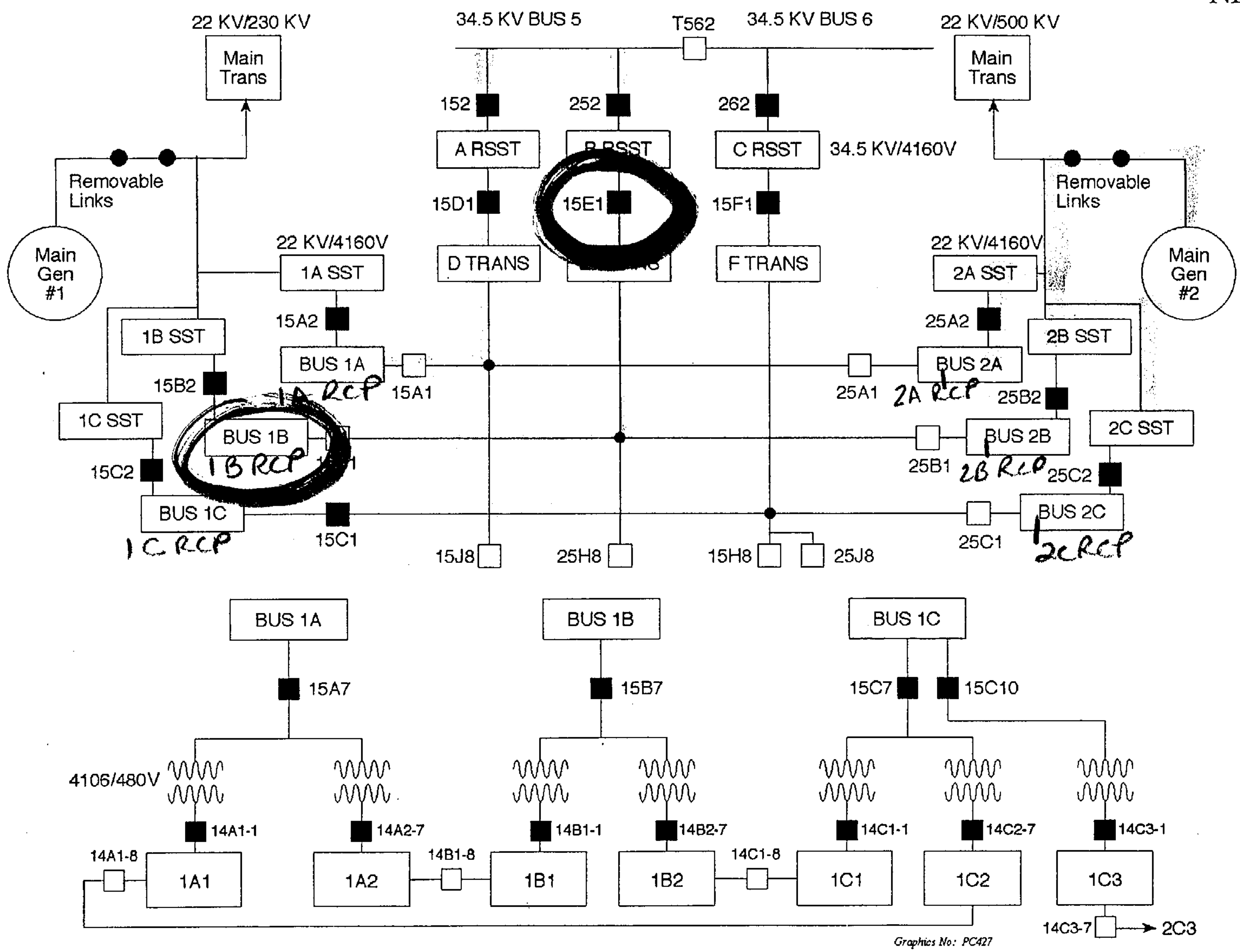
c. All RCPs running except 1B and 2B.

d. All RCPs running except 2B.

ANSWER: b

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct:	Distractors plausible:	Distractors incorrect:
Unit one supplied from RSST, Unit 2 supplied by Station Service. Unit Two RCPs unaffected by RSST problem. Unit 1 "B" station Service is lost, thus the "B" RCP is lost.	a. RSST normally only powers Emergency busses. Only during shutdown conditions does it power station Service. c. If unit 2 were shutdown, both Unit's "B" RCPs would be lost. d. If Unit 2 were Shutdown and Unit 1 were at power, Unit 2 "B" RCP would be lost.	a. Unit 1 "B" RCP lost. c. Unit 2 "B" RCP not lost d. Unit 2 "B" RCP not lost.
K/A: SYS003.K2.01	Objective: 2201	Source: NEW
Reference: ND-90.2-LP-2	Level: Comprehension	



STATION SERVICE DISTRIBUTION

QUESTION 34: (1.0)

The following conditions exist:

- RCS pressure is 300 psig.
- RCS temperature is 251°F.
- The RCS is solid.
- Flow through 1-RH-HCV-1142 is 50 gpm.

If a spurious SI occurs, which ONE of the following identifies RCS pressure response?

a. increases due to three running charging pumps.

b. increases due to letdown isolation.

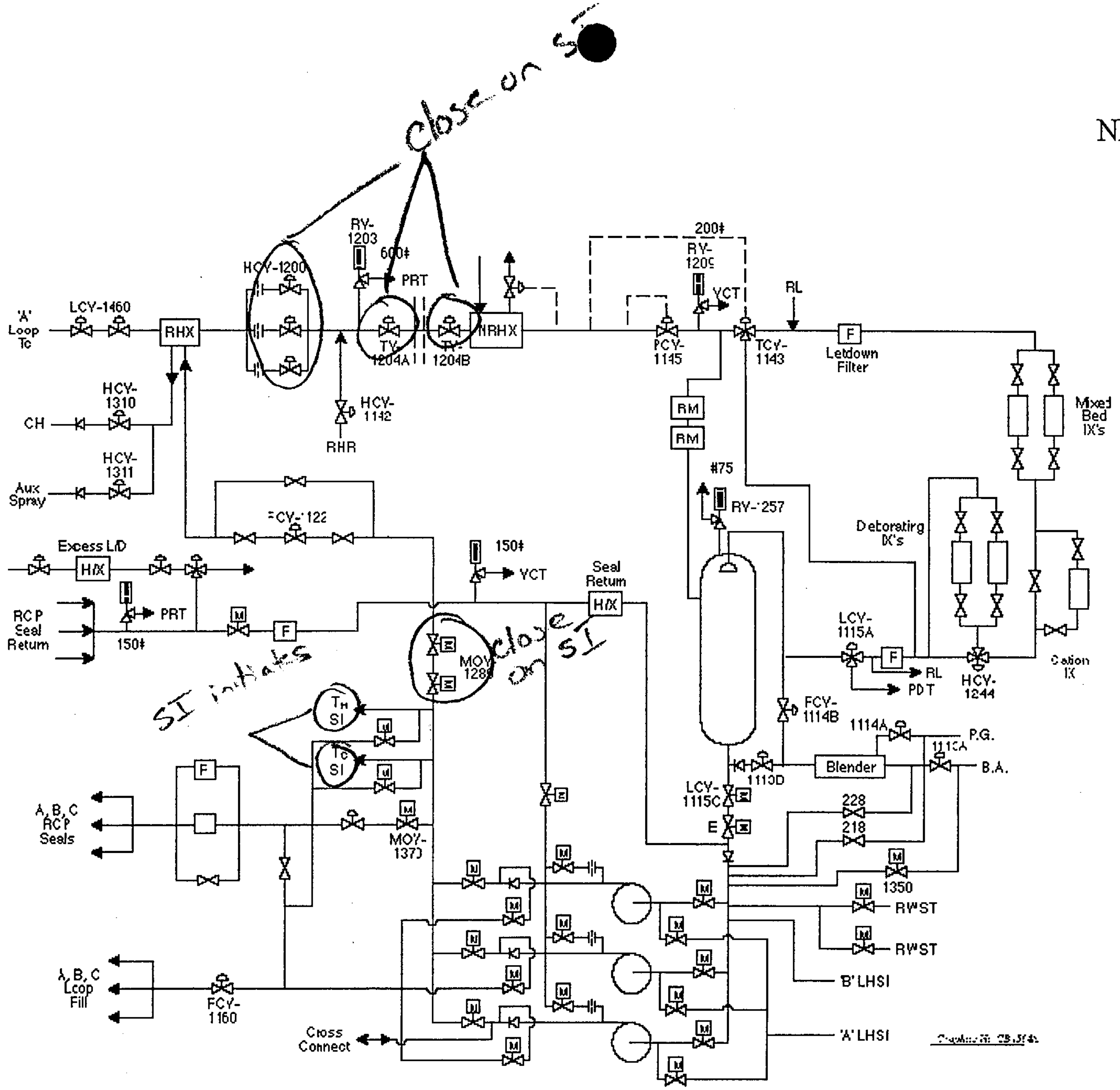
c. decreases due to the running RHR pump tripping.

d. decreases due to administrative isolation of containment I/A.

ANSWER: b

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct: TCV-1204A and B close upon SI which isolates the letdown from RHR. No letdown with charging causes RCS pressure to increase.	Distractors plausible: a – SI starts all charging pumps. C –RHR pumps will trip after 7 seconds if an SI and UV exists. D – this is an administrative requirement during an SI.	Distractors incorrect: a – all but one charging pump will be in PTL in this plant condition. C – UV does not exist. D – The fail position of the valves is to cause RCS pressure to increase.
K/A: SYS004.A1.03	Objective: 1792	Source: NEW
Reference: ND-88.3-LP-2	Level: Comprehension	



CHARGING AND LETDOWN SYSTEM DRAWING

QUESTION 35: (1.0)

With Unit 1 at 100% power, a partial phase "1" containment isolation signal results in closure of letdown isolation valve 1-CH-TV-1204A. No other valves or components are affected by the signal.

Which ONE of the following is correct concerning the affect of this transient on letdown flow?

a. Indicated Letdown flow goes to zero; actual Letdown flow continues to the PRT.

b. Indicated Letdown flow goes to zero; actual Letdown flow continues to the PDT.

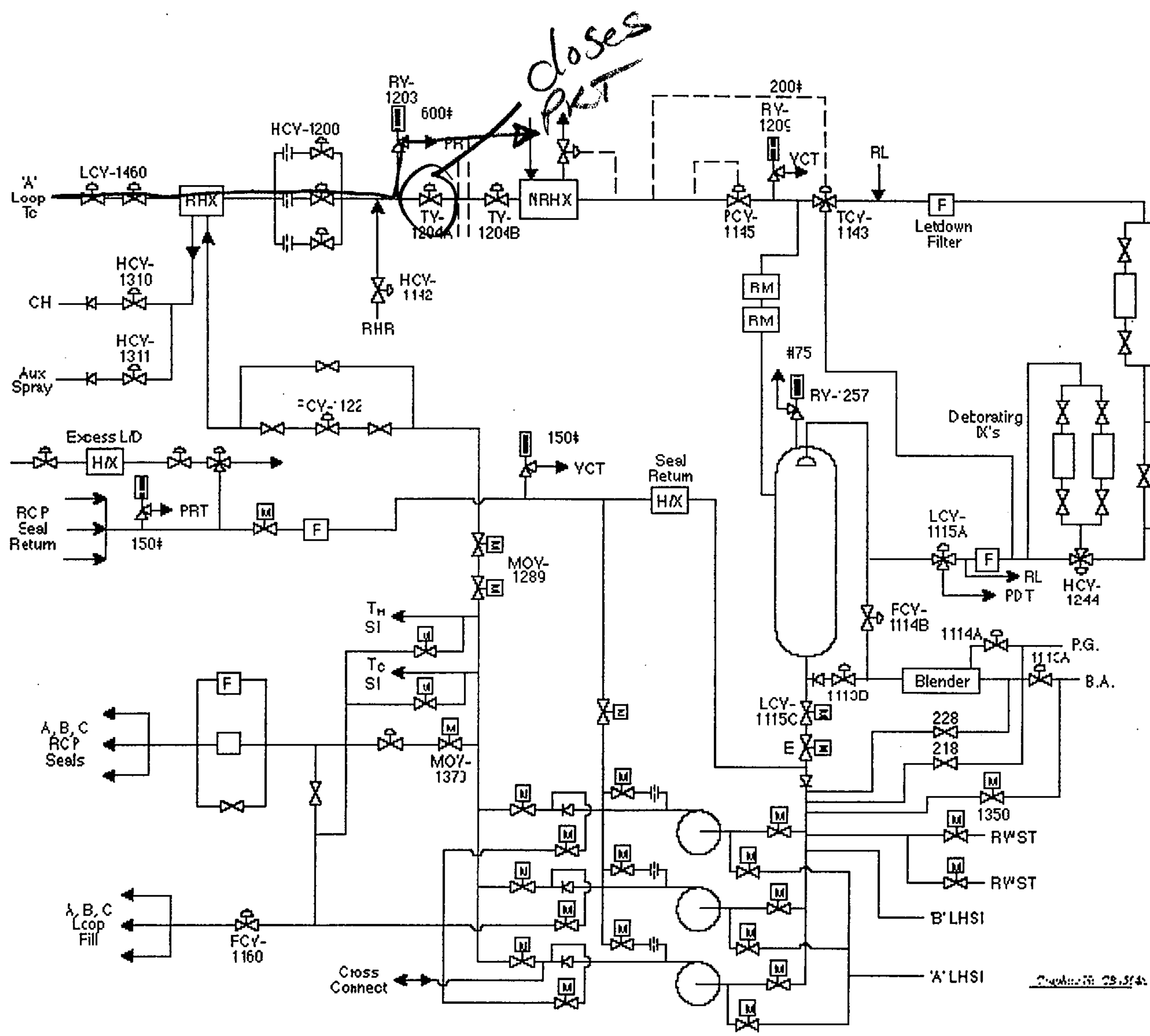
c. Indicated Letdown flow fluctuates as the relief valve lifts; actual Letdown flow continues to the PRT.

d. Indicated Letdown flow fluctuates as the relief valve lifts; actual Letdown flow continues to the PDTT.

ANSWER: a

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct: letdown flow transmitter is downstream of containment isolation trip valves; relief valve is upstream of trip valve TV-1204A; relief valve lifts at 600 psig and discharges to PRT.	Distractors plausible: b & d – candidate misconception concerning discharge flow path of the letdown relief valve. c & d – candidate misconception concerning the location of the letdown flow transmitter in the flow path.	Distractors incorrect: b & d – letdown relief valve discharges to the PRT, not the PDTT or PDT. c & d – letdown flow transmitter is downstream of the containment isolation trip valves; indicated flow will be zero.
K/A: SYS004.A2.12	Objective: 1792	Source: NEW
Reference: ND-88.3-LP-2	Level: Comprehension	



CHARGING AND LETDOWN SYSTEM DRAWING

QUESTION 36: (1.0)

Following a LOCA with core damage, the following conditions were determined to have existed:

- Maximum clad temperature reached 2175°F.
- Cladding oxidation was 15% of original cladding thickness.
- Hydrogen generation from Zirc-water reaction was 3%.
- One train of ESF equipment failed to operate.

Which ONE of the above conditions exceeded (violated) 10CFR50.46, ECCS acceptance criteria?

a. Clad temperature.

b. Clad oxidation.

c. Hydrogen generation.

d. ESF equipment availability.

ANSWER: c

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct: Actual allowable is <1%	Distractors plausible: a. Allowable up to 2200. b. Allowable up to 17% c. One train assumed inoperable, however the fact that one train failed may lead to the belief this is why damage occurred.	Distractors incorrect: a. <2200 degrees b. < 17% d. One train of Safeguards is all that is required to provide adequate protection.
K/A: SYS013.K3.01	Objective: 2303	Source: NEW
Reference: ND-91-LP-2	Level: Knowledge	

2. These criteria are:

Write on chalkboard the following ECCS Acceptance Criteria: Cladding temperature < 2200°F OK Cladding oxidation < 17% of cladding thickness OK Hydrogen generation < 1% of amount if all Zr reacted UNSAT Geometry core remains intact and coolable Long-term cooling core remains in configuration to allow long-term decay heat removal OK w/ 1 train

a. Cladding Temperature < 2200°F

(1) This first limit on clad temperature essentially determines the success with which the other four criteria are satisfied.

(2) Zircaloy can react with water and steam in a high temperature environment to produce hydrogen.

(3) Zirc cladding oxidation occurs throughout the life of the plant, but at normal operating temperature, the accumulated oxidation layer thickness is inconsequential. However, as cladding temperatures rise, the oxidation rate increases exponentially.

(4) 1800°F - oxidation rate significant
2000°F - only takes 1 hour to reach 17% thickness limit
2200°F - oxidation rate accelerates significantly
4800°F - process is auto-catalytic

QUESTION 37: (1.0)

Which ONE of the following evolutions is prohibited?

a. Entry into the Incore Sump Room with the flux thimbles retracted.

b. Access into an RCP Motor Cubicles during reduced inventory conditions.

c. Entry into an airborne radioactivity area without a SCBA.

d. A ten minute entry into 15 Rem/Hr area to protect property during a Site Area Emergency.

ANSWER: a

[RO: Tier 3]
[SRO: Tier 3]

Answer correct: Entry into the Incore Sump Room is prohibited with the flux thimbles retracted.	Distractors plausible: b/c/d – all of these conditions present situations where a higher than normal exposure could be received that might be misunderstood to prevent entry.	Distractors incorrect: Although these areas have higher than normal exposure – this level of exposure is allowed under administrative control.
K/A: GEN-2.3.10	Objective: 2576	Source: New
Reference: ND-93.2-LP-7	Level: Knowledge	

Description of Event

The following is a description of the events leading up to and resulting in the unexpected personnel radiation exposure.

On February 8, 1988, the Reactor Engineers performed the monthly flux map on Surry Unit 2 in accordance with Periodic Test Procedure number 2-PT-28.2. The "B" in-core detector was inoperable and the four other detectors were used to collect data. The "A" detector was used in the "Emergency" mode to monitor the core locations normally monitored with the B detector. Following recording of the flux trace of core location J-5, core location D-10 was selected on the "B" ten-path transfer device. During this scan, the "A" detector became stuck in the reactor core and could not be withdrawn with the drive unit. Since the minimum number of required thimbles had been monitored to complete the flux map, the test was terminated at this point. The position of the stuck detector was not recorded on the Periodic Test.

The Instrument Department subsequently submitted a work request to unstick and replace the "A" detector. The work request was submitted on February 11, 1988, and initially scheduled to be performed on February 24, 1988, prior to the next monthly map. The schedule was later delayed to the next week to coordinate the work with other planned containment entries.

Pre-job briefings were held by the Instrument Department on the evening of March 2nd and the morning of March 3rd to discuss in detail the procedure to be followed to replace the detector. The work was to be performed under Instrument Maintenance Procedure number IMP-C-IFM-20. Since the detector was known to be stuck, a procedure deviation was written by the lead instrument technician to add steps to free up the detector. Under the approved deviation, the drive box was to be disassembled and the drive cable pulled by hand enough to unstick the detector. The drive box was then to be reassembled and the drive unit used to fully withdraw the detector and place it into shielded storage prior to proceeding with the replacement.

On March 3, 1988, two nuclear instrument technicians and a health physics technician entered the Unit 2 containment with the reactor at power to perform the planned maintenance. The lead nuclear instrument technician and his immediate supervisor were stationed in the Main Control Room. After communications were established, the job proceeded as planned. Constant health physics coverage was established and the drive box disassembled. The cable was pulled approximately two feet by hand. The cable still felt stuck, however the drive box was reassembled and an attempt made to drive the detector as planned with the drive unit. The drive motor clutch slipped without moving the cable in either direction. At this point, the instrument technicians decided to continue to pull the detector back by hand until it became free. The Instrument Supervisor in the Control Room concurred with this decision.

The drive box was again disassembled, and the cable pulled repeatedly in pulls of approximately 20 inches in length. Instrument Technician No. 1 pulled the cable while Instrument Technician No. 2 took up slack with the takeup reel. When the detector was pulled into the Seal Table Room, the area radiation monitor alarmed in the Control Room as expected. When the detector passed through the "B" ten-path transfer device, the cable became unstuck and was much easier to pull. However, since the cable appeared to be damaged, it was decided to continue to pull the detector back by hand to the inserted/withdrawn limit switch, rotate the five path transfer device, then push the detector into storage.

The cable pull then continued very slowly being cautious not to pull the detector beyond the crane wall. Simultaneously with getting the withdrawn light in the Control Room, the Health Physics Technician observed the radiation level to be rapidly increasing in the area being monitored in close proximity to the tubing entering the drive gear. At this point, the Health Physics Technician terminated the work and quickly moved the instrument technicians and himself off the work platform to a low dose area where he read their self-reading dosimeters. The Health Physics Technician then briefly returned to the platform to survey with the Teletector in an extended position. The survey measurements indicated that the gamma radiation field was on the order of 100 R/hr in the area between the "A" and "B" drive boxes and greater than 1000 R/hr in the area near the front of the "A" drive box. The technicians then exited the containment and returned their dosimetry to Health Physics where their TLD's were read. Measured whole body doses ranged from a low of 275 mrem to a maximum of 524 mrem. A debriefing was held with Station Management that afternoon to determine what happened. At that time, it was believed that the detector had been pulled beyond the crane wall resulting in the high radiation levels. A Station Deviation Report was written on March 4, 1988.

Cause of High Radiation Levels

Unexpected high radiation levels occurred in the area where the technicians were working when the activated cable was pulled through the crane wall. The carbon steel drive cable had become highly activated due to exposure to neutron flux while the detector was stuck in the core.

QUESTION 38: (1.0)

The following parameters are indicated on the ICCM:

- Wide range pressure – 1988 psig.
- Highest quadrant CETC – 606°F.
- Lowest quadrant CETC – 600 °F.
- Average of the 5 highest CETC – 603 °F.
- "A" loop Th – 596 °F.
- "B" loop Th – 594 °F.

Which ONE of the following identifies the expected displayed subcooling margin?

a. 40 °F

b. 36 °F

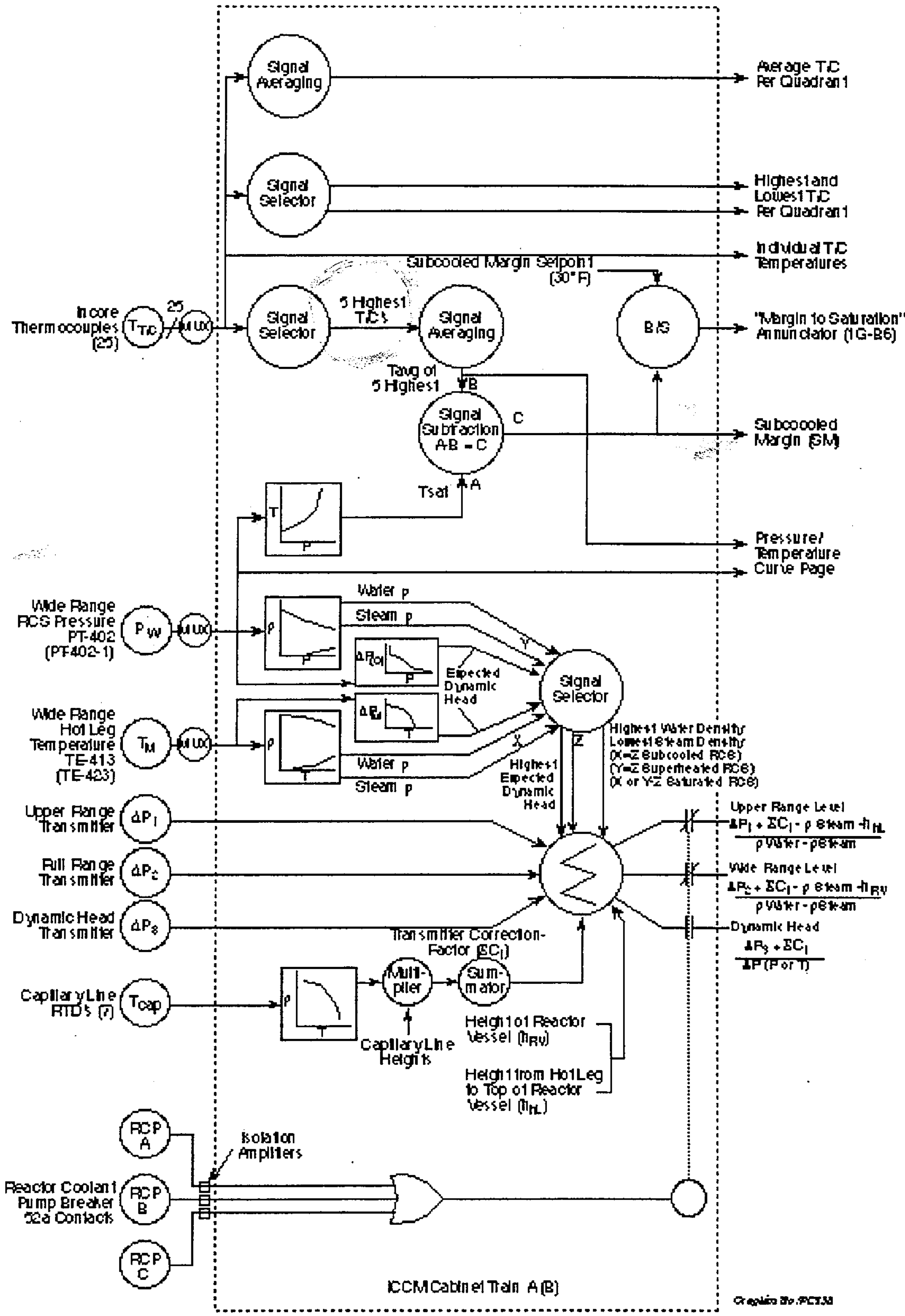
c. 33 °F

d. 30 °F

ANSWER: c

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct: Correct answer based on steam tables when using the Average of the 5 highest CETC and wide range pressure.	Distractors plausible: a – Result from using A loop Th b – Result from using the lowest quandrant CETC d – Result if using the highest quadrant CETC which is the most conservative.	Distractors incorrect: a/b/d – Wrong temperatures used for calculations.
K/A: SYS017.K4.01	Objective: 2691	Source: NEW
Reference: ND-93.4-LP-3	Level: Comprehension	



ICCM FUNCTIONAL BLOCK DIAGRAM

QUESTION 39: (1.0)

With Unit 2 at 100% power, a spurious Hi Hi CLS occurs.

Which ONE of the following identifies the effect on indicated partial pressure?

a. The indication is inaccurate, indicated pressure will be lower than pre-event values.

b. The indication is inaccurate, indicated pressure will be higher than pre-event values.

c. The indication is accurate, indicated pressure will be lower than pre-event values.

d. The indication is accurate, indicated pressure will be higher than pre-event values.

ANSWER: a

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct: Since the Component Cooling is isolated the Tsat will increase resulting in Ppartial decreasing.	Distractors plausible: b, c, d - misunderstanding of partial pressure calculation.	Distractors incorrect: b – partial pressure will decrease due to CARF discharge temperature increase. c/d - Indication invalid due to no cooling water to lower the air to the dewpoint temperature.
K/A: SYS022.A3.01	Objective: 1879	Source: NEW
Reference: ND-88.4-LP-5	Level: Comprehension	

Refer to/display H/T-5.3, Containment Vacuum Circuitry, and use for the following information.

(1) RTD-CV-101A1, 2 and 3 sense the air temperature at the discharge of the containment air recirc. coolers. At this point, the air temperature is the same as the saturation temperature of the water vapor in the air, and vapor pressure is easily found. For conservatism, the lowest of the three (3) temperatures is selected by the low signal select summator PM-CV-101A-1 (auctioneer unit).

ONLY IF Cooling available to lower to this point.

(2) The lowest temperature output from the summator is sent to function generator PM-CV-101A-2, which calculates vapor pressure.

(3) PM-CV-101A-3 takes the total pressure output of PT-CV-101A and subtracts from it the vapor pressure output of PM-CV-101A-2 to calculate partial pressure. The partial pressure is indicated on PI-CV-101A-2, in the Control Room. The output from the partial air pressure summator is compared with the partial pressure setpoint and used to operate alarms and the containment vacuum pump when it is in auto.

4. Air Flow Instrument (FT-CV-150)

a. A laminar flow element (FT-CV-150) measures flow in the combined discharge header of the containment vacuum pumps. The range of the MCR flow meter is 0-20 CFM.

b. Indication is provided in the Main Control Room and on the P-250 computer.

B. System Operation

QUESTION 40: (1.0)

Unit One is operating at 100% when the Reactor Operator notices that Main Feed Pump suction pressure suddenly decreased and stabilized at the lower pressure.

Which ONE of the following initiated the sudden pressure decrease in Main Feed Pump suction pressure?

a. Another Polishing Building ion exchanger bed was put into service.

b. FCV-CN-107 failed open.

c. AOV-CN-122 failed open.

d. The HP Heater Drain Pump Normal Level Control valve failed open.

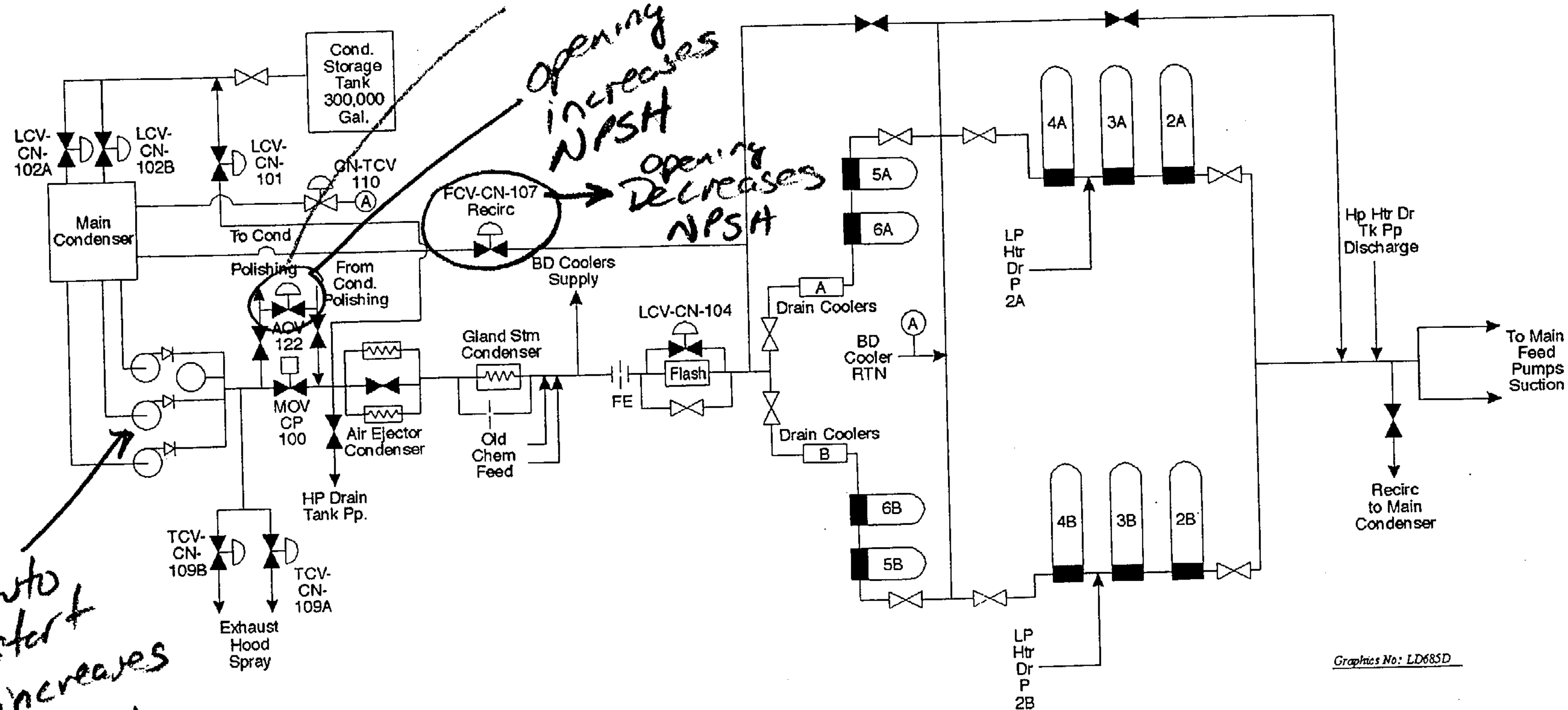
ANSWER: b

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct:	Distractors plausible:	Distractors incorrect:
FCV-CN-107 failing open would increase the flow of condensate to the main condenser and simultaneously decrease the amount flowing to the main feed pumps. This would result in Main Feed Pump discharge pressure decreasing.	a, c and d – Will affect Main Feed Pump (MFP) suction pressure.	a, c, and d - Will increase the MFP suction pressure.
K/A: SYS056.K1.03	Objective: 2033	Source: New
Reference: ND-89.3-LP-2	Level: Comprehension	

ND-89.3-H/T-2.2

Additional Bed increases NPSH



Opening increases NPSH

Opening Decreases NPSH

Auto start increases NPSH

CN pump Auto Starts
1)
2)
3)

CN pump Trips
1)
2)

CN pump Auto-Start Block
1)
2)

CN System Recirc Flowpaths
1)
2)
3)

Chemicals Added to CN System
1)
2)
3)

CP-MOV-100 Open Signals
1)
2)

CP-AOV-122 Open Signals
1)
2)
3)

CONDENSATE SYSTEM WORKSHEET

QUESTION 41: (1.0)

The following conditions exist:

- Unit 2 is at 100% power.
- All Main Feed Regulation Valve Controllers are in automatic.
- Rod Control is Manual.
- The turbine is in Operator Auto and IMP-OUT.
- Steam Flow, Feed Flow, and Pimp are selected for Channel IV.

Which ONE of the following identifies S/G level response if pressure transmitter 1-MS-PT-1447 fails low?

a. Steam Generator levels will decrease to the SF/FF mismatch with S/G low level Rx Trip setpoint.

b. Steam Generator levels will decrease to the S/G low low level Rx Trip setpoint.

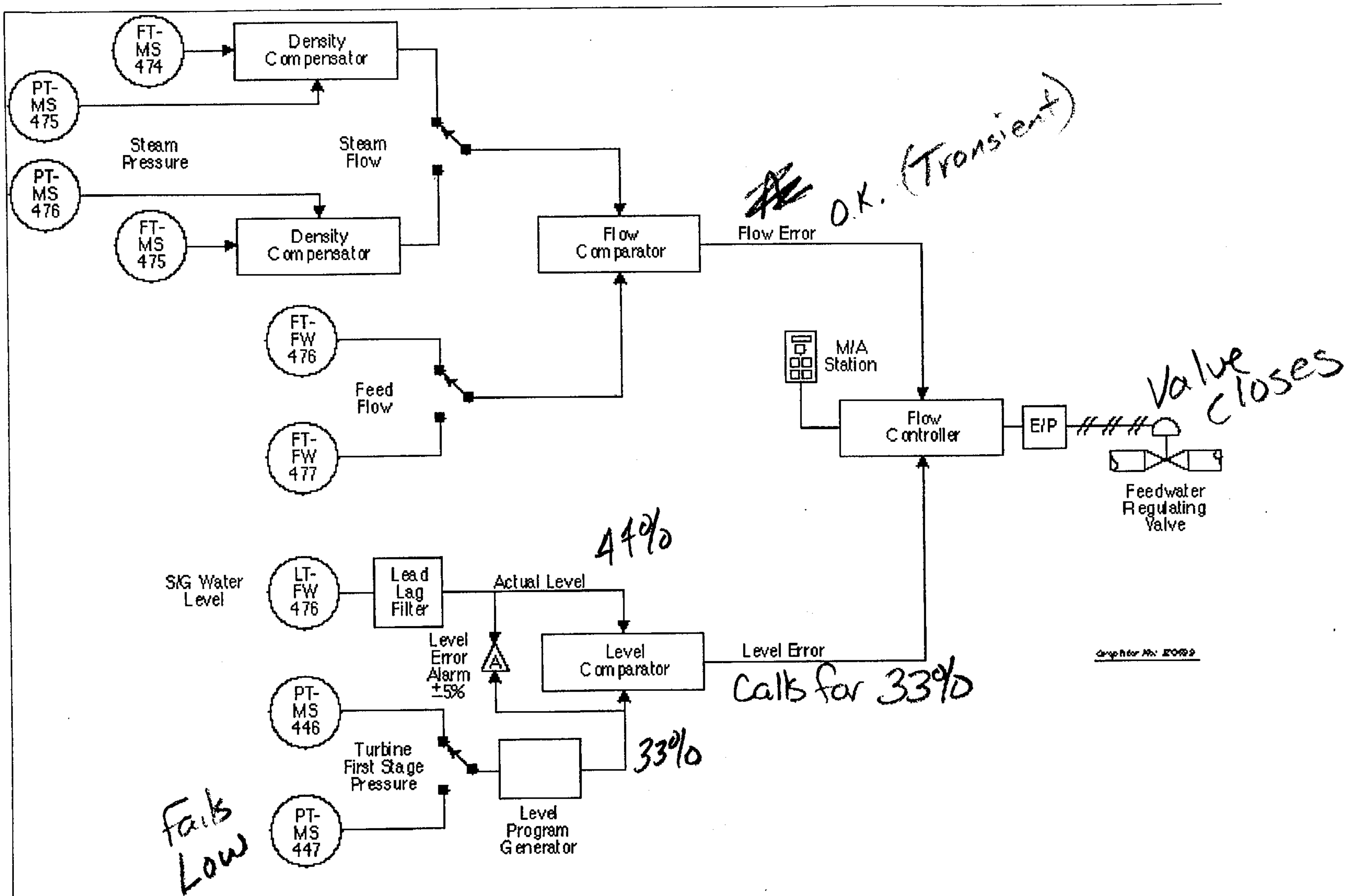
c. Steam Generator levels will decrease to 33%.

d. Steam Generator levels are not affected.

ANSWER: c

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct: This is the low limit of the steam generator level program.	Distractors plausible: a/b – S/G level will decrease. D – Misconception on which channel PT-447 .	Distractors incorrect: a/b level not go down to the 20% and 17% S/G level trips. D – S/G level will decrease.
K/A: SYS059.K3.03	Objective: 2667	Source: New
Reference: ND-93.3-LP-8	Level: Comprehension	



SGWLC DIAGRAM

QUESTION 42: (1.0)

With the Unit One Steam Driven Auxiliary Feed pump tagged out for maintenance, what is the Unit 1 source of auxiliary feedwater if both Units are in emergency procedure ECA-0.0?

a. Fire Protection and Domestic Water Tanks (1-FP-TK-1A/B).

b. Emergency Condensate Makeup Tank (1-CN-TK-3).

c. Distillate Water Storage Tank (1-WT-TK-101).

d. Emergency Condensate Storage Tank (1-CN-TK-1A).

ANSWER: a

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct: This is the only water source with sufficient driving head to makeup to a SG (via the Diesel Driven Fire Pump).	Distractors plausible: b. This is the normal backup water source to the Aboveground Emergency Condensate tank. c. ALL makeup water to the station (except Firewater) comes from TK-101. d. This is the normal source of water to the AFW pumps.	Distractors incorrect: b/c/d. Neither the motor driven, booster pumps, or cross ties have power to deliver this water source to the SGs.
K/A: SYS061.K1.07	Objective: 2934	Source: New
Reference: ECA-0.0	Level: Knowledge	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19.	__NOTIFY ELECTRICIANS TO MONITOR THE FOLLOWING: • Station batteries • Black batteries	
20.	__CHECK ECST LEVEL - GREATER THAN 20%	Locally make up from CST. <u>IF</u> CST <u>NOT</u> available, <u>THEN</u> consider use of one of the following: • Crosstie from Unit 2 • Fire water

* *

<u>CAUTION</u>:
- SG pressures should be maintained greater than 150 psig to prevent injection of SI accumulator nitrogen into the RCS.
- SG narrow range level should be maintained greater than 11% [22%] in at least one intact SG. If level can <u>NOT</u> be maintained, SG depressurization should be stopped until level is restored in at least one SG.

* *

<u>NOTE</u>:
- SGs should be depressurized at the maximum controllable rate to minimize RCS inventory loss.
- PRZR level may be lost and reactor vessel upper head voiding may occur due to depressurization of SGs. Depressurization should <u>NOT</u> be stopped to prevent these occurrences.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21.	__DEPRESSURIZE ALL INTACT SGs TO 175 PSIG:	
	a) Check AAC Diesel Generator - SUPPLYING BOTH BUSES 1J AND 2H	a) GO TO Step 21c.
	b) GO TO Step 22	

(STEP 21 CONTINUED ON NEXT PAGE)

QUESTION 43: (1.0)

Unit One is performing a normal startup and is currently at 2% power when a safety injection occurs. The reactor operator notices that both motor driven auxiliary feed pumps are running after the safety injection.

Which ONE of the following identifies the time after the safety injection that the motor driven auxiliary feed pumps started?

a. 0 seconds.

b. 10 seconds.

c. 27 seconds.

d. 50 seconds.

ANSWER: d

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct: The time delay associated with the AFW pumps is 50 seconds.	Distractors plausible: a. discounting any time delays, knowing AFW is an auto function of SI. In addition, EDG load sequencing is 0 Seconds. b. EDG load sequencing with a SI is 10 seconds. c. Normal anticipated response (trip from 100% power), due to AMSAC auto start signal (preempts SI signal). AMSAC actuates 27 seconds after SG levels <13%	Distractors incorrect: A/b. EDG load sequencing not active (no UV signal). There is a time delay associated with the AFW pumps c. Initial power is <37%, AMSAC is not enabled.
K/A: SYS061.K4.06	Objective: 2050	Source: New
Reference: ND-89.3-LP-4	Level: Comprehension	

1-FW-P-2

*LO-LO LEVEL 2/3 ch<17% NR IN ANY 2/3 S/Gs

*LOSS OF VOLTAGE ON 2/3 4160V STATION SERVICE BUSES

*AMSAC INITIATION ON 2/3 CH <13% IN ANY 2/3 S/Gs <u>AND</u> BOTH 1ST STAGE PRESSURES >37%

NOTE: After the AMSAC signal is initiated, the AFW pumps will continue to run until the AMSAC signal is manually reset.

1-FW-3A,3B

*LO-LO LEVEL 2/3 ch <17% NR IN ANY S/G

*LOSS OF VOLTAGE ON 2/2 RSS (X-FER BUSES) for affected unit

*ANY SI SIGNAL (AFTER 50 SEC T.D.)

*1/2 MFP BKRS OPEN ON BOTH MFPs

*AMSAC INITIATION

g. In the event an undervoltage condition occurs on a 4160v emergency bus after an SI or Hi-Hi CLS event has been initiated, the respective motor driven AFW pump will trip, and the automatic and manual start signals will be momentarily blocked (10 sec. for an SI; 140 sec. for a Hi-Hi CLS). The pump will auto-start again after the blocking signal is removed (times-out). This load sequencing will stagger the emergency loads starting on EDG, thus preventing an overload condition.

h. The turbine driven AFW pump will remain running after an AUTO START, even if the AUTO START signals clear, until the operator places the control switches for both PCV-MS-102 A & B to OPEN/RESET then returns them to the close position.

QUESTION 44: (1.0)

During release of the "A" WGDT, which ONE of the following will cause an automatic termination of the release?

a. FI-GW-101, WGDT effluent flow meter, fails high.

b. 1-GW-RI-101, Process Vent Radiation Monitor, is inadvertently de-energized.

c. 1-GW-AR-150A, Waste gas analyzer oxygen, fails high.

d. VG-RI-131-1, Vent Stack #2 effluent monitor, fails high.

ANSWER: b

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct: A de-energized RM will pick up relays to provide auto functions. WGDT is released through 1-FCV-101, which receives an auto close signal on high PROCESS VENT radiation.	Distractors plausible: a. A failed high conditions is non-conservative. c. High oxygen content is non-conservative d. Vent Vent auto actions is a common misconception.	Distractors incorrect: a. No auto functions associated with high flow. c. No auto actions associated with Waste gas analyzer d. No auto actions associated with High Vent Vent RM.
K/A: SYS071.A2.02	Objective: 2727	Source: NEW
Reference: ND-93-05-01	Level: Comprehension	

Several equipment failure conditions are monitored which produce a FAIL alarm and in some cases an error message. The fail condition is "TRUE" whenever any equipment failure is detected and "FALSE" when no equipment failures are detected. When a fail condition occurs, other than power failure, the red FAIL alarm indicator illuminates and the fail relay coil de-energizes. The Fail alarm logic is always fail-safe and the following is a brief description of the ratemeter failure modes:

- **<u>Power Failure</u>** - If power is lost to the ratemeter, the bargraph, alarm indicators, and the displays are blanked (turned off). The HIGH, WARN, and FAIL relay coils de-energize and open the alarm contacts.

- **<u>No Count Failure</u>** - If no pulses are received by the ratemeter for five minutes (30 minutes on the air ejector ratemeters), a no count failure is detected. A no count alarm usually indicates a failure in the detector or high voltage supply. The ratemeter display, however may read zero for five to thirty minutes or more without a low signal fail alarm. This is because the preamplifier is reporting a non-zero dose rate that is below the low range value.

- **<u>MPU Failure</u>** - If the fail timer circuit (Watchdog circuit), which checks the MPU (Main Processor Unit) function, is allowed to time out (because of a hardware failure), a failure condition will be indicated.

(5) Filter fault light - Indicates paper jam, out of paper, or take-up mechanism malfunction.

(6) Flow fault light - Indicates low flow condition for the monitor.

(7) Purge pushbutton - Operates valves to take suction from the Aux Building to purge the gas monitor. Valves are positioned to purge whenever the button in held in. Valves automatically return to normal when the button is released.

6. Automatic Functions on High Radiation Alarms

a. Certain effluent processes have automatic functions associated with high radiation alarms from various radiation monitors. The automatic function is to isolate the effluent flowpath or to isolate specific inputs into the effluent flowpath.

b. Each of the radiation monitors listed on the transparency has an automatic function that a high alarm condition will activate.

Refer to/display H/T-1.9, Radiation Monitors Automatic Actions, to assist with the following information.

(1) Process vent particulate and gas monitors (Victoreen & Kaman) (RM-RI-101/102 and GW-130-1).

(a) Shuts FCV-GW-160 and 260 - Isolates both units containment vacuum pump discharge.

(b) Shuts FCV-GW-101 - Isolates WGDT discharge.

(2) Component cooling water monitor (CC-RI-105/106) - Shuts CC surge tank vent valve.

(3) Condenser air ejector monitor (SV-RI-111)

(a) Opens TV-SV-102, Lines up air ejector discharge to containment. This TV closes on Hi CLS, but will auto reopen when CLS reset if High Alarm still present.

(b) Shuts TV-SV-103, Isolates discharge to atmosphere.

(4) Containment particulate and gas (RM-RI-159/160); and manipulator crane monitors (RM-RI-162)

(a) Trips affected unit's purge supply fans (4A and 4B).

(b) Shuts MOV-VS-100A, B, C and D, purge isolation valves.

(c) Shuts suction valves for containment instrument air compressor (TV-IA-101 A/B) which opens the outside suction valve.

c. If the automatic actions did not occur for a high radiation alarm, the operator is responsible to manually perform the isolations.

QUESTION 45: (1.0)

A High alarm on the _________ Radiation monitor will automatically close the Unit 1 Containment purge isolation MOVs.

a. Unit 1 Containment, 1-RM-RMS-163

b. Unit 1 Containment High Range, 1-RM-RMS-161

c. Unit 1 Containment Manipulator crane, 1-RM-RMS-162

d. Unit 1 Containment High Range (CHRRMS), 1-RM-RMS-127

ANSWER: c

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct: 1 of 3 Radiation monitors which will actuate the purge MOVs	Distractors plausible: a/b/c all containment monitors (located in area of concern)	Distractors incorrect: Only RM-RI-159/160/162 will initiate this function
K/A: SYS072.K4.01	Objective: 2728	Source: NEW
Reference: ND-93-05-01	Level: Knowledge	

(a) Shuts FCV-GW-160 and 260 - Isolates both units containment vacuum pump discharge.

(b) Shuts FCV-GW-101 - Isolates WGDT discharge.

(2) Component cooling water monitor (CC-RI-105/106) - Shuts CC surge tank vent valve.

(3) Condenser air ejector monitor (SV-RI-111)

(a) Opens TV-SV-102, Lines up air ejector discharge to containment. This TV closes on Hi CLS, but will auto reopen when CLS reset if High Alarm still present.

(b) Shuts TV-SV-103, Isolates discharge to atmosphere.

(4) Containment particulate and gas (RM-RI-159/160); and manipulator crane monitors (RM-RI-162)

(a) Trips affected unit's purge supply fans (4A and 4B).

(b) Shuts MOV-VS-100A, B, C and D, purge isolation valves.

(c) Shuts suction valves for containment instrument air compressor (TV-IA-101 A/B) which opens the outside suction valve.

c. If the automatic actions did not occur for a high radiation alarm, the operator is responsible to manually perform the isolations.

QUESTION 46: (1.0)

The Unit 1 charging pumps are in the following configuration:

- 1-CH-P-1A – auto-off
- 1-CH-P-1B – auto-off
- 1-CH-P-1C – running on the "H" bus (15H6)

The Load Tap changer on 'C' RSS transformer fails and immediately lowers transformer voltage to 3037 volts AC. Which charging pumps will be running on Unit 1, 65 seconds after the transient?

a. 1-CH-P-1A, 1-CH-P-1B, 1-CH-P-1C

b. 1-CH-P-1B, 1-CH-P-1C only

c. 1-CH-P-1A, 1-CH-P-1B only

d. 1-CH-P-1C only

ANSWER: b

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: 3037 is 73% of nominal (UV signal 75%). This causes "B" charging pump to auto start and disables the UV trip of "A" and "C" charging pumps. The SI signal will start the "A" charging pump resulting in all three running	Distractors plausible: b – Belief that UV trip is still active with the EDG carrying the emergency bus; c Belief that "C" charging pump trips and lockouts out (86 device) on n UV condition d – unrecognized condition that an undervoltage condition exists.	Distractors all- does not identify all running pumps.
K/A: 006-A3.05	Objective: 1818	Source: New
Reference: ND-88.3-LP-5	Level: Comprehension	

(c) 90% voltage for 7 seconds with SI

(4) All other charging pump breakers open

b. Charging pump B (15J5)

(1) Low charging header pressure - $\leq$ 1176 psig for $\geq$ 10 seconds.

(2) SI Train B

(3) Undervoltage or degraded voltage on H bus. ← From "C" RSST

(4) Breakers for charging pump "A" and "C" normal open.

c. Charging pump C normal (15H6)

(1) Low charging header pressure - $\leq$ 1176 psig for $\geq$ 10 seconds.

(2) SI Train A or B

(3) Undervoltage or degraded voltage on J bus

Previously running

(4) Breakers for charging pump "A" and "B" open.

d. Charging pump C alternate (15J2) - There are no auto start signals for C charging pump when it is powered from the J Bus.

from the auxiliary shutdown panel.

b. Auto starts, stops, and interlocks are the same regardless of control location.

Distribute AIA-5.1, Charging Pumps Starts and Trips.

5. Manual starts

   a. No auto trip locked in

   b. Breaker racked in.

   c. Place control switch to start.

6. Automatic starts

Refer trainees to H/T-5.3, Charging Pumps Starts

   a. Charging pump A (15H5)

      (1) Low charging header pressure - $\leq$ 1176 psig for $\geq$ 10 seconds.

      (2) SI Train A

      (3) Undervoltage or degraded voltage on J bus:

         (a) 75% voltage for 2 seconds

         (b) 90% voltage for 60 seconds

QUESTION 47: (1.0)

Given the following plant conditions:

- A spurious safety injection occurred with Unit 1 at 100% power.
- SI has been terminated, and charging and letdown are in service.
- RCS pressure is stable at 2235 psig, solid plant pressure control.
- PRZR water temperature is 590°F.

Using the steam tables provided, determine the pressure at which the RCS would stabilize if a PRZR bubble were drawn at the current plant conditions.

a. 1417 psig.

b. 1432 psig.

c. 1438 psig.

d. 1453 psig.

