U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos:	50-277, 50-278
Report Nos:	50-277/99-302, 50-278/99-302 (OL)
License Nos:	DPR-44 and DRP-56
Licensee:	PECO Nuclear
Facility:	Peach Bottom Atomic Power Station
Location:	Delta, Pennsylvania
Dates:	September 13-16, 1999 (Administration) September 20-24, 1999 (Grading)
Chief Examiner:	J. Williams, Senior Operations Engineer/Examiner, Region I
Examiners:	C. Sisco, Operations Engineer/Examiner, Region I S. Dennis, Operations Engineer/Examiner, Region I
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EXECUTIVE SUMMARY

Peach Bottom Atomic Power Station
Inspection Report Nos. 50-277 & 278 /99-302 (OL)

Operations

- Two reactor operator (RO) applicants and three senior reactor operator instant (SROI) applicants were administered exams. All applicants passed all portions of the exam.
- Overall, the applicants were well prepared for the exam.
- There were no post-exam facility comments.
- The facility used an examination preparation team of experienced training department staff who assisted the NRC examiners in an excellent manner.
- There were several times during the exam when the simulator did not perform as expected. These instances perturbed, but did not invalidate the examinations.
- Examination security was well maintained during the week of the exam.
- The responses to two violations associated with the licensed operator requalification training (LORT) program were reviewed by the examiners and found to be acceptable. The two open items were closed.

REPORT DETAILS

I. Operations

05 Operator Training and Qualifications

05.1 Reactor Operator (RO) and Senior Reactor Operator Instant (SROI) Initial Examinations

a. Scope

The NRC examiners reviewed the written and operating initial examinations prepared by the facility in accordance with the guidelines of the "Examination Standards for Power Reactors," (NUREG-1021, Revision 8). The review was conducted both in the Region I office and at the Peach Bottom facility. On September 14-16, 1999, the NRC examiners administered the operating portion of the exam to all applicants. On September 13, 1999, the written examinations were administered by the facility.

b. Observations and Findings

Grading and Results

The results of the examinations are summarized below:

	SROI Pass	Fail	RO Pass	Fail
Written	3	0	2	0
Operating	3	0	2	0
Overall	3	0	2	0

Examination Preparation and Quality

The written exams, job performance measures (JPMs) and simulator scenarios were developed by the facility using the guidelines of the examiner standards. The exam as submitted to the NRC met the guidelines of the examiner standards. There were no unacceptable test items. Enhancements were made to 12 of 126 written exam questions to improve clarity. There were no post-exam facility comments. All individuals with knowledge of the exam signed a security agreement.

The facility used an examination preparation team of experienced training department staff who assisted the NRC examiners in an excellent manner.

Written Examination Performance

All applicants passed the written exam. The facility training department staff performed an analysis of the ten written exam questions missed by at least half the applicants for generic weaknesses. The training department staff subsequently re-verified that the questions met the guidance of NUREG 1021, and were technically accurate. The questions were reviewed with the applicants and will be reviewed for training program improvement. These actions were determined to be acceptable.

Operating Test Administration and Performance

Simulator and job performance measure (JPM) actions, by the applicants, were acceptable.

During the operating test, there were several instances where the simulator did not respond as expected. Specific examples are noted in Attachment 1. These examples of simulator response perturbed, but did not invalidate the examination. The problems appeared to have resulted from a very recent simulator shutdown and startup and were not modeling discrepancies between the plant and simulator. They were not observed to be present during the exam validation process.

The facility took the necessary precautions to maintain examination integrity during the administration of the exam.

c. Conclusions

Two reactor operator (RO) applicants and three senior reactor operator instant (SROI) applicants were administered exams. All applicants passed all portions of the exam.

Overall, the applicants were well prepared for the exam.

There were no post-exam facility comments.

The facility used an examination preparation team of experienced training department staff who assisted the NRC examiners in an excellent manner.

There were several times during the exam when the simulator did not perform as expected. These instances perturbed, but did not invalidate the examinations.

Examination security was well maintained during the week of the exam.

08 Miscellaneous Operations Issues

08.2 Open Items Inspected

CLOSED (VIO 50-277 & 278/98-04-02) The annual operating exams did not follow licensed operator requalification training (LORT) program procedures for JPM difference requirements from week to week. The NRC inspectors reviewed and verified the corrective actions described in PECO Energy Company letter dated June 17, 1998, in response to the violation. The inspector also reviewed the annual operating tests given February and March 1999 to verify that JPM difference guidelines were satisfied. Based upon these reviews, this item is closed.

CLOSED (VIO 50-277 & 278/98-04-03) The annual operating test did not sample items required by 10 CFR55.45 dealing with executing the emergency plan. The NRC inspector reviewed and verified the corrective actions described in PECO Energy Company letter dated June 17, 1998, in response to the violation. Based upon this review, this item is closed.

V. Management Meetings

XI Exit Meeting Summary

On September 22, 1999, NRC observations regarding the examination were discussed with members of the facility staff. The NRC expressed appreciation for the cooperation and assistance that was provided during both the preparation and examination week by licensed operator training personnel.

LIST OF ITEMS OPENED AND/OR CLOSED

Closed

NUMBER	TYPE	DESCRIPTION
50-277 & 278/98-04-02	VIO	JPM difference requirements on annual operating tests were not in accordance with LORT program procedures.
50-277 & 278/98-04-03	VIO	Sampling items in 10 CFR50.45 on the annual operating test.

Attachments:
1. Facility Simulation Report
2. RO Written Examination with Answer Key
3. SRO Written Examination with Answer Key

Attachment 1

Facility Simulation Report

Facility Licensee: Peach Bottom Atomic Power Station

Facility Docket Nos.: 50-277 and 50-278

Operating Test Administered on: September 14-16, 1999

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and, without further verification and review, are not indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information that may be used in future evaluations. No licensee action is required in response to these observations.

While conducting the simulator portion of the operating tests, examiners observed the following items:

- The control rod position indication on the full core display showed one rod at minus eight when scrams occurred.
- In scenario number one, off gas system alarms were received as the 'B' steam jet air ejector (SJAE) was placed in service.
- In scenario number five, when the applicant locked the recirculation pump scoop tube, small power oscillations occurred that were mistakenly identified as thermal hydraulic instabilities.
- The simulator responded erratically when the applicants attempted to start emergency service water (ESW) and/or emergency cooling water (ECW) pumps to supply cooling to the emergency diesel generator (EDG) during a fast start of the EDG.

Attachment 2

RO WRITTEN EXAM WITH ANSWER KEY

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region: I / II / III / IV
Date: September 13, 1999	Facility/Unit: Peach Bottom 2 & 3
License Level: RO / SRO	Reactor Type: W / CE / BW / GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	______ Points
Applicant's Score	______ Points
Applicant's Grade	______ Percent

Test ID: 1999 RO Exam Level: RO Date: 9/13/99

Title: 1999 PEACH BOTTOM NRC RO EXAM Facility: PBAPS

ID Number: 5

1) A Reactor Operator has just begun night work following his long break (seven days).

- Day 1 he works 1845 - 0645
- Day 2 he works 1845 - 0645
- Day 3 he works 2245 - 0645 (4 hours vacation)
- Day 4 he works 1845 - 0945 (to cover for a sick RO)

At 1200, the RO gets a call at home from the Shift Operations Assistant (SOA) requesting that he return to work as soon as possible to fill a vacancy. To stay within the bounds of A-40, without deviations, the RO may:

A) Return to work immediately, but can only work 8 hours.

B) Return to work at 1745, but can only work 9 hours.

C) Return to work at 1845, and work a regular 12 hour shift.

D) Return to work at 2145, and work up to 16 hours.

2) Given the following conditions:

- With Unit 2 operating at 50% power, a packing leak is discovered on an accessible motor operated valve in a safety-related system.
- The leak is not severe and it has been decided to backseat the valve during the next shift.
- All plant systems are operating as designed.

In accordance with OM-C-7.5, "Valves", which of the following describes how this valve should be backseated?

A) The appropriate System Manager should manually backseat the valve using TMT.

B) The Operator in the Main Control Room should electrically backseat the valve.

C) Maintenance personnel should manually backseat the valve.

D) An Equipment operator at the motor control center should electrically backseat the valve.

3) An operator, performing an Independent Verification of a check-off list (COL), discovers that a manually operated valve is danger tagged in the "open" position? The COL required position for the valve is "closed".

In accordance with OM-C-11.1, "Independent Verification", which of the following describes the required action(s)?

A) The COL step should NOT be initialed, the clearance number and valve position should be noted on the COL.

B) The COL position should be changed to the actual valve position, then the step should be initialed and dated.

C) The COL step should be marked "N/A" and the remainder of the COL should be completed.

D) The COL should NOT be completed until a temporary change noting the discrepancy is prepared in accordance with A-3.

4) Both units are operating in MODE 1 at full power with no Surveillance Testing or other evolutions in progress. The Shift Manger receives a call indicating that one of the licensed operators has been selected for a random substance screening. The selected operator is currently the Unit 2 Reactor Operator (URO). The testing would require the operator to leave the main control room for approximately 45 minutes. Determine the MINIMUM shift response to this condition.

A) A temporary relief of the URO by the on-shift Plant Reactor Operator (PRO).

B) A temporary relief of the URO by the fourth Reactor Operator (4th RO).

C) A complete turnover of the URO position to the Plant Reactor Operator (PRO).

D) No relief is required since licensed operators are exempt from random substance testing while holding a licensed position.

5) In accordance with OM-C-11.2, "Double Verification", which of the following would REQUIRE a second individual to actually witness the activity while it is occurring?

A) Restoration of a throttled valve to its required locked position.

B) Fuse removal as directed by the T-200 procedures.

C) Restoration of a clearance on an ECCS system.

D) A routine surveillance test being performed in a Radiation Area.

6) Unit 3 is in MODE 5 with refueling activities in progress.

Which of the following conditions would require the Reactor Operator to notify the Fuel Floor Supervisor to suspend core alterations in accordance with FH-6C, "Core Component Movement - Core Transfers".

A) A control rod in a defueled cell is withdrawn.

B) The white rod permissive light on the C05 panel is NOT lit when the refuel platform is over the core with fuel loaded on the main hoist.

C) Wide Range Neutron Count Rate doubles when a fifth fuel bundle is seated around the "A" WRNM detector.

D) Receipt of the "A Fuel Pool Serv Water Booster Pump Overcurrent" alarm.

7) Which of the following combinations of tags may be applied to the same component at the same time in accordance with the Clearance and Tagging Manual?

A) A Special Condition Tag (SCT) and a tagged component bearing a green suspension label.

B) Two Special Condition Tags (SCTs).

C) A Danger Tag and Special Condition Tag (SCT).

D) An Information Tag and a Danger Tag.

8) A check-off list (COL) Independent Verification (IV) is required to be completed on 8 system valves located in an area with dose rates of 120 mR/hr.

What is the maximum time available to complete the verification before exceeding the guidelines for Shift Management to consider waiving the IV?

A) 2 minutes.

B) 5 minutes.

C) 10 minutes

D) 12 minutes

9) Given the following conditions:

- A male, fully qualified radiation worker at Peach Bottom has just returned from 4 weeks of outage support at Limerick.
- Total Effective Dose Equivalent (TEDE) received at Limerick was 250 mrem.
- This workers' current TEDE from Peach Bottom for 1999 is 225 mrem.

What is the MAXIMUM annual non-emergency Total Effective Dose Equivalent (TEDE) that can be received at Peach Bottom for the remainder of 1999 WITHOUT exceeding the Federal Exposure Limits.

A) 4475 mrem

B) 4525 mrem

C) 4750 mrem

D) 4775 mrem

10) To return the plant to a stable condition following a transient, Operations personnel need to enter a High Radiation Area that does not have an existing Radiation Permit (RWP).

Which of the following will meet the MINIMUM requirements for an Equipment Operator to enter the area.

A) Must be accompanied by an Advanced Rad Worker (ARW) qualified individual.

B) Must be accompanied by a Level II Radiation Protection Technician qualified individual.

C) Entry into the area is not permitted without the Radiation Protectection Manger (RPM) permission.

D) Entry into the area is not permitted until activation of the Emergency Plan.

11) The Plant Reactor Operator (PRO) has just received a fire alarm from the Turbine Building.

The PRO is REQUIRED to make a call for off-site fire fighting support:

A) After 10 minutes if an actual fire is confirmed.

B) Immediately if equipment for safe shutdown is jeopardized.

C) When 2 or more fire alarms are received in the same area.

D) After 20 minutes if the Incident Commander reports the fire is NOT controlled.

12) You were working as the fourth Reactor Operator on your crew during the night shift when an emergency occurred on Unit 3. The Shift Manager, acting as the Emergency Director, has assigned you the responsibility of being the NRC Communicator. From the following, select one responsibility of this position.

A) Establish communications with the TSC to report Trip Table status.

B) Establish communications with the PBAPS NRC Resident Inspector.

C) Initiating the Emergency Response Organization call out.

D) Initiating the Emergency Response Data System.

13) A Main Control Room annunciator has a "blue" dot on its window.

Which of the following describes the status of the equipment monitored by that annunciator?

The monitored equipment has a deficiency that:

A) Affects the performance of the Transient Response Implementation Plan (TRIP) procedures.

B) Is NOT considered a Main Control Room deficiency.

C) Affects the performance of the Emergency Response Procedures (ERP).

D) Does not impact any safety related plant equipment.

14) Given the following conditions:

- A Unit 2 reactor startup is in progress with control rod withdrawals occurring.
- Rod Worth Minimizer (RWM) Group 1 contains 12 control rods that are to be withdrawn from Notch "00" to Notch "48".
- 11 rods from this group are withdrawn to Notch "48" and the remaining rod to Notch "42".
- A control rod in Group 2 is then selected but not withdrawn.

Which of the following is the expected response of the RWM?

The RWM will display:

A) One withdraw error and further rod withdrawals will be blocked except for the rod with the withdraw error.

B) One withdraw error and if a second withdraw error is made further rod withdrawals will be blocked except for the two rods with the withdraw errors.

C) One insert error and further rod withdrawals will be blocked except for the rod with the insert error.

D) One insert error and if a second insert error is made, further rod withdrawals will be blocked except for the two rods with the insert errors.

15) During steady power reduction from 100% to 65% power on Unit 3 the Unit Reactor Operator notes Wide Range reactor water level indications, which had been reading about 10 inches less than Narrow Range, are slowly rising. Actual reactor water level remains unchanged.

Which of the following describes what is occurring?

A) The density compensation signal (reactor pressure) has failed full "downscale".

B) The density compensation signal (reactor pressure) is lowering as power is reduced resulting in a lowering d/p on the level instrument, therefore an indicated level rise.

C) The Digital Feedwater redundant feedback signals have failed full "upscale".

D) The reduction in recirculation flow is raising the pressure at the variable leg tap resulting in a lowering d/p on the level instrument, therefore an indicated level rise.

16) Given the following conditions:

- The E-42 4KV Bus has lost power.
- The fast transfer and Diesel Generator start both failed to occur automatically.
- The E-4 Diesel Generator (DG) was started with the "Quick Start" pushbutton.
- The E-42 breaker is closed and the DG is now carrying all the loads on the E-42 4KV Bus.

Which of the following describes the current Mode of operation of the DG and what is required to synchronize the DG back to the Grid?

The E-4 DG is operating in:

A) Droop (Parallel), the DG Quick Start pushbutton must be pressed again and synch must be completed within 3 minutes.

B) Isochronous (Unit), the DG Quick Start pushbutton must be pressed again and synch must be completed within 3 minutes.

C) Droop (Parallel), the DG Auto Start Bypass pushbutton must be pressed and synch must be completed within 3 minutes.

D) Isochronous (Unit), the DG Auto Start Bypass pushbutton must be pressed and synch must be completed within 3 minutes.

17) Which of the following conditions will result in Recirc flow controller output being limited to 30%?

A) Total feedwater flow greater than 85% and any condensate pump trip.

B) Individual feedpump flow less than 20% and Reactor level less than 17".

C) Total feedwater flow greater than 20% and Reactor level less than 17".

D) Reactor scram and Reactor level less than 17".

18) The following conditions exist on Unit 2 after a LOCA.

- Drywell pressure 7 psig, rising slowly
- Reactor pressure 400 psig, dropping slowly
- Reactor level -75"
- All low pressure ECCS pumps were manually secured
- Level is being maintained with condensate injection.
- "D" RHR pump was placed in Torus Sprays at 1000 gpm and Drywell sprays at 1000 gpm

If level were to drop to -200" what would be the response of the LPCI system with no additional operator actions?

A) A, B, C RHR pumps would start, "D" RHR would continue to run, LPCI outboard injection valve (MO-154) would auto open, spray valves would auto close.

B) A, B, C RHR pumps would start, "D" RHR would continue to run, LPCI outboard injection valve (MO-154) would auto open, spray valves would remain open.

C) A, B, C RHR pumps would NOT start, "D" RHR would continue to run, LPCI outboard injection valve (MO-154) would remain closed, spray valves would remain open.

D) A, B, C RHR pumps would NOT start, "D" RHR would continue to run, LPCI outboard injection valve (MO-154) would remain closed, spray valves would auto close.

19) Following a valid HPCI initiation due to high Drywell pressure on Unit 3, HPCI was secured using the "Short Term HPCI System Shutdown When an Initiation Condition IS Present" method of SO-23.2.2A-3, "HPCI System Shutdown". The PRO has been directed to initiate HPCI injection into the Reactor Vessel from this condition.

Under these conditions, HPCI Turbine speed during startup is controlled by:

A) The ramp generator initiated by the opening of HPCI steam supply valve, MO-3-23-014.

B) The slow opening of the HPCI Turbine Stop Valve, HO-3-23-4513.

C) The ramp generator initiated by opening of the HPCI Turbine Control Valve, HO-3-23-4512.

D) The ramp generator initiated by opening of the HPCI Turbine Stop Valve, HO-3-23-4513.

20) An ADS blowdown has occurred following a LOCA. The ADS valve control switches remain in "Auto". Pressure is 200 psig and lowering slowly. All Core Spray and RHR pumps were initially injecting. "D" Core Spray pump has tripped. All RHR pumps were secured when level recovered above -100". Level is being restored using A, B, and C Core Spray pumps.

An additional Core Spray pump needs to be shutdown to control level recovery.

Which of the following statements accurately describe the response of the ADS system to pump shutdown?

A) ADS blowdown will stop when the "A" Core Spray pump is shutdown.

B) ADS blowdown will stop when the "B" Core Spray pump is shutdown.

C) ADS blowdown will stop when the "C" Core Spray pump is shutdown.

D) An ADS seal in prevents inadvertent blowdown termination by pump shutdown.

21) An ATWS condition has occurred on Unit 3. Reactor level is 23 inches and pressure is 1000 psig with the turbine still running. The CRS has directed the URO to inject Standby Liquid Control (SBLC). The URO positions the SLBC switch to "PUMP 'A' RUN". Identify the expected SBLC System response.

A) Squib continuity light are lit, pump discharge pressure is 1450 psig.

B) Squib continuity lights are lit, pump discharge pressure is 1100 psig.

C) Squib continuity lights are out, pump discharge pressure is 1450 psig.

D) Squib continuity lights are out, pump discharge pressure is 1100 psig.

22) Unit 2 was operating at 100% power when the "A" Recirc pump tripped. The pump was isolated in accordance with the procedure.

Which of the following statements describes the relationship between INDICATED total core flow and ACTUAL total core flow?

Total core flow indicated on DPFR-2-3-095 dP/F will be:

A) Less than actual by an amount TWICE idle loop flow.

B) Less than actual by an amount EQUAL to idle loop flow.

C) Greater than actual by an amount TWICE idle loop flow.

D) Greater than actual by an amount EQUAL to idle loop flow.

23) The RCIC system was being restored to its normal alignment following maintenance, when a high steam flow isolation occurred due to stroking open the valves too quickly. A few minutes later, a feedwater transient results in a scram and the need for RCIC system operation. Current level is -50 inches and dropping slowly. The SRO has directed that RCIC be recovered and injection initiated into the vessel at 600 gpm.

After depressing the isolation reset pushbutton, which of the following actions will be necessary to inject with RCIC?

A) The RCIC turbine trip throttle valve will need to be reset from the control room.

B) The RCIC turbine trip throttle valve will need to be reset locally.

C) The MO-131, steam admission valve, must be stroked open manually.

D) RCIC will automatically align and inject when the isolation reset pushbutton is depressed.

24) Given the following conditions:

- Unit 2 has experienced a loss of all AC power (station blackout).
- The Reactor Core Isolation Cooling (RCIC) system automatically initiated.
- Reactor water level is now -52 inches and rising.
- The Control Room Supervisor directs the Unit Reactor Operator to isolate RCIC.

What will be the expected RCIC system response when the operator depresses the Manual Isolation Pushbutton?

A) A normal RCIC system isolation and turbine trip will occur.

B) A RCIC turbine trip and system isolation will occur except the Inboard Steam Isolation Valve (MO-15) will not close.

C) No RCIC isolation actions or turbine trip will occur.

D) A RCIC turbine trip and system isolation will occur except the Outboard Steam Isolation Valve (MO-16) will not close.

25) A small Main Steam Line leak has occurred in the Turbine Building on Unit 2. The following plant conditions exist:

- The Reactor is scrammed.
- The "A" Main Steam Line has failed to isolate.
- Reactor Pressure is 800 psig.
- The PRO shutdown all low pressure ECCS pumps immediately after they started on LO-LO-LO level since no injection or minimum flow path was available.
- HPCI is blocked.
- RCIC has been maintaining reactor level steady at -165" for 12 minutes.

Starting the "A" RHR pump will:

A) Result in an ADS blowdown after a 9 minute time delay.

B) Result in an immediate ADS blowdown.

C) Result in an ADS blowdown after a 105 second time delay.

D) NOT result in an ADS blowdown.

26) A small recirc leak has resulted in 8 psig drywell pressure and -170" reactor level on Unit 2. After the Drywell Cooling Fans tripped, the CRS directed them to be restored using T-223. The trip was bypassed and the fans were restored in fast speed. 15 minutes later, the STA reports that the amber bypass light over the Drywell Cooler Fan bypass switch (43-5-0165) has gone out.

Describe the effect this will have on Drywell Cooler Fan operation and why.

A) Fans will continue to run, light goes out when both trip signals clear.

B) Fans will continue to run, light goes out if either trip signal clears.

C) Fans will trip, light goes out when both trip signals clear.

D) Fans will trip, light goes out when either trip signal clears.

27) Unit 2 is operating in MODE 1 at full power. The 2B RPS MG set output breakers trip on underfrequency.

Under these conditions, which of the following PCIS isolations will occur and result in isolation valves repositioning?

A) A Group II Outboard half isolation.

B) A Group III Outboard half isolation.

C) An Outboard MSIV auto isolation.

D) A full RWCU isolation.

28) Unit 2 is operating in MODE 1 at full power when PISH-2-5-12A the drywell pressure input to RPS and PCIS fails high resulting in an "A" channel RPS half scram and associated annunciators.

Determine the expected Primary Containment Isolation System (PCIS) response to this condition.

The "GROUP II/III INBOARD ISOL RELAYS NOT RESET" annunciator will:

A) Alarm, but NO valves will reposition.

B) Alarm, and the inboard isolation valves will reposition.

C) NOT alarm, and NO valves will reposition.

D) NOT alarm, but the inboard isolation valves will reposition.

29) A T-112 BLOWDOWN is in progress with reactor pressure at 75 psig above Torus pressure. The URO notes that, although the ADS valve switches are in "OPEN", the SRV's are indicating closed. The green and white lights are lit for each of the ADS valves. All other SRV's have only green lights lit. The "SAFETY RELIEF VALVE OPEN" annunciator is NOT lit.

Given the above conditions, determine the current expected position of the ADS valves.

A) Fully open.

B) Partially open, but not far enough for proper indication.

C) Failed closed.

D) Fully closed, due to low steam pressure.

30) Given the following conditions:

- Unit 2 was operating at 100% power.
- Annunciator 220 F-5, "Inverter Trouble", was received indicating a loss of 20Y050.
- The reactor was later scrammed and the turbine tripped.

Which of the following is the reason why this failure requires reactor pressure control via the Safety Relief Valves?

The static inverter loss will:

A) Cause a full Group I Main Steam Isolation Valve closure.

B) Cause a "full open" signal to the Turbine Bypass Valves requiring the EHC Pumps to be tripped to prevent a rapid depressurization.

C) Result in a loss of Turbine Bypass Valve opening capability.

D) Result in a closure of the Inboard Main Steam Isolation Valves.

31) Unit 3 is performing a GP-2 Plant Startup. Power is being raised from 50% to 100%. The Plant Reactor Operator is monitoring Electrohydraulic Control (EHC) system performance.

Pressure averaging manifold pressure is initially:

A) Equal to reactor pressure with an increasing dP as turbine load is raised.

B) Equal to reactor pressure with a constant dP as turbine load is raised.

C) Less than reactor pressure with an increasing dP as turbine load is raised.

D) Less than reactor pressure with a lowering dP as turbine load is raised.

32) Unit 2 is operating at 100% power, with the Digital Feedwater Control System (DFCS) in three-element control with the "B" Narrow Range Level automatically selected. A fault in the level detector causes it to fail downscale. Which of the following will occur?

A) DFCS will sense a low level and increase RFPT speed, thereby causing Reactor Vessel level to increase.

B) The "B" Narrow Range Level Detector will be automatically de-selected and the HIGHEST remaining narrow range level signal will be automatically selected.

