U. S. NUCLEAR REGULATORY COMMISSION
REGION I

DOCKET/REPORT NOS.:	50-277/95-20 50-278/95-20
LICENSEE:	PECO Energy
FACILITY:	Peach Bottom Unit Nos. 2 and 3
EXAMINATION DATES:	August 17, 1995, to September 1, 1995
EXAMINERS:	D. Florek, Sr Operations Engineer C. Carroll, Examiner, Sonalysts R. Miller, Examiner, Sonalysts

CHIEF EXAMINER: Don Florek 10/19/95
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Operator Licensing and
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Operator Licensing and
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Division of Reactor Safety

EXAMINATION SUMMARY

Examination Report 50-277/95-20 and 50-278/95-20 (OL)

Initial examinations were administered to eight senior reactor operator (SRO) instant applicants and three reactor operator (RO) applicants during the period of August 17, 1995, and September 1, 1995, at the Peach Bottom Atomic Power Station, Units 2 and 3.

<u>OPERATIONS</u>

All RO applicants passed the examination. Three of eight SRO applicants passed the examination. One senior reactor operator applicant failed the written examination. One senior reactor applicant failed both the written and simulator portion of the operating examination. Three senior reactor applicants failed the simulator portion of the operating examination. Several generic weaknesses were noted during the operating portion of the examination.

The SRO applicants frequently lost sight of the big picture and poorly-monitored critical plant parameters and provided insufficient and untimely directions to the other members of the crew.

The SRO applicants had difficulty in prioritizing assigned tasks and frequently held crew briefs at inappropriate times.

The SRO applicants did not effectively plan ahead in either the normal operation or transient portion of the simulator examination.

During a loss of DC bus 20D21, the applicants did not use systematic methods for identification of affected equipment.

The applicants were not able to establish control at the alternate shutdown procedure in the manner specified in SE-2.

Several applicants demonstrated an inability to determine the status of control rod positions and/or reactor power in a timely manner. This inability adversely affected the ability of the SROs to implement the correct TRIP procedures.

Several of the instant SROs had difficulty in locating equipment and controls both in the control room and plant.

None of the SRO applicants questioned were able to identify the dominant risk-significant probabilistic safety-assessment accident sequences or the risk-significant operator errors.

DETAILS

1.0 INTRODUCTION

The NRC administered initial examinations to five senior reactor operator (SRO) instant applicants, three senior reactor operator upgrade applicants, and three reactor operator (RO) applicants. The examinations were administered in accordance with NUREG-1021, "Examiner Standards," Revision 7.

2.0 PREEXAMINATION ACTIVITIES

The facility staff reviewed the written examinations from August 2-3, 1995. The simulator scenarios and job performance measures (JPMs) were validated from August 15-17, 1995, on the facility's simulator and in the plant. The facility staff who were involved with these reviews signed security agreements to ensure that the initial examinations were not compromised.

3.0 EXAMINATION RESULTS AND RELATED FINDINGS, OBSERVATIONS AND CONCLUSIONS

3.1 Examination Results

The results of the examinations are summarized below:

	SRO Pass/Fail	RO Pass/Fail
Written	6/2	3/0
Operating	4/4	3/0
Overall	3/5	3/0

In a letter, dated August 31, 1995 (see Attachment 3), PECO Energy provided 10 comments on the written examination. The NRC accepted seven of the comments and did not accept three comments. As a result, six questions were determined to have two correct answers, two questions were deleted from the SRO written examination, and one question was deleted from the RO written examination. The NRC resolution of facility comments is summarized in Attachment 4.

3.2 Facility Generic Strengths and Weaknesses

The following is a summary of the strengths and weaknesses noted during initial examination administration. This information is being provided to aid the licensee in upgrading their training program.

Written Examination

Questions related to the following specific knowledge/ability topics were missed by at least half of the applicants, indicating a generic weakness in the subject:

SRO-30/RO-33 Knowledge of the combination of MSIV closures that would cause a 1/2 scram.

SRO-67/RO-72 Ability to determine, for a given set of accident conditions, the successful method for insertion of control rods.

SRO-81 Knowledge of the reason for initiating drywell sprays within the safe region of the drywell spray initiation limit curve.

SRO-86/RO-85 Knowledge of shift requirements for return to normal shift rotation after a six-month absence.

SRO-88 Knowledge of the definition of reactor coolant pressure boundary.

SRO-91 Knowledge of the administrative requirements for temporarily being unable to meet the medical requirements of an operator's license.

SRO-99 Knowledge of the administrative reporting requirements for entry into a limiting condition for operation.

RO-41 Knowledge of the interlocks bypassed by the containment spray override keylock switch (S18A/B).

RO-97 Knowledge of the operator actions for a decrease in fuel pool and reactor cavity water level while fuel is being moved.

Scenario Examination

Three scenarios were used to perform the dynamic simulator portion of the examination. One of the scenarios was based on a facility examination bank scenario, and the other two scenarios were newly developed by the NRC. The first NRC scenario, which was run three times, was a loss of a DC bus that was recovered--a loss of high-pressure feed condition generated by a loss of off-site power with a small recirc loop leak and three control rods not fully inserted. The second NRC scenario, which was run twice, was a loss of rod position indication--a cardox initiation in the cable spreading room, which required an operator to initiate a manual scram and activation of the alternate shutdown panel, a loss of feedwater pumps and two turbine bypass valves fail open. The third scenario--a facility examination bank-based scenario run three times--was a control rod drift that caused fuel failure, an unisolable reactor water cleanup leak and reactor vessel instrumentation flashing due to high temperature.

Several generic weaknesses were identified during the simulator portion of the operating examination. No generic strengths were specifically identified. The operator and crew response to the examination scenario that was based on a facility examination bank scenario identified no generic weaknesses, and the crew and individual responses were essentially the same and as expected. The generic weaknesses identified below were identified based on the observed performance of the applicants in both of the newly-developed NRC scenarios.

- The SRO applicants frequently lost sight of the big picture, poorly-monitored critical plant parameters, and provided insufficient and untimely directions to the other members of the crew.
- The SRO applicants had difficulty in prioritizing assigned tasks and frequently held crew briefs at inappropriate times.
- The SRO applicants did not effectively plan ahead in either the normal operation or transient portion of the simulator examination.
- During a loss of DC bus 20D21, the applicants did not use systematic methods for identification of affected equipment.
- The applicants were not able to establish control at the alternate shutdown procedure in the manner specified in SE-2.
- Several applicants demonstrated an inability to determine the status of control rod positions and/or reactor power in a timely manner. This inability adversely affected the ability of the SROs to implement the correct TRIP procedures.

Walkthrough Examination

The generic weaknesses identified in the walkthrough portion of the operating examination are identified below.

- Several of the instant SROs had difficulty in locating equipment and controls both in the control room and plant.
- None of the SRO applicants questioned were able to identify the dominant risk-significant probabilistic safety-assessment accident sequences or the risk-significant operator errors.

3.3 Licensee Actions on Previous Inspection Findings

(CLOSED) (Unresolved Item 277, 278/95-07-01) This unresolved item related to the licensee actions to assure evaluators and trainers do not accept performance where shift briefings are held when the plant is in the region susceptible to power oscillations. PECO Energy had issued several required readings to the training and operating staff to reinforce the management and procedural expectation that, when scrams are required that reduce the recirculation pump flow to minimum, there shall be no delay for briefs or discussions from the point from which the recirc controls are run back to the manual scram. In addition, the Manager of Operator Training frequently monitored simulator training and evaluation sessions and used a simulator performance monitoring check list to assure management expectations during this transition period of power/flow monitoring. Based on the licensee actions this item is closed.

4.0 EXIT MEETING

An exit meeting was conducted on September 1, 1995. Preliminary generic strengths and weaknesses on the operating tests were presented. The chief examiner was provided a signed copy of the facility comments on the written examination. The support given by all of the PECO Energy personnel enabled the examination to be developed and administered very efficiently and effectively.

Persons contacted and attendees at the exit meeting are listed below:

Licensee Personnel

G. Edwards, Plant Manager
A. Fulvio, Manager of NQA
G. Gellrich, Sr. Manager of Operations
G. Jardel, Instructor
D. McClellan, Manager Operator Training
P. Nielsen, Principal Instructor
R. Smith, Regulatory Interface
J. Stankiewicz, Director of Training
A. Wasong, Manager of Experience Assessment

NRC Personnel

D. Florek, Sr. Operations Engineer

Attachments:
1. RO Examination and Answer Key
2. SRO Examination and Answer Key
3. Facility comments on written examinations
4. NRC resolution of facility comments on the written examinations
5. Simulation Facility Report

ATTACHMENT 1

RO EXAMINATION AND ANSWER KEY

ATTACHMENT 1

RO EXAMINATION AND ANSWER KEY

U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
REACTOR OPERATOR LICENSE
REGION 1

APPLICANT'S NAME: ______________________

FACILITY: Peach Bottom 2 & 3

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: August 17, 1995

INSTRUCTIONS TO APPLICANT:

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires a final grade of at least 80%. Examination papers will be picked up four (4) hours after the examination starts.

TEST VALUE	APPLICANT'S SCORE	FINAL GRADE
100		

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

____
Applicant's Signature

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

MULTIPLE CHOICE

001 a b c d ___

002 a b c d ___

003 a b c d ___

004 a b c d ___

005 a b c d ___

006 a b c d ___

007 a b c d ___

008 a b c d ___

009 a b c d ___

010 a b c d ___

011 a b c d ___

012 a b c d ___

013 a b c d ___

014 a b c d ___

015 a b c d ___

016 a b c d ___

017 a b c d ___

018 a b c d ___

019 a b c d ___

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021 a b c d ___

022 a b c d ___

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024 a b c d ___

025 a b c d ___

026 a b c d ___

027 a b c d ___

028 a b c d ___

029 a b c d ___

030 a b c d ___

031 a b c d ___

032 a b c d ___

033 a b c d ___

034 a b c d ___

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036 a b c d ___

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038 a b c d ___

039 a b c d ___

040 a b c d ___

041 a b c d ___

042 a b c d ___

043 a b c d ___

044 a b c d ___

045 a b c d ___

ANSWER SHEET

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

046	a	b	c	d	___	069	a	b	c	d	___
047	a	b	c	d	___	070	a	b	c	d	___
048	a	b	c	d	___	071	a	b	c	d	___
049	a	b	c	d	___	072	a	b	c	d	___
050	a	b	c	d	___	073	a	b	c	d	___
051	a	b	c	d	___	074	a	b	c	d	___
052	a	b	c	d	___	075	a	b	c	d	___
053	a	b	c	d	___	076	a	b	c	d	___
054	a	b	c	d	___	077	a	b	c	d	___
055	a	b	c	d	___	078	a	b	c	d	___
056	a	b	c	d	___	079	a	b	c	d	___
057	a	b	c	d	___	080	a	b	c	d	___
058	a	b	c	d	___	081	a	b	c	d	___
059	a	b	c	d	___	082	a	b	c	d	___
060	a	b	c	d	___	083	a	b	c	d	___
061	a	b	c	d	___	084	a	b	c	d	___
062	a	b	c	d	___	085	a	b	c	d	___
063	a	b	c	d	___	086	a	b	c	d	___
064	a	b	c	d	___	087	a	b	c	d	___
065	a	b	c	d	___	088	a	b	c	d	___
066	a	b	c	d	___	089	a	b	c	d	___
067	a	b	c	d	___	090	a	b	c	d	___
068	a	b	c	d	___	091	a	b	c	d	___

ANSWER SHEET

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

092 a b c d ___

093 a b c d ___

094 a b c d ___

095 a b c d ___

096 a b c d ___

097 a b c d ___

098 a b c d ___

099 a b c d ___

100 a b c d ___

(********** END OF EXAMINATION **********)

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. After the examination has been completed, you must sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination. This must be done after you complete the examination.

3. Restroom trips are to be limited and only one applicant at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

4. Use black ink or dark pencil ONLY to facilitate legible reproductions.

5. Print your name in the blank provided in the upper right-hand corner of the examination cover sheet and each answer sheet.

6. Mark your answers on the answer sheet provided. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.

7. The point value for each question is indicated in parentheses after the question.

8. If the intent of a question is unclear, ask questions of the examiner only.

9. When turning in your examination, assemble the completed examination with examination questions, examination aids and answer sheets. In addition, turn in all scrap paper.

10. Ensure all information you wish to have evaluated as part of your answer is on your answer sheet. Scrap paper will be disposed of immediately following the examination.

11. To pass the examination, you must achieve a grade of 80% or greater.

12. There is a time limit of four (4) hours for completion of the examination.

13. When you are done and have turned in your examination, leave the examination area (EXAMINER WILL DEFINE THE AREA). If you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION: 001 (1.00)

Shutdown cooling (SDC) was in operation with RHR pump "A". A level transient has occurred which has resulted in the following:

- Reactor vessel level is -10 inches.
- SDC automatically isolated as required.

Which one of the following provides the MINIMUM operator actions necessary to inject into the RPV with RHR pump A?

a. Close MO-17 & MO-18 (SDC inboard/outboard isolations).
   Depress both shutdown control pushbuttons (10A-S32A/B).
   Close MO-15 (SDC suction) and open MO-13 (LPCI Torus suction).
   Start RHR pump "A".

b. Close MO-17 & MO-18 (SDC inboard/outboard isolations).
   Depress the "A" shutdown control pushbutton (10A-S32A).
   Close MO-15 (SDC suction) and open MO-13 (LPCI Torus suction).
   Open MO-25A (LPCI Inboard injection).

c. Depress both shutdown control pushbuttons (10A-32A/B).
   Close MO-15 (SDC suction) and open MO-13 (LPCI Torus suction).
   Start RHR pump "A".
   Open MO-25A (LPCI Inboard injection).

d. Close MO-17 & MO-18 (SDC inboard/outboard isolations).
   Close MO-15 (SDC suction) and open MO-13 (LPCI Torus suction).
   Start RHR pump "A".
   Open MO-25A (LPCI Inboard injection).

QUESTION: 002 (1.00)

Reactor Core Isolation Cooling (RCIC) has initiated due to low RPV water level. Level has been restored and normal feedwater has been reestablished. The Control Room Supervisor (CRS) has directed that RCIC be shutdown. The following plant conditions exist:

- Reactor water level is +25 inches and stable.
- Drywell pressure is 2.5 psig and stable.

Which one of the following describes the proper method for shutting down RCIC?

a. Depress the RCIC Manual Isolation pushbutton and realign RCIC to STANDBY.

b. Close the MO-21 RCIC injection valve and then close and reopen MO-4487 trip throttle valve.

c. Close the MO-16 outboard steam isolation valve and realign RCIC to STANDBY.

d. Depress the RCIC Turbine Trip pushbutton and realign RCIC to STANDBY.

QUESTION: 003 (1.00)

Unit 2 is operating at 75% power when the "A" Recirc MG Set Lube Oil pressure drops to 18 psig for 7 seconds, then returns to normal.

Which one of the following describes the response of the "A" Recirc MG Set?

a. Standby AC Lube Oil pumps start.

b. Scoop Tube locks.

c. DC powered lube oil pump starts.

d. Drive motor breaker and field breaker trip.

QUESTION: 004 (1.00)

Which one of the following describes a Reactor Recirculation System pump start limitation and its purpose?

a. Temperature differential between the bottom head region and the steam dome is limited to prevent excessive moisture carryover.

b. The operating pump speed is limited to prevent excessive vibration of the jet pumps in the idle loop.

c. Loop to loop temperature differentials are limited to prevent excessive thermal stresses in the idle loop jet pumps.

d. Loop to dome temperature differentials are limited to prevent thermal hydraulic instability.

QUESTION: 005 (1.00)

While attempting to free a stuck control rod, the operator observes drive water flow indication does NOT change when rod insertion is attempted.

Which one of the following is a possible cause of this indication?

a. Associated drive water stabilizing valve failed open.

b. Drive water pressure control valve failed closed.

c. Cooling water pressure control valve failed closed.

d. Directional control valve failed closed.

QUESTION: 006 (1.00)

Which one of the following describes the effect that a trip of all the condensate pumps will have on the Control Rod Drive Hydraulic System?

The running CRD pump(s) will ________.

a. draw a suction from the CST via the hotwell makeup and reject line

b. develop a low discharge pressure resulting in accumulator trouble alarms

c. overheat due to low flow

d. trip on low suction pressure

QUESTION: 007 (1.00)

Which one of the following describes the RCIC suction valve interlocks?

a. Torus suction valves (MO-39 and MO-41) being full open will cause CST suction valve (MO-18) to auto close.

b. CST suction valve (MO-18) being full open will cause torus suction valves (MO-39 and MO-41) to auto close.

c. Torus suction valves (MO-39 and MO-41) cannot be opened if the CST suction valve (MO-18) is open.

d. CST suction valve (MO-18) opens for an auto RCIC initiation irrespective of the position of the torus suction valves (MO-39 and MO-41).

QUESTION: 008 (1.00)

Which one of the following statements describes the operation of the ADS 105 second timer?

a. The timer must be manually reset to prevent blowdown if the triple low water level signal clears before the timer times-out.

b. The timer will auto reset if the high drywell pressure signal clears before the timer times-out.

c. If the ADS valves are open and the ADS initiation signal is still present, depressing the timer reset pushbutton will close the ADS valves for 105 seconds.

d. The timer time-out can be stopped by placing either keylock Inhibit Switch "A" OR "B" to INHIBIT to prevent ADS valves from opening.

QUESTION: 009 (1.00)

Which one of the following describes the consequences of a loss of 125 VDC panel 20D2111 on the Automatic Depressurization System?

a. Causes the "A" channel to shift to its alternate power supply. ADS initiation is still possible via the "A" or "B" channel.

b. Causes the "B" channel to shift to its alternate power supply. ADS initiation is still possible via the "A" or "B" channel.

c. Causes the "A" channel to deenergize. ADS initiation is still possible through the "B" channel.

d. Causes the "B" channel to deenergize. ADS initiation is still possible through the "A" channel.

QUESTION: 010 (1.00)

Which one of the following statements describes the operation of the LPCI Lockout reset pushbutton (S1A)?

Pushing this bottom will _______.

a. cause LPCI flow to be diverted to torus cooling following a LPCI injection if the LOCA signal is still present

b. automatically open the Recirc pump discharge valve if the LOCA signal has cleared

c. allows MO-25 (LPCI injection valve) to be manually closed after the LOCA signal has cleared

d. reset the LOCA closure lockout of the containment spray valves if the LOCA signal is still present

QUESTION: 011 (1.00)

Select the choice below that completes the following statements.

The reactor is operating at 100% power when the High Pressure Coolant Injection system malfunctions and starts injecting into the reactor. As a result, feedwater temperature decreases, _______.

a. reactor pressure remains constant, and reactor power increases until a reactor scram occurs.

b. reactor pressure remains constant, and reactor power increases but will not cause a scram.

c. Reactor pressure decreases, and reactor power increases but will not cause a scram.

d. Reactor pressure decreases, and reactor power increases until a reactor scram occurs.

QUESTION: 012 (1.00)

During HPCI turbine operation, the operator is cautioned against operating the turbine below 2200 RPM.

Which one of the following is the reason for this precaution?

a. To prevent going below the range of the speed control circuit.

b. To prevent possible cavitation in the pump impeller and volute.

c. To ensure sufficient pump discharge pressure to prevent water hammer in the discharge line.

d. To ensure sufficient oil pressure exists to keep the HPCI turbine stop valve (HO-4513) open.

QUESTION: 013 (1.00)

The HPCI system is in its normal standby readiness lineup.

Which one of the following power supplies, if deenergized, would prevent HPCI from being either manually or automatically initiated?

a. 250 VDC bus (20D11)

b. 480 VAC MCC (20B38)

c. 125 VDC bus (20D22)

d. 480 VAC MCC (30B37)

QUESTION: 014 (1.00)

With the reactor shutdown, a startup of reactor recirculation pump A is about to begin.