ANSWER: a

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: at 590°F in PRZR, drawing a bubble would result in RCS pressure decreasing to P_{SAT} for 590°F; using steam tables (extrapolating between 588°F and 592°F) = 1431.65 psia – 14.696 = 1416.95 psig	Distractors plausible: b – failure to convert to psig; c – failure to extrapolate (use value for 592°F); d – failure to extrapolate and failure to convert to psig.	Distractors incorrect: b – per "answer correct" description, there is only one correct answer.
K/A: SYS010-K5.01	Objective: 1403	Source: New
Reference: Steam tables, ND-83-LP-5	Level: Comprehension	

Table 1. Saturated Steam: Temperature Table—*Continued*

Temp Fahr t	Abs Press. Lb per Sq In. p	Specific Volume Sat. Liquid v_f	Specific Volume Evap v_{fg}	Specific Volume Sat. Vapor v_g	Enthalpy Sat. Liquid h_f	Enthalpy Evap h_{fg}	Enthalpy Sat Vapor h_g	Entropy Sat. Liquid s_f	Entropy Evap s_{fg}	Entropy Sat. Vapor s_g	Temp Fahr t
460.0	466.87	0.01961	0.97463	0.99424	441.5	763.2	1204.8	0.6405	0.8299	1.4704	460.0
464.0	485.56	0.01969	0.93588	0.95557	446.1	758.6	1204.7	0.6454	0.8213	1.4667	464.0
468.0	504.83	0.01976	0.89885	0.91862	450.7	754.0	1204.6	0.6502	0.8127	1.4629	468.0
472.0	524.67	0.01984	0.86345	0.88329	455.2	749.3	1204.5	0.6551	0.8042	1.4592	472.0
476.0	545.11	0.01992	0.82958	0.84950	459.9	744.5	1204.3	0.6599	0.7956	1.4555	476.0
480.0	566.15	0.02000	0.79716	0.81717	464.5	739.6	1204.1	0.6648	0.7871	1.4518	480.0
484.0	587.81	0.02009	0.76613	0.78622	469.1	734.7	1203.8	0.6696	0.7785	1.4481	484.0
488.0	610.10	0.02017	0.73641	0.75658	473.8	729.7	1203.5	0.6745	0.7700	1.4444	488.0
492.0	633.03	0.02026	0.70794	0.72820	478.5	724.6	1203.1	0.6793	0.7614	1.4407	492.0
496.0	656.61	0.02034	0.68065	0.70100	483.2	719.5	1202.7	0.6842	0.7528	1.4370	496.0
500.0	680.86	0.02043	0.65448	0.67492	487.9	714.3	1202.2	0.6890	0.7443	1.4333	500.0
504.0	705.78	0.02053	0.62938	0.64991	492.7	709.0	1201.7	0.6939	0.7357	1.4296	504.0
508.0	731.40	0.02062	0.60530	0.62592	497.5	703.7	1201.1	0.6987	0.7271	1.4258	508.0
512.0	757.72	0.02072	0.58218	0.60289	502.3	698.2	1200.5	0.7036	0.7185	1.4221	512.0
516.0	784.76	0.02081	0.55997	0.58079	507.1	692.7	1199.8	0.7085	0.7099	1.4183	516.0
520.0	812.53	0.02091	0.53864	0.55956	512.0	687.0	1199.0	0.7133	0.7013	1.4146	520.0
524.0	841.04	0.02102	0.51814	0.53916	516.9	681.3	1198.2	0.7182	0.6926	1.4108	524.0
528.0	870.31	0.02112	0.49843	0.51955	521.8	675.5	1197.3	0.7231	0.6839	1.4070	528.0
532.0	900.34	0.02123	0.47947	0.50070	526.8	669.6	1196.4	0.7280	0.6752	1.4032	532.0
536.0	931.17	0.02134	0.46123	0.48257	531.7	663.6	1195.4	0.7329	0.6665	1.3993	536.0
540.0	962.79	0.02146	0.44367	0.46513	536.8	657.5	1194.3	0.7378	0.6577	1.3954	540.0
544.0	995.22	0.02157	0.42677	0.44834	541.8	651.3	1193.1	0.7427	0.6489	1.3915	544.0
548.0	1028.49	0.02169	0.41048	0.43217	546.9	645.0	1191.9	0.7476	0.6400	1.3876	548.0
552.0	1062.59	0.02182	0.39479	0.41660	552.0	638.5	1190.6	0.7525	0.6311	1.3837	552.0
556.0	1097.55	0.02194	0.37966	0.40160	557.2	632.0	1189.2	0.7575	0.6222	1.3797	556.0
560.0	1133.38	0.02207	0.36507	0.38714	562.4	625.3	1187.7	0.7625	0.6132	1.3757	560.0
564.0	1170.10	0.02221	0.35099	0.37320	567.6	618.5	1186.1	0.7674	0.6041	1.3716	564.0
568.0	1207.72	0.02235	0.33741	0.35975	572.9	611.5	1184.5	0.7725	0.5950	1.3675	568.0
572.0	1246.26	0.02249	0.32429	0.34678	578.3	604.5	1182.7	0.7775	0.5859	1.3634	572.0
576.0	1285.74	0.02264	0.31162	0.33426	583.7	597.2	1180.9	0.7825	0.5766	1.3592	576.0
580.0	1326.17	0.02279	0.29937	0.32216	589.1	589.9	1179.0	0.7876	0.5673	1.3550	580.0
584.0	1367.7	0.02295	0.28753	0.31048	594.6	582.4	1176.9	0.7927	0.5580	1.3507	584.0
588.0	1410.0	0.02311	0.27608	0.29919	600.1	574.7	1174.8	0.7978	0.5485	1.3464	588.0
592.0	1453.3	0.02328	0.26499	0.28827	605.7	566.8	1172.6	0.8030	0.5390	1.3420	592.0
596.0	1497.8	0.02345	0.25425	0.27770	611.4	558.8	1170.2	0.8082	0.5293	1.3375	596.0
600.0	1543.2	0.02364	0.24384	0.26747	617.1	550.6	1167.7	0.8134	0.5196	1.3330	600.0
604.0	1589.7	0.02382	0.23374	0.25757	622.9	542.2	1165.1	0.8187	0.5097	1.3284	604.0
608.0	1637.3	0.02402	0.22394	0.24796	628.8	533.6	1162.4	0.8240	0.4997	1.3238	608.0
612.0	1686.1	0.02422	0.21442	0.23865	634.8	524.7	1159.5	0.8294	0.4896	1.3190	612.0
616.6	1735.9	0.02444	0.20516	0.22960	640.8	515.6	1156.4	0.8348	0.4794	1.3141	616.0
620.0	1786.9	0.02466	0.19615	0.22081	646.9	506.3	1153.2	0.8403	0.4689	1.3092	620.0
624.0	1839.0	0.02489	0.18737	0.21226	653.1	496.6	1149.8	0.8458	0.4583	1.3041	624.0
628.0	1892.4	0.02514	0.17880	0.20394	659.5	486.7	1146.1	0.8514	0.4474	1.2988	628.0
632.0	1947.0	0.02539	0.17044	0.19583	665.9	476.4	1142.2	0.8571	0.4364	1.2934	632.0
636.0	2002.8	0.02566	0.16226	0.18792	672.4	465.7	1138.1	0.8628	0.4251	1.2879	636.0
640.0	2059.9	0.02595	0.15427	0.18021	679.1	454.6	1133.7	0.8686	0.4134	1.2821	640.0
644.0	2118.3	0.02625	0.14644	0.17269	685.9	443.1	1129.0	0.8746	0.4015	1.2761	644.0
648.0	2178.1	0.02657	0.13876	0.16534	692.9	431.1	1124.0	0.8806	0.3893	1.2699	648.0
652.0	2239.2	0.02691	0.13124	0.15816	700.0	418.7	1118.7	0.8868	0.3767	1.2634	652.0
656.0	2301.7	0.02728	0.12387	0.15115	707.4	405.7	1113.1	0.8931	0.3637	1.2567	656.0
660.0	2365.7	0.02768	0.11663	0.14431	714.9	392.1	1107.0	0.8995	0.3502	1.2498	660.0
664.0	2431.1	0.02811	0.10947	0.13757	722.9	377.7	1100.6	0.9064	0.3361	1.2425	664.0
668.0	2498.1	0.02858	0.10229	0.13087	731.5	362.1	1093.5	0.9137	0.3210	1.2347	668.0
672.0	2566.6	0.02911	0.09514	0.12424	740.2	345.7	1085.9	0.9212	0.3054	1.2266	672.0
676.0	2636.8	0.02970	0.08799	0.11769	749.2	328.5	1077.6	0.9287	0.2892	1.2179	676.0
680.0	2708.6	0.03037	0.08080	0.11117	758.5	310.1	1068.5	0.9365	0.2720	1.2086	680.0
684.0	2782.1	0.03114	0.07349	0.10463	768.2	290.2	1058.4	0.9447	0.2537	1.1984	684.0
688.0	2857.4	0.03204	0.06595	0.09799	778.8	268.2	1047.0	0.9535	0.2337	1.1872	688.0
692.0	2934.5	0.03313	0.05797	0.09110	790.5	243.1	1033.6	0.9634	0.2110	1.1744	692.0
696.0	3013.4	0.03455	0.04916	0.08371	804.4	212.8	1017.2	0.9749	0.1841	1.1591	696.0
700.0	3094.3	0.03662	0.03857	0.07519	822.4	172.7	995.2	0.9901	0.1490	1.1390	700.0
702.0	3135.5	0.03824	0.03173	0.06997	835.0	144.7	979.7	1.0006	0.1246	1.1252	702.0
704.0	3177.2	0.04108	0.02192	0.06300	854.2	102.0	956.2	1.0169	0.0876	1.1046	704.0
705.0	3198.3	0.04427	0.01304	0.05730	873.0	61.4	934.4	1.0329	0.0527	1.0856	705.0
705.47*	3208.2	0.05078	0.00000	0.05078	906.0	0.0	906.0	1.0612	0.0000	1.0612	705.47*

$588 = 1410 - 14.7$

$592 = 1453.3 - 14.7$

$588 = 1395$

$592 = 1439$

$$588 + \frac{592-588}{2} = 590$$

$$1395 + \frac{1439-1395}{2} = 1417$$

QUESTION 48: (1.0)

During the initial RCP start following RCS fill and vent, you are assigned to operate reverse acting, letdown pressure controller, 1-CH-PCV-1145 in manual.

After the RCP is started, you notice RCS pressure decreasing. To restore pressure, you will __________.

a. **decrease** the demand on the controller to close CH-PCV-1145

b. **increase** the demand on the controller to open CH-PCV-1145

c. **decrease** the demand on the controller to open CH-PCV-1145

d. **increase** the demand on the controller to close CH-PCV-1145

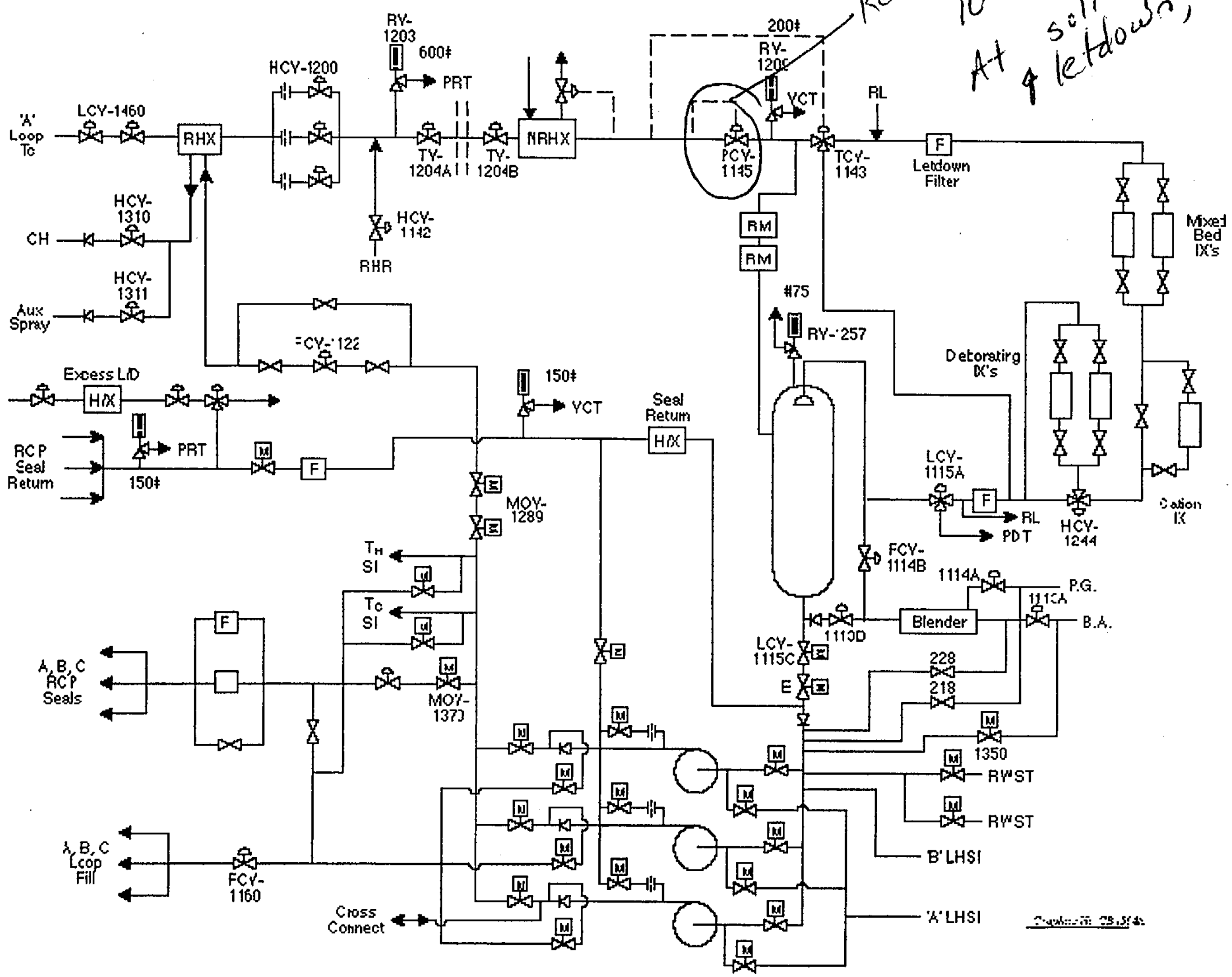
ANSWER: d

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: 100 demand will close the valve, limiting letdown, increasing RCS pressure.	Distractors plausible: All-variations of misconceptions of reverse acting and effects of manipulations of 1-CH-PCV-1145	Distractors incorrect: a. Incorrect valve direction. b. Would cause RCS pressure to decrease. c. Incorrect valve direction.
K/A: SYS011-A4.05	Objective: 1792	Source: New
Reference: ND-88.3-LP-2	Level: Knowledge	

ND-88.3-H/T-2.2

Reverse acting
100% demand = open
At solid plant ↑ letdown, ↓ RCS pressure



CHARGING AND LETDOWN SYSTEM DRAWING

QUESTION 49: (1.0)

With Unit 1 at 5% power, a loss of "A" DC Bus occurs.

Without operator intervention, which ONE of the following states how the Reactor Protection System will respond?

a. Both reactor trip breakers remain closed.

b. "A" reactor trip breaker remains closed, "B" reactor trip breaker opens.

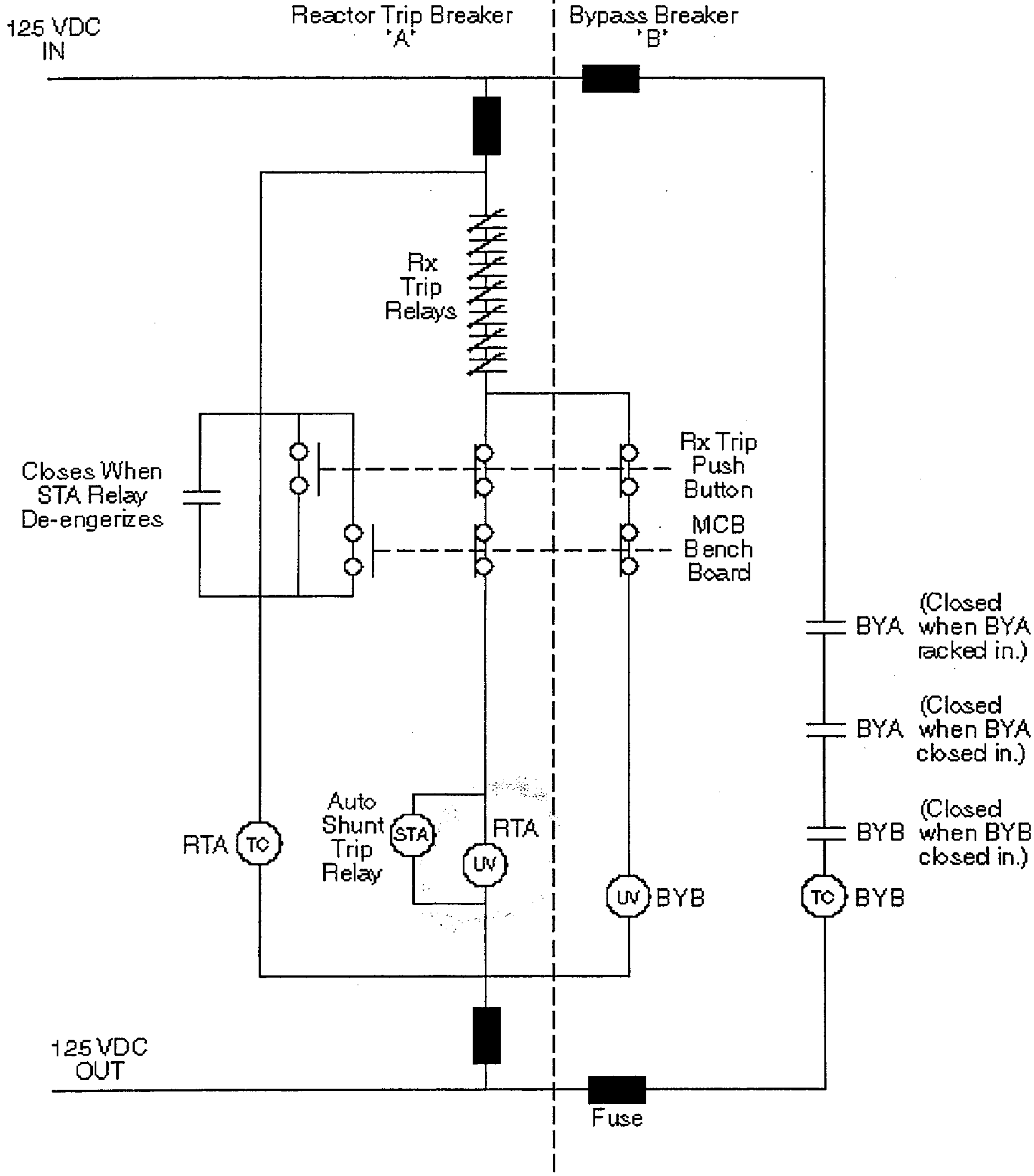
c. "B" reactor trip breaker remains closed, "A" reactor trip breaker opens.

d. Both reactor trip breakers open.

ANSWER: c

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: 'A' DC bus supplies power to the "A" Rx Trip breaker (opening it). No other reactor trip coincidence will be exceeded.	Distractors plausible: all – candidate misconception of other reactor trip coincidence being exceeded i.e. Attempt to tie cross talk via the turbine (turbine trip will not cause reactor trip < P-7, 10% power)	Distractors incorrect: all – train "B" is available to actuate, but has no trip signals.
K/A: 012-K2.01	Objective: 2587	Source: New
Reference: ND-93.3-LP-10	Level: Comprehension	



NOTE: Trip Coils (TC) energize to trip breaker.
Under Voltage Coils (UV) de-energize to trip breaker.
Auto Shunt Trip Realy (STA) de-energizes to actuate.

Rx TRIP BREAKER A AND BYPASS BREAKER TRIP LOGICS

QUESTION 50: (1.0)

The following conditions exist:

- A Small Break LOCA has occurred.
- ES-1.2, Post LOCA Cooldown and Depressurization is in progress.

Which ONE of the following conditions prevents RHR from being placed in service?

a. All RCS hot leg temperatures are 337°F.

b. RCS pressure is 475 psig.

c. 1-RC-PT-1403 is failed low.

d. Pressurizer level less than 22%.

ANSWER: b

[RO: Tier 1/Group 2]
[SRO: Tier 1/Group 2]

Answer correct: Required permissive pressure to place RHR in service is less than 460 psig.	Distractors plausible: a. There is a maximum temperature for RHR operation; c.This PT provides permissive input to the RHR inlet MOVs; d. To place RHR in service, temperature, pressure, and level must be considered, considering a LOCA is in progress, inventory is a concern.	Distractors incorrect: a. The maximum temperature for RHR operation is 350°F. c. Transmitter is failing in the wrong direction to preclude RHR being placed in service. d. No restrictions exist in the procedures for level.
K/A: EPE.E03.EK2.2	Objective: 3062	Source: NEW
Reference: ES-1.2	Level: Knowledge	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*32.	__CHECK IF RHR SYSTEM CAN BE PLACED IN SERVICE:	
	a) Consult with TSC to determine if RHR should be warmed	
	b) Check the following.	b) GO TO Step 33.
	• RCS hot leg temperature - LESS THAN 350°F	
	• RCS pressure - LESS THAN 450 PSIG [325 PSIG]	
	c) Consult with TSC to determine if RHR should be placed in service	
33.	__CHECK IF OVERPRESSURE MITIGATION SYSTEM CAN BE PLACED IN SERVICE:	
	a) Check RCS pressure - LESS THAN 365 PSIG.	a) GO TO Step 34. <u>WHEN</u> RCS pressure is less than 365 psig, <u>THEN</u> do Steps 33b and 33c.
	• PI-1-403 (NQ)	
	b) Check PRZR PORV block valves - OPEN	b) Open valves.
	c) Put both Overpressure Mitigation system key switches in - ENABLE (keys 53 and 54)	

QUESTION 51: (1.0)

Given the following plant conditions:

- Unit 1 is at 100% power.
- All plant equipment is operable.
- An I&C tech inadvertently isolates and vents "A" condenser pressure transmitter 1-CN-PT-101A.

Which ONE of the following is correct concerning the affects of this?

a. Trip signal to main turbine, no affect on condenser steam dumps or "A" condenser pressure indication.

b. Trip signal to main turbine and loss of "A" condenser pressure indication, no affect on condenser steam dump capability.

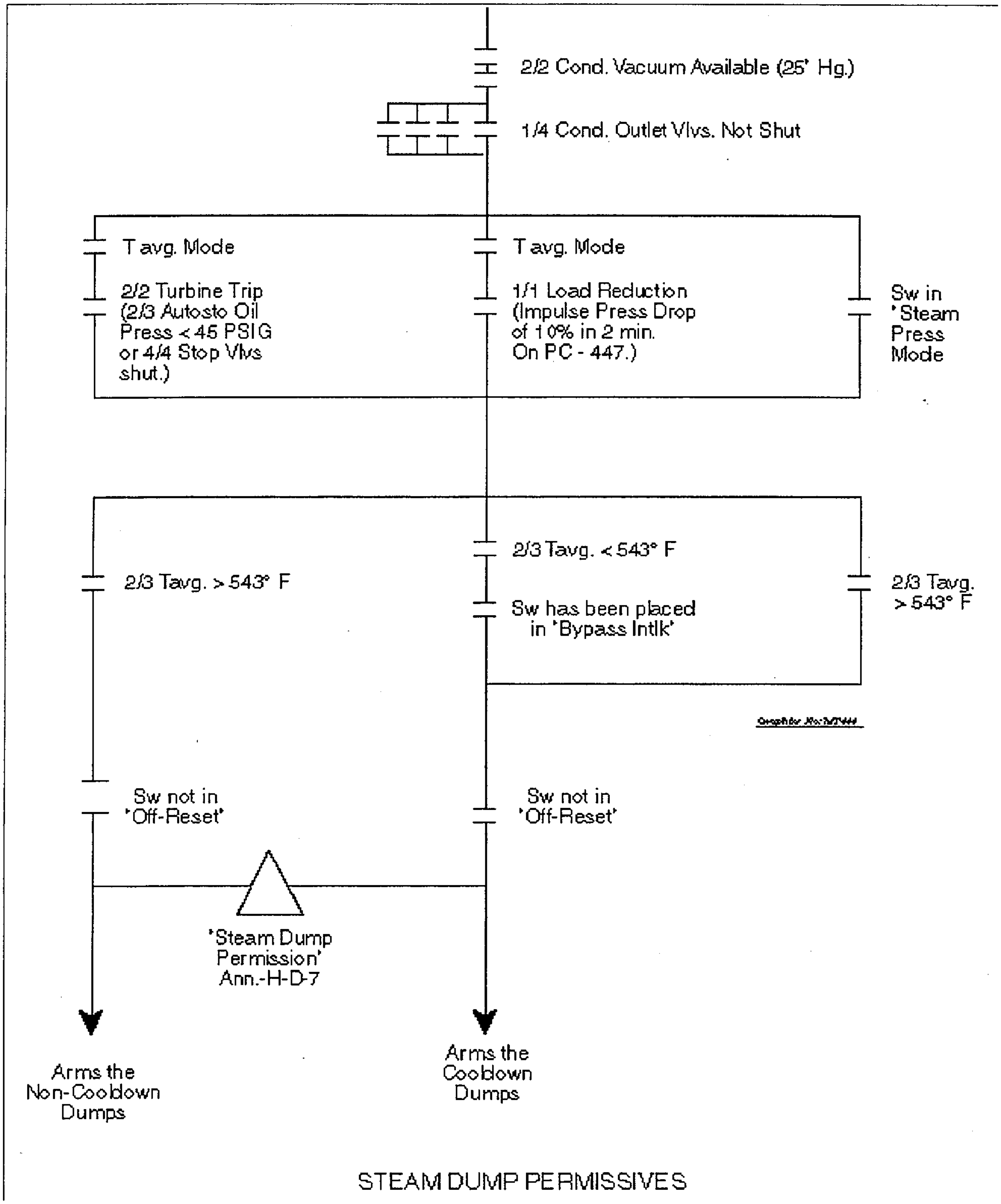
c. Loss of condenser steam dump capability and loss of "A" condenser pressure indication, no affect on main turbine.

d. Loss of condenser steam dump capability, trip signal to main turbine, and loss of "A" condenser pressure indication.

ANSWER: c

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: Turbine trip is mechanical (on the trip block). Steam dump capability is affected due to permissive not met (adequate heat sink). Indication is also lost.	Distractors plausible: all – candidate misconception concerning condenser pressure instruments' outputs, specifically the fact that low vacuum is a turbine trip.	Distractors incorrect: "A" condenser pressure transmitter supplies a signal to condenser steam dumps permissive, does not supply turbine trip coincidence.
K/A: SYS016-A2.03	Objective: 2672	Source: New
Reference: ND-93.3-LP-9	Level: Knowledge	



STEAM DUMP PERMISSIVES

QUESTION 52: (1.0)

Unit 1 is at 100% power with no equipment out of service and all three containment air recirculation fans (CARFs) running.

If a spurious Hi Hi CLS occurs, which ONE of the following identifies the **immediate** affects on the CARFs?

a. All three CARFs trip.

b. "A" and "B" trip, "C" continues to run.

c. "B" trips, "A" and "C" continue to run.

d. "B" and "C" trip, "A" continues to run.

ANSWER: b

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 1]

Answer correct: "A" fan is powered from "H" bus;"C" fan is powered from "J" bus.; train "A" HI HI CLS trips "A" fan, train "B" HI HI CLS trips "B" fan, but has no effect on "C" fans (powered from station service).	Distractors plausible: Misconception of which Fans trip on a loss of power.	Distractors incorrect: all – only the "A" and "B" fans trip.
K/A: SYS026-K3.01	Objective: 2323	Source: New
Reference: ND-91-LP-5	Level: Knowledge	

Upon a HI-HI CLS initiation, either automatically by 3/4 channels >23.0 psia or manually by simultaneously pressing both CLS pushbuttons, the following functions occur:

a. Starts:

(1) Containment Spray Pumps

(2) Inside Recirc Spray Pumps (2 min T.D.)

(3) Outside Recirc Spray Pumps (5 min T.D.)

(4) #1 Emergency Diesel Generator

(5) #3 Emergency Diesel Generator (also sends trip signal to other unit's #3 EDG output bkr)

(6) Recirc spray Hx SW rad mon sample pumps

b. Trips A and B containment Air Recirculation Fans

c. The GDC-17 Auto-Start Inhibit Circuit is activated such that it blocks auto-start of large non-class 1E pumps for 315 seconds.

d. Phase III isolation closes:

(1) RCP motor CC

(2) RCP thermal barrier CC

QUESTION 53: (1.0)

Given the following plant conditions:

- Unit is on-line at 100% power.
- All control systems are in AUTOMATIC.
- "C" S/G level channel III fails low.

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

a. Feed Flow < Steam Flow Reactor Trip occurs.

b. No affect on "C" S/G Level control.

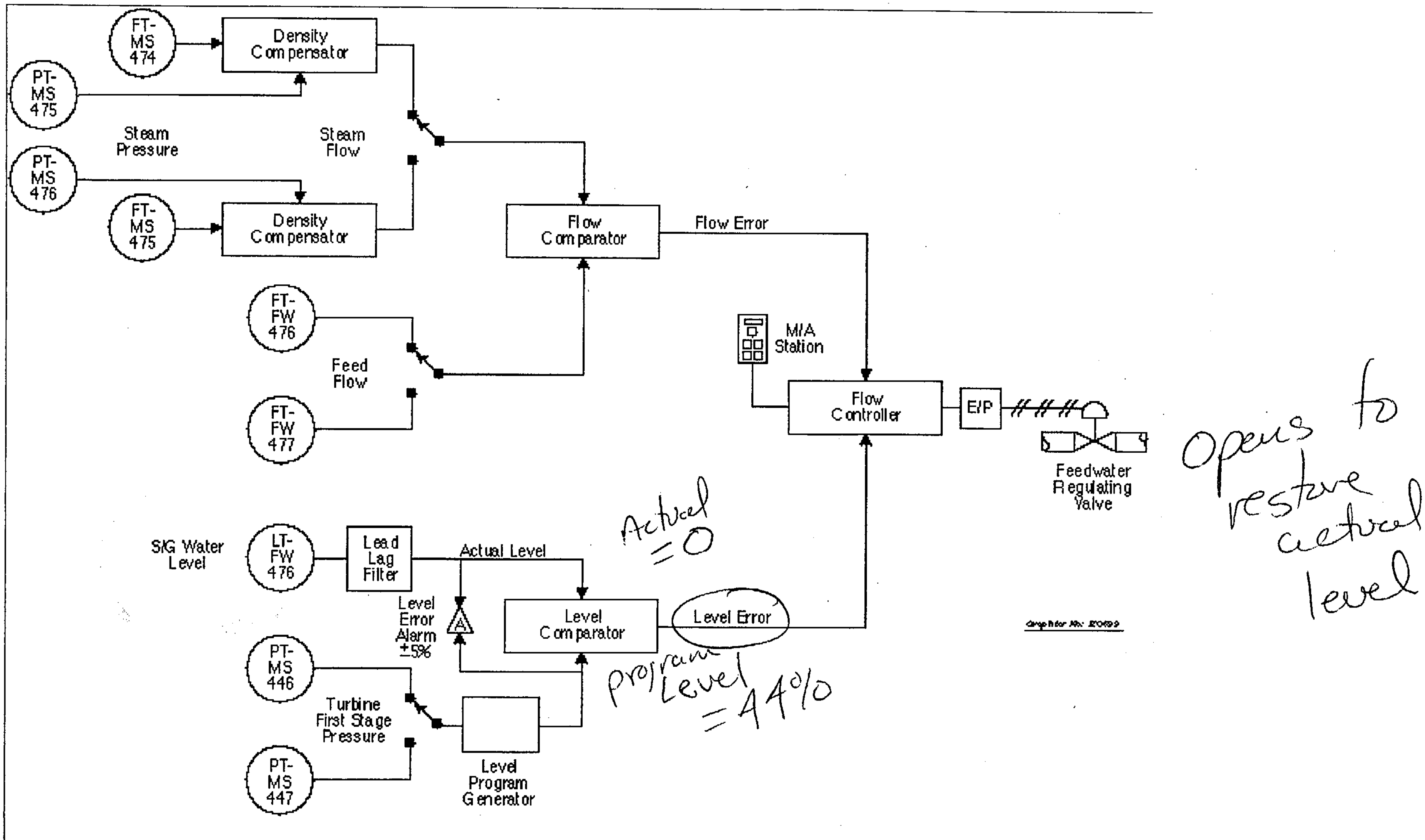
c. "C" FRV valve closes and "C" S/G level decreases.

d. "C" FRV opens and "C" S/G level increases.

ANSWER: d

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: Channel 3 is the sole input to SGWLC of actual level. The FRV will open in response to the percieved low level condition, raising actual level.	Distractors plausible: a & b – SF and FF channels are selectable for input to SGWLC; d – candidate misconception concerning the sequence of events and equipment alignment	Distractors incorrect: a- feed flow is > steam flow b – S/G level WILL change; feedflow increases.
K/A: SYS035-K6.03	Objective: 2669	Source: NEW
Reference: ND-93.3-LP-8	Level: Comprehension	



SGWLC DIAGRAM

QUESTION 54: (1.0)

Given the following plant conditions:

- Unit 1 generator output breaker has just been closed following a reactor startup.
- All control systems are in AUTOMATIC and aligned per startup procedures.
- Steam Dump Valve 1-MS-TCV-105A is de-energized and tagged out for repair.

If main steam header pressure transmitter PT-464 fails high, which ONE of the following is correct?

a. All steam dump valves will remain in their present position.

b. All steam dumps (except 1-MS-TCV-105) fully open and RCS cooldown stops at 543°F.

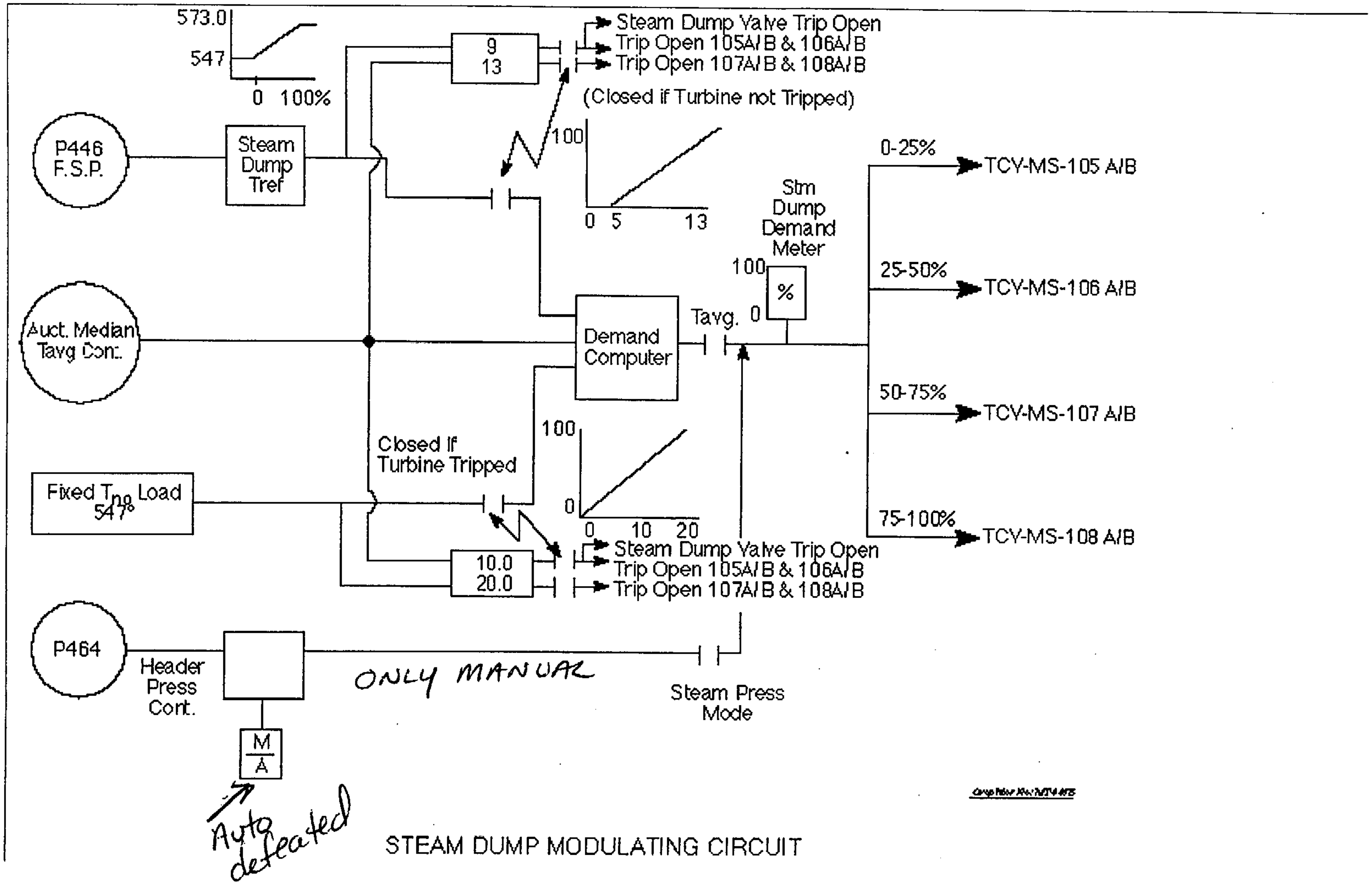
c. All steam dumps (except 1-MS-TCV-105) fully open and uncontrolled RCS cooldown condition continues until MSTVs are manually closed.

d. All steam dumps (except 1-MS-TCV-105) fully open and safety injection occurs when RCS T_{ave} decreases to <543°F.

ANSWER: a

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: No steam dumps open because the Automatic feature of this controller has been defeated.	Distractors plausible: all – candidate misconception concerning automatic operation of steam dumps in steam pressure mode.	Distractors incorrect: b/c/d the automatic feature of the header pressure controller has been defeated.
K/A: SYS039-A2.04	Objective: 2673	Source: New
Reference: ND-93.3-LP-9	Level: Comprehension	



STEAM DUMP MODULATING CIRCUIT

QUESTION 55: (1.0)

Given the following plant conditions:

- #1 EDG is carrying 1H emergency bus.
- The team is about to transfer 1H bus back to its normal source.
- The diesel operator places the 15H8 synchronizing switch to ON.
- The synchroscope begins rotating very fast in the slow direction (counterclockwise).

The diesel operator will have to ________ #1 EDG speed until the synchroscope is rotating slowly in the __________ direction.

a. decrease; slow

b. decrease; fast

c. increase; slow

d. increase; fast

ANSWER: d

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: EDG speed droop is NOT in service when the EDG is carrying the bus by itself; when preparing to parallel the EDG, speed droop is placed in service when the associated synch switch is turned on; this causes the EDG speed to slow and the synch scope rotates fast in the slow direction; EDG speed must be increased such that it is slightly "faster" than the source it is being paralleled to.	Distractors plausible: all – candidate misconception regarding operation of the EDG speed droop, speed control and synch circuit.	Distractors incorrect: a & b – EDG speed must be increased; c – synch scope must be rotating slowly in the fast direction.
K/A: SYS062-A4.03	Objective: 1165	Source: New
Reference: ND-80-LP-12	Level: Comprehension	

C. Paralleling AC Sources

Remember that 3 phase AC consists of 3 sinusoidal waveforms, each "rotating" in a set sequence. Suppose then, that you wanted to connect one 3 phase AC supply to another. How would you ensure that they will match up? Paralleling is the name of the technique for connecting AC sources.

There are four (4) precautions or physical checks that are performed for paralleling. These checks ensure that the AC voltages are matched in amplitude, frequency and phase before they are connected.

1. A synchroscope is used to aid the operator in paralleling AC sources. It consists of a synchroscope, lights, and 2 voltmeters; one for running voltage and the other for incoming voltage.

Refer to/display H/T-12.2, Synchroscope.

2. Running and incoming voltage.

   a. The running voltage is the voltage of the grid or the bus that the generator is to be synchronized to.

   b. The incoming voltage is the output voltage of the main generator or the emergency diesel generator.

   c. It is very important that the voltages be matched.

      (1) Even though the indicated voltages are approximately 120v, remember that they are stepped down from some higher voltage.

(2) A difference of 1 volt on the synchroscope meters can be a difference of 2000v (230kv) or 5000v (500kv) across the breakers in the switchyard. When the breakers are closed, this can result in a short term current draw of 200,000 or 500,000 amps.

(3) Although this current draw is extremely short term (and the breakers are designed to quench the arc), it does degrade the contacts, especially repeated high current draws over time.

3. The synchroscope itself is actually a type of motor called a synchro. It senses a difference in potential between the incoming and running voltage supplies due to the difference in frequencies. The speed of the rotation is proportional to the magnitude of the difference in potential, the direction is dependent upon which is at a higher potential, running or incoming.

a. When the incoming frequency is higher than the running frequency, the synchroscope spins in the "fast" direction.

b. When the incoming frequency is lower than the running frequency, the synchroscope spins in the "slow" direction.

c. The physical location of the synchroscope needle shows how apart the peaks of the voltage waves are; i.e., whether or not they are in phase. For the two voltages to be in phase, the needle must be at 12 o'clock.

Refer to/display H/T-12.3, Synchroscope Relationship to Voltage Phases. Point out to trainees how 12 o'clock on the synchroscope corresponds to voltages being in phase.

d. The SYNCH LIGHTS provide an additional indication of 12 o'clock. As the synchroscope spins, the lights go from dim to bright and back to dim.

(1) When the synchroscope is at 12 o'clock, the bulb is dim.

(2) At 6 o'clock, or maximum phase difference, the bulb is brightest. It is hard to tell when the bulb is at its brightest, but easy to tell when the filament is out, so the lights are human factored.

4. One parameter not monitored by a synchroscope is the phase rotation sequence. Remember that in 3 phase, the voltages peak in the sequence A-B-C. If one side (running) is in the sequence A-B-C, and the other side (incoming) is in the sequence A-C-B; this condition will <u>not</u> be detected by the synchroscope.

a. It is possible to check phase rotation with a "phase rotation meter." However, for breakers like the generator output breakers, phase rotation is verified during initial installation. The operators must accept that phase rotation is correct.

b. Operators may recall that for motors, they are "jogged" to ensure proper phase rotation. The motor is "bumped" to ensure proper rotation.

5. Using the synchroscope

a. Most breakers are interlocked to that the synchroscope must be turned on to allow remote operation of the breaker.

b. To parallel a breaker, first turn on the synchroscope.

c. Raise the speed of the incoming machine so that the synchroscope is spinning slowly in the fast direction. This ensures that the generator will pick up load when its output breaker is closed. This will be explained next in this lesson.

d. Match voltages on the synchroscope voltmeters by adjusting the voltage regulator of the incoming generator.

e. To ensure that the voltages are in phase, the output breaker should be closed with the synchroscope needle at 12 o'clock to obtain this, the operator needs to turn the switch at 11 o'clock to allow for the delay time in the closing circuitry.

f. When the breaker has been closed, verify by observing the red light on the breaker switch and checking that the generator has picked up load.

6. Consequences of Improper Paralleling

a. At one utility (Trojan) in late 1990, a 230 kv breaker was paralleled 150 degrees out of phase. The generator output breaker is supposed to close in 0.27 seconds. Due to a broken air pilot solenoid valve, the breaker closed 8 seconds later than it was supposed to.

b. The transient caused by the out of phase synchronization caused a 140 MWe excursion on the grid, tripped a nearby substation, and caused turbine building vibrations that could be felt in the control room.

c. At Surry, before the Main Generator output breakers are closed, a switching order is issued to open the breaker disconnects, and cycle the output breakers. This allows the operators to check the closure time before actual synchronization.

d. Another possible problem with incorrect paralleling is generator motor. If the output breakers are closed and the generator is not going fast enough to pick

QUESTION 56: (1.0)

Station battery 1-A capacity is 1800 amp-hours at a given rate of discharge.

If the discharge rate __________, the battery capacity __________.

a. increases; decreases

b. increases; increases

c. increases; remains constant

d. decreases; remains constant

ANSWER: a

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 1]

Answer correct: battery capacity decreases as discharge rate increases.	Distractors plausible: all – candidate misconception concerning battery capacity and discharge rates.	Distractors incorrect: b & c – battery capacity decreases as discharge rate increases; d – battery capacity increases as discharge rate decreases.
K/A: 063-A1.01	Objective: 1171	Source: New
Reference: ND-80-LP-13	Level: Knowledge	

B. Battery Charging

1. Most alkaline batteries used in household appliances are discarded after the battery dies. When the battery is dead, the chemical process has stopped. Surry uses rechargeable lead-acid batteries.

2. A battery charge is a reversal of the chemical reaction that produces electricity.

   a. A voltage is placed across the cell, higher than its nominal terminal voltage. This reverses the chemical reaction and drives the lead and lead oxide back onto the cathode and anode.

   b. Charges are done periodically to ensure cell capacity. Battery capacity is rated in ampere-hours, which means that the battery will deliver so many amps for a rated period of time.

> Tell trainees:
>
> For example, if a battery is rated at 2000 ampere-hours, then, in theory, it will provide:
>
> 2000 amps for 1 hour
> 1000 amps for 2 hours
> 500 amps for 4 hours, and so on.
>
> In reality, the battery capacity reacts close, but not exactly according to this theory (the higher the discharge rate, the less time it takes for the battery to reach its minimum capacity).

   c. The batteries have a capacity of 1800 amp-hours. This is sufficient to provide a minimum of two (2) hours of operation of Vital DC loads.

3. Types of charges - two types of battery charges: float and equalizers.

QUESTION 57: (1.0)

Given the following plant conditions:

- #1 and #3 EDG's are loaded to 2750 KW each.
- The total amount of fuel in the underground fuel oil storage tanks is at the minimum required by Technical Specifications.
- Due to equipment failures, the aboveground fuel oil storage tank is not available.

#1 and #3 EDG's will run out of fuel oil in approximately __________.

a. one hour

b. 3.5 days

c. four hours

d. 7 days

ANSWER: b

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: per Technical Specifications, tank has a minimum capacity to enable a fully loaded diesel to run for 7 days.	Distractors plausible: a – time that an EDG can run on its base tank; c – this the time the AAC diesel can run fully loaded on its day tank; d – time one diesel can run fully loaded on 35,000 gallons.	Distractors incorrect: per UFSAR, day tank has a minimum capacity to enable the diesel to run for one hour.
K/A: 064-K6.08	Objective: 2237	Source: New
Reference: ND-90.3-LP-1	Level: Knowledge	

(2) If the EDG is inoperable for any cause other than preplanned maintenance or testing, the operability of the other EDG must be demonstrated daily. For the purposes of this test, the EDG need be run for 2 hours, provided 2 AC offsite sources have been verified.

(3) If the EDG is not returned to service in 7 days, place the Unit in a 6 hour clock to HSD, and a 30 hour clock to CSD.

b. One EDG fuel oil flowpath may be inoperable up to 24 hours provided the other flowpath is verified operable. If 24 hours elapse and the flowpath is not restored, the EDG is inoperable. If the EDG battery, battery charger, or DC circuits are inoperable, the EDG is inoperable.

3. If the primary offsite power source is not operable, the Unit can be operated for 7 days provided backfeed is available. IF this requirement cannot be met, the Unit must be placed in CSD.

4. A train of the opposite Units emergency power may be inoperable for a period of 14 days. During this period, the following conditions apply:

a. If the offsite power source becomes unable to energize the opposite Unit's operable train, operation may continue if the EDG is carrying the emergency bus.

b. If the opposite Unit's operable train EDG becomes inoperable, operation may continue for 72 hours provided offsite power is energizing the bus.

c. If the inoperable train is returned to service, may revert to the 14 day clock.

5. Continuous load on EDG is limited to 2750kW.

6. The EDG has a continuous rating of 2750kW. 35,000 gallons of fuel in the below ground tanks allows for one EDG to operate at full load for 7 days.

Distribute SOER 83-01, Diesel Generator Failures, and discuss with trainees causes and corrective actions taken. Distribute LER 91-004, Two of Three Emergency Diesel Generators Inoperable, and discuss event with the class.

Summary

The EDGs are the single most important safety related components in the plant. If offsite power is lost, the EDGs are designed to rapidly start and re-energize the emergency buses to provide power to all safety related equipment. If the EDGs fail to start or load, the response to the loss of offsite power becomes much more complex and taxing for the operator. For this reason the operator's knowledge of EDG operation, Annunciator Response procedures, and Abnormal procedures can directly aid or inhibit plant response to this event.

Use objectives to review and clarify material as needed.

QUESTION 58: (1.0)

The Reactor was tripped from 100% power in response to increased steam generator tube leakage.

The post trip **indication** of primary-to-secondary leakage on the NRC radiation monitors will tend to be __________.

a. lower than at power indication

b. approximately equal to at power indication

c. higher than at power indication

d. so inaccurate as to be unusable

ANSWER: a

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: per	Distractors plausible: b – misconception about where the activity is coming from; therefore; thinking that it is the same; c – increased steam pressure could increase the actual leak rate. D - Because the actual leak rate is lower than at power but could be used for trending.	Distractors incorrect: b – Activity is from N-16 which is not being produced once we are shurdown. C leakage increases but N-16 production ceases d – accuracy of the NRC monitor is good even at lower ranges.
K/A: SYS073-K1.01	Objective: 1905	Source: New
Reference: ND-95.2-LP-6	Level: Knowledge	

b. Assuming the leak rate is < 50 GPM, the polishing building is bypassed to prevent the spread of contamination.

c. Chemistry and HP are informed to initiate leak rate calculation procedures. These will provide an accurate primary to secondary leak rate, which is especially useful for leak rates measured in the gallons per day range.

d. Steps 5 and 6 are performed to identify the S/G with the tube leak. During power operation, N^{16} traveling down the steam lines clearly will identify the leaking S/G. If the ERFCS computer is not operable, local readings on the NRC monitors can be used.

e. Blowdown radiation monitors also are used, however, since one monitor will be lined up to sample 2 S/G blowdowns and the long runs of piping involved may delay identification from this process.

f. Secondary contamination is minimized by ensuring blowdown is processed by the IXs and by shifting aux steam supply to the other unit or by starting up the auxiliary boilers, if necessary. Additionally, the blowdown may be reduced or stopped. Other S/G chemistry parameters as well as contamination will factor in the determination. Turbine building sumps are sampled for contamination, and sump pumps are secured as necessary.

g. When the leaking steam generator is identified positively, the steam supply from that S/G is isolated to the SDAFW pump. This both limits the spread of contamination to the pump and prevents an uncontrolled radioactive release should the pump start from an automatic condition in conjunction with an increase in the leak rate.

h. RCS leak rate is checked to verify compliance with Tech Specs. S/G tube leakage is checked based on criteria that is more limiting than Tech Specs. The

QUESTION 59: (1.0)

With both units operation at 100% power the 2G transformer trips due to internal fault.

Which ONE of the following identifies the impact on unit operation?

a. 2G transformer must be restored or Unit 2 must be shutdown.

b. 2G bus automatically re-energizes and running CW pumps remain running, allowing continued dual unit operation.

c. 2G bus automatically re-energizes and Unit 2 CW pumps must be manually restarted to allow continued dual unit operation.

d. 2G bus must be manually crosstied and Unit 2 CW pumps restarted to allow continued dual unit operation.

ANSWER: c

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: The 1G to 2G breaker will automatically close but the unit 2 pumps must be restarted.	Distractors plausible: a. because it is unit 2 which has the loss of CW pumps b – the 2G bus automatically re-energizes but misunderstanding of CW pump breakers. d – Some equipment in the plant must be manually cross-tied.	Distractors incorrect: a – Depending on the water temperature both Units may be able to operate at full power on only one unit's CW pumps. B – CW pumps do not "ride the bus". D – The 2G bus will automatically cross tie.
K/A: SYS075-A2.02	Objective: 2213	Source: New
Reference: ND-90.2-LP-1	Level: Comprehension	

4. Bus 1G feeds 480 MCCs 1G1-1 southwest and MCCs 1G1-1 northeast.

5. MCC 1G1-1 northeast and southwest feed various 480 loads and MCCs at low level.

6. Each 4160v breaker on bus 1G and 2G has a 2-position selector switch. These positions select the location where the breaker is operated.

   a. In the local switch position, each breaker is opened or closed by actuation of a control switch on breaker cubicle.

   b. In the remote switch position, each breaker can only be operated from the Screenwell supervisory panel in the Main Control Room. Most breakers can be remotely opened or closed. The 15G1 feeder breaker and 15G8 tie breaker can only be opened.

   c. The bus tie breaker 15G8 has an automatic close-in feature.

      (1) When either bus 1G or 2G feeder breaker is opened, an undervoltage condition is sensed on its respective bus.

      (2) If three conditions are met, the 15G8 breaker will automatically close and re-energize the de-energized bus. These conditions are:

         (a) One bus feeder breaker open with the other bus feeder closed.

         (b) No electrical fault sensed on 1G or 2G bus.

         (c) No electrical fault trip of the 15G8 breaker.

7. 480v Screenwell Distribution

i. The CW pumps may be controlled from:

   (1) Control Room intake structure control panel (common to both units, located in unit one Control Room), or

   (2) Locally with switch on face of breaker.

   (3) Control location selected at low level intake with switch on face of each pump breaker.

j. The CW pump will automatically trip on any of the following:

   (1) Phase or ground overcurrent to 86 relay.

   (2) Bus undervoltage on G bus.

   (3) High high water level in circulating water pump motor pit (2/3 level switches on wall of pit).

k. The CW pump motor breaker position is interlocked with the following:

   (1) Intake vacuum priming level control valve in the pump discharge line receives open signal when breaker closes.

   (2) Intake vacuum priming vacuum breakers on discharge line receive open signal when breaker is open.

QUESTION 60: (1.0)

Unit 1 is in Refueling Shutdown with the temporary jumper to the CC heat exchangers in service. The Unit 1 "B" & "D" high level intake structures are stop logged and drained. Unit 2 has sustained minor flooding on the "2C" Circulating Water Line.

What adverse effects will result if stop logs are placed at the Unit 2 "C" high level intake structure.

a. Loss of Component Cooling Service Water.

b. Loss of charging pump cooling water.

c. Unit 2 Turbine Trip on low intake canal level.

d. Loss of #5 MER chiller cooling water.

ANSWER: c

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: 3 out of 4 intake canal level channels < 23.5 feet while cause a dual unit turbine trip resulting in Unit 2 reactor trip.	Distractors plausible: a – misunderstanding of the sources of water to component cooling b – misunderstanding of sources of water to the charging pumps. d – misunderstanding of sources of water to the #5 MER.	Distractors incorrect: a – CC comes from Unit 1C. b – Charging pump CC has three sources and one of them still remains from 2A. – d MER #5 is supplied from 2A and 2C.
K/A: 075/GEN-2.1.32	Objective: 2116	Source: New
Reference: AP-13.0, ND-89.5-LP-2	Level: Knowledge	

WATER BOX 2A

- 2-SW-MOV-201B, BC HX SW SUPPLY
- 2-SW-MOV-203C and 2-SW-MOV-203D, RS HX B & C SW SUPPLY
- 2-CW-LE-202, Intake Canal Level Protection CH. III
- 2-SW-11, SW to CHG PP SW and Control Room Chillers
- 2-SW-532 and 2-SW-533, SW to MER 5 Control Room Chillers

WATER BOX 2B

- 2-SW-MOV-202A, SW PUMP SUPPLY and Isolation to 2-SW-P-100

WATER BOX 2C

- 2-SW-MOV-201A, BC HX SW SUPPLY
- 2-SW-MOV-203A and 2-SW-MOV-203B, RS HX A & D SW SUPPLY
- 2-CW-LE-203, Intake Canal Level Protection CH. IV
- 2-SW-474, SW to CHG PP SW and MCR Chillers
- 2-SW-530 and 2-SW-531, SW to MER 5 Control Room Chillers

WATER BOX 2D

- 2-SW-MOV-202B, SW PUMP SUPPLY and Isolation to 2-SW-P-100

<u>WATER BOX 1A</u>

- 1-SW-MOV-101B, BC HX SW SUPPLY
- 1-SW-MOV-103C and 1-SW-MOV-103D, RS HX B & C SW SUPPLY

<u>WATER BOX 1B</u>

- 1-SW-MOV-102A, CC HX SW SUPPLY
- 1-SW-12, Isolation to 1-SW-P-100
- 1-CW-LE-102, Intake Canal level protection CH. I

<u>WATER BOX 1C</u>

- 1-SW-MOV-101A, BC HX SW SUPPLY
- 1-SW-MOV-103A and 1-SW-MOV-103B, RS HX A & D SW SUPPLY
- 1-SW-939, Temporary SW jumper to CCHXs

<u>WATER BOX 1D</u>

- 1-SW-MOV-102B, CC HX SW SUPPLY
- 1-SW-495, SW to CHG PP SW and Control Room Chillers
- 1-CW-LE-103, Intake Canal level protection CH. II

(4) The detectors in Unit 1 are channels 1 & 2, Unit 2 are channels 3 & 4.