C) The "B" Narrow Range Level Detector will be automatically de-selected, and the LOWEST remaining narrow range level signal will be automatically selected.

D) A default valve of +23" will be automatically selected by the master level controller.

33) Following a reactor scram from a power condition, Reactor Feedwater Pump (RFP) speed automatically goes up to compensate for the shrink experienced as the voids in the reactor collapse.

To protect the pumps from overspeed under these conditions, RFPs are limited to 85% following a scram:

A) With all three condensate pump running.

B) With less than three condensate pumps running.

C) With individual feedwater flows greater than 20%.

D) With individual feedwater flows less than 20%.

34) Both Units were operating in MODE 1 at full power when the "3A" RPS bus was manually transferred to its alternate feed. Determine the expected condition of the Standby Gas Treatment (SBGT) system as a result of this transient.

A) "B" SBGT fan has started, SBGT "A" filter inlet and outlet dampers have opened.

B) "C" SBGT has started, SBGT "B" filter inlet and outlet dampers have opened.

C) "B" SBGT fan has started, SBGT "B" filter inlet and outlet dampers have opened.

D) "C" SBGT fan has started, SBGT "A" filter inlet and outlet dampers have opened.

ID Number: 53

35) Peach Bottom 4KV is aligned as follows:

- The #2 Emergency Auxiliary Transformer (OAX04) is out of service.
- All eight 4KV busses are being supplied by the #3 Emergency Auxiliary Transformer (OBX04).
- The 2A RHR and 2A HPSW pumps are running in Torus Cooling.

Determine the expected plant response to an automatic trip of the E-312 breaker.

A) The E-1 Diesel Generator will start after .25 seconds.

B) The E-1 Diesel Generator will start after .5 seconds.

C) The 2A RHR and HPSW pumps will trip and restart on power restoration.

D) The E-124 Load Center will trip and lockout.

36) Unit 3 is in MODE 1 at full power with "B" RPS on its normal alternate feed. All other equipment is in its normal alignment. Both of the inservice off-site start up feeds trip simultaneously. All diesel generators start and close in on their buses as designed.

Determine the response of Unit 3 to this loss of AC power event. Unit 3 will:

A) Scram immediately due to the loss of power to the RPS system.

B) Scram immediately due to turbine stop and control valve closure.

C) NOT scram immediately due to "B" RPS being powered by its alternate source.

D) NOT scram immediately due to the RPS MG Sets maintaining power until the diesel generators load their buses.

37) Peach Bottom has experienced a complete loss of off-site power. The E-1 and E-2 Diesel Generators started and loaded their busses normally but the "A" ESW pump did not start. The E-3 and E-4 Diesels did not start. The E-1 and E-2 were then shutdown due to not having cooling water. The Power System Director (PSD) has been requested to configure Conowingo Station for Peach Bottom Station Blackout. The CRS has directed a backfeed in accordance with SE-11 Attachment D, "Backfeeding Safe Shutdown Loads with E-1 & E-2 Diesel Generators Available".

Given that both the 2SUE and 3SUE busses are available for backfeeding, the PRO should select:

A) The 2SUE bus because it is the normal power source for the bus that feeds the "B" ESW pump.

B) The 2SUE bus because this will allow use of the SBO line to power 4KV buses.

C) The 3SUE bus because it is the normal power source for the bus that feeds the "B" ESW pump.

D) The 3SUE bus because this will allow use of the SBO line to power 4KV buses.

38) Which of the following statements describe the power supply to the Backup Scram Solenoid valves and their expected condition upon receipt of a full scram.

A) Station batteries, energize on a Full Scram.

B) Station batteries, de-energize on a Full Scram.

C) Reactor Protection Bus, energize on a Full Scram.

D) Reactor Protection Bus, de-energize on a Full Scram.

39) Given the following conditions:

- Unit 2 is making preparations for a reactor startup from a refueling outage.
- Reactor Building ambient temperature is 74 degrees F.
- The Reactor Building Equipment Operator is charging the hydraulic control unit accumulators with nitrogen to a pressure of 590 psig.
- Several days later with the Unit at 100% power, Reactor Building temperatures have stabilized at 92 degrees F.

Which of the following describes the expected impact on the Control Rod Drive Hydraulic system operations for these conditions? (Refer to attached figure.)

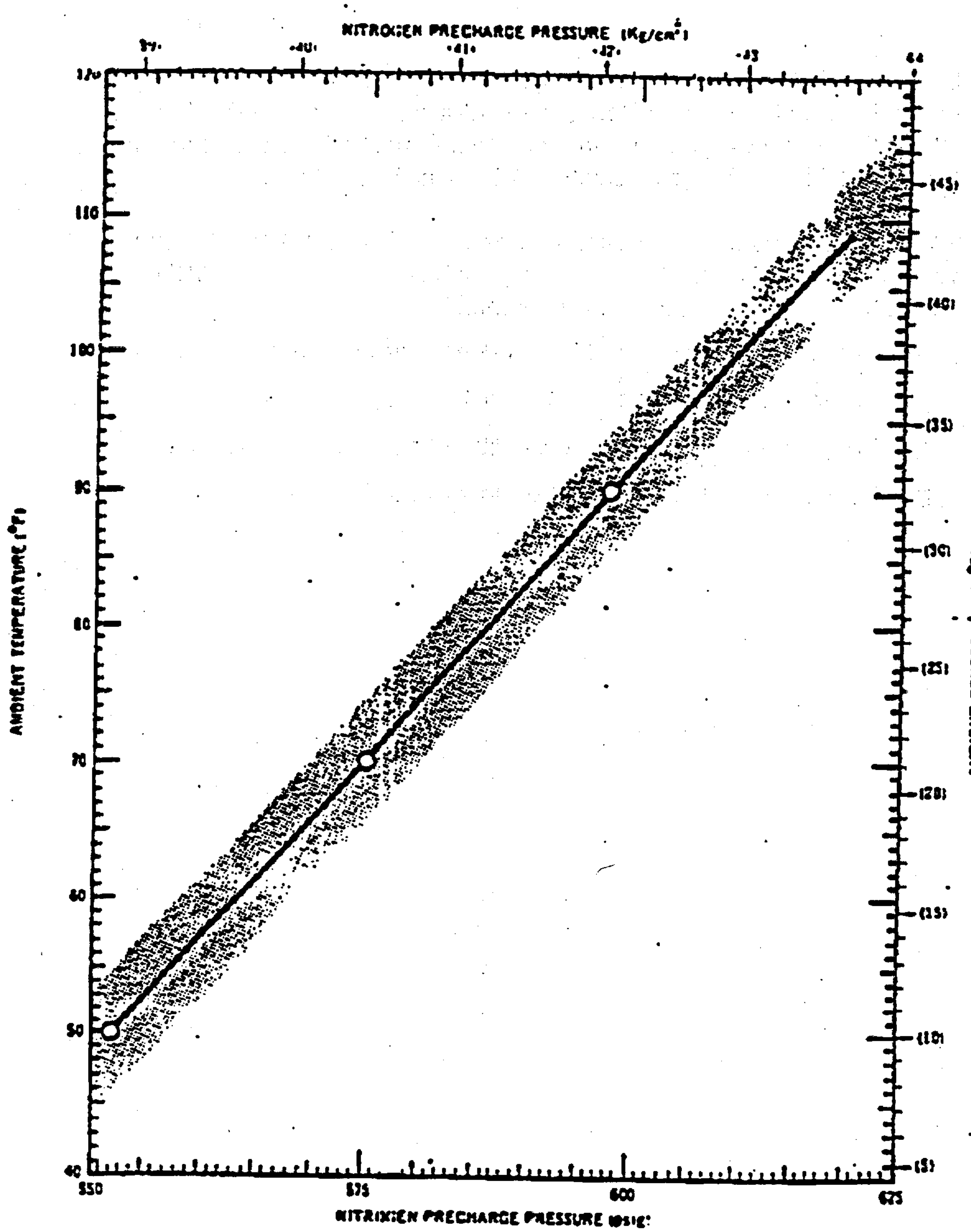
The individual control rod:

A) Normal insertion speeds will be slower and may result in control rod drift alarms.

B) Scram speeds will be slower and will result in reduced reactivity addition rates.

C) Normal insertion speeds will be faster and may result in "double notching".

D) Scram speeds will be faster and may result in mechanism damage.



SO 3.7.A-2, Figure 1
Accumulator Precharge Nitrogen Pressure
Verses Ambient Temperature

40) Unit 2 is in MODE 2 with a heat up in progress. Vessel level is being maintained by the Control Rod Drive Hydraulic system and the Reactor Water Cleanup system in dump mode to the main condenser through the "RWCU Filter Bypass Valve", MO-74, which is full open.

What is the basis for the caution in procedure SO 12.1.A-2, "Reactor Water Cleanup System Start" which prohibits opening the "RWCU Outlet Valve" MO-68 in this system lineup?

A) Excessive heat load on the Non-Regenerative Heat Exchanger.

B) Group II isolation on greater than 125% system flow.

C) Excessive RWCU pump flows without control room indication.

D) Auto closure of CV-55 "Dump Flow Control Valve".

41) Given the following conditions:

- Unit 3 has had a complete loss of the E13 4160VAC Bus.
- This results in a loss of power to the "A" Residual Heat Removal (RHR) Pump and to the "A" Loop Inboard LPCI Injection Valve (MO-25A).
- A valid LOCA signal occurs.

What must occur to result in a final, design RHR injection flowrate for these conditions of 30,000 gpm.

A) The RHR Loop Cross-Tie Valve (MO-20) must be unlocked and opened by an operator.

B) An operator must manually transfer the Inboard LPCI Injection Valve (MO-25A) to the alternate power supply.

C) The Outboard LPCI Injection Valve (MO-154A) must automatically open to inject through the normally open MO-25A.

D) The Inboard LPCI Injection Valve (MO-25A) must automatically transfer to the alternate power supply.

42) Unit 2 is in MODE 4 with "A" RHR pump in Shutdown Cooling (SDC) returning to the vessel through the MO-25A, "Inboard Disch". Loss of inventory causes level to drop to -20". SDC isolates and the "A" RHR pump trips.

Which of the following statements describes the response of LPCI should level continue to drop to
< -160" with no additional operator actions.

A) "A" RHR restarts, "B", "C", and "D" start and inject.

B) "A" RHR restarts, "B", "C", and "D" start and run on min. flow.

C) RHR "B", "C", and "D" start and inject.

D) RHR "B", "C", and "D" start and run on min flow.

43) Following a reactor scram and scram reset, the Unit Reactor Operator notes that the full core display for rod 02-23 is blank with no position indicated and no green back light. All other rods indicate 00 with a green back light.

If the blank display is due to a rod 02-23 Position Indicating Probe (PIP) problem, the operational impact will be the inability to select and move:

A) Other rods in REFUEL due to a lack of "REFUEL MODE SELECT PERMISSIVE".

B) Other rods due to "RPIS INOPERATIVE".

C) Rod 02-23 due to lack of position indication and backlight.

D) Rod 02-23 due to "ROD SELECT BLOCK TIMER MALFUNCTION".

44) Unit 2 is operating at 100% power when a leak develops in the TBCCW system. Shortly thereafter TBCCW Head Tank level drops out of sight low. The operating TBCCW pump starts cavitating and discharge pressure drops to 0 psig.

Assuming no operator actions are taken, which of the following statements describe the operational impact of this event?

A) "ISO-PHASE BUS TROUBLE" alarm is received immediately, "ISO-PHASE BUS LOSS OF COOLING" is received 10 minutes later, and an automatic turbine runback is initiated.

B) "ISO-PHASE BUS TROUBLE" and ISO-PHASE BUS LOSS OF COOLING" alarms are received immediately, and an automatic turbine runback is initiated.

C) "ISO-PHASE BUS TROUBLE" alarm is received immediately, followed by "ISO-PHASE BUS LOSS OF COOLING" alarm 10 minutes later. No automatic turbine runback will occur in this condition.

D) "ISO-PHASE BUS TROUBLE" and "ISO-PHASE BUS LOSS OF COOLING" alarms are received immediately. No automatic turbine runback will occur in this condition.

45) Unit 2 is operating at 95% power with all condensate pumps and all feedpumps running. The "A" CONDENSATE PUMP trips on motor overload. The RO verified that vessel level was maintained in the normal band.

Which of the following statements describe the plant response to this trip?

A) A Reactor Recirculation pump runback to 30% occurred due to the "A" condensate pump trip.

B) A Reactor Recirculation pump runback to 45% occurred due to the "A" condensate pump trip.

C) A Reactor Recirculation runback did NOT occur since vessel level was maintained in the normal band.

D) A Reactor Recirculation runback did NOT occur since total feed flow is > 85%.

46) Both Units were operating at full power when the following alarm and indications were received:

- "CONTROL ROOM RAD MONITOR DIV. II INITIATED" (003 A-3)
- MCR Radiation Monitors RI-0760B and RI-0760D were reading approximately 14,000 cpm with their red high lights lit.

Thirty seconds later, the following alarms and indication were received:

- "CONTROL ROOM VENT SUPPLY FAN HI-LO" (003 A-1)
- "CONTROL ROOM VENT SUPPLY LO FLOW CREV START" (003 A-5)
- FR-0765 is reading 200 scfm and dropping.

The Control Room Emergency Ventilation System:

A) Has NOT realigned since the complete initiation logic has not been satisfied.

B) Has NOT realigned as indicated by the low flow condition.

C) Has realigned due to a low flow condition.

D) Has realigned due to a high radiation condition.

47) Unit 2 has experienced a total loss of instrument air with the Instrument Air headers reading 0 psig.

Which of the following statements describe the pneumatic sources available to operate ALL of the Safety Relief Valves (SRVs).

A) Seismic Grade Instrument Gas (via T-261).

B) Instrument Nitrogen (via GP-8E).

C) Backup N2 bottles (via SV 8130 A &B).

D) Relief Valve accumulators.

48) The Standby Liquid Control (SBLC) injection sparger has become clogged with debris.

Which of the following instruments will be impacted by this event?

A) Calibrated Jet Pump flow indication.

B) Core Spray line break detection.

C) Control Rod Drive (CRD) cooling water differential pressure.

D) Core Plate differential pressure.

49) The Traversing In-core Probe (TIP) system is in use with a probe in the core when the reactor scrams on low level following a loss of feedwater. HPCI automatically starts and recovers level, containment parameters are normal.

Which of the following statements describe the expected response of the TIP system to this transient.?

A) TIP automatically retracts from the core, TIP Ball valves close, TIP Nitrogen Purge valves close.

B) TIP automatically retracts from the core, TIP Ball valves close, TIP Nitrogen purge valves remain open.

C) TIP automatically retracts from the core, TIP Ball valves and TIP Nitrogen purge valves remain open.

D) NO TIP system response to a reactor low level condition.

50) The following conditions exist on Unit 3:

- A leak on the "3A" RWCU pump discharge has resulted in Reactor Building Ventilation Stack Radiation Levels rising.
- The Reactor Building Equipment Cell Exhaust had just been aligned to Standby Gas Treatment (SBGT) in accordance with SO 9 when a loss of instrument air occurred.
- The "3A" and "3B" instrument air headers and the Unit 3 service air header are fully depressurized.
- ON-119 has been entered.

Under these conditions, Standby Gas Treatment:

A) Will remain in service. Reactor Building Ventilation will isolate.

B) Will remain in service. Reactor Building Ventilation will NOT isolate.

C) Will NOT remain in service. Reactor Building Ventilation will isolate.

D) Will NOT remain in service. Reactor Building Ventilation will NOT isolate.

51) Unit 2 is operating at 100% power with all 10 Condensate Filter Demineralizers are in service in the AUTOMATIC mode. Condensate filter demin system dP momentarily spikes high and returns to normal. The Reactor Operator acknowledges receipt of the "CONDENSATE FILTER-DEMIN TROUBLE" alarm (20C207L A-2) and notes that the Condensate Filter Demineralizer Bypass Valve (MO-2114) is full open.

After the dP spike returns to normal, Feedpump suction pressure:

A) Will drop, due to Condensate Demin "E" Valve closure.

B) Will drop, due to Condensate Demin "E" Valve opening.

C) Will NOT drop, due to Condensate Demin "E" Valve closure.

D) Will NOT drop, due to Condensate Demin "E" Valve opening.

52) A refueling outage is in progress on Unit 2, with the reactor cavity flooded and the Fuel Pool gates removed. CRDH is in service and RWCU is rejecting inventory to maintain fuel pool level. A trip of the in-service CRD pump occurs. With no operator action, which of the following will occur as a result of the CRD pump trip?

Fuel pool cooling pumps (FCP) will trip on:

A) Low skimmer surge tank level to prevent Fuel Pool pump down.

B) Low skimmer surge tank level to provide FPC pump protection.

C) Low Fuel Pool level to prevent Fuel Pool pump down.

D) Low booster pump suction pressure to provide FPC pump protection.

53) The following annunciators are alarming on Unit 2:

- "B RECIRC FLUID DRIVE SCOOP TUBE BRAKE ON" alarm 214 J-1
- "B RECIRC FLUID DRIVE SCOOP TUBE LOCK" alarm 213 C-3

Given that the "B" Recirc pump is continuing to run, select the condition that caused these alarms.

A) Loss of brake circuit continuity.

B) Recirc Lube Oil pressure at < 15 psi for 20 sec.

C) Loss of scoop tube positioner power.

D) Recirc Lube Oil temperature at 221 degrees F.

54) A LOCA occurs on Unit 2 causing drywell pressure to rise to 12 psig and reactor pressure to drop 200 psig and continue to lower.

The Plant Reactor Operator monitoring the response of "B" LPCI reports receipt of "SYSTEM II RHR INJ. VALVES OVERCURRENT" alarm 226 D-3.

What is the expected response of MO-2-10-25B "Inboard Disch" valve to these conditions?

A) Valve position lights are lit, valve continues to stroke open automatically.

B) Valve position lights are NOT lit, valve continues to stroke open automatically.

C) Valve position lights are lit, valve stroke stops but may be opened manually.

D) Valve position lights are NOT lit, valve stroke stops but may be opened manually.

55) The CRS directs you to use the arm and depress pushbutton to start the Core Spray system. Normal off-site power is available.

After arming and depressing "CS A INITIATION" pushbutton (14A-510A), what is the expected response of the Core Spray system?

A) "A" and "C" Core Spray pumps start immediately.

B) "A" and "C" Core Spray pumps start after a time delay.

C) "A", "B" ,"C", and "D" Core Spray pumps start immediately.

D) "A", "B", "C", and "D" Core Spray pumps start after a time delay.

56) Unit 2 is in MODE 2 with a start up in progress. Which of the following valid alarm conditions will prevent control rod insertion using "Emergency Rod In".

A) "APRM FLOW BIAS OFF NORMAL" alarm 211 A-4.

B) "RWM ROD BLOCK" alarm 211 F-5.

C) "RBM DOWNSCALE" alarm 211 C-4.

D) "A WRNM TRIP/INOP" alarm 210 G-3.

57) Unit 2 is operating at 100% power when the PRO responds to a "OFF-GAS TROUBLE" alarm. The "A" SJAE and "A" Jet Compressor are in service. At the Recombiner Panel (00C196) the PRO acknowledges the first in annunciator for Jet Compressor "STEAM FLOW LOW" alarm 231 A-3.

If this alarm condition persists, what will be the expected response of the off-gas system?

A) Jet Compressor "STEAM" supply valve MO-2990A close.

B) Recombiner "RECYCLE" valves AO-2791 and AO-2792 opens.

C) SJAE "A 1st STAGE" steam supply valves AO 2238 A/B/C close.

D) SJAE "OFF-GAS INLET" valves AO 2236A/B/C close.

58) A plant startup is in progress on Unit 2 with Reactor power at 40%. A loss of stator cooling has occurred and the Turbine Generator has runback. Generator megavars indicate +100 megavars lagging. You have been directed to reduce generator VARS to minimum.

Select the operator action listed below which will reduce generator VARS to minimum.

A) RAISE the AUTO VOLTAGE REGULATOR RHEOSTAT setpoint.

B) LOWER the AUTO VOLTAGE REGULATOR RHEOSTAT setpoint.

C) RAISE the MAN DC VOLT REG setpoint.

D) Lower the MAN DC VOLT REG setpoint.

59) Unit 3 was operating in MODE 1 at full power when it experienced a loss of all off-site power. All 4 Diesel Generators have started and closed in on their buses.

Under these conditions, which of the following components will continue to receive cooling water flow?

A) Instrument Nitrogen Compressor Coolers.

B) Station Air Compressors.

C) Reactor Water Cleanup Non-regenerative Heat Exchangers.

D) Condensate Pump Coolers.

60) Sufficient NPSH for unrestricted Reactor Recirculation pump operation is assured by:

A) The height of water above the pump suction

B) Feedwater flow

C) Steam Dryer return flow

D) RPV pressure

61) Unit 2 is in MODE 1 at 100% power, both Recirc pumps in service at 76% speed.

Which of the following conditions will result in an "A" Recirc Drive Motor Breaker trip?

A) Aux Bus #1 Low Voltage.

B) Exciter field breaker opens.

C) Recirc Pump Cooling Water Low Flow (RBCCW).

D) Recirc Pump Motor High Vibration.

62) Which of the following is the most likely cause of a "RBCCW Head Tank High Level" alarm?

A) Broken tube inside the in-service RBCCW Heat Exchanger.

B) RBCCW makeup valve (AO-2440) failure.

C) Tube rupture in RWCU regenerative Heat Exchanger.

D) Reactor Recirc Pump Seal Cooler internal leak.

63) Unit 3 was operating at full power when a reactor scram occurred due to a low reactor level condition. Mechanical binding prevented the scram inlet valve on control rod 26-31 from opening.

Determine the expected indications for this control rod.

A) Full Core Display blue light lit, CRDM damage will prevent rod insertion.

B) Full Core Display blue light lit, rod will move in slowly.

C) Full Core Display blue light NOT lit, CRDM damage will prevent rod insertion.

D) Full Core Display blue light NOT lit, rod will move in slowly.

64) Unit 3 is in a refueling outage with a core shuffle in progress.

Which of the following unanticipated conditions is a symptom requiring entry into ON-124, Fuel Floor and Fuel Handling Problems?

A) Inservice, operable "A" Wide Range Neutron Monitor count rate doubles between CCTAS steps during fuel handling.

B) An irradiated fuel support piece is dropped in the fuel transfer canal (cattle chute) during movement to spent fuel pool.

C) A Fuel Storage Pool High Radiation alarm is received.

D) A Refuel Floor Vent Exhaust High Radiation alarm is received.

65) To deinert the containment for personnel entry, a Unit 2 Drywell vent and purge is in progress exhausting through the Inboard and Outboard 18" Vents (AO-2506 and AO-2507) using SBGT.

Which of the following conditions will result in an auto closure of these valves.

A) Drywell radiation monitor reading exceeds the setpoint of 3.4 E-3 uCi/cc.

B) Main Stack radiation Hi Hi exceeds the setpoint of 1 E-2 uCi/cc.

C) 2 vent exh stack rad monitor Hi Hi exceeds 5E-5 uCi/cc.

D) Containment High Range Rad Monitor Hi, exceeds 16 R/hr.

66) Unit 3 was operating at 50% power when it experienced a loss of vacuum transient. Currently power is 20% and condenser vacuum is 24.5" and steady. A circ water problem has been discovered to be the cause of the vacuum loss. Maintenance estimates that it will be 4 hours until the circ water problem will be corrected. Under these conditions, the next operator action is to:

A) Continue to reduce power.

B) Hold power constant.

C) Trip the main turbine.

D) Scram and enter T-100.

67) Unit 2 is experiencing a low condenser vacuum transient and has entered OT-106, Condenser Low Vacuum. Vacuum is currently 24.5" Hg and dropping. The "C CONDENSER LO VAC" annunciator (203 D-2) is lit.

During a brief the CRS states that a full reactor scram will not be received until the "A" or "B" low vacuum alarms come in.

The CRS statement is:

A) Correct, since the "C" condenser provides only a "A" Channel RPS input.

B) Correct, since the scram setpoint cannot be achieved without all three condenser losing vacuum.

C) Incorrect, since the "C" condenser inputs into both RPS channels.

D) Incorrect, since low vacuum in any condenser (A, B, C) can result in a full scram.

68) Unit 2 has experienced a high drywell pressure transient. The reactor has been scrammed and the URO is controlling level manually. Due to overfeeding with HPCI, reactor level has exceeded the band of +5" to +35", HPCI was then secured.

Current conditions are:

- Reactor Pressure 1000 psig.
- Narrow range level indicators are upscale.
- Wide Range Level indicators reads +45" to +50" and steady.
- LI-2-2-3-86 is reading +75" and steady (figure 1 is attached).