Which one of the following statements describes the reactor recirculation pump start sequence?

a. The generator field breaker is initially open, but will shut approximately 10 seconds after the drive motor starts.

b. The pump must develop at least 5 psid within 21 seconds of the field breaker shutting or an MG set incomplete sequence trip will occur.

c. When the drive motor breaker is closed, the generator will initially go to 50% speed, but after the field breaker shuts, generator speed will decrease to 28% to 30%.

d. When the drive motor breaker is closed, the generator will initially go to 100% speed, but after the field breaker shuts, generator speed will decrease to 28% to 30%.

QUESTION: 015 (1.00)

Unit 2 and 3 are operating at 100% power with the Control Room Ventilation System in its normal alignment when the operator observes the following alarms and indications:

- "CONTROL ROOM RAD MONITOR DIV. I INITIATED" alarm is active.
- "CONTROL ROOM RAD MONITOR DIV. II INITIATED" alarm is active.
- "CONTROL ROOM VENT SUPPLY FLOW HI-LO" alarm is active.
- Control Room Vent flow recorder (FR-0765) indicates 40 scfm.
- Control Room Radiation Monitors:
  RI-0760A and RI-0760B red high lights are lit.
  RI-0760C and RI-0760D amber failure lights are lit.

Which one of the following describes the expected response of the Control Room Ventilation System?

a. Control Room Emergency Ventilation Fan OAV-30 or OBV-30 will start after a 60 second time delay.

b. Control Room Emergency Ventilation Fan OAV-30 or OBV-30 will immediately start.

c. Control Room Ventilation dampers remain aligned for normal operation and the standby Fresh Air Supply fan will start after a 60 second time delay.

d. Control Room Ventilation dampers will immediately shift the discharge of the running Fresh Air Supply fan to an Outside Air Cleanup Filter train.

QUESTION: 016 (1.00)

Which one of the following describes the expected plant response if the condensate pump discharge header pressure drops below 400 psig during a plant startup?

a. Condensate recirculation valve will close.

b. Condensate pump suction valve will close.

c. SJAE Condenser condensate inlet valve will close.

d. Condensate reject control valve will open.

QUESTION: 017 (1.00)

The plant is operating at 100% power when the "A" condensate pump trips.

Which one of the following describes the consequences of this malfunction?

a. The reactor recirculation pumps runback to 45% speed and one feed pump trips.

b. The reactor recirculation pumps and the feed pumps both runback to 45% speed.

c. The reactor recirculation pumps runback to 45% speed and the feed pumps runback to 85% speed.

d. One feed pump runs back to 85% speed and one feed pump trips.

QUESTION: 018 (1.00)

While placing the second reactor feed pump (RFP) in service during a startup, discharge check valve slamming is detected.

Which one of the following describes the actions which must be taken?

a. Immediately shut the affected RFP discharge valve.

b. Immediately trip the affected RFP.

c. Reduce the speed of the affected RFP.

d. Raise the speed of the affected RFP.

QUESTION: 019 (1.00)

Which one of the following conditions will result in the largest decrease in feedwater inlet temperature at 100% reactor power?

a. Loss of extraction steam to a fifth stage feedwater heater.

b. Inadvertent Reactor Core Isolation Cooling Injection.

c. Inadvertent High Pressure Coolant Injection.

d. Inadvertent closure of the SJAE condenser condensate inlet valve.

QUESTION: 020 (1.00)

The following conditions exist:

- ALL emergency diesel generators (EDG) started in response to a MCA (Maximum Credible Accident) start signal.
- One EDG received a jacket temperature high alarm and trip signal immediately after loading.
  -- The trip is automatically bypassed by the MCA start.
  -- The EDG successfully supplied power as required.

Which one of the following describes the expected response of this EDG during restoration of off-site power?

a. The EDG will trip when the Unit/Parallel switch is placed in PARALLEL if the MCA signal is clear.

b. The trip will be enabled after a manual control room stop if the MCA signal is clear.

c. The EDG will trip as soon as the MCA signal is clear.

d. The EDG will trip when either off-site supply breaker to its emergency bus is closed in parallel with the EDG.

QUESTION: 021 (1.00)

Plant conditions are as follows:

- MSIVs are closed, and RPV pressure is being controlled between 960 psig and 1060 psig with the SRVs.
- RPV water level is 23 inches and steady.
- Drywell pressure is 0.75 psig.
- The #2 Emergency Auxiliary Transformer (0AX04) is out of service.
- All eight 4 KV busses are being supplied by the #3 Emergency Auxiliary Transformer (0BX04).
- The 2A RHR and 2A HPSW pumps are running in Torus Cooling Mode.

The E-312 breaker then trips due to a breaker failure.

Which one of the following statements describes the expected plant response?

a. The E-1 Diesel Generator will auto start but not tie to the E-12 Bus.

b. The E-1 Diesel Generator will auto start and tie to the E-12 Bus, but 2A RHR and 2A HPSW will not auto start.

c. The E-1 Diesel Generator will auto start and tie to the E-12 Bus, and 2A RHR pump will auto start but 2A HPSW pump will not auto start.

d. The E-1 Diesel Generator will auto start and tie to the E-12 Bus, and both the 2A RHR pump and 2A HPSW pump will auto start.

QUESTION: 022 (1.00)

The E-2 Diesel Generator is tied to the E-22 bus for testing and is running in PARALLEL with offsite power. The following parameters are observed by the operator:

DG Voltage:	4.20 KV
DG Frequency:	60 Hz
DG Load:	2300 KW
DG KVAR:	1900 KVAR

Which one of the following statements describes the proper operator action?

a. Reduce D/G voltage to 4.16 KV

b. Reduce D/G KW using the GOVERNOR control switch to achieve a 0.75 power factor.

c. Reduce D/G KVAR using the AUTO VOLT REG control switch to achieve a 0.75 power factor.

d. Maintain current stable D/G operation since all parameters are within normal operating bands.

QUESTION: 023 (1.00)

Which one of the following describes the operation of the Reactor Water Cleanup system in the DUMP MODE of operation?

a. The dump mode will isolate at 140 psig (decreasing) upstream of the Clean Up Drain Header Control Valve (CV-55).

b. The dump mode will isolate at 5 psig (decreasing) upstream of the Clean Up Drain Header Control Valve (CV-55).

c. The RWCU system will isolate if the regenerative heat exchanger outlet temperature exceeds 200 deg. F.

d. The RWCU system will isolate if the non-regenerative heat exchanger outlet temperature exceeds 130 deg. F.

QUESTION: 024 (1.00)

Which one of the following conditions will cause the Reactor Water Cleanup System to ISOLATE?

a. RWCU suction line flow (300%).

b. RPV level (+29 inches)

c. RWCU pump flow (70 gpm)

d. Drywell pressure (2.0 psig)

QUESTION: 025 (1.00)

Which one of the following describes the gas flow through the Off-Gas system?

a. Composed mostly of hydrogen and oxygen in a 2:1 ratio which is catalytically recombined.

b. Receives its driving force from the third stage air ejector which receives its normal steam supply from the main turbine cross around header.

c. Superheated to about 3350 degrees F by an electric preheater to maximize the efficiency of the catalytic recombiner.

d. Will automatically isolate the Steam Supply valves to the Jet Compressors in the event that a "STACK GAS HIGH RADIATION" alarm condition occurs.

QUESTION: 026 (1.00)

Unit 2 is at 100% power when the following alarm is received:

- JET COMPRESSOR STEAM FLOW LOW

Which one of the following statements describes the plant response and/or the operator action that will be required?

a. If normal steam flow cannot be restored by the on-line pressure control valve (PCV), the standby PCV must be manually placed on-line.

b. If steam flow cannot be restored above the alarm setpoint within 30 seconds, the Off-Gas inlet valve will close.

c. To increase steam flow, the off-line set of Steam Jet Air Ejectors can be manually placed in service.

d. Reactor power must be reduced to less than 70% to minimize the amount of noncondensibles that are being built up in the main condenser.

QUESTION: 027 (1.00)

Which one of the following describes the operation of the individual manual/automatic (M/A) stations and the master control station for the Reactor Feed Pump Turbines (RFPTs)?

With all three RFPT M/A stations ________.

a. in "MANUAL" and the master controller in "MANUAL", RFPT speed may be adjusted by use of the master controller control knob

b. in "MANUAL" and the master controller in "AUTO", RFPT speed may be adjusted by use of the master controller control knob.

c. in "AUTO" and the master controller in "MANUAL", one slow turn of the master controller control knob produces the same RFPT speed change as one fast turn, but at a slower rate of speed increase.

d. in "AUTO" and the master controller in "MANUAL", the faster the master controller control knob is turned, the larger the magnitude of the RFPT speed change.

QUESTION: 028 (1.00)

Given the following information and the attached figure of the Electro-hydraulic Pressure Control (EHC) System:

PAM Pressure	950 psig
EHC Pressure Setpoint	920 psig
Load Limit	105 percent
Pressure Regulator "A"	Controlling

The PAM Pressure transmitter (PT-2184) has just failed downscale to 0 psig (input to pressure regulator "A").

Which one of the following describes the EHC system response?

a. "A" Pressure Regulator remains in control - a large reactor pressure INCREASE occurs

b. "A" Pressure Regulator remains in control - a large reactor pressure DECREASE occurs

c. "B" Pressure Regulator takes control - reactor pressure INCREASES slightly

d. "B" Pressure Regulator takes control - reactor pressure DECREASES slightly

QUESTION: 029 (1.00)

Plant conditions for Unit 2 are as follows:

- Drywell pressure is 1.45 psig.
- Reactor water level is -56 inches.
- Reactor pressure is 70 psig.
- Mode Switch is in SHUTDOWN.

Note: Primary Containment Isolation System (PCIS) Groups are listed below:

I	Main steam system isolation valves
II	Reactor auxiliary systems isolation valves
III	Ventilation system isolation valves and dampers
IV	HPCI
V	RCIC
VI	Core Spray
VII	RHR

Which one of the following states the PCIS group isolations that are active?

a. I, II, VI, VII

b. II, III, IV, V

c. III, IV, V, VI

d. I, V, VI, VII

QUESTION: 030 (1.00)

The Digital Feed Control System (DFCS) is controlling reactor water level in three element control.

Which one of the following describes the response of the Digital Feed Control System (DFCS) to a loss of ONE (1) steam flow signal?

a. DFCS auto transfers to single element control, C05 total steam flow recorder will increase.

b. DFCS auto transfers to single element control, C05 total steam flow recorder will decrease.

c. DFCS remains in three element control, C05 total steam flow recorder will increase.

d. DFCS remains in three element control, C05 total steam flow recorder will decrease.

QUESTION: 031 (1.00)

Which one of the following statements describe the Rod Block Monitor (RBM) System channel "B" nulling process?

a. The RBM high/inop trips are inhibited during the nulling sequence to prevent interrupting rod motion.

b. A rod block is generated during the nulling sequence.

c. Bypassing the "A" APRM will reset the nulling sequence clock and initiate a new nulling sequence.

d. Depressing the "Set Up" pushbutton when the selected trip level is reached initiates a new nulling sequence.

QUESTION: 032 (1.00)

Which one of the following statements describes the LPRM inputs to the Rod Block Monitor?

a. The RBM considers an LPRM operable and uses its input for the averaging circuit if the LPRM function switch is in OPERATE.

b. The RBM is bypassed when the average of all the operable LPRMs to a RBM indicates less than 30%.

c. An LPRM that is reading downscale will not be counted in the RBM count circuit.

d. An LPRM upscale condition will automatically bypass the LPRM input to the RBM.

QUESTION: 033 (1.00)

Which one of the following describes a Main Steam Isolation Valve (MSIV) alignment that would cause a HALF scram ONLY via the Reactor Protective System MSIV valve logic?

a. Both MSIVs in A steam line closed.

b. One MSIV in A steam line closed, one MSIV in B steam line closed.

c. One MSIV in A steam line closed, one MSIV in D steam line closed.

d. One MSIV in B steam line closed, one MSIV in C steam line closed.

QUESTION: 034 (1.00)

T-216-2, "Control Rod Insertion by Manual Scram or Individual Scram Test Switches", provides instructions for manually scramming control rods using the individual rod scram test switches.

Which one of the following describes the mechanism by which these switches affect control rod insertion?

a. A single switch, when taken to the TEST position, will DE-ENERGIZE both scram pilot solenoids for its associated control rod.

b. A single switch, when taken to the TEST position, will ENERGIZE both scram pilot solenoids for its associated control rod.

c. Two switches are provided for each associated control rod, and, when taken to the TEST positions, will DE-ENERGIZE their associated scram pilot solenoids.

d. Two switches are provided for each associated control rod, and, when taken to the TEST positions, will ENERGIZE their associated scram pilot solenoids.

QUESTION: 035 (1.00)

Which one of the following gives the approximate IRM range at which the reactor FIRST reaches the Point of Adding Heat (POAH)?

a. Mid-range on range 4

b. Between ranges 5 and 6

c. Between ranges 7 and 8

d. Mid-range on range 9

QUESTION: 036 (1.00)

Unit 2 is conducting a startup following a 3-day forced outage due to drywell leakage. Power ascension has increased into the RUN mode and is at 12% when APRM B fails downscale. The operator reaches up and inadvertently downranges IRM H, causing it to read 121/125 on Range 8.

Which one of the following describes the plant response?

a. A rod block ONLY will occur due to the APRM downscale, and an alarm ONLY will occur due to the IRM HI HI trip.

b. A rod block ONLY will occur due to the IRM HI HI trip, and an alarm ONLY will occur due to the APRM downscale.

c. A FULL SCRAM will result from the combination of the APRM downscale and the IRM HI HI trip.

d. A HALF SCRAM will result from the combination of the APRM downscale and the IRM HI HI trip.

QUESTION: 037 (1.00)

During a reactor startup with the Reactor Mode Switch in STARTUP, all of panel 2BD45 (24 VDC power to IRMs B, D, F, H and SRMs B and D) is deenergized.

Which one of the following describes the status of rod block and scram signals?

a. Full scram ONLY.

b. Rod block ONLY.

c. Rod block and half scram.

d. Rod block and full scram.

QUESTION: 038 (1.00)

While the HPCI system is in operation for a surveillance test on Unit 2, a spent fuel bundle is dropped in the Unit 3 spent fuel pool. The refuel floor exhaust duct radiation levels reach 30 mrem/hr and a Group III isolation occurs. However, the bypass damper (PO-00522) on the Standby Gas Treatment System (SBGT) does NOT close on the Group III isolation signal.

Which one of the following describes how SBGT system operation will be affected?

a. The increased system air flow through the charcoal filters would reduce the ability of the filters to remove radioactive iodine from the air in Unit 3.

b. The increased system air flow could cause implosion of the HPCI barometric condenser.

c. The decreased system air flow will increase the humidity of the incoming air which would decrease the overall effectiveness of the charcoal filter.

d. The decreased system air flow may prevent a negative pressure from being developed in the Reactor Building and Refuel Floor of Unit 3.

QUESTION: 039 (1.00)

Which one of the following statements describes the closing operation of the Main Steam Isolation Valves (MSIVs)?

a. Pneumatic pressure is NOT used to close the MSIVs.

b. Instrument air supplies pneumatic pressure to the inboard MSIVs during the valve closure operation.

c. Both the AC and DC solenoids must de-energize to close the MSIVs.

d. During test operation the MSIVs close under spring pressure alone.

QUESTION: 040 (1.00)

Unit 2 is operating at 100% power when a reactor scram occurs and an SRV actuates. When the SRV closes the tailpipe relief line vacuum breaker sticks OPEN.

Which one of the following describes the consequence of a subsequent actuation of this SRV?

a. It will have NO consequences since the vacuum breaker is normally open.

b. Suppression chamber pressure will INCREASE since noncondensed steam would be vented directly to the torus air space.

c. Water will be "pulled up" into the relief line and could result in over-pressurization of the tailpipe.

d. Drywell pressure will INCREASE since non-condensed steam would be vented directly to the drywell atmosphere.

QUESTION: 041 (1.00)

Following a LOCA, the Containment Spray Override-keylock Switch (S18A/B) has been placed to the "MANUAL OVERRD" position.

Which one of the following Containment Spray initiation interlocks will be bypassed by this action?

a. RPV level above -226 inches and LOCA present or sealed in.

b. RPV level above -226 inches and Containment Spray Valve Control Switch (17A/B) in MANUAL.

c. RPV level above -226 inches and Drywell pressure above 1 psig.

d. Only RPV level above -226 incnes.

QUESTION: 042 (1.00)

Which one of the following plant conditions will generate a Process Radiation Monitoring System isolation signal?

a. Refuel Floor Exhaust Ventilation Radiation Monitors, RIS-2-17-458A and RIS-2-17-458B, indicating 7 mr/hr.

b. Main Stack Radiation Monitor, RIS-50A, reaches the HIGH alarm setpoint (300 cps) while venting the drywell through the 18 inch vent valves to SBGT.

c. Main Steam Radiation Monitors, RIS-2-17-251A and RIS-2-17-251D indicate 11 R/hr (11 x NFPB), and RIS-2-17-251B and RIS-2-17-251C indicate 8.5 R/hr (8.5 x NFPB).

d. Liquid Radwaste Radiation Monitor, RIS-17-350, reaches the HIGH alarm setpoint with a radwaste discharge in progress.

QUESTION: 043 (1.00)

A loss of offsite AC power occurs such that Emergency Diesel Generators automatically start and restore their associated 4KV Emergency busses.

Which one of the following describes the response of the 125 VDC Station Battery Charger when 480V MCC power returns?

a. The battery chargers will automatically return to the "equalizing" mode.

b. The battery chargers will automatically return to the "float" mode.

c. The battery chargers trip and must be manually restored.

d. The battery chargers load shed and then automatically return to normal operation when voltage is restored on the emergency busses.

QUESTION: 044 (1.00)

Which one of the following describes the effect of a Loss of Offsite Power on the Fire Protection system?

a. If power is lost to the Motor Driven Fire Pump controller for more than 8 seconds, the pump WILL NOT automatically start. It can be manually started.

b. If power is lost to the Motor Driven Fire Pump controller for more than 8 seconds, the pump WILL NOT automatically start and CANNOT be manually started.

c. Loss of AC power to the diesel battery charger will result in a failure of the Diesel Driven Fire Pump to automatically start. It CAN be manually started.

d. Loss of AC power to the diesel battery charger will result in a failure of the Diesel Driven Fire Pump to automatically start, and it CANNOT be manually started.

QUESTION: 045 (1.00)

Unit 2 is operating at 90% power when the following plant conditions occur:

- "CONDENSER LOW VACUUM ALARM" is received.
- Condenser vacuum is 25.4 inches and slowly decreasing.

Which one of the following provides the required IMMEDIATE operator action(s)?

a. Verify proper operation of the on-line SJAE, place the standby SJAE in service, and enter T-100 "Scram" procedure if condenser vacuum continues dropping.

b. Scram the reactor, enter T-100 "Scram" procedure, and close the MSIVs.

c. Reduce reactor power per GP-9-2 "Fast Power Reduction" until condenser vacuum stops dropping.

d. Reduce reactor power per GP-9-2 "Fast Power Reduction" until "CONDENSER LOW VACUUM" alarm clears or an "APRM HIGH" alarm is received.

QUESTION: 046 (1.00)

Unit 3 is operating at 100% power when Recirculation Pump B trips. Plant conditions are as follows:

- APRMs indicate 71%.
- Calculated core flow is 46%.
- No thermal hydraulic instability has been observed.
- No operator actions have been taken.

Which one of the following is the operator action that must be IMMEDIATELY performed? (OT-112, Figures A and B, Power to Flow Maps are attached for reference.)

a. Insert rods in reverse sequence to exit Region 2 and continually monitor for thermal hydraulic instability.

b. Manually scram the reactor and enter T-100 "Scram" procedure.

c. No rod insertion is necessary but continually monitor for thermal hydraulic instability.

d. Fully insert the Table 1 rods of GP-9-3 "Fast Reactor Power Reduction".