(5) The four detectors go to two trains of a 3/4 matrix to provide the trip and isolation signals.

(6) If 3/4 intake canal level channels decrease to 23.5 feet elevation, the following actions take place:

  (a) The turbine trips (20 AST-1 and 2).

  (b) All condenser circulating water inlet and outlet motor operated valves close.

  (c) All service water flow paths through the station are isolated EXCEPT:

    1) Recirc spray heat exchanger SW

    2) Service water to Control Room chillers

    3) Charging pump service water

  (d) Reason: save water for absolutely vital uses (ultimate long term containment cooling, maintaining high head safety injection pumps operable, maintaining personnel habitability in the Control Room to allow control of the plant, maintain instrumentation reliability in the Control Room).

QUESTION 61: (1.0)

A rupture in the instrument air (IA) piping causes IA header pressure to decrease to 92 psig. Turbine building IA header pressure remains **above** 90 psig.

Which ONE of the following correctly states **all of the actuations** that occur to restore IA pressure?

a. Running IA compressor loads, standby IA compressor starts and loads.

b. Standby IA compressor starts and loads.

c. Running SA compressor loads, standby SA compressor starts and loads.

d. Standby SA compressor starts and loads.

ANSWER: c

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: running compressor loads at 100 psig, standby compressor starts and loads at 95 psig;	Distractors plausible: a/b/d – misunderstanding about interrelationships between the service and instrument air systems.	Distractors incorrect: a/b – normally y the IA compressors are not running and start at 90 psig; d – The running compressor must load prior to the standby compressor starting
K/A: SYS079-K4.01	Objective: 2333	Source: New
Reference: ND-92.1-LP-1	Level: Comprehension	

d. Air receivers

(1) Used to store air and act as a surge volume for the system, and have local pressure indication.

(2) SA and IA receivers are rated at 280 cubic feet, SA-TK-2 rated at 678.6 cubic feet.

3. Instrumentation and Controls

a. Service air compressors

(1) An instrument panel and control and indication panel are provided on each compressor for monitoring and operating the compressor.

(2) Amber lights are provided for alarm indication of compressor trips.

(3) Reset/Start Pushbutton - Resets any of the automatic shutdowns and is used to start the compressor.

(4) Stop pushbutton.

(5) Each compressor has a disconnect switch (1/2-SA-DS-1), located on the fire wall east of the compressors, used to isolate power to the compressor locally.

(6) Lead/Lag control switch, located east of the compressors, is a two position switch used to control which of the two compressors is controlling system pressure. Position 1 selects 1-SA-C-1 as the lead compressor. Position 2 selects 2-SA-C-1 as the lead compressor.

- The lead compressor will load at 100 psig and unload at 110 psig. The lag compressor will auto start and load at 95 psig and unload at 105 psig.

b. Blue and Gray compressors - These compressors have a control panel which provides indications necessary for compressor operation. These compressors are very rarely used.

Have the trainees refer to AIA-1.1, Blue/Grey for Compressor Indications and Controls, for Blue/Gray compressor controls.

c. Pressure indication in Control Room of service air header pressure: located on vertical board 1-1 and 2-1.

4. Alarms

Have the trainees refer to AIA-1.2, Station Air System Alarms, for the following discussion.

A "Service Air Compressor 1 Trouble" Alarm (1B-E5) annunciates when any of the following conditions occur:

a. Compressor motor overload

b. High oil temperature – 176°F

coalescing prefilter located upstream of the dryers as well as particulate after-filters positioned downstream. The prefilter removes liquid aerosols of water and oil and have a particulate performance rating of 100% of 0.6 microns and larger. The accumulated liquids are drained from the pre-filter housing through an automatic drain valve. The after-filters have a particulate performance rating of .9 microns absolute.

3. Instrumentation and controls

a. Instrument Air compressors

(1) The controls are three position switches, hand-off-auto. In HAND, the compressor motor runs continuously, the compressor loads and unloads at 100 and 110 psig, respectively. In AUTO, the compressor starts if pressure reaches 90 psig; load and unload setpoints are the same.

(2) Following an auto start, the auto start mercoid switch must be reset manually; it will not reset itself on high pressure.

(3) On any start signal, interlocks must be satisfied to start the compressor as follows:

- No motor overload.
- Discharge air temperature less than 444°F.
- Cooling water temperature less than 140°F.

QUESTION 62: (1.0)

A fire has been reported in the ESGR. There are flames issuing from the UPS 1A2.

Which ONE of the following actions should you direct the Fire Team to take?

a. Actuate the HP CO_2 System.

b. Actuate the LP CO_2 System.

c. Actuate the Halon System.

d. Rig fire hoses to the panel.

ANSWER: c

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 2]

Answer correct: The ESGR has Halon for fire suppression and this is where the UPS 1A2 is located.	a/b/d – Misunderstanding about what fire suppression is used for the ESGR and these suppression systems are used other places in the plant.	Distractors incorrect: a/b/d – Halon is used in the ESGR which is were the UPS 1A2 is located.
K/A: SYS086/GEN-2.4.27	Objective: 2363	Source: New
Reference: ND-92.2-LP-1	Level: Knowledge	

(3) The $HPCO_2$ system can be locked out by valving out the CO_2 bottles.

4. Indication Panel (MCR)

a. There is a panel in the MCR behind each unit's control board that provides indication of the status of portions of the $LPCO_2$ System.

b. There are initiation indication and alarms for the areas that have automatic initiation.

Ask: What areas have automatic initiation? Answer: Switchgear rooms, Cable tray rooms, Cable vaults, Cable tunnels, Turbine bearing enclosures, Generator bearing enclosures.

c. The Unit 1 panel also has an alarm for low battery airflow for both units' battery rooms.

C. Halon System

1. General description

a. The function of the Halon System is to provide fire suppression to the following areas:

Refer to/display H/T-1.29, Halon Protected Areas.

- Emergency switchgear and relay rooms

- Security - Main Access Control Area

QUESTION 63: (1.0)

A licensed RO is about to swap charging pump service water pumps, which is designated as a "Skill-of-the-Craft" task. The RO has performed the task several times this shift.

Per OPAP-0002, Operations Department Procedures, which ONE of the following is correct concerning the procedure adherence requirements?

a. The RO can perform the task without a procedure in hand, but must perform the task in accordance with the procedure.

b. The RO can perform the task without a procedure in hand, since the task is not covered by an approved procedure.

c. The RO must have the procedure in hand when performing the task, but is not required to sign-off procedure steps.

d. The RO can perform the task without procedure in hand only if he requests a peer check for each equipment manipulation.

ANSWER: a

[RO: Tier 3]
[SRO: Tier 3]

Answer correct: Skill-of-the-craft tasks may be performed without a procedure in hand if the operator is familiar with the task as long as the task is performed IAW the applicable procedure (1-OP-19.2 for this task).	Distractors plausible: b – some tasks are not yet covered by procedure; c – candidate misconception concerning the procedure adherence policy with respect to skill-of-the-craft tasks; d – peer check is a recent addition to the operators' tool bag to prevent errors during task performance.	Distractors incorrect: b – this task is covered by an approved procedure; c – the RO is not required to have the procedure in hand; d – peer check is not required nor addressed in OPAP-0002 for procedure adherence.
K/A: GEN-2.1.20	Objective: 8019	Source: North Anna Bank item #3001 (modified) (New for Surry)
Reference: OPAP-0002	Level: Knowledge	

6.0 INSTRUCTIONS

6.1 Procedure Adherence Policy

6.1.1 Operations Procedures shall be performed in a step-by-step manner with each step being completed prior to the performance of the next step, unless exception is permitted by the procedure being performed or as specified by this procedure.

6.1.2 Startup and Shutdown Procedures

a. Startup and Shutdown procedures are a series of controlling procedures that coordinate plant startup and shutdown evolutions.

b. Operations not requiring the completion of preceding steps may be performed out of sequence at the discretion of the Shift Supervisor as long as they do not modify the intent of the procedures. This is due to the overlap of steps in the procedures while transitioning from one procedure to the next.

6.1.3 Procedure instructions shall be adhered to and followed by Operations Department personnel during the performance of activities requiring procedure usage.

6.1.4 Adherence to procedures shall be accomplished by use of one of the following methods:

a. **Method 1 - Memorization**
Method by which the procedural steps for the required actions are committed to memory. This method does not permit any deviation from the Procedural Adherence Policy. Procedures for which actions should be committed to memory are the Immediate Actions in Emergency Operating Procedures, Abnormal Procedures, Annunciator Response Procedures.

b. **Method 2 - Skill-of-the-Craft**

1. The tasks shown in Skill-of-the-Craft Tasks (Attachment 1) are examples that may be performed without a procedure in hand, however, they shall be performed in accordance with the applicable procedure if the evolution is covered by an approved procedure. If a Qualified Operator performing the task is not familiar with the task, the procedure shall be in hand when performing the task or another Qualified Operator familiar with the task shall be present. Copies of the procedure shall be available at the work location for reference during performance of the task, if necessary.

QUESTION 64: (1.0)

Which ONE of the following consequences does a spurious SI present (assuming all systems function as designed)?

a. Pressurized Thermal Shock due to RWST water injection.

b. Degraded containment conditions due to PRT rupture.

c. Degraded heat sink due to Main Feed Pump trip.

d. Overload of the Station Service Electrical Distribution.

ANSWER: b

[RO: Tier 2/Group 3]
[SRO: Tier 3/Group 3]

Answer correct: Pressurizer level will increase due to three charging pumps running without letdown resulting in the pressurizer going solid and eventually rupturing the PRT rupture disks at 100 psig.	Distractors plausible: a – The colder RWST would be a concern during FR-C.1 recovery event. b – The main feed pumps are tripped on an SI. d – The starting of SI equipment will cause electrical load to increase.	Distractors incorrect: a – The colder RWST water will mix with the RCS water minimizing thermal shock. c – The main feed pumps will trip but the auxiliary feed pumps will start. d – the SI components will load on Reserve Station Service.
K/A: SYS007.K3.01	Objective: 3035	Source: NEW
Reference: ND-95.3-LP-3	Level: Knowledge	

QUESTION 65: (1.0)

Following a refueling shutdown, the RCS is in the process of being heated up and pressurized. The RO notes the CC surge tank level has been increasing for the past hour.

Isolation of which ONE of the following would potentially stop this increase?

a. RCP seal water return heat exchanger.

b. Neutron Shield Tank heat exchanger.

c. Primary Drains Tank cooler.

d. RHR pumps seal cooler.

ANSWER: d

[RO: Tier 2/Group 3]
[SRO: Tier 2/Group 3]

Answer correct: The RHR pumps have a higher discharge pressure than the CC pumps.	Distractors plausible: a/b/c – these systems are cooled by Component Cooling.	Distractors incorrect: a/b/c – these systems operate at a lower pressure than RHR; therefore, CC water would be forced into them.
K/A: SYS008.A2.02	Objective: 1910	Source: NEW
Reference: ND-88.5-LP-1 AIA1.1	Level: Knowledge	

COMPONENT COOLING SYSTEM LOADS

CARF/NST

17. Containment Instrument Air Compressor
18. Containment Air Recirc Fan Coolers
19. Neutron Shield Tank Coolers

CRDM Shroud Cooling/RCP

19. Shroud Cooling Coils
21. RCP Thermal Barrier Heat Exchangers †
22. RCP Motor Air Coolers
23. RCP Bearing Lube Oil Coolers

Hot Pipe Containment Penetration Cooling (>150^{0}F)

24. Containment Penetration Coolers
    a. Letdown
    b. Blowdown
    c. Main Steam
    d. Main Feed

Excess Letdown/RHR

25. Excess Letdown Heat Exchanger †
26. Primary Drains Cooler
27. RHR Heat Exchanger †
28. RHR Pump Seals †
29. Primary Shield Wall Coolers - for each loop penetration

Notes:

† Possible source of leakage into Component Cooling

QUESTION 66: (1.0)

Which ONE of the following prevents fuel assembly movement with cavity water level less than 23 feet?

a. Administrative requirements only.

b. Fuel mast interlock.

c. Fuel elevator interlock.

d. Manipulator crane interlock.

ANSWER: a

[RO: Tier 2/Group 3]
[SRO: Tier 2/Group 2]

Answer correct: There are no mechanical or electrical interlock to prevent a fuel assembly from being moved with low water level.	Distractors plausible: b/c/d – these are actual mechanical interlocks on the fuel movement equipment.	Distractors incorrect: b/c/d – none of these interlocks will prevent moving fuel with a low level in the cavity.
K/A: SYS034.A1.02	Objective: 2469	Source: NEW
Reference: ND-92.5-LP-1	Level: Knowledge	

6. At least one residual heat removal pump and heat exchanger shall be operable to circulate reactor coolant. The residual heat removal loop may be removed from operation for up to 1 hour per 8-hour period during the performance of core alterations or reactor vessel surveillance inspections.

7. Two residual heat removal pumps and heat exchangers shall be operable to circulate reactor coolant when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

8. At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange during movement of fuel assemblies.

9. With the reactor vessel head unbolted or removed, any filled portions of the Reactor Coolant System and the refueling canal shall be maintained at a boron concentration which is:

   a. Sufficient to maintain K-effective equal to 0.95 or less, and

   b. Greater than or equal to 2300 ppm and shall be checked by sampling at least once every 72 hours.

10. Direct communication between the Main Control Room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

11. No movement of irradiated fuel in the reactor core shall be accomplished until the reactor has been subcritical for a period of at least 100 hours.

QUESTION 67: (1.0)

During performance of turbine valve freedom testing, the test engineer erroneously directs a local turbine trip.

Which ONE of the following components initiates the reactor trip signal?

a. 74/AST.

b. 63-AST-4/5/6.

c. 4/4 Governor valves closed.

d. 20/ET.

ANSWER: b

[RO: Tier 2/Group 3]

Answer correct: The turbine trip / reactor trip signal is generated by 2/3 auto stop oil pressure switches , 45 psig or 4/4 Stop valves closed	Distractors plausible: a – This contact is used in the turbine trip block. c – 4/4 Stop valves closed generates a turbine trip. d – 20/ET provides a backup turbine trip from the turbine trip / reactor trip signal.	Distractors incorrect: a – This contact is used to detect a SOV turbine trip has occurred. c – 4/4 Stop valves closed generates a turbine trip not governor valves. d – 20/ET provides a backup turbine trip.
K/A:SYS045.K4.11	Objective: 2011	Source:
Reference: ND-89.2-LP-8	Level: Knowledge	

c. Master trip solenoid valve SOV-20/AST-2 is normally deenergized and serves as a backup to the action of the control block. In the event of an electrical trip, this solenoid energizes, opening its associated valve. This causes the auto-stop header to drain to the bearing oil reservoir in the pedestal, depressurizing the header, opening the interface emergency trip valve, initiating a turbine trip.

d. The master trip solenoids are powered from independent D.C. power sources. SOV 20/AST-1 is powered from "A" main station battery and SOV 20/AST-2 is powered from "B" main station battery. Thus, a loss of a single D.C. power supply does not prevent or cause a turbine trip (single-failure criteria).

C. Pressure Switches - 63-1/AST Through 63-6/AST

The pressure in the auto-stop oil header is continually monitored by six PRESSURE SWITCHES (63-1/AST through 63-6/AST). The pressure switches provide various operational and protection signals to other systems as follows:

1. Pressure switch 63-1/AST actuates at 45 psig and provides an input to the first out annunciators for low bearing oil, low vacuum, thrust bearing trip alarms, the overspeed, and solenoid trip alarms. These annunciators will not actuate until ASO pressure is less than 45 psig (contact for 63-1/AST is in series with the first out alarms).

2. Pressure switch 63-2/AST actuates at 45 psig and provides an input to the auto-stop latch circuit. This signal "informs" the EHC system that the turbine is latched. When the EHC sees that the turbine is not latched it sends a full close signal to the governor valves. The switch also causes the UNIT TRIP light on the turbine control panel to illuminate and the LATCH pushbutton to extinguish when ASO pressure <45 psig, and UNIT TRIP light to extinguish and the LATCH pushbutton to illuminate when pressure is > 45 psig. Furthermore, it is a permissive to allow the following lights to illuminate:

   (a) GV ∇, GVΔ, GV FAST when in turbine manual,

   (b) LOAD when a main generator output breaker is shut,

   (c) SPEED when both main generator output breakers are open.

3. Pressure switch 63-3/AST has no function. It was originally used to open the main turbine drain valves on a turbine trip. This function is now accomplished by relay 63X/AST-3 in parallel with 20/ET.

4. Pressure switches 63/AST-4 through 6 actuate at 45 psig and provide channel I, II, and III signals for the following:

   a. RPS reactor trip Train A and B. When 2 out of 3 channels actuate and power is >P-7, a reactor trip signal is generated.

   b. RX TRIP CH 1 AUTO STOP OIL DUMP (F-A4) indicates 63/AST-4 has actuated. Channels II and III have similar alarms 1F-A5 and 1F-A6 respectively for 63/AST-5 and -6.

   c. RX TRIPPED BY TURB TRIP (E-A8) first out annunciator, indicates 2/3 channels are <45 psig and power is >P-7.

QUESTION 68: (1.0)

During a spurious Hi Hi CLS, the team is unable to reset the Hi Hi CLS signal.

The team must locally secure the _______ pumps to ensure Containment pressure does not drop below ____ psia which would potentially damage the _______.

a. Containment Spray, 9, Containment Dome Liner

b. Containment Spray, 8, Containment Basemat

c. Inside Recirc Spray, 9, Containment Dome Liner

d. Inside Recirc Spray, 8, Containment Basemat

ANSWER: b

[RO: Tier 2/Group 3]
[SRO: Tier 2/Group 2]

Answer correct: Containment Spray which is using cold RWST water will cause excessive depressurization of containment resulting in basemat damage.	Distractors plausible: a/c – Misunderstanding of the weaker component in the containment. d – Misunderstanding of the temperature of the cooling water.	Distractors incorrect: a/c – The basemat is the limiting component. d – Inside recirc spray will not be cold enough to cause line damage to depressurization.
K/A: SYS103.G2.1.32	Objective: 1863	Source: NEW
Reference: TS-3.8, ND-88.4-LP-2	Level: Knowledge	

b. With the containment air partial pressure outside the acceptable operation range, restore the air partial pressure to within acceptable limits within 1 hour or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. If the air partial pressure cannot be maintained greater than or equal to 9.0 psia, the reactor shall be brought to the hot shutdown. The lower limits are imposed to ensure that the containment pressure does not decrease below the design value of 8 psia. The shell and dome plate liner of the containment are capable of withstanding an internal pressure as low as 3 psia; however, the bottom mat liner is capable of withstanding an internal pressure of only 8 psia.

d. Figure 3.8 is based on two peak pressures occurring in containment following a DBA. If the hot leg ruptures there will not be a second peak, if the cold leg ruptures then there will be a second peak which will occur while spray is activated. Flow to steam generators will result in flashing. The horizontal line of the figure prevents peak pressure from being excessive on the first peak and the sloping line limits the pressure rise on the second peak (if the cold leg ruptures), this discussion is in the UFSAR and alluded to in the basis section of the T.S.

Distribute LER 89-016, Inadvertent Reactivity Addition by Boron Dilution Without Containment Integrity Intact Due to Leaking RCP Standpipe Makeup Valve, and LER 92-005, Loss of Refueling Containment Integrity, and discuss events with the class.

b

provide AFW from the cross-connect from the ... connect must be initiated and the RCPs tripped within 10 minutes after the onset of the accident occurring at 100% power. The simulator validation of this accident assumed hot full power conditions (Reference 45). HELB scenarios initiated at reduced power or zero-power conditions may provide more operator response time due to higher steam generator inventory and less rapid reduction in steam generator inventory. Therefore, the 10-minute requirement is only applicable to the limiting event initiated at hot full power conditions. (Reference 32). (LONF AADBD SUEAC 246 and 956.)**

2.10 Appendix R (NAPS Only):

- Operators will control RCS temperature at the no-load value within 30 minutes after the initiation of the transient, as directed by the EOPs (Reference 28). (AADBD cases not available.)

- For a reactor trip coincident with loss of both motor driven AFW pumps and all RCPs, ensure the TDAFWP is aligned to at least 2 SGs within 70 minutes (References 15 and 28 NAPS). Surry AFW pump to SG alignment differs from that of North Anna. Each of Surry's pumps are normally aligned to all three SGs. Therefore, this requirement is not applicable to Surry. (AADBD cases not available.)

2.11 Locked Rotor Accident

- Operator starts one MCR emergency fan (pressurization mode) prior to indication of depleted bottled air system, assumed to be one hour after SGTR (Reference 48 and 49 NAPS, 50 SPS). (LOCROT AADBD SUEAC 744, 745, 746, 758, 759, 760.)**

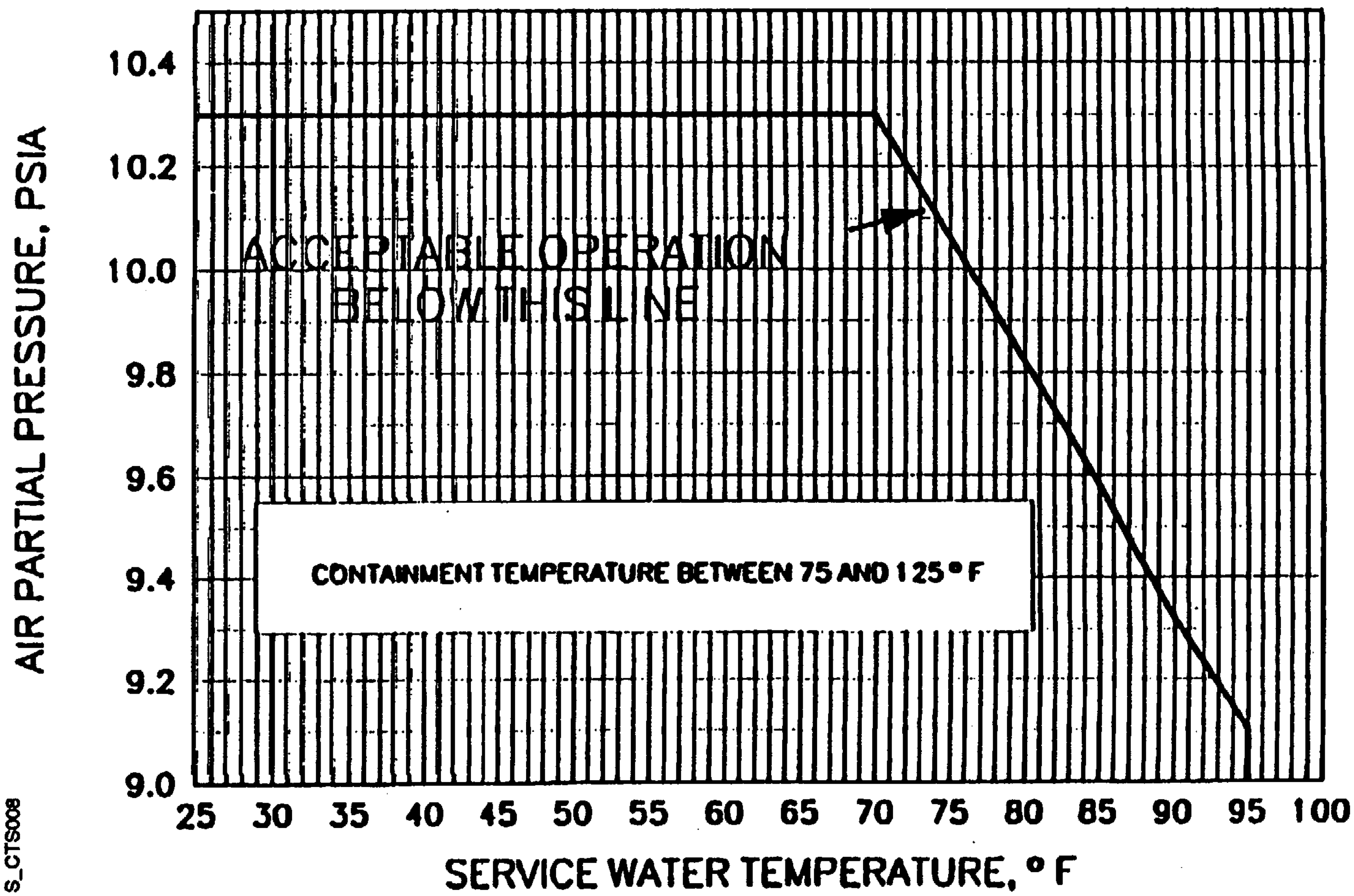
2.12 Low Temperature Overpressure (LTOPS) Mass Addition Accident

- Terminate uncontrolled mass addition to the RCS (from an initial condition requiring the Low Temperature Overpressure Pressure System in service) within ten minutes from event initiation (Reference 16 NAPS, 17 SPS). This requirement is due to the sizing of the PORV backup air/nitrogen supply. (AADBD cases not available.)

2.13 Spurious Containment Spray

- SPS ONLY: Terminate spurious containment spray actuation prior to containment pressure falling less than 8.0 psia (Reference 14). The North Anna containments are designed to withstand a lower negative pressure than the Surry containments. In

SURRY TECHNICAL SPECIFICATION CURVE MAX CONTAINMENT ALLOWABLE AIR PARTIAL PRESSURE INDICATION VS. SW TEMP



QUESTION 69: (1.0)

Which ONE of the following identifies the maximum steady state power level that does not exceed the maximum steady state power level allowed by the <u>facility license</u>?

a. 100%

b. 107%

c. 109%

d. 118%

ANSWER: a

[RO: Tier 2/Group 1]
[SRO: Tier 2/Group 1]

Answer correct: Maximum allowed by license.	Distractors plausible: b. Reactor trip setpoint c. TS required reactor trip setpoint. d. Safety Limit setpoint	Distractors incorrect: b/c/d all > 100% which is the maximum allowed by license.
K/A: SYS015.G.2.1.10	Objective: 2563	Source: NEW
Reference: TS license ND-93.2-LP-4	Level: Knowledge	

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70, and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. <u>Maximum Power Level</u>

      The licensee is authorized to operate the facility at steady state power levels not in excess of 2546 megawatts (thermal). = 100% power

   B. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 222, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   C. <u>Reports</u>

      The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. <u>Records</u>

      The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

   E. Deleted by Amendment 65

   F. Deleted by Amendment 66 and again by Amendment 71

   G. <u>Steam Generator Repair Program</u>

      (1) The Surry Power Steam Generator Repair Program for Unit No. 1 is approved.

      (2) During the steam generator repair program the following conditions shall be met:

         (a) All fuel shall be removed from the reactor pressure vessel and stored in the spent fuel pool.

         (b) Temporary containment and ventilation systems shall be installed and operated for all cutting and grinding operations involving components with removable radioactive contamination greater than 2200 DPM per 100 cm^2 except

QUESTION 70: (1.0)

Which ONE of the following manipulations is prohibited during a Reactor Startup?

a. Withdrawing the shutdown banks and performing a Normal Boration.

b. Withdrawing the shutdown banks and raising RCS temperature.

c. Withdrawing the shutdown banks and performing an Alternate Dilution.

d. Withdrawing the shutdown banks 14 hours after a Reactor Trip from 100% power.

ANSWER: c

[RO: Tier 3]
[SRO: Tier 3]

Answer correct: Can not have two different positive reactivity changes that are done simultaneously. Simultaneously	Distractors plausible: a/b/d – Trainee misconception about what causes a positive reactivity change in the reactor. Also not realizing that two positive reactivity additions are not allowed simultaneously.	Distractors incorrect: a/b/d - each of these are adding only one positive form of reactivity at a time.
K/A: GEN-2.2.1	8040	Source: New
Reference: VPAP-1410	Level: Comprehension	

3. A Reactor Engineer or qualified designee shall be in the control room area during a core on-load to monitor the core's response to fuel loading and to provide technical assistance to the Refueling SRO.

e. Reactivity Management for dry cask storage evolutions will be controlled by Nuclear Analysis and Fuel.

6.2.6 **Approach to Criticality**

a. A Reactor Engineer and the STA shall be present in the Control Room during approach to criticality and Special Tests that affect the core or measure core parameters. These individuals shall advise the Shift Supervisor and may coordinate test activities to further ensure that proper control is maintained.

b. Estimated Critical Position calculations shall be prepared and independently verified.

c. Reactor start-ups shall be designated as Infrequently Conducted and Complex Evolutions in accordance with VPAP-0108.

d. Conduct of operations governing the approach to criticality shall include the following:

1. Pre-job brief before the approach to criticality.
   - Emphasizing the operator at the control's responsibility for the core and to expect criticality at any time especially during positive reactivity additions
   - Emphasis on the need for conservative actions and strict compliance with procedures
   - Guidance on actions to be taken if expected responses are not observed

2. Ensuring that dedicated start-up team members are not distracted during the approach to criticality.

e. If the start-up is suspended near the point of criticality for an extended period of time, then the core shall be made sufficiently subcritical to avoid an inadvertent criticality, as directed by startup procedures.

f. Two operator controlled positive reactivity additions shall not be made simultaneously (e.g., dilution and rod withdrawal is not allowed).

QUESTION 71: (1.0)

Maintenance would like to remove the danger tags from a **4160-volt breaker** so they can cycle it in the TEST position. The tag-outs cannot be cleared.

Which ONE of the following correctly describes the operator actions required to facilitate this request?

a. Remove the danger tags in accordance with an approved partial clearance, rack the breaker to TEST, then return the danger tags to the Operations Annex.

b. Remove the danger tags in accordance with an approved partial clearance, rack the breaker to TEST, then destroy the danger tags and return the tagging records to the Operations Annex.

c. Remove the danger tags in accordance with an approved temporary release, rack the breaker to TEST, then place a special order blue tag on the breaker racking device and return the danger tags to the Operations Annex.

d. Remove the danger tags in accordance with an approved temporary release, rack the breaker to TEST, then place in tags in a temporary release envelope and attach the envelope to the breaker racking device.

ANSWER: d

[RO: Tier 3]
[SRO: Tier 3]

Answer correct: per VPAP-1402, breaker testing (TEST position) is done using temporary release; the sequence listed is correct per OPAP-0010.	Distractors plausible: a & b – partial clearances are used to remove danger tags for testing purposes; c – temporary releases are used to remove danger tags for testing purposes.	Distractors incorrect: a & b – partial clearances are not used to clear tags for breaker testing in TEST position; c – special order blue tag is not placed on the breaker racking device, danger tags are not returned to Annex.
K/A: GEN-2.2.13	Objective: 8038	Source: New
Reference: VPAP-1402; OPAP-0010	Level: Knowledge	

b. If the Shift Supervisor or Surry Water Treatment Coordinator (for tag-outs in the Condensate Polishing Building only) concurs with the request, the Tag-Out shall be modified by Operations personnel in accordance with OPAP-0010, Tag-Outs.

c. The supervisor shall verify that the Tag-Out modification provides a safe working boundary by performing an in-field verification before performing task.

6.5.5 **Temporary Releases of Tag-Outs for Breaker Testing**

NOTE: A Temporary Release is used to remove a Tag-Out on a breaker to allow it to be moved from the disconnect position to the test position and back to the disconnect position after testing is completed. The Temporary Release should normally be in effect less than one shift.

a. If Electrical Maintenance or Control Operations needs to test a breaker that has a Danger Tag on it and the Tag-Out cannot be cleared, the requestor shall submit a Temporary Release (Attachment 3) to the Shift Supervisor.

1. If a Temporary Release is required, the individual requesting the Temporary Release shall notify affected craft supervisors. The Shift Supervisor or Testing Supervisor (SRO) shall determine which craft supervisors must be notified of the Temporary Release, and determine which Work Orders shall be placed on hold, by reviewing the scope of work for each Work Order covered by the Tag-Out.

2. If Operations personnel determine that multiple supervisors are required to be notified, the supervisor requesting the Temporary Release shall obtain approval signatures of the other individuals on the Temporary Release (Attachment 3).

3. The cognizant Electrical Maintenance or Control Operations craft supervisors requesting the Temporary Release shall ensure that applicable Work Orders are placed on hold in accordance with VPAP-2002, Work Requests and Work Order Tasks.

b. If the Temporary Release can be completed, assign a Qualified Operator to review the information on the Temporary Release.

6.7.3 The Qualified Operator assigned the task of reviewing the required Temporary Release information shall:

a. Determine the Danger Tag numbers on the breaker to be tested and verify the accuracy of the Temporary Release.

b. Verify the following information is listed on the Temporary Release for each Danger Tag:

- Tagging Record number
- Danger Tag number
- Department of the Craft Supervisor
- Name of the Craft Supervisor, including Master Tag-Out foremen
- Signature of the Cognizant Craft Supervisors

c. Complete a Temporary Electrical Release Envelope Tag.

6.7.4 The Shift Supervisor or Surry Supervisor Water Treatment (for tagouts in the Condensate Polishing Building) shall assign a Qualified Operator to perform the Temporary Release and assign a Qualified Operator to perform the independent verification.

6.7.5 The SRO should notify the Electrical Department to remove the Ground Placement Tag and grounding device, if applicable.

6.7.6 The Qualified Operator assigned the Temporary Release shall:

a. Verify Danger Tags installed on the breaker match the Temporary Release listing.

b. Remove Danger tag(s) from the breaker.

c. Place the breaker in the Test Position in accordance with Shift Supervisor direction.

d. Sign and date the Temporary Electrical Release Envelope and place the Danger tag(s) in it.

QUESTION 72: (1.0)

Given the following Unit 1 conditions:

- All control rods are fully withdrawn.
- A failure in the rod control system causes "D" bank rods to step outward.
- The RO notes the failure and places rod control in MANUAL.
- Group step counters for "D" bank indicate 243 steps.
- IRPIs for "D" bank rods indicate (on the average) 230 steps.

Which ONE of the following is correct concerning the disparity between the group step counters and the IRPIs?

a. This is expected; the IRPIs should eventually drift up to indicate approximately 243 steps.

b. This is expected; the IRPIs should continue to indicate approximately 230 steps.

c. This is **not** expected; the IRPIs should have tracked with the group step counters; they are **still operable** per TS-3.12.C.

d. This is **not** expected; the IRPIs should have tracked with the group step counters; they are **inoperable** per TS-3.12.C.

ANSWER: b

[RO: Tier 2/Group 2]
[SRO: Tier 2/Group 1]

Answer correct: rods are physically incapable of being withdrawn beyond 230 steps; if demanded, the CRDMs will withdraw the rods to 230 steps, then ratchet as long as demanded (presumably without dropping the affected rods).	Distractors plausible: a – IRPIs are known to lag the group step demand counters somewhat, and also to drift up and down based on temperature c – IRPIs should track the group step counters, albeit with some lag possible; d – if IRPIs differ from group step counters >12 steps, they are inoperable per TS-3.1.3.2.	Distractors incorrect: a – since rods are physically at 230 steps, IRPIs should remain at or near 230 steps; c & d – the disparity IS expected, since rods are physically at 230 steps.
K/A: SYS014-K5.01	Objective: 2632	Source: New
Reference: ND-93.3-LP-3	Level: Knowledge	

Full Out Steps −128 = Bank Overlap		384 + Full Out Steps = Total Steps
225	97	609
226	98	610
227	99	611
228	100	612
229	101	613

> Refer to the trainees to 0-DRP-004, Precautions, Limitations, and Setpoints Attachment 2 Section 6, Rod Control System for the latest BOU thumbwheel switch settings. Point out that S2 denotes the full out steps for the unit.

9. Periodically the full out rod position is changed to limit jack shaft wear at the location where the grippers latch the rods in the full out position. Actual all out is 230 steps

> Refer to/re-display H/T-3.5, Rx Control Unit.

10. The four Slave Cyclers produce lift, move, and stationary current orders and send the orders to the Power Cabinets. The slave cyclers are activated by pulses from the master cycler. The slave cyclers produce the necessary output signals to control the firing sequence of the stationary, movable, and lift CRDM coils. The firing sequence determines rod direction. The slave cyclers also provide output signals to the digital step counters, the Pulse-to-Analog (P/A) Converter Unit, and the P-250 Computer. The P/A Converter and the P-250 Computer determine a bank's position based on Group One step position only.

QUESTION 73: (1.0)

An accident at SPS results in a radioactive plume passing over the high level intake structure. An operator at the high level intake structure receives a whole body dose of 1 rem per hour and he is exposed for 8 hours.

This operator is in the ______ zone and ______ exceeded the legal limit of 10CFR100.

a. exclusion, has

b. exclusion, has not

c. low population, has

d. low population, has not

ANSWER: b

[RO: Tier 3]
[SRO: Tier 3]

Answer correct: per 10CFR100, the high level intake structure falls within the exclusion area and the total dose does not exceed 25 rem.	Distractors plausible: all – misconception of exclusion area and allowable dose as delineated in 10CFR100.	Distractors incorrect: a – 8 rem is less than 25; c and d – high level intake structure is not in the low population zone.
K/A: GEN-2.3.2	Objective: 8051	Source: New
Reference: ND-80.2-LP-5	Level: Knowledge	

must comply with certain regulations concerning public safety in the event of a major accident. The purpose of 10CFR100, "Reactor Site Criteria," is to describe criteria which guide the NRC in evaluating the proposed site for construction. These criteria include intended use of the reactor, application of engineering standards to design, safety features and radioactive release boundaries, population density, and physical characteristics such as seismology, meteorology, geology, and hydrology. The NRC makes an evaluation of these criteria and, if acceptable, issues a license to construct a nuclear facility.

2. With respect to radiation protection, one of the key criteria of 10CFR100 is population density and use of the land surrounding the proposed site.

3. 10CFR100 defines the following boundaries associated with radioactive releases, including in that definition a description of the basis behind the establishment of its boundaries, an identification of the actual area boundaries and the dose limits associated with each of those boundaries:

- Exclusion Area
- Low Population Zone

4. In 10CFR100, the exclusion area is defined as "that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area."

5. The low population zone is defined as "the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident."

6. Each nuclear plant site has a defined radius for the exclusion and low population zones. The outer boundary of the exclusion area is "of such size that an individual located at

any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure." The exclusion area is bounded by a 1650 foot radius circle centered at the Unit 1 reactor containment building. The circle size was determined by the shortest distance to the site boundary and is sufficient, in conjunction with the plant design, to ensure that the dose limitations of Part 100 are met. The exclusion area is owned by and is under control of Virginia Power.

7. The low population zone is "of such size that an individual located at any point in its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure. The low population zone is bounded by a 3 mile radius circle centered at the Unit 1 reactor containment building. The nearest boundary of Newport News (the nearest densely populated center) is 4.7 miles." This distance is known as the population center distance, which is greater than one and one-third times the low-population-zone boundary distances as required by Part 100. In addition, the dose limitations of Part 100 are met with considerable conservatism.

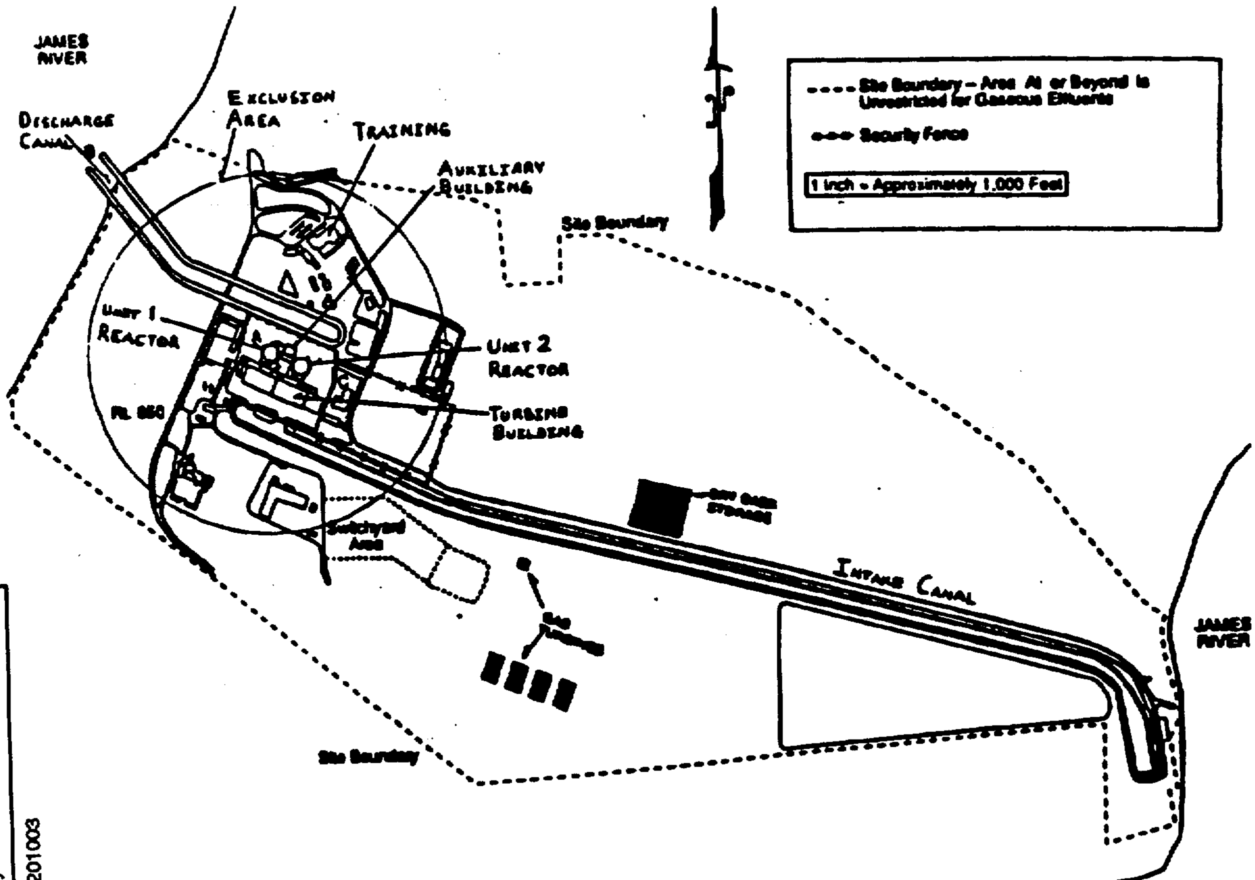
D. Describe the Technical Specifications That Aid in Meeting the Requirements of 10CFR20, 50, and 100

1. Tech Spec Section 3.11

a. The concentration of 0_2 in the waste gas holdup system shall be limited to ≤ 2% by volume whenever the H_2 concentration could exceed 4% by volume.

(1) With 0_2 > 2% but < 4% by volume, reduce the 0_2 concentration to < 2% within 48 hours.

Figure 2.1-3
SITE BOUNDARY AND MAJOR STRUCTURES



S0201003

QUESTION 74: (1.0)

Vital Bus II was de-energized in Unit 1 while at hot shutdown. The RO is performing the actions of ES-0.1.

Which ONE of the following identifies which RCP must be stopped and how much time is allowed to complete this action?

a. RCP 1A, 5 minutes.

b. RCP 1A, 2 minutes.

c. RCP 1B, 5 minutes.

d. RCP 1B, 2 minutes.

ANSWER: d

[RO: Tier 3]
[SRO: Tier 3]

Answer correct: The ARP states the RCP must be stopped within 2 minutes of a loss of CC to the motor/lube oil cooler.	Distractors plausible: a/b – Misconception of which VB supplies which RCP CC valves. c – Within 5 minutes is the WOG definition of immediate action.	Distractors incorrect: a/b – The 1B RCP would not be affected. c – Immediate pump trip is not required (within 5 minutes by WOG but rather within 2 minutes by ARP)
K/A: GEN-2.4.10	Objective: 1704	Source: New
Reference: ARP-C-B1, AP-10.02	Level: Knowledge	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: If CC flow to the RCP is lost, the RCP must be stopped within 2 minutes or before either the upper or lower bearing temperature has increased to 200°F.

* *

3. ___VERIFY RCP B CC FLOW - 0 GPM

Do the following:

a) Return RCP cooler CC flow to normal as necessary.

b) Monitor RCP bearing temperatures.

Bearing	P-250 Point
Upper Thrust	T0434A
Lower Thrust	T0436A
Upper Radial	T0433A
Lower Radial	T0435A

c) IF any bearing temperature reaches 175°F, THEN GO TO ARP 1C-F2, RCP BEARING HI TEMP.

d) GO TO Step 9.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE: A de-energized AC Vital Bus shall be re-energized within 2 hours OR the unit must be placed in Hot Shutdown within the next 6 hours.	
[1]	__EVALUATE FAILURE OF VITAL BUS 1-II: • Check Vital Bus 1-II voltage on MCR voltmeter - LESS THAN 117 VOLTS AND • Check the following TVs - CLOSED • 1-BD-TV-100B • 1-BD-TV-100D • 1-BD-TV-100F • 1-CC-TV-105B	GO TO Step 7.
[2]	__CHECK UNIT - AT POWER	IF unit on RHR with HX B in service, THEN initiate 1-AP-27.00, LOSS OF DECAY HEAT REMOVAL CAPABILITY. GO TO Step 4.
[3]	__TRIP THE REACTOR AND INITIATE 1-E-0, REACTOR TRIP OR SAFETY INJECTION	
4.	__STOP 1-RC-P-1B	

QUESTION 75: (1.0)

The plant has experienced a large-break LOCA twenty minutes ago from 100% power. The crew has transitioned from 1-E-0, Reactor Trip or Safety Injection, to 1-E-1, Loss of Reactor or Secondary Coolant. The following conditions exist:

- "A" S/G N/R level is 20%, AFW flow is 140 gpm.
- "B" S/G N/R level is 10%, AFW flow is 135 gpm.
- "C" S/G N/R level is 10%, AFW flow is 145 gpm.
- S/G pressure in all S/Gs is 1035 psig.
- RCS pressure is 100 psig and decreasing.
- No RCPs are running.
- Core Exit T/Cs are 705°F.
- All cold-leg temperatures are 280°F.
- RVLIS full-range level is 53%.
- Containment pressure is 37 psia.