Actual level should be verified using:

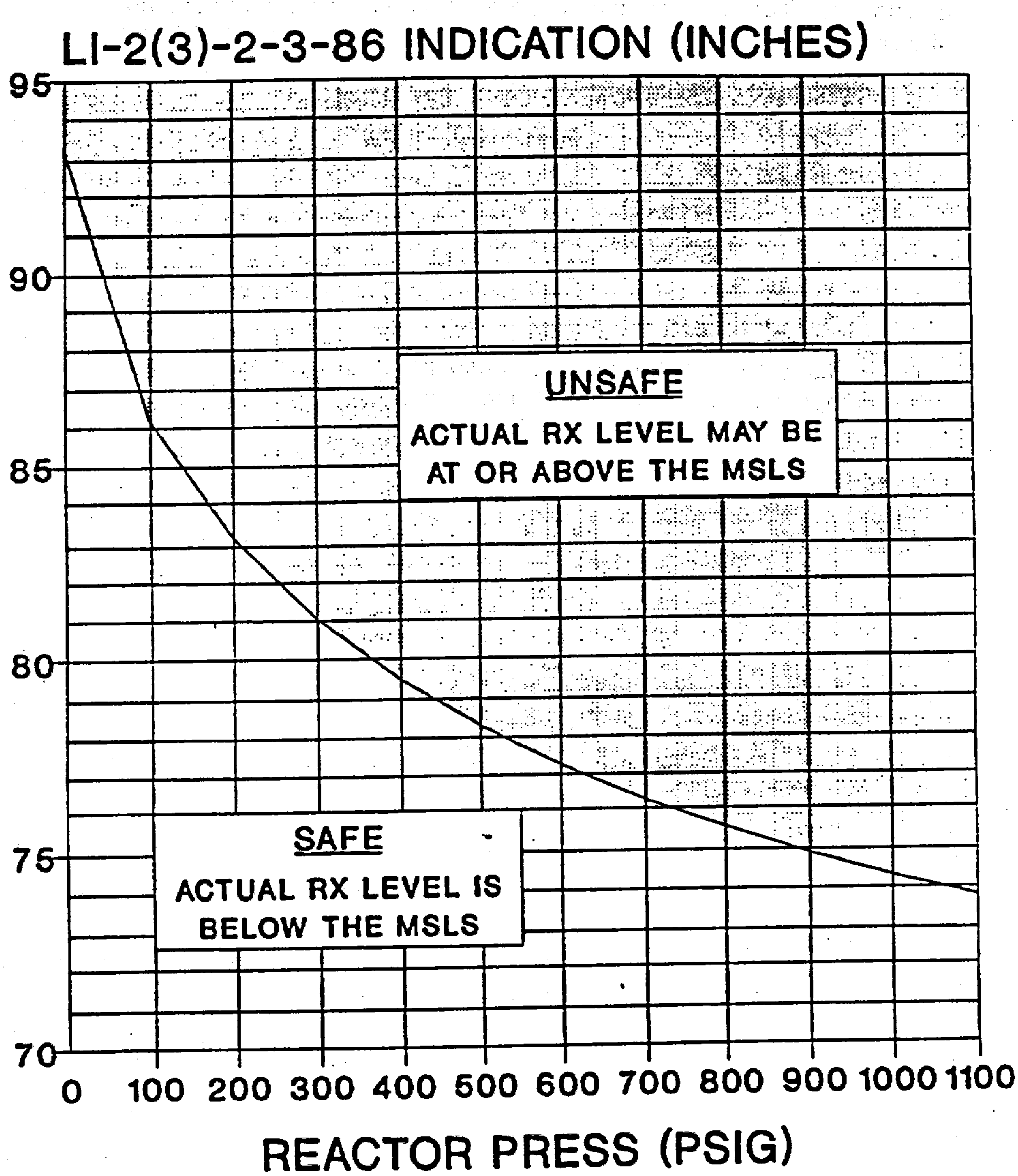
A) LI-2-2-3-86 and is currently ABOVE the Main Steam Lines.

B) LI-2-2-3-86 and is currently BELOW the Main Steam Lines.

C) Wide Range indication and is currently ABOVE the Main Steam Lines.

D) Wide Range indication and is currently BELOW the Main Steam Lines.

FIGURE 1


LI-2(3)-2-3-86 INDICATION (INCHES)
95
90
85
80
75
70
UNSAFE
ACTUAL RX LEVEL MAY BE AT OR ABOVE THE MSLS
SAFE
ACTUAL RX LEVEL IS BELOW THE MSLS
0 100 200 300 400 500 600 700 800 900 1000 1100
REACTOR PRESS (PSIG)

69) ON-118, loss of Turbine Building Closed Cooling Water (TBCCW), directs that if TBCCW cannot be restored, Main Turbine Generator load should be reduced to less than 18,000 amps in accordance with GP-9-2.

The basis for this ON-118 step is to:

A) Permit heat generated by the isophase bus bars to be absorbed by the environment.

B) Reduce the heat load so that RBCCW is not overloaded when it backs up TBCCW.

C) Permit condensate pumps to be alternated to prevent condensate pump overheating.

D) Reduce the heat load on TBCCW so that the station air compressors will NOT trip on high temperature.

70) ON-119, Loss of Instrument Air, directs that the reactor be scrammed if any rod begins to drift in due to lowering scram pilot air header pressure.

What is the bases for this direction?

A) To ensure that the scram discharge volume is fully isolated during the scram.

B) To ensure that various scram valve opening pressures do not result in a random rod pattern.

C) To ensure that the individual control rod scram inlet valves do not open before the scram outlet valves.

D) To ensure that sufficient volume exists in the scram discharge volume to complete a full scram.

71) A loose fitting has resulted in the loss of instrument air to the in-service Control Rod Drive (CRD) Flow Control Valve (AO-19).

Determine which of the following conditions could result from this instrument air loss.

A) Control Rod Drive accumulator alarms due to low pressure.

B) Control Rod Drive alarms due to high temperatures.

C) Control Rods begin to drift due to excessive flow.

D) High rod speeds during control rod withdrawal.

72) Unit 2 has experienced a loss of shutdown cooling, ON-125, Loss of Shutdown Cooling, directs you to determine the expected decay heat load using Operator Aid 95-04 located on the back of Panel 20C005A.

The information necessary to determine expected heat load using this Operator Aid is:

A) Current heat up rate.

B) Current WRNM indicated power.

C) Power history before shutdown.

D) Elapsed time since shutdown.

73) A Unit 3 reactor startup was in progress with reactor pressure at 500 psig and reactor power at 1% when the running "A" Control Rod Drive (CRD) pump tripped. A start of the "B" CRD pump is in progress. CRD charging header pressure was 920 psig and dropping when 3 accumulator trouble alarms were received on withdrawn control rods.

Under these conditions, ON-107, Loss of CRD Regulation Function, directs that a full reactor scram be inserted. The basis for this direction is that:

A) At this reactor pressure, operable HCU accumulators are required to ensure proper scram force.

B) At this reactor pressure the CRDM ball check valves will NOT reposition to permit reactor pressure to insert the control rods.

C) This condition may result in unanalyzed rod patterns due to rods inserting randomly on low accumulator pressure.

D) This condition exceeds the Tech Spec Limit for the number of withdrawn control rods that can be declared "slow".

74) Unit 2 has experienced a drywell steam leak with an ATWS. Current conditions are as follows:

- Reactor pressure being maintained 950-1050 psig.
- TI-2501 point 126 is not available.
- TI-2501 point 127 indicates 520 degrees F.
- Narrow range RPV level indicates +5 inches.
- Wide range RPV level indicates -115 inches.
- Fuel Zone RPV level range indicates -125 inches.
- Refuel range RPV level (Shutdown Range Instrument LI-2-2-3-86) indicates -21 inches.

Evaluate the above conditions and then use Table DW/T-1 from T-102 (attached), "Primary Containment Control" to determine which RPV level indication ranges may be used.

A) Narrow Range and Refuel Range

B) Narrow Range and Wide Range

C) Wide Range and Fuel Zone Range

D) Fuel Zone Range and Refuel Range

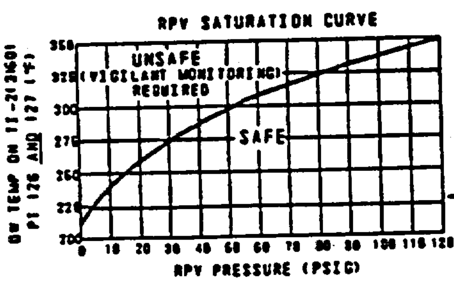
TABLE DW/T-1

RPV LEVEL INSTRUMENT STATUS

AN RPV LEVEL INSTRUMENT MAY BE USED TO DETERMINE RPV LEVEL ONLY WHEN THE FOLLOWING CONDITIONS ARE SATISFIED:

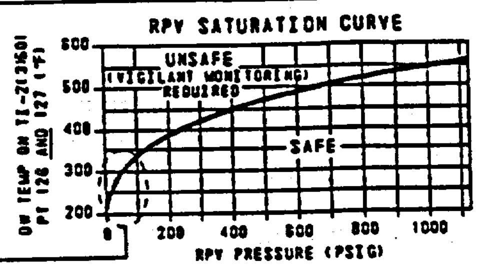
NOTES: 1. IF BOTH POINTS 126 AND 127 ARE AVAILABLE, THEN BOTH POINTS MUST PLOT "SAFE" TO CONSIDER A LEVEL INSTRUMENT AVAILABLE

2. IF EITHER POINT 126 OR 127 IS NOT AVAILABLE, THEN THE REMAINING POINT MUST PLOT "SAFE" TO CONSIDER A LEVEL INSTRUMENT AVAILABLE



SEE DETAIL

ALL RPV LEVEL INSTRUMENTS



IF DW TEMP AND RPV PRESS ARE ON THE UNSAFE SIDE OF THE RPV SATURATION CURVE

AND AN INSTRUMENT EXHIBITS AN UNEXPLAINED TREND OR OSCILLATION,

THEN THAT INSTRUMENT IS UNAVAILABLE

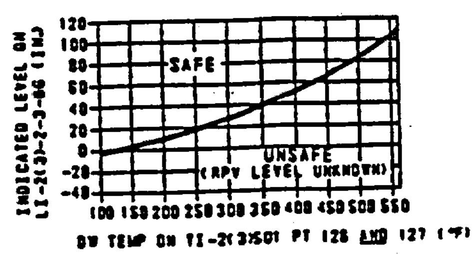
WIDE AND NARROW RANGE INSTS ONLY

FOR EACH OF THE INSTRUMENTS IN THE TABLE, THE INSTRUMENT READS ABOVE THE MIN INDICATED LEVEL OR THE TEMP NEAR THE DW REFERENCE LEG VERTICAL RUNS (TI-2(3)501 PT 126 AND 127) ARE BELOW THE MAX RUN TEMP.

INSTRUMENT	MIN INDICATED LEVEL IS ABOVE	OR	MAX RUN TEMP IS BELOW
NARROW RANGE	10 IN.	OR	450°F
WIDE RANGE	-120 IN.	OR	500°F

SHUTDOWN RANGE INST LI-2(3)-2-3-86 ONLY

LI-2(3)-2-3-86 READS ON THE SAFE SIDE OF THE CURVE



75) T-102, Primary Containment Control, provides direction to maintain Torus level in the band of 14.5 ft. to 14.9 ft.. In accordance with the TRIP Bases what is the first concern during a rising torus level transient?

A) Submerging the Reactor Building to Torus Vacuum Breaker Line.

B) Excessive stress on SRV tail pipes.

C) Submergence of the Torus Spray Header.

D) Excessive stress on ECCS suction piping.

76) T-102 step PC/P-6 directs use of Torus Sprays before Torus pressure reaches 9 psig if Torus level is below 21 ft.

What are the bases for the 9 psig and 21 feet limitations?

A) Threshold for downcomer chugging and Torus Spray header becomes submerged.

B) Threshold for downcomer chugging and Torus to drywell vacuum breakers become submerged.

C) Threshold for evaporative cooling and Torus to drywell vacuum breakers become submerged.

D) Threshold for evaporative cooling and Torus Spray header becomes submerged.

77) Plant conditions on Unit 3 are as follows:

- A steam leak exists in the Unit 3 Reactor Building.
- The Reactor has been shutdown and depressurized to a steady value of 30 psig.
- TR-3-13-139 point 22 indicates 325 degrees F.
- Wide Range RPV level indicates -150 inches.
- Fuel Zone RPV level indicates -172 inches.

Evaluate the above conditions and then use Table SC/T-4 from T-103, "Secondary Containment Control" (attached) to determine which RPV level range may be used.

A) Wide Range may be used, Vigilant Monitoring required.

B) Wide Range may be used, Vigilant Monitoring NOT required.

C) Fuel Zone may be used, Vigilant Monitoring required.

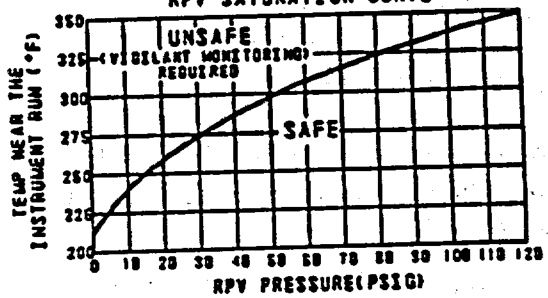
D) Fuel Zone may be used, Vigilant Monitoring NOT required.

TABLE SC/T-4
RPV LEVEL INSTRUMENT STATUS

AN RPV LEVEL INSTRUMENT MAY BE USED TO DETERMINE RPV LEVEL ONLY WHEN THE FOLLOWING CONDITIONS ARE SATISFIED:

ALL RPV LEVEL INSTRUMENTS

RPV SATURATION CURVE



__IF__ THE TEMP NEAR THE RX BLDG INSTRUMENT RUNS ARE IN THE UNSAFE REGION OF THE RPV SATURATION CURVE AS DETERMINED BY THE LOCAL INSPECTION OR TR-2(3)-13-139 PT 22

__AND__ AN INSTRUMENT EXHIBITS AN UNEXPLAINED TREND OR OSCILLATION,

__THEN__ THAT INSTRUMENT IS UNAVAILABLE

WIDE RANGE AND FUEL ZONE INSTS ONLY

FOR EACH OF THE INSTRUMENTS IN THE TABLE, THE INSTRUMENT READS __ABOVE__ THE MIN INDICATION LEVEL __OR__ THE TEMP NEAR THE RX BLDG REFERENCE LEG VERTICAL RUNS ARE __BELOW__ THE MAX RUN TEMP AS DETERMINED BY LOCAL INSPECTION OR TR-2(3)-13-139 PT 22.

INSTRUMENT	MIN INDICATED LEVEL IS ABOVE	OR	MAX RUN TEMP IS BELOW
WIDE RANGE	-120 IN.	OR	140°F
FUEL ZONE	-305 IN.	OR	315°F

78) A Designated Alternate (DA) is moving an old jet pump in the Unit 2 fuel pool when it falls off the auxiliary hoist. It is reported to the Control Room that a jet pump fell on an irradiated fuel bundle and damaged some fuel pins.

The Control Room also receives the following alarms and indications

- Refueling Floor Vent Exhaust Hi Radiation (218 A-1)
- Reac. Bldg. Zone Vent Exhaust Hi Radiation (218 B-1)
- Reac. Bldg. Or Refueling Floor Vent Exh. Hi Rad Trip (218 D-4)
- Refueling Floor Radiation Trip Units A and D High lights are lit.

Evaluate these conditions and determine the expected ventilation lineup.

A) Reactor Building Ventilation trips.
Refuel Floor Ventilation trips.
SBGT initiates and aligns to the entire Reactor Building/Refuel Floor.

B) Reactor Building Ventilation continues to run.
Refuel Floor Ventilation trips.
SBGT initiates and aligns to the Refuel Floor.

C) Reactor Building Ventilation continues to run.
Refuel Floor Ventilation continues to run.
SBGT initiates and aligns to the Refuel Floor.

D) Reactor Building Ventilation continues to run.
Refuel Floor Ventilation continues to run.
SBGT remains in standby.

79) Unit 2 is in T-103, "Secondary Containment Control", due to high water level condition in Secondary Containment. The Reactor has been conservatively scrammed and the Group II/III isolations (from the level shrink) are complete.

The CRS is currently attempting to determine whether a Primary System is discharging into the Reactor Building. Given the above conditions, evaluate the following and determine which constitutes a primary system discharging into the Reactor Building.

A) Leakage from a pipe flange on the discharge of the Reactor Water Cleanup Non-regenerative Heat Exchanger.

B) Steam leakage from a rupture on the piping of the #2 Main Steam stop valve inlet.

C) Leakage from a weld crack on the "A" RHR suction piping penetration to the Torus.

D) Steam leakage from the Standby Liquid Control Injection line just outboard of the drywell penetration.

80) Unit 3 was operating in MODE 1 at 75% power when a fire was reported in the Reactor Building 135' elevation. The Crew has entered ON-114, the procedure for an "actual fire", and the CRS has directed that the Equipment Operator isolate the RPV Condensing Chamber Backfill System.

The basis for isolation of this system under these conditions is to prevent inaccurate level indication and unreliable automatic initiations due to:

A) Lowering Instrumentation Variable Leg density.

B) Raising Instrumentation Variable Leg density.

C) Lowering Instrumentation Reference Leg density.

D) Rising Instrumentation Reference Leg density.

81) Unit 2 was operating in MODE 1 at 40% power when it experienced a loss of 20Y050. All required control room actions have been completed.

Under these conditions, operator actions will be impacted by a loss of power to:

A) The RBCCW backup of DWCW which will require manual transfer.

B) The lighting in vital areas which will require the use of flashlights.

C) The Fire Alarm Panel which will require continuous roving fire watches.

D) The Control Room radios which will require the use of alternate communications.

82) Unit 2 is operating in MODE 1 at 100% power when the following occurs:

- "REACTOR HI PRESS" alarm 210 G-2 annunciates.
- Reactor Pressure indicates 1075 psig and rising slowly.

In accordance with OT-102 "Reactor High Pressure" which of the following is an appropriate immediate operator action?

A) Control reactor pressure by raising the Bypass Jack setting.

B) Control reactor pressure by lowering the Max Combined Flow Limit Pot.

C) Control reactor pressure by lowering reactor power.

D) Control reactor pressure by raising the Max Combined Flow Limit Pot.

83) Unit 2 is operating at 87% power when the "A" Condensate pump shaft coupling shears. The Condensate pump continues to run at low motor amps.

Given that all three Reactor Feedpumps (RFPs) remain in service and no Operator action is taken, what is the expected plant response to this event?

A) A Recirc runback to 45% speed will occur immediately.

B) A Recirc runback to 45% speed will occur when level is less than +17".

C) A Recirc runback to 30% speed will occur immediately.

D) A Reactor scram will occur when level is less than +1".

84) Unit 2 is at 100% power when Drywell pressure begins to rise.

In accordance with OT-101 "HIGH DRYWELL PRESSURE" follow up actions the following parameters and alarms are noted.

- "A RECIRC PUMP SEAL STAGE 2 HI FLOW" alarm 214 A-1
- PI-2-02-2-033A "Seal 1 Inner" 1056 psig
- PI-2-02-2-032A "Seal 2 Outer" 1043 psig

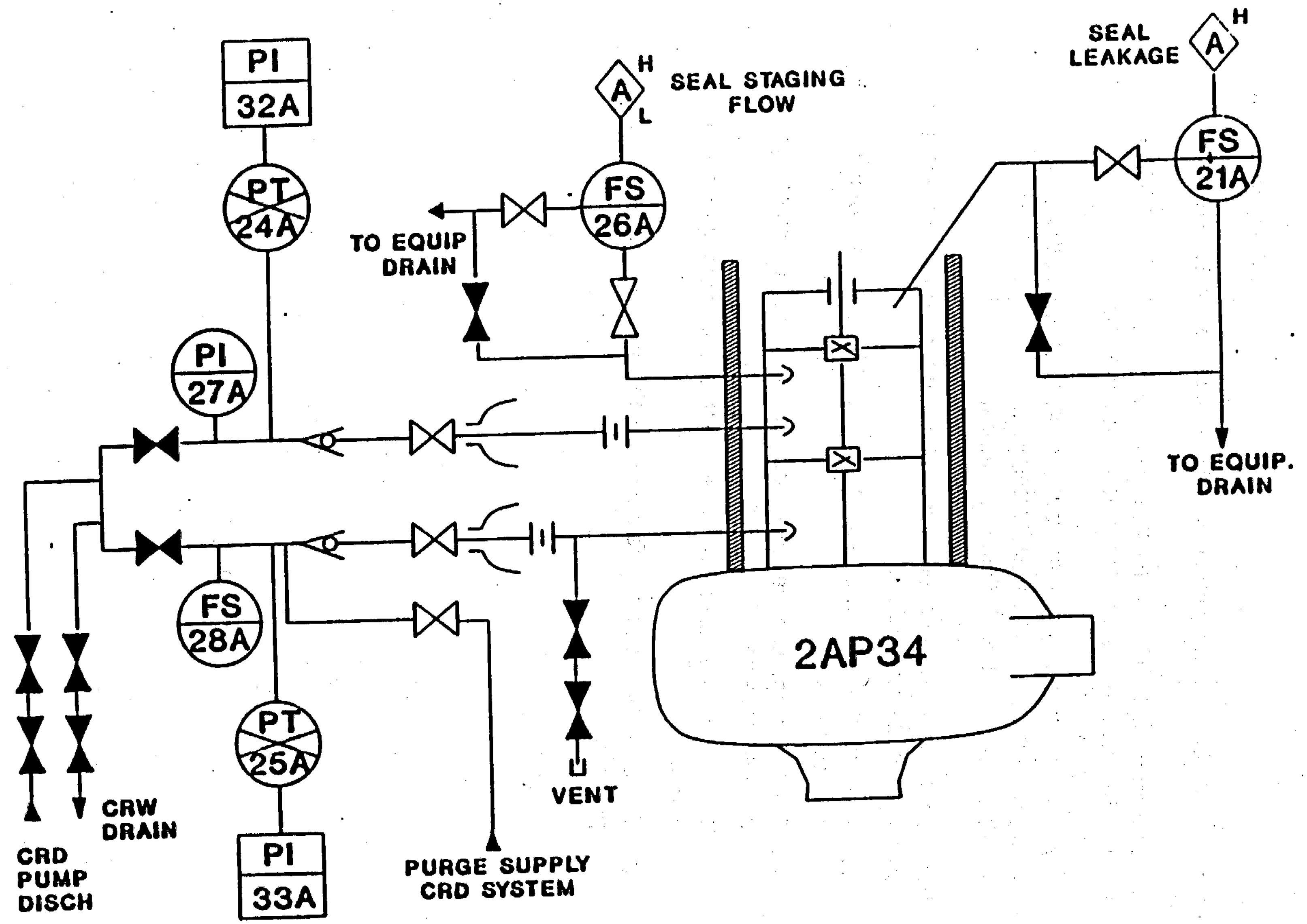
Evaluate these indications, using the attached drawing, and select the appropriate statement below.

A) The 1st stage seal has failed but it is NOT the source of high drywell pressure.

B) The 2nd stage seal has failed but it is NOT the source of high drywell pressure.

C) The 1st stage seal has failed and is the source of high drywell pressure.

D) The 2nd stage seal has failed and is the source of high drywell pressure.



RECIRCULATION PUMP SEAL PIPING
T-LOT-0030-3

Rev 0

85) Unit 3 was operating at 70% power when it experienced a rising drywell pressure. Using OT-101, High Drywell Pressure, the source of the leak has been determined to be the "A" Recirculation pump seals. The CRS has directed you to trip and isolate the "A" Recirculation pump.

Given these conditions, what is the proper sequence for isolating the recirculation pump and why?

A) Shut the suction valve first since it can close against a higher dP.

B) Shut the discharge valve first since it can close against a higher dP.

C) Shut the suction valve first since it is limited to closing against a lower dP.

D) Shut the discharge valve first since it is limited to closing against a lower dP.

86) Unit 2 is at 100% power.

Which of the following events would require power to be reduced or maintained in accordance with OT-104, "Positive Reactivity Insertion"?

A) "A" Reactor Feedpump min flow valve fails open.

B) EHC pressure set setpoint drops 10 psi.

C) Condensate pump trip.

D) Loss of RBCCW to RWCU Non-regen Heat Exchanger.

87) Which of the following is the reason why the Main Steam Isolation Valves (MSIV) are closed prior to evacuating the Main Control Room in accordance with SE-1, "Plant Shutdown from the Remote Shutdown Panel"?

A) With MSIVs closed, all reactor inventory and pressure control may take place at the Remote Shutdown Panel.

B) Since plant release points cannot be monitored at the Remote Shutdown Panel, closing the MSIVs precludes any concern for off-site releases.

C) The MSIV closure outside the Main Control Room requires access to plant areas that may not be accessible during an evacuation.

D) If the MSIVs are closed from outside the Main Control Room, there is no method for verification of complete closure.

88) Unit 2 Reactor Operator is controlling reactor level using HPCI at the Unit 2 Alternate Shutdown Panel following Control Room Abandonment. Indicated reactor level on LI-2-2-3-112 is currently 20" and reactor pressure is 500 psig. Using SE-10 Attachment 9, provided, determine the current reactor level and the expected HPCI response if an actual high level condition occurs.

A) Actual level is > 40", HPCI will automatically trip on high level condition.