QUESTION: 047 (1.00)

A normal unit 2 reactor shutdown is in progress per GP-3 "Normal Plant Shutdown". The reactor has been scrammed and the turbine has just tripped. The operator observes that all turbine bypass valves have fully opened and reactor pressure is 905 psig and decreasing.

Which one of the following is the required IMMEDIATE operator action?

a. Close the MSIVs and control reactor pressure with SRVs.

b. Manually jack the bypass valves closed to stabilize pressure.

c. Runback the Max Combined Flow Limiter pot to stabilize main steam line pressure above 900 psig but close MSIVs if pressure decreases below 900 psig.

d. Runback the Max Combined Flow Limiter pot to stabilize main steam line pressure and leave the MSIVs open even if steam line pressure is stabilized below 900 psig.

QUESTION: 048 (1.00)

Select the choice below that completes the following statements. (Assume no operator actions are taken.)

Unit 2 is operating at 90% power when a safety relief valve inadvertently opens and sticks open. As a result, feedwater flow rate will stabilize ________.

a. HIGHER than main steam flow rate, and PAM (Pressure Averaging Manifold) pressure will stabilize at its previous value

b. HIGHER than main steam flow rate, and PAM (Pressure Averaging Manifold) pressure will decrease and stabilize at a LOWER pressure

c. LOWER than main steam flow rate, and PAM (Pressure Averaging Manifold) pressure will stabilize at its previous value

d. LOWER than main steam flow rate, and PAM (Pressure Averaging Manifold) pressure will decrease and stabilize at a LOWER pressure

QUESTION: 049 (1.00)

Unit 2 is operating at 90% power when the operator observes the following plant conditions:

- Reactor pressure 1055 psig and slowly increasing.
- APRMs slowly increasing.

Which one of the following is the FIRST IMMEDIATE operator action?

a. Operate the Bypass Valve Jack to control reactor pressure below 1085 psig.

b. Operate the Bypass Valve Jack to control reactor pressure below 1050 psig.

c. Lower the EHC pressure set to control reactor pressure below 1050 psig.

d. Lower the EHC pressure set to control reactor pressure below 1085 psig.

QUESTION: 050 (1.00)

SELECT the choice below that completes the following statements.

During execution of OT-101 "High Drywell Pressure", the operator determines that both seals on Reactor Recirc Pump B have failed. Per OT-101, the recirc pump is tripped and isolated. When isolating the recirc pump, the operator is directed to ________.

a. simultaneously close the suction and discharge valves to quickly isolate the seal leak.

b. first close the seal purge isolation valves to reduce radioactive leakage from the pump seals when the pump is isolated.

c. first close the pump suction valve because a high differential pressure across the valve may prevent its closure if the discharge valve is closed.

d. first close the pump discharge valve due to its ability to operate against a large differential pressure.

QUESTION: 051 (1.00)

Unit 3 is operating at 85% power at 1020 psig when the following plant conditions occur:

- Reactor pressure spikes to 1043 psig and then stabilizes at 1030 psig.
- Reactor power increases to 91% and then stabilizes at 85%.

Which one of the following failures would cause this plant response?

a. The on-line EHC regulator's setpoint has failed high and the backup regulator is in control.

b. One MSIV disk has separated from its stem and has failed fully closed.

c. One SRV has inadvertently lifted and has failed to fully reseat upon re-closing.

d. The extraction steam to the fifth-point feedwater heater has isolated.

QUESTION: 052 (1.00)

Following a transient, Unit 3 has scrammed and plant conditions are as follows:

- Reactor pressure has been stabilized at 500 psig.
- MSIVs are open.

Which one of the following is the MAXIMUM ALLOWABLE reactor water level indication on LI-3-2-3-86 for which the MSIVs may remain open? (Refer to OT-110, Figure 1, LI-2(3)-2-3-86 Indication.)

a. +108 inches

b. +93 inches

c. +78 inches

d. +60 inches

QUESTION: 053 (1.00)

A transient on unit 2 has resulted in flooding of the main steam lines and a reactor pressure of 500 psig that is rapidly increasing (150 psig per minute).

Which one of the following will increase the SRV hydraulic discharge loads when the SRV(s) is/are opened?

a. Increased drywell temperature and pressure.

b. Reduced subcooling of the water in the main steam lines.

c. Prolonging the time that an SRV is open to reduce reactor pressure.

d. Opening the SRV at a reactor pressure of 1030 psig instead of 700 psig.

QUESTION: 054 (1.00)

Unit 2 is operating at power when a recirculation flow reduction event results in entry into Region 2 of the Power to Flow Map (attached for reference). Plant conditions prior to the event were as follows:

- Reactor power was 90%.
- APRMs indicated 90% +/- 1%.
- All LPRMs were above downscale alarms and below upscale alarms.
- LPRMs near center of core indicated 95% +/- 2%.
- Period meter indicated infinity.

After the flow reduction event and core flow first reaches its lowest flow rate, which ONE of the following neutron instrumentation responses indicates reactor instability?

a. APRMs are oscillating between 58% and 62% every second.

b. Period meter is oscillating between a -50 second period and a +50 second period every 2 seconds.

c. LPRMs near the center of the core are oscillating between 64% and 72% every 10 to 15 seconds.

d. LPRM downscale alarms occur briefly and then clear at 10 seconds, 25 seconds, and 60 seconds.

QUESTION: 055 (1.00)

Which one of the following describes the effect of losing both divisions of the 24 VDC power system?

a. Core Spray will inadvertently initiate.

b. Reactor feedwater pumps will lock up.

c. A main turbine trip occurs only if turbine first stage pressure indicates greater than 30% power.

d. A reactor scram occurs if the Mode Switch is in STARTUP.

QUESTION: 056 (1.00)

While at 100% power the Unit 2 operator observes the following plant conditions.

- "INVERTER TROUBLE" alarm
- Loss of control rod position indication.

Which one of the following describes the expected plant response?

a. Turbine bypass valves will fail closed if the main turbine trips and its speed coasts below 1650 rpm.

b. Reactor recirc pumps will trip if pump speed is greater than 30%.

c. RFP "C" will lockup and all RFP minimum flow recirc valves will fail open.

d. The hydraulic jack for each running RFP will automatically take control of RFP speed and lock pump speed at its current value.

QUESTION: 057 (1.00)

Unit 3 is operating at 100% power when the operator observes the following plant conditions.

- CRD charging water header pressure is 1450 psig.
- In-service CRD flow control valve red indicating light is ON and the green indicating light is OFF.
- CRD system flow is 120 gpm.

Which one of the following CRD component failures has occurred?

a. Running CRD pump impeller and wear ring failure.

b. CRD drive water pressure control valve has closed.

c. CRD stabilizing valves have failed open.

d. CRD flow controller has failed.

QUESTION: 058 (1.00)

Unit 2 is operating at 90% power when tne operator observes a "ROD DRIFT" alarm and identifies rod 06-27 is drifting out. Three minutes later rod 50-25 begins slowly drifting out.

Which one of the following is the REQUIRED operator action per ON-121 "Drifting Control Rod" procedure?

a. Insert the drifting rods to Full-In using the EMERG IN control switch.

b. Isolate the drifting CRDM HCU and vent the CRDM overpiston area to insert the rods.

c. Individually scram each drifting rod at Panel 20C016.

d. Manually scram the reactor and enter T-100 "Scram" procedure.

QUESTION: 059 (1.00)

Unit 3 is operating at 100% power and experiencing a loss of instrument air pressure in the Turbine Building.

Which one of the following describes the expected plant response?

a. The hotwell reject valves will fail open, eventually causing a trip of the condensate pumps on low suction pressure.

b. The condensate short path recirculation valve fails open, causing a reduction in RFP suction pressure and discharge capacity.

c. The RFP minimum flow recirc valves will fail open, causing a reduction in reactor feedwater flow.

d. The condensate filter demin inlet and outlet valves will fail closed and the demin bypass valve will open, causing a trip of the RFPs.

QUESTION: 060 (1.00)

During a hot summer month, an SRV opens and then re-closes but does not reseat.

Which one of the following describes the operator action required to meet both Technical Specifications and operating procedures?

a. Commence a shutdown to HOT SHUTDOWN if torus temperature reaches 105 deg. F.

b. Perform an external visual examination of the torus if torus temperature exceeds 120 deg. F.

c. Initiate a manual reactor scram if torus temperature reaches 110 deg. F.

d. Commence a shutdown to COLD SHUTDOWN if torus temperature reaches 120 deg. F.

QUESTION: 061 (1.00)

Unit 2 is operating at 100% power when a plant transient occurs. Plant conditions are as follows:

- APRMs decrease by 8%.
- Reactor total steam flow decreases by 1E6 lbm/hr.
- Main Generator output decreases by 70 MW.
- Core pressure drop decreases by 4 psid.
- Reactor recirculation loop "A" drive flow increases by 2,000 gpm.

Which one of the following is the cause of this transient?

a. Main steam line "A" SRV lifts and sticks partially open.

b. Reactor recirculation pump "B" shaft shear.

c. Reactor recirculation loop "A" flow instrumentation failure.

d. Jet pump failure in reactor recirculation loop "A".

QUESTION: 062 (1.00)

Entry into T-103, Secondary Containment Control, requires verification of the isolation of the Reactor Building and Refuel Floor Ventilation and the initiation of SBGTS for a Reactor Building Ventilation Exhaust radiation above the high alarm setpoint.

Which one of the following is the reason for these actions?

a. To maintain a negative Reactor Building-to-drywell differential pressure.

b. To maintain a negative Reactor Building-to-atmosphere differential pressure.

c. To rapidly reduce the airborne radioactivity in the Reactor Building for personnel access.

d. To direct the effluent to the vent stack to be monitored for offsite radiation releases.

QUESTION: 063 (1.00)

A reactor startup is in progress and the operator is withdrawing control rods to position 48 using the ROD NOTCH OVERRIDE switch when the following indications are received:

- "ROD DRIFT" alarm.
- "ROD OVERTRAVEL" alarm.
- No rod position is indicated.

Which one of the following will cause the above indications?

a. The rod has drifted beyond the last even numbered position and is still settling to position 48.

b. The operator provided a witndraw signal to the rod for an excessive period of time after reaching position 48.

c. The rod is uncoupled and its position in the core is unknown.

d. The Reactor Manual Control rod drive timer has malfunctioned.

QUESTION: 064 (1.00)

Which one of the following describes a plant condition that could cause airborne plant release rates to exceed the Technical Specification limits?

a. Continuing to operate the SJAE when reactor power is less than 5% due to insufficient dilution flow.

b. Continuing to operate the SJAE when reactor power is less than 5% due to insufficient flow to cool the offgas adsorbers.

c. Starting the mechanical vacuum pump when reactor power is greater than 5% because it discharges directly to the offgas stack.

d. Starting the mechanical vacuum pump when reactor power is greater than 5% because it discharges directly to the Turbine Building Ventilation discharge.

QUESTION: 065 (1.00)

Which one of the following is the reason that ON-113, Loss of RBCCW, cautions the operator to maintain CRD seal purge to the reactor recirc pumps in service during a loss of RBCCW?

a. To reduce the chance of radioactive reactor coolant from leaking into the RBCCW pump coolers

b. To prevent hot reactor coolant from overheating the pump shaft and the lower motor bearing

c. To prevent recirc pump bearing failure due to thermal shock when RBCCW is restored

d. To restrict the flow of hot reactor coolant into the recirc pump seals

QUESTION: 066 (1.00)

ON-102 "Air Ejector Discharge High Radiation" directs the operator to maintain air ejector discharge radiation levels below 700 mr/hr.

Which one of the following alarms would indicate that the air ejector discharge radiation level has reached or exceeded 700 mr/hr?

a. AIR EJECTOR DISCHARGE HI RADIATION

b. AIR EJECTOR DISCHARGE HI HI RADIATION

c. VENT EXH STACK RAD MONITOR HI

d. VENT EXH STACK RAD MONITOR HI HI

QUESTION: 067 (1.00)

Which one of the following is the reason for initiating a DC load shed as soon as possible per SE-11 (Sheet 5), "Loss of Off-Site Power with No Diesel Generators Available"?

a. To prevent spurious operation of ECCS components during restoration of AC power

b. To extend the amount of time that the DC loads required for adequate core cooling are available

c. To preserve the DC power that is necessary to initiate a Diesel Generator start to rated rpm and voltage

d. To reduce the buildup of hydrogen in the battery rooms to prevent a hydrogen combustion hazard

QUESTION: 068 (1.00)

Unit 2 Control Room has been evacuated and the unit is being cooled down from the HPCI ASD panel. The operator observes the following parameters:

TIME	PRESSURE	REACTOR LEVEL (LI 2-2-3-112)
2245	800 psig	-19 inches
2300	700 psig	-18 inches
2315	600 psig	-17 inches
2330	500 psig	-16 inches
2345	400 psig	-15 inches

Which one of the following describes the trend of ACTUAL reactor water level over the past hour and the current water level? (SE-10 "Shutdown Outside Control Room" Attachment 9, Figure 1, Actual Rx Level as a Function of Rx Press and Indicated Level, is attached.)

a. Level has been decreasing and is currently less than zero inches.

b. Level has been increasing and is currently less than zero inches.

c. Level has been decreasing and is currently between 0 and +40 inches.

d. Level has been increasing and is currently between 0 and +40 inches.

QUESTION: 069 (1.00)

Unit 2 was scrammed from 100% power, the Control Room was evacuated at 2115, and the unit is being cooled down from the HPCI ASD panel. The operator has recorded the following information:

TIME	PRESSURE
2130	Peak reactor pressure of 1100 psig occurred.
2215	950 psig
2230	800 psig
2245	650 psig
2300	500 psig

Which one of the following is the LOWEST reactor pressure to which the reactor can be depressurized at 2315 without exceeding the Technical Specification cooldown rate limit? (SE-10 "Shutdown Outside Control Room" Attachment 10, Table 1, Steam Saturation Temperatures, is attached.)

a. 450 psig

b. 410 psig

c. 370 psig

d. 350 psig

QUESTION: 070 (1.00)

T-102 "Primary Containment Control" provides the following caution:

Operation of HPCI or RCIC with Torus suction AND Torus temperature above 190 deg. F may result in equipment damage.

Which one of the following is the basis for this caution?

a. Insufficient pump cooling may result in impeller binding and mechanical seal failure.

b. Turbine exhaust check valve may chatter resulting in valve failure or steam hammer.

c. Inadequate NPSH may result in pump cavitation causing excessive vibration and pitting damage of pump components.

d. The torus is no longer capable of condensing all of the turbine exhaust resulting in increased torus pressures.

QUESTION: 071 (1.00)

Select the choice below that completes the following statements.

A LOCA is in progress, causing highly elevated secondary containment temperatures. The indicated reactor vessel water level will be ________.

a. lower than actual due to boiling in the reference legs.

b. higher than actual due to a rupture failure of the diaphragm in the dp cell.

c. inaccurate because boiling may occur in both the variable and reference legs.

d. inaccurate because the dp cell may experience steam binding

QUESTION: 072 (1.00)

An ATWS is in progress on Unit 2. After the insertion of the initial scram signal, the operator observes the following:

- CRD pumps are tripped.
- RPV pressure is 350 psig.
- About half of the blue scram lights are NOT lit.
- Control Rod Drive Scram Solenoid Group 1 and 3 in RPS Channel B are lit.
- SCRAM DISCH VOL HI WATER LEVEL TRIP 50 GAL alarm has been received.
- All ACCUMULATOR TROUBLE alarm lights on the full core display are lit.

Which one of the following alternate rod insertion methods is MOST likely to successfully insert any withdrawn rods?

a. Removal of the group scram solenoid fuses per T-213 "Scram Solenoid Deenergization".

b. Resetting the scram, draining the scram discharge volume, and manually scramming per T-216 "Rod Insertion by Manual Scram or Individual Scram Test Switches".

c. Placing the individual scram test switches for the withdrawn rods to the full down position per T-213 "Scram Solenoid Deenergization".

d. Manual venting of the CRD withdrawal lines per T-215 "Control Rod Insertion by Withdraw Line Venting".

QUESTION: 073 (1.00)

A LOCA is in progress on Unit 2. The plant has been successfully scrammed and plant conditions are as follows:

- RPV water level:
  Shutdown Range (LI-86) indicates +10 inches.
  Narrow Range (LI-94A,B,&C) indicate +5 to +8 inches.
  Wide Range (LI-85A&B) indicate -10 inches.
  Fuel Zone instruments indicate -25 inches.
- Drywell temperature (TI-2-501):
  Point 126 indicates 270 deg. F.
  Point 127 indicates 267 deg. F.
- Reactor Building temperature (TR-2-13-139):
  Point 22 indicates 155 deg. F.
- RPV pressure is 200 psig.

Which one of the following instruments is NOT AVAILABLE for RPV level indication?

a. Shutdown Range

b. Wide Range

c. Narrow Range

d. Fuel Zone

QUESTION: 074 (1.00)

Which one of the following reactor water level anomalies can ONLY be detected or observed during a rapid RPV depressurization below 450 psig?

a. Boiling in the variable leg.

b. Boiling in the reference leg.

c. Degassing in the reference leg.

d. Hydrogen buildup in the condensing chamber.

QUESTION: 075 (1.00)

An ATWS is in progress on Unit 2. Plant conditions are as follows:

- MSIVs are open.
- Main generator load is 300 MWe.
- Reactor recirc pumps are at 25% speed.
- APRMs indicate 25%.

Which one of the following is the reason that T-101, Step RC/Q-15, directs the operator to trip the reactor recirc pumps at least 10 seconds apart?

a. To ensure RPV water level swell is not enough to reach the reactor feed pump and main turbine trip setpoint.

b. To ensure RPV water level swell is not enough to undesireably flood the main steam lines and result in carryover to any operating turbines.

c. To ensure RPV water level shrinkage is not enough to undesireably initiate HPCI or RCIC.

d. To ensure RPV water level shrinkage is not enough to cause an overspeed trip condition on any running reactor feed pumps.

QUESTION: 076 (1.00)

An event has occurred on Unit 2 that has resulted in the lowering of Torus level. Plant conditions are as follows:

- Torus level is 10.0 feet and steady.
- Torus temperature is 135 deg. F.
- RPV pressure is 1000 psig.

Which one of the following describes the adverse consequences of plant equipment operation at this Torus level?

a. Operation of RCIC will increase Torus pressure causing RCIC to trip on high turbine exhaust pressure.

b. Operation of HPCI will increase Torus pressure and threaten containment integrity.

c. Opening SRVs will rapidly increase Torus pressure.

d. A LOCA will result in overpressurizing the containment.

QUESTION: 077 (1.00)

T-117 "Level Power Control" is being executed on Unit 2. Step LQ-20 directs the operator to restore and maintain RPV level above -172 inches with Condensate/Feedwater, CRD, RCIC, HPCI, and LPCI, regardless of ECCS suction requirements.

Which one of the following is the reason that Step LQ-20 EXCLUDES the use of Core Spray to recover RPV level?

a. To ensure one ECCS injection source is protected from damage due to vortexing or insufficient NPSH.

b. Because there are no heat exchangers in the Core Spray system to remove core decay heat.

c. To minimize the injection of cold water directly into the core which could add significant positive reactivity.

d. To ensure the natural circulation that has been established in the RPV is not interrupted.

QUESTION: 078 (1.00)

T-102 "Primary Containment Control" is being executed. Step PC/P-9 indicates:

IF Drywell pressure drops below 2 psig,
THEN Terminate drywell sprays.

Which one of the following is the basis for Step PC/P-9?

a. To prevent cycling the Drywell-to-Torus vacuum breakers.

b. To prevent opening the Reactor Building-to-Torus vacuum breakers.

c. To prevent developing an excessive differential pressure between the Drywell and the Torus.

d. To prevent developing an excessive differential between the water level inside the downcomer and the Torus.