Using the attached procedure, which ONE of the following identifies the required procedure to be implemented immediately?

a. 1-FR-C.2

b. 1-FR-P.1

c. 1-FR-Z.1

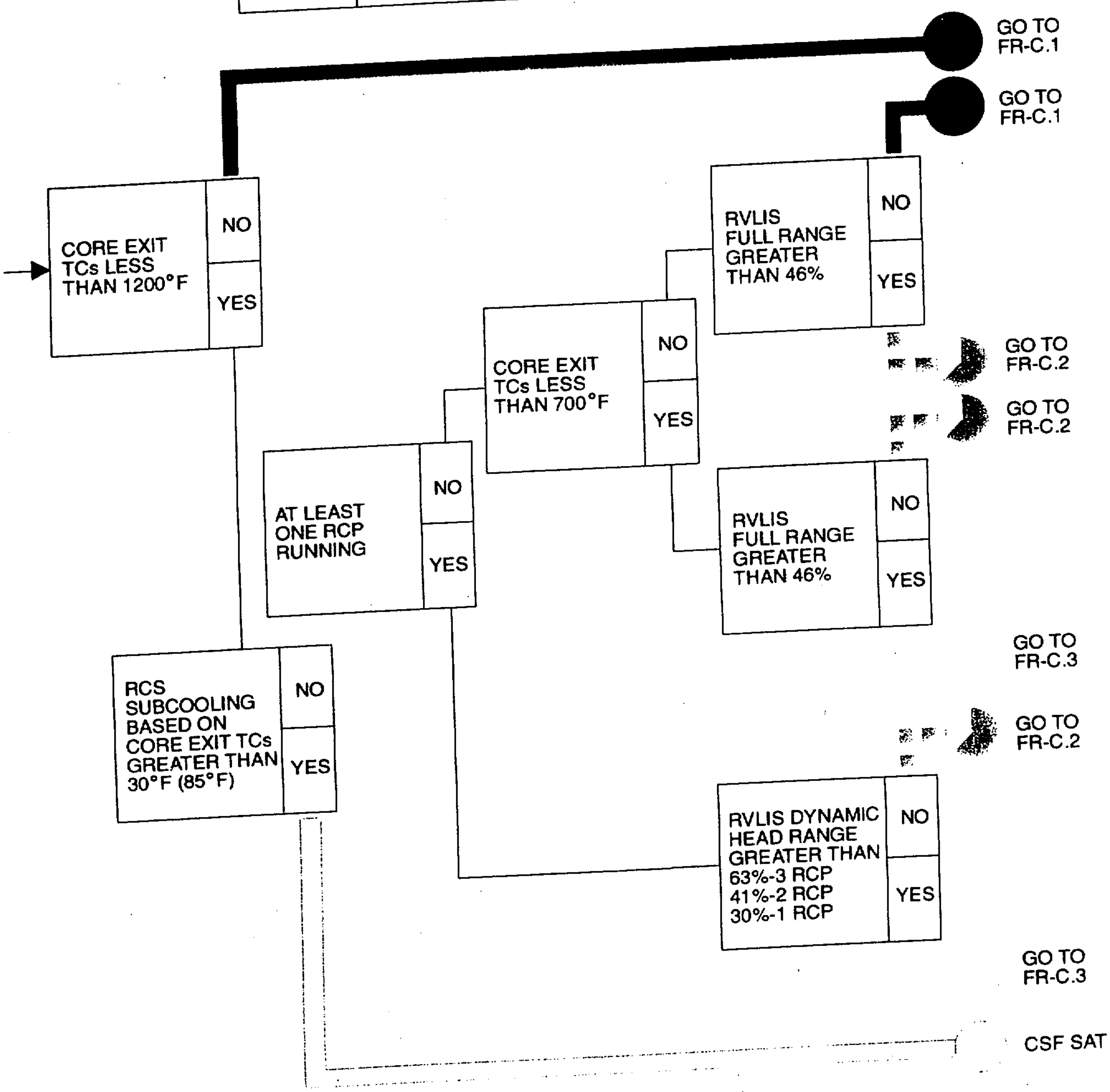
d. 1-FR-H.1

ANSWER: d

[RO: Tier 3]
[SRO: Tier 3]

Answer correct: adverse CTMT conditions (CTMT pressure is > 20 psia) require ≥ S/G N/R level > 22% or AFW flow > 450 gpm; since all S/Gs are less than 22% N/R and total AFW flow is only 420 gpm, heat sink is a red path (for the stated conditions, it is the only red path).	Distractors plausible: All – candidate misinterpretation of the data presented and/or misapplication of 1-E-0.	Distractors incorrect: a – 1-FR-C.2 is applicable for the stated conditions, but it is only an orange path and is superceded by the heat sink red path. b – 1-FR-C.1 is not applicable, since reactor vessel level is above the required value. c – 1-FR-Z.1 is applicable for the stated conditions, but it is only an orange path and is superceded by the heat sink red path.
K/A: GEN-2.4.21	Objective: 2943	Source: New
Reference: CSFST's, and OPAP-002	Level: Comprehension	

Number:	Title:	Revision:
F-2	CORE COOLING	Rev-1A



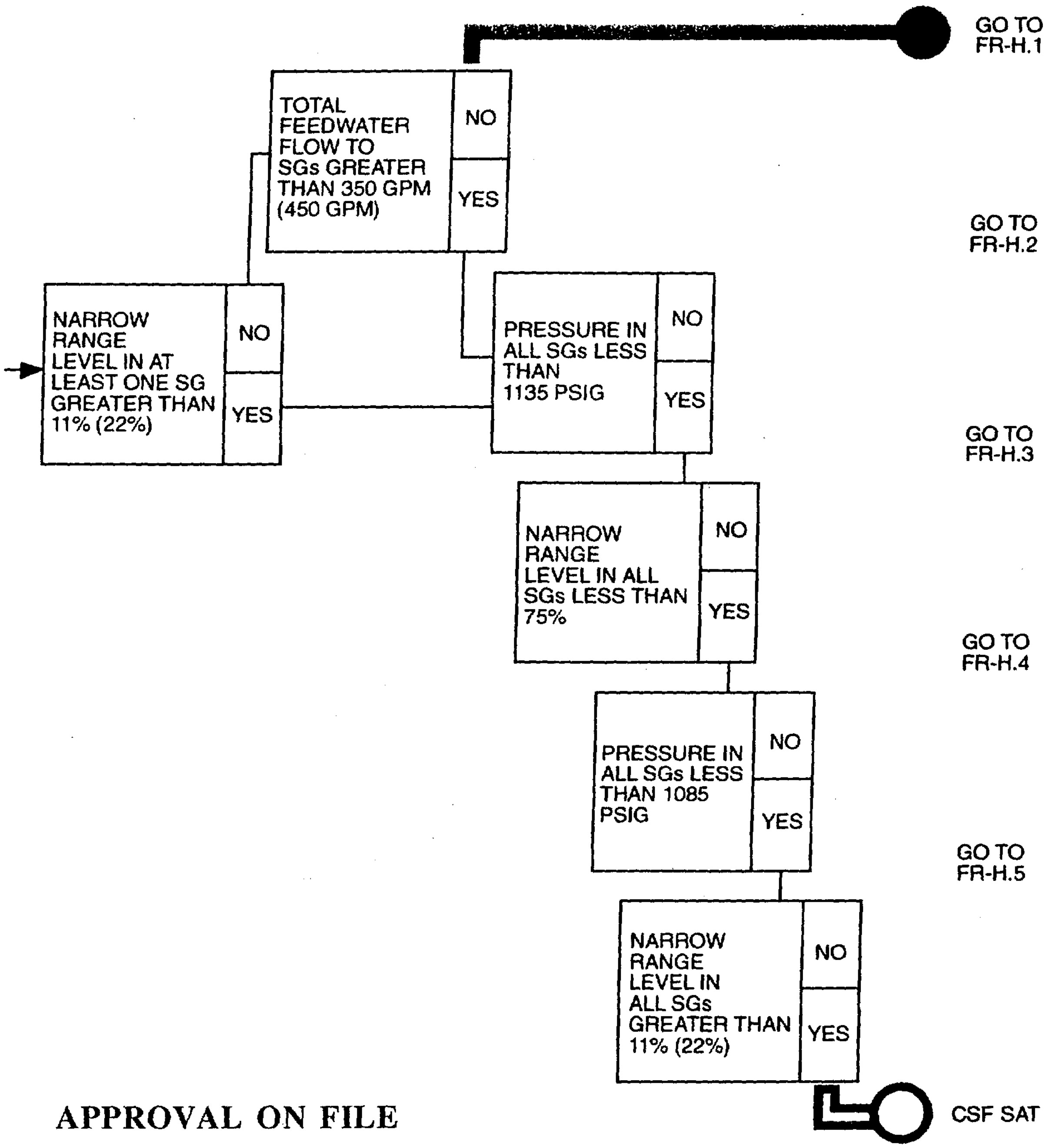
Drawing No. CB380

APPROVAL ON FILE

SNSOC CHAIRMAN

DATE

Number:	Title:	Revision:
F-3	HEAT SINK	4

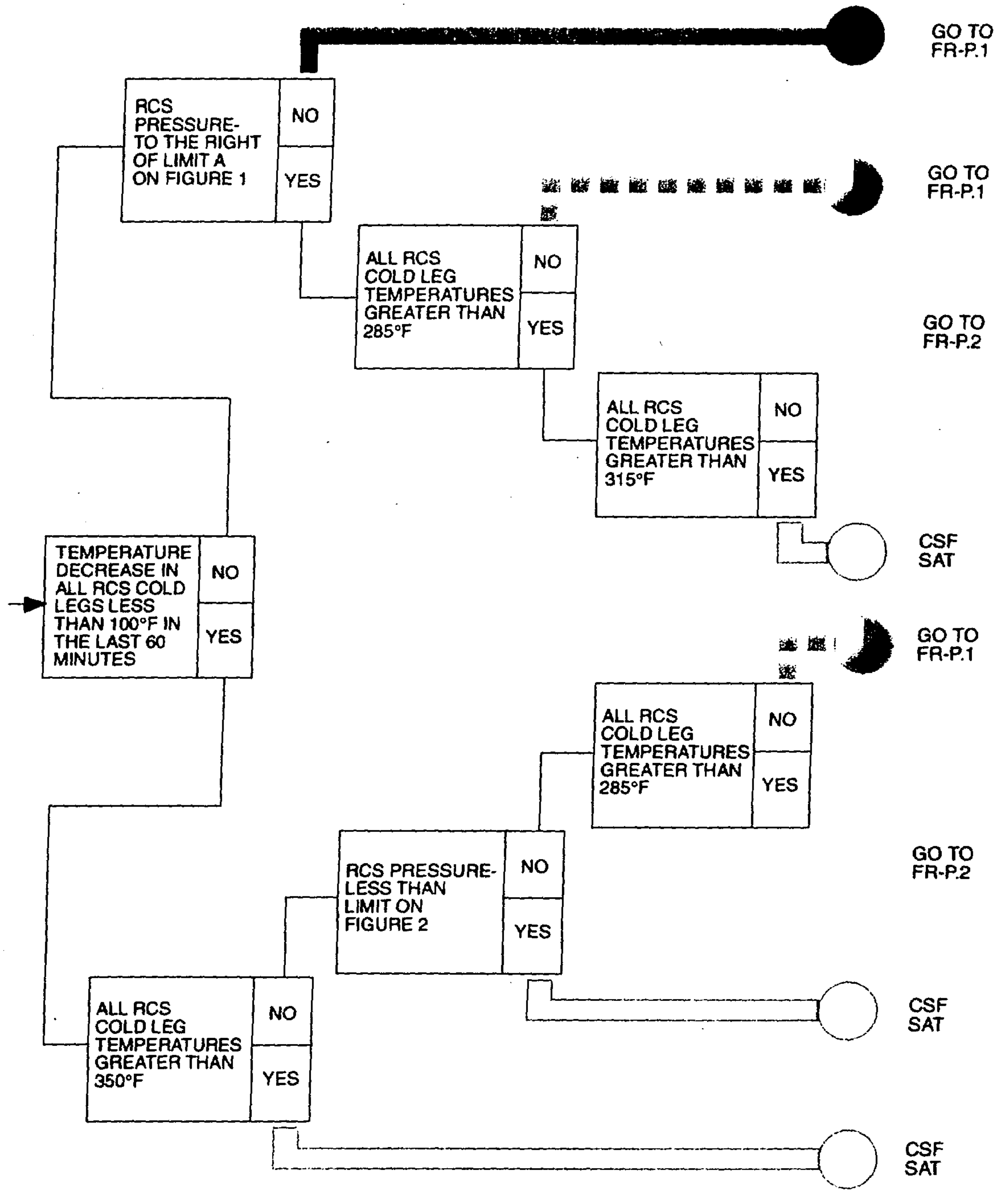


APPROVAL ON FILE

SNSOC CHAIRMAN　　　　DATE

Graphics No. CB381

Number:	Title:	Revision:
F-4	INTEGRITY	2



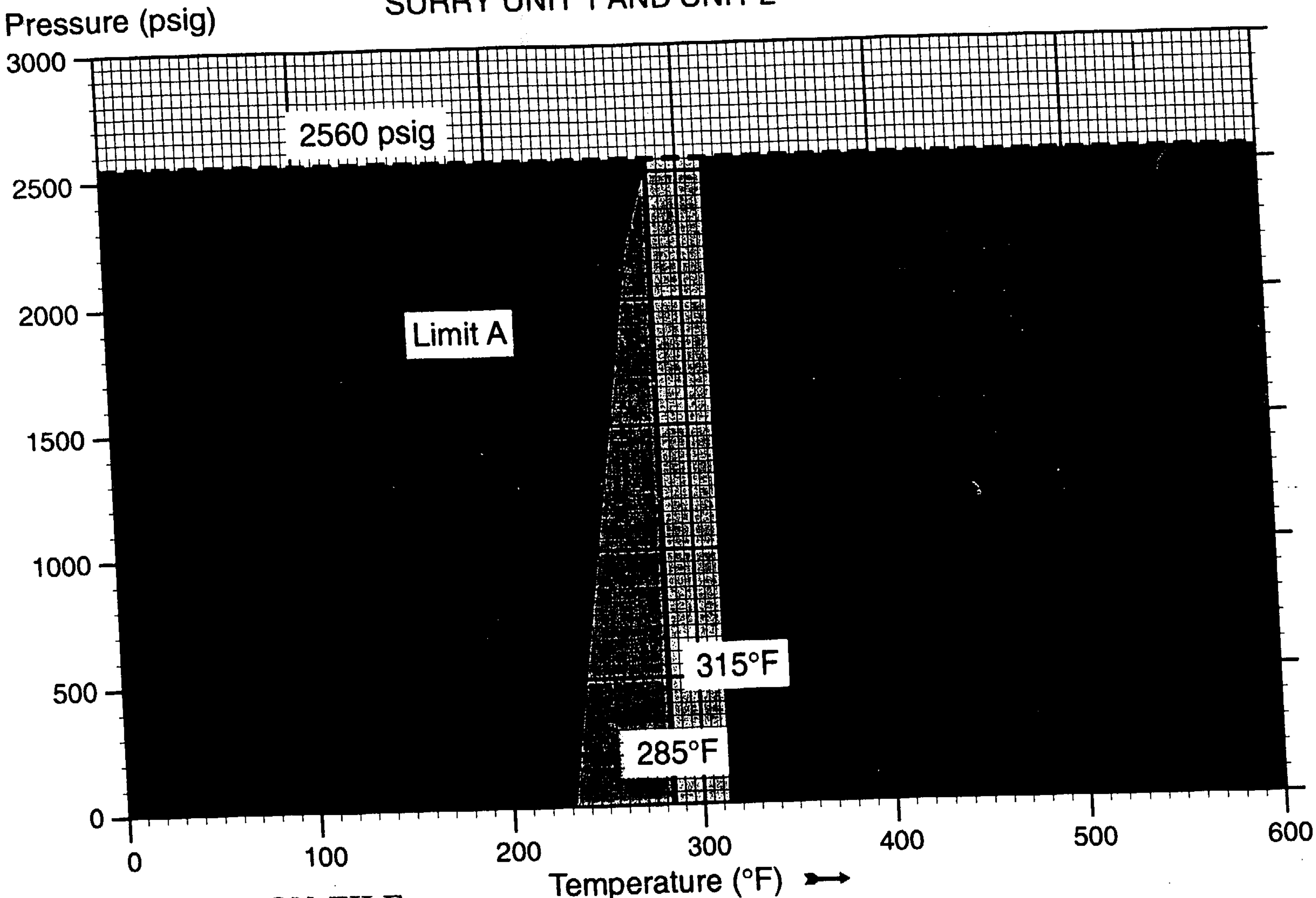
Graphics No. CB383

APPROVAL ON FILE

SNSOC CHAIRMAN   DATE

Number:	Title:	Revision:
F-4	INTEGRITY	2

FIGURE 1 - OPERATIONAL LIMITS CURVE
SURRY UNIT 1 AND UNIT 2



APPROVAL ON FILE

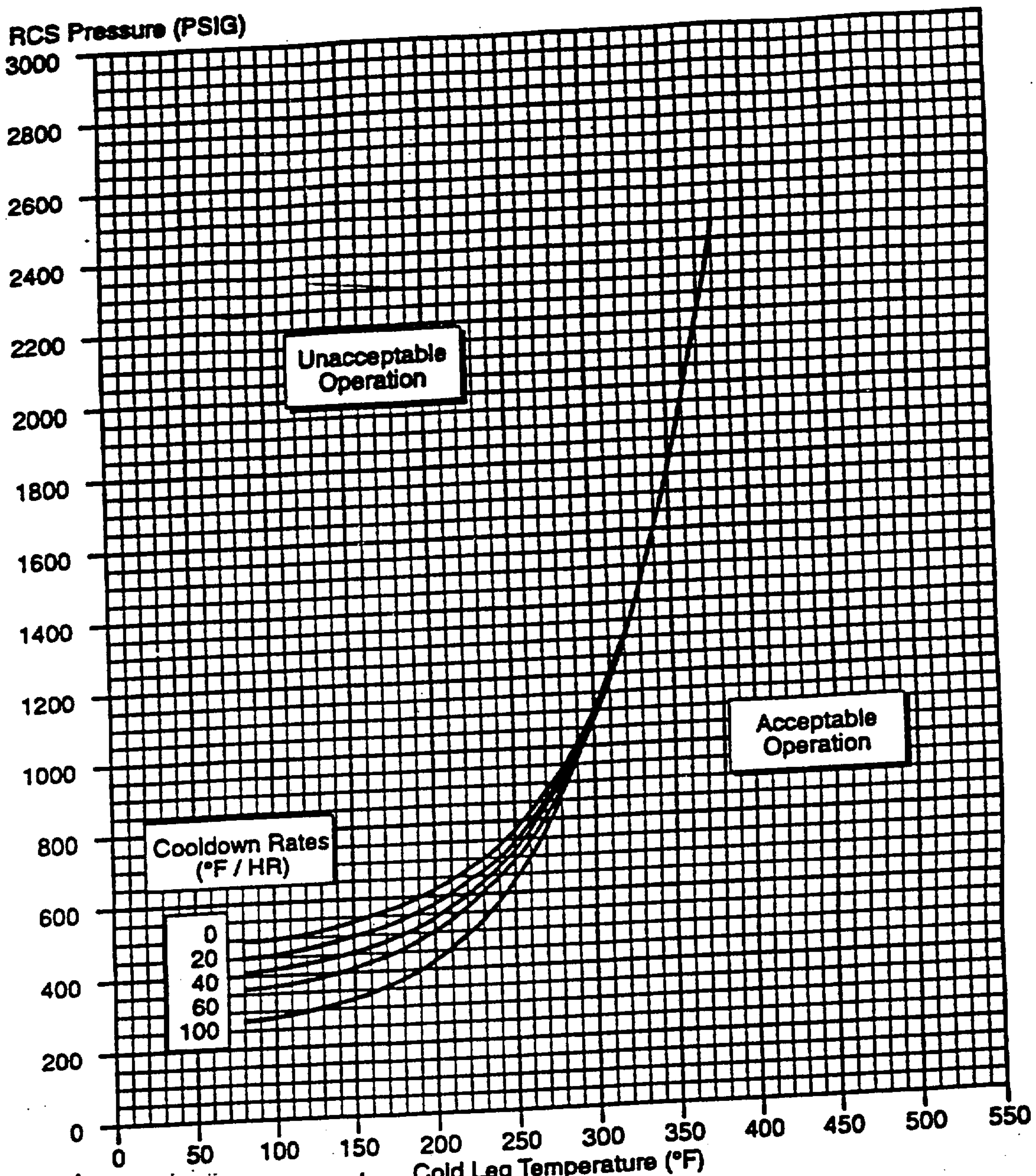
SNSOC Chairman

Date

Drawing No: WT316

Number:	Title:	Revision:
F-4	INTEGRITY	2

Figure 2
RCS COOLDOWN RESTRICTIONS

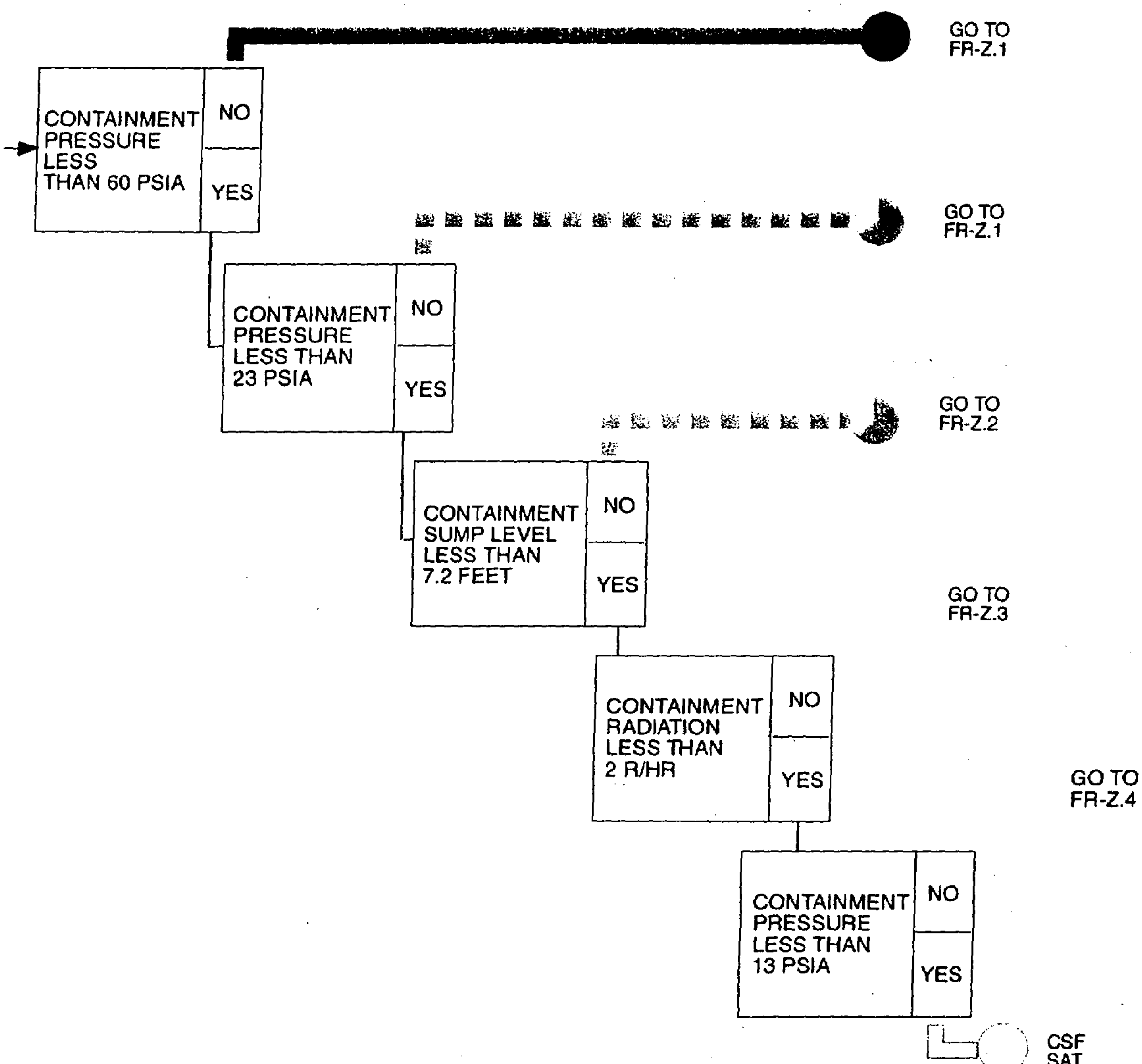


Graphics No. C81300

TBSowers
SNSOC Chairman

1/25/96
Date

Number:	Title:	Revision:
F-5	CONTAINMENT	Rev-1A

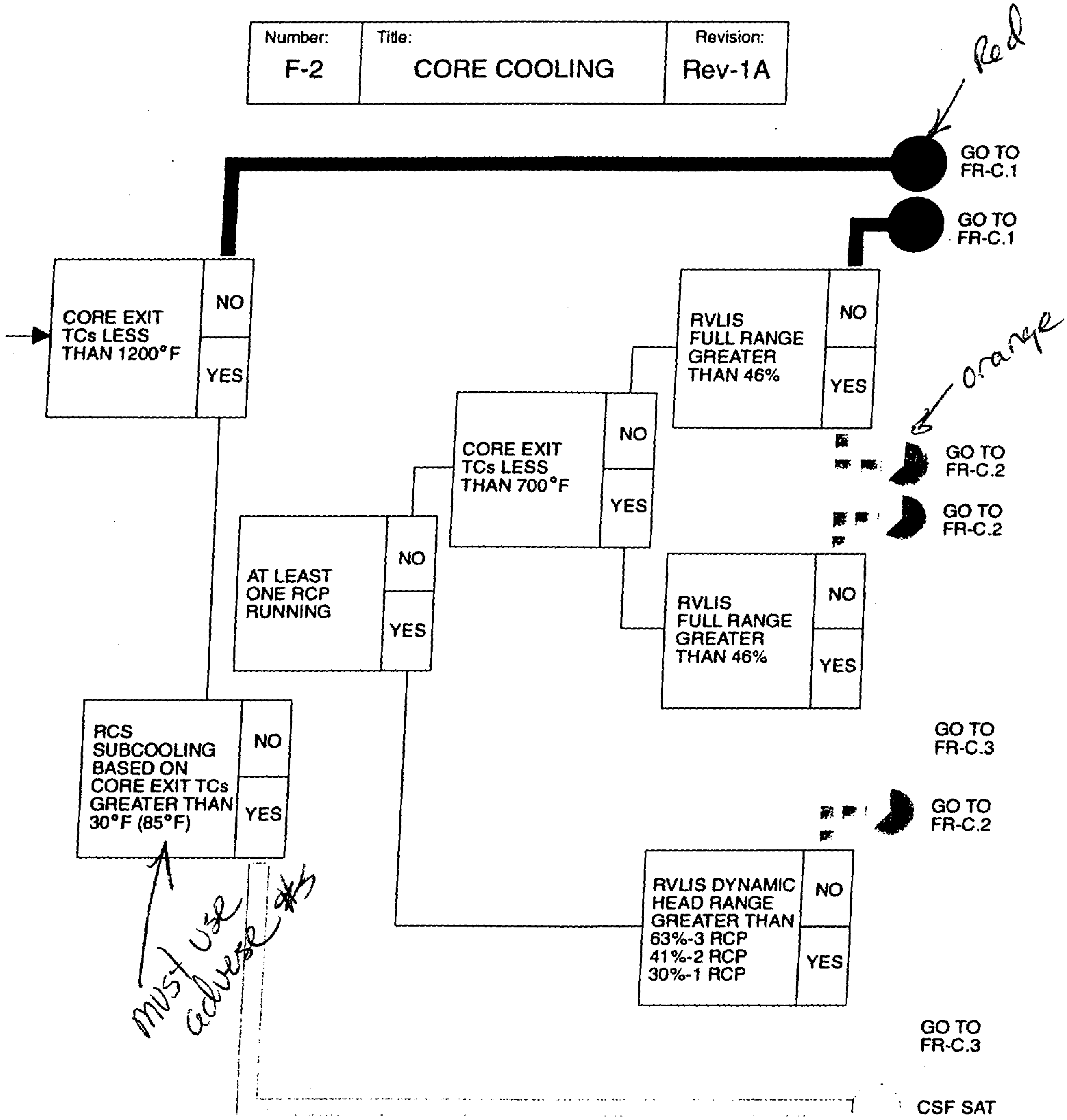


Drawing No. CB382

APPROVAL ON FILE
SNSOC CHAIRMAN

DATE

Number:	Title:	Revision:
F-2	CORE COOLING	Rev-1A



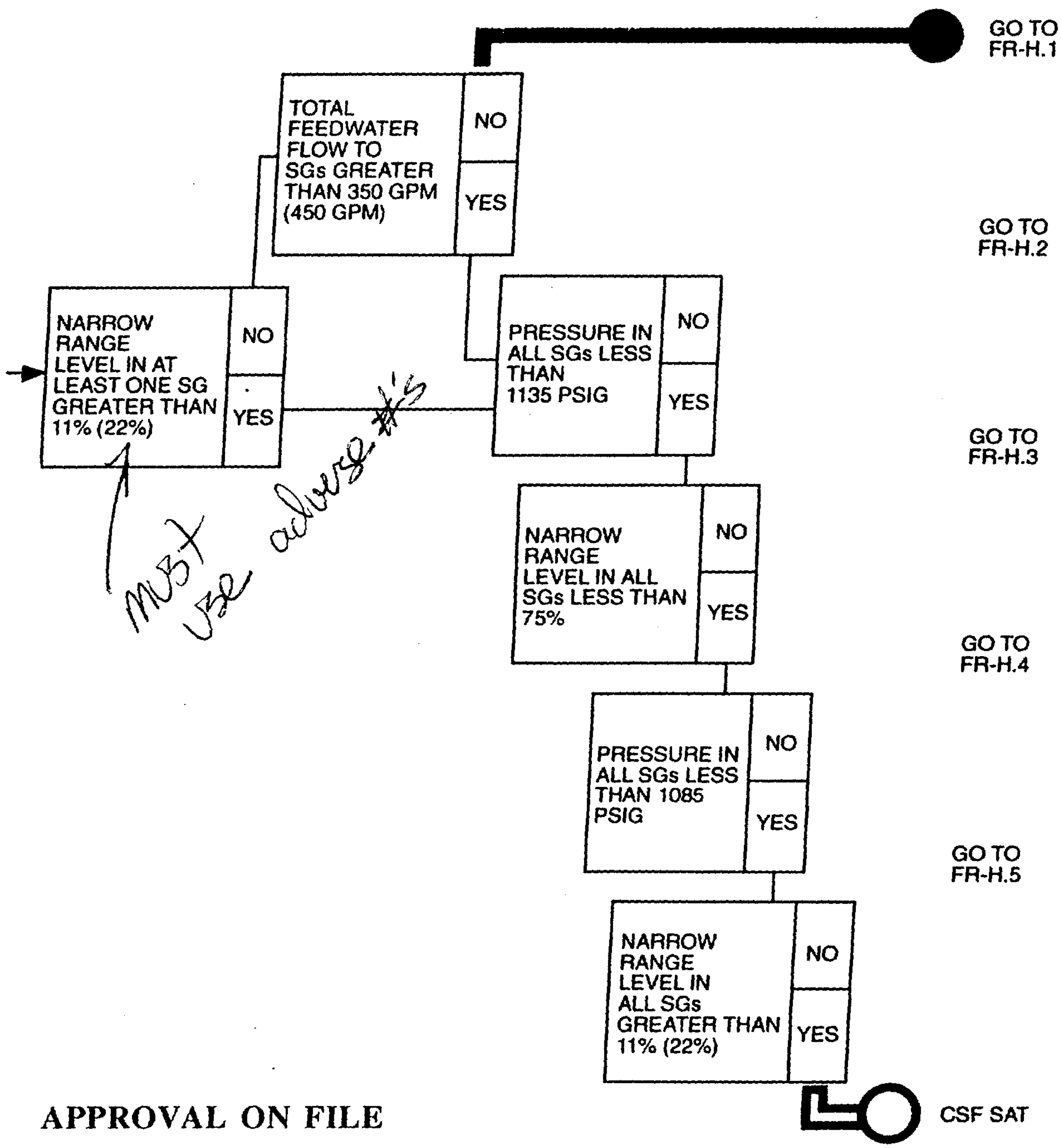
APPROVAL ON FILE

Drawing No. CB:380

SNSOC CHAIRMAN DATE

Number:	Title:	Revision:
F-3	HEAT SINK	4

Rev

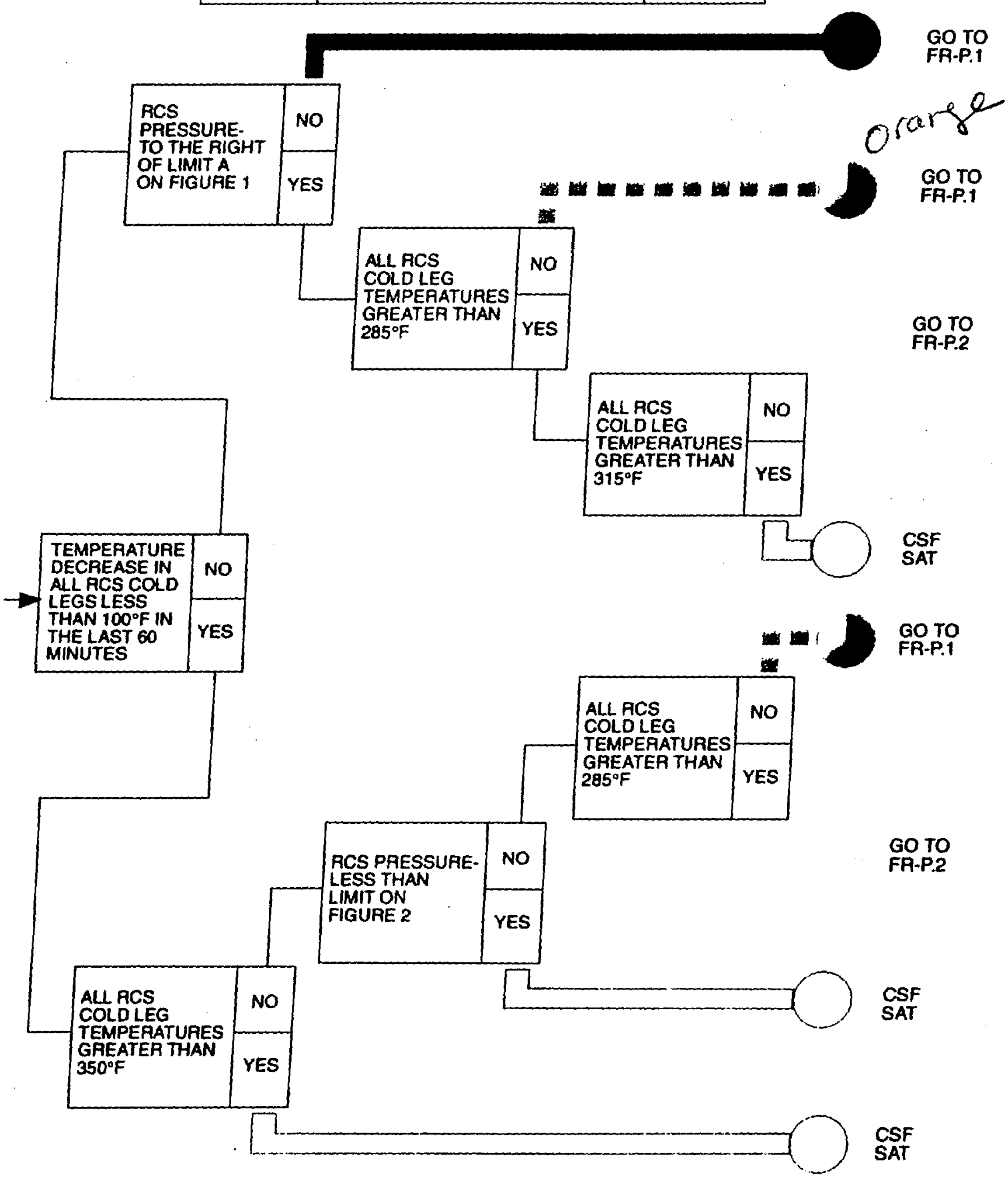


APPROVAL ON FILE

SNSOC CHAIRMAN DATE

Graphics No. CB381

Number:	Title:	Revision:
F-4	INTEGRITY	2



APPROVAL ON FILE

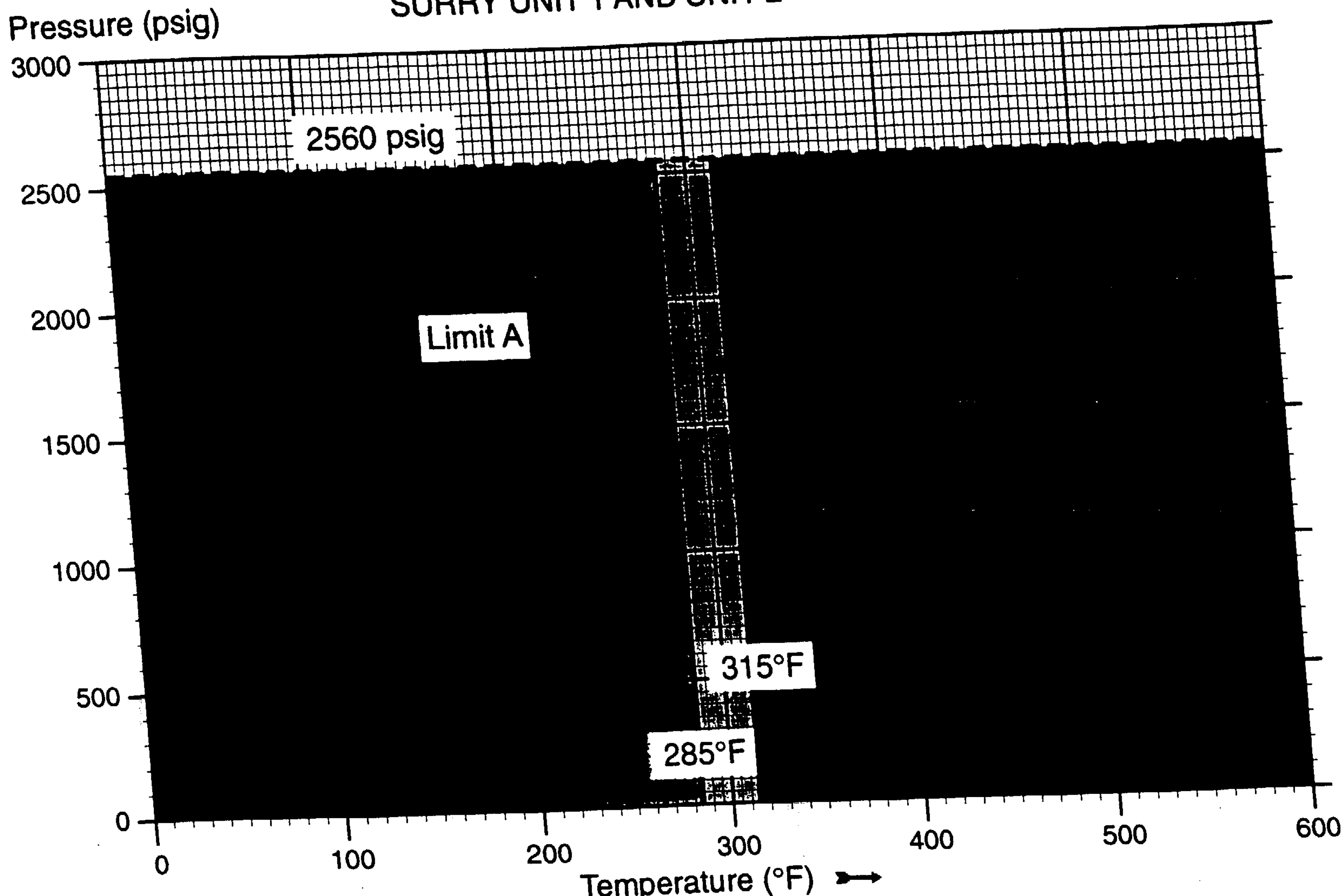
Graphics No. CB383

SNSOC CHAIRMAN

DATE

Number:	Title:	Revision:
F-4	INTEGRITY	2

FIGURE 1 - OPERATIONAL LIMITS CURVE
SURRY UNIT 1 AND UNIT 2



APPROVAL ON FILE

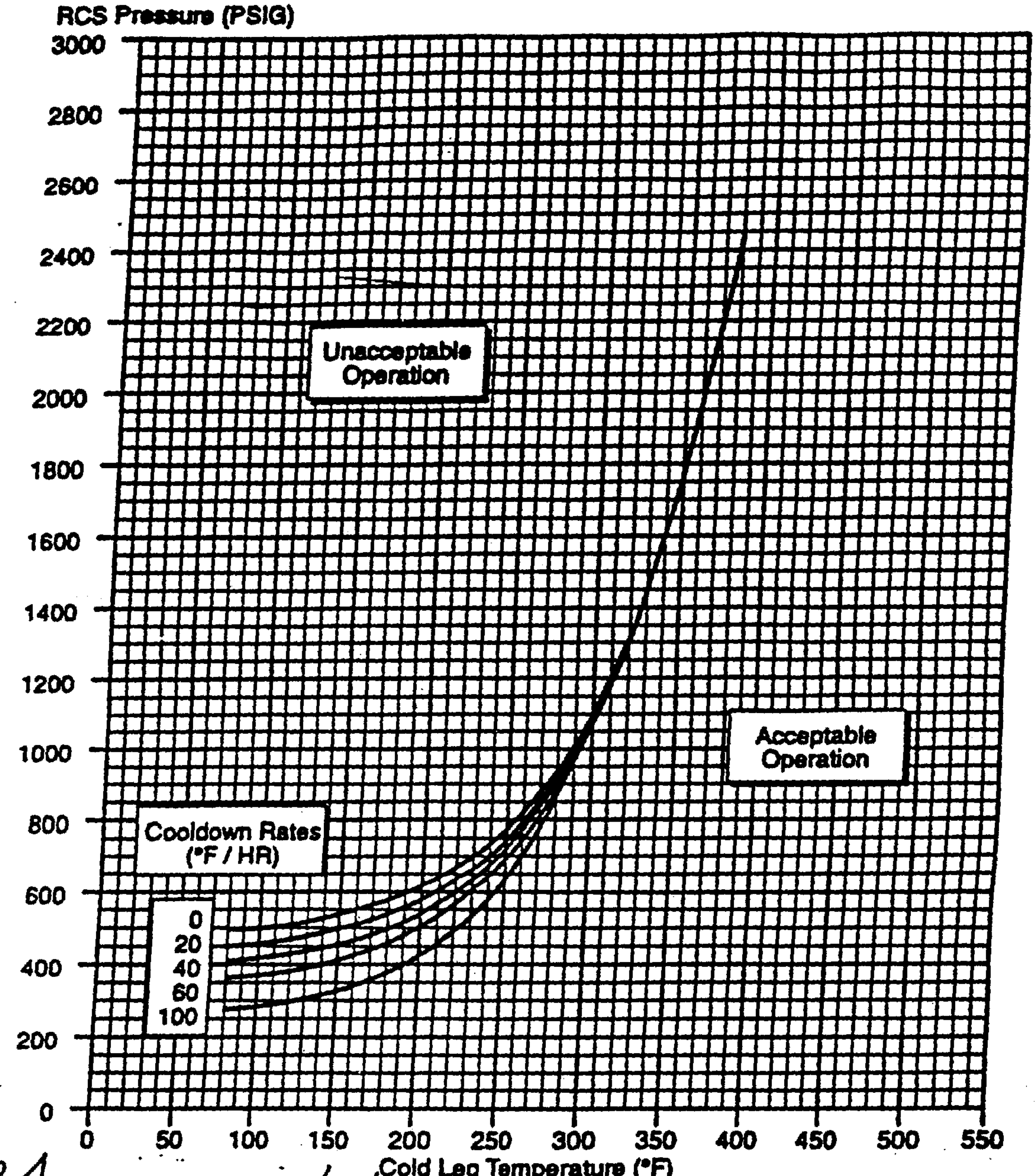
SNSOC Chairman

Date

Drawing No: WT316

Number:	Title:	Revision:
F-4	INTEGRITY	2

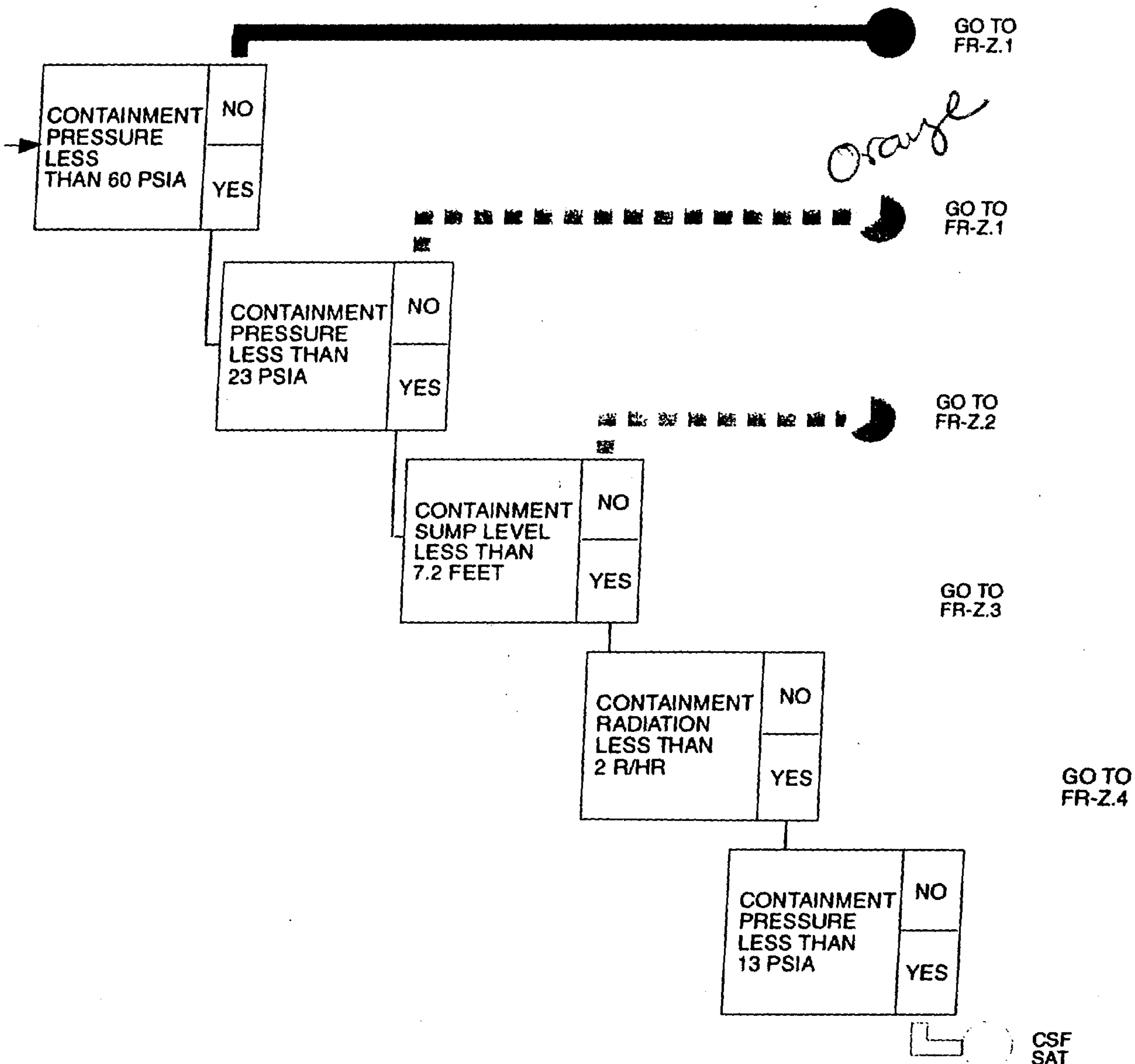
Figure 2
RCS COOLDOWN RESTRICTIONS



TBSowers
SNSOC Chairman

1/25/96
Date

Number:	Title:	Revision:
F-5	CONTAINMENT	Rev-1A



APPROVAL ON FILE

SNSOC CHAIRMAN

DATE

Drawing No. CB382

u. The control room team may take actions necessary to control system parameters within their established operating range (e.g., controlling AFW flow to maintain SG levels within the proper band, adjusting PRZR master controller to maintain RCS pressure, or maintaining RCS Tave at 547·)

6.4.5 Critical Safety Function (CSF) Status Trees

CSF Status Trees are normally monitored by the STA. In the event the STA is unavailable, a licensed Operator, as directed by the SRO in charge of Control Room activities, may be assigned to monitor the CSF Status Trees until the STA arrives.

a. Monitoring of the CSF Status Trees shall begin when directed by E-0 or when a transition is made from E-0 and any appropriate FR shall be implemented upon transition from E-0 to any other procedure or when directed by E-0.

b. Monitoring of CSF Status Trees shall be maintained upon transition either to or from any procedure. The only exception to this is when any EOP specifically directs not to perform a FR procedure.

c. The CSF Status Trees have different rules of usage than EOPs and are monitored in parallel with the performance of the EOPs.

d. CSF Status Tree monitoring shall be continuous when a RED or ORANGE terminus is encountered.

e. The SRO in charge of the Control Room activities shall be immediately informed if a RED or ORANGE terminus exists. The SRO in charge of the Control Room activities shall also be regularly advised of YELLOW or GREEN plant conditions.

f. The CSF Status Tree shall be entered at the left side of the tree and each question of the tree branch shall be answered based on the existing Unit conditions.

g. Each CSF Status Tree shall be monitored to completion at the tree's terminus.

h. The appearance of a RED or ORANGE path CSF Status Tree usually implies that some Unit equipment is not available or is significantly degraded.

i. If a RED path terminus is encountered, the Recovery Procedure in progress shall be stopped immediately and the Functional Restoration Procedure (FR) required by the RED path terminus performed except in ECA-0.0, the first several steps of ECA-0.1 and 0.2, during ES-1.3, and when ECA-2.1 or other FRs create RED path conditions and procedural transition is not appropriate.

j. If an E, ES, or ECA series EOP is suspended to perform a RED or ORANGE path Functional Restoration Procedure (FR), Operator judgement is required in subsequent procedures to avoid inadvertent reinstatement of a RED or ORANGE path by undoing a critical step of the original FR.

Priority listed by F1 F2 F3 F4 F5

k. If a RED path of higher priority arises during the performance of any RED path FR, then the higher priority path should be addressed first and the lower priority RED path FR suspended unless specifically addressed by the original FR (e.g., FR-C.2, Caution before step 11).

l. If an ORANGE path is encountered, the remaining CSF Status Trees shall be monitored. If a RED path is not encountered, the Recovery Procedures in progress shall be suspended and the FR required by the ORANGE path shall be performed.

m. If a RED path or higher priority ORANGE path arises during the performance of any ORANGE path FR, then the RED path or higher priority ORANGE path should be addressed first and the lower priority ORANGE path FR suspended unless specifically addressed by the original FR (e.g., FR-C.2, Caution before step 11).

n. FRs entered from a RED or ORANGE path shall be performed to completion unless pre-empted by a higher priority path or a loss of all AC power.

o. A YELLOW terminus does not require immediate Operator attention. It is frequently indicative of an off-normal or temporary condition which will be restored to normal by actions that are already in progress.

   1. A YELLOW path may provide an early indication of a developing RED or ORANGE path condition.

   2. For YELLOW path conditions only, the cognizant SRO may decide whether or not to implement any YELLOW path condition FRs. Unit conditions should be evaluated to determine if implementation of the FR is appropriate.

p. After restoration of the CSF Status Trees from a RED or ORANGE path condition, recovery actions may continue when the FR is completed. Usually the FR will return the Operator to the procedure and step in effect. The Operator may be directed to another procedure due to conditions created during performance of the FR.

q. The Recovery Procedures are optimal assuming that equipment is available as required for safety. Some adjustments may be required to the Recovery Procedures because of certain equipment failures.

r. If loss of AC power to the emergency busses occurs and ECA performance begins, none of the FRs can be implemented because none of the electrically powered equipment used to restore a Critical Safety Function will be operable.

1. A NOTE before Step 1 of ECA-0.0 states "CSFs should be monitored for information only. FRs should not be implemented."

2. After Step 5 of ECA-0.0, the Operating Team shall continue performance of ECA-0.0 and transition to ECA-0.1 or ECA-0.2 **if** power is restored to an Emergency Bus. This is necessary because actions taken in ECA-0.0 after Step 5 must be carefully undone to prevent further damage to Station equipment. FRs will not apply until specified in ECA-0.1 or ECA-0.2

s. Certain procedures (e.g., ESs and ECAs) take precedence over the FRs. Typically, a NOTE before Step 1 of the FR will notify the Operator not to implement the FR under specific conditions.

6.4.6 **Changes to EOPs**

EOPs shall be revised in accordance with VPAP-0506, EOP Development, Revision, and Maintenance.

6.5 Abnormal Procedure (AP) Usage

6.5.1 The Immediate Action steps of APs shall be committed to memory so that the APs do not have to be directly referenced when the condition requiring usage occurs.

6.5.2 The Immediate Action steps of APs should be reviewed after performance to ensure required actions have been taken.

6.5.3 Subsequent Action steps of the APs shall be completed as listed in the APs.

QUESTION 76: (1.0)

Given the following plant conditions:

- The team is responding to a loss of both emergency busses.
- Neither bus could be re-energized and all required equipment is in Pull-to-Lock.
- S/G depressurization resulted in automatic actuation of safety injection.

The team is directed to reset the SI signal to __________.

a. enable securing equipment that automatically started

b. enable securing equipment that was manually started

c. prevent equipment from automatically starting when an emergency bus is restored

d. prevent equipment from automatically starting when it is returned to Auto

ANSWER: d

[RO: Tier 1/Group 1]

Answer correct: per 1-ECA-0.0, SI is reset to allow manual loading of equipment on a recovered emergency bus.	Distractors plausible: a – this is the reason SI is reset elsewhere in the EOPs; b – if any emergency bus equipment **had** been manually started, the SI signal would prevent it from being secured; c – if an SI signal were present, any emergency bus equipment in auto-standby would auto-start.	Distractors incorrect: a – all emergency bus powered equipment was placed in PULL-TO-LOCK prior to S/G depressurization, so none should auto-start when SI actuates; b – since both emergency busses are de-energized and all equipment is in PULL-TO-LOCK, none will be running; c – equipment will not auto-start when the emergency bus is restored because all of the control switches are in PULL-TO-LOCK.
K/A: APE055-EK3.02	Objective: 2934	Source: New
Reference: 1-ECA-0.0; WOG B/G document for ECA-0.0, ND-95.3-LP-17	Level: Knowledge	

b. The SI signal is **<u>reset to permit manual loading of equipment on the AC emergency bus following AC power restoration (rk)</u>**. This provides bus overload protection in case power is restored from a low capacity power supply.

39. **<u>STEP 24</u>: CHECK SI SIGNAL STATUS.**

a. The purpose of this step is to check if an SI signal exists.

b. The secondary depressurization initiated previously will result in SI actuation, if not already actuated, on low pressurizer pressure or another signal. The team should check SI actuation status and reset SI as soon as the reset delay time has expired.

(1) This reset action is consistent with the guideline philosophy of defeating automatic loading of the emergency bus upon AC power restoration.

(2) Resetting SI will permit the team to manually load SI equipment as instructed.

c. If SI has not been actuated, the team will go to step 28, skipping the steps that verify containment isolation. If SI is subsequently actuated, the team should return to steps 24b through 27 to verify containment isolation. Step 24a) RNO is a type of Continuous Action Step.

40. **<u>STEP 25</u>: VERIFY CONTAINMENT ISOLATION VALVES CLOSED.**

a. The purpose of this step is to verify isolation of phase I containment penetrations.

QUESTION 77: (1.0)

With 1-RC-LT-1461 aligned to the upper channel of Pressurizer Level Control, the transmitter develops a 0.5 gpm reference leg leak while at power with all control systems in Automatic.

Which ONE of the following identifies system response?

a. Pressurizer level continually increases to a water solid condition.

b. Pressurizer level decreases until letdown isolates, then increases.

c. Pressurizer level continually decreases to the Low Pressurizer Pressure reactor trip setpoint.

d. Pressurizer level continually increases to the High Pressurizer Level reactor trip setpoint.

ANSWER: b

[RO: Tier 1/Group 2]

Answer correct: The controlling channel failing high causes charging flow to decrease to less than letdown flow. The lower pressurizer level channel will see the decrease and isolate letdown . Charging will now exceed letdown causing level to increase.	Distractors plausible: A, D - Level will increase after first decreasing. C - Lowering pressure level causes RCS pressure to decrease. ALL: must determine effects of reference leg leak and integrate that comprehension with knowledge of the charging system.	Distractors incorrect: A, D - The controlling channel indication fails high causing actual level to initially decrease. C – Pressurizer Level does not decrease enough to cause a substantial loss of pressurizer pressure.
K/A: APE008.AA2.10	Objective: 2661	Source: NEW
Reference: ND-93.3-LP-7, ND-93.1-LP-1	Level: Comprehension	

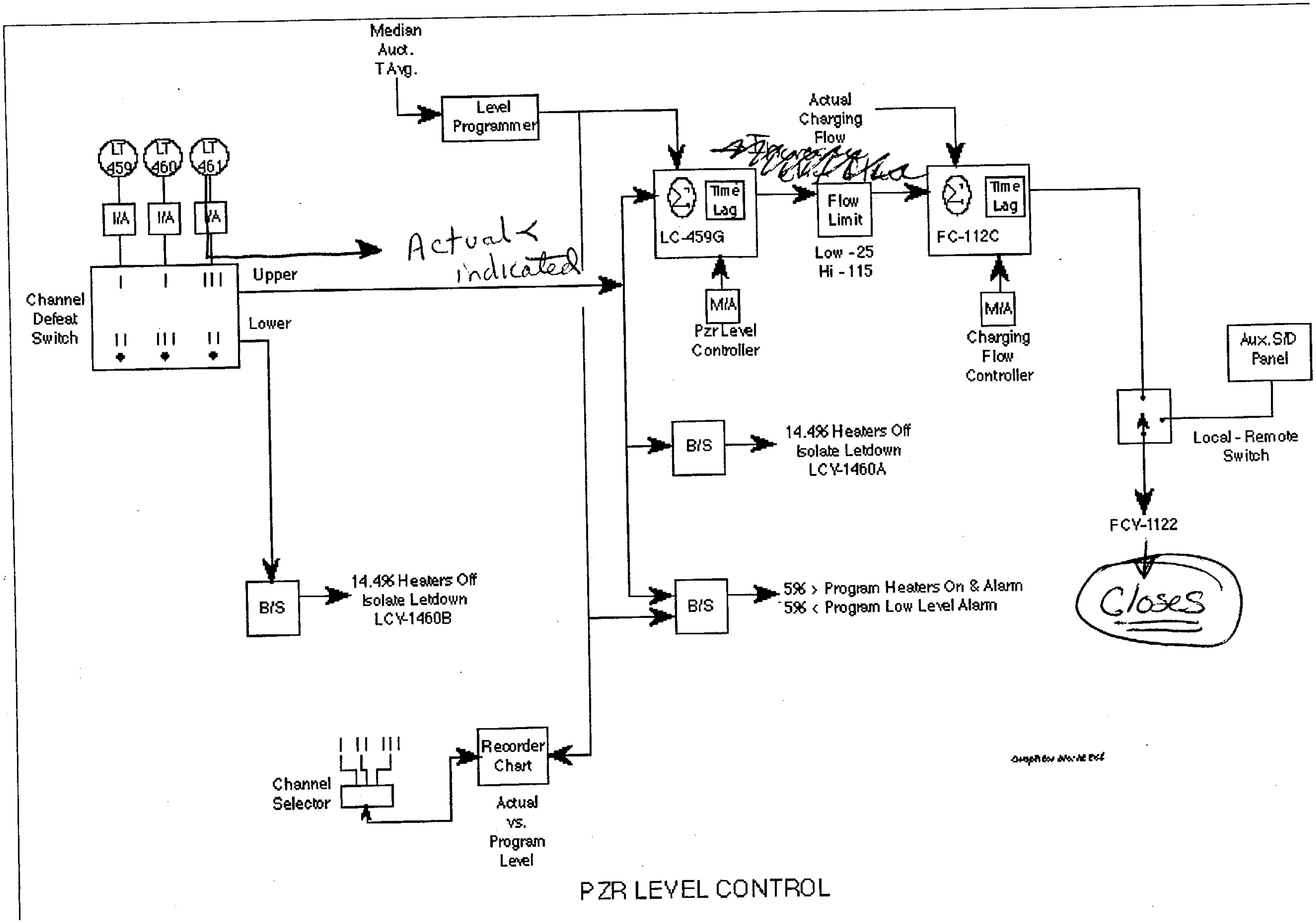
(e) There are several errors common to wet leg type systems:

1) Reference leg temperature. Since the system must use a condensing pot, the temperature of the fluid in the reference leg can be different than the temperature in the tank. If the system cannot use a condensing pot, the reference leg can still be susceptible to heating by ambient temperature changes. If this temperature should increase, the density of the reference leg would decrease, thus decreasing the pressure on the Low side of the D/P Transmitter causing an indicated level greater than actual.