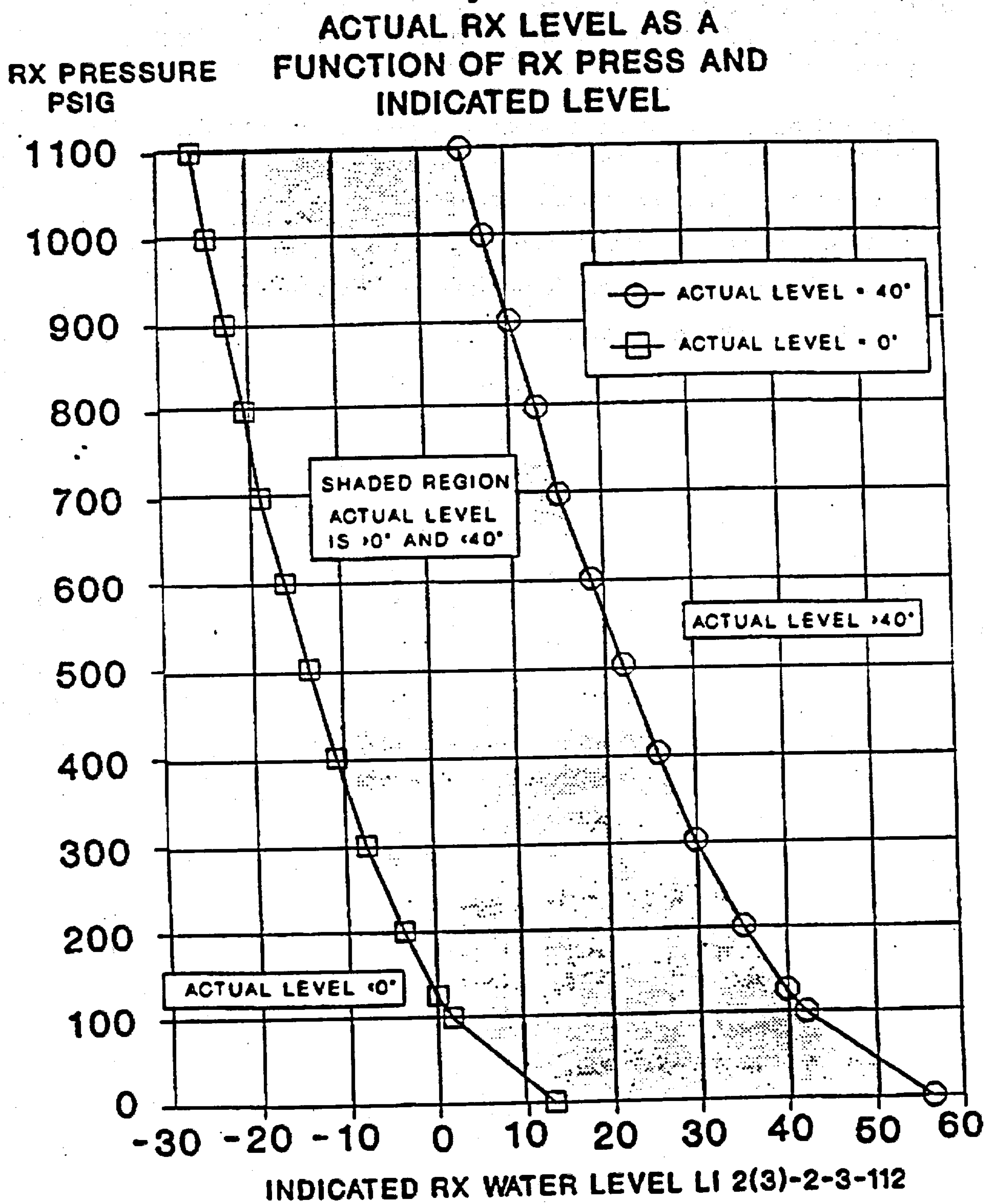
B) Actual level is > 40", HPCI must be manually tripped on high level condition.

C) Actual level is between 0" and 40", HPCI will automatically trip on a high level condition.

D) Actual level is between 0" and 40", HPCI must be manually tripped on a high level condition.

SE-10 Attachment 9
Figure 1

SE-10 Attachment 9
Figure 1



89) Unit 2 was operating at full power in MODE 1 when a positive reactivity event occurred due to a control rod drifting out. The CRS has directed you to monitor for evidence of fuel damage.

Which of the following indications would be the first indication of a small fuel pin leak from this transient?

A) Main Steam Line Radiation Recorders.

B) Air Ejector Discharge Log Monitor Recorders.

C) Off-Gas Adsorber Outlet Radiation Indication.

D) Main Stack Gas Recorder.

90) Following a LOCA on Unit 2 the CRS directs restoration of Drywell Cooling, using T-223, "Drywell Cooler Fan Bypass" for Drywell pressure control. The Unit Reactor Operator reports that the Drywell Cooler fans cannot be placed inservice without an engineering evaluation due to plant conditions falling on the UNSAFE side of T-223 Figure 1, "Drywell Chilled Water (DWCW) Saturation Curve.

Which of the following describes the basis for restricting Drywell Fan restoration when on the UNSAFE side of the curve?

A) Water hammer and rupture of piping inboard of DWCW Isolation valves when flow is restored.

B) Inadvertent lifting of overpressure relief valves inboard of the DWCW Isolation valves when flow is restored.

C) Overcurrent trips of the Drywell Cooler Fans if restarted with a LOCA condition.

D) Overpressurization and rupture of piping inboard of the closed DWCW Isolation valves with a LOCA condition.

91) Unit 2 was operating at 100% power when a total loss of Instrument Air occurred resulting in a plant scram. T-101, "RPV Control" was entered on high reactor pressure at the time of the scram. Normal scram actions have been completed, no other actions have been performed.

In accordance with T-101, RPV pressure control leg, which of the following is the correct method for pressure control under these conditions?

A) Manual operation of SRVs between 950 psig and 1050 psig.

B) Automatic operation of the EHC system at 920 psig.

C) Manual operation of ADS SRVs to stabilize pressure below 1050 psig.

D) Automatic operation of SRVs at their setpoint.

92) Unit 3 has experienced a reactor scram following a steam leak in the Drywell. The CRS directs restoration of Drywell Instrument Nitrogen from T-101, RPV Control, to permit manual reactor pressure control. Restoring Instrument Nitrogen to the Drywell in accordance with GP-8E, "Primary Containment Isolation Bypass":

A) May contribute to a flammable environment in the Drywell.

B) Will only supply nitrogen to the "B" Instrument Nitrogen Header.

C) May deplete CAD nitrogen tank inventory.

D) Will only be permitted if Instrument Air Header pressure is greater than Drywell pressure.

93) Unit 3 has experienced a transient and the following is observed:

- Torus pressure: 9 psig
- Torus temperature: 200 degrees F
- Torus level: 14 feet
- Reactor pressure: 1000 psig
- RHR "A" Loop Flow: 23,000 gpm
- Core Spray "B" Loop Flow: 7500 gpm
- All other low pressure ECCS pump are NOT in service.

Use the attached T-102 Sheet 3 curves to determine if Net Positive Suction Head (NPSH) requirements are being met.

A) There is sufficient NPSH for the "A" Loop of the RHR ONLY.

B) There is sufficient NPSH for the "B" Loop of Core Spray ONLY.

C) There is sufficient NPSH for both the "A" Loop of RHR and the "B" Loop of Core Spray.

D) There is NOT sufficient NPSH for either the "A" Loop of RHR or the "B" Loop of Core Spray.

94) For a lowering suppression pool level T-102, "Torus Level", directs that if Torus level cannot be maintained above 9.5' secure HPCI. It does not direct that RCIC be secured until < 6'.

What is the basis for securing HPCI but not RCIC at 9.5'?

A) HPCI turbine exhaust becomes uncovered at 9.5', RCIC turbine exhaust becomes uncovered at 6'.

B) HPCI turbine exhaust becomes uncovered at 9.5', RCIC turbine exhaust is an insignificant containment input.

C) HPCI NPSH becomes a concern at 9.5', RCIC turbine exhaust becomes uncovered at 6'.

D) HPCI NPSH becomes a concern at 9.5', RCIC turbine exhaust is a insignificant containment input.

95) T-111, "Level Restoration" was entered on Unit 3 following a loss of all off-site power and a failure of all diesel generators to start. Current plant conditions are as follows:

- Reactor pressure is 800 psig.
- Reactor level -195" and dropping slowly.
- HPCI tripped on a loss of lube oil.
- RCIC is blocked out of service.

Evaluate these plant conditions and determine the status of Adequate Core Cooling (ACC).

A) ACC exists until level is below -200".

B) ACC exists until level is below -210".

C) ACC does NOT exist, since level is below -172".

D) ACC does NOT exist, since injection is not present.

96) Level recorder LR-2-02-3-110A blue pen is fed by LT-2-02-3-072C "Wide Range" and LT-2-02-3-073C "Fuel Zone" level transmitters.

If level transmitter LT-73C failed upscale and then actual reactor level dropped to -172", what would be the impact on vessel level indications and ECCS initiation from reactor level?

A) LR-110A blue pen input would swap at -100", low level ECCS initiations would NOT be impacted.

B) LR-110A blue pen input would swap at -100", low level ECCS initiations would be impacted.

C) LR-110A blue pen input would NOT swap at -100", low level ECCS initiations would NOT be impacted.

D) LR-110A blue pen input would NOT swap at -100" low level ECCS initiations would be impacted.

97) Following an ATWS and Group I Isolation on Unit 2, the following conditions exist:

- Reactor power: 30%
- Reactor level: -100"
- Torus temperature: 115 degrees F.
- SRV's A, B, C, G open

T-117 level power control directs RPV injection be terminated and prevented using T-240.

For the conditions listed above, which of the following concerns is the basis for performing T-240?

A) Uncontrolled injection of large amounts of cold water.

B) Power generation which is a threat to primary containment.

C) Neutron flux oscillations which challenge fuel clad integrity.

D) Power excursions while establishing minimum alternative RPV flooding pressure.

98) Unit 2 was operating at 100% power when a Reactor high pressure scram condition occurred due to a total loss of instrument air. Control rods failed to insert, reactor pressure peaked at 1180 psig.

The following plant conditions currently exist:

- Reactor power: 35%
- Reactor level: +23"
- Reactor pressure: 1140 psig
- Full core display blue lights lit
- A & B Air Header pressure: 0 psig

Determine which of the following TRIP procedures will insert the control rods.

A) T-213, "Scram Solenoid De-Energization"

B) T-214, "Isolating and Venting the Scram Air Header"

C) T-215, "Control Rod Insertion by Withdraw Line Venting"

D) T-216 "Control Rod Insertion by Manual Scram or Individual Scram Test Switches"

99) A steam leak exists in the Unit 3 Turbine Building. T-104, "Radioactivity Release", has been entered due to high ventilation stack radiation alarms. The Equipment Operator (EO) then reports that Turbine Building Ventilation is tripped.

Under these conditions, determine the appropriate response to the EO's report that Turbine Building Ventilation is tripped.

A) Restart ventilation to monitor the release.

B) Restart ventilation to lower the radioactive release.

C) Maintain ventilation tripped to prevent an unmonitored release.

D) Maintain ventilation tripped to lower the radioactive release.

100) For which of the following conditions would direction be given to initiate Drywell Sprays regardless of whether Adequate Core Cooling is assured?

A) To prevent exceeding the Pressure Suppression Pressure Limit.

B) To maintain Drywell pressure below the Drywell Spray Initiation Limit.

C) To mitigate the consequence of a H2 deflagration.

D) To mitigate the consequences of containment overpressurization.

SCANTRON FORM NO 90010-PECO

PART 1

KEY

FEED THIS DIRECTION

PECO NUCLEAR

STATION PB / LGS

FORM RO

COURSE TITLE 1999 Peach Bottom RO Exam

NAME (PRINT) Examination Key
last first mi

SOCIAL SECURITY NUMBER

COMPANY / PECO PAYROLL #

DATE 13 SEPT 99

I HAVE REVIEWED AND UNDERSTAND THE CORRECTED QUIZ; ALL WORK ON THIS EXAMINATION IS MY OWN, I HAVE NEITHER GIVEN NOR RECEIVED ASSISTANCE

signature

IMPORTANT

- USE #2 PENCIL
- EXAMPLE: A B ● D E
- ERASE *COMPLETELY* TO CHANGE

50

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

PART 2

Attachment 3

SRO WRITTEN EXAM WITH ANSWER KEY

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region: I / II / III / IV
Date: September 13, 1999	Facility/Unit: Peach Bottom 2 & 3
License Level: RO / SRO	Reactor Type: W / CE / BW / GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	________ Points
Applicant's Score	________ Points
Applicant's Grade	________ Percent

Test ID: 1999 SRO Exam Level: SRO Date: 9/13/99

Title: 1999 PEACH BOTTOM NRC SRO EXAM Facility: PBAPS

ID Number: 1

1) Given the following conditions:
- The Control Room Supervisor (CRS) has delegated completion of GP-3, "Normal Plant Shutdown" for Unit 3 to a fully qualified Senior Reactor Operator (SRO)
- This has been logged in the Unified Control Room Log
- During the Unit 3 shutdown a problem requires entry into T-103, "Secondary Containment Control"
- Unit 2 is operating at 75% power during this time

Which of the following delineates the responsibility for command and control authority on each of the two Units for these conditions?

A) The CRS shall retain command and control over both Units at all times

B) The Unit 3 SRO retains command and control over Unit 3 until an emergency no longer exists. The CRS retains command and control over Unit 2.

C) The Senior Manager - Operations shall assume command and control over both Units upon his arrival.

D) The Unit 3 SRO immediately transfers Unit 3 command and control to the Shift Manager and provides support and backup to the CRS on both Units.

2) A Reactor Operator has just begun night work following his long break (seven days).

- Day 1 he works 1845 - 0645
- Day 2 he works 1845 - 0645
- Day 3 he works 2245 - 0645 (4 hours vacation)
- Day 4 he works 1845 - 0945 (to cover for a sick RO)

At 1200, the RO gets a call at home from the Shift Operations Assistant (SOA) requesting that he return to work as soon as possible to fill a vacancy. To stay within the bounds of A-40, without deviations, the RO may:

A) Return to work immediately, but can only work 8 hours.

B) Return to work at 1745, but can only work 9 hours.

C) Return to work at 1845, and work a regular 12 hour shift.

D) Return to work at 2145, and work up to 16 hours.

3) Both units are operating in MODE 1 at full power with no Surveillance Testing or other evolutions in progress. The Shift Manger receives a call indicating that one of the licensed operators has been selected for a random substance screening. The selected operator is currently the Unit 2 Reactor Operator (URO). The testing would require the operator to leave the main control room for approximately 45 minutes. Determine the MINIMUM shift response to this condition.

A) A temporary relief of the URO by the on-shift Plant Reactor Operator (PRO).

B) A temporary relief of the URO by the fourth Reactor Operator (4th RO).

C) A complete turnover of the URO position to the Plant Reactor Operator (PRO).

D) No relief is required since licensed operators are exempt from random substance testing while holding a licensed position.

4) Unit 3 is in MODE 1 at 80% power.

- An applicable Tech Spec Surveillance with a 24 hour frequency was last performed satisfactorily at 0900 on 1/1/99.
- The LCO Required Actions direct that the equipment be restored to OPERABLE status in 4 hours, or be in MODE 3 in 12 hours AND MODE 4 in 36 hours.

If equipment problems prevent the surveillance from being performed, when is the unit required to be in MODE 4?

A) By 2100 on 1/3/99.

B) By 0100 on 1/4/99.

C) By 0300 on 1/4/99.

D) By 0700 on 1/4/99.

5) A Post Maintenance Test (PMT) requires the performance of a portion of a ST to stroke time a valve following maintenance to prove OPERABILITY. The CRS notes that the acceptance criteria for valve stroke time needs to be changed due to a recent Tech Spec revision.

Which of the following is the MINIMUM required to use the ST to complete this PMT.

A) A "Partial Procedure Use Change".

B) A "Permanent Revision Temporary Change".

C) A "Single Use Temporary Change".

D) A "Procedure Revision".

6) Which of the following combinations of tags may be applied to the same component at the same time in accordance with the Clearance and Tagging Manual?

A) A Special Condition Tag (SCT) and a tagged component bearing a green suspension label.

B) Two Special Condition Tags (SCTs).

C) A Danger Tag and Special Condition Tag (SCT).

D) An Information Tag and a Danger Tag.

7) You are the responsible SRO supervising the beginning of core reload as the refuel platform main hoist lowers the third fuel bundle around the "A" Wide Range Neutron Monitoring detector.

Which of the following conditions would require suspension of core alterations?

A) The Unit Reactor Operator reports that the white rod withdraw permissive light on panel C05 is NOT lit.

B) "A" Wide Range Neutron Monitor count rate doubles as the bundle is seated in the fuel support piece.

C) The Refuel Platform Operator reports that the "Rod Block Interlock #1" light on the refuel platform is NOT lit.

D) The Unit Reactor Operator reports that the inservice RHR shutdown cooling pump has just been removed from service.

8) During a HPCI system maintenance window, the HPCI System Manger wishes to perform a diagnostic activity on the HPCI High Steam Flow Isolation Logic. The activity will involve lifting leads, checking electrical continuity and potentially cleaning and tightening of electrical connections. The activity is expected to take 1 hour. HPCI is considered to be Tech Spec INOP due to being isolated with the Aux Oil pump in "Pull to Lock". A 50.59 review has determined there is no un-reviewed safety question.

Which of the following procedures is required to control this activity?

A) A-C-023 "Plant Evolution/Special Test (PEST) Program".

B) A-C-041 "Troubleshooting, Rework and Testing (TRT) Control Process".

C) A-C-025 "Fix it Now (FIN) Process".

D) MOD-C-7 "Temporary Plant Alteration (TPA)".

9) Given the following conditions:

- A scheduled Unit 2 surveillance is required to be performed on a system in a radiation area.
- All radiological precautions have been taken and a pre-evolution brief has been completed.

Using the As Low As Reasonably Achievable (ALARA) guidelines, which of the following is the PREFERRED method for completing this surveillance? (Consider only the personnel aspects of this surveillance.)

A) One individual installing shielding in a 90 mR/hr area for 30 minutes then performing the surveillance in a 9 mR/hr area for 60 minutes.

B) Two individuals performing the surveillance in a 90 mR/hr area for 35 minutes.

C) One individual performing the surveillance in a 90 mR/hr area for 60 minutes.

D) Two individuals installing shielding in a 90 mR/hr area for 15 minutes then performing the surveillance in a 9 mR/hr for 35 minutes.

10) During a declared emergency, it is necessary to raise the PECO Administrative Dose Control Levels to the NRC annual exposure limits.

Which of the following describes how this extension is authorized?

A) During a declared emergency, all Peach Bottom qualified Radiation Workers are automatically extended to the NRC TEDE limit.

B) The Radiation Protection Manager provides the Emergency Director with verbal case-by-case extension authorizations to the NRC limit.

C) The Control Room Supervisor provides immediate verbal extension authorization for Operations personnel.

D) The Emergency Director approves a "Dose Extension Form".

11) Which of the following is the REQUIRED immediate action if a Locked High Radiation Area door is found open with no control of area access?

A) Inform Security and establish Positive Access Control.

B) Inform the on-shift Health Physics Technician and lock the area after checking for unauthorized personnel.

C) Inform Security, lock the area and have Health Physics check for exposures in excess of those expected.

D) Inform the Health Physics Supervisor and establish Positive Access Control

12) To return the plant to a stable condition following a transient, Operations personnel need to enter a High Radiation Area that does not have an existing Radiation Permit (RWP).

Which of the following will meet the MINIMUM requirements for an Equipment Operator to enter the area.

A) Must be accompanied by an Advanced Rad Worker (ARW) qualified individual.

B) Must be accompanied by a Level II Radiation Protection Technician qualified individual.

C) Entry into the area is not permitted without the Radiation Protectection Manger (RPM) permission.

D) Entry into the area is not permitted until activation of the Emergency Plan.

13) Given the following conditions:

- A plant transient occurred on Unit 3 at 1415 resulting in an Unusual Event declaration at 1425.
- While completing the Unusual Event Notification Form an Alert was declared at 1435 for an unrelated event.

The State and Local Agencies shall be notified of the Alert no later than:

A) 1440.

B) 1450.

C) 1515.

D) 1535.

14) Unit 3 was operating in MODE 1 at 50% power when a plant transient required the crew to scram the unit. The following conditions exist:

- All rods are inserted
- Reactor pressure dropped to 940 psig and has stabilized at approximately 1000 psig
- Reactor level dropped to -20", then quickly recovered to its present value of +20" and going up
- HPCI auto started and is injecting into the reactor vessel
- A Main Stack High Radiation Alarm is present
- An Area Radiation Monitor is alarming (HPCI reading 12 mR/hr)

Select which of the following TRIP procedures should be entered and executed under these conditions.

A) Scram Condition (T-100)

B) Primary Containment Control (T-102)

C) Secondary Containment Control (T-103)

D) Radioactive Release (T-104)

15) Peach Bottom Units 2 and 3 are operating in MODE 1 at full power when the following timeline commences:

- At 1600, the Power System Director (PSD) notifies the Control Room that a hurricane is forecast to hit the station with sustained winds of 100 miles per hour.
- At 1630, with Plant Manager and PSD concurrence both Units are shutdown using GP-4, Manual Reactor Scram. Unit 2's shutdown was uneventful but Unit 3 has 6 control rods with unknown position after the scram.
- At 1700, the hurricane hits Peach Bottom and causes a complete loss of off-site power. All four Diesel Generators failed to start. The crew was successful at manually starting the E-3 Diesel Generator but have been unable to close either of its output breakers.

It is currently 1718, using the attached ERP-101, Classification of Emergencies Tables, classify these conditions to determine the appropriate current Emergency Action Level (EAL) to be declared.

A) Unusual Event

B) Alert

C) Site Area Emergency

D) General Emergency

TABLE 1
GENERAL CONDITIONS

<u>NOTE</u>:
This table is to be used as a guide for "big picture" emergency classification.
<u>IF</u> conditions listed are met
<u>AND</u> specific EALs of other tables <u>do not</u> address current emergency conditions,
<u>THEN</u> classify using this table.

UNUSUAL EVENT	1) Situation threatens normal level of plant safety. No releases of radioactive material off-site are expected.
ALERT	1) Situation does or could represent a substantial degradation in the level of plant safety 2) Conditions exist that warrant precautionary activation of Technical Support Center and placing Emergency Operations Facility and other key emergency personnel on standby 3) Release of radioactive material warrants off-site response or monitoring, but does not require protective actions.
SITE AREA EMERGENCY	1) The level of safety has or could be degraded to the point of losing a plant function needed to protect the public 2) Conditions exist that warrant: (a) Activation of EOF/ENC <u>AND</u> (b) Activation of off-site monitoring teams <u>OR</u> Protective measures recommendations to public near the site 3) A significant release of radioactive material has occurred or could take place onsite or near the site boundary.
GENERAL EMERGENCY	1) Substantial core damage <u>AND</u> loss of, or high potential for loss of primary Containment integrity. 2) Conditions exist that warrant all on-site and off-site emergency facilities being activated to aid in implementation of protective actions. 3) A significant release of radioactive material has occurred or could take place offsite in a short period of time. 4) Protective Actions Recommendations for off-site areas are made for PBAPS. * PAR evacuate a full 360 degrees for 2 miles evacuate affected and 2 adjacent sectors for 2-5 miles

TABLE 2
FUEL DAMAGE

UNUSUAL EVENT CM-5	1) Off-gas radiation rise of 500 mR/hr within 30 minutes	
	2) Off-gas radiation >2.5E+03 mR/hr	{RR-2(3)-17-152}
	3) Reactor coolant activity >4 uCi/gm dose equivalent I-131 per Tech. Spec. 3.4.6.	
ALERT	1) Containment radiation >4.0E+02 R/hr	{RI-8(9)103 A/C} {RI-8(9)103 B/D}
	2) Off-gas radiation >2.5E+04 mR/hr	{RR-2(3)-17-152}
	3) Reactor coolant activity >300 uCi/gm dose equivalent I-131 with a Rx scram due to main steam line high radiation.	
	4) Spent fuel damage resulting in refuel floor high radiation <u>OR</u> refuel floor ventilation exhaust high radiation	{RIS-2(3)-17-458(A,B,C,D)} {RR-2-(3)-17-456}
SITE AREA EMERGENCY	1) Containment radiation >4.0E+03 R/hr	{RI-8(9)103 A/C} {RI-8(9)103 B/D}
	2) Major spent fuel damage or uncovering of spent fuel confirmed by high fuel floor radiation levels <u>AND</u>	{U/2 ARM 3-7,3-8,3-9,3-10} {U/3 ARM7-9,7-10,7-11,7-12}
	(a) observation <u>OR</u>	
	(b) refuel floor high radiation <u>OR</u>	{RIS-2(3)-17-458(A,B,C,D)}
	(c) refuel floor ventilation exhaust high radiation	{RR-2(3)-17-456}
GENERAL EMERGENCY	1) Containment radiation >4.0E+04 R/hr with Containment pressure >10 psig	{RI-8(9)103 A/C} {RI-8(9)103 B/D} {PR-2(3)508}
	(for a known or probable failure of Primary Containment, see Table 4)	
	* PAR - evacuate a full 360 degrees for 2 miles - evacuate affected and 2 adjacent sectors for 2-5 miles	
	2) Containment radiation >3.0E+05 R/hr with Containment pressure >10 psig	{RI-8(9)103 A/C} {RI-8(9)103 B/D} {PR-2(3)508}
	(for a known or probable failure of Primary Containment, see Table 4)	
	* PAR - evacuate a full 360 degrees for 5 miles - evacuate affected and 2 adjacent sectors for 5-10 miles	

TABLE 3
REACTOR COOLANT SYSTEM (RCS)

UNUSUAL EVENT	1) RCS leakage exceeding Tech. Spec. LCO 3.4.4 limits. a. No pressure boundary leakage b. greater than 5 gpm unidentified leakage c. greater than 25 gpm total leakage over the previous 24 hour period d. greater than 2 gpm increase in unidentified leakage within the previous 24 hour period in MODE 1 2) Stuck open relief valve <u>OR</u> safety valve.	
ALERT	1) RCS leakage greater than 50 gpm (25,000 lbm/hr) 2) Scram condition with Reactor level below -160" <u>OR</u> unknown.	{LI-2(3)-02-3-091} {LI-2(3)-02-3-113} {PR/LR-2(3)-02-3-404A} * {LR-2(3)-02-3-110A} * {LR-2(3)-02-3-110B} * (Blue Pen Only)
SITE AREA EMERGENCY	1) Scram condition with Reactor level below -160" <u>OR</u> unknown <u>AND</u> Containment pressure >10 psig.	{LI-2(3)-02-3-091} {LI-2(3)-02-3-113} {PR/LR-2(3)-02-3-404A} * {LR-2(3)-02-3-110A} * {LR-2(3)-02-3-110B} * (Blue Pen Only) {PR-2-(3)508}
GENERAL EMERGENCY CM-3	1) Scram condition with Reactor level below -226" for greater than 3 minutes <u>AND</u> Containment pressure >20 psig. * PAR evacuate a full 360 degree for 2 miles evacuate affected and 2 adjacent sectors for 2-5 miles	{LI-2(3)-02-3-091} {LI-2(3)-02-3-113} {PR/LR-2(3)-02-3-404A} * {LR-2(3)-02-3-110A} * {LR-2(3)-02-3-110B} * (Blue Pen Only) {PR-2(3)-508}

TABLE 4
PRIMARY CONTAINMENT

UNUSUAL EVENT	1) Failure of a Primary Containment Penetration to isolate due to a valid isolation condition (both valves in a two valve penetration fail to close).
ALERT	1) Unexpected radiation levels rise by a factor of 1000. 2) Unexpected airborne activity of >1000 DAC hours excluding isotopes with half lives <2 hrs. 3) Torus room flood (6 inches) with a corresponding level drop in the Torus. {panel 2(3)24, alarm E-5} {panel 2(3)0C003, LI-2(3)919}
SITE AREA EMERGENCY	1) Primary Containment radiation >4.0E+2 R/hr {RI-8(9)103A/C} {RI-8(9)103B/D} **AND** Main Stack >6.9E+0 uCi/cc {RR-0-17-051} 2) Primary Containment radiation >4.0E+2 R/hr {RI-8(9)103A/C} {RI-8(9)103B/D} **AND** Vent Stack >1.0E-3 uCi/cc {RR-2(3)979} 3) Primary Containment radiation >4.0E+3 R/hr with a known or probable failure of Primary Containment Integrity. {RI-8(9)103A/C} {RI-8(9)103B/D}
GENERAL EMERGENCY	1) Primary Containment Radiation >4.0E+4 R/hr with a known or probable failure of Primary Containment Integrity. {RI-8(9)103A/C} {RI-8(9)103B/D} (for Primary Containment Intact, see Table 2) * PAR evacuate a full 360 degree for 2 miles evacuate affected and 2 adjacent sectors for 2-5 miles 2) Primary Containment radiation >3.0E+5 R/hr with a known or probable failure of Primary Containment Integrity. {RI-8(9)103A/C} {RI-8(9)103B/D} (for Primary Containment Intact, see Table 2) * PAR evacuate a full 360 degree for 5 miles evacuate affected and 2 adjacent sectors for 5-10 miles

TABLE 5
RADIOACTIVE RELEASE

NOTE:

CDE = COMMITTED DOSE EQUIVALENT
TEDE = TOTAL EFFECTIVE DOSE EQUIVALENT
TPARD = TOTAL PROTECTIVE ACTION RECOMMENDATION DOSE
BASIS EPA-400, "MANUAL OF PROTECTIVE ACTION GUIDES AND PROTECTIVE ACTIONS FOR NUCLEAR INCIDENTS.