QUESTION: 079 (1.00)

In T-101-2 "RPV Control", if SRVs are cycling, the operator is directed to manually open SRVs and control RPV pressure between 960 and 1060 psig.

Which one of the following is the reason for establishing the minimum RPV pressure at 960 psig?

a. To ensure the turbine bypass valves do not have the opportunity to stick closed

b. To prevent MSIVs from closing on low main steam line pressure

c. To minimize the amount of steam that is sent to the suppression pool

d. To prevent excessive loss of reactor coolant inventory

QUESTION: 080 (1.00)

An accident is in progress on Unit 2. Torus sprays have been initiated but CANNOT maintain torus pressure below 9 psig.

Which one of the following is the reason for initiating drywell sprays when torus pressure cannot be maintained below 9 psig?

a. To prevent chugging and eventual fatigue failure at the junction of the downcomers and the vent header

b. To prevent excessive cyclic stresses on the SRV tailpipes

c. To prevent exceeding the torus pressure limit if a DBA LOCA occurs

d. To prevent exceeding the structural design limits of the torus if an emergency depressurization is required

QUESTION: 081 (1.00)

Following a reactor scram, T-100 "Scram", Step S-10, directs the following operator action:

VERIFY GEN LOCKOUT.

Which one of the following is the method that the operator will use to perform step S-10?

a. Check that the main generator disconnects indicate open.

b. Check that the 13 KV system has transferred from the unit to the startup transformers.

c. Check that the emergency transformers have transferred to the Startup Bus.

d. Check that the main generator output and field breakers indicate open.

QUESTION: 082 (1.00)

Unit 3 is being maintained in Cold Shutdown with the RPV head bolts tensioned per GP-12 "Core Cooling". One loop of RHR is in shutdown cooling with one pump operating at a flow rate of 8,000 gpm.

Which one of the following sets of indications should be used by the operator to ensure that shutdown cooling flow through the core is sufficient to prevent thermal stratification and remove decay heat? (Refer to GP-12, Figure 3, Vessel Level vs. Flow Through One Recirc Loop.)

	LI-86 Shutdown Range	LI-94 Narrow Range
a.	30 inches	30 inches
b.	35 inches	35 inches
c.	35 inches	38 inches
d.	38 inches	35 inches

QUESTION: 083 (1.00)

Step SC/L-1 of T-103 "Secondary Containment Control" directs the following action:

MONITOR AND CONTROL SECONDARY CONTAINMENT WATER LEVELS.

Which one of the following is an appropriate operator action that would be taken to accomplish this step?

a. Isolate any leaking ECCS system even though it is currently the only system available to maintain RPV level.

b. Direct maintenance to immediately install sandbags around the outside of the room doors.

c. Plot the rate of increase in the sump water levels to determine whether sump capacity is sufficient.

d. Start all available sump pumps and operate them to remove water from the sumps.

QUESTION: 084 (1.00)

Following a LOCA, which one of the following will result in the most rapid production of hydrogen in the drywell?

a. High temperature steam/nitric acid corrosion of stainless steel components

b. Radiolytic decomposition of water by fission product gammas in the torus

c. Boric acid corrosion of various metal components when SBLC is injected

d. Zirc water reaction of the fuel cladding when the core is uncovered

QUESTION: 085 (1.00)

A reactor operator has been off-shift for six months.

Which one of the following describes the MINIMUM actions which must be taken before the operator may return to the normal shift rotation?

a. Work the first 6 hours of a shift for 7 days under the supervision of a qualified RO to include a complete plant tour.

b. Work five 12-hour shifts under the supervision of a qualified RO to include participation in shift turnovers and pre- and post- ALARA job briefings.

c. Work 40 hours as an extra operator on-shift under the supervision of a qualified SRO to include pre- and post- job briefings.

d. Work five 8-hour shifts under the supervision of a qualified RO to include a complete plant tour and participation in shift turnovers.

QUESTION: 086 (1.00)

An on-coming Reactor Operator has worked the following control room shift schedule during a refueling outage.

Day:	1	2	3	4	5	6	7	8	9	10	11	12	13	14
Hrs:	12	12	12	12	12	0	0	0	0	10	10	10	14	?

Which one of the following is the MAXIMUM number of hours the operator may work on day 14 without obtaining special authorization from the Plant Manager?

a. 6 hours

b. 8 hours

c. 10 hours

d. 14 hours

QUESTION: 087 (1.00)

An operator is conducting an independent verification of a locked manual valve. The Locked Valve List requires the valve to be throttled 15 turns open (40% open). When the operator observes the valve he finds the following conditions:

- The valve locking device is painted orange.
- The valve is fully open.

Which one of the following is the action that the operator is expected to take per administrative procedures?

a. Unlock and close the valve, then open it 15 turns and lock it in the required position, then note the situation in the comments section of the valve list.

b. Note the current position of the valve on the Locked Valve List and complete the verification of the remainder of the valve lineup.

c. Report to SRO on-duty that the valve is in the correct position and the Locked Valve List is incorrect; note the position on the Locked Valve List and complete the lineup verification.

d. Leave the valve in its current position and report to the SRO on-duty that the valve position, locking device color coding, and Locked Valve List are in disagreement.

QUESTION: 088 (1.00)

An auxiliary equipment operator has just reported to the reactor operator that he has initiated an (Equipment Trouble Tag) ETT for a plant component.

Which one of the following actions must be completed in accordance with A-C-26 "Maintenance Work Process"?

a. The auxiliary operator will initiate a work order for routing through shift management and work planning.

b. The auxiliary operator will process an Action Request (A/R) for the component tagged with the ETT.

c. The reactor operator will notify the Fix-It-Now (FIN) team leader to review the ETT to determine if FIN can make repairs.

d. The reactor operator will complete and place an ETT sticker on the associated control room panel for that equipment.

QUESTION: 089 (1.00)

A surveillance test is in progress.

Which one of the following conditions would require an operator to STOP a surveillance TESTING procedure IMMEDIATELY in accordance with A-C-43 "Surveillance Testing Program"?

a. Testing has placed the system outside of Technical Specification limits but the system can be restored to specifications within 1 hour after the test is completed.

b. A system parameter has been observed to be outside of the normal band, but the procedure step provides guidance for adjustment of the abnormal system parameter.

c. The procedure does not have a blank for the recording of a specified parameter required for satisfactory performance of the system being tested.

d. The procedure is taking much longer to perform than expected as explained in the pre-test briefing of the personnel involved in the test.

QUESTION: 090 (1.00)

A reactor startup is in progress on Unit 3. The reactor is critical and the operator has just observed that the reactor period is 80 seconds and slowly increasing.

Which one of the following is the operator action required by GP-2 "Reactor Startup"?

a. Insert control rods to increase reactor period to 100 seconds.

b. Withdraw control rods to decrease reactor period to 50 seconds.

c. Wait until reactor period increases to greater than 100 seconds to withdraw control rods.

d. Wait until SRM count rate stabilizes before withdrawing additional control rods.

QUESTION: 091 (1.00)

Which one of the following describes a radiologically controlled area that is posted as a YELLOW ZONE?

a. A buffer area that is established around any accessible Red Zone.

b. An area that is established around any open bag containing hot particles.

c. Any area outside of the restricted area but inside the site boundary for which PECo restricts access.

d. An area in which an individual will inhale greater than 12 DAC-hours in a 40 hour work-week.

QUESTION: 092 (1.00)

An operator observes that an area is posted as a VERY HIGH RADIATION AREA and HP reports it is a Level I area.

Which one of the following radiation dose rates would be expected inside this area?

a. Greater than or equal to 1 rem/hr but less than 10 rem/hr at 30 cm.

b. Greater than or equal to 10 rem/hr but less than 500 rem/hr at 30 cm.

c. Greater than or equal to 1 rad/hr but less than 10 rad/hr at 1 meter.

d. Greater than or equal to 10 rad/hr but less than 500 rad/hr at 1 meter.

QUESTION: 093 (1.00)

Which one of the following describes when the reactor operator assigned as the "operator at the controls" may leave the controls area and enter the area behind the control panels?

a. When an off-duty actively licensed reactor operator is available to momentarily observe the control panels without formally relieving the assigned operator.

b. To perform normal procedural steps as long as the opposite unit's reactor operator assigned as the "operator at the controls" remains in the control area.

c. With CRS permission to verify receipt of an alarm or take corrective action during an emergency condition even if no other operator is available to observe the control panels.

d. To ensure the timely completion of the shift logs for plant parameters of a time critical nature even if no other operator is available to observe the control panels.

QUESTION: 094 (1.00)

Which one of the following is designated to number a new operator aid request and make the entry in the Operator Aid Log in accordance with OM-C-9.2 "Operator Aids"?

a. The individual that is requesting the operator aid.

b. The Control Room Supervisor or the Shift Manager.

c. The Senior Manager of Operations.

d. The Operations Support Group.

QUESTION: 095 (1.00)

Select the choice below that completes the following statements in accordance with OM-P-7.6 "Fuses and Quality Parts".

An operator has just replaced a blown fuse in a circuit, and the circuit has been restored to normal. The operator must then provide the blown fuse to the _______ with a written description of the circumstances surrounding the failure.

a. Shift Technical Advisor

b. Electrical Maintenance Supervisor

c. Operations Support Group Manager

d. Instrument and Controls Supervisor

QUESTION: 096 (1.00)

Select the choice below that completes the following statements.

Fuel is being loaded on Unit 2. The white Rod Withdraw Permissive light on panel C05A will be ILLUMINATED if the refueling bridge is over the core and the ________.

a. main hoist is unloaded but NOT fully raised, and the auxiliary hoist is unloaded and fully raised

b. main hoist is unloaded and fully raised, and the auxiliary hoist is unloaded but NOT fully raised

c. main hoist is carrying a 500 pound load, and the auxiliary hoist is unloaded and fully raised

d. main hoist is unloaded and fully raised, and the auxiliary hoist is carrying a 450 pound load

QUESTION: 097 (1.00)

Select the choice below that completes the following statements.

Unit 2 is in a refueling outage and fuel is being loaded to the core. If the spent fuel pool and reactor cavity water level unexpectedly decrease by 1 foot and continue decreasing by 3 inches/min., the operations shift is required ________.

a. to immediately stop all fuel movement, leave the refuel bridge in its current condition, and evacuate the fuel floor to Turbine Building elevation 165 ft.

b. to immediately stop all fuel movement, de-energize the loaded hoists and cranes in their current condition, and initiate investigation into the level loss.

c. to lower any irradiated component suspended on the refuel bridge to its lowest position and evacuate the fuel floor to Reactor Building elevation 135 ft.

d. to lower any irradiated component suspended on a hoist or crane to the nearest underwater storage location and notify Health Physics.

QUESTION: 098 (1.00)

Following a transient, Unit 2 conditions are as follows:

- RPV pressure is 1000 psig and steady.
- CRS has determined that torus level can NOT be maintained above the Heat Capacity Level Limit curve.
- Drywell temperature and pressure are increasing, but have not reached TRIP entry values.

Which one of the following is the action that must be taken per the Trip Procedures?

a. Initiate torus and drywell sprays.

b. Initiate only drywell sprays.

c. Perform a normal RPV depressurization.

d. Perform an emergency blowdown.

QUESTION: 099 (1.00)

An abnormal event is in progress on Unit 3. The unit has been successfully scrammed and plant conditions are as follows:

- Torus temperature is 93 deg. F.
- Torus level is approaching the SRV Tailpipe Level Limit curve and is on the SAFE side of the curve by only 0.1 foot.
- RPV pressure is 1000 psig.

Which one of the following is the required action?

a. Increase torus level with the Condensate Transfer system, HPCI, or HPSW.

b. Decrease torus temperature by placing both loops of RHR in torus cooling.

c. Immediately initiate an emergency RPV blowdown with all operable SRVs.

d. Reduce reactor pressure with bypass valves not to exceed normal cooldown rates.

QUESTION: 100 (1.00)

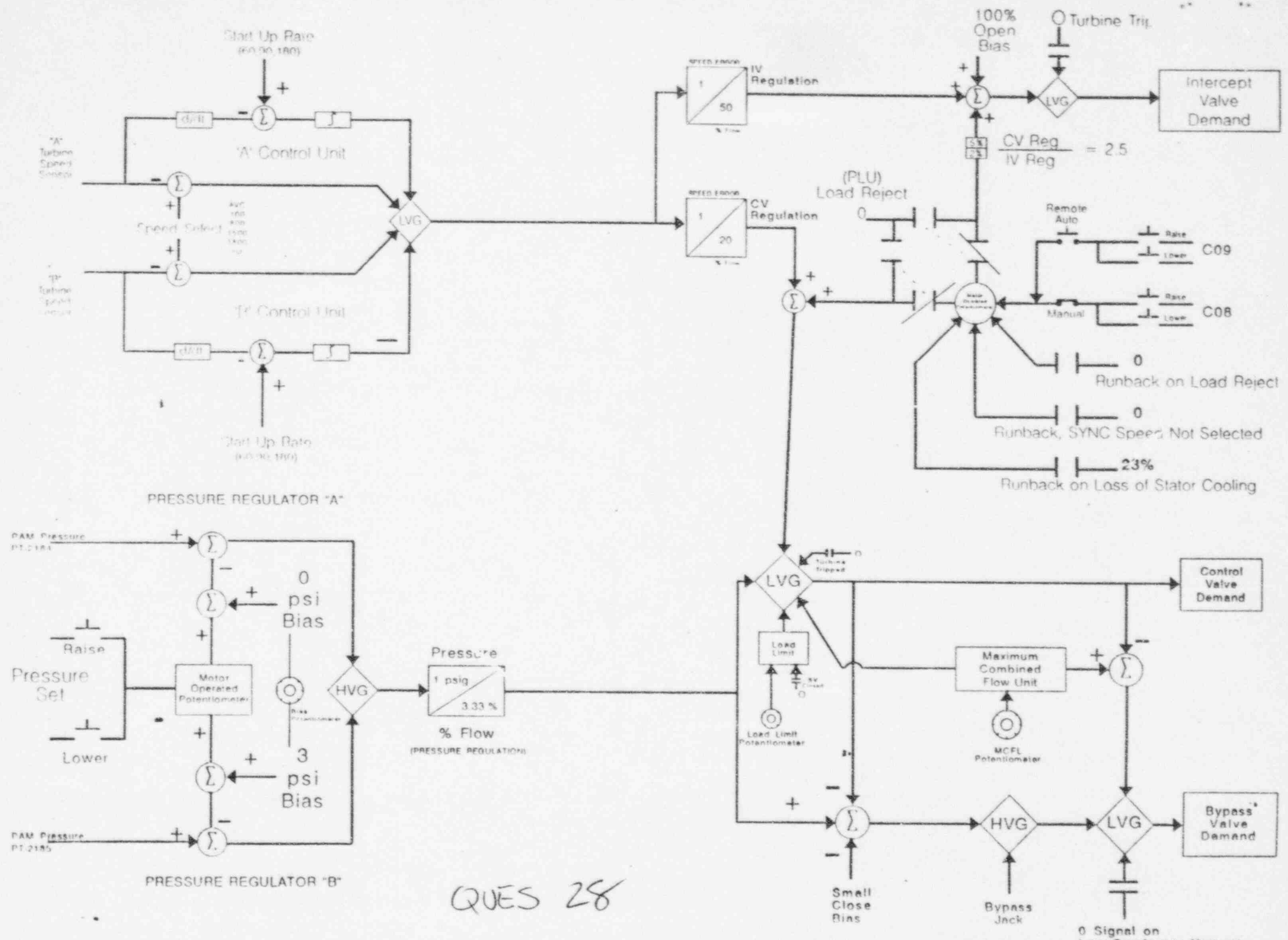
Following a LOCA and a partial ATWS on Unit 3, containment conditions are as follows:

- Torus pressure is 15 psig and slowly increasing.
- Torus temperature is 210 deg. F. and steady.
- Torus level is 12 feet and steady.

Which one of the following is the maximum flow rate for RHR pumps "A" and "B"?

	RHR Pump "A"	RHR Pump "B"
a.	10,000 gpm	0 gpm
b.	10,000 gpm	10,000 gpm
c.	10,000 gpm	9,700 gpm
d.	9,700 gpm	9,700 gpm

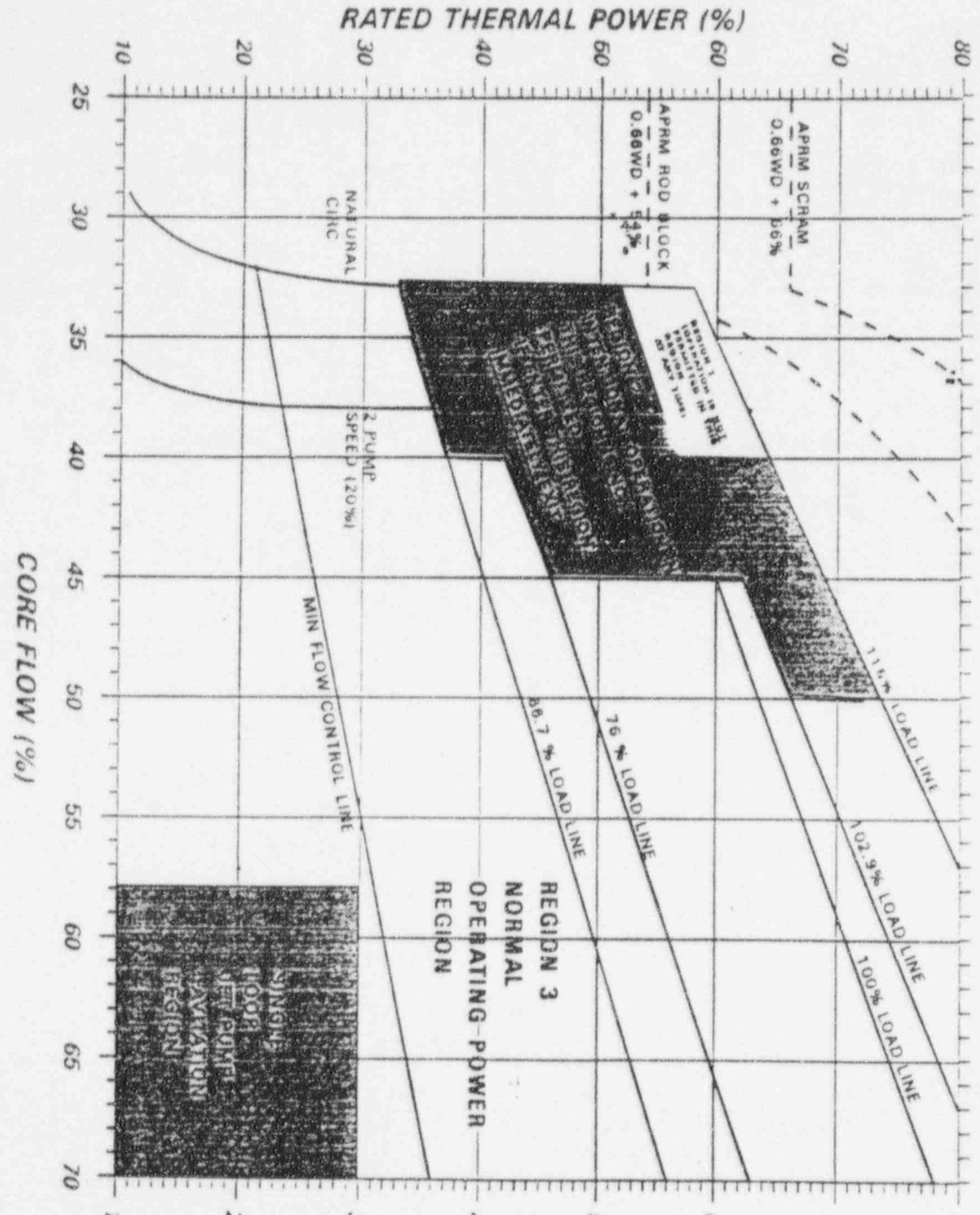
(********** END OF EXAMINATION **********)