2) Reference leg flashing. In hot systems that use condensing pots, if the tank pressure should rapidly decrease, the reference leg could partially or fully flash off. This would decrease the pressure on the Low side of the D/P transmitter causing indicated level to be much greater than actual. Hydrogen could also come rapidly out of solution removing water from the reference leg and causing a similar effect.

3) For a reference leg leak, the leak would decrease the pressure on the Low side of the D/P transmitter thus indicated level will be greater than actual (e.g., ~6% increase for the pressurizer transmitters on a partial loss of the reference leg).

4) For a variable leg leak, the pressure on the High side of the D/P transmitter would increase, thus the indicated level will be less than actual.



PZR LEVEL CONTROL

QUESTION 78: (1.0)

Which ONE of the following explains why a **negative** number may be displayed on the ICCM subcooled margin monitor during a large-break LOCA?

a. The ICCM is not qualified for adverse containment conditions.

b. The core exit thermocouples are invalid in a steam environment.

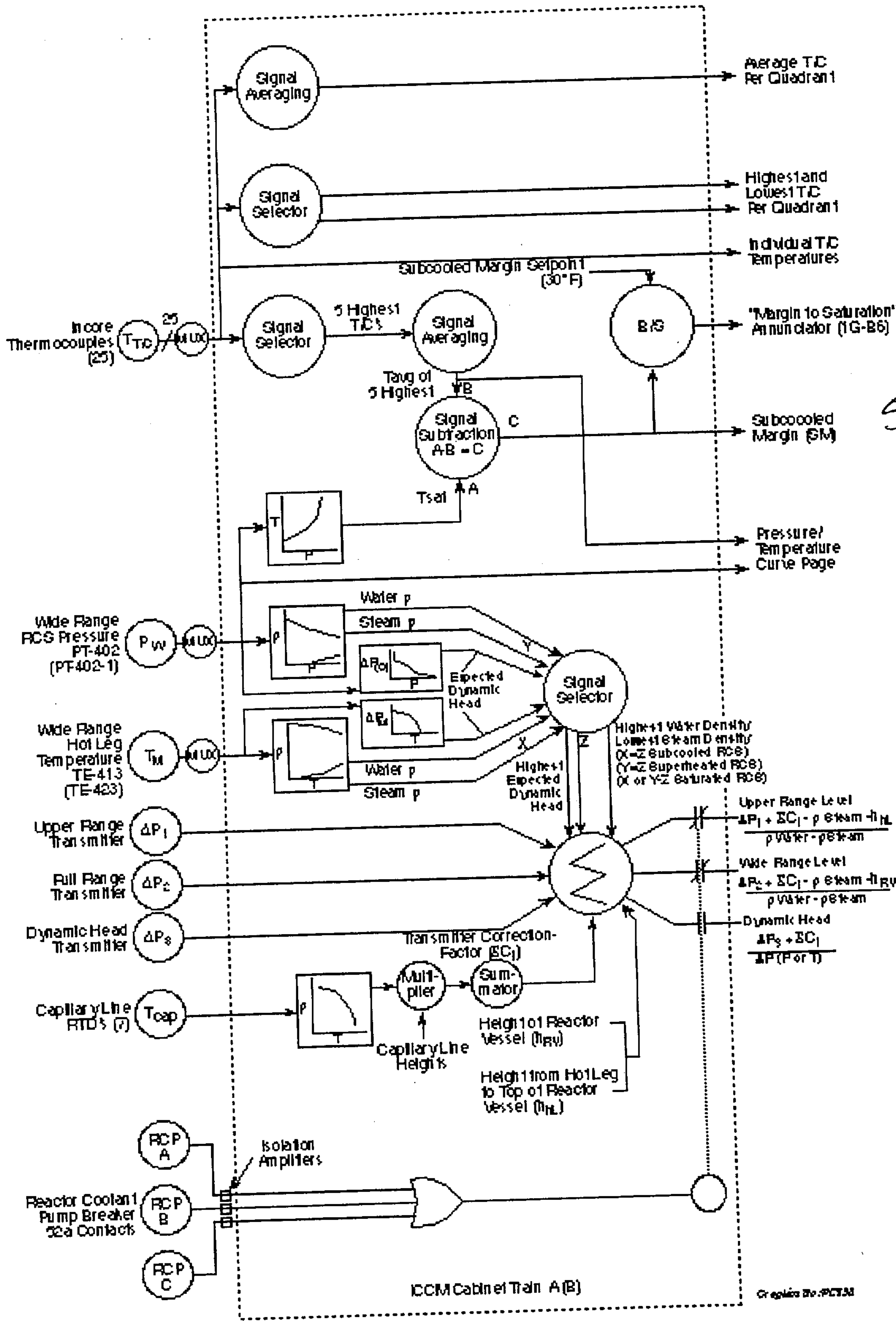
c. The calculated value is outside its normal range of indication.

d. The number of degrees of superheat is preceded by a negative sign.

ANSWER: d

[RO: Tier 1/Group 2]

Answer correct: margin to saturation for subcooled conditions is a positive number; margin to saturation for superheated conditions is a negative number.	Distractors plausible: a – certain instruments are not qualified for post-accident conditions; b – the CETCs are only valid up to a certain temperature; c – when certain instruments monitored by the plant computer are outside their normal range, they display a slightly negative number.	Distractors wrong: a – the ICCM is qualified for adverse containment conditions; b – the CETCs are designed to operate in a steam environment; c – the negative sign does NOT indicate that the calculated value is outside its normal range of indication.
K/A: EPE011-EA1.14	Objective: 2691	Source: New
Reference: ND-93.4-LP-3	Level: Knowledge	



Shows actual subcooling or Superheating

ICCM FUNCTIONAL BLOCK DIAGRAM

QUESTION 79: (1.0)

An unisolable RCS leak exists in the Auxiliary Building Basement.

Which ONE of the following describes the expected procedure transitions?

a. E-0, Reactor Trip or Safety Injection, to E-1, Loss of Reactor or Secondary Coolant, to ECA-1.2, LOCA outside Containment, to ECA-1.1, Loss of Emergency Coolant Recirculation.

b. E-0, Reactor Trip or Safety Injection, to E-1, Loss of Reactor or Secondary Coolant, to ECA-1.1, Loss of Emergency Coolant Recirculation.

c. E-0, Reactor Trip or Safety Injection, to E-1, Loss of Reactor or Secondary Coolant, to ECA-1.2, LOCA outside Containment to ES-1.2, Post LOCA cooldown and depressurization.

d. E-0, Reactor Trip or Safety Injection, to ECA-1.2, LOCA outside Containment, to ECA-1.1, Loss of Emergency Coolant Recirculation.

ANSWER: d

[RO: Tier 1/Group 2]

Answer correct:	Distractors plausible:	Distractors incorrect:
This is the correct procedure flow path to determine where the break is outside containment and then to attempt to isolate it.	a. diagnostic steps in E-0 have displayed an error in transitioning to E-1 for a LOCA outside CTMT on the step for a RCS leak INSIDE containment.	ALL: incorrect transitions.
K/A: EPE.E11.EA2.1	Objective: 3035	Source: New
Reference: E-0, ECA-1.2,1.1, ND-95.3-LP-3	Level: Knowledge	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
26.	__CHECK FOR OUTSIDE CONTAINMENT INVENTORY LOSS: a) Radiation Monitors - NORMAL • Kaman vent-vent • Auxiliary Building Control Area b) Sump annunciators - NOT LIT • VSP-F-4 • B-D-1 • B-D-2 • B-F-3	Determine cause of abnormal conditions. IF the cause is a loss of RCS inventory outside CTMT, THEN GO TO 1-ECA-1.2, LOCA OUTSIDE CONTAINMENT.
*27.	__CHECK SG LEVELS:	
	a) Any narrow range level - GREATER THAN 11%	a) Maintain total feed flow greater than 350 gpm until narrow range level in at least one SG is greater than 11%.
	b) Check emergency buses - BOTH ENERGIZED	b) Locally isolate AFW header on deenergized bus if necessary to control level: • 1-FW-141 for H bus • 1-FW-156 for H bus • 1-FW-171 for H bus OR • 1-FW-140 for J bus • 1-FW-155 for J bus • 1-FW-170 for J bus
	c) Control feed flow to maintain narrow range level between 11% and 50%	c) IF narrow range level in any SG continues to increase in an uncontrolled manner, THEN GO TO 1-E-3, STEAM GENERATOR TUBE RUPTURE.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	__TRY TO IDENTIFY AND ISOLATE BREAK:	
	a) Close LHSI to cold legs • 1-SI-MOV-1890C	
	b) Check RCS pressure - INCREASING	b) Open 1-SI-MOV-1890C AND continue efforts to identify and isolate leakage: • Check for annunciators: 1) AUX Building Sump HI Level (VSP-F-4) 2) RS Pit A Hi Level (B-D-1) 3) RS Pit B Hi Level (B-D-2) 4) SFGDS Area Sump Hi Level (B-F-3) • Locally check auxiliary building and safeguards recirculation loops for leakage. • Notify TSC of any damage control needs. • GO TO 1-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.
	c) Place LHSI pumps in PTL	
	d) Close LHSI pump suctions from RWST • 1-SI-MOV-1862A • 1-SI-MOV-1862B	
	e) GO TO 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT	

- END -

QUESTION 80: (1.0)

During the rapid RCS cooldown directed by E-3, Steam Generator Tube Rupture, steam flow is limited to 1.0×10^6 lbm/hr.

Which ONE of the following identifies the basis for this caution?

a. Prevent entry into FR-P.1.

b. Prevent restart of high head SI pumps.

c. Prevent Main Steam Isolation.

d. Prevent Condenser Air ejector discharge isolation.

ANSWER: c

[RO: Tier 1/Group 2]

Answer correct: Steam flow isolation signal not blocakble.	Distractors plausible: a. Rapid cooldowns can cause problems with PTS. b. High Steam flow SI will reinitiate SI flow. d. Header to line will perform this function, knowledge of the different protections afforded by each steam induced SI signal is required.	Distractors incorrect: a. Target temperatures set to prevent entry into the FR. Final temperatrue and not rate determines FR applicability b. High steam flow SI is blocked. d. Header to line will not occur when steaming to the condenser.
K/A: APE038.G2.4.48	Objective: 2915	Source: NEW
Reference: 1-E-3 caution prior to step 15, ND–95.3-LP-13	Level: Knowledge	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: Flow on each Main Steamline should be kept less than 1.0×10^6 PPH to prevent Main Steamline isolation during RCS cooldown with the Steam Dumps.

* *

NOTE:
- Low PRZR pressure SI signal should be blocked when PRZR pressure decreases to less than 2000 psig.
- High steam flow SI signal should be blocked when Tave decreases to less than 543°F.
- RCP trip criteria does NOT apply after initiation of an operator controlled cooldown.

15. __INITIATE RCS COOLDOWN:

a) Determine required core exit temperature (ONE TIME):

LOWEST RUPTURED SG PRESSURE (PSIG)	CORE EXIT TEMPERATURE (°F)
BETWEEN 1001 AND 1085	495 [440]
BETWEEN 901 AND 1000	485 [430]
BETWEEN 801 AND 900	470 [415]
BETWEEN 701 AND 800	455 [400]
BETWEEN 601 AND 700	440 [385]
BETWEEN 501 AND 600	420 [365]
BETWEEN 401 AND 500	400 [345]
BETWEEN 350 AND 400	385 [335]

b) Place Steam Dump Mode Select switch in Steam Pressure mode

(STEP 15 CONTINUED ON NEXT PAGE)

QUESTION 81: (1.0)

Given the following plant conditions:

- Unit 1 is at 100% power with stable T_{ave}.
- Charging flow is noted to be increasing.
- Annunciator 1D-E5, CHG PP TO REGEN HX HI-LO FLOW, has just alarmed.
- VCT level is decreasing and PRZR level is increasing.
- All other plant parameters are normal.

The conditions described are most likely due to __________ and the team should respond by __________.

a. increasing RCS leakage; entering 1-AP-16.00, Excessive RCS Leakage

b. increasing RCS leakage; isolating letdown and maximizing charging flow

c. failure of charging flow control; isolating instrument air to FCV-1122

d. failure of charging flow control; taking manual control of charging

ANSWER: d

[RO: Tier 1/Group 3]

Answer correct: for given conditions, RCS leakage is **not** increasing, most likely cause is failure of charging flow control; per 1-AR-C-C5, take manual control of FCV-1122.	Distractors plausible: a – candidate misconception concerning RCS inventory balance, AP-16 would be correct for increasing leakage; b – candidate misconception concerning RCS inventory balance, isolating letdown and maximizing charging flow would be correct for increasing leakage; c – failure of charging flow control is correct.	Distractors incorrect: a & b – RCS leakage is **not** indicated by VCT level decreasing with PRZR level increasing; c – FCV-1122 fails **open** on loss of IA
K/A: APE028-AA1.06	Objective: 2661	Source: New
Reference: 1-ARP-D-E5	Level: Comprehension	

VIRGINIA POWER
SURRY POWER STATION

ANNUNCIATOR RESPONSE PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1D-E5	CHG PP TO REGEN HX HI-LO FLOW	2
		PAGE
		1 of 3

1D-37

REFERENCES

1. UFSAR 9.1
2. 11448-ESK-10D, 10AJ
3. 1-DRP-005, Instrumentation Setpoints
4. 0-DRP-004, Precautions, Limitations and Setpoints
5. DCP 97-031, Hathaway Sys |2

PROBABLE CAUSE

1. Alarm actuates when FC-CH-122B-1 senses Charging flow greater than or equal to 110 gpm.

   High Charging flow may be caused by one or more of the following:

   - Low Pressurizer level.
   - RCS contraction due to TAVE decrease.
   - Primary System leak.
   - Flow control valve 1-CH-FCV-1122 in manual and high flow rate has been set.
   - Malfunction of 1-CH-FCV-1122 in AUTO.

2. Alarm actuates when FC-CH-122B-2 senses Charging flow less than or equal to 30 gpm.

   Low Charging flow may be caused by one or more of the following:

   - High Pressurizer level.
   - RCS expansion due to TAVE increase.
   - Flow control valve 1-CH-FCV-1122 in manual and low flow rate has been set.
   - Malfunction of 1-CH-FCV-1122 in AUTO.

3. Instrumentation failure has occurred.

APPROVAL RECOMMENDED	APPROVED	DATE
REVIEWED	CHAIRMAN STATION NUCLEAR SAFETY AND OPERATING COMMITTEE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	__VERIFY ALARM ON 1-CH-FI-1122A: • LESS THAN OR EQUAL TO 30 GPM <u>OR</u> • GREATER THAN OR EQUAL TO 110 GPM	Initiate a Work Request and GO TO Step 8.
2.	__CHECK CHG FLOW - ANY OF THE FOLLOWING CONDITIONS EXIST • 1-CH-FCV-1122 in MANUAL • CHG line isolated for maintenence • CHG line isolated due to operator action • Safety Injection	GO TO Step 4.
3.	__RETURN TO PROCEDURE IN EFFECT	
4.	__CHECK ACTUAL PRZR LEVEL AGAINST SETPOINT - DEVIATION EXISTS DUE TO TAVE-TREF MISMATCH	GO TO Step 6.
5.	__RETURN TAVE TO SETPOINT USING ANY OF THE FOLLOWING: • Rods • Boration • Dilution • Steam Demand	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	__CHECK 1-CH-FCV-1122 FOR PROPER OPERATION • Demand - NORMAL OUTPUT AND STABLE • CHG PP AMPS - STABLE • CHG Flow - STABLE	IF 1-CH-FCV-1122 shows abnormal operation, THEN do the following: a) Place 1-CH-FCV-1122 in MANUAL. b) Attempt to restore CHG flow. c) Locally investigate cause. d) Initiate a Work Request.
7.	__VERIFY CHG FLOW - NORMAL	Do the following: a) Place 1-CH-FCV-1122 in MANUAL. b) Initiate a Work Request as necessary. c) IF a total loss of CHG flow has occurred, THEN GO TO 1-AP-8.00, LOSS OF NORMAL CHARGING FLOW.
8.	__NOTIFY SHIFT SUPERVISOR	

- END -

QUESTION 82: (1.0)

The Unit 1 Discharge tunnel Radiation monitor is currently reading 1.2E1.

What radiation units are being monitored?

a. mr/hr.

b. CPM.

c. DPM.

d. R/hr.

ANSWER: b

[RO: Tier 2/Group 1]

Answer correct: These are the unit for this radiation monitor.	Distractors plausible: a – these units are used for area radiation monitors. c – Disintegration's per minute is a fundamental unit of radioactive decay. D – these units are used for the containment high range monitors.	Distractors incorrect: a/c/d – these units are not used for this radiation monitor.
K/A: SYS068.K5.03	Objective: 2727	Source: NEW
Reference: ND-93.5-LP-1	Level: Knowledge	

process monitor Area monitor Monitor

(1) Digital readout - Provides indication to ~~two~~ decimal ~~places~~ times ten raised to some power. Will read out in CPM or mR. Can also provide detector voltage and rate of change of activity.

(2) Bargraph - Provides indication in CPM or mR. The bargraph indicator is green when all conditions are normal. When off-normal conditions occur, the bargraph display will change as described in the following sections.

b. ON/OFF pushbutton - Turns detector and circuitry on or off.

c. Alarm status indicators - Will light up when any of the five alarm conditions exist.

(1) High alarm - Will give the alarm whenever the setpoint is reached. The high alarm light will come on and the annunciator will alarm. The bargraph color will change to red. The alarm will flash until the alarm acknowledge pushbutton is pushed. The high alarm setpoint can be checked by pushing the HIGH pushbutton.

(2) Warning alarm - Will give the alarm whenever the setpoint is reached. The warning alarm light will come on and the annunciator will alarm. The bargraph color will change to amber. The alarm will flash until the alarm acknowledge pushbutton is pushed. The warning alarm setpoint can be checked by pushing the WARN pushbutton.

(3) Fail alarm

QUESTION 83: (1.0)

Which ONE of the following containment pressures is the maximum that will allow manual reset of the Hi-Hi CLS System once actuated?

a. < 23.0 psia.

b. < 17.7 psia.

c. < 14.2 psia.

d. < 10.3 psia.

ANSWER: c

[RO: Tier 2/Group 1]

Answer correct: The reset for Hi Hi CLS is from the reset of Hi CLS at 14.2 psia.	Distractors plausible: A – This would be less than the actuation setpoint which would be the logical reset point. B – This is the actuation setpoint for Hi CLS which could be confused with the reset point. D – This is a valid number for Containment Pressure in Technical Specifications.	Distractors incorrect: A – ≥ 23.0 psia is the actuation setpoint of Hi Hi CLS. B - ≥ 17.7 psia is the actuation setpoint for Hi CLS. D – This is the maximum allowable Containment Air Partial Pressure by Technical Specifications.
K/A: SYS013.K5.02	Objective: 2321	Source: New
Reference: ND-91-LP-5	Level: Knowledge	

multiplying relays (MRs) energized. The matrix formed by the energized relays maintain the output relays in CLS train A and train B energized.

f. When containment total pressure reaches 17.7 psia (3.0 psig), the A/D switch opens and deenergizes the MRs.

The alarm CLS HIGH CONTAINMENT PRESSURE CHANNEL 1 (E-E-1) annunciates. If this pressure is sensed on three out of four channels, the "3/4 matrix 1A/1B" is deenergized and the CLS HIGH TRAIN A and/or TRAIN B alarm (B-B-4 and B-B-5) annunciate. Along with the alarm, all of the output relays are deenergized and the HI CLS functions for that train actuate.

g. The output of PT-LM-100A is also sent to another A/D converter. This converter is associated with the HI-HI CLS subsystem. Under normal conditions this comparator switch is open and the associated relays are de-energized.

h. If containment pressure rises to 23.0 psia (8.3 psig), the CLS HIGH-HIGH CONTAINMENT PRESSURE CHANNEL 1 alarm annunciates (E-A-1). If 3/4 channels rise to 23.0 psia, the "3/4 matrix 2A/2B" is energized and the CLS HIGH-HIGH TRAIN A and/or TRAIN B alarm (B-C-4 and B-C-5) annunciate.

i. CLS HI and HI-HI can be manually reset when 2/4 channels go below 14.2 psia (-0.5 psig). The CLS TRAIN A (B) RESET PERMISSIVE annunciator alarms (BD4 & 5) when 2/4 channels reach 14.2 psia. The CLS reset pushbuttons must be simultaneously depressed after receipt of this alarm in order to enable securing of CLS functions.

j. In order to be able to reset HI-HI CLS functions, both HI and HI-HI CLS must be reset (reset both when <14.2 psia). In other words, cannot reset HI-

QUESTION 84: (1.0)

Unit 1 is operating at 100% power with "D" bank rods at 218 steps when an electrical failure deenergizes vital bus I-III. You have noticed that the rods cannot be withdrawn.

Which ONE of the following is preventing rod motion?

a. Intermediate range high flux rod stop.

b. Power range high flux rod stop.

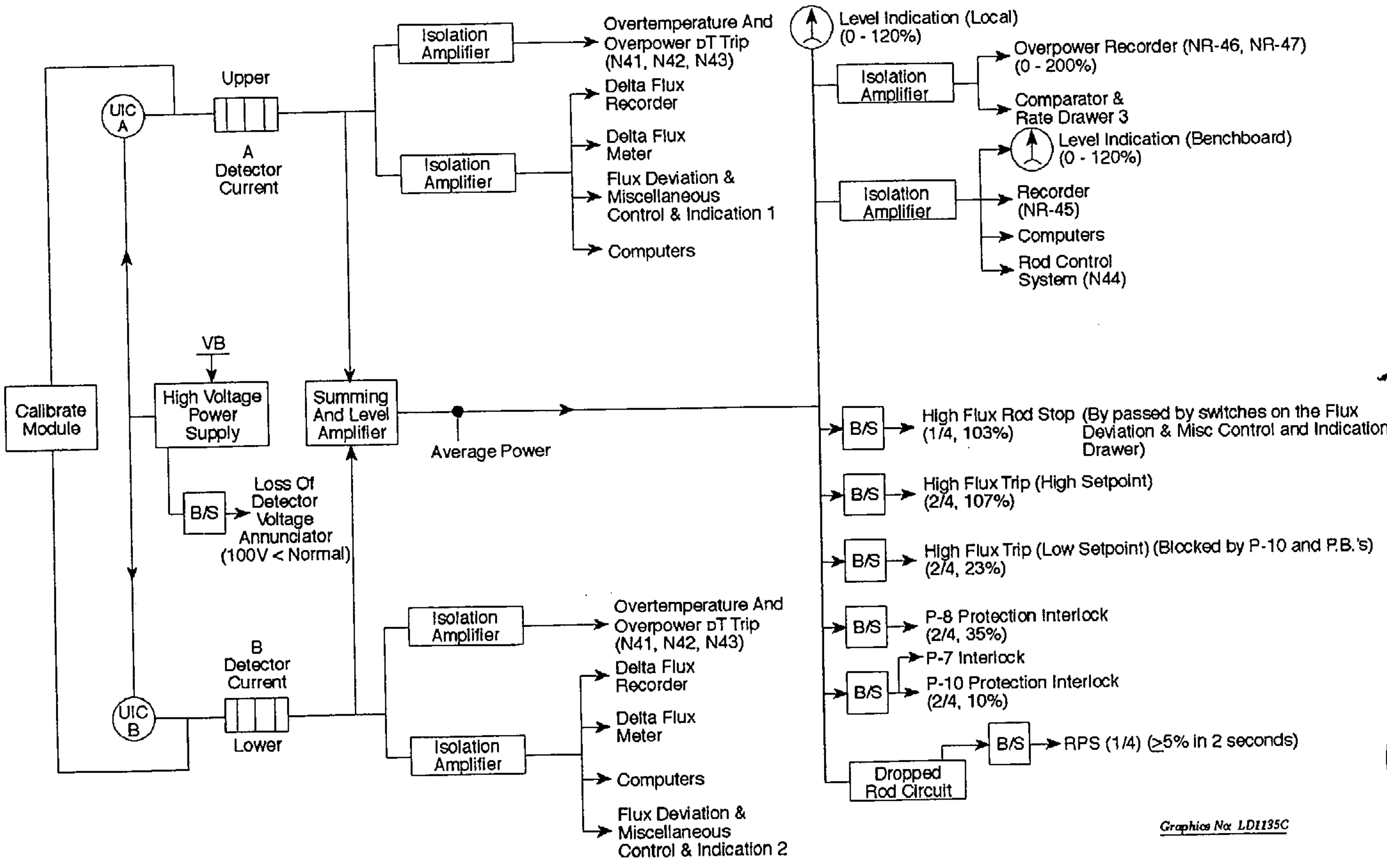
c. Overtemperature delta-T rod stop.

d. Overpower delta-T rod stop.

ANSWER: b

[RO: Tier 2/Group 1]

Answer correct:	Distractors plausible:	Distractors incorrect:
Loss of VB I-III will trip all the bistables associated with power-range channel III (N-43) including the high flux rod stop; coincidence for the rod stop is one of four PR NI's.	All will stop automatic and manual rod withdrawal..	a,intermediate-range NI's are powered from vital bus I-1 and 1-II. b,OT delta T rod stop coincidence is 2/3 channels. D. OP delta T rod stop coincidence is 2/3 channels.
K/A: SYS.015.K2.01	Objective: 2556	Source: New
Reference: ND-93.2-LP-4	Level: Comprehension	



Trip 1 of 4 bistables makes up 1/4 coincidence

Upper/Lower Ion Chamber Current to ΔT Protection System for Δø Penalty

Overpower Rod Stop 1/4 at 103%

Overpower Rx Trip 2/4 at 107%

Overpower Rx Trip (Low) 2/4 at 23%, manually blocked by P-10

P-8 3/4 PR < 35% blocks Low Flow Trip in 1/3 Loops

P-10 2/4 PR 10%, allows manual block of overpower trip (low stpt), IR High Flux Trip and Rod Stop, Blocks reset of SR High Voltage

Dropped Rod (1/4) 5% reduction in 2 secs

1 - Compares Upper Detectors to the Avg of the Upper Detectors.

2 - Compares Lower Detectors to the Avg of the Lower Detectors.

3 - Compares Channel Avgs to the lowest channel.

POWER RANGE CHANNEL

QUESTION 85: (1.0)

During a LOCA the pressure of isolated SI Accumulators is monitored as RCS pressure decreases below Accumulator pressure.

What is the reason for this monitoring of SI Accumulator pressure?

a. Verifies the injection phase of safety injection has occurred.

b. Ensures that Technical Specification capacities are maintained.

c. Used to determine if accumulator check valve failure has taken place.

d. SI Accumulator isolation MOV limit switches are not environmentally qualified.

ANSWER: d

[RO: Tier 2/Group 1]

Answer correct: The MOV limit switches for the SI accumulator isolation valves are not EQ. Therefore, accumulator pressure must be monitored to verify their isolation upon RCS depressurization.	Distractors plausible: a. This would be the normal action for a limiting design basis LOCA. b. Technical specifications control volume, Cb, and pressure of the accumulators. c. A check valve leakage has occurred in plant.	Distractors incorrect: a. Isolated accumulators do not inject. b. Technical Specifications apply to the accumulators only when the reactor is critical. c. MOV isolates the accumulator.
K/A: SYS022.K3.01	Objective: 3052	Source: New
Reference: ND-95.3-LP-7.doc	Level: Knowledge	

satisfied.

b. The accumulators are isolated to prevent discharge into the RCS when the hot leg temperature criteria are satisfied.

(1) At this point where RCS pressure is less than the LHSI pump shutoff head, a significant amount of accumulator nitrogen has been discharged into the RCS.

(2) The 360°F hot leg temperature setpoint was selected so that the RCS saturation pressure for this temperature exceeds the accumulator N_2 pressure after accumulator water has been discharged. The Psat associated with 360°F is 154 psia. Below this temp, Psat decreases and additional N_2 may be introduced into the RCS.

(3) To prevent accumulator isolation on an erroneous reading, at least two RTDs should be less than 360°F.

c. **RCS depressurization can be performed concurrently with accumulator venting provided RCS pressure is maintained greater than the accumulator nitrogen pressure. (rk)**

d. Should a coincident loss of the "J" Bus EDG and offsite power occur, neither the accumulator isolation MOV or containment IA would be available for accomplishing this step. In this case, RCS pressure would have to be maintained sufficient to prevent accumulator nitrogen injection until either electrical power or containment IA is restored. This requires that SG pressure be maintained greater than 150 psig.

29. **<u>STEP 22</u>: VERIFY PRESSURE FOR ANY ISOLATED SI ACC(S) REMAINS STABLE AS RCS PRESSURE DECREASES BELOW ACCUMULATOR**

PRESSURE.

a. The purpose of this step is to direct the operator to ensure that SI Accumulator discharge MOVs are holding as RCS pressure decreases.

b. An Engineering evaluation has determined that the SI Accumulator discharge MOV position limit switches are not EQ. Thus, under adverse CTMT conditions, the switches could indicate inaccurately.

c. This step is a **CONTINUOUS ACTION STEP** as denoted by the asterisk (*) preceding it. **(rk)**

30. **<u>CAUTION PRIOR TO STEP 23</u>: IF FUEL DAMAGE IS SUSPECTED (BASED ON HIGH CETC INDICATIONS OR HIGH RCS ACTIVITY), SGs SHOULD NOT BE DEPRESSURIZED BELOW RCS PRESSURE.**

a. The purpose of this caution is to minimize the potential for a radiological release.

b. If fuel damage is suspected based on high CETC temperatures or high sample activities, it is advantageous to keep the secondary side pressure above the primary side pressure in order to minimize radiological releases.

31. **<u>STEP 23</u>: CHECK IF INTACT SGs SHOULD BE DEPRESSURIZED TO RCS PRESSURE.**

a. The purpose of this step is to cooldown and depressurize the secondary side if intact SG pressures are greater than RCS pressure.

b. At this point, RCS pressure is low and the plant is on cold leg recirc. However, the secondary side may still be relatively hot and at a pressure higher

QUESTION 86: (1.0)

The following Unit conditions exist:

- Unit One is shutdown.
- The Unit One A Main Feed pump is feeding all three S/Gs at a rate of 800 gpm each.
- The Unit One reactor operator is holding depressed the Train "A" and "B" feedwater isolation reset buttons in order to completely fill the steam generators for wet layup.

Which ONE of the following events would trip the Unit One "A" Main Feed Pump?

a. High Water Level in the Steam Generators.

b. 'B' Main Feed Pump Recirc Valve fails closed.

c. Safety Injection Initiation Train B.

d. Load Shed Actuation on all Station Service Busses.

ANSWER: b

[RO: Tier 2/Group 1]

Answer correct: The main feed pumps trip if their recirc valve is not open within 15 seconds of < 2800 gpm discharge flow.	Distractors plausible: a/c - these are valid main feed pump trips. d – this is a valid trip for the 1B Main Feed Pump	Distractors incorrect: a\c – the S/G High water level and SI main feed pump trips are blocked by the Isolation buttons being depressed by the operator. d – load shed does not affect 1A main feed pump.
K/A: SYS059.K4.16	Objective: 2040	Source: New
Reference: ND-89.3-LP-3	Level: Comprehension	

c. The pump and motor bearings are lubricated by each pumps' lubrication oil system. Radial movement of the pump shaft is restricted by two (2) Journal bearings mounted on opposite sides of the pump. A 6 inch Kingsbury thrust bearing minimizes pump axial motion.

d. The main feed pumps have a set of pre-start conditions which must be satisfied prior to start. These interlocks are:

Refer to/re-display H/T-3.3, Page 1, Main Feed System Fact Sheet, and use with the following information.

(1) $^{>}$10 psig - Lube Oil Pressure

(2) $^{>}$100 psig - Suction Pressure

(3) Lockout reset (no overcurrent or ground)

e. Each main feed pump has ten (10) trips associated with it. They are:

(1) Overcurrent or Ground

(2) Lube Oil Press. $^{<}$4 psig.

(3) S.I. (Train A or B) (May be bypassed with "Reset" buttons)

(4) Low Suction Header Press. $^{<}$55 psig - 15 SEC. T.D.Both pumps

(5) $^{<}$2800 gpm MFW Flow & Recirc Valve not open, 15 SEC. T.D.

(6) Other Motor Overcurrent or Ground.

(7) Bus Undervoltage - Approximately 70%

(8) S/G Hi-Hi Level - 2/3 Channels on 1/3 S/G's $^{>}$75% (May Be Bypassed with "RESET" buttons)

(9) Manual

(10) Load Shedding 1B & 2A MFPs

Refer to/display H/T-3.3, Page 2, Main Feed System Fact Sheet, and use with the following information.

2. Discharge MOVs are provided for each main feed pump. They provide the ability to isolate the high pressure discharge of the pump. They have no automatic opening signal and must be manually opened by a switch on the Main Control Board. Their opening logics allow them to be opened and remain open if **both** motors for the associated pump are running. They can be closed either MANUALLY or will automatically close if EITHER MOTOR BREAKER OPENED for the associated pump (for backup to auto FRV closure).

3. Pump Recirculation Subsystem

Refer to/display H/T-3.4, Recirculation System, and use with the following information.

a. The purpose of the recirculation subsystem is to provide for minimum flow to be diverted to the main condenser to prevent MFP overheating during low flow conditions and allow pump coastdown upon stop.

Have trainees refer to Technical Specification Interpretation #14, Feedwater Isolation, and discuss with class.

5. The FEED REG BYPASS VALVES are controlled by a manual control station controller. The valves are used to regulate feedwater flow to the S/G at low power levels. They are air operated and fail closed upon loss of IA. The block signals that will automatically shut these valves are:

   a. SI,

   b. 2/3 channels S/G Hi-Hi level on 1/3 S/Gs (75%) (block respective valve).

   c. These signals (Hi-Hi level and SI) can be bypassed by pushing the S/G Level Reset pushbuttons.

      (1) Both pushbuttons can be depressed any time after actuation of either the SI signal or Hi-Hi S/G level. The SI signal does not have to be reset. After both pushbuttons have been depressed, a MFP can be started and the MFRV bypass HCVs can be operated.

      (2) If both FW isolation reset pushbuttons are depressed prior to either SI or Hi-Hi S/G level and held in until after the signal is locked in, the MFPs will not trip and the MFRV bypass HCVs will not close.

6. The FIRST POINT FEEDWATER HEATERS are used to raise the temperature of the feed pump discharge from approximately 375°F to approximately 435°F. This action adds efficiency to the cycle and reduces thermal stress on components. The 1st point FW heaters receive extraction steam from the 1st point extraction, drains from the 4th pass MSR reheat pots and drains from the Reheater Drain receivers.

QUESTION 87: (1.0)

A male radiation worker's total effective dose equivalent for the current quarter is 1837 mrem and for the current year is 3823 mrem.

The worker __________ be authorized access to the RCA __________.

a. can; if a dose extension request is prepared and authorized

b. can; with no additional authorization other than an RWP

c. cannot; because his quarterly dose has exceeded the administrative limit

d. cannot; because his quarterly and annual dose have both exceeded the administrative limit

ANSWER: a

[RO: Tier 3]

Answer correct: per VPAP-2101, a worker whose dose has exceeded admin limits will be denied RCA access; a dose extension request may be prepared and authorized to restore RCA access.	Distractors plausible: all – candidate misconception concerning dose limits and VPAP requirements.	Distractors incorrect: b – per VPAP-2101, an RWP is not sufficient authorization to enter the RCA; the worker needs an approved dose extension request; c & d – for the stated conditions, the worker can be authorized RCA access with an approved dose extension request.
K/A: GEN-2.3.4	Objective: 8049	Source: New
Reference: VPAP-2101	Level: Comprehension	

i. If a worker needs to enter a posted High Radiation Area Exceeding 15 rem/hr or a Posted Very High Radiation Area, then, in addition to requirements for entering a Locked High Radiation Area, the worker shall ensure an RWP briefing is attended before the entry, and that continuous communications are established with an HP Technician during work in the area. **[Commitment 3.2.6]**

NOTE: If a worker's dose exceeds administrative limits, the worker will be denied RCA access. To allow worker access to an RCA, a dose extension request must be prepared, normally by the worker's supervisor. Dose extension requests must be signed by the affected worker and Station management.

j. If a dose extension request is submitted for worker signature, before signing the request the worker should review the extension and ask any appropriate questions.

6.3.2 **Regulatory Dose Limits and Controls**

Limits provided in this Step are 10 CFR 20 limits. Exceeding these limits is a violation of 10 CFR 20 and shall be appropriately addressed and reported.

NOTE: Total Effective Dose Equivalent (TEDE) means the sum of deep-dose equivalent (from external exposures) and committed effective dose equivalent (from internal exposures). Deep-dose equivalent may be referred to as external whole body dose.

a. **Dose Limits for Radiation Workers**

Dose limits for occupationally exposed adult radiation workers, including occupational dose received at all facilities, including Surry and North Anna, are:

Type	Worker Dose Limit
Total Effective Dose Equivalent (TEDE)	5 rem per calendar year
Lens of Eye (lens dose equivalent)	15 rem per calendar year
Skin (shallow dose equivalent)	50 rem per calendar year
Extremities (shallow dose equivalent)	50 rem per calendar year

6.3.3 Administrative Dose Limits

NOTE: Dose limits in 6.3.3 do not apply to a Declared Pregnant Woman. Declared Pregnant Woman administrative dose control is addressed in 6.3.5.

NOTE: Dose limits in 6.3.3 are implemented by controls specified in 6.3.4.

Administrative dose limits are established to minimize the potential for exceeding federal limits. If a worker exceeds an administrative dose limit without exceeding a 10 CFR 20 or Technical Specifications (TS) limit, the event shall not be considered a violation of either 10 CFR 20 or TS. Exceeding administrative limits shall require a radiological incident investigation and a Deviation (i.e., Plant Issue) in accordance with VPAP-1501, Deviations. Investigation results shall be used to determine reportability and shall become Station records.

a. **Radiation Worker Quarterly Administrative Dose Limits**

Type	Radiation Worker Quarterly Administrative Dose Limit
Total Effective Dose Equivalent (TEDE)	2.0 rem/calendar quarter

b. **Radiation Worker Annual Administrative Dose Limits**

Type	Radiation Worker Annual Administrative Dose Limits
Total Effective Dose Equivalent (TEDE)	4.0 rem/calendar year
Lens of Eye (lens dose equivalent)	12.0 rem/calendar year
Skin (shallow dose equivalent)	40.0 rem/calendar year
Extremities (shallow dose equivalent)	40.0 rem/calendar year

c. **System Worker Annual Administrative Dose Limits**

Type	System Worker Annual Administrative Dose Limits
Total Effective Dose Equivalent (TEDE)	0.5 rem/calendar year

6.3.4 Administrative Dose Controls - General Requirements

NOTE: An integral part of administrative dose controls is the control of access to RCAs. RCA access control is addressed in 6.6.1.

a. The following control is in place to provide reasonable assurance that a worker will not exceed administrative dose limits.

If a worker has a quarterly or annual dose within 200 mrem of an administrative dose limit, the worker will be denied RCA access unless specifically authorized by the Supervisor Exposure Control and Instrumentation.

EXAMPLE: If a radiation worker has more than 1.8 rem deep-dose equivalent (whole body gamma plus neutron dose) in a calendar quarter or more than 3.8 rem TEDE in a calendar year, then that worker will be denied RCA access unless a dose extension request is approved. System employees will be denied access at 0.30 rem TEDE any time during a calendar year.

b. Request:

Type	Administrative Dose Limits
Total Effective Dose Equivalent (TEDE)	4.75 rem/year
Lens of Eye (lens dose equivalent)	14.0 rem/year
Skin (shallow dose equivalent)	45.0 rem/year
Extremities (shallow dose equivalent)	45.0 rem/year

c. An extension request shall be acknowledged by the affected worker and approved by:

- Department Superintendent (or Superintendent cognizant of worker duties)
- Superintendent Radiological Protection
- Site Vice President or Manager Station Operations and Maintenance or Manager Station Safety and Licensing

QUESTION 88: (1.0)

Using the references provided, determine which ONE of the following represents the current High Alarm setpoint for the NRC radiation monitor?

a. 4.24 mrem/hr.

b. 4200.00 mrem/hr.

c. 42.4 mrem/hr.

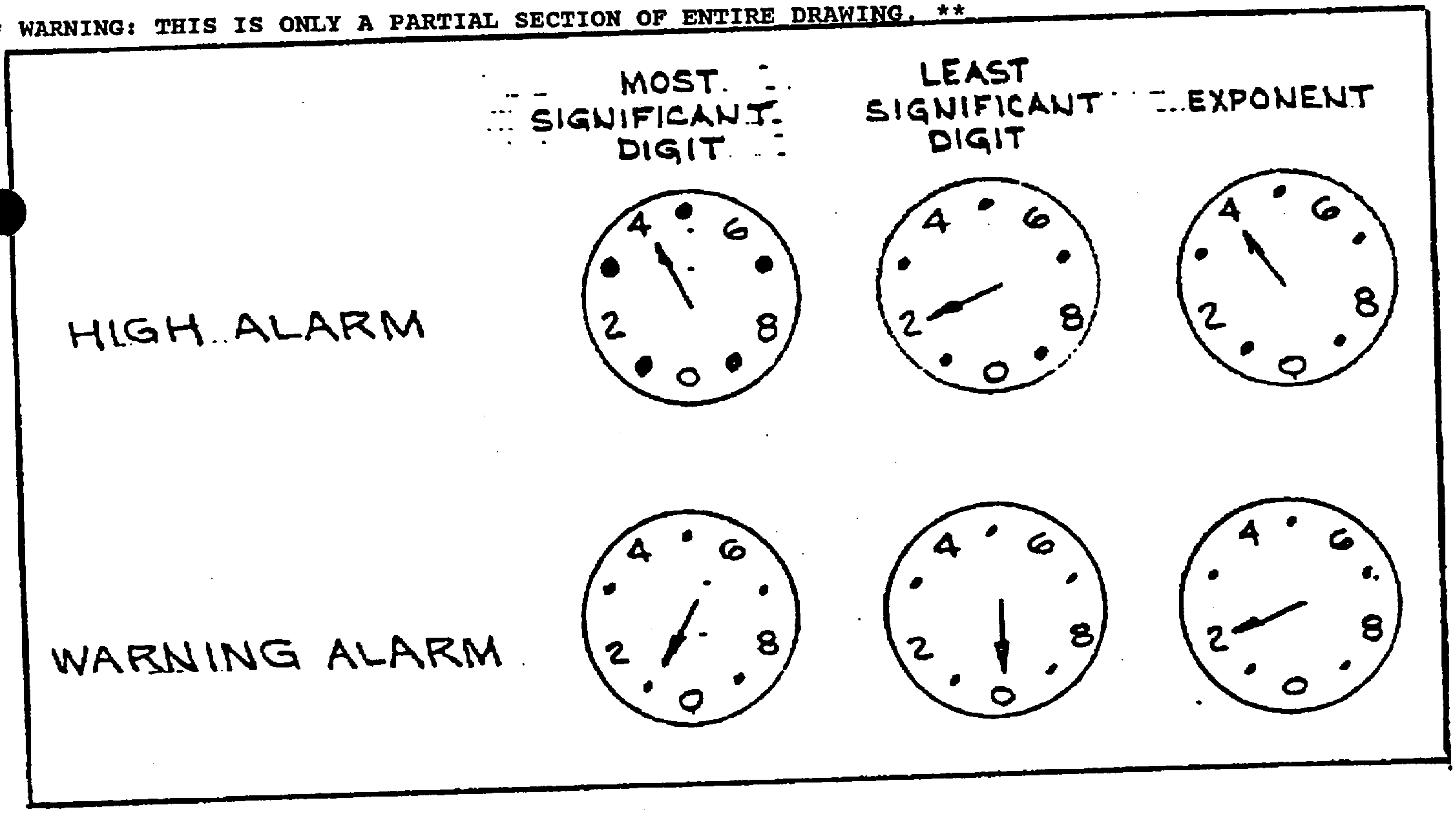
d. 420000.0 mrem/hr.

ANSWER: b

[RO: Tier 2/Group 1]

Answer correct: The setpoint is interpreted as 42 x $10e^4$ mrem/hr.	Distractors plausible: a/c – Logical combinations of the numbers 424 not realizing that the last number represents an exponent . d – improper application of the exponent.	Distractors incorrect: a/c/d – not the correct setpoint of 4200.00 mrem/hr.
K/A: SYS071.A4.25	Objective: 2744	Source: NEW
Reference: ND-93.5-LP-3, OPT-RM-001	Level: Knowledge	

** WARNING: THIS IS ONLY A PARTIAL SECTION OF ENTIRE DRAWING. **



c. The control units for the NRC monitors are located in #1 ESGR on the wall next to the "B" DC panel.

d. The power supply for these detectors is selectable from either unit's semi-vital bus. A three-position select switch (Unit 1-OFF-Unit 2), and two (2) white lights indicate the source of power.

e. Each control unit contains a microprocessor which selects the appropriate detector, makes the measurement, converts the data to mrem/hr, displays the data, and provide alarms.

f. The readout in mrem/hr is updated every 4 seconds. The readout has 2 significant digits and an exponent.

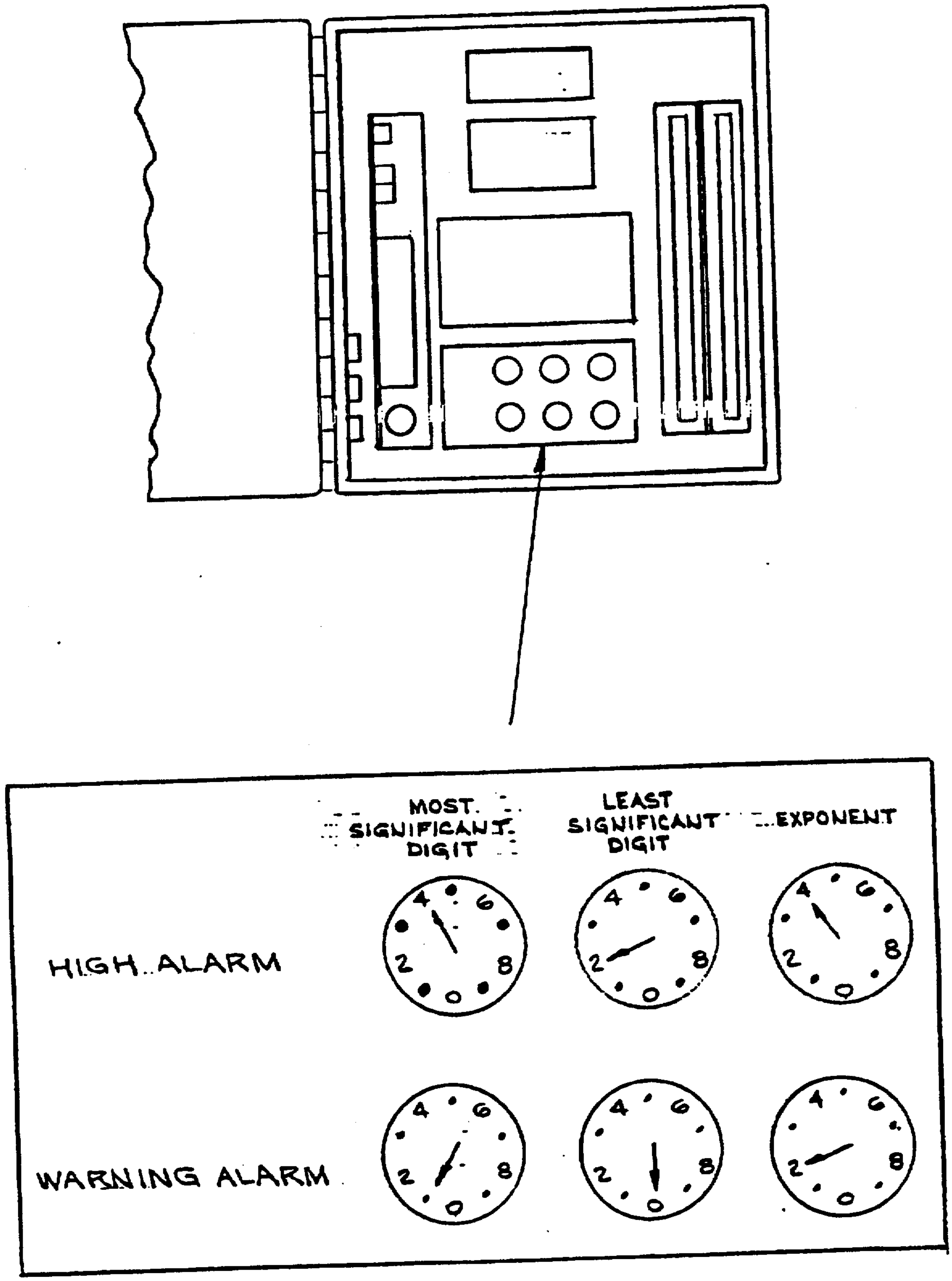
> Write on board - Reading - .43 X E2.
>
> Ask: What does this reading mean?
>
> Answer: 43 mrem/hr.

3. Technical Specifications

a. Tech Spec 3.7 has a table (3.7-6) which lists the Accident Monitoring Systems which must be operable. The NRC radiation monitors are part of this list.

b. According to this list, three monitors must be operable for each units main steam lines and each units TDAFW pump exhaust monitors must be operable. The Kaman monitors for process vent and vent-vent meet the requirements for this table. The NRC monitors for process vent and vent-vent are the preplanned alternates.

TA-900 CONTROL UNIT



EXAMPLE SHOWN WITH
- HIGH ALARM SET TO $.42 \times 10^4$
- WARNING ALARM SET TO $.10 \times 10^2$

NRC 200190

FIG. 2

ALARM SETTINGS

ATTACHMENT 6

(Page 1 of 1)

NRC RADIATION MONITORS

NOTE: Performance of this attachment satisfies the monthly channel check requirements of Tech Spec Table 4.1-2.

1. Verify tamper seals are in place on each monitor. IF a tamper seal is missing, THEN verify alert and alarm setpoints at or below indicated value, AND place a new tamper seal on the monitor door.
2. IF tamper seals were found intact, THEN enter N/A in the Alarm and Alert blanks.
3. Record actual reading, Power light status and Failure light Status (LIT / NOT LIT).
4. Do not record power supply. (It is shown only to aid in locating RM ratemeters.)

	2-MS-RI-224 Tamper Seal Installed ____ Alarm $0.25x10^1$______ Alert $0.13x10^0$______ Reading: __________ Power Light ________ Failure Light ________	**1-GW-RI-122** Tamper Seal Installed ____ Alarm $0.53x10^5$______ Alert $0.27x10^4$______ Reading: __________ Power Light ________ Failure Light ________	**1-MS-RI-124** Tamper Seal Installed ____ Alarm $0.25x10^1$______ Alert $0.13x10^0$______ Reading: __________ Power Light ________ Failure Light ________
1-MS-RI-129 Tamper Seal Installed ____ Alarm $0.28x10^2$______ Alert $0.14x10^1$______ Reading: __________ Power Light ________ Failure Light ________	**2-MS-RI-225** Tamper Seal Installed ____ Alarm $0.25x10^1$______ Alert $0.13x10^0$______ Reading: __________ Power Light ________ Failure Light ________	POWER SUPPLY	**1-MS-RI-125** Tamper Seal Installed ____ Alarm $0.25x10^1$______ Alert $0.13x10^0$______ Reading: __________ Power Light ________ Failure Light ________
2-MS-RI-229 Tamper Seal Installed ____ Alarm 0.___ ______ Alert $0.14x10^1$______ Reading: __________ Power Light ________ Failure Light ________	**2-MS-RI-226** Tamper Seal Installed ____ ______ Alert $0.13x10^0$______ Reading: __________ Power Light ________ Failure Light ________	**1-VG-RI-123** Tamper Seal Installed ____ ______ Alert $0.17x10^0$______ Reading: __________ Power Light ________ Failure Light ________	**1-MS-RI-126** Tamper Seal Installed ____ Alarm $0.25x10^1$______ Alert $0.13x10^0$______ Reading: __________ Power Light ________ Failure Light ________

Completed by: ______________________________ Date: ______________

QUESTION 89: (1.0)

The power to area radiation monitor 1-RM-RMS-157, MCR Area Monitor, has been lost.

Which ONE of the following indications will alert the operator that this detector has failed?

a. Digital Ratemeter "RANGE" light on.

b. Digital Ratemeter "FAIL" light on.

c. "Alert/Failure" annunciator illuminated.

d. Local detector "HIGH" light flashing.

ANSWER: c

[RO: Tier 2/Group 1]

Answer correct: The loss of power would cause the Alert/Failure Annunciator to energize due to the monitor being designed as fail safe. This means that the alarm relay on a loss of power causes the annunciator to actuate. The annunciator system is on a different power source.	Distractors plausible: a/b – these are other status monitoring lights on the front of the panel. d – This is another status light near the detector.	Distractors incorrect: a – the Range light energizes on a off scale high or low signal but requires power to turn on. b – The Fail light energizes on a drawer electronics or detector failure but requires power to turn on. d – the local detector high light is energized upon a valid radiation alarm and requires power to actuate.
K/A: SYS072.A2.02	Objective: 2726	Source: NEW
Reference: ND-93.5-LP-1	Level: Comprehension	

Several equipment failure conditions are monitored which produce a FAIL alarm and in some cases an error message. The fail condition is "TRUE" whenever any equipment failure is detected and "FALSE" when no equipment failures are detected. When a fail condition occurs, other than power failure, the red FAIL alarm indicator illuminates and the fail relay coil de-energizes. The Fail alarm logic is always fail-safe and the following is a brief description of the ratemeter failure modes:

- **Power Failure** - If power is lost to the ratemeter, the bargraph, alarm indicators, and the displays are blanked (turned off). The HIGH, WARN, and FAIL relay coils de-energize and open the alarm contacts.

- **No Count Failure** - If no pulses are received by the ratemeter for five minutes (30 minutes on the air ejector ratemeters), a no count failure is detected. A no count alarm usually indicates a failure in the detector or high voltage supply. The ratemeter display, however may read zero for five to thirty minutes or more without a low signal fail alarm. This is because the preamplifier is reporting a non-zero dose rate that is below the low range value.