UNUSUAL EVENT	1) Gaseous release exceeding ODCMS 3.8.C.1 as evidenced by a calculated offsite dose rate from the Main Stack, Vent Stack, Torus Hardened Vent, or unmonitored release that exceeds either 0.057 mRem/hr TPARD using a 60 minute average release data <u>OR</u> 0.170 mRem/hr child thyroid CDE using a 60 minute average release data. 2) Liquid release exceeding ODCMS 3.8.B.1 3) Iodine Release exceeding ODCMS 3.8.C.1.b
ALERT	1) Calculated offsite dose rate >0.57 mRem/hr TPARD using 15 min. avg. release data. 2) Calculated offsite dose rate >1.7 mRem/hr child thyroid CDE using 15 min. avg. release data.
SITE AREA EMERGENCY	1) Projected offsite total dose >100 mRem TPARD. 2) Projected offsite thyroid dose >500 mRem child thyroid CDE. 3) Projected offsite skin dose >5,000 mRem. 4) Actual offsite dose rate >25 mRem/hr TEDE. 5) Measured offsite air concentration >6.5N8 uCI/cc iodine.
GENERAL EMERGENCY	1) Projected offsite total dose >1000 mRem TPARD. 2) Projected offsite thyroid dose >5000 mRem child thyroid CDE. 3) Projected offsite skin dose >50,000 mRem. 4) Actual offsite dose rate >250 mRem/hr TEDE. 5) Measured offsite air concentration >6.5N7 uCi/cc iodine. * PAR evacuate 360 degrees for 5 miles evacuate affected and 2 adjacent sectors for 5-10 miles

TABLE 6
FIRE

UNUSUAL EVENT	1) Fire in protected area lasting 10 minutes or more after initial attempts to extinguish it.
ALERT	1) Fire which has lasted over 20 minutes after initial attempts to extinguish it and which <u>could</u> make any of the following safety systems INOPERABLE: - ADS - ECW - ESW - HPCI - HPSW - PCIS - RCIC - SBGTS - SLC - RHR - RPS - Core Spray - Control Rod Drive HCU's - Control Room Ventilation - 2 Emergency Diesel Generators - Loss of Emergency Switchgear - Primary Containment - Secondary Containment
SITE AREA EMERGENCY	1) Fire which removes those Safety Systems required to perform a single plant function (i.e., both HPCI & ADS when required by Tech. Specs., all of Low Pressure ECCS when required by Tech. Specs.).
GENERAL EMERGENCY	N/A

TABLE 7
SEVERE NATURAL PHENOMENA

UNUSUAL EVENT	1) Earthquake felt in plant or detected and confirmed on station seismic instrumentation per SO 67.7.A. 2) Conowingo Pond level <104 feet without prior notification by the Power System Director. {LI-2(3)278A,B,C} 3) Conowingo Pond level >111 feet with predicted flow in excess of 600,000 cfs. {LI-2(3)278A,B,C} 4) Hurricane forecasted to hit the station with sustained winds of 75 mph or greater, as notified by the Power System Director. 5) A tornado within site boundaries.
ALERT	1) "OPERATING BASIS EARTHQUAKE" exceeded per SE-5 and felt in the plant. 2) An uncontrollable loss of Conowingo Pond level as confirmed by the Power System Director. 3) Conowingo Pond level >112 feet. 4) Hurricane or tornado which strikes the power block with identifiable plant damage.
SITE AREA EMERGENCY	1) "MAXIMUM CREDIBLE EARTHQUAKE" detected on station seismic instrumentation (0.12g) per UFSAR Sec. 2.5.3, Sec.12.2 and Appendix C. 2) Conowingo Pond level <87 feet as confirmed by the Power System Director. 3) Conowingo Pond level >113 feet.
GENERAL EMERGENCY	N/A

TABLE 8
LOSS OF POWER

UNUSUAL EVENT	1) All offsite power to the emergency buses unavailable for >60 seconds. 2) No diesel generators available when required for >60 seconds.
ALERT	1) Loss of all offsite power <u>AND</u> <u>NO</u> diesel generators energize their associated buses. 2) Loss of DC power as evidenced by verifying <105 volts on all four 125 V distribution panels. {panel 2(3)09, alarms C-3 & C-4} {panel 2(3)20, alarms H-3 & H-4}
SITE AREA EMERGENCY	1) Loss of all offsite power for >15 minutes <u>AND</u> <u>NO</u> diesel generators energize their associated buses for >15 minutes. 2) Loss of DC power for longer than 15 minutes as evidenced by verifying <105 V on all four 125 V distribution panels. {panel 2(3)09, alarms C-3 & C-4} {panel 2(3)20, alarms H-3 & H-4}
GENERAL EMERGENCY	N/A

<u>NOTE</u>:

DIESEL GENERATORS SHUTDOWN DUE TO LACK OF COOLING WATER ARE CONSIDERED "UNAVAILABLE".

TABLE 9
LOSS OF ASSESSMENT OR COMMUNICATIONS

UNUSUAL EVENT	1) Loss of communications capability including (refer to Reportability Reference Manual) Loss of the ENS Network AND Loss of the OMNI Network AND Loss of the GTE System 2) Unplanned loss of most or all safety system annunciators OR indicators for >15 minutes requiring increased surveillance to safely operate the unit(s).
ALERT	1) Unplanned loss of most or all safety system annunciators OR indicators for 15 minutes requiring increased surveillance to safely operate the unit(s) AND EITHER a significant plant transient is in progress OR the plant monitoring system (PMS) is unavailable.
SITE AREA EMERGENCY	1) Loss of safety system annunciators AND indicators AND PMS AND a significant plant transient is in progress.
GENERAL EMERGENCY	N/A

NOTE:

SIGNIFICANT PLANT TRANSIENTS INCLUDE BUT ARE NOT LIMITED TO: SCRAM, RECIRC RUNBACK INVOLVING GREATER THAN 25% THERMAL POWER CHANGE, ECCS INJECTIONS, OR THERMAL POWER OSCILLATIONS OF 10% OR GREATER.

TABLE 10
HAZARDS TO STATION OPERATION

UNUSUAL EVENT	1) Aircraft crash on OR near site OR unusual aircraft activity over facility 2) Significant explosion on OR near site 3) Significant toxic gas OR flammable gas release on OR near site.
ALERT	1) Aircraft crash OR missile impact within the protected area. 2) Significant explosion within the protected area affecting plant operation. 3) Uncontrolled significant release of toxic OR flammable gas within the protected area.
SITE AREA EMERGENCY	HAZARDS WITH EITHER UNIT NOT IN MODE 4 1) Aircraft crash OR missile impact with major damage in any vital area. 2) Explosion causing severe damage to 2 OR more diesel generators OR to ECCS equipment such that the systems required to perform a single plant function become inoperable (i.e., both HPCI & ADS when required by Tech. Specs., all of low pressure ECCS when required by Tech. Specs.). 3) Uncontrolled release of toxic OR flammable gas detected in the Control Room (e.g. Chlorine, Cardox).
GENERAL EMERGENCY	N/A

TABLE 11
CONTROL ROOM EVACUATION

UNUSUAL EVENT	N/A
ALERT	1) Evacuation of Main Control Room is anticipated <u>OR</u> required <u>AND</u> control is established at Remote Shutdown Panels or Alternative Shutdown Panels.
SITE AREA EMERGENCY	1) Evacuation of Main Control Room <u>AND</u> control of Reactor Shutdown Systems <u>is not</u> established at Remote Shutdown Panels or Alternative Shutdown Panels in 15 minutes.
GENERAL EMERGENCY	N/A

TABLE 12
THREAT TO SECURITY

UNUSUAL EVENT	1) Credible sabotage or bomb threat 2) Credible intrusion and attack threat 3) Attempted intrusion and attack 4) Attempted sabotage discovered 5) Hostage situation or extortion threat.
ALERT	1) Actual attack and intrusion into a protected area 2) Suspected bomb or sabotage device discovered.
SITE AREA EMERGENCY	1) Imminent loss of physical control of the facility with imminent occupation of the Control Room or other vital areas.
GENERAL EMERGENCY	1) Actual loss of physical control of the facility with occupation of the Control Room or other vital areas. * PAR evacuate 360 degrees for 2 miles

NOTE:

"CREDIBLE THREAT" MEANS (1) PHYSICAL EVIDENCE SUPPORTING THE THREAT EXISTS, (2) INFORMATION INDEPENDENT FROM THE ACTUAL THREAT MESSAGE EXISTS, THAT SUPPORTS THE THREAT, OR (3) A SPECIFIC GROUP OR ORGANIZATION CLAIMS RESPONSIBILITY FOR THE THREAT.

TABLE 13
PLANT SYSTEMS/EQUIPMENT FAILURE

UNUSUAL EVENT	1) Inability to reach required shutdown mode within Tech. Spec. LCO required action completion time. 2) Turbine rotating component failure causing rapid plant shutdown.
ALERT	1) Cold shutdown unattainable. 2) Failure to initiate a scram when required via the reactor protection system AND via Rx mode switch AND via manual scram pushbuttons AND via alternate rod insertion (ARI). 3) Scram condition AND the Rx is NOT shutdown. 4) Turbine failure causing casing penetration.
SITE AREA EMERGENCY	1) Hot shutdown unattainable. 2) Scram condition, Rx NOT shutdown AND torus temperature above 110 degrees F.
GENERAL EMERGENCY	N/A

NOTE:

THE REACTOR IS CONSIDERED SHUTDOWN WHEN REACTOR POWER IS BELOW MID-RANGE ON IRM RANGE 7 or WRNM indicates below 1.00E0%.

16) ON-114 is designed to mitigate actual fires reported in selected areas at Peach Bottom. An actual fire reported in which of the following areas would require entry into this procedure.

A) The Low Level Radwaste Storage Facility.

B) The Inner Screen Structure.

C) The SU-25 Start-Up House.

D) The Water Treatment Plant.

17) Unit 2 is experiencing a Hydraulic Anticipated Transient Without Scram (ATWS). Currently the Reactor Operator is manually inserting control rods using T-220, also the "A" Standby Liquid Control (SBLC) pump is injecting boron into the reactor vessel. Reactor Engineering has been directed to complete a calculation to determine the reactor's shutdown condition.

Select from the following conditions the one that describes when the ATWS will be considered to be terminated by T-101, RPV Control.

A) 1 Control Rod is at position 08, 10 Control Rods are at position 02, all other rods are at position 00, 28% of the SBLC tank has been injected.

B) 1 Control Rod is at position 44, all other rods are at position 00, 2% of the SBLC tank has been injected.

C) 3 Control Rods are at position 06, all other Control Rods are at position 00, 45% of the SBLC tank has been injected.

D) 12 Control Rods positions are unknown, the SBLC tank has been fully injected into the vessel.

18) Which of the following conditions will result in Recirc flow controller output being limited to 30%?

A) Total feedwater flow greater than 85% and any condensate pump trip.

B) Individual feedpump flow less than 20% and Reactor level less than 17".

C) Total feedwater flow greater than 20% and Reactor level less than 17".

D) Reactor scram and Reactor level less than 17".

19) The following conditions exist on Unit 2 after a LOCA.

- Drywell pressure 7 psig, rising slowly
- Reactor pressure 400 psig, dropping slowly
- Reactor level -75"
- All low pressure ECCS pumps were manually secured
- Level is being maintained with condensate injection.
- "D" RHR pump was placed in Torus Sprays at 1000 gpm and Drywell sprays at 1000 gpm

If level were to drop to -200" what would be the response of the LPCI system with no additional operator actions?

A) A, B, C RHR pumps would start, "D" RHR would continue to run, LPCI outboard injection valve (MO-154) would auto open, spray valves would auto close.

B) A, B, C RHR pumps would start, "D" RHR would continue to run, LPCI outboard injection valve (MO-154) would auto open, spray valves would remain open.

C) A, B, C RHR pumps would NOT start, "D" RHR would continue to run, LPCI outboard injection valve (MO-154) would remain closed, spray valves would remain open.

D) A, B, C RHR pumps would NOT start, "D" RHR would continue to run, LPCI outboard injection valve (MO-154) would remain closed, spray valves would auto close.

20) Following a valid HPCI initiation due to high Drywell pressure on Unit 3, HPCI was secured using the "Short Term HPCI System Shutdown When an Initiation Condition IS Present" method of SO-23.2.2A-3, "HPCI System Shutdown". The PRO has been directed to initiate HPCI injection into the Reactor Vessel from this condition.

Under these conditions, HPCI Turbine speed during startup is controlled by:

A) The ramp generator initiated by the opening of HPCI steam supply valve, MO-3-23-014.

B) The slow opening of the HPCI Turbine Stop Valve, HO-3-23-4513.

C) The ramp generator initiated by opening of the HPCI Turbine Control Valve, HO-3-23-4512.

D) The ramp generator initiated by opening of the HPCI Turbine Stop Valve, HO-3-23-4513.

21) An ADS blowdown has occurred following a LOCA. The ADS valve control switches remain in "Auto". Pressure is 200 psig and lowering slowly. All Core Spray and RHR pumps were initially injecting. "D" Core Spray pump has tripped. All RHR pumps were secured when level recovered above -100". Level is being restored using A, B, and C Core Spray pumps.

An additional Core Spray pump needs to be shutdown to control level recovery.

Which of the following statements accurately describe the response of the ADS system to pump shutdown?

A) ADS blowdown will stop when the "A" Core Spray pump is shutdown.

B) ADS blowdown will stop when the "B" Core Spray pump is shutdown.

C) ADS blowdown will stop when the "C" Core Spray pump is shutdown.

D) An ADS seal in prevents inadvertent blowdown termination by pump shutdown.

22) An ATWS condition has occurred on Unit 3. Reactor level is 23 inches and pressure is 1000 psig with the turbine still running. The CRS has directed the URO to inject Standby Liquid Control (SBLC). The URO positions the SLBC switch to "PUMP 'A' RUN". Identify the expected SBLC System response.

A) Squib continuity light are lit, pump discharge pressure is 1450 psig.

B) Squib continuity lights are lit, pump discharge pressure is 1100 psig.

C) Squib continuity lights are out, pump discharge pressure is 1450 psig.

D) Squib continuity lights are out, pump discharge pressure is 1100 psig.

23) A Unit 2 start up is in progress with the plant in MODE 1 at 25% RTP. Main condenser pressure switches PS-5-11A and PS-5-11B were discovered to be out of calibration and INOP. Use the copy of Tech Specs provided to determine the actions required (if any) for this inoperability.

A) Restore RPS trip capability within 1 hour.

B) Place one trip system in trip within 6 hours.

C) Be in MODE 2 within 6 hours.

D) Required number of channels met, no action required.

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. OR A.2 Place associated trip system in trip.	12 hours 12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more automatic Functions with RPS trip capability not maintained. __OR__ Two or more manual Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 30% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTES-----------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE	FREQUENCY
SR 3.3.1.1.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2 ------------NOTE------------ [redacted] required to be performed until 12 [redacted] after THERMAL POWER ≥ 25% RTP. ------------------------ [redacted] the absolute difference between [redacted] power range monitor (APRM) [redacted] and the calculated power is ≤ 2% RTP while operating at ≥ 25% RTP.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.3	------------------NOTE------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	------------------NOTE------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.1.1.6	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.1.1.7	Adjust the channel to conform to a calibrated flow signal.	31 days
SR 3.3.1.1.8	Calibrate the local power range monitors.	1000 MWD/T average core exposure

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.10	-----NOTE----- Radiation detectors are excluded. --------------- Perform CHANNEL CALIBRATION.	92 days
SR 3.3.1.1.11	-----NOTES----- 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --------------- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.1.12	-----NOTES----- 1. Neutron detectors are excluded. 2. Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. APRM flow units and associated flow transmitters are excluded. --------------- Perform CHANNEL CALIBRATION.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.13	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.	24 months
SR 3.3.1.1.14	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.15	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.16	Calibrate each radiation detector.	24 months
SR 3.3.1.1.17	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.18	Verify the RPS RESPONSE TIME is within limits.	24 months
SR 3.3.1.1.19	Calibrate APRM flow units and associated flow transmitters.	24 months

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Wide Range Neutron Monitors					
a. Period-Short	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 13 seconds
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 13 seconds
b. Inop	2	3	G	SR 3.3.1.1.5 SR 3.3.1.1.17	NA
	5(a)	3	H	SR 3.3.1.1.6 SR 3.3.1.1.17	NA
2. Average Power Range Monitors					
a. Startup High Flux Scram	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 15.0% RTP
b. Flow Biased High Scram	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 0.66 W + 63.9% RTP(b)
c. Scram Clamp	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 118.0% RTP
[illegible]	1	2	F	SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.17	≥ 2.5% RTP
e. In[illegible]	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.17	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) 0.66 W + 63.9% - 0.66 ΔW RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. Reactor Pressure — High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 1085.0 psig
4. Reactor Vessel Water Level — Low (Level 3)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 1.0 inches
5. Main Steam Isolation Valve — Closure	1	8	F	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 10% closed
6. Drywell Pressure — High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 2.0 psig
7. Scram Discharge Volume Water Level — High	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 50.0 gallons
	5(a)	2	H	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 50.0 gallons
8. Turbine Stop Valve — Closure	≥ 30% RTP	4	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 10% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure — Low	≥ 30% RTP	2	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 500.0 psig
10. Turbine Condenser — Low Vacuum	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 23.0 inches Hg vacuum
11. [redacted]	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 15 X Full Power Background
12. [redacted]	1,2	1	G	SR 3.3.1.1.14 SR 3.3.1.1.17	NA
	5(a)	1	H	SR 3.3.1.1.14 SR 3.3.1.1.17	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
13. Manual Scram	1,2	1	G	SR 3.3.1.1.9 SR 3.3.1.1.17	NA
	5(a)	1	H	SR 3.3.1.1.9 SR 3.3.1.1.17	NA
14. RPS Channel Test Switch	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.17	NA
	5(a)	2	H	SR 3.3.1.1.4 SR 3.3.1.1.17	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

24) Unit 2 was operating at 100% power when the "A" Recirc pump tripped. The pump was isolated in accordance with the procedure.

Which of the following statements describes the relationship between INDICATED total core flow and ACTUAL total core flow?

Total core flow indicated on DPFR-2-3-095 dP/F will be:

A) Less than actual by an amount TWICE idle loop flow.

B) Less than actual by an amount EQUAL to idle loop flow.

C) Greater than actual by an amount TWICE idle loop flow.

D) Greater than actual by an amount EQUAL to idle loop flow.

25) The RCIC system was being restored to its normal alignment following maintenance, when a high steam flow isolation occurred due to stroking open the valves too quickly. A few minutes later, a feedwater transient results in a scram and the need for RCIC system operation. Current level is -50 inches and dropping slowly. The SRO has directed that RCIC be recovered and injection initiated into the vessel at 600 gpm.

After depressing the isolation reset pushbutton, which of the following actions will be necessary to inject with RCIC?

A) The RCIC turbine trip throttle valve will need to be reset from the control room.

B) The RCIC turbine trip throttle valve will need to be reset locally.

C) The MO-131, steam admission valve, must be stroked open manually.

D) RCIC will automatically align and inject when the isolation reset pushbutton is depressed.

26) Given the following conditions:

- Unit 2 has experienced a loss of all AC power (station blackout).
- The Reactor Core Isolation Cooling (RCIC) system automatically initiated.
- Reactor water level is now -52 inches and rising.
- The Control Room Supervisor directs the Unit Reactor Operator to isolate RCIC.

What will be the expected RCIC system response when the operator depresses the Manual Isolation Pushbutton?

A) A normal RCIC system isolation and turbine trip will occur.

B) A RCIC turbine trip and system isolation will occur except the Inboard Steam Isolation Valve (MO-15) will not close.

C) No RCIC isolation actions or turbine trip will occur.

D) A RCIC turbine trip and system isolation will occur except the Outboard Steam Isolation Valve (MO-16) will not close.

27) A small Main Steam Line leak has occurred in the Turbine Building on Unit 2. The following plant conditions exist:

- The Reactor is scrammed.
- The "A" Main Steam Line has failed to isolate.
- Reactor Pressure is 800 psig.
- The PRO shutdown all low pressure ECCS pumps immediately after they started on LO-LO-LO level since no injection or minimum flow path was available.
- HPCI is blocked.
- RCIC has been maintaining reactor level steady at -165" for 12 minutes.

Starting the "A" RHR pump will:

A) Result in an ADS blowdown after a 9 minute time delay.

B) Result in an immediate ADS blowdown.

C) Result in an ADS blowdown after a 105 second time delay.

D) NOT result in an ADS blowdown.

28) A small recirc leak has resulted in 8 psig drywell pressure and -170" reactor level on Unit 2. After the Drywell Cooling Fans tripped, the CRS directed them to be restored using T-223. The trip was bypassed and the fans were restored in fast speed. 15 minutes later, the STA reports that the amber bypass light over the Drywell Cooler Fan bypass switch (43-5-0165) has gone out.

Describe the effect this will have on Drywell Cooler Fan operation and why.

A) Fans will continue to run, light goes out when both trip signals clear.

B) Fans will continue to run, light goes out if either trip signal clears.

C) Fans will trip, light goes out when both trip signals clear.

D) Fans will trip, light goes out when either trip signal clears.

29) Unit 2 is operating in MODE 1 at full power. The 2B RPS MG set output breakers trip on underfrequency.

Under these conditions, which of the following PCIS isolations will occur and result in isolation valves repositioning?

A) A Group II Outboard half isolation.

B) A Group III Outboard half isolation.

C) An Outboard MSIV auto isolation.

D) A full RWCU isolation.

30) Unit 2 is operating in MODE 1 at full power when PISH-2-5-12A the drywell pressure input to RPS and PCIS fails high resulting in an "A" channel RPS half scram and associated annunciators.

Determine the expected Primary Containment Isolation System (PCIS) response to this condition.

The "GROUP II/III INBOARD ISOL RELAYS NOT RESET" annunciator will:

A) Alarm, but NO valves will reposition.