Start Up Rate
'A' Control Unit
'B' Control Unit
Speed Select
LVG
Start Up Rate
PRESSURE REGULATOR "A"
IV Regulation
CV Regulation
100% Open Bias
Turbine Trip
Intercept Valve Demand
CV Reg / IV Reg = 2.5
(PLU) Load Reject
Remote Auto
Raise
Lower
C09
Manual
C08
0
Runback on Load Reject
0
Runback, SYNC Speed Not Selected
23%
Runback on Loss of Stator Cooling
0 psi Bias
Raise
Pressure Set
Lower
Motor Operated Potentiometer
HVG
Pressure
1 psig
3.33 %
% Flow
(PRESSURE REGULATION)
3 psi Bias
PRESSURE REGULATOR "B"
LVG
Turbine Tripped
Load Limit
Load Limit Potentiometer
Control Valve Demand
Maximum Combined Flow Unit
MCFL Potentiometer
Small Close Bias
HVG
Bypass Jack
LVG
Bypass Valve Demand
0 Signal on Low Condenser Vacuum (7" Hg VAC)

QUES 28

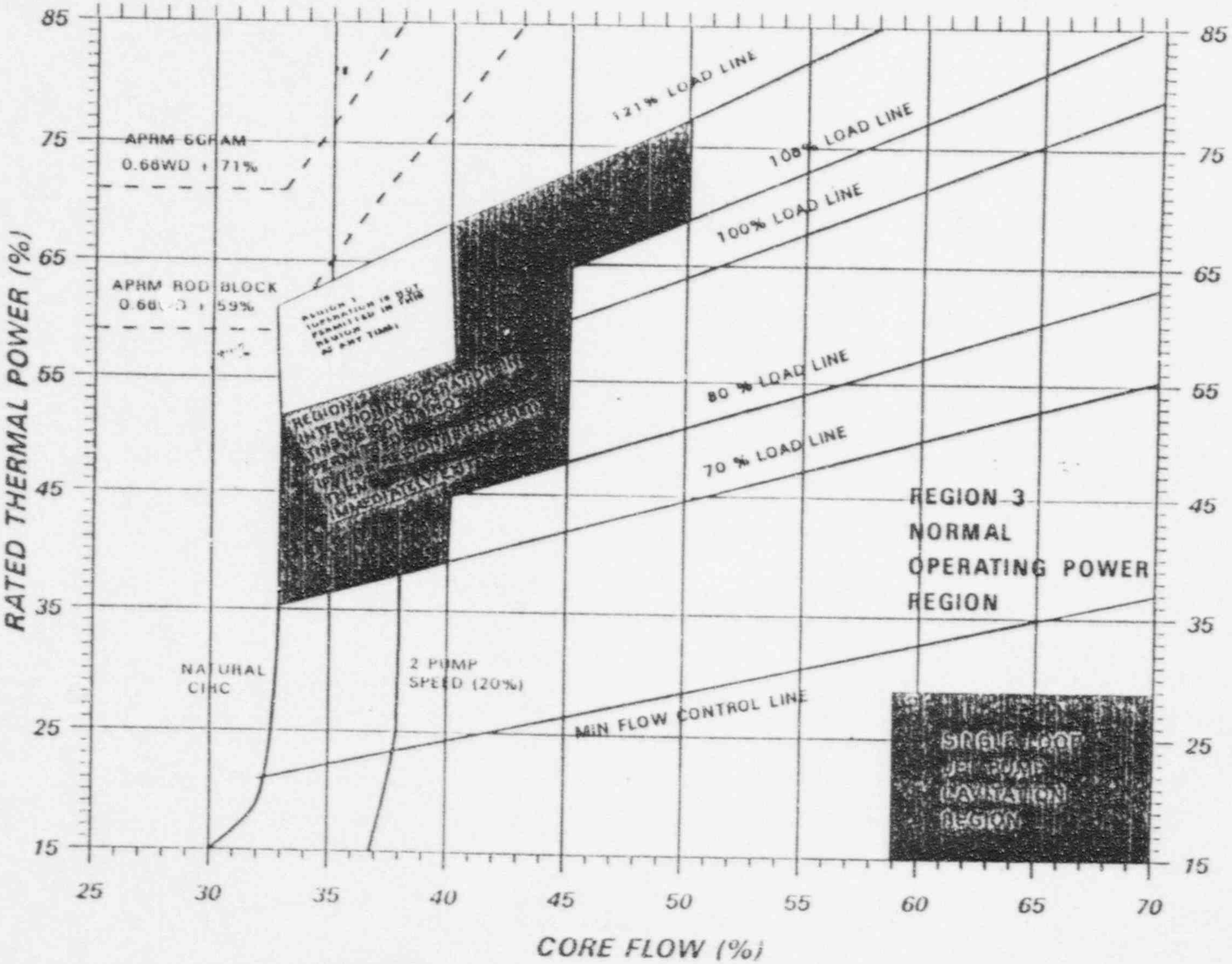
FIGURE A
UNIT 2 ONLY



QUES 46

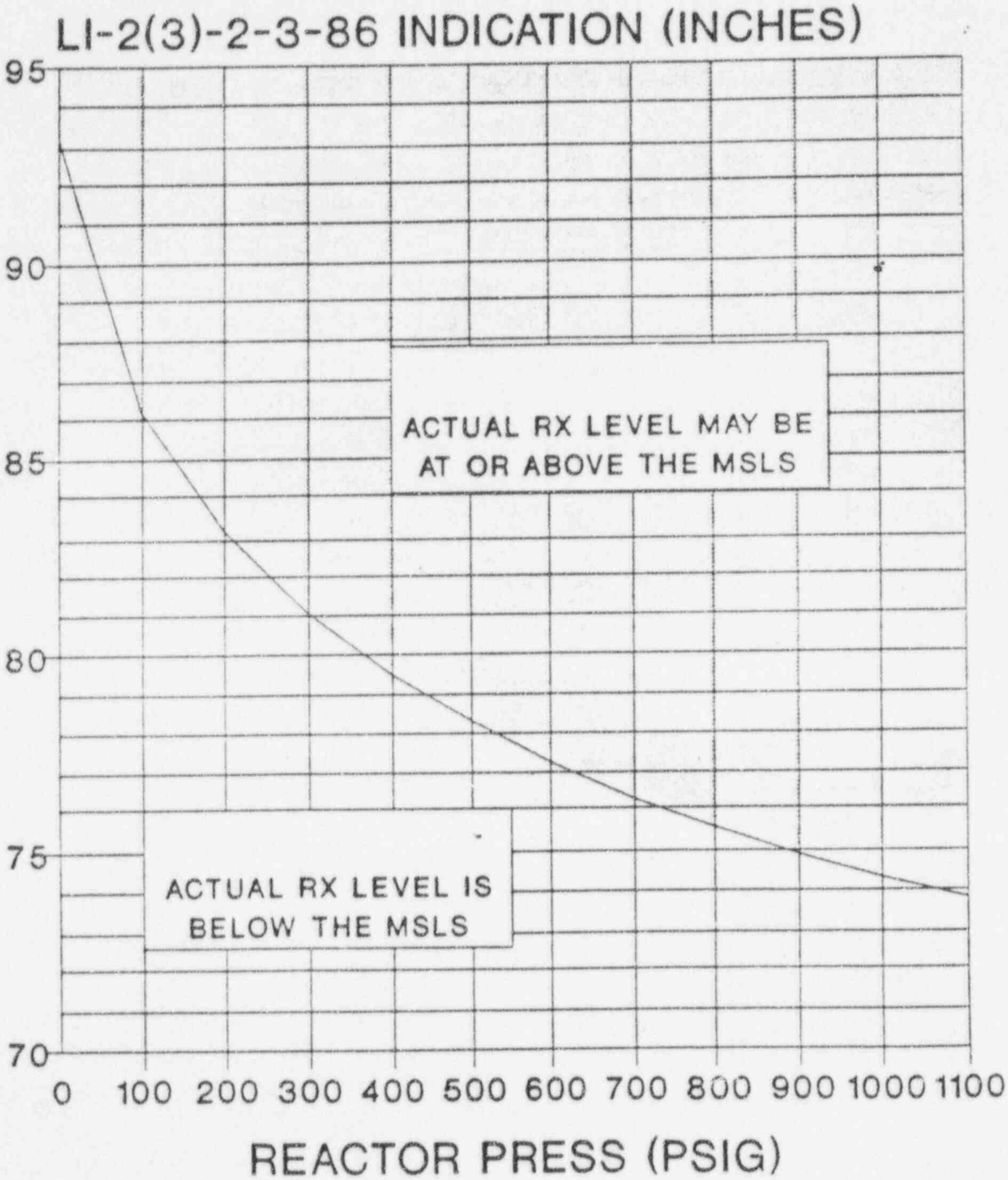
FIGURE B

UNIT 3 ONLY



QUES 46

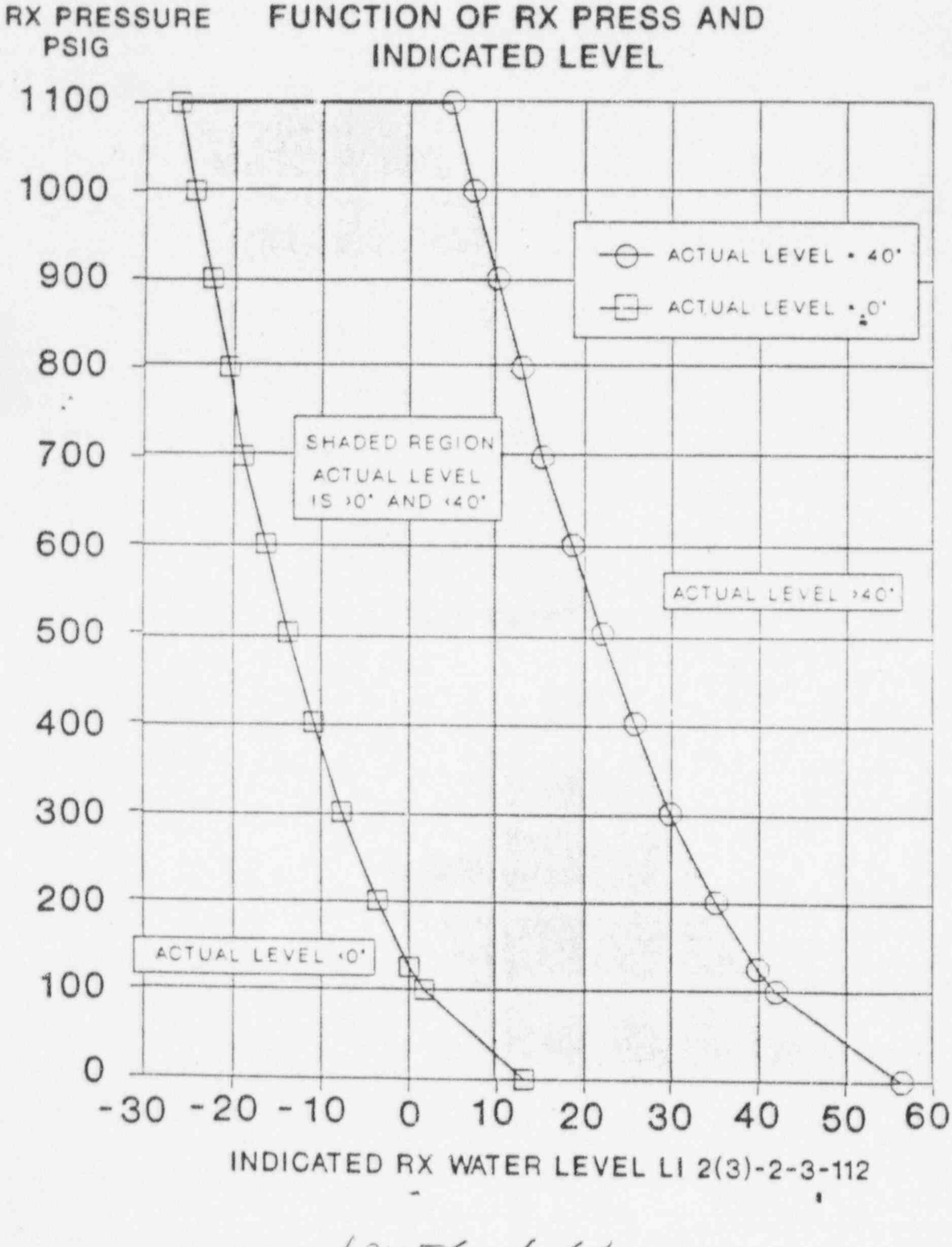
FIGURE 1



QUES 52

SE-10 Attachment 9
Figure 1

ACTUAL RX LEVEL AS A FUNCTION OF RX PRESS AND INDICATED LEVEL



QUES 68

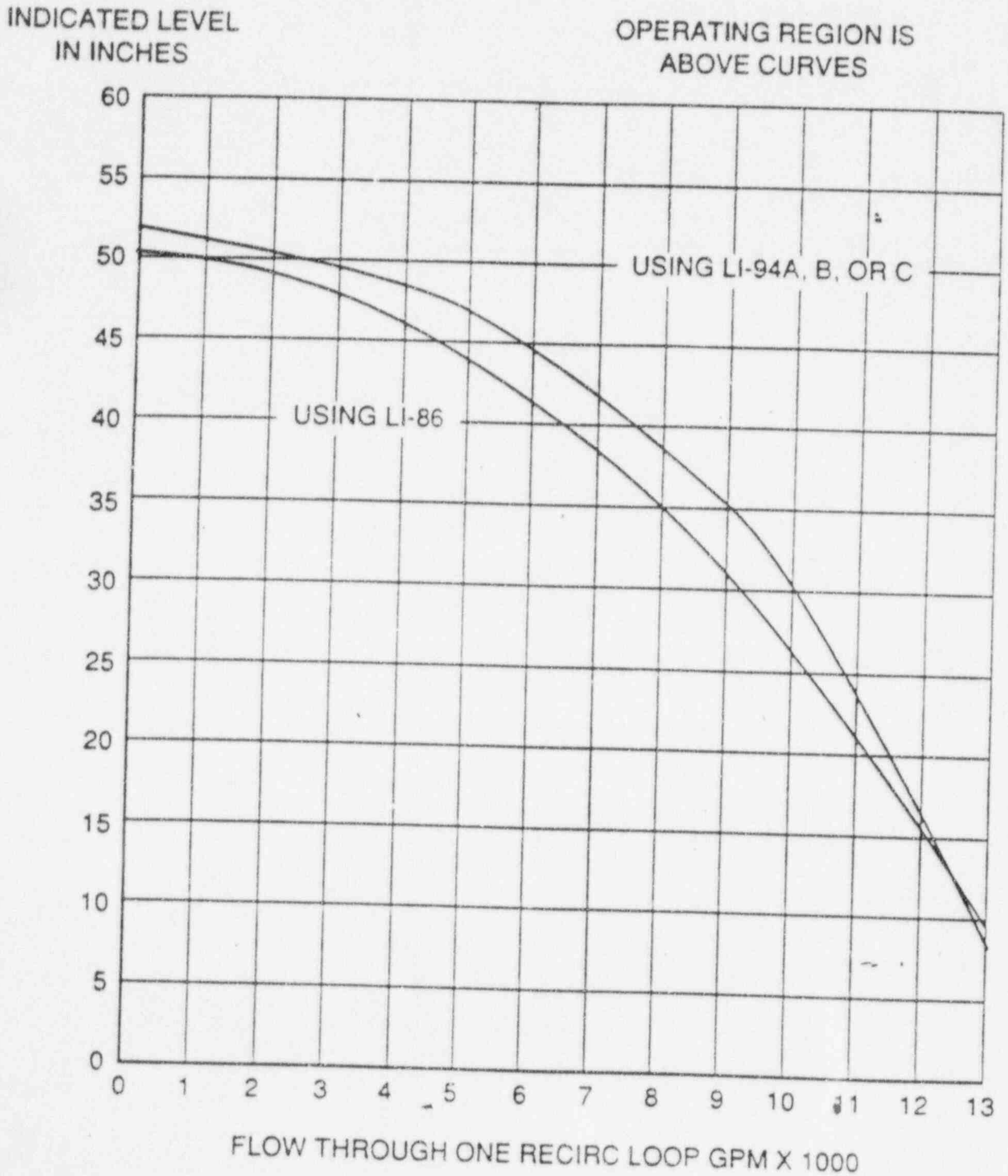
Table 1 - Steam Saturation Temperatures

Rx Press Psig	Saturation Temp deg-F	Rx Press Psig	Saturation Temp deg-F	Rx Press Psig	Saturation Temp deg-F
1100	558	600	489	100	338
1090	557	590	488	90	331
1080	556	580	485	80	324
1070	555	570	483	70	316
1060	554	560	481	60	307
1050	553	550	480	50	297
1040	551	540	478		
1030	550	530	476		
1020	549	520	474		
1010	548	510	472		
1000	547	500	470		
990	545	490	468		
980	544	480	466		
970	543	470	464		
960	541	460	462		
950	540	450	459		
940	539	440	457		
930	538	430	455		
920	536	420	453		
910	535	410	451		
900	534	400	448		
890	532	390	446		
880	531	380	443		
870	530	370	441		
860	528	360	438		
850	527	350	436		
840	525	340	433		
830	524	330	430		
820	523	320	428		
810	521	310	425		
800	520	300	422		
790	518	290	419		
780	517	280	416		
770	515	270	412		
760	514	260	409		
750	513	250	406		
740	511	240	402		
730	510	230	399		
720	508	220	395		
710	507	210	392		
700	505	200	388		
690	503	190	384		
680	502	180	379		
670	500	170	375		
660	499	160	370		
650	497	150	365		
640	496	140	360		
630	494	130	355		
620	492	120	350		
610	491	110	344		

QUES 69

FIGURE 3

Vessel Level Vs. Flow Through One Recirc Loop



QUES 82

ANSWER: 001 (1.00)

c.

REFERENCE:

Facility Question Bank No. 4578
LOT-0370

203000K301 [4.3/4.1]

203000A405 ..(KA's)

ANSWER: 002 (1.00)

d.

REFERENCE:

LOT-0380, SO 13.2.A-2
KA217000A401
(3.7/3.7)

217000A401 ..(KA's)

ANSWER: 003 (1.00)

a.

REFERENCE:

PLOT 0030, p. 25-26
Learning Objective 5.f

202001K408 [2.8/2.9]

202001K408 ..(KA's)

ANSWER: 004 (1.00)

b.

REFERENCE:

PLOT 0030-1, p. 28
Learning Objective 6.g

202001K502 [3.1/3.2]

202001K502 ..(KA's)

ANSWER: 005 (1.00)

d.

REFERENCE:

LOT 0070, system drawing
Learning objective 3.

201001A109 [2.9/2.8]

201001A109 ..(KA's)

ANSWER: 006 (1.00)

a.

REFERENCE:

Control Rod Hydraulic system, LOT 0070, p. 24.
Learning Objective 6.a
Facility Question Bank No. 3883

201001K101 [3.1/3.1]

201001K101 ..(KA's)

ANSWER: 007 (1.00)

a.

REFERENCE:

Reactor Core Isolation Cooling, LOT-380 p. 20-21.

217000K407 [3.6/3.6]

217000K407 ..(KA's)

ANSWER: 008 (1.00)

c.

REFERENCE:

Automatic Depressurization System, LOT 0330, p. 15, 16
Learning Objective 5.b

218000K501 [3.8/3.8]

218000K501 ..(KA's)

ANSWER: 009 (1.00)

c.

REFERENCE:

Automatic Depressurization System, LOT-0330 p. 8
Learning Objective 6.f

218000K606 [3.4/3.6]

218000K606 ..(KA's)

ANSWER: 010 (1.00)

c.

REFERENCE:

Residual Heat Removal, H-LOT 0370-1, p. 1,2

203000G009 [4.3/3.9]

203000G009 ..(KA's)

ANSWER: 011 (1.00)

b.