- **MPU Failure** - If the fail timer circuit (Watchdog circuit), which checks the MPU (Main Processor Unit) function, is allowed to time out (because of a hardware failure), a failure condition will be indicated.

QUESTION 90: (1.0)

Given the following plant conditions:

- Unit 1 is stable at 20% power with turbine control in IMP-IN.
- Rod Control is in MANUAL with Tave and Tref matched.
- "B" RCP trips.
- The reactor does **not** trip automatically.
- **No operator actions** are taken.
- The transient continues until the plant stabilizes.

Which ONE of the following is correct?

a. The indicated final steady-state values for "A" and "C" loop delta-T will be lower than their pre-event values.

b. The indicated final steady-state value for Tref will be higher than its pre-event value.

c. The actual final steady-state value for "B" loop delta-T will be higher than its pre-event value.

d. The actual final steady-state core delta-T will be higher than its prevent value.

ANSWER: d

[RO: Tier 2/Group 2]

Answer correct: With the turbine in IMP-IN, turbine control will modulate to maintain steam flow constant; as flow through "B" loop decreases, less heat is removed from "B" S/G and more heat is removed form the active loops; final steady-state result is delta-T in active loops increases and delta-T in "B" loops decreases; overall core delta-T increases to maintain constant heat transfer.	Distractors plausible: all – candidate misconception concerning the effects of a flow reduction on RCS parameters.	Distractors incorrect: a – active loop delta-T's increase from pre-event values. b – final steady-state Tref will be equal to initial value. c – "B" loop delta-T will be lower than its pre-event value.
K/A: SYS002-K5.01	Objective: 1406	Source: New
Reference: North Anna Simulator validation	Level: Comprehension	

QUESTION 91: (1.0)

Fire water is initiated to extinguish a fuel pit bridge fire.

With this continued dilution in progress, at what Boron concentration will the Technical Specification minimum ≤ 0.95 K_{eff} limit be violated?

a. 2500 ppm.

b. 2250 ppm.

c. 2300 ppm.

d. A fully diluted spent fuel pool will not violate the limit.

ANSWER: d

[RO: Tier 2/Group 2]

Answer correct: The spent fuel pool is designed to maintain Keff ≤ 0.95 without boric acid in the water.	Distractors plausible: a – Upper Boron Concentration Spec for the RWST b – Lower Boron Concentration Spec for an Accumulator; c – TS 5.4-2 requires 2300 ppm in the spent fuel pool;	Distractors incorrect: b – multiple samples already indicate that boron concentration is low, there is no need to obtain additional samples prior to taking action; c – boration is not necessary, rod unlatching is a core alteration and must be suspended until the gate valve is closed; d – boration is not necessary.
K/A: 033-A2.01	Objective: 2494	Source: New
Reference: TS-5.4, ND-92.5-LP-6	Level: Knowledge	

assemblies to ensure $k_{eff} \leq 0.95$, even if unborated water were used to fill the spent fuel storage pit. The spent fuel pool is divided into a two-region storage pool. Region 1 comprises the first three rows of fuel racks (324 storage locations) adjacent to the Fuel Building Trolley Load Block. Region 2 comprises the remainder of the fuel racks in the fuel pool. During spent fuel cask handling, Region 1 is limited to storage of spent fuel assemblies which have decayed at least 150 days after discharge and shall be restricted to those assemblies in the "acceptable" domain of Figure 5.4-1. Administrative controls with written procedures will be employed in the selection and placement of these assemblies. The enrichment of the fuel stored in the spent fuel racks shall not exceed 4.3 weight percent of U-235.

C. Whenever there is spent fuel in the spent fuel pit, the pit shall be filled with borated water at a boron concentration not less than 2300* ppm to match that used in the reactor cavity and refueling canal during refueling operations.

D. The only drain which can be connected to the spent fuel storage area is that in the reactor cavity. The strict step-by-step procedures used during refueling ensure that the gate valve on the fuel transfer tube which connects the spent fuel storage area with the reactor cavity is closed before draining of the cavity commences. In addition, the procedures require placing the bolted blank flange on the fuel transfer tube as soon as the reactor cavity is drained.

* This limit takes effect at the time the Unit 2 reactor cavity is flooded following the end of Operating Cycle 10.

<u>References</u>

FSAR Section 9.5 Fuel Pit Cooling System

FSAR Section 9.12 Fuel Handling System

QUESTION 92: (1.0)

Given the following plant conditions:

- The "A" waterbox is being removed from service for leak repairs.
- The operator inadvertently closes 1-VP-4 (air ejector suction from **"B"** waterbox) **instead of** 1-VP-3 (air ejector suction from **"A"** waterbox).
- CW flow through "A" waterbox is isolated per procedure.

As a result, the _________ air ejector(s) will become steam-bound and __________.

a. "A"; condenser vacuum will degrade

b. "B"; condenser vacuum will degrade

c. "A" and "B"; condenser vacuum will degrade

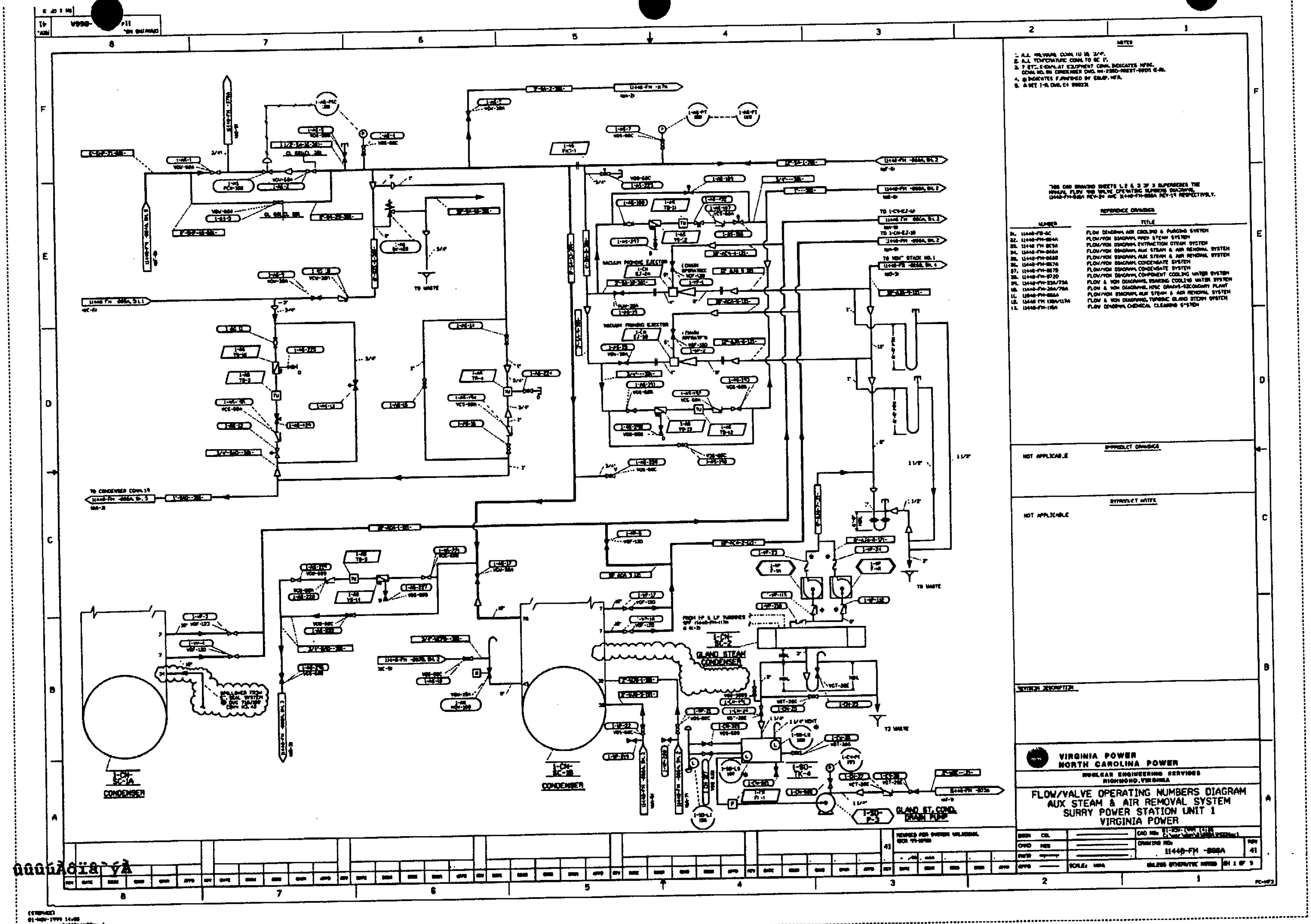
d. "A"; condenser vacuum will be maintained by "B" air ejector via the suction cross-tie

ANSWER: a

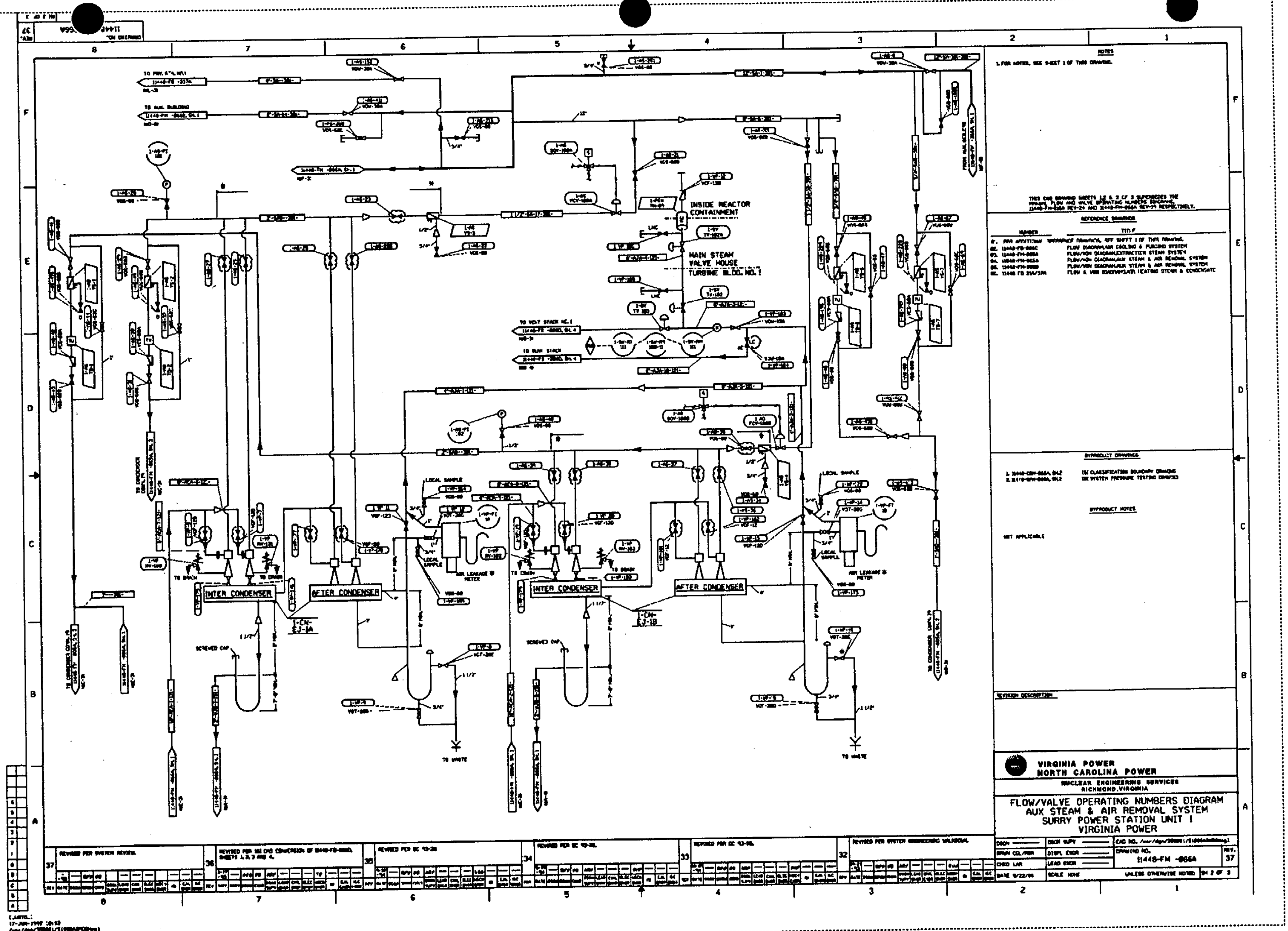
[RO: Tier 2/Group 2]

Answer correct: "A" air ejector is normally aligned to "A" condenser ("A" and "B" waterboxes); per MOP (and based on plant event) if air ejector suction is not isolated prior to securing CW flow through a waterbox, condenser vacuum **will** degrade.	Distractors plausible: b – vacuum will degrade, candidate misconception concerning normal alignment of air ejectors; c – "A" air ejector will become steam-bound, vacuum will degrade, there is a cross-tie on the suction of the air ejectors; d – "A" air ejector will become steam-bound.	Distractors incorrect: b & c – "B" air ejector is not affected, since the suction cross-tie is normally closed; d – the suction cross-tie is normally closed.
K/A: SYS055-K3.01	Objective: 2030	Source: New
Reference: 11448-FM-66A sh. 1 & 2, ND-89.3-LP-2, NCRODP-25	Level: Comprehension	

* DO NOT USE FOR WORK IN PLANT. DATA LOSS POSSIBLE AT THIS SCALE & DPI. **



** DO NOT USE FOR WORK IN PLANT. DATA LOSS POSSIBLE AT THIS SCALE & DPI. **



QUESTION 93: (1.0)

Which ONE of the following identifies the power supply to the Unit 1 “A” Electric Hydrogen Recombiner Unit?

a. 1H1-2 MCC.

b. 2H1-2 MCC.

c. 1J1-2 MCC.

d. 2J1-2 MCC.

ANSWER: a

[RO: Tier 2/Group 3]

Answer correct: This is the correct power supply to the Hydrogen Recombiner.	Distractors plausible: b – This the power supply for the unit 2 hydrogen recombiner. c – This is the power supply to the “B” Hydrogen Recombiner. d – This is the	Distractors incorrect: b/c/d – these are not the correct power supply.
K/A: SYS028.K2.01	Objective: 2703	Source: NEW
Reference: ND-93.4-LP-4	Level: Knowledge	

the lower part of the upper louvers located on three sides of the unit.

(4) The middle bank of heaters in each Recombination Unit has three thermocouples. These thermocouples can be monitored on the Control Panel in the MCR Annex.

c. Power supply panel

(1) The Power Supply Panel contains all the necessary electrical equipment, to provide the power required by the heaters in the Recombination Unit.

(2) Location - Cable Vault.

(3) Power supply - 'A' Recombiner - MCC 1H1-2 6D
'B' Recombiner - MCC 1J1-2 6D

d. Control panel

(1) The Control Panel contains all the control and monitoring equipment required for operating the Recombination Unit.

(2) Location - Control Room Annex.

3. Operation

a. The EHRS is used to reduce the hydrogen concentration in the containment atmosphere following a LOCA.

b. The EHR is placed into service IAW Attachment 4 of E-1, (Loss of Reactor or

QUESTION 94: (1.0)

A Unit 1 Instrument Air line rupture has caused a complete loss of air pressure with both Units at power. Subsequently, the reactor operator notices that Containment Instrument Air pressure is currently 60 psig and decreasing.

Which ONE of the following caused the decrease in Containment Instrument Air?

a. Normally cross-tied with the Station Instrument Air System.

b. 1-IA-TV-100 (Normal Discharge) trips closed due to low Containment Instrument Air pressure interlock.

c. 1-IA-TV-100 (Normal Discharge) fails closed due to loss of dome supply air.

d. 1-IA-TV-101A (Normal Suction) fails closed due loss of dome supply air.

ANSWER: c

[RO: Tier 2/Group 3]

Answer correct: The normal discharge valve for containment I/A is held open by station Instrument Air. Thus a loss of station instrument air will cause a loss of containment instrument air as the Normal discharge fails closed.	Distractors plausible: a – This would be true if the unit was shutdown. B – there is a low containment discharge valve closure at ≤ 30 psig in the containment instrument air system. D - Misunderstanding of the interlock between normal suction and outside suction swapover.	Distractors incorrect: a – Cross tie with Station Instrument Air is not performed until ≤ 200 °F. b – The given containment instrument air pressure is greater than the trip setpoint. d – If either normal suction closes then the outside side would automatically open with minimal affect on containment instrument air.
K/A: SYS078.K4.02	Objective: 2335	Source: NEW
Reference: ND-92-LP-1	Level: Comprehension	

(a) Each valve has pushbutton controls with integral indication of valve position located on the Control Room vertical board at the containment isolation panel.

(b) Valves will automatically close on CLS HI signal, or high containment radiation level, indicated by the manipulator crane, containment particulate, or containment gaseous radiation monitors.

(c) When opening the valve, the valve must be fully open before the red pushbutton can be released.

(3) Alternate suction valve (AOV-IA-103) is controlled by the positions of the normal valves. If both normal suction valves are open, the alternate suction valve SOV is deenergized and the alternate suction is closed. If either normal suction valve closes, the alternate suction valve opens automatically.

AOV supplied from normal Station IA.

- There is no manual electrical control for the alternate suction valve. Valve position is indicated in the Control Room on the vertical board.

d. Compressor discharge trip valve (TV-IA-100) - controlled from vertical board 1-1 (containment isolation panel). Automatically closes on CLS High High, or low instrument air header pressure of 30 psig, sensed by PS-IA-109 A or B. Position indicated on the Control Room vertical board near the control switch.

e. Indication of system pressure is provided by PT-IA-106, which senses the pressure in the B receiver.

f. A rotameter installed in the compressor common discharge line provides local

QUESTION 95: (1.0)

Which ONE of the following would prevent making a mode change to Refueling Shutdown?

a. RCS temperature 143°F.

b. Reactor head detensioned but still installed.

c. RCS is subcritical by 5125 pcm.

d. 4 incore flux thimbles not retracted.

ANSWER: a

[RO: Tier 3]

Answer correct: RCS temperature must be ≤ 140°F in order to be in Cold Shutdown.	Distractors plausible: b –Fuel scheduled to be moved is a requirement which the head detensioned implies. C – Misunderstanding of 5% $\Delta k/k$ requirement converted to pcm. D – Flux thimbles still installed would prevent fuel movement.	Distractors incorrect: b – The head being detensioned would imply fuel movement is to be performed which is part of the definition. C – The subcritical margin of 5000 pcm is satisfied. D – Flux thimbles not being retracted would not prevent entering Refueling Mode.
K/A: GEN-2.1.22	Objective: 1719	Source: New
Reference: TS-1.0 Definitions, ND-88.1-LP-9	Level: Knowledge	

1.0 DEFINITIONS

The following frequently used terms are defined for the uniform interpretation of the specifications.

A. RATED POWER

A steady state reactor core heat output of 2546 MWt.

B. THERMAL POWER

The total core heat transferred from the fuel to the coolant.

C. REACTOR OPERATION

1. REFUELING SHUTDOWN

5000 pcm

When the reactor is subcritical by at least 5% Δk/k and T_{avg} is ≤ 140°F and fuel is scheduled to be moved to or from the reactor core.

2. COLD SHUTDOWN

When the reactor is subcritical by at least 1% Δk/k and T_{avg} is ≤ 200°F.

3. INTERMEDIATE SHUTDOWN

When the reactor is subcritical by at least 1.77% Δk/k and
200°F < T_{avg} < 547°F.

4. HOT SHUTDOWN

When the reactor is subcritical by at least 1.77% Δk/k and T_{avg} is ≥ 547°F.

QUESTION 96: (1.0)

The System Operator has requested that you operate the main generator in a VARS IN condition. The following conditions exist:

- Maximum attainable Hydrogen pressure is 45 psig.
- Main Generator output is 800 Megawatts.

Using the supplied reference, which ONE of the following identifies the maximum VARS IN that can be carried by Surry Unit 1?

a. 280 MVAR's.

b. 270 MVAR's.

c. 260 MVAR's.

d. 210 MVAR's.

ANSWER: d

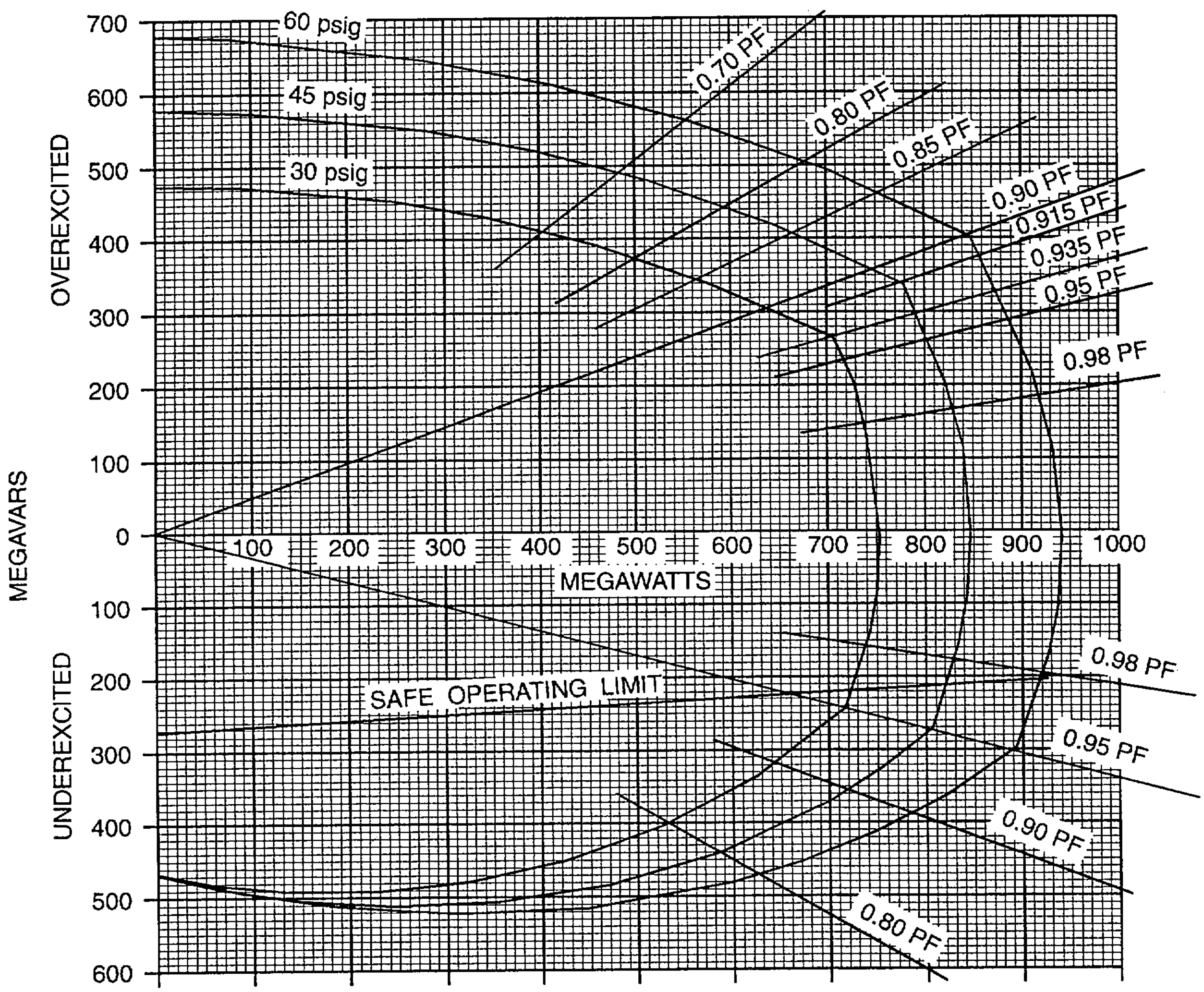
[RO: Tier 3]

Answer correct: Correct answer from curve from 1-DRP-003 att. 48.	Distractors plausible: a – this is where the 45 psig and 800 Mw intersect disregarding the safe operation limit line b – at 800 Mw 0.95 pf which looks like an operating limit(solid line) c – using VARS IN vice OUT to determine limit.	Distractors incorrect: a/b/c - incorrect based on curve from 1-DRP-003 att. 48.
K/A: GEN-2.1.25	Objective: 2184	Source: New
Reference: 1-DRP-003 att 48 ND-90.1-LP-6	Level: Comprehension	

ATTACHMENT 48

(Page 1 of 2)

CAPABILITY CURVES



HYDROGEN INNER-COOLED TURBINE GENERATOR
CALCULATED CAPABILITY CURVES
941, 700 KVA, 0.90 PF, 0.58 SCR, 60 PSIG, 3 PH
60 CYCLES, 1800 RPM

Graphics No: KM1117

b. EXCITER FIELD BREAKER TRIP (J-B-8) - Indicates the Exciter field breaker has tripped. Causes: Exciter filed grounded or shorted, or Main transformer overexcitation (OPX or V/Hz).

c. VOLT REG AUTO TRIP (J-C-7) - Caused by the limits on the regulator exceeded (OPX), loss of the field breaker, regulator in test or withdrawn, regulator PT fuse blown.

d. GEN FIELD GROUNDED (J-C-8) - Indicates a ground on the generator rotor.

e. GEN PT BLOWN FUSE (J-H-6) - Indicates a blown fuse on the metering or relay PT, a spike on the system as given by the loss of a large Unit.

f. GEN/MN XFMR OVEREXCITATION RELAY TROUBLE (K-C-1) - Loss of DC to the Beckwith relay (V/Hz), blown voltage regulator PT fuse, V/Hz difference between any two potential phases, or a V/Hz relay internal error.

D. Generator Capability Curve

Refer to/display HT-6.9, Generator Capability Limits

1. Operation of the generator is limited by the Generator Capability Curve.

2. The curve is based on protecting various components of the generator during operation from overheating under various MW, MVAR, and H_2 pressure conditions. SCR refers to short circuit ratio, determined during testing. SCR is the amount of heat the windings can handle based on the current generated by a direct short across the output of the generator.

3. When the Generator is overexcited - VARS are out, the generator is considered to be

Inductive with a Lagging power factor (pf).

When the Generator is underexcited - VARS are in, the generator is considered to be Capacitive with a Leading power factor (pf).

4. The generator should be limited within the envelope of the H_2 gas line, the .9 pf overexcited line, and .95 pf underexcited lines.

5. Operation above the Safe Operating Limit line (ADMIN Limit) will prevent reaching the Minimum Excitation Limit.

6. Limitations

   a. Between a pf of 0-.9 overexcited, generator limits are based on rotor winding temperature limits.

   b. Between a pf of .9 overexcited and .95 underexcited, generator limits are based on stator winding temperature limits.

   c. Between a pf of .95 underexcited and 0, generator limits are based on stator core temperature limits.

7. Below the safe operating limits, three (3) additional limits apply:

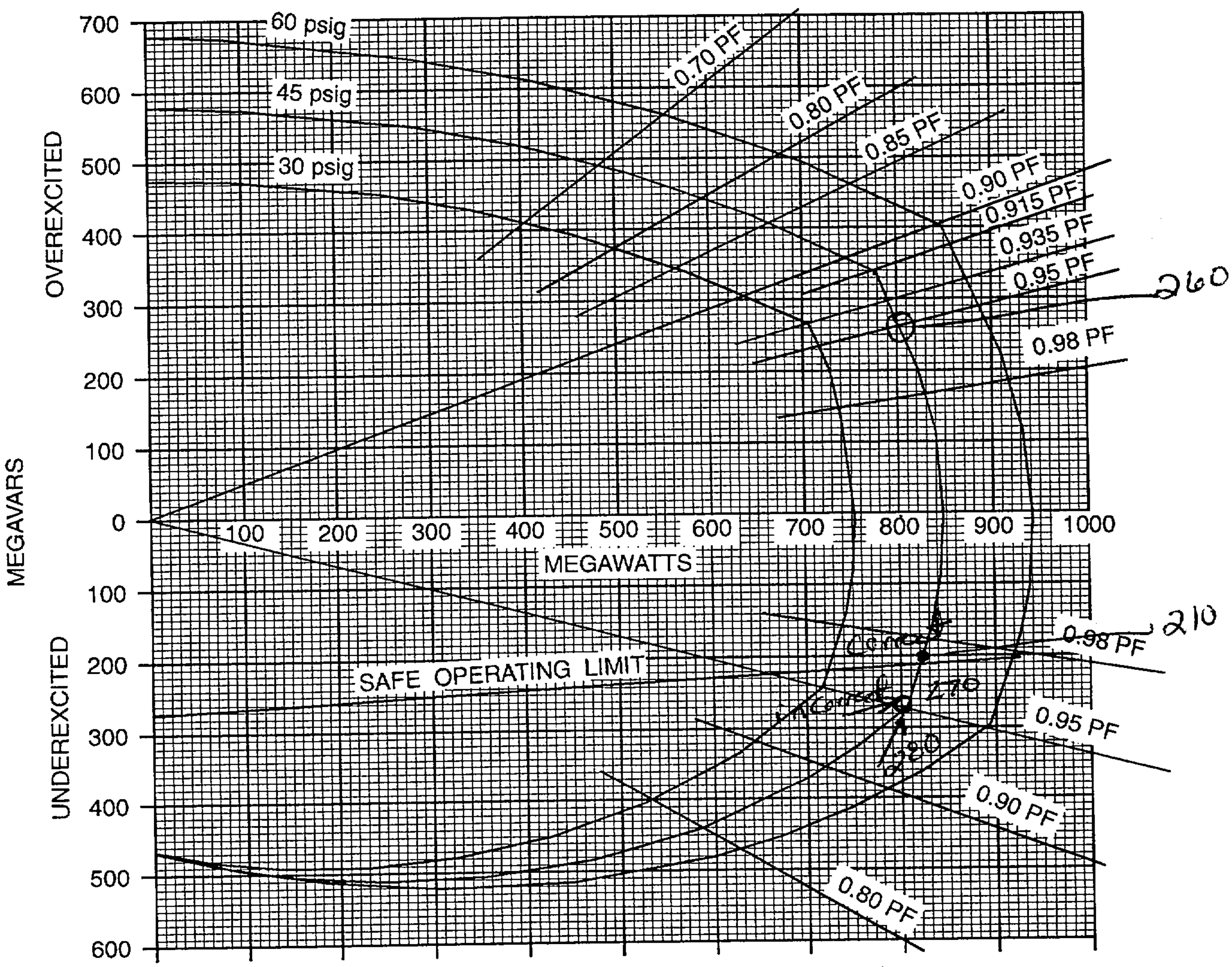
   a. Minimum excitation limit.

   b. KLF relay actuation.

ATTACHMENT 48

(Page 1 of 2)

CAPABILITY CURVES



Graphics No: KM1117

HYDROGEN INNER-COOLED TURBINE GENERATOR
CALCULATED CAPABILITY CURVES
941, 700 KVA, 0.90 PF, 0.58 SCR, 60 PSIG, 3 PH
60 CYCLES, 1800 RPM

QUESTION 97: (1.0)

The Turbine Building operator reports that the air side seal oil pump has tripped on thermal overload.

Which ONE of the following identifies the actions required to coordinate restoration of power to the seal oil pump?

a. Electricians must be contacted to reset the thermal overload device.

b. The Turbine Building operator has the authority to reset the thermal overload device once to determine the cause of the trip.

c. After determining the cause, the SRO may approve one reset of the thermal overload device.

d. The pump must be tagged out, bridged, and meggered prior to reset of the thermal overload device.

ANSWER: c

[RO: Tier 3]

Answer correct: This is correct in accordance with OPAP-006	Distractors plausible: a – Electricians work with the operators in returning equipment to service. b – This seems reasonable to determine the cause of the fault. d – This would be the correct response after resetting the overload once.	Distractors incorrect:.a – Operators may reset overloads. b – The turbine building operator must have the SRO's permission prior to resetting the thermal overload. d – This is not required for one reset of the thermal overload.
K/A: GEN-2.1.8	Objective: 8023	Source: New
Reference: OPAP-006	Level: Comprehension	

b. The Tag-Out should include the annunciator in accordance with OPAP-0010, Tag-Outs. **(Surry)**

6.7.6 A walkdown of Station annunciator panels should be performed weekly. **(North Anna)**

a. Any annunciators in alarm should be noted along with the cause of the alarm and the corrective action required for clearing the alarm.

b. The weekly review of the Station annunciator panels should be performed by an Operations Supervisor. The Operations Supervisor should submit a report to the Operation and Maintenance Superintendents.

6.8 Response to Indications

6.8.1 Instrument readings should be believed and treated as accurate unless proven otherwise.

6.8.2 If an unexpected reading occurs on an instrument, other indications should be observed to check the reading, if possible.

6.8.3 Abnormal or unexpected indications should be promptly investigated to determine the cause of the problem so immediate corrective action can be taken and promptly reported to the Unit SRO.

6.8.4 If an instrument or indication is determined to be malfunctioning or inaccurate, the device should be appropriately labeled, the Unit SRO notified, and a Maintenance Work Request initiated.

6.9 Resetting Protective Devices

6.9.1 Prior to resetting a protective device, the cause for the device tripping should be determined.

6.9.2 Prior to resetting a protective device, at a minimum, it shall be verified that no abnormal condition exists.

6.9.3 At the discretion of the Shift Supervisor, thermal overload devices may be reset once.

6.9.4 Reactor Trips require a full investigation in accordance with VPAP-1404, Reactor Control.

QUESTION 98: (1.0)

Unit 2 is in a refueling outage with core on-load in progress. The Refueling SRO requests that the running RHR pump be secured to increase water clarity. The RO checks the logs and determines that the running RHR pump was re-started at 1500 today.

Which ONE of the following times correctly states when the running RHR pump may be secured and for how long may it be secured?

a. 2300 today, 1 hour.

b. 1500 today, 6 hours.

c. 1900 today, 12 hours.

d. 1500 tomorrow, 8 hours.

ANSWER: a

[RO: Tier 3]

Answer correct: This is 8 hours after the last start of the pump (1500)	Distractors plausible: b/c/d – contain typical math errors and typical T.S. clocks	Distractors incorrect: b/c/d – None of these have the correct time after pump start or length of run.
K/A: GEN-2.2.30	Objective: 2469	Source: New
Reference: T.S. 3.10.A.6, ND-92.5 - LP-1	Level: Knowledge	

6. At least one residual heat removal pump and heat exchanger shall be operable to circulate reactor coolant. The residual heat removal loop may be removed from operation for up to 1 hour per 8-hour period during the performance of core alterations or reactor vessel surveillance inspections.

7. Two residual heat removal pumps and heat exchangers shall be operable to circulate reactor coolant when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

8. At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange during movement of fuel assemblies.

9. With the reactor vessel head unbolted or removed, any filled portions of the Reactor Coolant System and the refueling canal shall be maintained at a boron concentration which is:

   a. Sufficient to maintain K-effective equal to 0.95 or less, and

   b. Greater than or equal to 2300 ppm and shall be checked by sampling at least once every 72 hours.

10. Direct communication between the Main Control Room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

11. No movement of irradiated fuel in the reactor core shall be accomplished until the reactor has been subcritical for a period of at least 100 hours.

QUESTION 99: (1.0)

The Reactor Operator is in the process of performing a normal shutdown when she observes that letdown flow has decreased.

Which ONE of the following conditions would cause this decrease in letdown flow?

a. Decreasing RCS pressure.

b. Decreasing VCT pressure.

c. A decrease in the auto setpoint for PCV-1145.

d. Channeling of the mixed bed.

ANSWER: a

[RO: Tier 2/Group 1]

Answer correct:	Distractors plausible:	Distractors incorrect:
Letdown flow is a function of the d/p across the letdown orifice that directly determined by RCS pressure.	b. VCT pressure could be seen as affecting downstream pressure if PCV-1145 is not considered. c. Decreasing the auto setpoint could be picked if the operation of controllers is not properly understood. d. Mixed bed pressure drop could be seen as affecting downstream pressure if PCV-1145 is not considered.	b. VCT pressure would have to increase to lower D/P thus lowering flow. However, PCV-1145 maintains the pressure constant on the downstream side of the orifice. c. Lowering the auto setpoint of PCV-1145 would cause pressure on the downstream side of the orifice to decrease; therefore, flow would increase. d. Channeling of the mixed bed would cause pressure on the downstream side of the orifice to decrease; therefore, flow would increase. However, PCV-1145 maintains the pressure constant on the downstream side of the orifice.
K/A: SYS004.K1.30	Objective: 2523	Source: New
Reference: ND-93.1-LP-1	Level: Comprehension	

(20) Units check

$$\frac{lbm}{\sec} = ft^2 \sqrt{\frac{lbm}{ft^3}\left(\frac{lbf}{ft^2}\right)\frac{lbm - ft}{\sec^2 - lbf}}$$

End of non-testable material.

(21) Simplified form

$\dot{m} = K\sqrt{\Delta P \rho}$

$\Delta P \downarrow, \dot{m} \downarrow$

Taking the square root converts differential pressure to volumetric flow rate. Multiplying by the density converts volumetric flow rate to mass flow rate.

Refer to/display H/T-1.8, Orifice Plate.

b. Flow nozzle

The flow nozzle works on the same principle as the venturi, but due to its inherent construction, it is less energy efficient.

(1) The flow nozzle is less accurate than the venturi.

(2) On the other hand, it is also less expensive.

c. Flow orifice

The flow orifice works like the venturi and the flow nozzle in that it develops a low pressure area, but it is still less accurate.

QUESTION 100: (1.0)

Given the following plant conditions:

- The team is responding to an ATWS in accordance with 1-FR-S.1, Response to Nuclear Power Generation/ATWS.
- The RO is performing the immediate actions without assistance from the third licensed RO.
- Manual trip was attempted but the reactor would **not** trip.
- Control rods are in manual due to continuous rod insertion caused by instrument failure.

In response to this, the RO should __________.

a. trip the turbine, then manually insert control rods

b. Place rods in auto, then trip the turbine

c. trip the turbine, then place rods in automatic

d. manually insert control rods until all control rods are at zero steps, then trip the turbine

ANSWER: b

[RO: Tier 3]

Answer correct: OPAP-0002 states that the first two steps of this procedure must be completed in order given (rods driven in then turbine tripped)	Distractors plausible: a – this would be correct if rods do not insert automatically after the turbine is tripped; c– correct actions in the wrong order; d – with all control rods at zero steps, reactor power would be in the intermediate range.	Distractors incorrect: a – automatic rod insertion should be verified, rods should only be manually inserted if automatic rod insertion cannot be verified; c – rods must be addressed first d – turbine trip must be performed within 30 seconds.
K/A: GEN-2.4.1	Objective: 2993	Source: New
Reference: 1-FR-S.1, OPAP-002, ND-95.3-LP-36	Level: Knowledge	

2. Attachments are often used for procedure actions that can be accomplished by an Operator who is not part of the Control Room Shift Team.

6.4.4 **Implementation of EOPs After Initial Entry**

a. Communication for EOP implementation shall occur between the cognizant Shift Team members and the EOP Reader. No other personnel shall become involved in discussions, debates, or questions unless prompted by the cognizant SRO.

b. If an EOP contains a Continuous Action Page, it shall be frequently monitored. It contains information that may be applicable at any step of the procedure.

c. The Shift Team members involved in the performance of the EOPs shall accurately convey to the EOP Reader the status of equipment and conditions.

d. **Safety System Resets**

1. If the system (e.g., Safety Injection, Containment Isolation) has not been reset once, then the Reactor Operator shall reset the system when directed by procedure with concurrence from the cognizant SRO whether it has actuated or not. This action is not required if the procedure step includes the words **if necessary**.

2. Once the system has been reset once, it is not necessary to reposition the reset switches every time a procedure directs a reset. Verify that the applicable status light still reflects the reset condition. If it does not, another reset shall be performed.

e. **Immediate Action Steps**

1. EOP Immediate Action Steps should be performed from memory. Immediate Action Steps are designated by brackets around the individual step number in the applicable procedures (e.g., [1.])

2. The first four Immediate Action Steps of E-0 and the Immediate Action Steps of FR-S.1 shall be performed in sequence or sequentially. All other Immediate Action steps do not have specific step sequence requirements.

3. Immediate Action Steps that have been performed shall be verified when the EOP is entered.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1]	__VERIFY REACTOR TRIP: • Rod bottom lights - LIT • Reactor Trip and bypass breakers - OPEN • Neutron Flux - DECREASING	Manually trip the reactor. <u>IF</u> reactor will <u>NOT</u> trip, <u>THEN</u> verify automatic insertion. <u>IF</u> <u>NOT</u>, <u>THEN</u> manually insert control rods.
[2]	__VERIFY TURBINE TRIP: • Verify all turbine stop valves - CLOSED	Manually trip the turbine. <u>IF</u> turbine will <u>NOT</u> trip, <u>THEN</u> reduce load using limiter. <u>IF</u> turbine load can <u>NOT</u> be reduced, <u>THEN</u> close MSTVs.
3.	__VERIFY AFW PUMPS RUNNING:	
	a) MD AFW pumps - RUNNING	a) Manually start pumps.
	b) TD AFW pump - RUNNING IF NECESSARY	b) Manually open steam supply valves. • 1-MS-SOV-102A • 1-MS-SOV-102B

QUESTION 101: (1.0)

The team is checking if SI can be terminated in accordance with 1-E-1, Loss of Reactor or Secondary Coolant. RCS subcooling and heat sink are both adequate.

Because RCS pressure is increasing and PRZR level is off-scale low, the team is directed to try to stabilize RCS pressure with normal PRZR spray and to **not** terminate SI at this time.

Which ONE of the following correctly states the basis for stabilizing RCS pressure?

a. Minimizes the potential for brittle failure of the reactor vessel.

b. Prevents continued reduction in safety injection flow.

c. Prevents the PRZR safety valves and/or PORVs from lifting.

d. Prevents excessive primary-to-secondary delta-P across the S/G tube sheets.

ANSWER: b

[SRO: Tier 1/Group 1]

Answer correct: SPS uses centrifugal HHSI pumps; as RCS pressure increases, SI flow decreases; for the stated conditions, stabilizing RCS pressure will prevent SI flow from decreasing further and may eventually result in PRZR level coming on scale.	Distractors plausible: a – increasing RCS pressure **does** increase the potential for brittle failure of the vessel; c– increasing RCS pressure **could** eventually result in lifting PRZR safety valves and/or PORVs; d – increasing RCS pressure **does** increase the delta-P across the S/G tube sheets and a maximum limit exists for this parameter.	Distractors incorrect: a – E-1 is not the controlling procedure for this concern which would be FR-P series. C – this is only a concern after the pressurizer is filled with water. D – We would still be within the limits for D/P across the S/G.
K/A:EPE011-EA2.11	Objective: 3052	Source: New
Reference: 1-E-1, ND-95.3-LP-7	Level: Knowledge	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 6.	__CHECK IF SI FLOW SHOULD BE REDUCED:	
	a) RCS subcooling based on CETCs - GREATER THAN 30°F [85°F]	a) GO TO Step 7.
	b) Secondary heat sink: • Total feed flow to INTACT SGs - GREATER THAN 350 GPM [450 GPM] OR • Narrow range level in at least one intact SG - GREATER THAN 11% [22%]	b) GO TO Step 7.
	c) RCS pressure - STABLE OR INCREASING	c) GO TO Step 7.
	d) PRZR level - GREATER THAN 22% [43%]	d) Try to stabilize RCS pressure with normal PRZR spray. GO TO Step 7. To accomplish d)
	e) GO TO 1-ES-1.1, SI TERMINATION	
* 7.	__CHECK IF HI HI CLS INITIATED: • RS pump(s) - RUNNING OR • Any Hi Hi CLS annunciator - LIT	GO TO Step 13.

QUESTION 102: (1.0)

RCS cooldown is in progress per 1-ES-0.2, Natural Circulation Cooldown with CRDM Fans.

Which ONE of the following correctly describes why SI accumulators MOV's cannot be closed until RCS pressure is less than 1000 psig?

a. Prevents use of the "Defeat" switches to close the valves.

b. Ensures RCS leakage can be soley mitigated by the LHSI pumps.

c. Ensures RCS leakage can be soley mitigated by the HHSI pumps.

d. Ensures compliance with T.S. 3.3, Safety Injection System.

ANSWER: d

[SRO: Tier 1/Group 1]

Answer correct: This is a requirement by T.S. 3.3.A.2.d	Distractors plausible: a – trainee misconception concerning interlocks for accumulator discharge valves. B/C – the reduction in RCS pressure will cause SI pump flow to increase while reducing break flow.	Distractors incorrect: a – The defeat switches must be in defeat to close the valves but are not pressure interlocked. B/c – the accumulators may be needed for injection of water so isolating them early is not desired.
K/A: E09-EA2.2	Objective: 3044	Source: New
Reference: 1-ES-0.2, T.S. 3.3.A.2.d ND-93.5-LP-5	Level: Comprehension	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18.	___CHECK IF SI ACCUMULATORS SHOULD BE ISOLATED:	
	a) Check power to Accumulator discharge isolation valves - AVAILABLE	a) Locally close the following breakers (Zone 1 key required): • 1H1-2N 5B • 1J1-2E 1B • 1J1-2E 1C
	b) Check RCS pressure - LESS THAN 1000 PSIG	b) GO TO Step 20. <u>WHEN</u> RCS pressure less than 1000 psig, <u>THEN</u> do Steps 18c, 18d and 19.
	c) Isolate SI Accumulators: 1) Put ACC interlock key switches in DEFEAT: (keys 11, 12, and 13) • 1-SI-MOV-1865A • 1-SI-MOV-1865B • 1-SI-MOV-1865C 2) Close the following: • 1-SI-MOV-1865A • 1-SI-MOV-1865B • 1-SI-MOV-1865C	c) <u>IF</u> CTMT and Turbine BLDG IA available, <u>THEN</u> do the following to vent any unisolated SI ACC: 1) Consult with TSC or HP. 2) Verify or place in service the Process Vent system. 3) Close or verify closed 2-SI-TV-201A and B. 4) Close or verify closed 1-RC-HCV-1549. 5) Open ACC vent line isolation valve, HCV-1853A, B, or C. 6) Open 1-SI-TV-101A and B. 7) Adjust HCV-1936 to vent SI ACC(s). <u>IF</u> accumulators can <u>NOT</u> be vented, <u>THEN</u> do the following: 1) Maintain SG pressure greater than 150 psig. 2) Consult with TSC.
	d) Locally open the following breakers: • 1H1-2N 5B • 1J1-2E 1B • 1J1-2E 1C	

c. 550 psig is the steam generator pressure corresponding to condensate pump shutoff head.

Discuss the steps/contents of Attachment 4 as required.

22. <u>STEP 18</u>: **CHECK IF SI ACCUMULATORS SHOULD BE ISOLATED.**

a. The purpose of this step is to determine if appropriate plant conditions exist for isolating the SI accumulators.

b. The accumulator isolation valves should be closed to prevent dumping of the accumulator water into the RCS when RCS pressure drops below accumulator pressure.

c. The criteria for isolating accumulators is based on TS operability requirements.

d. Since power to the isolation valves is administratively removed during startup, local actions are required to close the breakers. Note that the key switch <u>must</u> be placed into defeat in order to close the MOVs (independent of RCS pressure).

e. Any accumulator that can not be isolated must be vented.

(1) Should a coincident loss of the #3 EDG and offsite power occur, neither the accumulator isolation MOV nor Ctmt IA would be available for accomplishing this step.

The accumulators (one for each loop) discharge into the cold leg of the reactor coolant piping when Reactor Coolant System pressure decreases below accumulator pressure, thus assuring rapid core cooling for large breaks. The line from each accumulator is provided with a motor-operated valve to isolate the accumulator during reactor start-up and shutdown to preclude the discharge of the contents of the accumulator when not required.

Accumulator Motor Operated Discharge Isolation Valves

Unit No. 1	Unit No. 2
MOV 1865A	MOV 2865A
MOV 1865B	MOV 2865B
MOV 1865C	MOV 2865C

However, to assure that the accumulator valves satisfy the single failure criteria, they will be locked, sealed or otherwise secured open by de-energizing the valve motor operators when the reactor coolant pressure exceeds 1000 psig. The operating pressure of the Reactor Coolant System is 2235 psig and accumulator injection is initiated when this pressure drops to 600 psia. De-energizing the motor operator when the pressure exceeds 1000 psig allows sufficient time during normal startup operation to perform the actions required to de-energize the valve. This procedure will assure that there is an OPERABLE flow path from each accumulator to the Reactor Coolant System during POWER OPERATION and that safety injection can be accomplished.

The removal of power from the valves listed above will assure that the systems of which they are a part satisfy the single failure criterion.

QUESTION 103: (1.0)

With Unit 1 at 100% power, a seismic event resulted in minor damage to all main steamlines in the Safeguards Building. The team entered 1-ECA-2.1, Uncontrolled Depressurization of all Steam Generators. Given the following conditions:

- "A" S/G Narrow Range level is 8%.
- "B" S/G Narrow Range level is 9%.
- "C" S/G Narrow Range level is 7%.
- RCS cooldown rate is 220°F per hour.

Which ONE of the following is the minimum <u>total</u> AFW flow to the three S/G's?

a. 300 GPM

b. 540 GPM

c. 180 GPM

d. 350 GPM

ANSWER: c

[SRO: Tier 1/Group 1]

Answer correct: per 1-ECA-2.1, the team must throttle aux feedwater to 60 gallons per minute per generator.	Distractors plausible: a – correct if adverse containment conditions existed; b – this is the minimum flow in the EOPs with no SI and RCPs running (post-trip); d – minimum flow in the EOPs with SI in service and no RCPs running (post-trip).	Distractors incorrect: all – based in conditions, flows are excessive.
K/A: E12-EK1.2	Objective: 2958	Source: New
Reference: WOG B/G document for 1-ECA-2.1, ND-95.3-LP-22	Level: Comprehension	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: A minimum of 60 gpm [100 gpm] feed flow must be maintained to each SG with a narrow range level less than 11% [22%].

* *

NOTE: Shutdown Margin should be monitored during RCS cooldown. |16

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	CONTROL FEED FLOW TO MINIMIZE RCS COOLDOWN:	
	a) Check cooldown rate in RCS cold legs - LESS THAN 100°F/HR	a) Lower feed flow to 60 gpm [100 gpm] to each SG. GO TO Step 2c.
	b) Check narrow range level in all SGs - LESS THAN 50%	b) Control feed flow to maintain narrow range level less than 50% in all SGs.
	c) Check RCS hot leg temperatures - STABLE OR DECREASING	c) Control feed flow or dump steam to stabilize RCS hot leg temperatures.

b. Depending upon the size of the effective break areas for the SGs, the cooldown rate experienced after reactor trip could exceed 100°F/hr. A reduction of feed flow to the SGs has three primary effects:

Write the following primary effects on the board:

- to minimize any additional cooldown resulting from the addition of feedwater,
- to prevent SG tube dryout by maintaining a minimum feed flow to the SGs, and
- to minimize the water inventory in the SGs that eventually is the source of additional steam flow to containment or the environment.

c. The minimum feed flow of 60 gpm represents the minimum verifiable flow rate corresponding to the WOG recommendation of 25 gpm.

d. As steam flow rate drops, the feed flow will eventually increase SG inventory. Feed is then controlled to maintain SG NR level at less than 50% to prevent overfeeding the SGs.

e. In addition, as SG pressure and steam flow rate drop, RCS hot leg temperatures will stabilize and start increasing. The team controls feed flow or dumps steam to stabilize the RCS hot leg temperatures. This allows the SI flow to establish conditions for SI termination and minimizes thermal stresses that may be generated.

f. The maximum cooldown rate of 100°F/hr may exceed the Surry Tech Spec limits. However, best estimate analyses performed indicate that this is not likely to cause significant flaw extension and is, therefore, a more significant limit for situations addressed by this procedure. This is consistent with generic

QUESTION 104: (1.0)

The following conditions exist:

- #2 EDG is secured for fuel oil filter replacement.
- Breaker 15H8 trips due to internal fault. This action initiates an electrical perturbation tripping "B" and "C" RSST's.
- All systems respond as designed.

Which ONE of the following identifies the most limiting Technical Specification action?

a. Restore #2 EDG to operable status within 7 days.

b. Establish Unit 2 backfeed to "E" transfer bus within 8 hours.

c. Restore 15H8 to operable status within 6 hours.

d. Place Unit 2 in HSD within 6 hours.

ANSWER: d

[SRO: Tier 1/Group 1]

Answer correct: for the stated conditions, both offsite power sources are inoperable for unit 2 and both unit 2 EDGs are inoperable because UV/DV protection is not functional when the EDG is carrying the bus; TS-3.0.1 is the most limiting action.	Distractors plausible: all – candidate misconception concerning EDG operability when supplying emergency bus.	Distractors incorrect: a – this presumes unit 2 EDGs are operable, they are not; c – this presumes all EDGs are operable, they are not; d – this presumes unit 2 EDGs are operable, they are not.
K/A: EPE055/GEN-2.1.33	Objective: 2237	Source: New
Reference: TS-3.0.1, TS-3.16, ND-90.3-LP-1	Level: Comprehension	

3.16 EMERGENCY POWER SYSTEM

Applicability

Applies to the availability of electrical power for safe operation of the station during an emergency.

Objective

To define those conditions of electrical power availability necessary to shutdown the reactor safely, and provide for the continuing availability of Engineered Safeguards when normal power is not available.

Specification

A. A reactor shall not be made critical nor shall a unit be operated such that the reactor coolant system pressure and temperature exceed 450 psig and 350°F, respectively, without:

1. Two diesel generators (the unit diesel generator and the shared backup diesel generator) OPERABLE with each generator's day tank having at least 290 gallons of fuel and with a minimum on-site supply of 35,000 gal of fuel available.

2. Two 4,160V emergency buses energized. → 3.01 clock
   6 hrs to HSD
   30 hrs to CSD

3. Four 480V emergency buses energized.

4. Two physically independent circuits from the offsite transmission network to energize the 4,160V and 480V emergency buses. One of these sources must be immediately available (i.e. primary source) and the other must be capable of being made available within 8 hours (i.e. dependable alternate source).

5. Two OPERABLE flow paths for providing fuel to each diesel generator.

6. Two station batteries, two chargers, and the DC distribution systems OPERABLE.

7. Emergency diesel generator battery, charger and the DC control circuitry OPERABLE for the unit diesel generator and for the shared back-up diesel generator.