B) Alarm, and the inboard isolation valves will reposition.

C) NOT alarm, and NO valves will reposition.

D) NOT alarm, but the inboard isolation valves will reposition.

31) A T-112 BLOWDOWN is in progress with reactor pressure at 75 psig above Torus pressure. The URO notes that, although the ADS valve switches are in "OPEN", the SRV's are indicating closed. The green and white lights are lit for each of the ADS valves. All other SRV's have only green lights lit. The "SAFETY RELIEF VALVE OPEN" annunciator is NOT lit.

Given the above conditions, determine the current expected position of the ADS valves.

A) Fully open.

B) Partially open, but not far enough for proper indication.

C) Failed closed.

D) Fully closed, due to low steam pressure.

32) Given the following conditions:

- Unit 2 was operating at 100% power.
- Annunciator 220 F-5, "Inverter Trouble", was received indicating a loss of 20Y050.
- The reactor was later scrammed and the turbine tripped.

Which of the following is the reason why this failure requires reactor pressure control via the Safety Relief Valves?

The static inverter loss will:

A) Cause a full Group I Main Steam Isolation Valve closure.

B) Cause a "full open" signal to the Turbine Bypass Valves requiring the EHC Pumps to be tripped to prevent a rapid depressurization.

C) Result in a loss of Turbine Bypass Valve opening capability.

D) Result in a closure of the Inboard Main Steam Isolation Valves.

33) Unit 3 is performing a GP-2 Plant Startup. Power is being raised from 50% to 100%. The Plant Reactor Operator is monitoring Electrohydraulic Control (EHC) system performance.

Pressure averaging manifold pressure is initially:

A) Equal to reactor pressure with an increasing dP as turbine load is raised.

B) Equal to reactor pressure with a constant dP as turbine load is raised.

C) Less than reactor pressure with an increasing dP as turbine load is raised.

D) Less than reactor pressure with a lowering dP as turbine load is raised.

34) Unit 2 is operating at 100% power, with the Digital Feedwater Control System (DFCS) in three-element control with the "B" Narrow Range Level automatically selected. A fault in the level detector causes it to fail downscale. Which of the following will occur?

A) DFCS will sense a low level and increase RFPT speed, thereby causing Reactor Vessel level to increase.

B) The "B" Narrow Range Level Detector will be automatically de-selected and the HIGHEST remaining narrow range level signal will be automatically selected.

C) The "B" Narrow Range Level Detector will be automatically de-selected, and the LOWEST remaining narrow range level signal will be automatically selected.

D) A default valve of +23" will be automatically selected by the master level controller.

35) Following a reactor scram from a power condition, Reactor Feedwater Pump (RFP) speed automatically goes up to compensate for the shrink experienced as the voids in the reactor collapse.

To protect the pumps from overspeed under these conditions, RFPs are limited to 85% following a scram:

A) With all three condensate pump running.

B) With less than three condensate pumps running.

C) With individual feedwater flows greater than 20%.

D) With individual feedwater flows less than 20%.

36) Unit 2 and Unit 3 are both in MODE 1 at 100% power. During an attempt to start the Standby Gas Treatment (SGT) system, the "A" SGT fan (OAV020) failed to start and the "Standby Gas Treatment B Filter Inlet" 00476-1 failed to open.

Using the Tech Specs provided, determine the required actions for Unit 2 AND Unit 3.

A) Unit 2 - Restore SGT within 7 days.
Unit 3 - Restore SGT within 7 days.

B) Unit 2 - Enter LCO 3.0.3 immediately.
Unit 3 - No action required.

C) Unit 2 - Restore SGT within 7 days.
Unit 3 - No action required.

D) Unit 2 - Enter LCO 3.0.3 immediately.
Unit 3 - Restore SGT within 7 days.

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- C.1 Place OPERABLE SGT subsystem in operation. OR	 Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2.1	Suspend movement of irradiated fuel assemblies in secondary containment.	Immediately
		<u>AND</u>	
	C.2.2	Suspend CORE ALTERATIONS.	Immediately
		<u>AND</u>	
	C.2.3	Initiate action to suspend OPDRVs.	Immediately
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1	Enter LCO 3.0.3	Immediately
E. Two SGT subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	E.1	--------NOTE-------- LCO 3.0.3 is not applicable. -------------------- Suspend movement of irradiated fuel assemblies in secondary containment.	Immediately
	<u>AND</u>		
	E.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	E.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 15 minutes with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- C.1 Place OPERABLE SGT subsystem in operation. OR	Immediately (continued)

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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2.1	Suspend movement of irradiated fuel assemblies in secondary containment.	Immediately
		<u>AND</u>	
	C.2.2	Suspend CORE ALTERATIONS.	Immediately
		<u>AND</u>	
	C.2.3	Initiate action to suspend OPDRVs.	Immediately
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1	Enter LCO 3.0.3	Immediately
E. Two SGT subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	E.1	--------NOTE--------- LCO 3.0.3 is not applicable. -------------------- Suspend movement of irradiated fuel assemblies in secondary containment.	Immediately
	<u>AND</u>		
	E.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	E.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 15 minutes with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months

37) Both Units were operating in MODE 1 at full power when the "3A" RPS bus was manually transferred to its alternate feed. Determine the expected condition of the Standby Gas Treatment (SBGT) system as a result of this transient.

A) "B" SBGT fan has started, SBGT "A" filter inlet and outlet dampers have opened.

B) "C" SBGT has started, SBGT "B" filter inlet and outlet dampers have opened.

C) "B" SBGT fan has started, SBGT "B" filter inlet and outlet dampers have opened.

D) "C" SBGT fan has started, SBGT "A" filter inlet and outlet dampers have opened.

38) Peach Bottom 4KV is aligned as follows:

- The #2 Emergency Auxiliary Transformer (OAX04) is out of service.
- All eight 4KV busses are being supplied by the #3 Emergency Auxiliary Transformer (OBX04).
- The 2A RHR and 2A HPSW pumps are running in Torus Cooling.

Determine the expected plant response to an automatic trip of the E-312 breaker.

A) The E-1 Diesel Generator will start after .25 seconds.

B) The E-1 Diesel Generator will start after .5 seconds.

C) The 2A RHR and HPSW pumps will trip and restart on power restoration.

D) The E-124 Load Center will trip and lockout.

39) Unit 3 is in MODE 1 at full power with "B" RPS on its normal alternate feed. All other equipment is in its normal alignment. Both of the inservice off-site start up feeds trip simultaneously. All diesel generators start and close in on their buses as designed.

Determine the response of Unit 3 to this loss of AC power event. Unit 3 will:

A) Scram immediately due to the loss of power to the RPS system.

B) Scram immediately due to turbine stop and control valve closure.

C) NOT scram immediately due to "B" RPS being powered by its alternate source.

D) NOT scram immediately due to the RPS MG Sets maintaining power until the diesel generators load their buses.

40) Peach Bottom has experienced a complete loss of off-site power. The E-1 and E-2 Diesel Generators started and loaded their busses normally but the "A" ESW pump did not start. The E-3 and E-4 Diesels did not start. The E-1 and E-2 were then shutdown due to not having cooling water. The Power System Director (PSD) has been requested to configure Conowingo Station for Peach Bottom Station Blackout. The CRS has directed a backfeed in accordance with SE-11 Attachment D, "Backfeeding Safe Shutdown Loads with E-1 & E-2 Diesel Generators Available".

Given that both the 2SUE and 3SUE busses are available for backfeeding, the PRO should select:

A) The 2SUE bus because it is the normal power source for the bus that feeds the "B" ESW pump.

B) The 2SUE bus because this will allow use of the SBO line to power 4KV buses.

C) The 3SUE bus because it is the normal power source for the bus that feeds the "B" ESW pump.

D) The 3SUE bus because this will allow use of the SBO line to power 4KV buses.

ID Number: 56

41) Which of the following statements describe the power supply to the Backup Scram Solenoid valves and their expected condition upon receipt of a full scram.

A) Station batteries, energize on a Full Scram.

B) Station batteries, de-energize on a Full Scram.

C) Reactor Protection Bus, energize on a Full Scram.

D) Reactor Protection Bus, de-energize on a Full Scram.

42) Given the following conditions:

- Unit 2 is making preparations for a reactor startup from a refueling outage.
- Reactor Building ambient temperature is 74 degrees F.
- The Reactor Building Equipment Operator is charging the hydraulic control unit accumulators with nitrogen to a pressure of 590 psig.
- Several days later with the Unit at 100% power, Reactor Building temperatures have stabilized at 92 degrees F.

Which of the following describes the expected impact on the Control Rod Drive Hydraulic system operations for these conditions? (Refer to attached figure.)

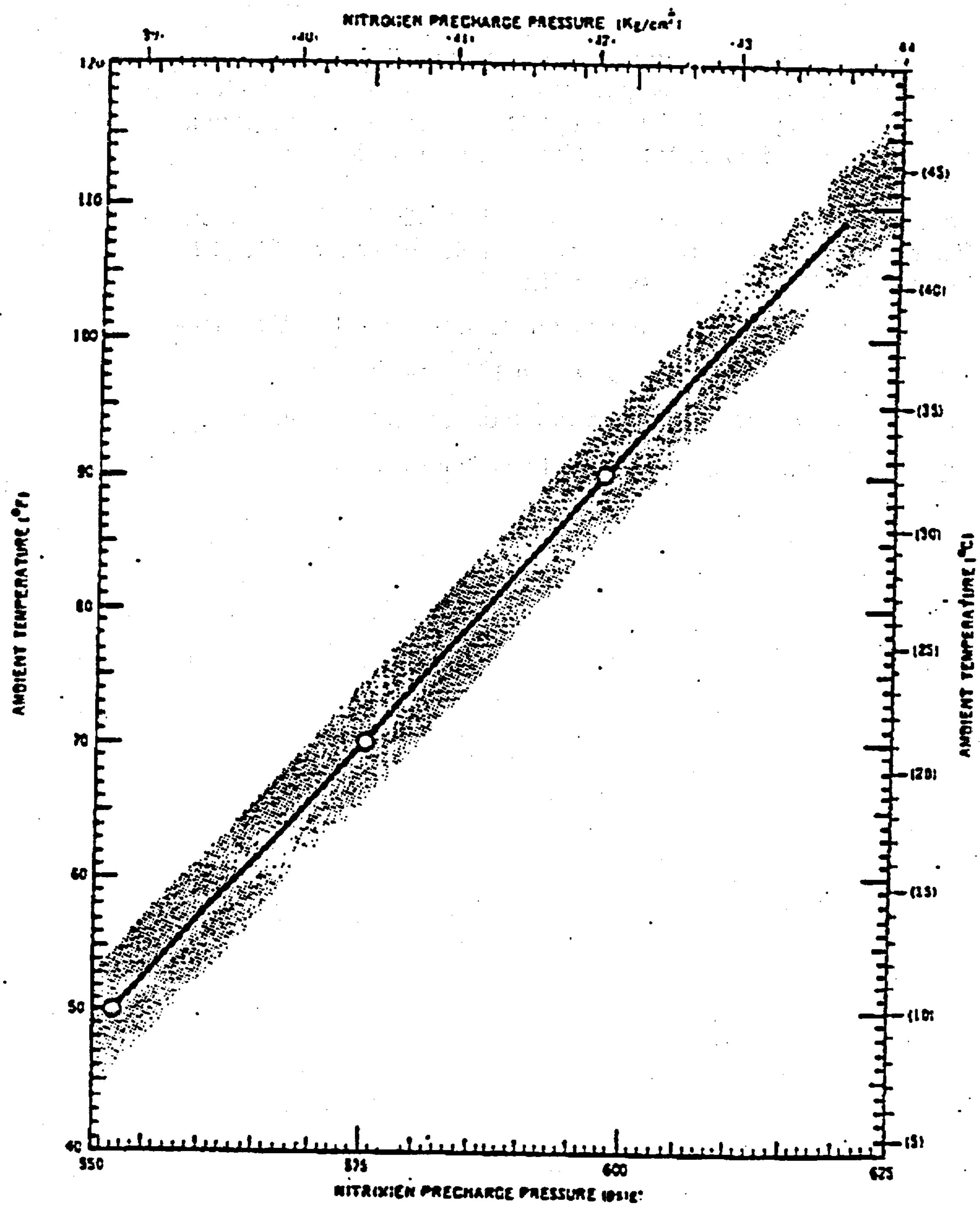
The individual control rod:

A) Normal insertion speeds will be slower and may result in control rod drift alarms.

B) Scram speeds will be slower and will result in reduced reactivity addition rates.

C) Normal insertion speeds will be faster and may result in "double notching".

D) Scram speeds will be faster and may result in mechanism damage.



SO 3.7.A-2, Figure 1
Accumulator Precharge Nitrogen Pressure
Verses Ambient Temperature

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43) Unit 2 is in MODE 2 with a heat up in progress. Vessel level is being maintained by the Control Rod Drive Hydraulic system and the Reactor Water Cleanup system in dump mode to the main condenser through the "RWCU Filter Bypass Valve", MO-74, which is full open.

What is the basis for the caution in procedure SO 12.1.A-2, "Reactor Water Cleanup System Start" which prohibits opening the "RWCU Outlet Valve" MO-68 in this system lineup?

A) Excessive heat load on the Non-Regenerative Heat Exchanger.

B) Group II isolation on greater than 125% system flow.

C) Excessive RWCU pump flows without control room indication.

D) Auto closure of CV-55 "Dump Flow Control Valve".

44) Given the following conditions:

- Unit 3 has had a complete loss of the E13 4160VAC Bus.
- This results in a loss of power to the "A" Residual Heat Removal (RHR) Pump and to the "A" Loop Inboard LPCI Injection Valve (MO-25A).
- A valid LOCA signal occurs.

What must occur to result in a final, design RHR injection flowrate for these conditions of 30,000 gpm.

A) The RHR Loop Cross-Tie Valve (MO-20) must be unlocked and opened by an operator.

B) An operator must manually transfer the Inboard LPCI Injection Valve (MO-25A) to the alternate power supply.

C) The Outboard LPCI Injection Valve (MO-154A) must automatically open to inject through the normally open MO-25A.

D) The Inboard LPCI Injection Valve (MO-25A) must automatically transfer to the alternate power supply.

45) Unit 2 is in MODE 4 with "A" RHR pump in Shutdown Cooling (SDC) returning to the vessel through the MO-25A, "Inboard Disch". Loss of inventory causes level to drop to -20". SDC isolates and the "A" RHR pump trips.

Which of the following statements describes the response of LPCI should level continue to drop to
< -160" with no additional operator actions.

A) "A" RHR restarts, "B", "C", and "D" start and inject.

B) "A" RHR restarts, "B", "C", and "D" start and run on min. flow.

C) RHR "B", "C", and "D" start and inject.

D) RHR "B", "C", and "D" start and run on min flow.

46) Following a reactor scram and scram reset, the Unit Reactor Operator notes that the full core display for rod 02-23 is blank with no position indicated and no green back light. All other rods indicate 00 with a green back light.

If the blank display is due to a rod 02-23 Position Indicating Probe (PIP) problem, the operational impact will be the inability to select and move:

A) Other rods in REFUEL due to a lack of "REFUEL MODE SELECT PERMISSIVE".

B) Other rods due to "RPIS INOPERATIVE".

C) Rod 02-23 due to lack of position indication and backlight.

D) Rod 02-23 due to "ROD SELECT BLOCK TIMER MALFUNCTION".

47) At what point during a Unit 3 reactor startup can the Unit Reactor Operator expect a significant increase in Wide Range Neutron Monitoring (WRNM) nuclear instrumentation response during control rod withdrawals?

A) Control rod withdrawals made after steam is being drawn from the reactor.

B) Control rod withdrawals made as reactor power passes 1.00E0% on WRNMs.

C) Withdrawal of a center control rod during a fast recovery startup.

D) Initial rod withdrawals from 50% rod density in the startup.

48) Given the following conditions:

- Unit 2 is in Mode 5
- The Mode Selector Switch is in "Refuel"
- The Refueling Platform is over the spent fuel pool
- A fuel bundle has been loaded on the Main Hoist and raised out of the fuel pool storage rack

Which of the following actions would result in a rod block?

A) The Refueling Platform operator raises the Main Hoist to the "full up" position.

B) The Unit Reactor Operator places the Mode Selector Switch in "Startup/Hot Standby".

C) The Refueling Platform operator moves the platform over the reactor vessel.

D) The Unit Reactor Operator selects, but does NOT withdraw, a single control rod.

49) Unit 3 is in MODE 1 at full power with the following conditions:

- Main Generator Load is 1050 MWe.
- Power factor is .95 lagging.
- Generator hydrogen pressure is 60 psig.

The Power System Director contacts you and directs you to raise reactive loading to 400 MVARs. What is the maximum real load permitted by the attached generator capability curve for the new value of reactive loading?

A) 1040 megawatts

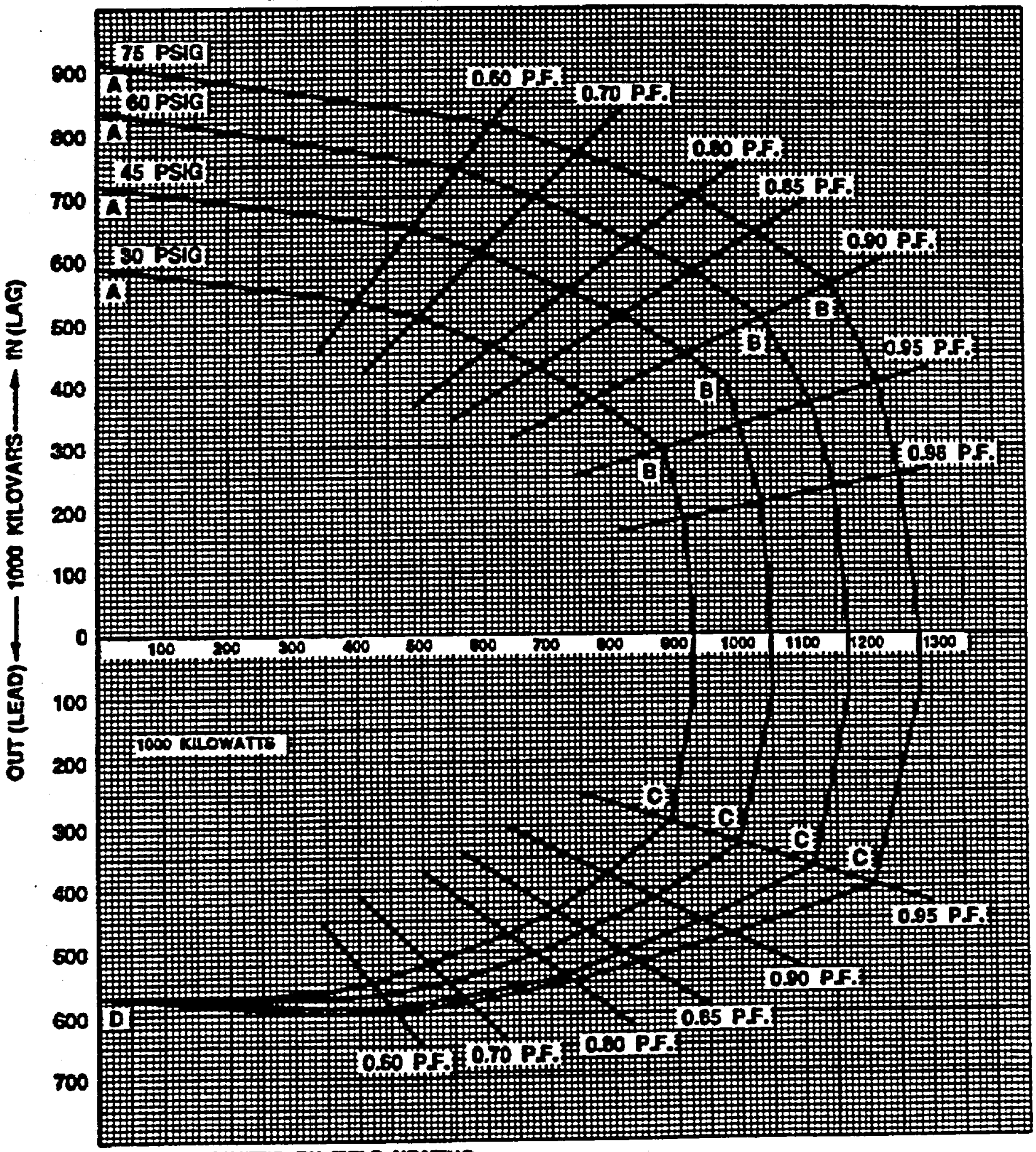
B) 1090 megawatts

C) 1190 megawatts

D) 1220 megawatts

FIGURE 1

ATB 4 POLE 1,280,000 KVA 1800 RPM 22,000 VOLTS
0.90 P.F. 0.60 SCR 75 PSIG HYDROGEN PRESSURE 500 VOLTS EXCITATION



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

50) Unit 2 is operating at 100% power when a leak develops in the TBCCW system. Shortly thereafter TBCCW Head Tank level drops out of sight low. The operating TBCCW pump starts cavitating and discharge pressure drops to 0 psig.

Assuming no operator actions are taken, which of the following statements describe the operational impact of this event?

A) "ISO-PHASE BUS TROUBLE" alarm is received immediately, "ISO-PHASE BUS LOSS OF COOLING" is received 10 minutes later, and an automatic turbine runback is initiated.

B) "ISO-PHASE BUS TROUBLE" and ISO-PHASE BUS LOSS OF COOLING" alarms are received immediately, and an automatic turbine runback is initiated.

C) "ISO-PHASE BUS TROUBLE" alarm is received immediately, followed by "ISO-PHASE BUS LOSS OF COOLING" alarm 10 minutes later. No automatic turbine runback will occur in this condition.

D) "ISO-PHASE BUS TROUBLE" and "ISO-PHASE BUS LOSS OF COOLING" alarms are received immediately. No automatic turbine runback will occur in this condition.

51) Unit 2 is operating at 95% power with all condensate pumps and all feedpumps running. The "A" CONDENSATE PUMP trips on motor overload. The RO verified that vessel level was maintained in the normal band.

Which of the following statements describe the plant response to this trip?

A) A Reactor Recirculation pump runback to 30% occurred due to the "A" condensate pump trip.

B) A Reactor Recirculation pump runback to 45% occurred due to the "A" condensate pump trip.

C) A Reactor Recirculation runback did NOT occur since vessel level was maintained in the normal band.

D) A Reactor Recirculation runback did NOT occur since total feed flow is > 85%.

52) Both Units were operating at full power when the following alarm and indications were received:

- "CONTROL ROOM RAD MONITOR DIV. II INITIATED" (003 A-3)
- MCR Radiation Monitors RI-0760B and RI-0760D were reading approximately 14,000 cpm with their red high lights lit.

Thirty seconds later, the following alarms and indication were received:

- "CONTROL ROOM VENT SUPPLY FAN HI-LO" (003 A-1)
- "CONTROL ROOM VENT SUPPLY LO FLOW CREV START" (003 A-5)
- FR-0765 is reading 200 scfm and dropping.

The Control Room Emergency Ventilation System:

A) Has NOT realigned since the complete initiation logic has not been satisfied.

B) Has NOT realigned as indicated by the low flow condition.

C) Has realigned due to a low flow condition.

D) Has realigned due to a high radiation condition.

53) Unit 2 has experienced a total loss of instrument air with the Instrument Air headers reading 0 psig.

Which of the following statements describe the pneumatic sources available to operate ALL of the Safety Relief Valves (SRVs).

A) Seismic Grade Instrument Gas (via T-261).

B) Instrument Nitrogen (via GP-8E).

C) Backup N2 bottles (via SV 8130 A &B).

D) Relief Valve accumulators.

54) The Traversing In-core Probe (TIP) system is in use with a probe in the core when the reactor scrams on low level following a loss of feedwater. HPCI automatically starts and recovers level, containment parameters are normal.

Which of the following statements describe the expected response of the TIP system to this transient.?

A) TIP automatically retracts from the core, TIP Ball valves close, TIP Nitrogen Purge valves close.

B) TIP automatically retracts from the core, TIP Ball valves close, TIP Nitrogen purge valves remain open.

C) TIP automatically retracts from the core, TIP Ball valves and TIP Nitrogen purge valves remain open.

D) NO TIP system response to a reactor low level condition.

55) The following conditions exist on Unit 3:

- A leak on the "3A" RWCU pump discharge has resulted in Reactor Building Ventilation Stack Radiation Levels rising.
- The Reactor Building Equipment Cell Exhaust had just been aligned to Standby Gas Treatment (SBGT) in accordance with SO 9 when a loss of instrument air occurred.
- The "3A" and "3B" instrument air headers and the Unit 3 service air header are fully depressurized.
- ON-119 has been entered.

Under these conditions, Standby Gas Treatment:

A) Will remain in service. Reactor Building Ventilation will isolate.

B) Will remain in service. Reactor Building Ventilation will NOT isolate.

C) Will NOT remain in service. Reactor Building Ventilation will isolate.

D) Will NOT remain in service. Reactor Building Ventilation will NOT isolate.

56) Unit 2 is operating at 100% power with all 10 Condensate Filter Demineralizers are in service in the AUTOMATIC mode. Condensate filter demin system dP momentarily spikes high and returns to normal. The Reactor Operator acknowledges receipt of the "CONDENSATE FILTER-DEMIN TROUBLE" alarm (20C207L A-2) and notes that the Condensate Filter Demineralizer Bypass Valve (MO-2114) is full open.