REFERENCE:

High Pressure Coolant Injection, LOT-0340, p. 10
Learning Objective 2c, 2d.

206000A217 [3.9/4.3]

206000A217 ..(KA's)

ANSWER: 012 (1.00)

d.

REFERENCE:

HPCI System Operating Procedure, SO 23.1, p. 2
LOT-0340, Learning Objective 5.k

206000G010 [3.9/3.8]

206000G010 ..(KA's)

ANSWER: 013 (1.00)

a. C c

REFERENCE:

P&IDs, E-26
LOT 0340, Learning Objective 3.a

206000K201 [3.2/3.3]

206000K201 ..(KA's)

ANSWER: 014 (1.00)

d.

REFERENCE:

Starting the first recirculation pump, SO 2.A.a.A-2
PLOT-0030-1, pg 30, Learning Objective 5.i

202002A401 [3.3/3.1]

202002A401 ..(KA's)

ANSWER: 015 (1.00)

b.

REFERENCE:

modified PB exam bank # 3778
ON-115, Rev 5, pg 2
LOT 0450, pg 8, Learning Objective 2a

288000A301 [3.8/3.8]

288000A301 ..(KA's)

ANSWER: 016 (1.00)

a.

REFERENCE:

LOT-0520, pg 20
Learning Objective 2.b

256000K102 [3.3/3.3]

256000K102 ..(KA's)

ANSWER: 017 (1.00)

c.

REFERENCE:

Condensate system, LOT-0520, p. 21
Learning Objective 7.f

256000K606 [3.3/3.3]

256000K606 ..(KA's)

ANSWER: 018 (1.00)

b.

REFERENCE:

Startup of Second or Third RFP, SO 6C.1.C-2, p. 5
Facility Exam Bank question 1659

259001G010 [3.2/3.3]

259001G010 ..(KA's)

ANSWER: 019 (1.00)

a.

REFERENCE:

PBAPS LOT-0530, pg 15
PBAPS LOT-1610, pg 4

From March, 1995 Examination

259001A204 [3.3/3.4]

259001A204 ..(KA's)

ANSWER: 020 (1.00)

b.

REFERENCE:

T-LOT-0670-21

From August 1994 Exam

264000A404 [3.7/3.7]

264000A404 ..(KA's)

ANSWER: 021 (1.00)

b.

REFERENCE:

Modified Examination Bank Q. 2771, E-1, E-188, SO 54.7.B
LOT-0660, pg 26, Figures 1 and 2, Learning Objective 4, 7, and 11
NOTE: No LPCI initiation signal exists so there is no start signal to the RHR or HPSW pumps.

262001A302 [3.2/3.3]

262001A302 ..(KA's)

ANSWER: 022 (1.00)

c.

REFERENCE:

SO 52A.1.A

264000A401 [3.3/3.4]

264000A401 ..(KA's)

ANSWER: 023 (1.00)

b.

REFERENCE:

Reactor Water Cleanup System, PLOT-0110 p. 7
Learning Objective 1.a

204000A304 [3.4/3.5]

204000G004 ..(KA's)

ANSWER: 024 (1.00)

a.

REFERENCE:

Reactor Water Cleanup, PLOT-0110, p. 11
Learning Objective 4.d

204000K404 [3.5/3.6]

204000K404 ..(KA's)

ANSWER: 025 (1.00)

a.

REFERENCE:

Off-Gas system, LOT-0510, p. 10
Learning Objective 1.b

271000K101 [3.1/3.1]

271000K101 ..(KA's)

ANSWER: 026 (1.00)

b.

REFERENCE:

Off-Gas system, LOT-510, pg 10, 17, 19
Learning Objective 6.c, 4.o

271000A211 [2.8/2.9]

271000A211 ..(KA's)

ANSWER: 027 (1.00)

d.

REFERENCE:

Feedwater Control system, LOT-0550, p. 37 and 40
Learning Objective 4.j
Facility Question Bank No. 2581
259002A401 [3.8/3.6]

259002A401 ..(KA's)

ANSWER: 028 (1.00)

c.

REFERENCE:

T-PLOT-0590-6
Electro-hydraulic Control Logic, p. 10

241000A101 [3.9/3.8]

241000A101 ..(KA's)

ANSWER: 029 (1.00)

b.

REFERENCE:

Significantly modified PB Bank #3885
LOT 0180, rev 008, pg 11 - 17, Learning Objective 5.b

223002A302 [3.5/3.5]

223002A302 ..(KA's)

ANSWER: 030 (1.00)

b.

REFERENCE:

Feedwater control system, LOT-0550, p. 22
Learning Objective 5.b

259002K603 [3.1/3.1]

259002K603 ..(KA's)

ANSWER: 031 (1.00)

b.

REFERENCE:

Rod Block Monitor, LOT-0280, p. 12
T-LOT-0280-6
Learning Objective 5.c

215002K403 [2.9/3.0]

215002K403 ..(KA's)

ANSWER: 032 (1.00)

c.

REFERENCE:

Rod Block Monitor, LOT-0280, p. 7
Learning Objective 2.b

215002K102 [3.1/3.2]

215002K102 ..(KA's)

ANSWER: 033 (1.00)

b.

REFERENCE:

Reactor Protective System, LOT-0300, p. 23
Learning Objective 2.1

212000K114 [3.6/3.7]

212000K114 ..(KA's)

ANSWER: 034 (1.00)

a.

REFERENCE:

Reactor Protective System, LOT-0300, p. 28
Learning Objective 5.j

211000K410 [3.3/3.6]

212000K410 ..(KA's)

ANSWER: 035 (1.00)

c.

REFERENCE:

Reactor Startup, GP-2, p. 47

215003A401 [3.3/3.3]

215003A401 ..(KA's)

ANSWER: 036 (1.00)

d.

REFERENCE:

Average Power Range Monitor, LOT-0270, p. 10
Intermediate Range Monitor, LOT-0250, p. 17
Learning Objective 5.b

215005A104 [4.1/4.1]

215005A104 ..(KA's)

ANSWER: 037 (1.00)

c.

REFERENCE:

Source Range Monitor, LOT-0240, p. 16
Intermediate Range Monitor, LOT-0250, p. 16

Facility Question Bank No. 3879

215003K602 [3.6/3.8]

215003K602 ..(KA's)

ANSWER: 038 (1.00)

d.

REFERENCE:

LOT-0210 pg 13

From March, 1995 Exam, minor modifications

261000K301 [3.6/3.3]

261000K301 ..(KA's)

ANSWER: 039 (1.00)

c. a d

REFERENCE:

Main Steam and Pressure Relief System, LOT-0120, p. 20
Learning Objective 3.a

239001K201 [3.2/3.3]

239001K201 ..(KA's)

ANSWER: 040 (1.00)

d.

REFERENCE:

Main Steam and Pressure Relief, LOT-0120, p. 15
Learning Objective 2.z

239001K403 [3.2/3.3]

239001K403 ..(KA's)

ANSWER: 041 (1.00)

a.

REFERENCE:

LOT-0370, pg. 35
From August, 1994 Exam

226001A401 [3.5/3.4]

226001A401 ..(KA's)

ANSWER: 042 (1.00)

c.

REFERENCE:

LOT 0720, pg 16, 25, 38, 42; Learning Objective 2.a

272000K402 [3.7/4.1]

272000K402 ..(KA's)

ANSWER: 043 (1.00)

b.

REFERENCE:

DC Distribution, LOT-0690, p. 7.
Learning Objective 2.b
Facility Question Bank No. 2626

263000K102 [3.2/3.3]

263000K102 ..(KA's)

ANSWER: 044 (1.00)

a.

REFERENCE:

Fire Protection System, LOT-0685, p. 16
Learning Objective 5.b

286000K403 [3.3/3.4]

286000K403 ..(KA's)

ANSWER: 045 (1.00)

c.

REFERENCE:

OT-106, Rev 13, pg 1
LOT-1540, Learning Objective 2
modified PB exam bank # 2544
295002G010 [3.8/3.7]

295002G010 ..(KA's)

ANSWER: 046 (1.00)

d.

REFERENCE:

OT-112, Rev 16, pg 1
LOT-1540, Learning Objective 1, 2, and 3
modified pb exam bank #2547

NOTE: Include OT-112 Figures A and B in exam.

295001G010 [3.8/3.7]

295001G010 ..(KA's)

ANSWER: 047 (1.00)

d.

REFERENCE:

OT-111, Rev 2, pg 1
LOT-1540, Learning Objective 2 and 3

241000G014 [3.6/3.5]

241000G014 ..(KA's)

ANSWER: 048 (1.00)

a.

REFERENCE:

OT-114, Rev 2, pg 1
OT-114 Bases

259002K603 [3.1/3.1]

259002K603 ..(KA's)

ANSWER: 049 (1.00)

b.

REFERENCE:

OT-102-3, Rev 0, pg 1
LOT-1540, Learning Objective 2

295007A105 [3.7/3.8]

295007A105 ..(KA's)

ANSWER: 050 (1.00)

c.

REFERENCE:

OT-101 Bases, Rev 8, pg 5
LOT-1540, Learning Objective 4 and 5 (Although SRO objectives, this function is performed by the RO, and as such the caution needs to be well understood by the RO.)
PB exam bank #4115

295010G006 [3.8/3.9]

295010G006 ..(KA's)

ANSWER: 051 (1.00)

b.

REFERENCE:

Modified PB Exam Bank # 4306

295007K201 [3.5/3.7]

295007K201 ..(KA's)

ANSWER: 052 (1.00)

c.

REFERENCE:

OT-110, Rev 2, Figure 1, pg 5
LOT-1540, Learning Objective 3

NOTE: Include figure 1 in exam.

295008A201 [3.9/3.9]

295008A201 ..(KA's)

ANSWER: 053 (1.00)

d.

REFERENCE:

OT-110 Bases, Rev 2, pg 7
LOT-1540, Learning Objective 4 and 5

295008K204 [3.1/3.3]

295008K204 ..(KA's)

ANSWER: 054 (1.00)

b. + a

REFERENCE:

OT-112, Rev 16, pg 1
OT-112 Bases, Rev 15

295001A106 [3.3/3.4]

295001A106 ..(KA's)

ANSWER: 055 (1.00)

d.

REFERENCE:

Modified PB Bank #4587
LOT-0690, Learning Objective 3.a (related objective)

295004K203 [3.3/3.3]

295004K203 ..(KA's)

ANSWER: 056 (1.00)

a.

REFERENCE:

ON-112-2, Rev 0, pg 1
LOT-1550, Learning Objective 1
PB exam bank #1698, modified for different answers and distractors.
262002K105 [2.7/2.9]

262002K105 ..(KA's)

ANSWER: 057 (1.00)

d.

REFERENCE:

ON-107, Rev 4, pg 1 and 2
LOT-1550, Learning Objective 2
LOT-0070, pg 8, Learning Objective 4.k

295022A202 [3.3/3.4]

295022A202 ..(KA's)

ANSWER: 058 (1.00)

d.

REFERENCE:

OT-110, Rev 2, Figure 1, pg 5
LOT-1540, Learning Objective 3

NOTE: Include figure 1 in exam.

295008A201 [3.9/3.9]

295008A201 ..(KA's)

ANSWER: 053 (1.00)

d.

REFERENCE:

OT-110 Bases, Rev 2, pg 7
LOT-1540, Learning Objective 4 and 5

295008K204 [3.1/3.3]

295008K204 ..(KA's)

ANSWER: 054 (1.00)

b. + a

REFERENCE:

OT-112, Rev 16, pg 1
OT-112 Bases, Rev 15

295001A106 [3.3/3.4]

295001A106 ..(KA's)

ANSWER: 055 (1.00)

d.

REFERENCE:

Modified PB Bank #4587
LOT-0690, Learning Objective 3.a (related objective)

295004K203 [3.3/3.3]

295004K203 ..(KA's)

ANSWER: 056 (1.00)

a.

REFERENCE:

ON-112-2, Rev 0, pg 1
LOT-1550, Learning Objective 1
PB exam bank #1698, modified for different answers and distractors.
262002K105 [2.7/2.9]

262002K105 ..(KA's)

ANSWER: 057 (1.00)

d.

REFERENCE:

ON-107, Rev 4, pg 1 and 2
LOT-1550, Learning Objective 2
LOT-0070, pg 8, Learning Objective 4.k

295022A202 [3.3/3.4]

295022A202 ..(KA's)

ANSWER: 058 (1.00)

d.

REFERENCE:

ON-121, Rev 3, pg 1
LOT-1550, Learning Objective 2

295014A101 [4.0/4.1]

295014A101 ..(KA's)

ANSWER: 059 (1.00)

c.

REFERENCE:

ON-119, Rev 11, Attachment 1
LOT-1550, Learning Objective 1

295019K203 [3.2/3.3]

295019K203 ..(KA's)

ANSWER: 060 (1.00)

c.

REFERENCE:

T.S. 3.7.A.1.c(3)
OT-114, Rev 6, pg 1

295013G010 [3.8/3.6]

295013G010 ..(KA's)

ANSWER: 061 (1.00)

d.

REFERENCE:

ON-100, Rev 2, pg 1
LOT-1550, Learning Objective 1

290002K303 [3.3/3.4]

290002K303 ..(KA's)

ANSWER: 062 (1.00)

b.

REFERENCE:

T-103 Bases, Rev 6, pg 4

295035K202 [3.6/3.8]

295035K202 ..(KA's)

ANSWER: 063 (1.00)

c.

REFERENCE:

ON-105 Bases, Rev 3, pg 1

201003A402 [3.5/3.5]

201003A402 ..(KA's)

ANSWER: 064 (1.00)

c.

REFERENCE:

OT-106 Bases, Rev 11, pg 3
LOT-0500, Fig. 5, Learning Objective 3b and 5d
LOT-1550, Learning Objective 3

295017G007 [3.2/3.6]

295017G007 ..(KA's)

ANSWER: 065 (1.00)

d.

REFERENCE:

ON-113 Bases, Rev 8, pg 3
LOT 1550, Learning Objective 3

295018G00/ [3.2/3.4]

295018G007 ..(KA's)

ANSWER: 066 (1.00)

b.

REFERENCE:

ON-102, Rev 4, pg 1
LOT-1550, Learning Objective 1 and 2
LOT-0720, pg 28, Learning Objective 3a and 3c

295038A203 [3.5/4.3]

295038A203 ..(KA's)

ANSWER: 067 (1.00)

b.

REFERENCE:

SE-113 Bases, Rev 5, pg 45
LOT-1555, Learning Objective 13a

295003K303 [3.5/3.6]

295003K303 ..(KA's)

ANSWER: 068 (1.00)

a.

REFERENCE:

SE-10 Attachment 9 Figure 1, Rev 0, pg 2
LOT-1555, Learning Objective 2a and 2b

295016A202 [4.2/4.3]

295016A202 ..(KA's)

ANSWER: 069 (1.00)

c.

REFERENCE:

SE-10, Attachment 10, Rev 0, pg 2

294001A108 [3.1/3.6]

294001A108 ..(KA's)

ANSWER: 070 (1.00)

c.

REFERENCE:

LOT-1560, Learning Objective 6
modified PB exam bank #1694

295026K101 [3.0/3.4]

295026K101 ..(KA's)

ANSWER: 071 (1.00)

c.

REFERENCE:

LOT-1560, Learning Objective 3

295032G007 [3.3/3.5]

295032G007 ..(KA's)

ANSWER: 072 (1.00)

d.

REFERENCE:

T-215, Rev 0, pg 1
T-216, pg 4 Note; T-213, pg 2 Note 3

295015K201 [3.8/3.9]

295015K201 ..(KA's)

ANSWER: 073 (1.00)

a.

REFERENCE:

T-102, Rev 9, Table DW/T-1
T-103, Rev 6, Table SC/T-4
LOT 1560, Learning Objective 8

295028A203 [3.7/3.9]

295028A203 ..(KA's)

ANSWER: 074 (1.00)

c.

REFERENCE:

LOT-0050-5, Rev 10, pg 1 of 2., Learning Objective 6f

295031K202 [3.8/3.9]

295031K202 ..(KA's)

ANSWER: 075 (1.00)

a.

REFERENCE:

T-101 Bases, Rev 14, step RC/Q-15, pg 7
LOT-1560, Learning Objective 3

295037K301 [4.1/4.2]

295037K301 ..(KA's)

ANSWER: 076 (1.00)

d. 9C

REFERENCE:

T-102 Bases, Rev 10, pg 7 and T-102 T/L-8 (Bases for trip procedure curves not located.)
NOTE: Verify with facility that downcomers are uncovered at 10 feet but SRV tailpipes are still covered.

295030K207 [3.5/3.8]

295030K207 ..(KA's)

ANSWER: 077 (1.00)

c.

REFERENCE:

T-117 Bases, Rev 9, pg 9
LOT 1560, Learning Objective 3
Modified PB Bank #2889

295037K106 [4.0/4.2]

295037K106 ..(KA's)

ANSWER: 078 (1.00)

b. + c

REFERENCE:

T-102 Bases, pg 12 and 13, PC/P-5 and -9
LOT-1560, Learning Objective 6

295010G007 [3.6/3.8]

295010G007 ..(KA's)

ANSWER: 079 (1.00)

c.

REFERENCE:

T-101 Bases, Rev 14, Step RC/P-3, pg 20

295025K301 [4.2/4.3]

295025K301 ..(KA's)

ANSWER: 080 (1.00)

a.

REFERENCE:

T-102 Bases, Rev 10, Step PC/P-5,-6,-7,-8, pg 12 and 13

295024K301 [3.6/4.0]

295024K301 ..(KA's)

ANSWER: 081 (1.00)

d.

REFERENCE:

T-100 Bases, Rev 7, Step S-10, pg 4
LOT-1560, Learning Objective 1

295005A208 [3.2/3.3]

295005A208 ..(KA's)

ANSWER: 082 (1.00)

c.

REFERENCE:

GP-12, Rev 17, pg 5 and Figure 3
OM-C-7.1, Rev 0, Step 3.0, pg 2 and 3
295021A203 [3.5/3.5]

295021A203 ..(KA's)

ANSWER: 083 (1.00)

d.

Deleted

REFERENCE:

T-103 Bases, Rev 6, Step SC/L-1, pg 5
LOT-1560, Learning Objective 8
295036A101 [3.0/3.2]

295036A101 ..(KA's)

ANSWER: 084 (1.00)

d.

REFERENCE:

LOT-0160, Rev 6, pg 6 and 7, Learning Objective 7

294001K115 [3.4/3.8]

294001K115 ..(KA's)

ANSWER: 085 (1.00)

d.

REFERENCE:

A-C-10, Rev 10, Step 7.5.2.1.c, pg 14
LOT-0005, Learning Objective 2
294001A103 [2.7/3.7]

294001A103 ..(KA's)

ANSWER: 086 (1.00)

b.

REFERENCE:

A-C-40, Rev 0, Step 7.2 and 7.3, pg 5
LOT-1570, Learning Objective 1.j
294001A103 [2.7/3.7]

294001A103 ..(KA's)

ANSWER: 087 (1.00)

d.

REFERENCE:

A-8, Rev 13, pg 5 and 7
LOT-1570, Learning Objective 1.b

294001K101 [3.7/3.7]

294001K101 ..(KA's)

ANSWER: 088 (1.00)

b.

REFERENCE:

A-C-26, Rev 0, Step 7.1.6, pg 7
LOT-1570, Learning Objective 1.g

294001A110 [3.6/4.2]

294001A110 ..(KA's)

ANSWER: 089 (1.00)

a.

REFERENCE:

A-C-43, Rev 0, Step 7.4.3 and 7.4.4, pg 7
LOT-1570, Learning Objective 1.n

294001A113 [4.5/4.3]

294001A113 ..(KA's)

ANSWER: 090 (1.00)

c.

REFERENCE:

GP-2, Rev 76, Step 6.1.15, pg 47

294001A102 [4.2/4.2]

294001A102 ..(KA's)

ANSWER: 091 (1.00)

a.