B. During power operation or the return to power from HOT SHUTDOWN, the requirements of specification 3.16-A may be modified by one of the following:

1.a. With either unit's dedicated diesel generator or shared backup diesel generator unavailable or inoperable:

1. Verify the operability of two physically independent offsite AC circuits within one hour and at least once per eight hours thereafter.

2. If the diesel generator became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the operability of the remaining OPERABLE diesel generator daily. For the purpose of operability testing, the second diesel generator may be inoperable for a total of two hours per test provided the two offsite AC circuits have been verified OPERABLE prior to testing.

3. If this diesel generator is not returned to an OPERABLE status within 7 days, the reactor shall be brought to HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

1.b. One diesel fuel oil flow path may be "inoperable" for 24 hours provided the other flow path is proven OPERABLE. If after 24 hours, the inoperable flow path cannot be returned to service, the diesel shall be considered "inoperable." When the emergency diesel generator battery, charger or DC control circuitry is inoperable, the diesel shall be considered "inoperable."

2. If a primary source is not available, the unit may be operated for seven (7) days provided the dependable alternate source can be OPERABLE within 8 hours. If specification A-4 is not satisfied within seven (7) days, the unit shall be brought to COLD SHUTDOWN.

3. One battery may be inoperable for 24 hours provided the other battery and battery chargers remain OPERABLE with one battery charger carrying the DC load of the failed battery's supply system. If the battery is not returned to OPERABLE status within the 24 hour period, the reactor shall be placed in HOT SHUTDOWN. If the battery is not restored to OPERABLE status within an additional 48 hours, the reactor shall be placed in COLD SHUTDOWN.

C. The continuous running electrical load supplied by an emergency diesel generator shall be limited to 2750 KW.

<u>Basis</u>

The Emergency Power System is an on-site, independent, automatically starting power source. It supplies power to vital unit auxiliaries if a normal power source is not available. The Emergency Power System consists of three diesel generators for two units. One generator is used exclusively for Unit 1, the second generator for Unit 2, and the third generator functions as a backup for either Unit 1 or 2. The diesel generators have a cumulative 2,000 hour rating of 2750 KW. The actual loads using conservative

ratings for accident conditions, require approximately 2,320 kw. Each unit has two emergency buses, one bus in each unit is connected to its exclusive diesel generator. The second bus in each unit will be connected to the backup diesel generator as required. Each diesel generator has 100 percent capacity and is connected to independent 4,160 v emergency buses. These emergency buses are normally fed from the reserve station service transformers. The normal station service transformers are fed from the unit isolated phase bus at a point between the generator terminals and the low voltage terminal of the main step-up transformer. The reserve station service transformers are fed from the system reserve transformer in the high voltage switchyard. The circuits which supply power through either system reserve transformer are called "primary source." In the event a system reserve transformer is inoperable, the remaining one may be cross-tied by a 34.5 bus to all three reserve station service transformers. Thus, a primary source is available to both units even if one of the two system reserve transformers is out of service. Verification of primary source operability is performed by confirming that the reserve station service transformers are energized.

In addition to the "primary sources," each unit has an additional off-site power source which is called the "dependable alternate source." This source can be made available in eight (8) hours by removing a unit from service, disconnecting its generator from the isolated phase bus, and feeding offsite power through the main step-up transformer and normal station service transformers to the emergency buses.

The generator can be disconnected from the isolated phase bus within eight (8) hours. A unit can be maintained in a safe condition for eight (8) hours with no off-site power without damaging reactor fuel or the reactor coolant pressure boundary.

Verification of the dependable alternate source operability is accomplished by verifying that the required circuits, transformers, and circuit breakers are available.

QUESTION 105: (1.0)

Which ONE of the following identifies the purpose of the station Black Battery?

a. Ensures the station batteries have sufficient power to supply vital DC loads for 2 hours.

b. Provides a safety related battery that is located above mean sea level.

c. Ensures adequate station DC power is available to place both units in cold shutdown within 8 hours upon a loss of all AC power.

d. Provides a backup battery for the station DC distribution system.

ANSWER: a

[SRO: Tier 1/Group 1]

Answer correct: The black battery was installed to remove the Emergency Lube Oil pump and the Seal Oil Backup pump from the station battery.	Distractors plausible: b – the station batteries are located in the ESGR which is subject to flooding. C – the removal of the Emergency Lube Oil pump and the Seal Oil Backup pumps allowed the station batteries to last longer. D – trainee misconception concerning DC distribution.	Distractors incorrect: b – the black battery is not safety related. C – the black battery only supplies the Emergency Lube oil pump and the Seal Oil Backup Pump. D – There are no interconnections with the DC distribution system.
K/A: EPE055-EK3.01	Objective: 2277	Source: New
Reference: ND-90.3-LP-6	Level: Knowledge	

- HPD trip circuit.

- PORV 1456, and 1 set of indicating lights for 1455C and 1456 position

- "B" train Reactor Head and Pzr head vent valves

- A, B, and C Feed reg valves

- SI Accumulator valves

B. Black Battery System

1. Prior to 1987, the largest loads on the Station Battery Buses were the Emergency Turbine Oil Pump (60 HP) and the Air Side Seal Oil Backup Pump (25 HP). These loads are considered to be nonsafety related.

2. To reserve the capacity of the Station Batteries for vital safety related equipment, these two pumps were shifted to a different battery power supply called the "Black Battery."

3. Two of the existing battery chargers that were replaced by the UPS system were moved to the Black Battery House. The Black Battery houses is located between the condensate storage tanks and the Unit 1 Main Transformers.

4. The house is divided into two sections.

   a. One section contains the U-1 and U-2 Black Batteries.

   b. The second section houses the motor starters for the Air Side Oil Backup Pump and the Emergency Turbine Lube Oil Pump, the battery chargers, the AMSAC UPS and the DC distribution panels for this equipment.

QUESTION 106: (1.0)

Given the following plant conditions:

- A fire in the MCR has forced evacuation.
- Unit 1 was manually tripped from 100% power.
- "A" S/G PORV has been reported to be continuously lifting.
- "A" S/G pressure is 980 psig and decreasing.

Per FCA-1.00, the team should __________.

a. locally block the PORV closed

b. establish S/G High/Low interface integrity at the ASDP

c. place the Cable Vault keyswitch panel switch for 1-MS-RV-101A in "EMERG CLOSE"

d. proceed to Instrument Rack MB8 and take 1-MS-RV-101A to Remote/Manual and close the valve

ANSWER: c

[SRO: Tier 1/Group 1]

Answer correct: The keyswitch for the PORV is for Appendix "R" isolation so that the valve can be closed in the event of a hot short.	Distractors plausible: a – there is a manual isolation valve for the PORV. B – trainee misconception concerning isolations at the ASDP. D – it is possible to control the valve from the instrument rack	Distractors incorrect: a/d – No procedural guidance to perform these actions. B – The High/Low switches at the ASDP do not include isolation for the S/G PORV's.
K/A: 067/GEN-2.4.27	Objective: 3136	Source: New
Reference:FCA-1.00 ND-95.6-LP- 3	Level: Comprehension	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	TRANSFER CONTROL TO AUXILIARY SHUTDOWN PANEL (Continued):	
	c) At the Auxiliary Shutdown Panel place CHG FLOW CONTROL in LOCAL position • ()-CH-FCV-()122	
	d) Take manual control of CHG FLOW CONTROL and adjust to control flow • ()-CH-FCV-()122	d) <u>IF</u> CHG flow can <u>NOT</u> be controlled, <u>THEN</u> locally close ()-CH-304, ()-CH-FCV-()122 Outlet <u>AND</u> control flow by throttling ()-CH-305, ()-CH-FCV-()122 Bypass valve.
	e) Locally open the following breakers: • ()-CH-MOV-()115B, ()H1-2N 1C • ()-CH-MOV-()115D, ()J1-2E 5A • ()-CH-MOV-()115C, ()H1-2N 2C • 1-CH-MOV-1115E, 1J1-2E 2C • 2-CH-MOV-2115E, 2J1-2W 10C	

* *

<u>CAUTION</u>: Only the PORV on the depressurizing SG should be closed.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14.	___CHECK PRESSURE IN ALL SGs - STABLE OR INCREASING	In the Cable Vaults, on the Key Switch panel, place key switch for MS PRESS CONT VLV FIRE EMERG CLOSE to EMERG CLOSE position: • ()-MS-RV-()01A • ()-MS-RV-()01B • ()-MS-RV-()01C

QUESTION 107: (1.0)

Unit 1 is in Intermediate Shutdown with the following plant conditions:

- RCS temperature = 360°F.
- PRZR pressure = 600 psig.
- PRT pressure = 35 psig.

Using the reference provided, determine which ONE of the following tailpipe temperatures would be indicative of substantial PORV seat leakage.

a. 483°F

b. 259°F

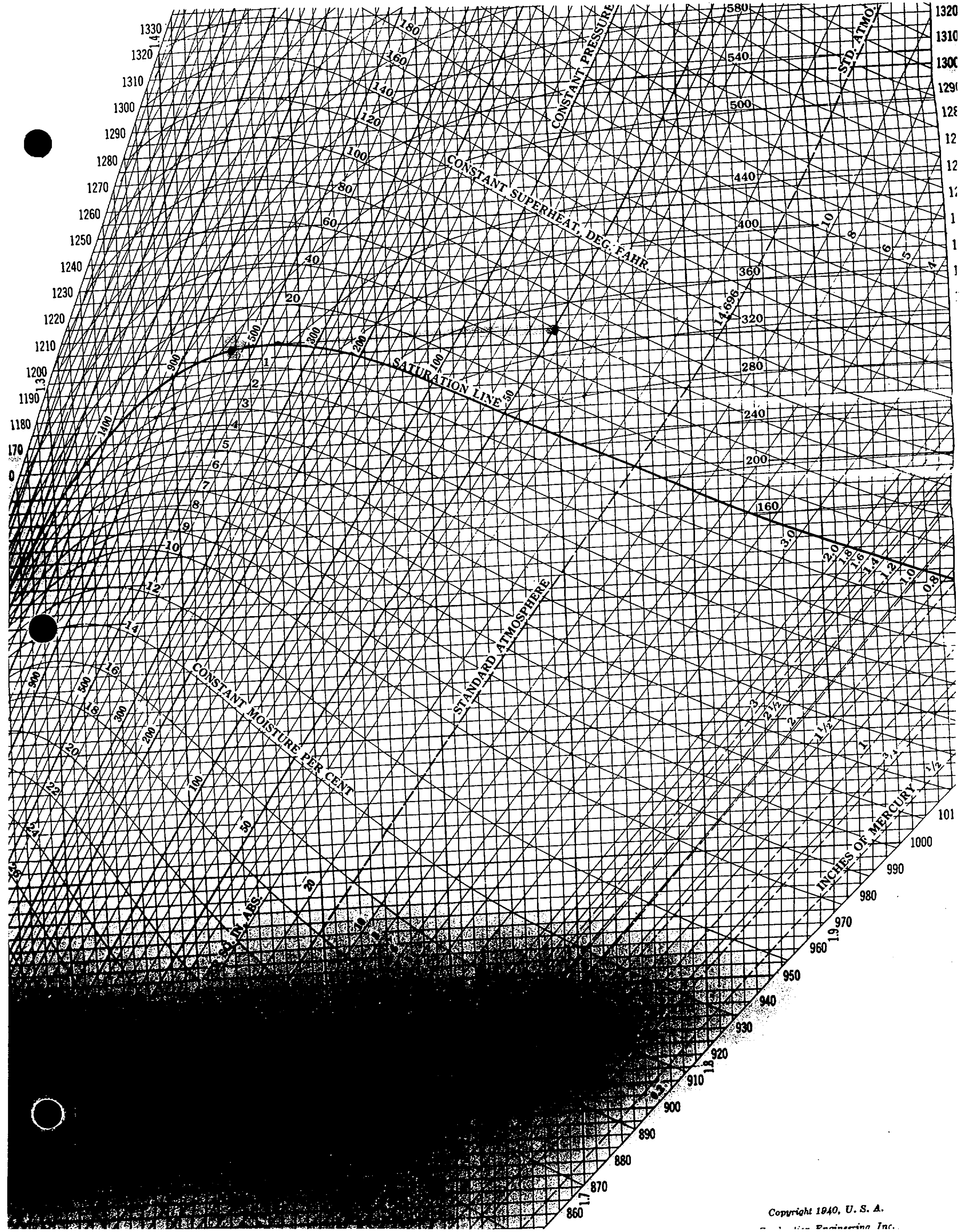
c. 335°F

d. 281°F

ANSWER: c

[SRO: Tier 1/Group 2]

Answer correct: Per Mollier diagram this the correct temperature for these conditions.	Distractors plausible: a – correct if candidate converts 35 psig to psia by subtracting 15 instead of adding; b – correct if candidate doesn't convert 35 psig to psia, but extrapolates to find T_{sat} for 35 psia instead; d – correct if candidate determines T_{sat} for 615 psia.	Distractors incorrect: all incorrect due to not being the value per the Mollier Diagram.
K/A:APE008-AK1.01	Objective: 1403	Source: New (also, see Vogtle NRC exam, Dec. '99)
Reference: Steam tables; ND-83-LP-5	Level: Comprehension	


CONSTANT PRESSURE
STD. ATMO.
CONSTANT SUPERHEAT, DEG. FAHR.
14.696
SATURATION LINE
CONSTANT MOISTURE PER CENT
STANDARD ATMOSPHERE
INCHES OF MERCURY

QUESTION 108: (1.0)

The team is responding to a LOCA and the following plant conditions exist:

- Reactor trip occurred 20 minutes ago.
- 1-ECA-1.1, Loss of Emergency Coolant Recirculation, is completed through step 15.
- **No** RCPs are running and the team was unable to establish CC flow to containment.
- **One** charging pump is running.
- SI flow = 280 gpm.
- The team was unable to start either LHSI pump.
- RVLIS full-range = 69%.
- RCS subcooling = 86°F.
- Core exit TCs are decreasing.
- Containment pressure = 18 psia.
- Containment high-range radiation recorder = 75%.

Using the reference provided, determine the crew's next course of action in accordance with 1-ECA-1.1, Loss of Emergency Coolant Recirculation.

a. Reset CLS isolation signals, stop LHSI pumps, isolate HHSI, and align normal charging.

b. Depressurize the RCS, check if RHR can be placed in service and continue with the procedure.

c. Raise RCS makeup flow to maintain RVLIS indication.

d. Recommend, to the TSC, throttling SI flow to 150 gpm.

ANSWER: b

[SRO: Tier 1/Group 2]

Answer correct: CTMT adverse setpoints apply, so subcooling is below the minimum required to terminate SI flow; SI flow is below the minimum required (290 gpm, per the attachment); team needs to start one additional charging pump and go to step 21 to verify adequate RCS makeup flow (yes) then depressurize the RCS and continue with the ECA.	Distractors plausible: a – candidate fails to recognize that CTMT adverse setpoints apply and therefore believes that subcooling is adequate; c – candidate misinterprets the procedure step not realizeing that RVLIS level is adequate. D – candidate fails to recognize that CTMT adverse setpoints apply and therefore believes that subcooling is adequate; also, candidate forgets that no LHSI pumps are running (as stated in the question stem).	Distractors incorrect: all – CTMT adverse setpoints apply, so subcooling is below the minimum required to terminate SI flow; SI flow is below the minimum required (290 gpm, per the attachment); team needs to start one additional charging pump and go to step 21 to verify adequate RCS makeup flow (yes) then depressurize the RCS and continue with the ECA
K/A: EPE.E11-EA2.2	Objective: 2950	Source: New
Reference:1-ECA-1.1 ND-95.3-LP-20	Level: Comprehension	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*16.	__CHECK IF SI CAN BE TERMINATED:	
	a) Check RVLIS indication: • Full range - GREATER THAN 63% IF NO RCP RUNNING OR • Dynamic range - GREATER THAN 36% IF ONE RCP RUNNING	a) GO TO Step 21.
	b) RCS subcooling based on CETCs - GREATER THAN 80°F [135°F]	b) IF minimum SI flow required as determined from Attachment 2 is less than or equal to 150 gpm, THEN GO TO Step 18. IF minimum SI flow required as determined from Attachment 2 is greater than 150 gpm, THEN do the following: 1) Consult with TSC to determine if SI valves should be throttled, using Attachment 3 to remove seal-in contacts from MOVs. 2) GO TO Step 21.
*17.	__CHECK IF CLS CAN BE RESET:	
	a) CTMT pressure - LESS THAN 14 PSIA	a) GO TO Step 18. WHEN CTMT pressure less than 14 psia, THEN do Steps 17b.
	b) Reset both trains of CLS if necessary	
18.	__STOP LHSI PUMPS AND PUT IN AUTO	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19.	__ISOLATE HHSI TO COLD LEGS:	
	a) Check CHG pump miniflow RECIRC valves - OPEN • 1-CH-MOV-1275A • 1-CH-MOV-1275B • 1-CH-MOV-1275C • 1-CH-MOV-1373	a) Manually open valves.
	b) Close HHSI to Cold Leg • 1-SI-MOV-1867C • 1-SI-MOV-1867D • 1-SI-MOV-1842	
20.	__ESTABLISH CHARGING FLOW:	
	a) Close CHG flow control • 1-CH-FCV-1122	
	b) Verify CHG line isolation - OPEN • 1-CH-HCV-1310A	b) Manually open valve.
	c) Open CHG line isolation MOVs • 1-CH-MOV-1289A • 1-CH-MOV-1289B	c) Locally open valve(s).
	d) Establish desired charging flow using CHG flow control	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21.	__VERIFY ADEQUATE RCS MAKEUP FLOW:	
	a) Check RVLIS indication: • Full range - GREATER THAN 63% IF NO RCP RUNNING OR • Dynamic range - GREATER THAN 36% IF ONE RCP RUNNING	a) Raise RCS makeup flow to maintain RVLIS indication as necessary.
	b) CETCs - STABLE OR DECREASING	b) Raise RCS makeup flow to maintain CETCs stable or decreasing.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: Voiding may occur in the RCS during RCS depressurization. Voiding will result in a rapidly increasing PRZR level.

* *

22. __DEPRESSURIZE RCS TO REDUCE SUBCOOLING:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a) Check RCS subcooling based on CETCs - GREATER THAN 30°F [85°F]	a) GO TO Step 23.
b) Use normal PRZR spray	b) IF normal spray NOT available, THEN use one PRZR PORV. IF RCS can NOT be depressurized using any PRZR PORV, THEN use auxiliary spray.
c) Depressurize RCS until EITHER of the following conditions satisfied: • RCS subcooling based on CETCs - BETWEEN 30°F [85°F] AND 40°F [95°F] OR • PRZR level - GREATER THAN 68% [60%]	c) IF RCS subcooling is less than 30°F [85°F], THEN raise RCS makeup flow as necessary to restore subcooling.
d) Stop RCS depressurization	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
23.	__CHECK IF RHR SYSTEM CAN BE PLACED IN SERVICE:	
	a) Consult with TSC to determine if RHR should be warmed	
	b) Check the following: • RCS hot leg temperatures - LESS THAN 350°F • RCS pressure - LESS THAN 450 PSIG [325 PSIG]	b) GO TO Step 24.
	c) Consult with TSC to determine if RHR should be placed in service	
24.	__CHECK IF OVERPRESSURE MITIGATION SYSTEM CAN BE PLACED IN SERVICE:	
	a) Check RCS pressure - LESS THAN 365 PSIG • PI-1-403 (NQ)	a) GO TO Step 25. <u>WHEN</u> RCS pressure is less than 365 psig. <u>THEN</u> do Steps 24b and 24c.
	b) Check PRZR PORV block valves - OPEN	b) Open valves.
	c) Put both Overpressure Mitigation system key switches in - ENABLE (keys 53 and 54)	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*16.	__CHECK IF SI CAN BE TERMINATED:	
	a) Check RVLIS indication: • Full range - GREATER THAN 63% IF NO RCP RUNNING OR • Dynamic range - GREATER THAN 36% IF ONE RCP RUNNING	a) GO TO Step 21.
	b) RCS subcooling based on CETCs - GREATER THAN 80°F [135°F]	b) IF minimum SI flow required as determined from Attachment 2 is less than or equal to 150 gpm, THEN GO TO Step 18. IF minimum SI flow required as determined from Attachment 2 is greater than 150 gpm, THEN do the following: 1) Consult with TSC to determine if SI valves should be throttled, using Attachment 3 to remove seal-in contacts from MOVs. 2) GO TO Step 21.
*17.	__CHECK IF CLS CAN BE RESET:	
	a) CTMT pressure - LESS THAN 14 PSIA	a) GO TO Step 18. WHEN CTMT pressure less than 14 psia, THEN do Steps 17b.
	b) Reset both trains of CLS if necessary	
18.	__STOP LHSI PUMPS AND PUT IN AUTO	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19.	__ISOLATE HHSI TO COLD LEGS:	
	a) Check CHG pump miniflow RECIRC valves - OPEN • 1-CH-MOV-1275A • 1-CH-MOV-1275B • 1-CH-MOV-1275C • 1-CH-MOV-1373	a) Manually open valves.
	b) Close HHSI to Cold Leg • 1-SI-MOV-1867C • 1-SI-MOV-1867D • 1-SI-MOV-1842	
20.	__ESTABLISH CHARGING FLOW:	
	a) Close CHG flow control • 1-CH-FCV-1122	
	b) Verify CHG line isolation - OPEN • 1-CH-HCV-1310A	b) Manually open valve.
	c) Open CHG line isolation MOVs • 1-CH-MOV-1289A • 1-CH-MOV-1289B	c) Locally open valve(s).
	d) Establish desired charging flow using CHG flow control	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21.	__VERIFY ADEQUATE RCS MAKEUP FLOW:	
	a) Check RVLIS indication: • Full range - GREATER THAN 63% IF NO RCP RUNNING OR • Dynamic range - GREATER THAN 36% IF ONE RCP RUNNING	a) Raise RCS makeup flow to maintain RVLIS indication as necessary.
	b) CETCs - STABLE OR DECREASING	b) Raise RCS makeup flow to maintain CETCs stable or decreasing.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: Voiding may occur in the RCS during RCS depressurization. Voiding will result in a rapidly increasing PRZR level.

* *

22. __DEPRESSURIZE RCS TO REDUCE SUBCOOLING:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a) Check RCS subcooling based on CETCs - GREATER THAN 30°F [85°F]	a) GO TO Step 23.
b) Use normal PRZR spray	b) IF normal spray NOT available, THEN use one PRZR PORV. IF RCS can NOT be depressurized using any PRZR PORV, THEN use auxiliary spray.
c) Depressurize RCS until EITHER of the following conditions satisfied: • RCS subcooling based on CETCs - BETWEEN 30°F [85°F] AND 40°F [95°F] OR • PRZR level - GREATER THAN 68% [60%]	c) IF RCS subcooling is less than 30°F [85°F], THEN raise RCS makeup flow as necessary to restore subcooling.
d) Stop RCS depressurization	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
23.	__CHECK IF RHR SYSTEM CAN BE PLACED IN SERVICE:	
	a) Consult with TSC to determine if RHR should be warmed	
	b) Check the following: • RCS hot leg temperatures - LESS THAN 350°F • RCS pressure - LESS THAN 450 PSIG [325 PSIG]	b) GO TO Step 24.
	c) Consult with TSC to determine if RHR should be placed in service	
24.	__CHECK IF OVERPRESSURE MITIGATION SYSTEM CAN BE PLACED IN SERVICE:	
	a) Check RCS pressure - LESS THAN 365 PSIG • PI-1-403 (NQ)	a) GO TO Step 25. <u>WHEN</u> RCS pressure is less than 365 psig, <u>THEN</u> do Steps 24b and 24c.
	b) Check PRZR PORV block valves - OPEN	b) Open valves.
	c) Put both Overpressure Mitigation system key switches in - ENABLE (keys 53 and 54)	

QUESTION 109: (1.0)

During a reactor startup with Intermediate range indication at 1.5×10^{-11}, N-31 fails high.

Which ONE of the following describes the impact on the Unit startup?

a. Power can be held at 1.5×10^{-11} provided N-31 is returned to operable status within 48 hours.

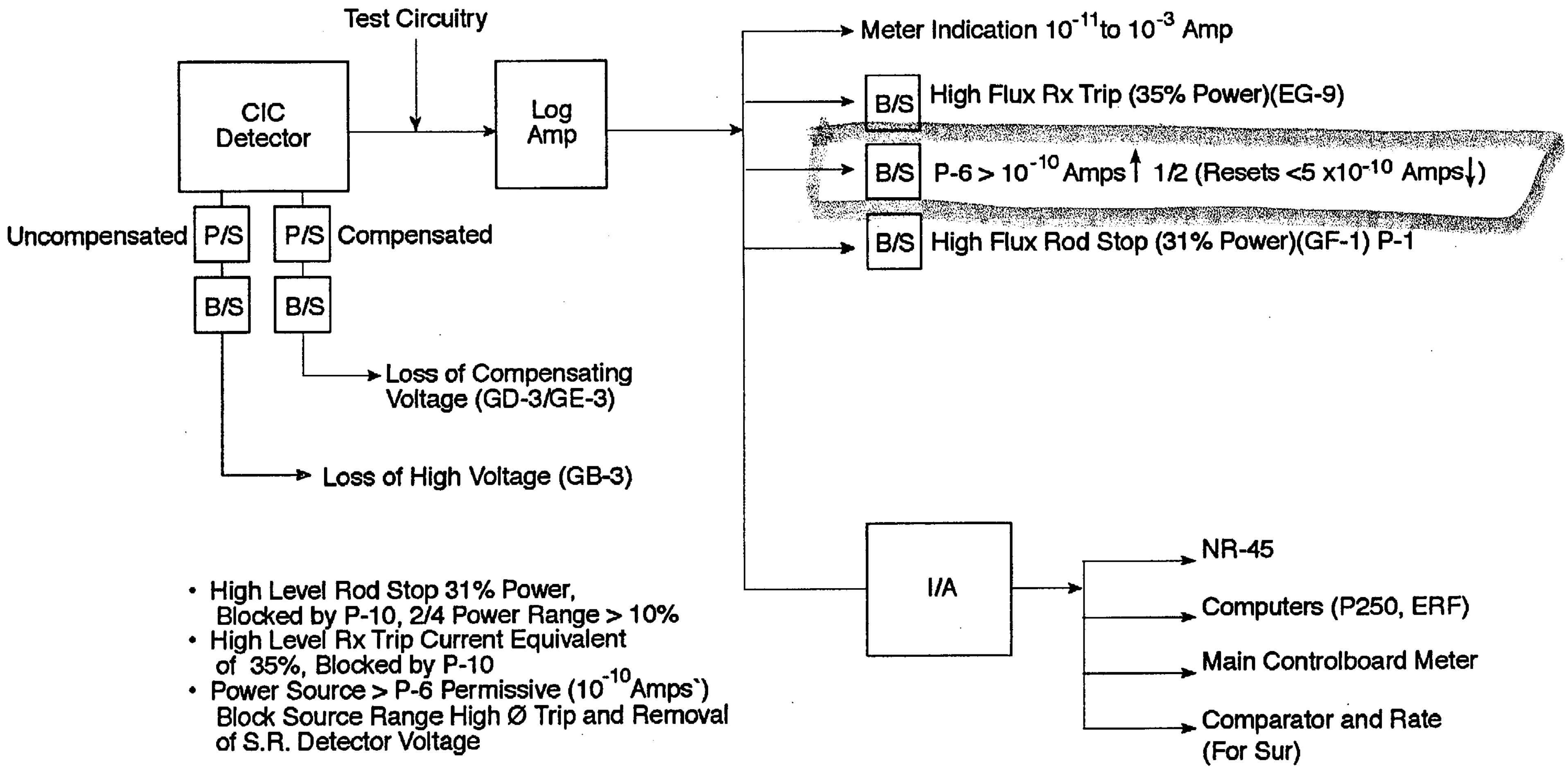
b. Immediate manual reactor trip and performance of E-0 is required.

c. The reactor trip breakers must be opened.

d. An automatic reactor trip will occur.

ANSWER: d

Answer correct: the reactor trip setpoint for a Source Range is 10e5 cps which can not have been blocked until intermediate range is greater than 10e10 amps.	Distractors plausible: a – this would be true if N-31 had failed low. b/c – trainee misunderstanding of AP-4.00.	Distractors incorrect: all – the unit will automatically trip.
K/A: APE032.AK3.01	Objective: 2551	Source: NEW
Reference: AP-4.00, ND-93.2-LP-3	Level: Comprehension	



- High Level Rod Stop 31% Power, Blocked by P-10, 2/4 Power Range > 10%
- High Level Rx Trip Current Equivalent of 35%, Blocked by P-10
- Power Source > P-6 Permissive (10^{-10} Amps`) Block Source Range High Ø Trip and Removal of S.R. Detector Voltage

GraphicNo. - SV214A

IR DETECTOR CIRCUIT

QUESTION 110: (1.0)

The team is performing a unit cooldown and depressurization in accordance with AP-24.01, Large Steam Generator Tube Leak. The Unit is currently at 800 psig in the RCS and the RO has recognized 1-RC-PT-1458 (Pressurizer Narrow Range Pressure) is failed high.

What complication will this present to the operating team?

a. All SI accumulators will not be capable of being closed.

b. The "A" Train of ICCM cannot calculate the Margin to Saturation.

c. Both trains of Over Pressure Mitigation System (OPMS) will be unavailable.

d. RHR cannot be placed in service.

ANSWER: d

Answer correct: This transmitter provides an open permissive to RH-MOV-1701 at less than 460 psig. Since it is failed high (>1000 psig) the valve can not be opened.	Distractors plausible: a – the accumulators are interlocked with an RCS pressure signal. B – The ICCM does receive an RCS pressure input. C – OPMS will be affected.	Distractors incorrect: a – Will not affect accumulator isolation as this comes from the protection transmitters. B – Will not affect ICCM because this comes from the wide range pressure transmitters. C – only one train of OPMS will be affected.
K/A: APE037.AA1.09	Objective: 1684	Source: NEW
Reference: AP-24.01, ND-88.1-LP-3	Level: Comprehension	

c. The common 0.75" line contains a restricting orifice which is sized to limit the rate of mass loss from the RCS, should the vent valves fail in the open position. With a full system differential pressure between the RCS and Containment, the line size ensures that the maximum rate of loss does not exceed the makeup capability of one charging pump.

d. The vent valves are designed to fail in the shut position and are manually controlled from the MCR.

8. Pressurizer Instrumentation

a. The pressurizer is monitored for pressure, level, and temperature. These signals are sent to the protection systems, control systems, and annunciators. The pressure control transmitters (1-RC-PT-1444 and 1445) are Rosemount type pressure transmitters. The pressure transmitters are attached to the reference legs of the pzr level cells. The output is used to generate the control signals for the pressurizer pressure control subsystem. 1-RC-PT-1458 is a narrow range (0 - 1000 psig) meter and provides input to the overpressure mitigation system (OPMS) and the 1-RH-MOV-1701 valve interlock.

b. The pressure transmitters 1-RC-PT-1455, -1456, and -1457 for the pressure protection subsystem are also Rosemount transmitters. Their output is used to generate the SI initiation signal and the high and low pressure reactor trip signals. The level control and protection subsystem uses differential pressure transmitters to measure the liquid level in the pzr vessel. The transmitters 1-RC-LT-1459, -1460, and -1461 provide input to both the level protection and level control subsystems. The transmitter LT-1462 provides a "cold" calibrated (140°F) level indication for use during CSD and provides no control or protection outputs.

c. The pressurizer is monitored for temperature in the liquid as well as the vapor

QUESTION 111: (1.0)

Unit 1 is operating at 100% power when 1-FW-P-1B trips due to an oil leak from the outboard bearing. The RO performs all appropriate actions.

Which ONE of the following automatic start signals will <u>initially</u> close the 1-FW-P-3A pump breaker?

a. AMSAC.

b. Steam generator low-low level.

c. Steam flow > feed flow with low S/G level.

d. 2 of 2 breakers open on 1 of 2 MFW pumps.

ANSWER: b

[SRO: Tier 1/Group 2]

Answer correct: with only 1 MFW pump available, AP-21 requires the unit to be manually tripped; S/G levels will shrink below the low-low level setpoint and start all AFW pumps.	Distractors plausible: a – AMSAC **will** actuate; c – steam flow > feed flow with low S/G level **will** occur; d – AFW pumps **will** auto-start based on MFW pump breaker position.	Distractors incorrect: a – AMSAC actuates, but **after** the S/G low-low level signal occurs; c – SF>FF with low S/G level only trips the reactor, it doesn't start AFW pumps; d – The correct logic is 1 of 2 breakers open on 2 of 2 MFW pumps.
K/A: 054-AA2.03	Objective: 2050	Source: New
Reference: ND-89.3-LP-4	Level: Comprehension	

1-FW-P-2

*LO-LO LEVEL 2/3 ch<17% NR IN ANY 2/3 S/Gs

*LOSS OF VOLTAGE ON 2/3 4160V STATION SERVICE BUSES

*AMSAC INITIATION ON 2/3 CH <13% IN ANY 2/3 S/Gs AND BOTH 1ST STAGE PRESSURES >37%

NOTE: After the AMSAC signal is initiated, the AFW pumps will continue to run until the AMSAC signal is manually reset.

1-FW-3A,3B

*LO-LO LEVEL 2/3 ch <17% NR IN ANY S/G

*LOSS OF VOLTAGE ON 2/2 RSS (X-FER BUSES) for affected unit

*ANY SI SIGNAL (AFTER 50 SEC T.D.)

*1/2 MFP BKRS OPEN ON BOTH MFPs

*AMSAC INITIATION

g. In the event an undervoltage condition occurs on a 4160v emergency bus after an SI or Hi-Hi CLS event has been initiated, the respective motor driven AFW pump will trip, and the automatic and manual start signals will be momentarily blocked (10 sec. for an SI; 140 sec. for a Hi-Hi CLS). The pump will auto-start again after the blocking signal is removed (times-out). This load sequencing will stagger the emergency loads starting on EDG, thus preventing an overload condition.

h. The turbine driven AFW pump will remain running after an AUTO START, even if the AUTO START signals clear, until the operator places the control switches for both PCV-MS-102 A & B to OPEN/RESET then returns them to the close position.

QUESTION 112: (1.0)

Which ONE of the following Emergency Diesel Generator components requires 125VDC power to perform its function?

a. Governor booster pump.

b. Fuel transfer pump.

c. Soak back oil pump.

d. Radiator louvers.

ANSWER: a

Answer correct: the governor booster pump is supplied from the EDG battteries	Distractors plausible: b/c/d – all of these items are EDG support items that receive power from the EDG electrical systems	Distractors incorrect: b/c/d – these are powered from the EDG MCC.
K/A: APE058.AK3.01	Objective: 2240	Source: NEW
Reference: ND- 90.3-LP-1	Level: Knowledge	

(b) When the normal source voltage is restored to > 90% of normal voltage, a 30 minute time delay is started. If the voltage remains > 90% of nominal, at the end of the time delay the ABT transfers to the normal source.

(c) If the alternate source voltage is lost during the 30 minute time delay, the ABT will transfer immediately to the normal source.

(d) The ABT will not transfer to a source that is < 440V (90% of normal).

Ask trainees: What source of power is required to start an emergency diesel generator? Answer: Emergency diesel 125v DC.

b. Emergency Diesel Generator 125v DC System

(1) This system consists of a 480v battery charger, a battery system, and wiring to the remote excitation cabinet.

(2) The DC bus feeds the following:

(a) Start Ckt #1 and start Ckt #2. Each of which supplies its own governor booster pump, engine start/stop controls and relays, and diesel protective relays.

(b) The EDG Annunciator Panel on the ECC. For the "Power Available" light on the ECC annunciator panel to be lit, the circuit breakers for start ckt #1 & #2, Alarm power, DC control power, and AC control power must be closed.

(c) The engine & governor control Ckt (D.C. control Ckt breaker) which supplies power to the governor control, the D.C. fuel oil pump, and field flash controls.

(d) D.C. for flashing the generator field.

Ask trainees: How does the operator know that the EDG battery has enough potential to start the booster pumps for the UG8 governor, start DC fuel oil pump, and the flash field? Answer: If the battery potential is low, the battery low voltage alarm is received in the Main Control Room (KF7). Ask trainees: At what remote location can EDG battery voltage be determined? Answer: #2 ESGR battery voltage recorders.

c. Emergency Generator Exciter Output

(1) The emergency diesel generator requires an external source of power to energize its field windings.

QUESTION 113: (1.0)

Given the following plant conditions:

- Unit 1 is at 100% power.
- Five minutes ago "F" xfer bus tripped.
- 1H bus was re-energized by its associated diesel.
- 20 seconds ago "D" transfer bus tripped on fault.
- Unit 1 remains at 100% power.

Which ONE of the following identifies the running Unit 1 AFW pump(s)?

a. "A" and "B" motor driven only.

b. Terry Turbine and "A" motor driven only.

c. Terry Turbine and "B" motor driven only.

d. "A" motor driven only.

ANSWER: d

[SRO: Tier 1/Group 3]

Answer correct: Both motor driven AFW pumps will receive an auto start signal from the station blackout signal but only the "A" pump will have power to run.	Distractors plausible: a – trainee not determining that the J bus will be de-energized. B/c – trainee misunderstanding the auto start signals for the Terry Turbine.	Distractors incorrect: a – the J bus will be de-energized; therefore, the "B" motor driven pump will not start. B/c – the terry turbine will not receive an autostart signal.
K/A: APE056-AA1.10	Objective: 2050	Source: New
Reference: ND-89.3-LP-4	Level: Comprehension	

1-FW-P-2

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*LOSS OF VOLTAGE ON 2/3 4160V STATION SERVICE BUSES

*AMSAC INITIATION ON 2/3 CH <13% IN ANY 2/3 S/Gs <u>AND</u> BOTH 1ST STAGE PRESSURES >37%

NOTE: After the AMSAC signal is initiated, the AFW pumps will continue to run until the AMSAC signal is manually reset.

1-FW-3A,3B

*LO-LO LEVEL 2/3 ch <17% NR IN ANY S/G

*LOSS OF VOLTAGE ON 2/2 RSS (X-FER BUSES) for affected unit

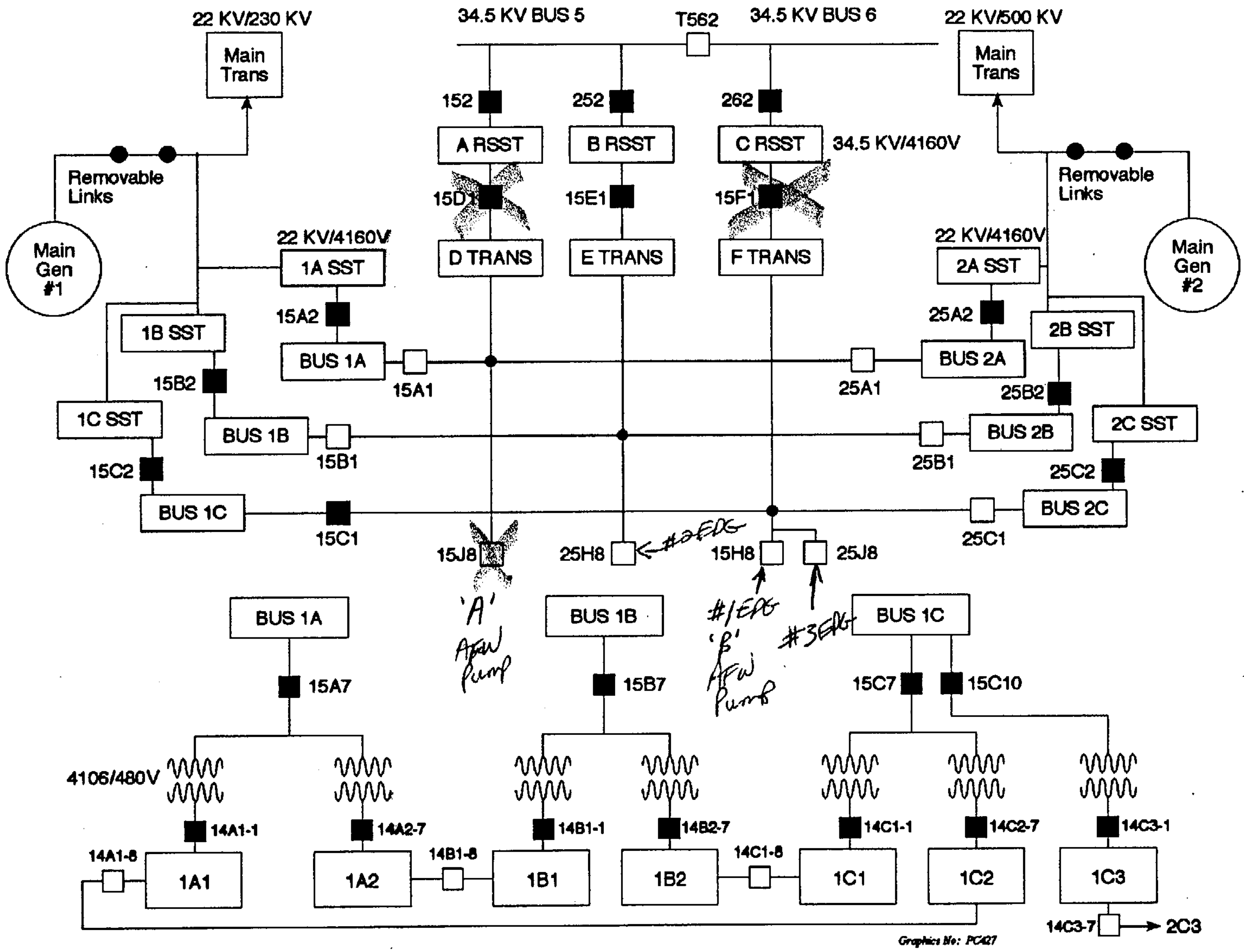
*ANY SI SIGNAL (AFTER 50 SEC T.D.)

*1/2 MFP BKRS OPEN ON BOTH MFPs

*AMSAC INITIATION

g. In the event an undervoltage condition occurs on a 4160v emergency bus after an SI or Hi-Hi CLS event has been initiated, the respective motor driven AFW pump will trip, and the automatic and manual start signals will be momentarily blocked (10 sec. for an SI; 140 sec. for a Hi-Hi CLS). The pump will auto-start again after the blocking signal is removed (times-out). This load sequencing will stagger the emergency loads starting on EDG, thus preventing an overload condition.

h. The turbine driven AFW pump will remain running after an AUTO START, even if the AUTO START signals clear, until the operator places the control switches for both PCV-MS-102 A & B to OPEN/RESET then returns them to the close position.



STATION SERVICE DISTRIBUTION

QUESTION 114: (1.0)

Following replacement of the packing on RWST chemical addition tank outlet valve 1-CS-MOV-102B, the valve is unisolated and a large spill of sodium hydroxide occurs.

Which ONE of the following individuals is responsible for coordinating the response effort?

a. Station Emergency Manager.

b. Shift Supervisor.

c. Recovery Manager.

d. Environmental Compliance Coordinator.

ANSWER: b

[SRO: Tier 2/Group 1]

Answer correct: per VPAP-2203, the Shift Supervisor coordinates the response to a hazardous substance spill.	Distractors plausible: a – the SEM functions to direct emergency response activities; c – after the LEOF is manned, the RM assumes some of the duties of the SEM during the response to an emergency; d – the Environmental Compliance Coordinator must be notified of all oil and hazardous substance spills.	Distractors incorrect: a & c – a hazardous substance spill does not require activation of the emergency response organization; d – the Environmental Compliance Coordinator is not responsible for directing the response to a hazardous substance spill.
K/A: 013/GEN-2.1.26	Objective: 8018	Source: New
Reference: VPAP-2203	Level: Knowledge	

5.3.3 Professional Engineering support for certification of the SPCC Plan in compliance with 40 CFR 112 and certification of secondary containment areas in compliance with 9 VAC 25-91-130.A.4.

5.4 Shift Supervisor

The Shift Supervisor is responsible for:

5.4.1 Notifying Virginia Power personnel and regulatory agencies, as required by this procedure and by VPAP-2802, Notifications and Reports.

5.4.2 Organizing response teams to stop, contain, and clean up oil discharges.

5.4.3 Coordinating containment and cleanup response.

5.5 Superintendent - Nuclear Site Services

The Superintendent - Nuclear Site Services is responsible for:

5.5.1 Providing manpower and equipment as directed by the Shift Supervisor for corrective action and spill response.

5.5.2 Providing trenching equipment and a trenching equipment operator to stand-by at the Vehicle Maintenance Garage when fuel deliveries are made at the Vehicle Maintenance Garage. **(North Anna)**

5.6 Superintendent - Radiological Protection

The Superintendent - Radiological Protection is responsible for providing information and support to respond to oil discharges that involve radiological contamination.

5.7 Superintendent - Nuclear Training

The Superintendent - Nuclear Training is responsible for administering and maintaining training requirements in accordance with the Oil Spill Prevention, Control and Countermeasures (SPCC) Plan.

5.8 Supervisor Facilities and Support

The Supervisor Facilities and Support is responsible for:

5.8.1 Maintaining and providing personnel trained in oil discharge cleanup.

5.8.2 Staging oil spill van at the Vehicle Motor Pool when gasoline deliveries are made to the Vehicle Motor Pool. **(North Anna)**

QUESTION 115: (1.0)

Given the following plant conditions:

- During core off-load, a fuel assembly is damaged while being placed in the SFP rack.
- The fuel handlers are unaware of the damage.
- Several days later, an increasing trend is noted on the SFP bridge crane area radiation monitor.
- The running SFP cooling pump automatically trips.

Which ONE of the following is **NOT** a reason why increased **off-site** exposure would result? Assume no operator actions.

a. Boiling of the SFP water releases particulate radioactivity to the atmosphere.

b. Overheating and subsequent failure of additional fuel assemblies.

c. Loss of purification flow increases activity of SFP water.

d. Loss of shutdown margin due to boron stratification.

ANSWER: d

[SRO: Tier 2/Group 2]

Answer correct: assuming the SFP water boron concentration is initially homogeneous, boron stratification would only occur if undiluted water were subsequently added.	Distractors plausible: a – candidate misconception concerning the potential for decay heat to cause boiling of SFP water, or the increased particulate release due to boiling; b – candidate misconception concerning the potential for decay heat to cause overheating of spent fuel assemblies; c – failure to realize that piping configuration requires SFP cooling to be in service before the purification system can be placed in service.	Distractors incorrect: a – loss of SFP cooling eventually results in boiling, which does release increased amounts of particulate activity to the atmosphere; b – loss of SFP cooling could eventually cause overheating and failure of recently-discharged fuel assemblies, depending on power history; c – SFP/RP piping configuration requires SFP cooling to be in service before the purification system can be placed in service.; loss of RP flow results in increased activity in SFP water, which increases amount of particulate activity released at the surface (even without boiling).
K/A: SYS033-A3.02	Objective: 2499	Source: New
Reference: ND-92.5-LP-6	Level: Comprehension	

LESSON PLAN

Introduction

The Spent Fuel Pit and its support systems keep the deadly spent fuel in a safe condition while it is decaying. The operator must know what equipment is involved to keep the radioactive spent fuel from becoming a danger to himself and the general public. The SFP and Reactor Cavity Purification systems minimize radioactivity and maximize water clarity. This lesson will cover this equipment.

Objectives

After receiving this instruction, the trainee will be able to:

A. Describe the purpose and construction of the Nuclear Fuel Storage Facility.

B. Describe the Spent Fuel Pit Cooling, Purification and Skimmer System.

C. Describe the function and operation of the Reactor Cavity Purification System.

D. Describe the Spent Fuel Pit and Support Systems.

QUESTION 116: (1.0)

With unit 1 in Cold Shutdown, Instrument Technicians are calibrating "A" S/G narrow-range level transmitter 1-FW-LT-1474.

Using the references provided, determine which ONE of the following correctly lists **<u>all of the valves</u>** associated with 1-FW-LT-1474 that the Instrument Technicians are authorized to manipulate.

a. 1-FW-ICV-3009, 3010, 3011, 3013, and 3014 **<u>only</u>**.

b. 1-FW-ICV-3009, 3010, 3011, 3012, 3013, 3014, and 3015 **<u>only</u>**.

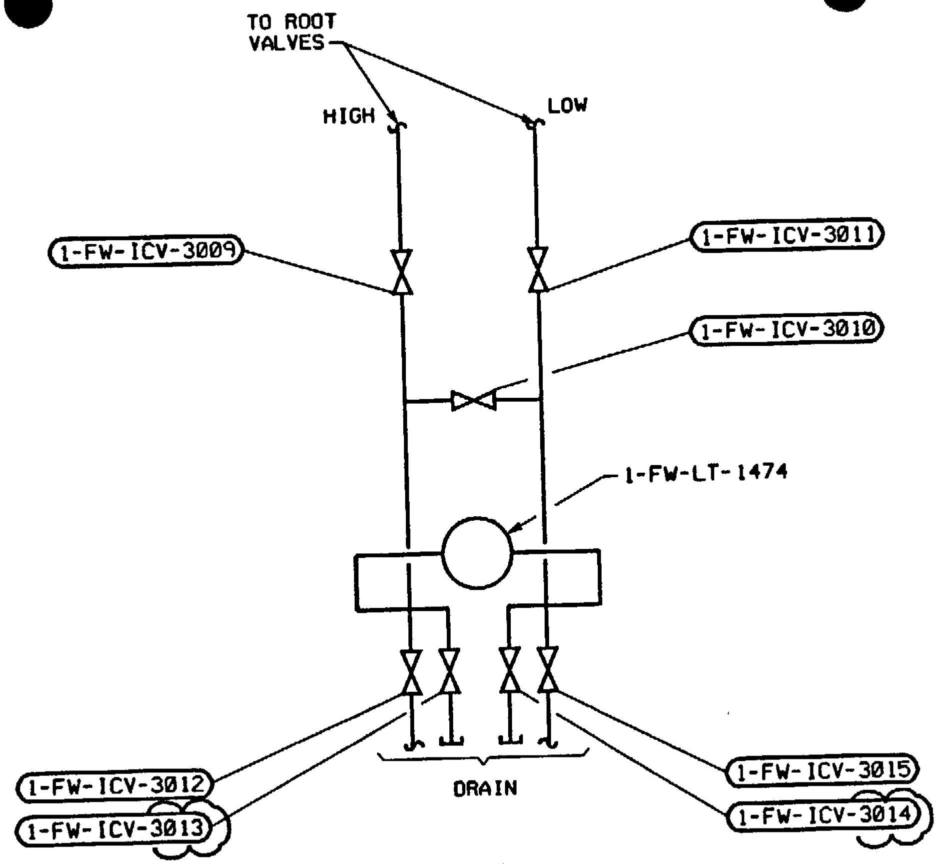
c. 1-FW-ICV-3009, 3010, 3011, 3013, 3014, 1-FW-5 and 6 **<u>only</u>**.

d. 1-FW-ICV-3009, 3010, 3011, 3012, 3015, and 1-FW-5 and 6 **<u>only</u>**.

ANSWER: b

[SRO: Tier 3]

Answer correct: per VPAP-1401, Instrument Technicians can only manipulate isolation valves if no root valve exists; in this example, 1-FW-5 and 6 are the isolation valves (root valves), so they can only be operated by Operations personnel; the Instrument Technicians can manipulate all of the **other** valves associated with the level transmitter.	Distractors plausible: all – candidate misconception concerning the function of the listed valves and the requirements for valve manipulation.	Distractors incorrect: a – does not list all of the instrument valves associated with the level transmitter; c – includes the root valves, and does not list all of the instrument valves associated with the level transmitter; d – includes the root valves.
K/A: GEN-2.1.1	Objective: 8037	Source: New
Reference: VPAP-1401; 1-FK-FW-LT-1474, 11448-FM-68A sheet 1 of 4.	Level: Comprehension	



NOTES:

1. THIS DRAWING IS TO BE USED FOR INSTRUMENT VALVE NUMBERS ONLY. SEE OTHER FK SERIES DRAWINGS FOR INSTRUMENT LOCATION, TUBE CLASS, VALVE TYPES, ETC. AS APPLICABLE.

REFERENCES:

11448-FM-68A, SH. 1

REV	DATE	DESCRIPTION	BY	CHKD	ENGR	APPD
1	/ /93	REVISED PER CONFIG.MAN.WALKDN.	DMR	DW		
0	3/31/92	ORIGINAL ISSUE / CONFIG. MAN.				