After the dP spike returns to normal, Feedpump suction pressure:

A) Will drop, due to Condensate Demin "E" Valve closure.

B) Will drop, due to Condensate Demin "E" Valve opening.

C) Will NOT drop, due to Condensate Demin "E" Valve closure.

D) Will NOT drop, due to Condensate Demin "E" Valve opening.

57) A refueling outage is in progress on Unit 2, with the reactor cavity flooded and the Fuel Pool gates removed. CRDH is in service and RWCU is rejecting inventory to maintain fuel pool level. A trip of the in-service CRD pump occurs. With no operator action, which of the following will occur as a result of the CRD pump trip?

Fuel pool cooling pumps (FCP) will trip on:

A) Low skimmer surge tank level to prevent Fuel Pool pump down.

B) Low skimmer surge tank level to provide FPC pump protection.

C) Low Fuel Pool level to prevent Fuel Pool pump down.

D) Low booster pump suction pressure to provide FPC pump protection.

58) Unit 3 was operating at 50% power when it experienced a loss of vacuum transient. Currently power is 20% and condenser vacuum is 24.5" and steady. A circ water problem has been discovered to be the cause of the vacuum loss. Maintenance estimates that it will be 4 hours until the circ water problem will be corrected. Under these conditions, the next operator action is to:

A) Continue to reduce power.

B) Hold power constant.

C) Trip the main turbine.

D) Scram and enter T-100.

59) Unit 2 is experiencing a low condenser vacuum transient and has entered OT-106, Condenser Low Vacuum. Vacuum is currently 24.5" Hg and dropping. The "C CONDENSER LO VAC" annunciator (203 D-2) is lit.

During a brief the CRS states that a full reactor scram will not be received until the "A" or "B" low vacuum alarms come in.

The CRS statement is:

A) Correct, since the "C" condenser provides only a "A" Channel RPS input.

B) Correct, since the scram setpoint cannot be achieved without all three condenser losing vacuum.

C) Incorrect, since the "C" condenser inputs into both RPS channels.

D) Incorrect, since low vacuum in any condenser (A, B, C) can result in a full scram.

60) Unit 2 has experienced a high drywell pressure transient. The reactor has been scrammed and the URO is controlling level manually. Due to overfeeding with HPCI, reactor level has exceeded the band of +5" to +35", HPCI was then secured.

Current conditions are:

- Reactor Pressure 1000 psig.
- Narrow range level indicators are upscale.
- Wide Range Level indicators reads +45" to +50" and steady.
- LI-2-2-3-86 is reading +75" and steady (figure 1 is attached).

Actual level should be verified using:

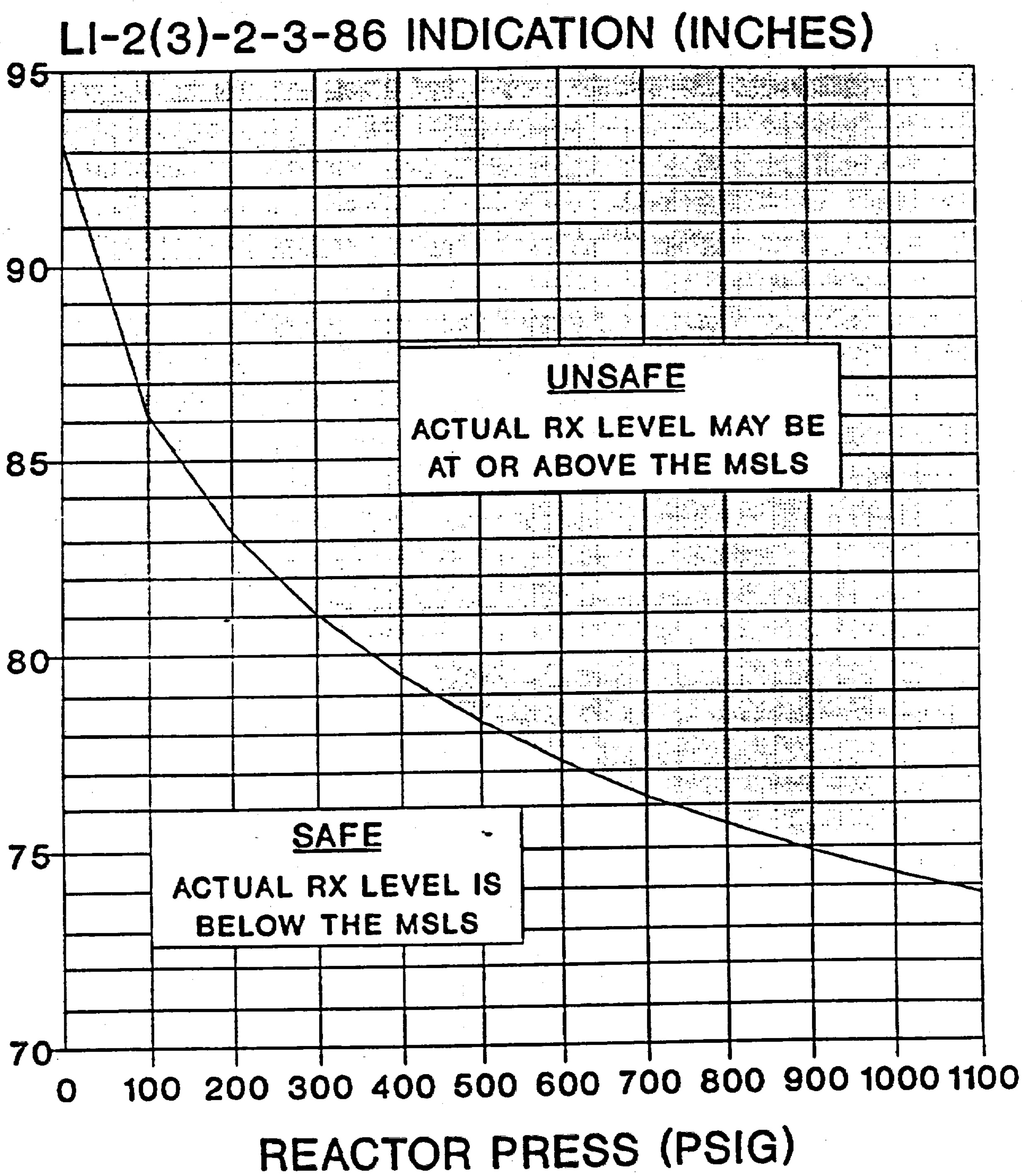
A) LI-2-2-3-86 and is currently ABOVE the Main Steam Lines.

B) LI-2-2-3-86 and is currently BELOW the Main Steam Lines.

C) Wide Range indication and is currently ABOVE the Main Steam Lines.

D) Wide Range indication and is currently BELOW the Main Steam Lines.

FIGURE 1


LI-2(3)-2-3-86 INDICATION (INCHES)
95
90
85
80
75
70
UNSAFE
ACTUAL RX LEVEL MAY BE AT OR ABOVE THE MSLS
SAFE
ACTUAL RX LEVEL IS BELOW THE MSLS
0
100
200
300
400
500
600
700
800
900
1000
1100
REACTOR PRESS (PSIG)

61) ON-118, loss of Turbine Building Closed Cooling Water (TBCCW), directs that if TBCCW cannot be restored, Main Turbine Generator load should be reduced to less than 18,000 amps in accordance with GP-9-2.

The basis for this ON-118 step is to:

A) Permit heat generated by the isophase bus bars to be absorbed by the environment.

B) Reduce the heat load so that RBCCW is not overloaded when it backs up TBCCW.

C) Permit condensate pumps to be alternated to prevent condensate pump overheating.

D) Reduce the heat load on TBCCW so that the station air compressors will NOT trip on high temperature.

62) ON-113, Loss of Reactor Building Closed Cooling Water (RBCCW), directs that the RCCW Head Tank Level be verified. When sent to verify RBCCW Head Tank Level, an Equipment Operator must:

A) Go to Turbine Building 165' elevation and check the level in the head tank sightglass.

B) Go to Turbine Building 165 elevation and check make-up valve position.

C) Go to the Refuel Floor and check the level in the head tank sightglass level.

D) Go to the Refuel Floor and check make-up valve position.

63) ON-119. Loss of Instrument Air, directs that the reactor be scrammed if any rod begins to drift in due to lowering scram pilot air header pressure.

What is the bases for this direction?

A) To ensure that the scram discharge volume is fully isolated during the scram.

B) To ensure that various scram valve opening pressures do not result in a random rod pattern.

C) To ensure that the individual control rod scram inlet valves do not open before the scram outlet valves.

D) To ensure that sufficient volume exists in the scram discharge volume to complete a full scram.

64) A loose fitting has resulted in the loss of instrument air to the in-service Control Rod Drive (CRD) Flow Control Valve (AO-19).

Determine which of the following conditions could result from this instrument air loss.

A) Control Rod Drive accumulator alarms due to low pressure.

B) Control Rod Drive alarms due to high temperatures.

C) Control Rods begin to drift due to excessive flow.

D) High rod speeds during control rod withdrawal.

65) Unit 2 has experienced a loss of shutdown cooling, ON-125, Loss of Shutdown Cooling, directs you to determine the expected decay heat load using Operator Aid 95-04 located on the back of Panel 20C005A.

The information necessary to determine expected heat load using this Operator Aid is:

A) Current heat up rate.

B) Current WRNM indicated power.

C) Power history before shutdown.

D) Elapsed time since shutdown.

66) A Unit 3 reactor startup was in progress with reactor pressure at 500 psig and reactor power at 1% when the running "A" Control Rod Drive (CRD) pump tripped. A start of the "B" CRD pump is in progress. CRD charging header pressure was 920 psig and dropping when 3 accumulator trouble alarms were received on withdrawn control rods.

Under these conditions, ON-107, Loss of CRD Regulation Function, directs that a full reactor scram be inserted. The basis for this direction is that:

A) At this reactor pressure, operable HCU accumulators are required to ensure proper scram force.

B) At this reactor pressure the CRDM ball check valves will NOT reposition to permit reactor pressure to insert the control rods.

C) This condition may result in unanalyzed rod patterns due to rods inserting randomly on low accumulator pressure.

D) This condition exceeds the Tech Spec Limit for the number of withdrawn control rods that can be declared "slow".

67) Unit 2 has experienced a drywell steam leak with an ATWS. Current conditions are as follows:

- Reactor pressure being maintained 950-1050 psig.
- TI-2501 point 126 is not available.
- TI-2501 point 127 indicates 520 degrees F.
- Narrow range RPV level indicates +5 inches.
- Wide range RPV level indicates -115 inches.
- Fuel Zone RPV level range indicates -125 inches.
- Refuel range RPV level (Shutdown Range Instrument LI-2-2-3-86) indicates -21 inches.

Evaluate the above conditions and then use Table DW/T-1 from T-102 (attached), "Primary Containment Control" to determine which RPV level indication ranges may be used.

A) Narrow Range and Refuel Range

B) Narrow Range and Wide Range

C) Wide Range and Fuel Zone Range

D) Fuel Zone Range and Refuel Range

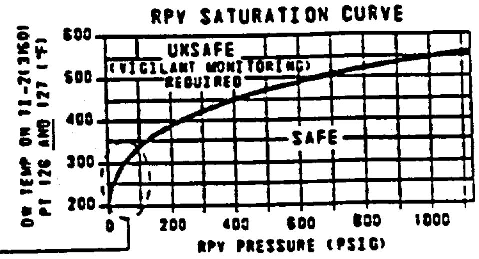
TABLE DW/T-1

RPV LEVEL INSTRUMENT STATUS

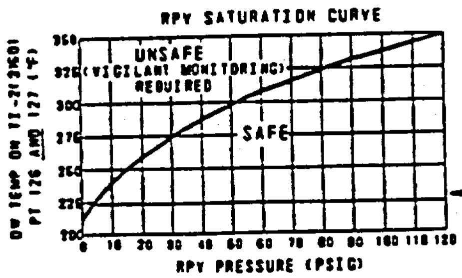
AN RPV LEVEL INSTRUMENT MAY BE USED TO DETERMINE RPV LEVEL ONLY WHEN THE FOLLOWING CONDITIONS ARE SATISFIED:

NOTES:
1. IF BOTH POINTS 126 AND 127 ARE AVAILABLE, THEN BOTH POINTS MUST PLOT "SAFE" TO CONSIDER A LEVEL INSTRUMENT AVAILABLE
2. IF EITHER POINT 126 OR 127 IS NOT AVAILABLE, THEN THE REMAINING POINT MUST PLOT "SAFE" TO CONSIDER A LEVEL INSTRUMENT AVAILABLE

ALL RPV LEVEL INSTRUMENTS



SEE DETAIL



IF DW TEMP AND RPV PRESS ARE ON THE UNSAFE SIDE OF THE RPV SATURATION CURVE
AND AN INSTRUMENT EXHIBITS AN UNEXPLAINED TREND OR OSCILLATION,
THEN THAT INSTRUMENT IS UNAVAILABLE

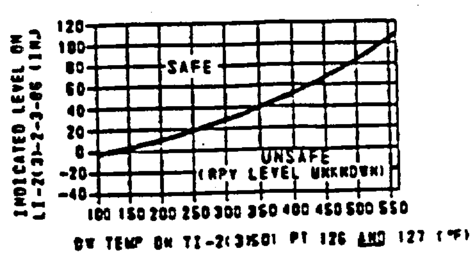
WIDE AND NARROW RANGE INSTS ONLY

FOR EACH OF THE INSTRUMENTS IN THE TABLE, THE INSTRUMENT READS ABOVE THE MIN INDICATED LEVEL OR THE TEMP NEAR THE DW REFERENCE LEG VERTICAL RUNS (TI-2(3)60I PT 126 AND 127) ARE BELOW THE MAX RUN TEMP.

INSTRUMENT	MIN INDICATED LEVEL IS ABOVE	OR	MAX RUN TEMP IS BELOW
NARROW RANGE	10 IN.	OR	450°F
WIDE RANGE	-120 IN.	OR	500°F

SHUTDOWN RANGE INST LI-2(3)-2-3-86 ONLY

LI-2(3)-2-3-86 READS ON THE SAFE SIDE OF THE CURVE



68) T-102, Primary Containment Control, provides direction to maintain Torus level in the band of 14.5 ft. to 14.9 ft. In accordance with the TRIP Bases what is the first concern during a rising torus level transient?

A) Submerging the Reactor Building to Torus Vacuum Breaker Line.

B) Excessive stress on SRV tail pipes.

C) Submergence of the Torus Spray Header.

D) Excessive stress on ECCS suction piping.

69) T-102 step PC/P-6 directs use of Torus Sprays before Torus pressure reaches 9 psig if Torus level is below 21 ft.

What are the bases for the 9 psig and 21 feet limitations?

A) Threshold for downcomer chugging and Torus Spray header becomes submerged.

B) Threshold for downcomer chugging and Torus to drywell vacuum breakers become submerged.

C) Threshold for evaporative cooling and Torus to drywell vacuum breakers become submerged.

D) Threshold for evaporative cooling and Torus Spray header becomes submerged.

70) Plant conditions on Unit 3 are as follows:

- A steam leak exists in the Unit 3 Reactor Building.
- The Reactor has been shutdown and depressurized to a steady value of 30 psig.
- TR-3-13-139 point 22 indicates 325 degrees F.
- Wide Range RPV level indicates -150 inches.
- Fuel Zone RPV level indicates -172 inches.

Evaluate the above conditions and then use Table SC/T-4 from T-103, "Secondary Containment Control" (attached) to determine which RPV level range may be used.

A) Wide Range may be used, Vigilant Monitoring required.

B) Wide Range may be used, Vigilant Monitoring NOT required.

C) Fuel Zone may be used, Vigilant Monitoring required.

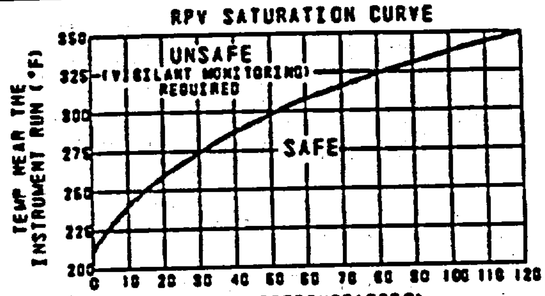
D) Fuel Zone may be used, Vigilant Monitoring NOT required.

TABLE SC/T-4
RPV LEVEL INSTRUMENT STATUS

AN RPV LEVEL INSTRUMENT MAY BE USED TO DETERMINE RPV LEVEL ONLY WHEN THE FOLLOWING CONDITIONS ARE SATISFIED:

ALL RPV LEVEL INSTRUMENTS

RPV SATURATION CURVE



IF THE TEMP NEAR THE RX BLDG INSTRUMENT RUNS ARE IN THE UNSAFE REGION OF THE RPV SATURATION CURVE AS DETERMINED BY THE LOCAL INSPECTION OR TR-2(3)-13-139 PT 22

AND AN INSTRUMENT EXHIBITS AN UNEXPLAINED TREND OR OSCILLATION,

THEN THAT INSTRUMENT IS UNAVAILABLE

WIDE RANGE AND FUEL ZONE INSTS ONLY

FOR EACH OF THE INSTRUMENTS IN THE TABLE, THE INSTRUMENT READS ABOVE THE MIN INDICATION LEVEL OR THE TEMP NEAR THE RX BLDG REFERENCE LEG VERTICAL RUNS ARE BELOW THE MAX RUN TEMP AS DETERMINED BY LOCAL INSPECTION OR TR-2(3)-13-139 PT 22.

INSTRUMENT	MIN INDICATED LEVEL IS ABOVE	OR	MAX RUN TEMP IS BELOW
WIDE RANGE	-120 IN.	OR	140°F
FUEL ZONE	-305 IN.	OR	315°F

71) A Designated Alternate (DA) is moving an old jet pump in the Unit 2 fuel pool when it falls off the auxiliary hoist. It is reported to the Control Room that a jet pump fell on an irradiated fuel bundle and damaged some fuel pins.

The Control Room also receives the following alarms and indications

- Refueling Floor Vent Exhaust Hi Radiation (218 A-1)
- Reac. Bldg. Zone Vent Exhaust Hi Radiation (218 B-1)
- Reac. Bldg. Or Refueling Floor Vent Exh. Hi Rad Trip (218 D-4)
- Refueling Floor Radiation Trip Units A and D High lights are lit.

Evaluate these conditions and determine the expected ventilation lineup.

A) Reactor Building Ventilation trips.
Refuel Floor Ventilation trips.
SBGT initiates and aligns to the entire Reactor Building/Refuel Floor.

B) Reactor Building Ventilation continues to run.
Refuel Floor Ventilation trips.
SBGT initiates and aligns to the Refuel Floor.

C) Reactor Building Ventilation continues to run.
Refuel Floor Ventilation continues to run.
SBGT initiates and aligns to the Refuel Floor.

D) Reactor Building Ventilation continues to run.
Refuel Floor Ventilation continues to run.
SBGT remains in standby.

72) Unit 2 is in T-103. "Secondary Containment Control", due to high water level condition in Secondary Containment. The Reactor has been conservatively scrammed and the Group II/III isolations (from the level shrink) are complete.

The CRS is currently attempting to determine whether a Primary System is discharging into the Reactor Building. Given the above conditions, evaluate the following and determine which constitutes a primary system discharging into the Reactor Building.

A) Leakage from a pipe flange on the discharge of the Reactor Water Cleanup Non-regenerative Heat Exchanger.

B) Steam leakage from a rupture on the piping of the #2 Main Steam stop valve inlet.

C) Leakage from a weld crack on the "A" RHR suction piping penetration to the Torus.

D) Steam leakage from the Standby Liquid Control Injection line just outboard of the drywell penetration.

73) Unit 3 was operating in MODE 1 at 75% power when a fire was reported in the Reactor Building 135' elevation. The Crew has entered ON-114, the procedure for an "actual fire", and the CRS has directed that the Equipment Operator isolate the RPV Condensing Chamber Backfill System.

The basis for isolation of this system under these conditions is to prevent inaccurate level indication and unreliable automatic initiations due to:

A) Lowering Instrumentation Variable Leg density.

B) Raising Instrumentation Variable Leg density.

C) Lowering Instrumentation Reference Leg density.

D) Rising Instrumentation Reference Leg density.

74) ON-114, for an Actual Reported Fire, has a note to inform the Operator that a loss of power to the Motor Driven Fire Pump for more than 8 seconds will defeat that pumps automatic start capability.

The basis for this feature is to prevent:

A) A simultaneous start with the Diesel Driven Fire Pump and resultant water hammer.

B) A spurious start due to loss of power to the fire header pressure instrumentation.

C) The pump from automatically starting with reduced bus voltage.

D) Overloading the diesel generators on a loss of off-site power.

75) Given the following conditions:

- A loss of off-site power has occurred.
- The E-1 and E-4 Diesel Generators (DG) are running.
- The E-43 4KV bus has an overcurrent lockout.
- No DG cooling water is available.
- Drywell pressure is 3.6 psig and slowly rising.

Why are jumpers, installed in the Control Room, the PREFERRED method for shutting down the two diesel generators?

A) This bypasses the 10 minute timer on the MCA signal enabling the DG Control Switch "Pull-To-Lock" position.

B) Local methods of DG shutdown are disabled for these conditions.

C) The DG shutdown actions need to be completed as quickly as possible.

D) Use of the DG Control Switch "Pull-To-Lock" position will not allow a restart should cooling be restored.

76) Unit 2 was operating in MODE 1 at 40% power when it experienced a loss of 20Y050. All required control room actions have been completed.

Under these conditions, operator actions will be impacted by a loss of power to:

A) The RBCCW backup of DWCW which will require manual transfer.

B) The lighting in vital areas which will require the use of flashlights.

C) The Fire Alarm Panel which will require continuous roving fire watches.

D) The Control Room radios which will require the use of alternate communications.

77) Following a reactor scram, the Unit Reactor Operator reported that all APRMs are downscale. Later, the Control Room Supervisor (CRS) directed all control rods be verified to be inserted to or beyond Notch "02".

Which of the following describes why the CRS needs this information?

The CRS:

A) Will direct boron injection (Standby Liquid Control) it this is not true.

B) Is assured the reactor is shutdown and will remain shutdown during the ensuing cooldown.

C) Will exit T-101, "RPV Control" and enter T-117, "Level/Power Control", if this is not true.

D) Is assured the Heat Capacity Temperature Limit will not be exceeded.

78) Unit 2 is operating in MODE 1 at 100% power when the following occurs:

- "REACTOR HI PRESS" alarm 210 G-2 annunciates.
- Reactor Pressure indicates 1075 psig and rising slowly.

In accordance with OT-102 "Reactor High Pressure" which of the following is an appropriate immediate operator action?

A) Control reactor pressure by raising the Bypass Jack setting.

B) Control reactor pressure by lowering the Max Combined Flow Limit Pot.

C) Control reactor pressure by lowering reactor power.

D) Control reactor pressure by raising the Max Combined Flow Limit Pot.

79) Unit 2 is operating at 87% power when the "A" Condensate pump shaft coupling shears. The Condensate pump continues to run at low motor amps.

Given that all three Reactor Feedpumps (RFPs) remain in service and no Operator action is taken, what is the expected plant response to this event?

A) A Recirc runback to 45% speed will occur immediately.

B) A Recirc runback to 45% speed will occur when level is less than +17".

C) A Recirc runback to 30% speed will occur immediately.

D) A Reactor scram will occur when level is less than +1".

80) Unit 2 is at 100% power when Drywell pressure begins to rise.

In accordance with OT-101 "HIGH DRYWELL PRESSURE" follow up actions the following parameters and alarms are noted.