REFERENCE:

HP-C-215, Rev 1, pg 9

294001K103 [3.3/3.8]

294001K103 ..(KA's)

ANSWER: 092 (1.00)

a.

REFERENCE:

HP-C-215, Rev 1, pg 9

294001K103 [3.3/3.8]

294001K103 ..(KA's)

ANSWER: 093 (1.00)

c.

REFERENCE:

LOT-0005
USNRC Regulatory Guide 1.114, Section B.2

294001A109 [3.3/4.2]

294001A109 ..(KA's)

ANSWER: 094 (1.00)

d.

REFERENCE:

OM-C-9.2, Rev 0, Step 6.2, pg 6
LOT-0007, Learning Objective None

294001A106 [3.4/3.6]

294001A106 ..(KA's)

ANSWER: 095 (1.00)

a.

REFERENCE:

OM-P-7.6, Rev 0, step 3.2, pg 4

294001K107 [3.3/3.6]

294001K107 ..(KA's)

ANSWER: 096 (1.00)

b.

REFERENCE:

FH-6C, Rev 38, Step 7.3.3, pg 11

234000K402 [3.3/4.1]

234000K402 ..(KA's)

ANSWER: 097 (1.00)

d.

REFERENCE:

ON-124, Rev 0, Step 6, pg 4
LOT-1550, Learning Objective 2

233000A202 [3.1/3.3]

233000A202 ..(KA's)

ANSWER: 098 (1.00)

d.

REFERENCE:

T-102, Rev 9, Step T/L-8

295026G012 [3.8/4.5]

295026G012 ..(KA's)

ANSWER: 099 (1.00)

d.

REFERENCE:

T-102, Curve T/L-3, Step T/L-19
LOT-1560, Learning Objective 6 and 8

295029K301 [3.5/3.9]

295029K301 ..(KA's)

ANSWER: 100 (1.00)

d.

REFERENCE:

T-102 sh 3 of 3, Rev 9, ECCS suction requirements tables
LOT-1560, Learning Objective 8

295030A101 [3.6/3.8]

(********** END OF EXAMINATION **********)

RO Exam BWR Reactor
Organized by Question Number

QUESTION	VALUE	REFERENCE
001	1.00	27094
002	1.00	9000001
003	1.00	9000002
004	1.00	9000003
005	1.00	9000004
006	1.00	9000005
007	1.00	9000006
008	1.00	9000007
009	1.00	9000008
010	1.00	9000009
011	1.00	9000010
012	1.00	9000011
013	1.00	9000012
014	1.00	9000013
015	1.00	9000014
016	1.00	9000015
017	1.00	9000016
018	1.00	9000017
019	1.00	9000018
020	1.00	9000019
021	1.00	9000020
022	1.00	9000021
023	1.00	9000022
024	1.00	9000023
025	1.00	9000024
026	1.00	9000025
027	1.00	9000026
028	1.00	9000027
029	1.00	9000028
030	1.00	9000029
031	1.00	9000030
032	1.00	9000031
033	1.00	9000032
034	1.00	9000033
035	1.00	9000034
036	1.00	9000035
037	1.00	9000036
038	1.00	9000037
039	1.00	9000038
040	1.00	9000039
041	1.00	9000040
042	1.00	9000041
043	1.00	9000042
044	1.00	9000043
045	1.00	9000044
046	1.00	9000045
047	1.00	9000048
048	1.00	9000049
049	1.00	9000050

RO Exam BWR Reactor
Organized by Question Number

QUESTION	VALUE	REFERENCE
050	1.00	9000051
051	1.00	9000053
052	1.00	9000054
053	1.00	9000055
054	1.00	9000056
055	1.00	9000057
056	1.00	9000058
057	1.00	9000059
058	1.00	9000061
059	1.00	9000063
060	1.00	9000066
061	1.00	9000067
062	1.00	9000068
063	1.00	9000070
064	1.00	9000072
065	1.00	9000073
066	1.00	9000074
067	1.00	9000077
068	1.00	9000078
069	1.00	9000079
070	1.00	9000081
071	1.00	9000082
072	1.00	9000084
073	1.00	9000085
074	1.00	9000086
075	1.00	9000088
076	1.00	9000089
077	1.00	9000090
078	1.00	9000091
079	1.00	9000092
080	1.00	9000093
081	1.00	9000094
082	1.00	9000095
083	1.00	9000097
084	1.00	9000102
085	1.00	9000103
086	1.00	9000104
087	1.00	9000105
088	1.00	9000113
089	1.00	9000114
090	1.00	9000115
091	1.00	9000116
092	1.00	9000117
093	1.00	9000119
094	1.00	9000122
095	1.00	9000123
096	1.00	9000124
097	1.00	9000125
098	1.00	9000126

RO Exam BWR Reactor
Organized by Question Number

QUESTION	VALUE	REFERENCE
099	1.00	9000127
100	1.00	9000128

	100.00	

	100.00	

R O E x a m   B W R R e a c t o r
O r g a n i z e d b y K A G r o u p

PLANT WIDE GENERICS

QUESTION	VALUE	KA
090	1.00	294001A102
085	1.00	294001A103
086	1.00	294001A103
094	1.00	294001A106
069	1.00	294001A108
093	1.00	294001A109
088	1.00	294001A110
089	1.00	294001A113
087	1.00	294001K101
092	1.00	294001K103
091	1.00	294001K103
095	1.00	294001K107
084	1.00	294001K115

PWG Total	13.00	

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
005	1.00	201001A109
006	1.00	201001K101
014	1.00	202002A401
001	1.00	203000A405
010	1.00	203000G009
011	1.00	206000A217
012	1.00	206000G010
013	1.00	206000K201
033	1.00	212000K114
034	1.00	212000K410
035	1.00	215003A401
037	1.00	215003K602
036	1.00	215005A104
002	1.00	217000A401
007	1.00	217000K407
008	1.00	218000K501
009	1.00	218000K606
029	1.00	223002A302
028	1.00	241000A101
047	1.00	241000G014
019	1.00	259001A204
018	1.00	259001G010
027	1.00	259002A401
030	1.00	259002K603

RO Exam BWR Reactor
Organized by KA Group

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
048	1.00	259002K603
038	1.00	261000K301
022	1.00	264000A401
020	1.00	264000A404

PS-I Total	28.00	

Group II

QUESTION	VALUE	KA
063	1.00	201003A402
003	1.00	202001K408
004	1.00	202001K502
023	1.00	204000G004
024	1.00	204000K404
032	1.00	215002K102
031	1.00	215002K403
041	1.00	226001A401
039	1.00	239001K201
040	1.00	239001K403
016	1.00	256000K102
017	1.00	256000K606
021	1.00	262001A302
056	1.00	262002K105
043	1.00	263000K102
026	1.00	271000A211
025	1.00	271000K101
042	1.00	272000K402
044	1.00	286000K403

PS-II Total	19.00	

Group III

QUESTION	VALUE	KA
097	1.00	233000A202
096	1.00	234000K402
015	1.00	288000A301
061	1.00	290002K303

PS-III Total	4.00	

RO Exam BWR Reactor
Organized by KA Group

PLANT SYSTEMS

QUESTION	VALUE	KA
PS Total	51.00	

EMERGENCY PLANT EVOLUTIONS

Group I

QUESTION	VALUE	KA
081	1.00	295005A208
049	1.00	295007A105
051	1.00	295007K201
050	1.00	295010G006
078	1.00	295010G007
058	1.00	295014A101
072	1.00	295015K201
080	1.00	295024K301
079	1.00	295025K301
074	1.00	295031K202
077	1.00	295037K106
075	1.00	295037K301

EPE-I Total	12.00	

Group II

QUESTION	VALUE	KA
054	1.00	295001A106
046	1.00	295001G010
045	1.00	295002G010
067	1.00	295003K303
055	1.00	295004K203
052	1.00	295008A201
053	1.00	295008K204
060	1.00	295013G010
068	1.00	295016A202
064	1.00	295017G007
065	1.00	295018G007
059	1.00	295019K203
057	1.00	295022A202
098	1.00	295026G012
070	1.00	295026K101
073	1.00	295028A203
099	1.00	295029K301
076	1.00	295030K207
066	1.00	295038A203

R O Exam BWR Reactor
Organized by KA Group

EMERGENCY PLANT EVOLUTIONS

Group II

QUESTION	VALUE	KA

EPE-II Total	19.00	

Group III

QUESTION	VALUE	KA
082	1.00	295021A203
071	1.00	295032G007
062	1.00	295035K202
083	1.00	295036A101

EPE-III Total	4.00	

EPE Total	35.00	

Test Total	100.00	

ANSWER KEY

MULTIPLE CHOICE

001 c

002 d

003 a

004 b

005 d

006 a

007 a

008 c

009 c

010 c

011 b

012 d

013 a + c

014 d

015 b

016 a

017 c

018 b

019 a

020 b

021 b

022 c

023 b

024 a

025 a

026 b

027 d

028 c

029 b

030 b

031 b

032 c

033 b

034 a

035 c

036 d

037 c

038 d

039 c + d

040 d

041 a

042 c

043 b

044 a

045 c

ANSWER KEY

046	d	069	c
047	d	070	c
048	a	071	c
049	b	072	d
050	c	073	a
051	b	074	c
052	c	075	a
053	d	076	d + c
054	b + a	077	c
055	d	078	b + c
056	a	079	c
057	d	080	a
058	d	081	d
059	c	082	c
060	c	~~083~~	~~d~~ Deleted
061	d	084	d
062	b	085	d
063	c	086	b
064	c	087	d
065	d	088	b
066	b	089	a
067	b	090	c
068	a	091	a

ANSWER KEY

092	a
093	c
094	d
095	a
096	b
097	d
098	d
099	d
100	d

(********** END OF EXAMINATION **********)

ATTACHMENT 2

SRO EXAMINATION AND ANSWER KEY

U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
SENIOR OPERATOR LICENSE
REGION 1

APPLICANT'S NAME: ______________________

FACILITY: Peach Bottom 2 & 3

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: August 17, 1995

INSTRUCTIONS TO APPLICANT:

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires a final grade of at least 80%. Examination papers will be picked up four (4) hours after the examination starts.

TEST VALUE	APPLICANT'S SCORE	FINAL GRADE
100		

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

____
Applicant's Signature

ANSWER SHEET

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

MULTIPLE CHOICE

001	a	b	c	d	___
002	a	b	c	d	___
003	a	b	c	d	___
004	a	b	c	d	___
005	a	b	c	d	___
006	a	b	c	d	___
007	a	b	c	d	___
008	a	b	c	d	___
009	a	b	c	d	___
010	a	b	c	d	___
011	a	b	c	d	___
012	a	b	c	d	___
013	a	b	c	d	___
014	a	b	c	d	___
015	a	b	c	d	___
016	a	b	c	d	___
017	a	b	c	d	___
018	a	b	c	d	___
019	a	b	c	d	___
020	a	b	c	d	___
021	a	b	c	d	___
022	a	b	c	d	___
023	a	b	c	d	___
024	a	b	c	d	___
025	a	b	c	d	___
026	a	b	c	d	___
027	a	b	c	d	___
028	a	b	c	d	___
029	a	b	c	d	___
030	a	b	c	d	___
031	a	b	c	d	___
032	a	b	c	d	___
033	a	b	c	d	___
034	a	b	c	d	___
035	a	b	c	d	___
036	a	b	c	d	___
037	a	b	c	d	___
038	a	b	c	d	___
039	a	b	c	d	___
040	a	b	c	d	___
041	a	b	c	d	___
042	a	b	c	d	___
043	a	b	c	d	___
044	a	b	c	d	___
045	a	b	c	d	___

ANSWER SHEET

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

046	a	b	c	d	___	069	a	b	c	d	___
047	a	b	c	d	___	070	a	b	c	d	___
048	a	b	c	d	___	071	a	b	c	d	___
049	a	b	c	d	___	072	a	b	c	d	___
050	a	b	c	d	___	073	a	b	c	d	___
051	a	b	c	d	___	074	a	b	c	d	___
052	a	b	c	d	___	075	a	b	c	d	___
053	a	b	c	d	___	076	a	b	c	d	___
054	a	b	c	d	___	077	a	b	c	d	___
055	a	b	c	d	___	078	a	b	c	d	___
056	a	b	c	d	___	079	a	b	c	d	___
057	a	b	c	d	___	080	a	b	c	d	___
058	a	b	c	d	___	081	a	b	c	d	___
059	a	b	c	d	___	082	a	b	c	d	___
060	a	b	c	d	___	083	a	b	c	d	___
061	a	b	c	d	___	084	a	b	c	d	___
062	a	b	c	d	___	085	a	b	c	d	___
063	a	b	c	d	___	086	a	b	c	d	___
064	a	b	c	d	___	087	a	b	c	d	___
065	a	b	c	d	___	088	a	b	c	d	___
066	a	b	c	d	___	089	a	b	c	d	___
067	a	b	c	d	___	090	a	b	c	d	___
068	a	b	c	d	___	091	a	b	c	d	___

ANSWER SHEET

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

092 a b c d ___

093 a b c d ___

094 a b c d ___

095 a b c d ___

096 a b c d ___

097 a b c d ___

098 a b c d ___

099 a b c d ___

100 a b c d ___

(********** END OF EXAMINATION **********)

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. After the examination has been completed, you must sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination. This must be done after you complete the examination.

3. Restroom trips are to be limited and only one applicant at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

4. Use black ink or dark pencil ONLY to facilitate legible reproductions.

5. Print your name in the blank provided in the upper right-hand corner of the examination cover sheet and each answer sheet.

6. Mark your answers on the answer sheet provided. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.

7. The point value for each question is indicated in parentheses after the question.

8. If the intent of a question is unclear, ask questions of the examiner only.

9. When turning in your examination, assemble the completed examination with examination questions, examination aids and answer sheets. In addition, turn in all scrap paper.

10. Ensure all information you wish to have evaluated as part of your answer is on your answer sheet. Scrap paper will be disposed of immediately following the examination.

11. To pass the examination, you must achieve a grade of 80% or greater.

12. There is a time limit of four (4) hours for completion of the examination.

13. When you are done and have turned in your examination, leave the examination area (EXAMINER WILL DEFINE THE AREA). If you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION: 001 (1.00)

Shutdown cooling (SDC) was in operation with RHR pump "A". A level transient has occurred which has resulted in the following:

- Reactor vessel level is -10 inches.
- SDC automatically isolated as required.

Which one of the following provides the MINIMUM operator actions necessary to inject into the RPV with RHR pump A?

a. Close MO-17 & MO-18 (SDC inboard/outboard isolations).
   Depress both shutdown control pushbuttons (10A-S32A/B).
   Close MO-15 (SDC suction) and open MO-13 (LPCI Torus suction).
   Start RHR pump "A".

b. Close MO-17 & MO-18 (SDC inboard/outboard isolations).
   Depress the "A" shutdown control pushbutton (10A-S32A).
   Close MO-15 (SDC suction) and open MO-13 (LPCI Torus suction).
   Open MO-25A (LPCI Inboard injection).

c. Depress both shutdown control pushbuttons (10A-32A/B).
   Close MO-15 (SDC suction) and open MO-13 (LPCI Torus suction).
   Start RHR pump "A".
   Open MO-25A (LPCI Inboard injection).

d. Close MO-17 & MO-18 (SDC inboard/outboard isolations).
   Close MO-15 (SDC suction) and open MO-13 (LPCI Torus suction).
   Start RHR pump "A".
   Open MO-25A (LPCI Inboard injection).

QUESTION: 002 (1.00)

Reactor Core Isolation Cooling (RCIC) has initiated due to low RPV water level. Level has been restored and normal feedwater has been reestablished. The Control Room Supervisor (CRS) has directed that RCIC be shutdown. The following plant conditions exist:

- Reactor water level is +25 inches and stable.
- Drywell pressure is 2.5 psig and stable.

Which one of the following describes the proper method for shutting down RCIC?

a. Depress the RCIC Manual Isolation pushbutton and realign RCIC to STANDBY.

b. Close the MO-21 RCIC injection valve and then close and reopen MO-4487 trip throttle valve.

c. Close the MO-16 outboard steam isolation valve and realign RCIC to STANDBY.

d. Depress the RCIC Turbine Trip pushbutton and realign RCIC to STANDBY.

QUESTION: 003 (1.00)

Which one of the following describes a Reactor Recirculation System pump start limitation and its purpose?

a. Temperature differential between the bottom head region and the steam dome is limited to prevent excessive moisture carryover.

b. The operating pump speed is limited to prevent excessive vibration of the jet pumps in the idle loop.

c. Loop to loop temperature differentials are limited to prevent excessive thermal stresses in the idle loop jet pumps.

d. Loop to dome temperature differentials are limited to prevent thermal hydraulic instability.

QUESTION: 004 (1.00)

While attempting to free a stuck control rod, the operator observes drive water flow indication does NOT change when rod insertion is attempted.

Which one of the following is a possible cause of this indication?

a. Associated drive water stabilizing valve failed open.

b. Drive water pressure control valve failed closed.

c. Cooling water pressure control valve failed closed.

d. Directional control valve failed closed.

QUESTION: 005 (1.00)

Which one of the following describes the effect that a trip of all the condensate pumps will have on the Control Rod Drive Hydraulic System?

The running CRD pump(s) will _______.

a. draw a suction from the CST via the hotwell makeup and reject line

b. develop a low discharge pressure resulting in accumulator trouble alarms

c. overheat due to low flow

d. trip on low suction pressure

QUESTION: 006 (1.00)

Which one of the following describes the RCIC suction valve interlocks?

a. Torus suction valves (MO-39 and MO-41) being full open will cause CST suction valve (MO-18) to auto close.

b. CST suction valve (MO-18) being full open will cause torus suction valves (MO-39 and MO-41) to auto close.

c. Torus suction valves (MO-39 and MO-41) cannot be opened if the CST suction valve (MO-18) is open.

d. CST suction valve (MO-18) opens for an auto RCIC initiation irrespective of the position of the torus suction valves (MO-39 and MO-41).

QUESTION: 007 (1.00)

Which one of the following statements describes the operation of the ADS 105 second timer?

a. The timer must be manually reset to prevent blowdown if the triple low water level signal clears before the timer times-out.

b. The timer will auto reset if the high drywell pressure signal clears before the timer times-out.

c. If the ADS valves are open and the ADS initiation signal is still present, depressing the timer reset pushbutton will close the ADS valves for 105 seconds.

d. The timer time-out can be stopped by placing either keylock Inhibit Switch "A" OR "B" to INHIBIT to prevent ADS valves from opening.

QUESTION: 008 (1.00)

Which one of the following describes the consequences of a loss of 125 VDC panel 20D2111 on the Automatic Depressurization System?

a. Causes the "A" channel to shift to its alternate power supply. ADS initiation is still possible via the "A" or "B" channel.

b. Causes the "B" channel to shift to its alternate power supply. ADS initiation is still possible via the "A" or "B" channel.

c. Causes the "A" channel to deenergize. ADS initiation is still possible through the "B" channel.

d. Causes the "B" channel to deenergize. ADS initiation is still possible through the "A" channel.

QUESTION: 009 (1.00)

Which one of the following statements describes the operation of the LPCI Lockout reset pushbutton (S1A)?

Pushing this bottom will ________.

a. cause LPCI flow to be diverted to torus cooling following a LPCI injection if the LOCA signal is still present

b. automatically open the Recirc pump discharge valve if the LOCA signal has cleared

c. allows MO-25 (LPCI injection valve) to be manually closed after the LOCA signal has cleared

d. reset the LOCA closure lockout of the containment spray valves if the LOCA signal is still present

QUESTION: 010 (1.00)

Select the choice below that completes the following statements.

The reactor is operating at 100% power when the High Pressure Coolant Injection system malfunctions and starts injecting into the reactor. As a result, feedwater temperature decreases, ________.

a. reactor pressure remains constant, and reactor power increases until a reactor scram occurs.

b. reactor pressure remains constant, and reactor power increases but will not cause a scram.

c. Reactor pressure decreases, and reactor power increases but will not cause a scram.

d. Reactor pressure decreases, and reactor power increases until a reactor scram occurs.