VIRGINIA POWER
NORTH CAROLINA POWER

NUCLEAR ENGINEERING SERVICES
RICHMOND, VIRGINIA

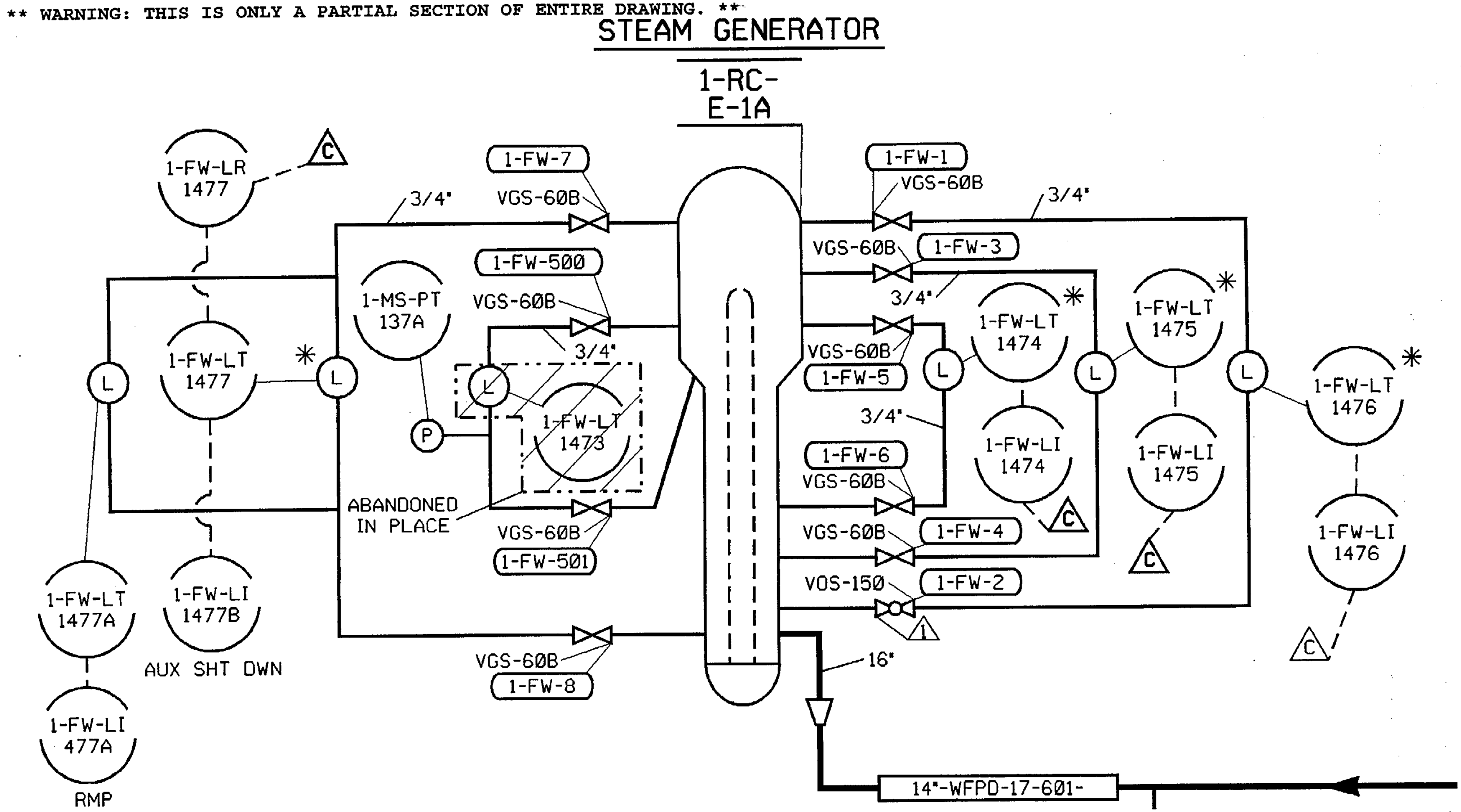
DRWN	RPD
CHKD	EESII
ENGR	TMN
APPD	ARF

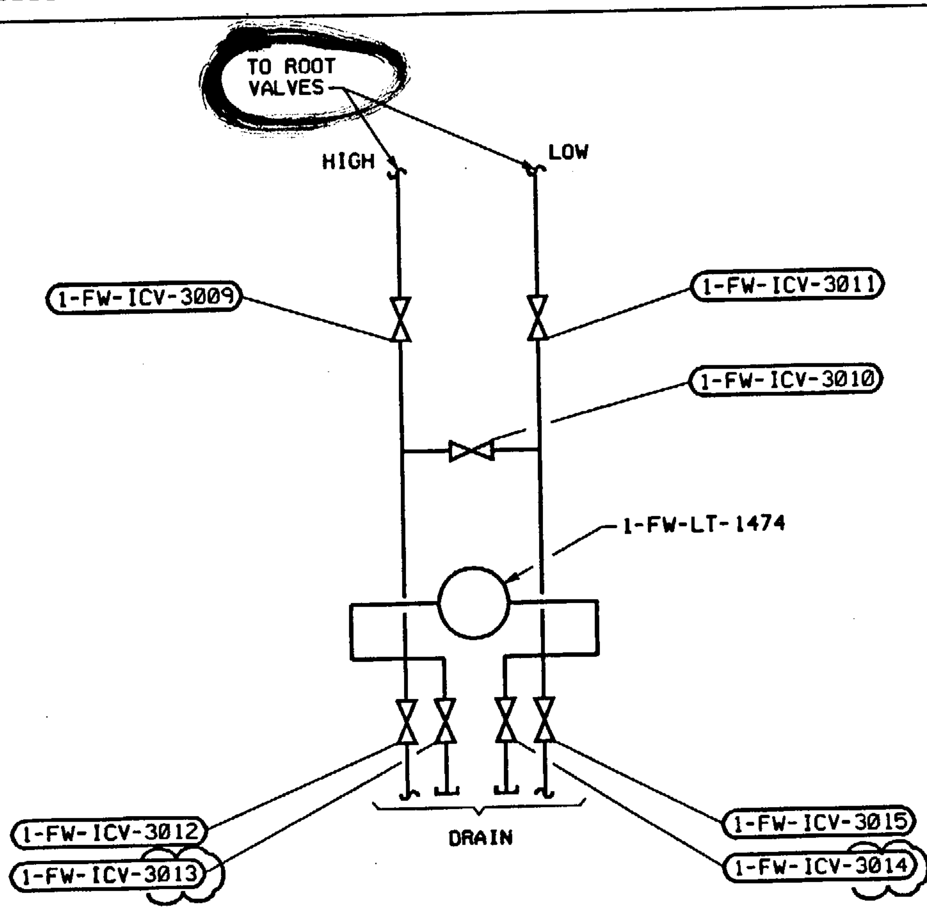
INSTRUMENT VALVE NUMBERS FOR
1-FW-LT-1474
SURRY POWER STATION - UNIT 1

DATE 3/31/92 | CAD NO. (WINDLES) 13604221LT1474.FW | DRAWING NO. 11448-FK-FW-LT-1474 | SH 1 OF 1 | REV 1

5-MAR-1997 15:01 PC:NA2

** WARNING: THIS IS ONLY A PARTIAL SECTION OF ENTIRE DRAWING. **





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REV	DATE	DESCRIPTION	BY	CHKD	ENGR	APPD

VIRGINIA POWER
NORTH CAROLINA POWER

NUCLEAR ENGINEERING SERVICES
RICHMOND, VIRGINIA

DRWN	RPD
CHKD	EESII
ENGR	TMN
APPD	ARF

INSTRUMENT VALVE NUMBERS FOR
1-FW-LT-1474
SURRY POWER STATION - UNIT 1

DATE 3/3 2 | CAD NO. (WINDLES) 13604223LT1474.FW | DRAWING NO. 11448-FK-FW-LT-1474 | SH OF 1 | REV 1

PC=NA2

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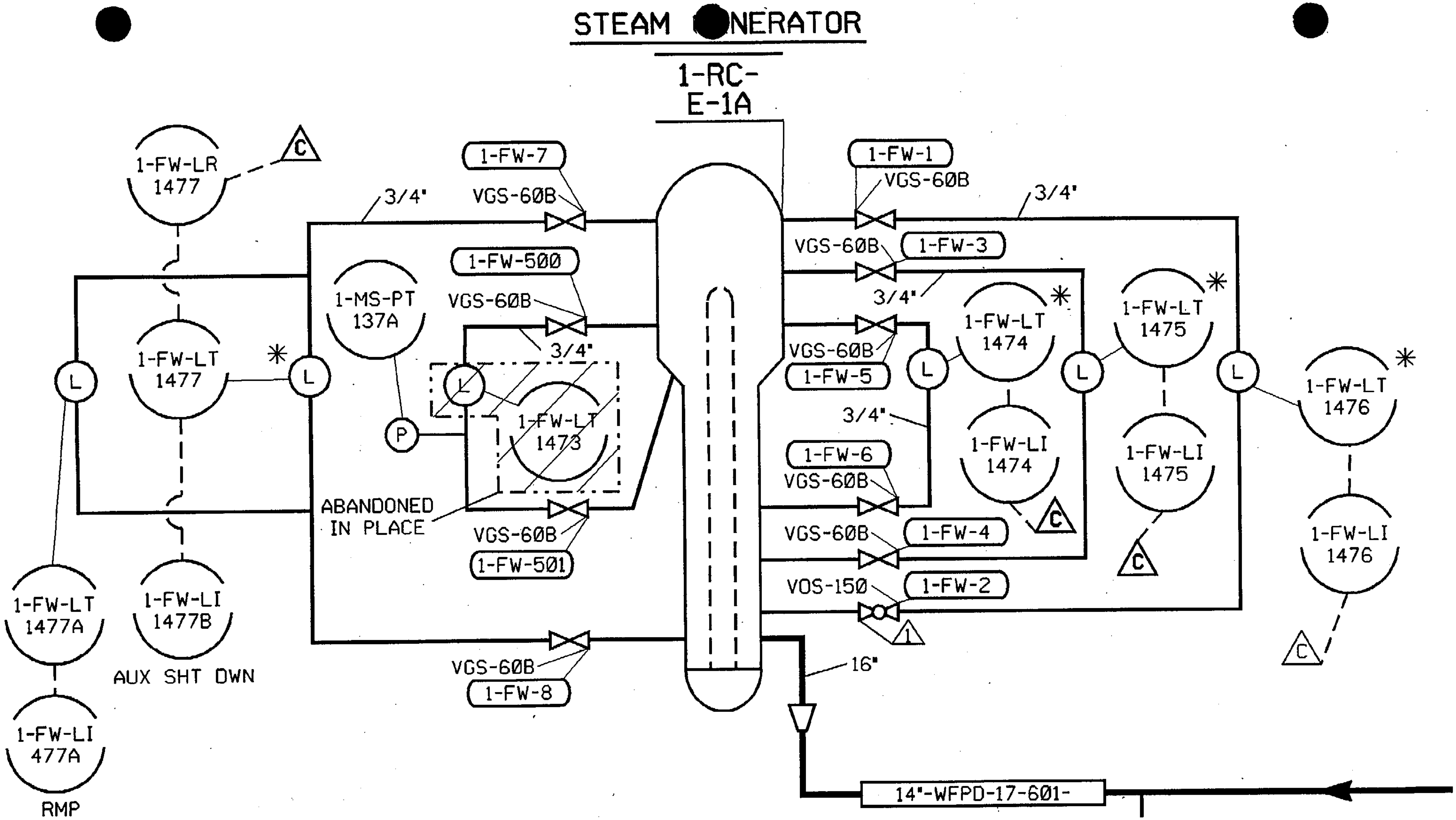

QUESTION 117: (1.0)

Given the following values for RCS chemistry parameters at 100% power:

- Chloride = 53 ppb.
- Fluoride = 44 ppb.
- Hydrogen = 12 cc/kg.
- Oxygen = 72 ppb.

Using the references provided, determine the most limiting action.

a. Immediately initiate unit shutdown; reduce RCS temperature to ≤250°F as quickly as possible.

b. Take action to restore hydrogen to within limit within 24 hours; if not restored, initiate unit shutdown to cold shutdown.

c. Take action to restore hydrogen to within limit within 24 hours; increase monitoring of RCS hydrogen and oxygen, gross beta gamma and suspended solids.

d. Restore oxygen and chloride to within limits within 7 days; if not restored, perform a technical evaluation and implement a program of corrective measures.

ANSWER: a

[SRO: Tier 3]

Answer correct: per VPAP-2201 table 25 (NOTE 4), with hydrogen ≤15 cc/kg and oxygen >50 ppb, plant shutdown should commence IAW action level 3.	Distractors plausible: b – this would be correct if candidate fails to read NOTE 4 of table 25; c – this would be correct without consideration of **oxygen value** as stated in NOTE 4 of table 25; d – oxygen and chloride exceed the action level 1 values; candidate misinterpretation of hydrogen values in table 25.	Distractors incorrect: b– doesn't take into consideration NOTE 4; c – doesn't take into consideration the oxygen value as stated in NOTE 4; d – combination of hydrogen and oxygen being OOS is more limiting
K/A: GEN-2.1.34	Objective: 1299	Source: New
Reference: VPAP-2201	Level: Comprehension	

Table 25
Primary System Chemistry - Power Operation
(Reactor Critical)

Analysis	Frequency	Typical Value	Action Levels			Tech. Spec. Ref.	
			1	2	3	NAPS	SPS
Aluminum (ppb)	(6)	≤ 50	--	--	--	--	--
Boron (ppm)	7/W	Variable	--	--	--	--	--
Calcium (ppb)	(6)	≤ 25	--	--	--	--	--
Chloride (ppb)	3 or 5/W(3)	≤ 50	> 50	> 150	> 1,500	Tb.3.4-1	3.1.F.1.b
Fluoride (ppb)	3 or 5/W(3)	≤ 50	> 50	> 150	> 1,500	Tb.3.4-1	3.1.F.1.c
Hydrogen $\frac{cc(STP)}{kg(H_2O)}$	3/W	≥ 25 - (1) ≤ 50	< 25 or > 50	≤ 15(4)	≤ 5	--	--
Lithium (ppm)	7/W	Within the target band for boron/ lithium(2)	--	--	--	--	--
Magnesium (ppb)	(6)	≤ 25	--	--	--	--	--
Oxygen (ppb)	3 or 5/W(3)	≤ 5	> 5	--	> 100	Tb.3.4-1	3.1.F.1.a
pH @25° C	3/W	Variable	--	--	--	--	--
Silica (ppb)	1/Q	≤ 1000 (5)	--	--	--	--	--
Specific Conductivity (µS/cm)	3/W	Variable	--	--	--	--	--
Sulfate (ppb)	1/W	≤ 50	> 50	> 150	> 1,500	--	--
Suspended Solids (ppb)	1/Q	≤ 100 (7)	> 200 (7)	--	--	--	--

NOTE: (1) To assist reactor coolant degassing, the reactor coolant dissolved hydrogen concentration may be reduced to < 25 cc/kg, but ≥ 15 cc/kg within 24 hours prior to shutdown without entry into Action Level 1.

(2) SURRY-"MODIFIED" Lithium Program [Increasing $pH_{(T)}$ from 6.9 to 7.4]
NORTH ANNA-"Coordinated" lithium program [constant $pH_{(T)}$ 6.9]

(3) NAPS - 3/W, SPS - 5/W

(4) Corrective action is recommended if this value is exceeded but no plant shutdown is suggested. Station Chemistry will increase RCS monitoring of Dissolved Hydrogen and Oxygen, Gross Beta Gamma, and Suspended Solids when operating at or below the Action Level 2 Guideline value of 15cc/kg. IF, while operating with RCS hydrogen between 5 - 15 cc/kg, dissolved oxygen exceeds 50 ppb or TSS and activity exceed 200% of steady state values, THEN, plant shutdown should commence in accordance with Action Level 3 guidelines.

(5) Reduce silica to ≤ 1,000 ppb as soon as possible after reaching 100 percent power. Operation with silica ≥ 1,000 ppb with reactor power at 100 percent is acceptable for up to one month, however, care should be taken to minimize the intrusion of aluminium, calcium, and/or magnesium into the reactor coolant.

(6) NAPS only -Following refueling, if the RCS silica concentration is ≥ 1 ppm, then RCS aluminium, calcium, and magnesium concentrations should be determined prior to criticality.

(7) SPS has a UFSAR suspended solids limit of 1.0 ppm for RHR and RCS.

ATTACHMENT 2

(Page 1 of 3)

Primary Chemistry Action Statements

1. **INTRODUCTION**

The chemistry limits and action levels that have been presented in 6.20 are achievable and appropriate for the protection of system materials, ensuring fuel performance, and controlling radiation field buildup.

Three Action Levels have been defined for remedial actions to be taken when the parameters are confirmed to be outside the control values. These Action Levels are not intended to supersede Technical Specifications but can be used in conjunction with those requirements.

2. **ACTION LEVEL 1**

The Action Level 1 value of a parameter represents the range, outside of which data or engineering judgment indicates that long-term system reliability may be affected, thereby warranting an improvement of operating practices. Action Level 1 values generally represent limits for normal plant operations.

Actions to be taken if a parameter exceeds the Action Level 1 value:

a. Efforts should be made to restore the parameter to within the appropriate limit within seven (7) days.

b. If the parameter has not been restored to within the appropriate range within seven (7) days, a technical evaluation should be performed and a program for implementing corrective measures instituted. Such a program may require equipment additions or modifications over the long-term.

3. **ACTION LEVEL 2**

The Action Level 2 value of parameter represents the value, outside of which data or engineering judgment indicates significant damage could be done to the system in the short-term, thereby warranting a prompt correction of the abnormal condition.

c. Efforts should be made to bring the parameter within the appropriate Action Level 2 value within twenty-four (24) hours.

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d. If the parameter has not been restored to within the Action Level 2 value within 24 hours, an orderly unit shutdown should be initiated and the plant brought to a cold shutdown condition as quickly as permitted by other plant constraints.

- If the chemistry value improves to below the Action Level 2 value prior to plant shutdown, full power operation may be resumed.

e. Following a unit shutdown caused by exceeding the time limit on an Action Level 2 value, a technical evaluation of the incident should be performed and appropriate corrective measures taken before the unit is restarted.

4. **ACTION LEVEL 3**

The Action Level 3 value of a parameter represents the limit beyond which data or engineering judgment indicates that it is inadvisable to continue to operate the plant.

Action to be taken if a parameter exceeds the Action Level 3 value:

f. An orderly unit shutdown should be initiated immediately, with a reduction of the reactor coolant temperature to $\leq 250\ ^{\circ}F$ as rapidly as other plant constraints permit.

g. If the chemistry should improve to below the limits of action Level 3 prior to completing the plant shutdown, power operation may be resumed subject to the requirements defined by the other Action Levels.

h. Following a unit shutdown caused by entering an Action Level 3 condition, a technical evaluation of the incident should be performed and appropriate corrective measures taken before the unit is restarted.

5. **CORRECTIVE ACTIONS**

Corrective actions should be implemented when a parameter is approaching or has exceeded an Action Level value. The following actions may be considered typical of those that can be taken:

i. Verify conditions.

j. Identify and isolate sources of impurities.

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k. Verify that the reactor coolant purification system is in service with the maximum available flow and that the ion exchanger bed removal efficiencies are adequate.

l. Increase sample and analysis frequencies for short-term trending purposes and for confirmatory analyses of critical chemistry parameters (the control parameters).

6. TECHNICAL EVALUATIONS

The technical evaluation process for prolonged abnormal water chemistry conditions should include the following:
The evaluation should address both the cause of the abnormal condition and inform the appropriate levels of management of the existence of the condition, the implications, and the possible corrective measures that can be taken over both the short and long terms.

6.1.7 **Station Valve Operations [Commitment 3.2.3]**

a. Operations Department personnel shall be the only Station personnel authorized to manipulate valves except:

1. Instrument valves (e.g., local isolation valves, equalizing valves, and vents and drains) may be operated by qualified Instrument Technicians. Instrument valves are normally downstream of a root valve and are used to isolate an instrument. If an instrument has only one valve for isolation, that valve shall be used as an instrument valve.

2. Sample valves may be operated by qualified Chemistry and Radiological Control Technicians.

3. Motor Operated Valves (MOVs) that require testing as part of a Maintenance Procedure may be operated by Maintenance Department personnel.

4. Valves that require testing as part of an Engineering Work Request (EWR) or Design Change Package (DCP) may be operated by Testing personnel.

5. Fire Protection valves may be operated by Safety and Loss Prevention personnel with the concurrence of the Operations Shift Supervisor.

6. Valves that are used to control a process (not related to plant operations) may be operated by individuals responsible for that process, as determined by Operations management.

b. Valve operation shall be in accordance with approved Station procedures, skill of the craft, or if action is required to protect Station equipment or personnel.

c. A procedure or process that directs operation of a valve shall include sufficient Shift Supervisor notification.

d. Valves that have Danger Tags attached to them shall not be operated.

6.1.8 **Control Room Notification**

a. Station personnel shall notify the Control Room or authorized SRO before performing any evolution that may affect Unit operation or instrumentation.

b. Station personnel shall notify the Control Room of abnormal conditions or emergencies.

Table 25
Primary System Chemistry - Power Operation
(Reactor Critical)

Analysis	Frequency	Typical Value	Action Levels			Tech. Spec. Ref.	
			1	2	3	NAPS	SPS
Aluminum (ppb)	(6)	≤ 50	--	--	--	--	--
Boron (ppm)	7/W	Variable	--	--	--	--	--
Calcium (ppb)	(6)	≤ 25	--	--	--	--	--
Chloride (ppb)	3 or 5/W(3)	≤ 50	> 50	> 150	> 1,500	Tb.3.4-1	3.1.F.1.b
Fluoride (ppb)	3 or 5/W(3)	≤ 50	> 50	> 150	> 1,500	Tb.3.4-1	3.1.F.1.c
Hydrogen <u>cc(STP)</u> kg(H_2O)	3/W	≥ 25 - (1) ≤ 50	< 25 or > 50	≤ 15(4)	≤ 5	--	--
Lithium (ppm)	7/W	Within the target band for boron/ lithium(2)	--	--	--	--	--
Magnesium (ppb)	(6)	≤ 25	--	--	--	--	--
Oxygen (ppb)	3 or 5/W(3)	≤ 5	> 5	--	> 100	Tb.3.4-1	3.1.F.1.a
pH @25° C	3/W	Variable	--	--	--	--	--
Silica (ppb)	1/Q	≤ 1000 (5)	--	--	--	--	--
Specific Conductivity (μS/cm)	3/W	Variable	--	--	--	--	--
Sulfate (ppb)	1/W	≤ 50	> 50	> 150	> 1,500	--	--
Suspended Solids (ppb)	1/Q	≤ 100 (7)	> 200 (7)	--	--	--	--

NOTE: (1) To assist reactor coolant degassing, the reactor coolant dissolved hydrogen concentration may be reduced to < 25 cc/kg, but ≥ 15 cc/kg within 24 hours prior to shutdown without entry into Action Level 1.

(2) SURRY-"MODIFIED" Lithium Program [Increasing $pH_{(T)}$ from 6.9 to 7.4]
NORTH ANNA-"Coordinated" lithium program [constant $pH_{(T)}$ 6.9]

(3) NAPS - 3/W, SPS - 5/W

(4) Corrective action is recommended if this value is exceeded but no plant shutdown is suggested. Station Chemistry will increase RCS monitoring of Dissolved Hydrogen and Oxygen, Gross Beta Gamma, and Suspended Solids when operating at or below the Action Level 2 Guideline value of 15cc/kg. IF, while operating with RCS hydrogen between 5 - 15 cc/kg, dissolved oxygen exceeds 50 ppb **or** TSS and activity exceed 200% of steady state values, THEN, plant shutdown should commence in accordance with Action Level 3 guidelines.

(5) Reduce silica to ≤ 1,000 ppb as soon as possible after reaching 100 percent power. Operation with silica ≥ 1,000 ppb with reactor power at 100 percent is acceptable for up to one month, however, care should be taken to minimize the intrusion of aluminium, calcium, and/or magnesium into the reactor coolant.

(6) NAPS only -Following refueling, if the RCS silica concentration is ≥ 1 ppm, then RCS aluminium, calcium, and magnesium concentrations should be determined prior to criticality.

(7) SPS has a UFSAR suspended solids limit of 1.0 ppm for RHR and RCS.

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Primary Chemistry Action Statements

1. **INTRODUCTION**

The chemistry limits and action levels that have been presented in 6.20 are achievable and appropriate for the protection of system materials, ensuring fuel performance, and controlling radiation field buildup.

Three Action Levels have been defined for remedial actions to be taken when the parameters are confirmed to be outside the control values. These Action Levels are not intended to supersede Technical Specifications but can be used in conjunction with those requirements.

2. **ACTION LEVEL 1**

The Action Level 1 value of a parameter represents the range, outside of which data or engineering judgment indicates that long-term system reliability may be affected, thereby warranting an improvement of operating practices. Action Level 1 values generally represent limits for normal plant operations.

Actions to be taken if a parameter exceeds the Action Level 1 value:

a. Efforts should be made to restore the parameter to within the appropriate limit within seven (7) days.

b. If the parameter has not been restored to within the appropriate range within seven (7) days, a technical evaluation should be performed and a program for implementing corrective measures instituted. Such a program may require equipment additions or modifications over the long-term.

3. **ACTION LEVEL 2**

The Action Level 2 value of parameter represents the value, outside of which data or engineering judgment indicates significant damage could be done to the system in the short-term, thereby warranting a prompt correction of the abnormal condition.

c. Efforts should be made to bring the parameter within the appropriate Action Level 2 value within twenty-four (24) hours.

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d. If the parameter has not been restored to within the Action Level 2 value within 24 hours, an orderly unit shutdown should be initiated and the plant brought to a cold shutdown condition as quickly as permitted by other plant constraints.

- If the chemistry value improves to below the Action Level 2 value prior to plant shutdown, full power operation may be resumed.

e. Following a unit shutdown caused by exceeding the time limit on an Action Level 2 value, a technical evaluation of the incident should be performed and appropriate corrective measures taken before the unit is restarted.

4. **ACTION LEVEL 3**

The Action Level 3 value of a parameter represents the limit beyond which data or engineering judgment indicates that it is inadvisable to continue to operate the plant.

Action to be taken if a parameter exceeds the Action Level 3 value:

f. An orderly unit shutdown should be initiated immediately, with a reduction of the reactor coolant temperature to ≤ 250 °F as rapidly as other plant constraints permit.

g. If the chemistry should improve to below the limits of action Level 3 prior to completing the plant shutdown, power operation may be resumed subject to the requirements defined by the other Action Levels.

h. Following a unit shutdown caused by entering an Action Level 3 condition, a technical evaluation of the incident should be performed and appropriate corrective measures taken before the unit is restarted.

5. **CORRECTIVE ACTIONS**

Corrective actions should be implemented when a parameter is approaching or has exceeded an Action Level value. The following actions may be considered typical of those that can be taken:

i. Verify conditions.

j. Identify and isolate sources of impurities.

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k. Verify that the reactor coolant purification system is in service with the maximum available flow and that the ion exchanger bed removal efficiencies are adequate.

l. Increase sample and analysis frequencies for short-term trending purposes and for confirmatory analyses of critical chemistry parameters (the control parameters).

6. TECHNICAL EVALUATIONS

The technical evaluation process for prolonged abnormal water chemistry conditions should include the following:

The evaluation should address both the cause of the abnormal condition and inform the appropriate levels of management of the existence of the condition, the implications, and the possible corrective measures that can be taken over both the short and long terms.

QUESTION 118: (1.0)

A unit 2 refueling outage is scheduled to begin on March 18th. The following sequence of events is anticipated to occur on that date:

- 0100 – Reactor power < 2%.
- 0130 – Enter Hot Shutdown.

The earliest time that core off-load can commence is __________.

a. 3/22 at 0500

b. 3/22 at 0530

c. 3/30 at 1030

d. 3/30 at 0230

ANSWER: b

[SRO: Tier 3]

Answer correct: when unit enters hot shutdown the reactor is called subcritical; 3/18 @ 0130 + 100 hrs. = 3/30 @ 0230	Distractors plausible: a – trainee misconception of when the 100 clock starts. c/d – trainee applying 200 hour requirement prior to reduced inventory operation.	Distractors incorrect: a – the 100 hours must start from being subcritical c/d – 100 hours must elapse after being subcritical.
K/A: GEN-2.2.26	Objective: 2473	Source: Vogtle NRC exam, Dec. '99
Reference: TS-3.10, ND-92.5-LP-1	Level: Comprehension	

QUESTION 119: (1.0)

Which ONE of the following will prevent core offload?

a. "A" RHR pump inoperable.

b. CTMT purge secured.

c. Fuel Building exhaust secured.

d. "A" LHSI pump unavailable.

ANSWER: c

[SRO: Tier 3]

Answer correct: This is the only piece of equipment listed that has a technical specification requirement that stops fuel movement.	Distractors plausible: all – misinterpretation of TS-3.10 and refueling procedures requirements.	Distractors incorrect: all – none of these are T.S. requirements to stop refueling.
K/A: GEN-2.2.28	Objective: 2473	Source: New
Reference; TS-3.10, ND-92.5-LP-1, OP-FH-001	Level: Comprehension	

_____ / _____ 5.4.10 Before starting the core off-load, verify or put CTMT PURGE on filtered exhaust IAW 1-OP-VS-001. IF CTMT purge is NOT on filtered exhaust, THEN verify that the CTMT PURGE MOVs are closed.

_____ / _____ 5.4.11 Put the Fuel Building ventilation system on filtered exhaust and the two TR A and two TR B REFUEL SFTY MODE switches in the REFUEL position IAW 0-OP-VS-002.

_____ / _____ 5.4.12 Verify that the Fuel Transfer System wet checkout has been completed IAW 1-OPT-FH-001, Fuel Handling System.

* *

CAUTION: The 0-NSP-RX-001, Chi-Squared Test, must be done within the 24 hour period before the first fuel assembly for the core off-load is removed from the core, and during the core off-load if a break in fuel movement is greater than or equal to 24 hours. **(Ref. 2.4.8)**

* *

_____ / _____ 5.4.13 Verify with Engineering that 0-NSP-RX-001 is complete and the acceptance criteria has been met. **(Ref. 2.4.8)**

_____ / _____ 5.4.14 Record 0-NSP-RX-001 completion date and time in this step and in the Unit 1 Control Room Narrative Log.

Date: ______________ Time: ________

3. At least one source range neutron detector shall be in service at all times when the reactor vessel head is unbolted. Whenever core geometry or coolant chemistry is being changed, subcritical neutron flux shall be continuously monitored by at least two source range neutron detectors, each with continuous visual indication in the Main Control Room and one with audible indication within the containment. During core fuel loading phases, there shall be a minimum neutron count rate detectable on two operating source range neutron detectors with the exception of initial core loading, at which time a minimum neutron count rate need be established only when there are eight (8) or more fuel assemblies loaded into the reactor vessel.

4. Manipulator crane area radiation levels and airborne activity levels within the containment and airborne activity levels in the ventilation exhaust duct shall be continuously monitored during refueling. A manipulator crane high radiation alarm or high airborne activity level alarm within the containment will automatically stop the purge ventilation fans and automatically close the containment purge isolation valves.

5. Fuel pit bridge area radiation levels and ventilation vent exhaust airborne activity levels shall be continuously monitored during refueling. The fuel building exhaust will be continuously bypassed through the iodine filter bank during refueling procedures, prior to discharge through the ventilation vent.

6. At least one residual heat removal pump and heat exchanger shall be operable to circulate reactor coolant. The residual heat removal loop may be removed from operation for up to 1 hour per 8-hour period during the performance of core alterations or reactor vessel surveillance inspections.

7. Two residual heat removal pumps and heat exchangers shall be operable to circulate reactor coolant when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

8. At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange during movement of fuel assemblies.

9. With the reactor vessel head unbolted or removed, any filled portions of the Reactor Coolant System and the refueling canal shall be maintained at a boron concentration which is:

   a. Sufficient to maintain K-effective equal to 0.95 or less, and

   b. Greater than or equal to 2300 ppm and shall be checked by sampling at least once every 72 hours.

10. Direct communication between the Main Control Room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

11. No movement of irradiated fuel in the reactor core shall be accomplished until the reactor has been subcritical for a period of at least 100 hours.

QUESTION 120: (1.0)

The on-shift procedure writer presents a PT revision to the Unit 1 SRO for his review and approval. The SRO is on the "Cognizant Management A" list, but is **NOT** on the "Cognizant Management B" list. The revision changes the acceptable stroke time for a containment isolation trip valve.

Which ONE of the following is correct concerning the SRO's review of the procedure change?

a. This is a **change of intent**; the SRO **is** authorized to approve the change.

b. This is a **change of intent**; the SRO is **not** authorized to approve the change.

c. This is a **non-intent change**; the SRO **is** authorized to approve the change.

d. This is a **non-intent change**; the SRO is **not** authorized to approve the change.

ANSWER: b

[SRO: Tier 3]

Answer correct: changes to the acceptance criteria for PTs is considered a change of intent, which requires Cognizant Management B approval.	Distractors plausible: all – candidate misconception concerning the definition of change of intent vs. non-intent changes, or Cognizant Mgmt A vs. B.	Distractors incorrect: a – the SRO is not authorized to approve change of intent revisions; c & d – this is a change of intent revision.
K/A: GEN-2.2.6	Objective: 8036	Source: New
Reference: VPAP-0502	Level: Comprehension	

4.13 Procedure Change

A change to a procedure (temporary, permanent or one time only) that is needed to complete an activity. Permanent changes should be prepared only when sufficient time is not available for processing a revision.

4.13.1 Changes and revisions are classified into two categories based on the intent of the change:

a. **Change of Intent**

The intent of a procedure is the specific task or goal that is to be accomplished by the procedure. A change to the intent includes any of the following:

- Deletion of the procedure
- Changes to:
  - Purpose
  - Initial conditions
  - Acceptance criteria or tolerances
  - Scaling or setpoints
  - The method for meeting a commitment identified in the procedure
  - Step verification (IV or SV) (modifications or deletions to IV or SV requirements authorized by the Shift Supervisor or SRO per VPAP-1405 are **not** procedure changes)
  - System/component as left condition(s)
- Changes that add or delete a subsection, add an alternative method for performing a task, involve a less conservative method of performing the task, or affect equipment qualifications
- Changes that decrease personnel safety or Fire Protection effectiveness
- Change that deletes or relocates a Hold Point
- Change to CAUTION or WARNING statements. This does not include adding CAUTION or WARNING statements
- A change to a procedure that is marked "Infrequently Conducted or Complex Test or Evolution."

b. **Non-Intent Changes**

Changes and revisions which clearly do not change the intent of the procedure are considered non-intent changes. Examples are provided below and may be found anywhere in the procedure.

- Typographical errors that impact plant nomenclature or that may impact the successful completion of the procedure
- Obviously incorrect valve lineups or sequences of steps
- Incorrectly specified instruments for data taking
- Changes caused by the discrepancies in nameplate data, mark numbers, and equipment labels that do not meet the definition of an intent change

4.13.2 Every change is also classified in one of the following three categories based on the duration or extent of usage.

a. **One Time Only**

A change that will only be used once to complete a specific activity.

b. **Temporary**

A change that may not be incorporated into a revision and has a specified expiration date. The expiration date should not be greater than 120 days.

c. **Permanent**

A change that is permanent and should be incorporated into the next revision.

4.14 Procedure Package

Documents compiled during the processing of a procedure that represent the record of the procedure preparation, review, and approval process. The contents of procedure packages are specified in 7.1.

4.15 Procedure Revision

A change that results in the issuance of the entire procedure with an increase to the revision level.

4.16 Procedure Manager

A system for maintenance of procedures that updates EPDS and DMIS. Procedure Manager without MIND interface only updates PROMIS files. Procedure Manager with MIND interface automatically posts procedures, updates PROMIS files, MIND files, and DMIS files.

g. **Approval Authority Determination**

1. The writer/requestor shall determine the required approval authority by completing the SNSOC Approval Determination and the Change of Intent Checklist portions of the PAR form.

2. IF any of the questions are marked "Yes" for SNSOC Approval Determination, then SNSOC approval is required. Check the SNSOC Approval Required block on the PAR.

3. If any of the questions are marked "Yes" on the Change of Intent Checklist, then Cognizant Management B approval is required. Check the Cognizant Management B Approval Required block of the PAR.

4. If SNSOC or Cognizant Management B approval is NOT required, Cognizant Management A approval is required. Check the Cognizant Management A Approval Required block of the PAR.

h. **Activity Screening**

NOTE: The Activity Screening shall include considerations of 10 CFR 72.48, if the screening involves an ISFSI related procedure.

1. The writer shall perform or obtain an Activity Screening from an individual qualified to prepare an Activity Screening Checklist in accordance with VPAP-3001, Safety Evaluations.

2. If any of the questions are marked "Yes", the procedure requires SNSOC approval.

QUESTION 121: (1.0)

The "A" WGDT has been in holdup for 14 days and requires release.

Which ONE of the following sequences is required to perform a gaseous release?

a. RP obtains and analyzes a gas sample, RP generates a release permit, Operations verifies release information and commences release.

b. RP obtains and analyzes a gas sample, Operations verifies the sample is within the existing batch release permit, and commences release.

c. RP obtains and analyzes a gas sample, Operations verifies the sample is within the existing continuous release permit, and commences release.

d. Based on initial tank contents and decay time since the tank was placed in holdup, RP generates a release permit, Operations verifies release information and commences release.

ANSWER: a

[SRO: Tier 3]

Answer correct: a – this is the correct sequence by procedure.	Distractors plausible: b – Batch release permits are used for liquids only, never for gaseous releases. C – Continuous release permits are used for Specific secondary systems (i.e. sumps) d – Activity levels could be determined from previous samples and isotope decay information; however, a current sample is required to generate a release permit.	Distractors incorrect:.b/c/d – none of these are the correct procedural sequence.
K/A: GEN-2.3.6	Objective: 2432	Source: New
Reference: VPAP-2103, ND-92.4-LP-1	Level: Knowledge	

(2) The concentration of hydrogen or oxygen in the WGDT shall be determined within the limits of Spec 3.11.A by continuous installed monitors as required by Table 3.7-5(a), which says monitors must be operable or take grab sample at least once per 24 hours and analyze within 4 hours. If degassing operations are in progress, the grab sample requirements are every 4 hours.

(3) The quantity of radioactivity contained in each gas storage tank shall be determined within Spec 3.11.B at least once per month when specific activity of RCS is $\leq$ 2200 micro-curies per gram of DE Xe-133. If RCS specific activity > 2200 μCi/gm DE Xe-133, the WGDT shall be sampled once per day.

17. Gaseous Waste Releases

a. Precautions and Limitations

Highlight the following Precautions and Limitations.

(1) When the Waste Gas Decay Tank H_2 concentration is greater than 4 percent by volume, the O_2 concentration must not be greater than 2 percent by volume.

(2) Before adjusting the Process Vent Accountability Sampler Flow (1.0 to 3.0 cfm), HP notification is required.

(3) If the WGDT Gas Analyzer is out of service and BR-79 is open, the WGDT sample shall be collected at least once every 24 hours and analyzed within four hours from the sampling.

(4) When the WGDT is being released, the indicated flow on FI-GW-101, WGDT Effluent Flow, will be less than the actual flow. Actual flow is determined using CFM = (ml from WGDT Discharge Record)(3.532 x 10^{-5})/(elapsed time in minutes). The formula at the bottom of the WGDT Discharge Record is used to determine the total ml released.

(5) The WGDT maximum pressure is 115 psig.

b. The WGDT release form generated by HP will either be a computerized discharge record or a manually calculated discharge record.

(1) The computerized form contains pertinent data such as the tank sampled, release form number, and the calculated doses for whole body, skin, and thyroid.

(2) The limiting release rate will be from one of these three limiting factors.

(3) The maximum administrative release rate for a WGDT is 3.0 cfm. This is based on the release rate monitoring instrumentation only reading up to 3.0 cfm.

(4) However, if one of the limiting doses corresponds to a lower release rate than 3.0 cfm, an associated lower release limit will be assigned by HP.

18. ARP-RM-J3, J4, K3, K4, Process Vent RM Alarm

Have the trainees refer to the latest revision of ARP-RM-J3, J4, K3, K4.

a. The ARPs provide guidance for operations personnel to address an indication of abnormally high radioactivity on process vent monitors (Victoreen and

QUESTION 122: (1.0)

Unit 1 is in Cold Shutdown and the team is preparing for entry into Refueling Shutdown.

Prior to placing containment purge in service, at least one containment air recirc fan must be in operation __________.

a. to ensure the purge isolation valves are operable

b. to provide a flow path for the purge **exhaust** fans

c. to provide a flow path for the purge **supply** fans

d. to prevent backflow through the ring header

ANSWER: a

[SRO: Tier 3]

Answer correct: containment integrity must be established prior to entry into refueling shutdown; purge isolation valves receive an auto-closure signal from CTMT gaseous/particulate radiation monitor, which is considered inoperable if no CTMT air recirc fans are running; at least one CTMT air recirc fan must be running to provide a representative sample to the CTMT gaseous/part. R/M.	Distractors plausible: all – candidate misconception regarding the flow CTMT purge system flow path.	Distractors incorrect: all – purge system has separate ductwork from the CTMT air recirc fans.
K/A: GEN-2.3.9	Objective: 2411	Source: New
Reference:OP-VS-001, TS-3.10 TS-3.9.9, ND-92.3-LP-4	Level: Comprehension	

3.10 REFUELING

Applicability

Applies to operating limitations during REFUELING OPERATIONS.

Objective

To assure that no accident could occur during REFUELING OPERATIONS that would affect public health and safety.

Specification

A. During REFUELING OPERATIONS the following conditions are satisfied:

1. The equipment access hatch and at least one door in the personnel airlock shall be properly closed. For those penetrations which provide a direct path from containment atmosphere to the outside atmosphere, the automatic containment isolation valves shall be operable or the penetration shall be closed by a valve, blind flange, or equivalent.

2. The Containment Ventilation Purge System and the area and airborne radiation monitors which initiate isolation of this system shall be tested and verified to be operable immediately prior to REFUELING OPERATIONS.

4.10 Any ventilation change to the Containment or Fuel Building, with the Fuel Transfer Tube open, which results in a building pressure change, will cause a corresponding level change in both the Fuel Pit and the Reactor Cavity. (A pressure differential of 1 psid is about equal to a 2 feet of water level difference.) **(Reference 2.4.3)**

4.11 The following components are supported by proper alignment and operation of the Containment Ventilation System.

- Containment Air Recirc Smoke Detectors 1-FP-DA-101A, 1-FP-DA-101B, and 1-FP-DA-101C
- Containment Air Radiation Monitors 1-RM-RMS-159 and 1-RM-RMS-160

4.12 Radiation monitors 1-RM-RMS-159/160 shall <u>not</u> be considered operable unless at least one CTMT AIR RECIRC FAN is operating and air is flowing into S/G Cubicle C, from the ductwork which contains the tap for 1-RM-RMS-159/160, as identified on Station Drawings 11448-FB-6A, Sheet 2, 11448-FB-7B and 11448-FB-7D.

4.13 No Containment Air Recirculation Smoke Detector shall be considered operable unless the associated Containment Air Recirc Fan, as shown in the table below, is operating.

Smoke Detector	Ctmt Air Recirc Fan
1-FP-DA-101A	1-VS-F-1A
1-FP-DA-101B	1-VS-F-1B
1-FP-DA-101C	1-VS-F-1C

4.14 The Auxiliary Ventilation Exhaust Filter Trains must not be placed in service during or following painting, fire, or chemical release in any ventilation zone communicating with the Auxiliary Ventilation System. If the Auxiliary Ventilation Exhaust Filter Trains must be placed in service, then the surveillance requirements of Technical Specification 4.12 must be satisfied. **(Reference 2.4.4)**

QUESTION 123: (1.0)

Following a steam generator tube rupture coincident with a loss of offsite power, the Station Emergency Manager notes that the following emergency action levels in EPIP-1.01, Emergency Manager Controlling Procedure, are **all** currently exceeded:

TAB	CONDITION	CLASSIFICATION
A-10	Failure of a safety/relief valve to close after pressure reduction, which may affect the health/safety of the public.	Notification of Unusual Event
B-2	Fuel failure with steam generator tube rupture.	General Emergency
B-4	Gross primary to secondary leakage with loss of offsite power.	Site Area Emergency
B-6	Gross primary to secondary leakage	Alert
E-1	Release imminent or in progress and site boundary doses projected to exceed 1 rem TEDE or 5 rem thyroid CEDE.	General Emergency

When the SEM initially accesses the event, __________ should be used for classified.

a. **all tabs**

b. tabs B-2 **or** E-1

c. tabs B-2, E-1 **and** B-4

d. tabs B-2 **and** E-1

ANSWER: d

[SRO: Tier 3]

Answer correct: when a particular emergency classification exists in more than one event category, all applicable event categories should be noted to ensure the emergency classification is not inadvertently downgraded.	Distractors plausible: all – candidate misconception concerning the requirements for event classification.	Distractors incorrect: a & c – there is no need to note emergency classifications below the highest applicable emergency classification; b – both tabs should be noted.
K/A: GEN-2.4.38	Objective: 3110	Source: New
Reference: EPIP-1.01; ND-95.5-LP 2	Level: Comprehension	

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. RCS leak rate exceeds makeup capacity ABOVE CSD CONDITION	• Primary system leak (LOCA) - IN PROGRESS AND • Safety Injection - REQUIRED AND • RCS subcooling based on Core Exit Thermocouples - LESS THAN 30° F OR RCS inventory cannot be maintained based on pressurizer level or RVLIS indication	SITE AREA EMERGENCY
2. RCS leak rate limit - EXCEEDED ABOVE CSD CONDITION	• Primary system leak determined to be - GREATER THAN 50 gpm AND • Pressurizer level can be - RESTORED AND MAINTAINED	ALERT
3. Leak rate requiring plant shutdown IAW T.S. ABOVE CSD CONDITION	Intentional reduction in power, load, or temperature IAW T.S. 3.1.C leakage limit Action Statement - HAS COMMENCED	NOTIFICATION OF UNUSUAL EVENT

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Release imminent or in progress and site boundary doses projected to exceed 1.0 Rem TEDE or 5.0 Rem Thyroid CDE ALL CONDITIONS	HP assessment indicates actual or projected doses at or beyond Site Boundary - GREATER THAN 1.0 Rem TEDE or 5.0 Rem Thyroid CDE	GENERAL EMERGENCY
2. Release imminent or in progress and site boundary doses projected to exceed 100 mrem TEDE or 500 mrem Thyroid CDE ALL CONDITIONS	HP assessment indicates actual or projected doses at or beyond Site Boundary - GREATER THAN 100 mrem TEDE or 500 mrem Thyroid CDE	SITE AREA EMERGENCY

QUESTION 124: (1.0)

Which ONE of the following Protective Action Recommendations is the **most conservative**?

a. Evacuate 360° from 0 to 5 miles; shelter 360° from 5 to 10 miles.

b. Shelter 360° from 0 to 2 miles; shelter downwind sectors from 2 to 5 miles.

c. Evacuate 360° from 0 to 5 miles; evacuate downwind sectors from 5 to 10 miles; shelter unaffected sectors from 5 to 10 miles.

d. Evacuate 360° from 0 to 2 miles; evacuate downwind sectors from 2 to 5 miles; shelter downwind sectors from 5 to 10 miles; shelter unaffected sectors from 2 to 10 miles.

ANSWER: c

[SRO: Tier 3]

Answer correct: this is PAR 1, which is the most conservative in accordance with EPIP-1.06.	Distractors plausible: all – candidate misconception regarding the conservatism of actions associated with protecting the public.	Distractors incorrect: all are less conservative than PAR 1.
K/A: GEN-2.4.44	Objective: 3111	Source: New
Reference: EPIP-1.06	Level: Comprehension	

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.06	PROTECTIVE ACTION RECOMMENDATION MATRIX	2
ATTACHMENT	SPS	PAGE
2		1 of 1

NOTE:
- For situations involving multiple Emergency Action Levels (EALs), the most conservative PAR (the PAR closest to 1) should be used.
- Downwind sectors are defined as primary plus two (2) buffer sectors.
- PAR 3, a radiologically-based PAR, does not appear on this matrix.

EAL	PROTECTIVE ACTION RECOMMENDATION
B - 7 B - 8 C - 4 C - 5 C - 6 C - 7 C - 8 D - 1 J - 1	Any of the following exist: • Personnel Hatch Monitor: RM-RMS-161 or 261 > 1.5 E+4 mR/hr • Any Cont. Hi Range Monitor: RM-RMS-127 or -227 RM-RMS-128 or -228 > 4.5 E+4 R/hr • Containment pressure: > 60 psia and NOT decreasing • Shift Supv. or SEM judgement that a release path from containment to the environment is likely or has occurred NO → PAR 4: • Evacuate 360° from 0 to 2 miles. • Evacuate downwind sectors from 2 to 5 miles. • Shelter downwind sectors from 5 to 10 miles. • Shelter unaffected sectors from 2 to 10 miles. YES → Is Primary sector R, A, B, E or F NO → PAR 1: • Evacuate 360° from 0 to 5 miles. • Evacuate downwind sectors from 5 to 10 miles. • Shelter unaffected sectors from 5 to 10 miles. YES → PAR 2: • Evacuate 360° from 0 to 5 miles. • Shelter 360° from 5 to 10 miles.
E - 1	PAR 5: • Evacuate 360° from 0 to 2 miles. • Shelter downwind sectors from 2 to 5 miles.
M - 1	PAR 6: • Shelter 360° from 0 to 2 miles. • Shelter downwind sectors from 2 to 5 miles.

QUESTION 125: (1.0)

During a shutdown LOCA, the SI termination criteria are __________ when RCS cold-leg temperatures are below 285°F, __________.

a. less restrictive; to prevent RCS overpressurization

b. less restrictive; to minimize RWST depletion

c. more restrictive; to ensure adequate reactor vessel refill

d. more restrictive; to account for RCS pressure drop when SI flow is reduced

ANSWER: a

[SRO: Tier 3]

Answer correct: with RCS cold-leg temperatures <285°F, a major concern is brittle failure of the reactor vessel; the criteria for reducing SI flow are less restrictive to prevent RCS re-pressurization, which would increase the probability of reactor vessel failure.	Distractors plausible: b – SI termination criteria **are** less restrictive and RWST depletion can be a concern during certain accident scenarios; c & d – candidate misconception concerning the SI termination criteria and basis for shutdown LOCA procedure guidance; c – reactor vessel refill is a concern during LOCAs; d – RCS pressure does decrease after reducing SI flow.	Distractors incorrect: b – RWST depletion is not the major concern; c & d – SI termination criteria are less restrictive.
K/A: GEN-2.4.9	Objective: 3211	Source: New
Reference: AP-16.01 ND-95.2-LP-7	Level: Comprehension	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	<u>NOTE</u>: RCP C is the preferred RCP to provide PRZR spray.	
29.	__CHECK IF AN RCP SHOULD BE STARTED:	
	a) Check all RCPs - STOPPED	a) Stop all but one RCP. GO TO Step 30.
	b) Check RCP Seal ΔP - GREATER THAN 210 PSID	b) GO TO Step 30.
	c) Check RCS subcooling based on CETCs - GREATER THAN 50°F [140°F]	c) GO TO Step 43.
	d) Check PRZR level - GREATER THAN 35% [50%]	d) RETURN TO Step 28.
	e) Try to start an RCP:	e) GO TO Step 30.
	1) Establish conditions for starting an RCP IAW 1-OP-RC-001, STARTING AND RUNNING ANY RCP	
	2) Start one RCP	
30.	__CHECK RCS COLD LEG TEMPERATURE - GREATER THAN 285°F	GO TO Step 33.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE: • After stopping any SI pump, RCS pressure should be allowed to stabilize or increase before stopping another SI pump. • The CHG pumps and LHSI pumps should be stopped on alternate trains when possible. • CHG pumps should be run in the following order of priority: C, B, A.	
31.	__CHECK IF ONE CHG PUMP SHOULD BE STOPPED:	
	a) Check CHG pumps - TWO RUNNING	a) GO TO Step 32.
	b) Determine required RCS subcooling based on the number of RCPs running: • Any RCP, 100°F • No RCPs, 110°F	
	c) Check RCS subcooling based on CETCs - GREATER THAN REQUIRED	c) <u>IF</u> RCS hot leg temperatures greater than 350 °F [295°F], <u>THEN</u> GO TO Step 43. <u>IF</u> RCS hot leg temperatures less than 350°F [295°F], <u>THEN</u> verify running <u>OR</u> start one LHSI pump. GO TO Step 31d. <u>IF</u> at least one LHSI pump can <u>NOT</u> be started, <u>THEN</u> GO TO Step 43.
	d) Check PRZR level - GREATER THAN 35% [50%]	d) Do <u>NOT</u> stop CHG pump. RETURN TO Step 28.
	e) Stop one CHG pump	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
32.	__CHECK IF NORMAL CHARGING SHOULD BE ESTABLISHED:	
	a) Check RCS subcooling based on CETCs - GREATER THAN REQUIRED FOR THE NUMBER OF RCPs RUNNING • Any RCP, 130°F • No RCPs, 145°F	a) IF RCS hot leg temperature greater than 350 °F [295°F], THEN GO TO Step 43. IF RCS hot leg temperatures less than 350°F [295°F], THEN verify running OR start one LHSI pump. GO TO Step 32b. IF at least one LHSI pump can NOT be started, THEN GO TO Step 43.
	b) Check PRZR level - GREATER THAN 35% [50%]	b) RETURN TO Step 28.
	c) GO TO Step 34	

NOTE: • After stopping any SI pump, RCS pressure should be allowed to stabilize or increase before stopping another SI pump.

• CHG pumps should be run in the following order of priority: C, B, A.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
33.	__CHECK IF ONE CHG PUMP SHOULD BE STOPPED:	
	a) Check CHG pumps - TWO RUNNING	a) GO TO Step 33e.
	b) Check RCS pressure - GREATER THAN 250 PSIG [475 PSIG]	b) GO TO Step 43.
	c) Check PRZR level - GREATER THAN 35% [50%]	c) Do NOT stop CHG pump. RETURN TO Step 28.
	d) Stop one CHG pump	
	e) Check CHG pump suction - RMT IN PROGRESS	e) Stop running LHSI pump(s).

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

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<u>CAUTION</u>: The CHG pump miniflow recirc valves must be opened before reducing CHG flow to less than 60 gpm.

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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
34.	__ESTABLISH 60 GPM CHARGING FLOW:	
	a) Close CHG flow control • 1-CH-FCV-1122	
	b) Verify CHG line isolation - OPEN • 1-CH-HCV-1310A	b) Manually open valve.
	c) Open CHG line isolation MOVs • 1-CH-MOV-1289A • 1-CH-MOV-1289B	c) Locally open valve(s).
	d) Establish at least 60 gpm charging flow using CHG flow control	
35.	__ISOLATE HHSI TO COLD LEGS: • Close 1-SI-MOV-1867C • Close 1-SI-MOV-1867D	Locally close valve(s).
36.	__CHECK IF CHARGING FLOW SHOULD BE CONTROLLED TO MAINTAIN PRZR LEVEL:	
	a) Check LHSI flow - NONE INDICATED	a) GO TO Step 43.
	b) Control charging flow to maintain PRZR level	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: RCP C is the preferred RCP to provide PRZR spray.

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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
37.	__CHECK RCP STATUS:	
	a) Check RCPs - AT LEAST ONE RUNNING	a) Try to start one RCP: 1) Establish conditions for starting an RCP IAW 1-OP-RC-001, STARTING AND RUNNING ANY RCP. 2) IF an RCP can NOT be started, THEN GO TO Step 38.
	b) Stop all but one RCP	
	c) GO TO Step 41	
38.	__CHECK RHR - IN SERVICE	GO TO Step 40.
39.	__GO TO STEP 41	