- "A RECIRC PUMP SEAL STAGE 2 HI FLOW" alarm 214 A-1
- PI-2-02-2-033A "Seal 1 Inner" 1056 psig
- PI-2-02-2-032A "Seal 2 Outer" 1043 psig

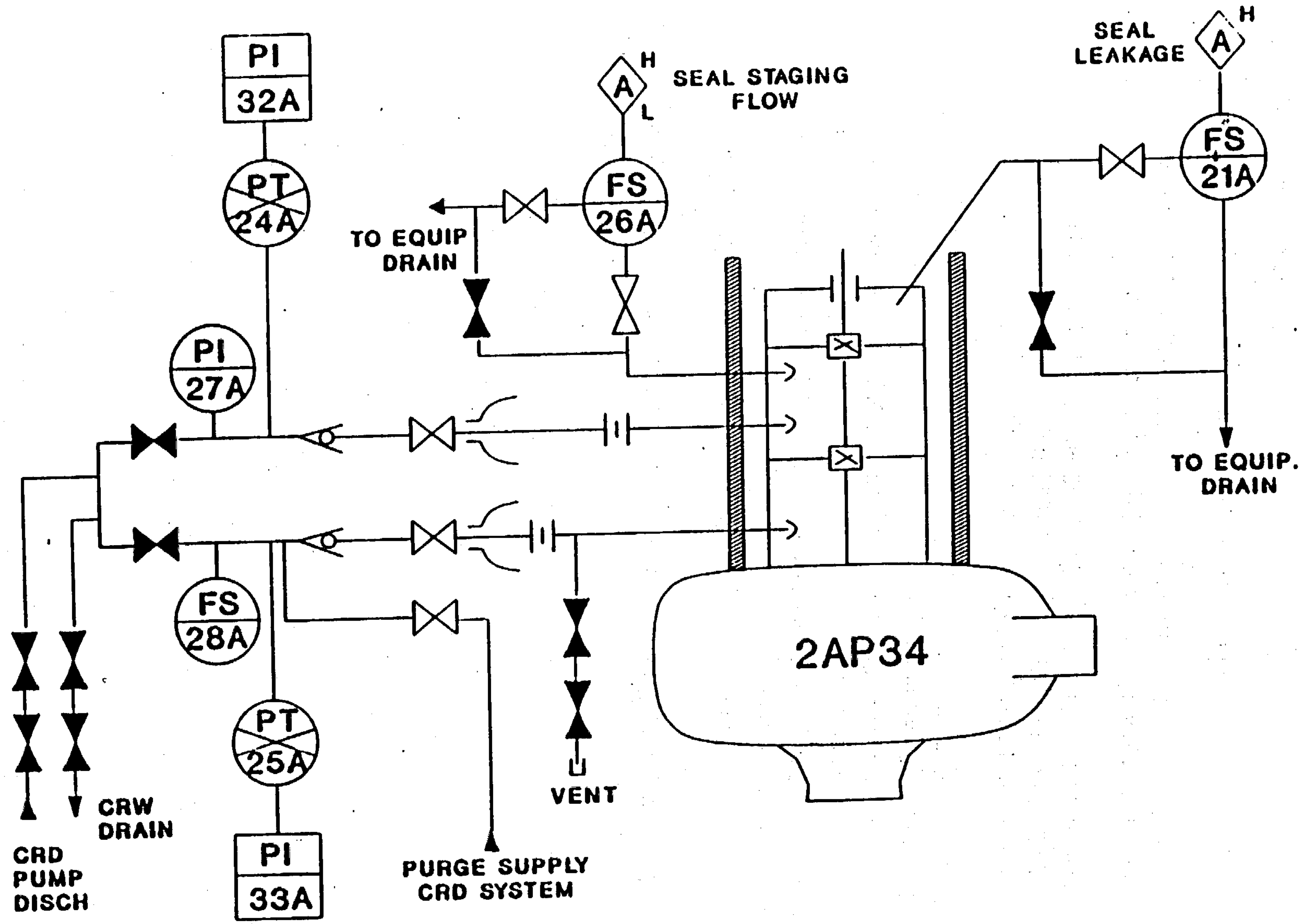
Evaluate these indications, using the attached drawing, and select the appropriate statement below.

A) The 1st stage seal has failed but it is NOT the source of high drywell pressure.

B) The 2nd stage seal has failed but it is NOT the source of high drywell pressure.

C) The 1st stage seal has failed and is the source of high drywell pressure.

D) The 2nd stage seal has failed and is the source of high drywell pressure.



RECIRCULATION PUMP SEAL PIPING

T-LOT-0030-3

Rev 0

81) Unit 3 was operating at 70% power when it experienced a rising drywell pressure. Using OT-101, High Drywell Pressure, the source of the leak has been determined to be the "A" Recirculation pump seals. The CRS has directed you to trip and isolate the "A" Recirculation pump.

Given these conditions, what is the proper sequence for isolating the recirculation pump and why?

A) Shut the suction valve first since it can close against a higher dP.

B) Shut the discharge valve first since it can close against a higher dP.

C) Shut the suction valve first since it is limited to closing against a lower dP.

D) Shut the discharge valve first since it is limited to closing against a lower dP.

82) Unit 2 is at 100% power.

Which of the following events would require power to be reduced or maintained in accordance with OT-104, "Positive Reactivity Insertion"?

A) "A" Reactor Feedpump min flow valve fails open.

B) EHC pressure set setpoint drops 10 psi.

C) Condensate pump trip.

D) Loss of RBCCW to RWCU Non-regen Heat Exchanger.

83) The following conditions exist following the receipt of an automatic scram signal:

- Reactor power: < 1.00 E 0%
- RPV pressure: 950 psig AND dropping
- RPV level: +25 inches AND steady
- Drywell pressure: .5 psig AND steady
- Scram Air Header pressure: 0 psig
- Control Rod 34-27 is at position 48
- All other Control Rods are fully inserted.
- Boron has NOT been injected to the RPV

Which one of the following procedures will provide direction for the successful insertion of Control Rod 34-27 in this situation?

A) GP-3, "Normal Plant Shutdown"

B) T-100, "Scram"

C) T-101, "RPV Control"

D) T-117, "Level/Power Control"

84) Which of the following is the reason why the Main Steam Isolation Valves (MSIV) are closed prior to evacuating the Main Control Room in accordance with SE-1, "Plant Shutdown from the Remote Shutdown Panel"?

A) With MSIVs closed, all reactor inventory and pressure control may take place at the Remote Shutdown Panel.

B) Since plant release points cannot be monitored at the Remote Shutdown Panel, closing the MSIVs precludes any concern for off-site releases.

C) The MSIV closure outside the Main Control Room requires access to plant areas that may not be accessible during an evacuation.

D) If the MSIVs are closed from outside the Main Control Room, there is no method for verification of complete closure.

85) Unit 2 Reactor Operator is controlling reactor level using HPCI at the Unit 2 Alternate Shutdown Panel following Control Room Abandonment. Indicated reactor level on LI-2-2-3-112 is currently 20" and reactor pressure is 500 psig. Using SE-10 Attachment 9, provided, determine the current reactor level and the expected HPCI response if an actual high level condition occurs.

A) Actual level is > 40", HPCI will automatically trip on high level condition.

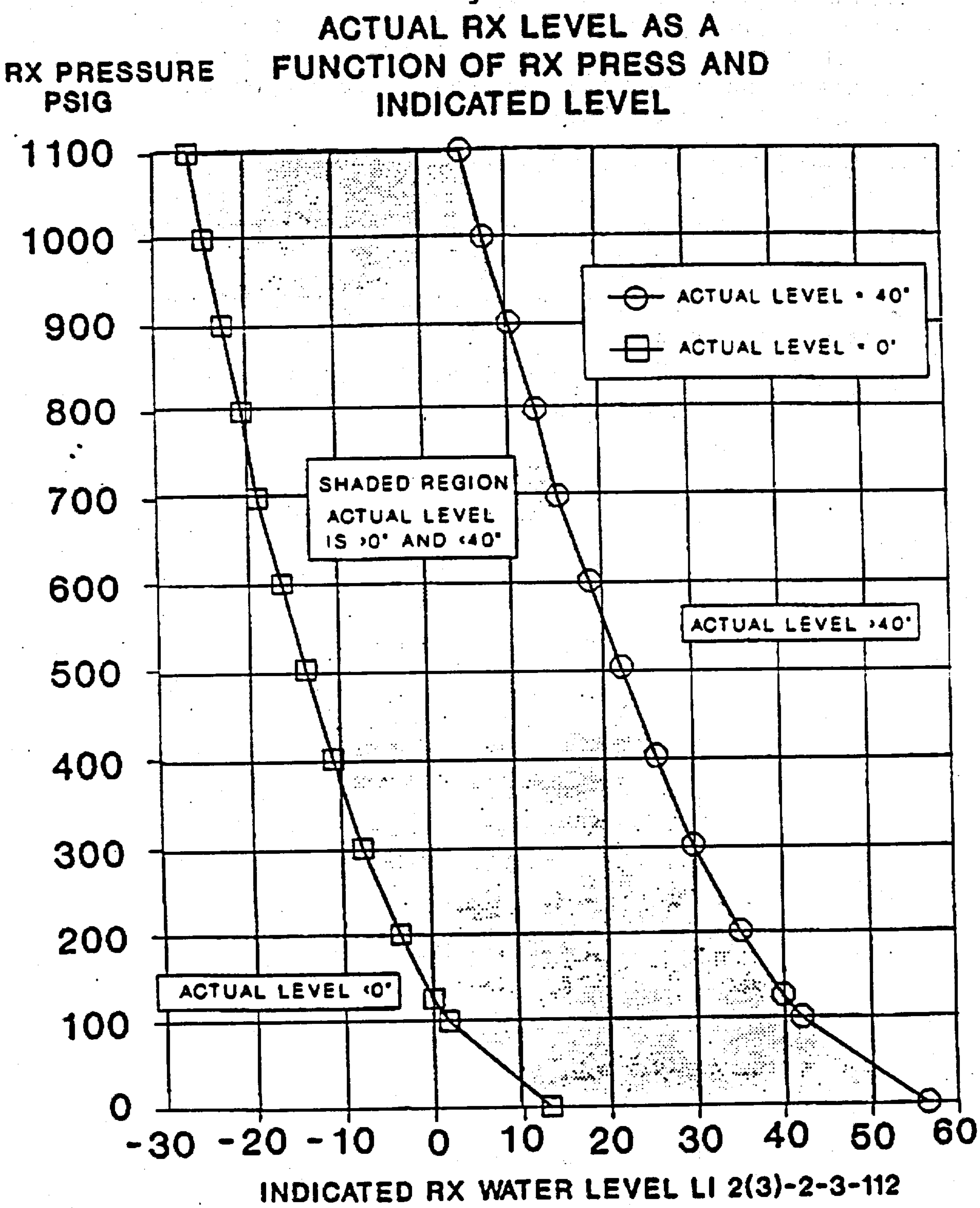
B) Actual level is > 40", HPCI must be manually tripped on high level condition.

C) Actual level is between 0" and 40", HPCI will automatically trip on a high level condition.

D) Actual level is between 0" and 40", HPCI must be manually tripped on a high level condition.

SE-10 Attachment 9
Figure 1

SE-10 Attachment 9
Figure 1



86) Unit 2 was operating at full power in MODE 1 when a positive reactivity event occurred due to a control rod drifting out. The CRS has directed you to monitor for evidence of fuel damage.

Which of the following indications would be the first indication of a small fuel pin leak from this transient?

A) Main Steam Line Radiation Recorders.

B) Air Ejector Discharge Log Monitor Recorders.

C) Off-Gas Adsorber Outlet Radiation Indication.

D) Main Stack Gas Recorder.

87) Given the following conditions:

- Unit 2 has experienced a loss of coolant accident with confirmed fuel failures.
- Drywell and Torus pressures reached 29 psig and sprays were initiated.
- Sprays were NOT manually secured when pressure reached 2.0 psig.
- Sprays did NOT automatically isolate at 1 psig.

Which of the following is the expected impact on the plant for these conditions?

A) The drywell oxygen concentration may rise.

B) Torus water level indication will be unavailable.

C) The running Residual Heat Removal Pumps may cavitate.

D) Failure of the Reactor Building - Torus Vacuum Breakers will make the Reactor Building a High Radiation Area.

88) Following a LOCA on Unit 2 the CRS directs restoration of Drywell Cooling, using T-223, "Drywell Cooler Fan Bypass" for Drywell pressure control. The Unit Reactor Operator reports that the Drywell Cooler fans cannot be placed inservice without an engineering evaluation due to plant conditions falling on the UNSAFE side of T-223 Figure 1, "Drywell Chilled Water (DWCW) Saturation Curve.

Which of the following describes the basis for restricting Drywell Fan restoration when on the UNSAFE side of the curve?

A) Water hammer and rupture of piping inboard of DWCW Isolation valves when flow is restored.

B) Inadvertent lifting of overpressure relief valves inboard of the DWCW Isolation valves when flow is restored.

C) Overcurrent trips of the Drywell Cooler Fans if restarted with a LOCA condition.

D) Overpressurization and rupture of piping inboard of the closed DWCW Isolation valves with a LOCA condition.

89) Unit 2 was operating at 100% power when a total loss of Instrument Air occurred resulting in a plant scram. T-101, "RPV Control" was entered on high reactor pressure at the time of the scram. Normal scram actions have been completed, no other actions have been performed.

In accordance with T-101, RPV pressure control leg, which of the following is the correct method for pressure control under these conditions?

A) Manual operation of SRVs between 950 psig and 1050 psig.

B) Automatic operation of the EHC system at 920 psig.

C) Manual operation of ADS SRVs to stabilize pressure below 1050 psig.

D) Automatic operation of SRVs at their setpoint.

90) Unit 3 has experienced a reactor scram following a steam leak in the Drywell. The CRS directs restoration of Drywell Instrument Nitrogen from T-101, RPV Control, to permit manual reactor pressure control. Restoring Instrument Nitrogen to the Drywell in accordance with GP-8E, "Primary Containment Isolation Bypass":

A) May contribute to a flammable environment in the Drywell.

B) Will only supply nitrogen to the "B" Instrument Nitrogen Header.

C) May deplete CAD nitrogen tank inventory.

D) Will only be permitted if Instrument Air Header pressure is greater than Drywell pressure.

91) Unit 3 has experienced a transient and the following is observed:

- Torus pressure: 9 psig
- Torus temperature: 200 degrees F
- Torus level: 14 feet
- Reactor pressure: 1000 psig
- RHR "A" Loop Flow: 23,000 gpm
- Core Spray "B" Loop Flow: 7500 gpm
- All other low pressure ECCS pump are NOT in service.

Use the attached T-102 Sheet 3 curves to determine if Net Positive Suction Head (NPSH) requirements are being met.

A) There is sufficient NPSH for the "A" Loop of the RHR ONLY.

B) There is sufficient NPSH for the "B" Loop of Core Spray ONLY.

C) There is sufficient NPSH for both the "A" Loop of RHR and the "B" Loop of Core Spray.

D) There is NOT sufficient NPSH for either the "A" Loop of RHR or the "B" Loop of Core Spray.

92) A full power ATWS occurred on Unit 2 which caused excessive heat input to the Torus and a Torus leak. The following conditions currently exist:

- Main Condenser is available.
- Six rods are stuck full out, all other rods are fully inserted.
- Reactor pressure: 950 psig
- Torus temperature: 175 degrees F. and steady
- Torus level 14 ft. and dropping

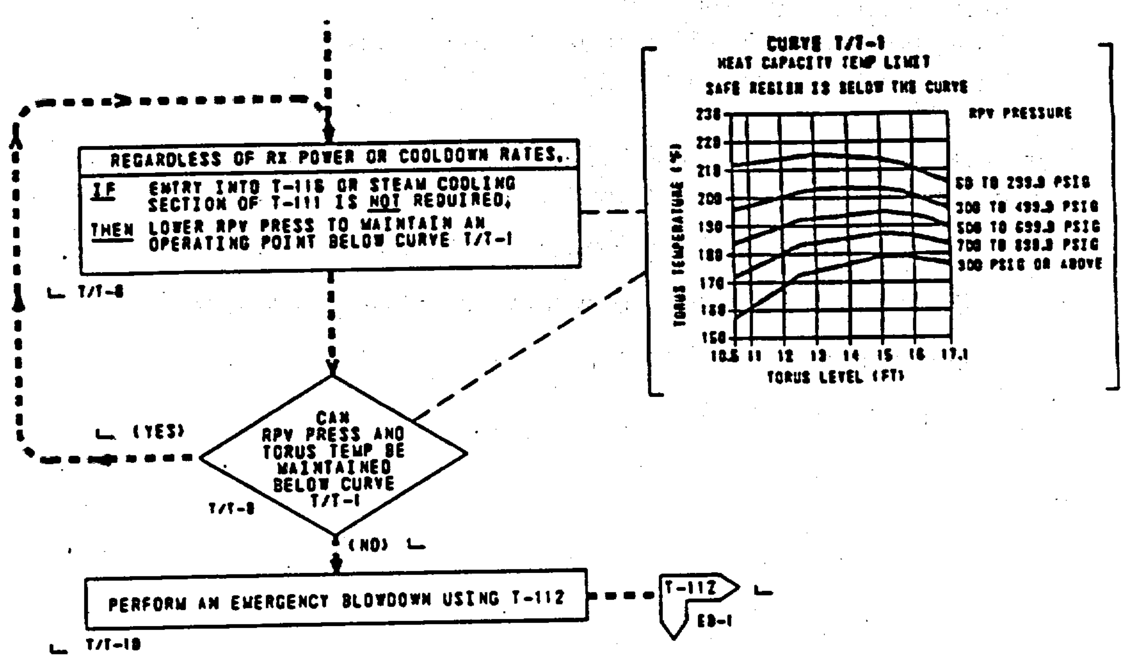
Use the attached portion of T-102 to determine which of the following actions are required as Torus level drops from 14 ft. to 12 ft.

A) Perform an Emergency Blowdown using T-112.

B) Perform an Emergency Blowdown using Bypass valves.

C) Depressurize to 900 psig.

D) Depressurize to 750 psig.


REGARDLESS OF RX POWER OR COOLDOWN RATES,
IF ENTRY INTO T-116 OR STEAM COOLING SECTION OF T-111 IS NOT REQUIRED,
THEN LOWER RPV PRESS TO MAINTAIN AN OPERATING POINT BELOW CURVE T/T-1
T/T-8
CAN RPV PRESS AND TORUS TEMP BE MAINTAINED BELOW CURVE T/T-1
T/T-9
(YES)
(NO)
PERFORM AN EMERGENCY BLOWDOWN USING T-112
T/T-10
T-112
ED-1
CURVE T/T-1
HEAT CAPACITY TEMP LIMIT
SAFE REGION IS BELOW THE CURVE
RPV PRESSURE
50 TO 299.9 PSIG
300 TO 499.9 PSIG
500 TO 699.9 PSIG
700 TO 899.9 PSIG
900 PSIG OR ABOVE
TORUS TEMPERATURE (°F)
230
220
210
200
190
180
170
160
150
10.5
11
12
13
14
15
16
17.1
TORUS LEVEL (FT)

93) For a lowering suppression pool level T-102, "Torus Level", directs that if Torus level cannot be maintained above 9.5' secure HPCI. It does not direct that RCIC be secured until < 6'.

What is the basis for securing HPCI but not RCIC at 9.5'?

A) HPCI turbine exhaust becomes uncovered at 9.5', RCIC turbine exhaust becomes uncovered at 6'.

B) HPCI turbine exhaust becomes uncovered at 9.5', RCIC turbine exhaust is an insignificant containment input.

C) HPCI NPSH becomes a concern at 9.5', RCIC turbine exhaust becomes uncovered at 6'.

D) HPCI NPSH becomes a concern at 9.5', RCIC turbine exhaust is a insignificant containment input.

94) T-111, "Level Restoration" was entered on Unit 3 following a loss of all off-site power and a failure of all diesel generators to start. Current plant conditions are as follows:

- Reactor pressure is 800 psig.
- Reactor level -195" and dropping slowly.
- HPCI tripped on a loss of lube oil.
- RCIC is blocked out of service.

Evaluate these plant conditions and determine the status of Adequate Core Cooling (ACC).

A) ACC exists until level is below -200".

B) ACC exists until level is below -210".

C) ACC does NOT exist, since level is below -172".

D) ACC does NOT exist, since injection is not present.

95) Level recorder LR-2-02-3-110A blue pen is fed by LT-2-02-3-072C "Wide Range" and LT-2-02-3-073C "Fuel Zone" level transmitters.

If level transmitter LT-73C failed upscale and then actual reactor level dropped to -172", what would be the impact on vessel level indications and ECCS initiation from reactor level?

A) LR-110A blue pen input would swap at -100", low level ECCS initiations would NOT be impacted.

B) LR-110A blue pen input would swap at -100", low level ECCS initiations would be impacted.

C) LR-110A blue pen input would NOT swap at -100", low level ECCS initiations would NOT be impacted.

D) LR-110A blue pen input would NOT swap at -100" low level ECCS initiations would be impacted.

96) Following an ATWS and Group I Isolation on Unit 2, the following conditions exist:

- Reactor power: 30%
- Reactor level: -100"
- Torus temperature: 115 degrees F.
- SRV's A, B, C, G open

T-117 level power control directs RPV injection be terminated and prevented using T-240.

For the conditions listed above, which of the following concerns is the basis for performing T-240?

A) Uncontrolled injection of large amounts of cold water.

B) Power generation which is a threat to primary containment.

C) Neutron flux oscillations which challenge fuel clad integrity.

D) Power excursions while establishing minimum alternative RPV flooding pressure.

97) Unit 2 was operating at 100% power when a Reactor high pressure scram condition occurred due to a total loss of instrument air. Control rods failed to insert, reactor pressure peaked at 1180 psig.

The following plant conditions currently exist:

- Reactor power: 35%
- Reactor level: +23"
- Reactor pressure: 1140 psig
- Full core display blue lights lit
- A & B Air Header pressure: 0 psig

Determine which of the following TRIP procedures will insert the control rods.

A) T-213, "Scram Solenoid De-Energization"

B) T-214, "Isolating and Venting the Scram Air Header"

C) T-215, "Control Rod Insertion by Withdraw Line Venting"

D) T-216 "Control Rod Insertion by Manual Scram or Individual Scram Test Switches"

98) A steam leak exists in the Unit 3 Turbine Building. T-104, "Radioactivity Release", has been entered due to high ventilation stack radiation alarms. The Equipment Operator (EO) then reports that Turbine Building Ventilation is tripped.

Under these conditions, determine the appropriate response to the EO's report that Turbine Building Ventilation is tripped.

A) Restart ventilation to monitor the release.

B) Restart ventilation to lower the radioactive release.

C) Maintain ventilation tripped to prevent an unmonitored release.

D) Maintain ventilation tripped to lower the radioactive release.

99) For which of the following conditions would direction be given to initiate Drywell Sprays regardless of whether Adequate Core Cooling is assured?

A) To prevent exceeding the Pressure Suppression Pressure Limit.

B) To maintain Drywell pressure below the Drywell Spray Initiation Limit.

C) To mitigate the consequence of a H2 deflagration.

D) To mitigate the consequences of containment overpressurization.

100) Unit 2 has experienced a large LOCA during which the core was uncovered and fuel cladding oxidation occurred. Adequate core cooling is now assured, containment pressure is 22 psig. Chemistry reports the following containment parameters:

- Torus O2: 4%
- Torus H2: 3%
- Drywell O2: 6%
- Drywell H2: 2%

Using this data and T-102 Sh. 2, "Primary Containment Control" tables (PC/G-1, PC/G-2), provided, select the appropriate TRIP Legs to be entered from those listed below.

A) DW/G-1 and T/G-1

B) DW/G-2 and T/G-1

C) DW/G-3 and T/G-2

D) DW/G-2 and T/G-2

TABLE PC/G-1
DW COMBUSTIBLE GAS CONTROL

<table>
<tr><td colspan="2" rowspan="4"></td><td colspan="4">DRYWELL OXYGEN LEVEL</td></tr>
<tr><td rowspan="3">BELOW 5%</td><td colspan="3">AT LEAST 5% OR UNKNOWN</td></tr>
<tr><td colspan="3">TORUS HYDROGEN LEVEL</td></tr>
<tr><td>BELOW 0.5%</td><td>0.5% TO 5.99%</td><td>AT LEAST 6% OR UNKNOWN</td></tr>
<tr><td rowspan="3">DRYWELL HYDROGEN LEVEL</td><td>BELOW 0.5%</td><td>NO ACTION REQUIRED</td><td>NO ACTION REQUIRED</td><td rowspan="2">DW/G-2</td><td rowspan="3">DW/G-3</td></tr>
<tr><td>0.5% to 5.99%</td><td rowspan="2">DW/G-1</td><td></td></tr>
<tr><td>AT LEAST 6% OR UNKNOWN</td><td></td><td></td></tr>
</table>

TABLE PC/G-2
TORUS COMBUSTIBLE GAS CONTROL

<table>
<tr><td colspan="2" rowspan="4"></td><td colspan="4">TORUS OXYGEN LEVEL</td></tr>
<tr><td rowspan="3">BELOW 5%</td><td colspan="3">AT LEAST 5% OR UNKNOWN</td></tr>
<tr><td colspan="3">DRYWELL HYDROGEN LEVEL</td></tr>
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<tr><td>0.5% to 5.99%</td><td rowspan="2">T/G-1</td><td></td></tr>
<tr><td>AT LEAST 6% OR UNKNOWN</td><td></td><td></td></tr>
</table>

PART 1

1	A
2	B
3	B
4	D
5	D
6	D
7	C
8	B
9	A
10	D
11	D
12	B
13	B
14	B
15	C
16	B
17	B
18	D
19	C
20	D
21	C
22	B
23	B
24	C
25	A
26	B
27	B
28	A
29	B
30	C
31	A
32	C
33	C
34	C
35	C
36	A
37	D
38	B
39	C
40	D
41	A
42	D
43	C
44	D
45	D
46	A
47	D
48	C
49	B
50	C

PECO NUCLEAR

STATION PB / LGS

COURSE TITLE 1999 Peach Bottom SRO Exam FORM SRO

NAME (PRINT) Examination Key (last) ______ (first) ______ (mi)

SOCIAL SECURITY NUMBER ___ - ___ - ___

COMPANY / PECO PAYROLL # ______

DATE 13 SEPT 99

I HAVE REVIEWED AND UNDERSTAND THE CORRECTED QUIZ; ALL WORK ON THIS EXAMINATION IS MY OWN, I HAVE NEITHER GIVEN NOR RECEIVED ASSISTANCE ______ (signature)

IMPORTANT

- USE #2 PENCIL
- EXAMPLE: A B ● D E
- ERASE *COMPLETELY* TO CHANGE

PART 2

51	B
52	D
53	A
54	A
55	A
56	D
57	B
58	C
59	C
60	D
61	A
62	D
63	B
64	B
65	D
66	A
67	C
68	B
69	A
70	C
71	A
72	D
73	C
74	D
75	C
76	D
77	B
78	C
79	D
80	A
81	C
82	A
83	A
84	A
85	D
86	B
87	A
88	A
89	D
90	A
91	B
92	D
93	B
94	B
95	A
96	B
97	C
98	A
99	C
100	B