QUESTION: 011 (1.00)

During HPCI turbine operation, the operator is cautioned against operating the turbine below 2200 RPM.

Which one of the following is the reason for this precaution?

a. To prevent going below the range of the speed control circuit.

b. To prevent possible cavitation in the pump impeller and volute.

c. To ensure sufficient pump discharge pressure to prevent water hammer in the discharge line.

d. To ensure sufficient oil pressure exists to keep the HPCI turbine stop valve (HO-4513) open.

QUESTION: 012 (1.00)

The HPCI system is in its normal standby readiness lineup.

Which one of the following power supplies, if deenergized, would prevent HPCI from being either manually or automatically initiated?

a. 250 VDC bus (20D11)

b. 480 VAC MCC (20B38)

c. 125 VDC bus (20D22)

d. 480 VAC MCC (30B37)

QUESTION: 013 (1.00)

With the reactor shutdown, a startup of reactor recirculation pump A is about to begin.

Which one of the following statements describes the reactor recirculation pump start sequence?

a. The generator field breaker is initially open, but will shut approximately 10 seconds after the drive motor starts.

b. The pump must develop at least 5 psid within 21 seconds of the field breaker shutting or an MG set incomplete sequence trip will occur.

c. When the drive motor breaker is closed, the generator will initially go to 50% speed, but after the field breaker shuts, generator speed will decrease to 28% to 30%.

d. When the drive motor breaker is closed, the generator will initially go to 100% speed, but after the field breaker shuts, generator speed will decrease to 28% to 30%.

QUESTION: 014 (1.00)

The plant is operating at 100% power when the "A" condensate pump trips.

Which one of the following describes the consequences of this malfunction?

a. The reactor recirculation pumps runback to 45% speed and one feed pump trips.

b. The reactor recirculation pumps and the feed pumps both runback to 45% speed.

c. The reactor recirculation pumps runback to 45% speed and the feed pumps runback to 85% speed.

d. One feed pump runs back to 85% speed and one feed pump trips.

QUESTION: 015 (1.00)

While placing the second reactor feed pump (RFP) in service during a startup, discharge check valve slamming is detected.

Which one of the following describes the actions which must be taken?

a. Immediately shut the affected RFP discharge valve.

b. Immediately trip the affected RFP.

c. Reduce the speed of the affected RFP.

d. Raise the speed of the affected RFP.

QUESTION: 016 (1.00)

Which one of the following conditions will result in the largest decrease in feedwater inlet temperature at 100% reactor power?

a. Loss of extraction steam to a fifth stage feedwater heater.

b. Inadvertent Reactor Core Isolation Cooling Injection.

c. Inadvertent High Pressure Coolant Injection.

d. Inadvertent closure of the SJAE condenser condensate inlet valve.

QUESTION: 017 (1.00)

The following conditions exist:

- ALL emergency diesel generators (EDG) started in response to a MCA (Maximum Credible Accident) start signal.
- One EDG received a jacket temperature high alarm and trip signal immediately after loading.
  - -- The trip is automatically bypassed by the MCA start.
  - -- The EDG successfully supplied power as required.

Which one of the following describes the expected response of this EDG during restoration of off-site power?

a. The EDG will trip when the Unit/Parallel switch is placed in PARALLEL if the MCA signal is clear.

b. The trip will be enabled after a manual control room stop if the MCA signal is clear.

c. The EDG will trip as soon as the MCA signal is clear.

d. The EDG will trip when either off-site supply breaker to its emergency bus is closed in parallel with the EDG.

QUESTION: 018 (1.00)

Plant conditions are as follows:

- MSIVs are closed, and RPV pressure is being controlled between 960 psig and 1060 psig with the SRVs.
- RPV water level is 23 inches and steady.
- Drywell pressure is 0.75 psig.
- The #2 Emergency Auxiliary Transformer (0AX04) is out of service.
- All eight 4 KV busses are being supplied by the #3 Emergency Auxiliary Transformer (0BX04).
- The 2A RHR and 2A HPSW pumps are running in Torus Cooling Mode.

The E-312 breaker then trips due to a breaker failure.

Which one of the following statements describes the expected plant response?

a. The E-1 Diesel Generator will auto start but not tie to the E-12 Bus.

b. The E-1 Diesel Generator will auto start and tie to the E-12 Bus, but 2A RHR and 2A HPSW will not auto start.

c. The E-1 Diesel Generator will auto start and tie to the E-12 Bus, and 2A RHR pump will auto start but 2A HPSW pump will not auto start.

d. The E-1 Diesel Generator will auto start and tie to the E-12 Bus, and both the 2A RHR pump and 2A HPSW pump will auto start.

QUESTION: 019 (1.00)

The E-2 Diesel Generator is tied to the E-22 bus for testing and is running in PARALLEL with offsite power. The following parameters are observed by the operator:

DG Voltage:	4.20 KV
DG Frequency:	60 Hz
DG Load:	2300 KW
DG KVAR:	1900 KVAR

Which one of the following statements describes the proper operator action?

a. Reduce D/G voltage to 4.16 KV

b. Reduce D/G KW using the GOVERNOR control switch to achieve a 0.75 power factor.

c. Reduce D/G KVAR using the AUTO VOLT REG control switch to achieve a 0.75 power factor.

d. Maintain current stable D/G operation since all parameters are within normal operating bands.

QUESTION: 020 (1.00)

Which one of the following describes the operation of the Reactor Water Cleanup system in the DUMP MODE of operation?

a. The dump mode will isolate at 140 psig (decreasing) upstream of the Clean Up Drain Header Control Valve (CV-55).

b. The dump mode will isolate at 5 psig (decreasing) upstream of the Clean Up Drain Header Control Valve (CV-55).

c. The RWCU system will isolate if the regenerative heat exchanger outlet temperature exceeds 200 deg. F.

d. The RWCU system will isolate if the non-regenerative heat exchanger outlet temperature exceeds 130 deg. F.

QUESTION: 021 (1.00)

Which one of the following conditions will cause the Reactor Water Cleanup System to ISOLATE?

a. RWCU suction line flow (300%).

b. RPV level (+29 inches)

c. RWCU pump flow (70 gpm)

d. Drywell pressure (2.0 psig)

QUESTION: 022 (1.00)

Which one of the following describes the gas flow through the Off-Gas system?

a. Composed mostly of hydrogen and oxygen in a 2:1 ratio which is catalytically recombined.

b. Receives its driving force from the third stage air ejector which receives its normal steam supply from the main turbine cross around header.

c. Superheated to about 3350 degrees F by an electric preheater to maximize the efficiency of the catalytic recombiner.

d. Will automatically isolate the Steam Supply valves to the Jet Compressors in the event that a "STACK GAS HIGH RADIATION" alarm condition occurs.

QUESTION: 023 (1.00)

Unit 2 is at 100% power when the following alarm is received:

- JET COMPRESSOR STEAM FLOW LOW

Which one of the following statements describes the plant response and/or the operator action that will be required?

a. If normal steam flow cannot be restored by the on-line pressure control valve (PCV), the standby PCV must be manually placed on-line.

b. If steam flow cannot be restored above the alarm setpoint within 30 seconds, the Off-Gas inlet valve will close.

c. To increase steam flow, the off-line set of Steam Jet Air Ejectors can be manually placed in service.

d. Reactor power must be reduced to less than 70% to minimize the amount of noncondensibles that are being built up in the main condenser.

QUESTION: 024 (1.00)

Which one of the following describes the operation of the individual manual/automatic (M/A) stations and the master control station for the Reactor Feed Pump Turbines (RFPTs)?

With all three RFPT M/A stations ________.

a. in "MANUAL" and the master controller in "MANUAL", RFPT speed may be adjusted by use of the master controller control knob

b. in "MANUAL" and the master controller in "AUTO", RFPT speed may be adjusted by use of the master controller control knob.

c. in "AUTO" and the master controller in "MANUAL", one slow turn of the master controller control knob produces the same RFPT speed change as one fast turn, but at a slower rate of speed increase.

d. in "AUTO" and the master controller in "MANUAL", the faster the master controller control knob is turned, the larger the magnitude of the RFPT speed change.

QUESTION: 025 (1.00)

Given the following information and the attached figure of the Electro-hydraulic Pressure Control (EHC) System:

PAM Pressure	950 psig
EHC Pressure Setpoint	920 psig
Load Limit	105 percent
Pressure Regulator "A"	Controlling

The PAM Pressure transmitter (PT-2184) has just failed downscale to 0 psig (input to pressure regulator "A").

Which one of the following describes the EHC system response?

a. "A" Pressure Regulator remains in control - a large reactor pressure INCREASE occurs

b. "A" Pressure Regulator remains in control - a large reactor pressure DECREASE occurs

c. "B" Pressure Regulator takes control - reactor pressure INCREASES slightly

d. "B" Pressure Regulator takes control - reactor pressure DECREASES slightly

QUESTION: 026 (1.00)

Plant conditions for Unit 2 are as follows:

- Drywell pressure is 1.45 psig.
- Reactor water level is -56 inches.
- Reactor pressure is 70 psig.
- Mode Switch is in SHUTDOWN.

Note: Primary Containment Isolation System (PCIS) Groups are listed below:

I	Main steam system isolation valves
II	Reactor auxiliary systems isolation valves
III	Ventilation system isolation valves and dampers
IV	HPCI
V	RCIC
VI	Core Spray
VII	RHR

Which one of the following states the PCIS group isolations that are active?

a. I, II, VI, VII

b. II, III, IV, V

c. III, IV, V, VI

d. I, V, VI, VII

QUESTION: 027 (1.00)

The Digital Feed Control System (DFCS) is controlling reactor water level in three element control.

Which one of the following describes the response of the Digital Feed Control System (DFCS) to a loss of ONE (1) steam flow signal?

a. DFCS auto transfers to single element control, C05 total steam flow recorder will increase.

b. DFCS auto transfers to single element control, C05 total steam flow recorder will decrease.

c. DFCS remains in three element control, C05 total steam flow recorder will increase.

d. DFCS remains in three element control, C05 total steam flow recorder will decrease.

QUESTION: 028 (1.00)

Which one of the following statements describe the Rod Block Monitor (RBM) System channel "B" nulling process?

a. The RBM high/inop trips are inhibited during the nulling sequence to prevent interrupting rod motion.

b. A rod block is generated during the nulling sequence.

c. Bypassing the "A" APRM will reset the nulling sequence clock and initiate a new nulling sequence.

d. Depressing the "Set Up" pushbutton when the selected trip level is reached initiates a new nulling sequence.

QUESTION: 029 (1.00)

Which one of the following statements describes the LPRM inputs to the Rod Block Monitor?

a. The RBM considers an LPRM operable and uses its input for the averaging circuit if the LPRM function switch is in OPERATE.

b. The RBM is bypassed when the average of all the operable LPRMs to a RBM indicates less than 30%.

c. An LPRM that is reading downscale will not be counted in the RBM count circuit.

d. An LPRM upscale condition will automatically bypass the LPRM input to the RBM.

QUESTION: 030 (1.00)

Which one of the following describes a Main Steam Isolation Valve (MSIV) alignment that would cause a HALF scram ONLY via the Reactor Protective System MSIV valve logic?

a. Both MSIVs in A steam line closed.

b. One MSIV in A steam line closed, one MSIV in B steam line closed.

c. One MSIV in A steam line closed, one MSIV in D steam line closed.

d. One MSIV in B steam line closed, one MSIV in C steam line closed.

QUESTION: 031 (1.00)

T-216-2, "Control Rod Insertion by Manual Scram or Individual Scram Test Switches", provides instructions for manually scramming control rods using the individual rod scram test switches.

Which one of the following describes the mechanism by which these switches affect control rod insertion?

a. A single switch, when taken to the TEST position, will DE-ENERGIZE both scram pilot solenoids for its associated control rod.

b. A single switch, when taken to the TEST position, will ENERGIZE both scram pilot solenoids for its associated control rod.

c. Two switches are provided for each associated control rod, and, when taken to the TEST positions, will DE-ENERGIZE their associated scram pilot solenoids.

d. Two switches are provided for each associated control rod, and, when taken to the TEST positions, will ENERGIZE their associated scram pilot solenoids.

QUESTION: 032 (1.00)

Unit 2 is operating at 90% power when the following plant conditions occur:

- "CONDENSER LOW VACUUM ALARM" is received.
- Condenser vacuum is 25.4 inches and slowly decreasing.

Which one of the following provides the required IMMEDIATE operator action(s)?

a. Verify proper operation of the on-line SJAE, place the standby SJAE in service, and enter T-100 "Scram" procedure if condenser vacuum continues dropping.

b. Scram the reactor, enter T-100 "Scram" procedure, and close the MSIVs.

c. Reduce reactor power per GP-9-2 "Fast Power Reduction" until condenser vacuum stops dropping.

d. Reduce reactor power per GP-9-2 "Fast Power Reduction" until "CONDENSER LOW VACUUM" alarm clears or an "APRM HIGH" alarm is received.

QUESTION: 033 (1.00)

Unit 3 is operating at 100% power when Recirculation Pump B trips. Plant conditions are as follows:

- APRMs indicate 71%.
- Calculated core flow is 46%.
- No thermal hydraulic instability has been observed.
- No operator actions have been taken.

Which one of the following is the operator action that must be IMMEDIATELY performed? (OT-112, Figures A and B, Power to Flow Maps are attached for reference.)

a. Insert rods in reverse sequence to exit Region 2 and continually monitor for thermal hydraulic instability.

b. Manually scram the reactor and enter T-100 "Scram" procedure.

c. No rod insertion is necessary but continually monitor for thermal hydraulic instability.

d. Fully insert the Table 1 rods of GP-9-3 "Fast Reactor Power Reduction".

QUESTION: 034 (1.00)

A followup action for OT-114 "Inadvertent Opening of a Relief valve" directs the operator to reduce reactor recirculation pump speed to minimum in accordance with GP-9-2(3) "Fast Reactor Power Reduction" prior to lowering the EHC pressure set to decrease PAM pressure to 900 psig.

Which one of the following is the reason for this reactor power reduction?

a. To attempt to reseat the SRV by reducing reactor pressure.

b. To reduce the heat load on the torus until the SRV is closed.

c. To prevent an excessive power excursion if the SRV re-closes.

d. To reduce reactor power below the PCIOMR envelope.

QUESTION: 035 (1.00)

During power operations on Unit 2, the operator observes that several turbine bypass valves are open and reactor pressure is decreasing. If the operator cannot stop the pressure decrease by running back the Max Combined Flow Limiter, OT-111 "Reactor Low Pressure" procedure directs a reactor scram.

Which one of the following is the reason for scramming the reactor?

a. To enable the operator to immediately terminate the pressure reduction by closing the MSIVs.

b. To place the mode switch in shutdown and maintain the reactor feed pumps and main condenser available.

c. To enable the operator to manually control reactor pressure below 900 psig with the bypass valve jack.

d. To reduce the initial heat load on the torus when the MSIVs close.

QUESTION: 036 (1.00)

Select the choice below that completes the following statements. (Assume no operator actions are taken.)

Unit 2 is operating at 90% power when a safety relief valve inadvertently opens and sticks open. As a result, feedwater flow rate will stabilize ________.

a. HIGHER than main steam flow rate, and PAM (Pressure Averaging Manifold) pressure will stabilize at its previous value

b. HIGHER than main steam flow rate, and PAM (Pressure Averaging Manifold) pressure will decrease and stabilize at a LOWER pressure

c. LOWER than main steam flow rate, and PAM (Pressure Averaging Manifold) pressure will stabilize at its previous value

d. LOWER than main steam flow rate, and PAM (Pressure Averaging Manifold) pressure will decrease and stabilize at a LOWER pressure

QUESTION: 037 (1.00)

Unit 2 is operating at 90% power when the operator observes the following plant conditions:

- Reactor pressure 1055 psig and slowly increasing.
- APRMs slowly increasing.

Which one of the following is the FIRST IMMEDIATE operator action?

a. Operate the Bypass Valve Jack to control reactor pressure below 1085 psig.

b. Operate the Bypass Valve Jack to control reactor pressure below 1050 psig.

c. Lower the EHC pressure set to control reactor pressure below 1050 psig.

d. Lower the EHC pressure set to control reactor pressure below 1085 psig.

QUESTION: 038 (1.00)

SELECT the choice below that completes the following statements.

During execution of OT-101 "High Drywell Pressure", the operator determines that both seals on Reactor Recirc Pump B have failed. Per OT-101, the recirc pump is tripped and isolated. When isolating the recirc pump, the operator is directed to ________.

a. simultaneously close the suction and discharge valves to quickly isolate the seal leak.

b. first close the seal purge isolation valves to reduce radioactive leakage from the pump seals when the pump is isolated.

c. first close the pump suction valve because a high differential pressure across the valve may prevent its closure if the discharge valve is closed.

d. first close the pump discharge valve due to its ability to operate against a large differential pressure.

QUESTION: 039 (1.00)

Select the choice below that completes the following statements.

During execution of OT-101 "High Drywell Pressure", the shift suspects that reactor recirc loop A suction and discharge valve stems are leaking. Per OT-101, the recirc pump suction and discharge valves are NOT to be backseated because ________.

a. operation of the LPCI system relies on the automatic closure of the recirc discharge valve, thus one LPCI subsystem must be declared inoperable and a 7 day LCO must be initiated

b. operation of the LPCI system relies on the automatic closure of the recirc discharge valve, thus requiring the reactor to be in COLD SHUTDOWN within 48 hours

c. both valves are required to isolate a recirc pump seal failure, thus requiring the reactor to be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN in the following 24 hours.

d. both valves are required to isolate a recirc pump seal failure, thus the valves must be declared inoperable and the reactor must be in COLD SHUTDOWN within 24 hours

Deleted

QUESTION: 040 (1.00)

Unit 3 is operating at 85% power at 1020 psig when the following plant conditions occur:

- Reactor pressure spikes to 1043 psig and then stabilizes at 1030 psig.
- Reactor power increases to 91% and then stabilizes at 85%.

Which one of the following failures would cause this plant response?

a. The on-line EHC regulator's setpoint has failed high and the backup regulator is in control.

b. One MSIV disk has separated from its stem and has failed fully closed.

c. One SRV has inadvertently lifted and has failed to fully reseat upon re-closing.

d. The extraction steam to the fifth-point feedwater heater has isolated.

QUESTION: 041 (1.00)

Following a transient, Unit 3 has scrammed and plant conditions are as follows:

- Reactor pressure has been stabilized at 500 psig.
- MSIVs are open.

Which one of the following is the MAXIMUM ALLOWABLE reactor water level indication on LI-3-2-3-86 for which the MSIVs may remain open? (Refer to OT-110, Figure 1, LI-2(3)-2-3-86 Indication.)

a. +108 inches

b. +93 inches

c. +78 inches

d. +60 inches

QUESTION: 039 (1.00)

Select the choice below that completes the following statements.

During execution of OT-101 "High Drywell Pressure", the shift suspects that reactor recirc loop A suction and discharge valve stems are leaking. Per OT-101, the recirc pump suction and discharge valves are NOT to be backseated because ________.

a. operation of the LPCI system relies on the automatic closure of the recirc discharge valve, thus one LPCI subsystem must be declared inoperable and a 7 day LCO must be initiated

b. operation of the LPCI system relies on the automatic closure of the recirc discharge valve, thus requiring the reactor to be in COLD SHUTDOWN within 48 hours

c. both valves are required to isolate a recirc pump seal failure, thus requiring the reactor to be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN in the following 24 hours.

d. both valves are required to isolate a recirc pump seal failure, thus the valves must be declared inoperable and the reactor must be in COLD SHUTDOWN within 24 hours

Deleted

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a. +108 inches

b. +93 inches

c. +78 inches

d. +60 inches

QUESTION: 042 (1.00)

A transient on unit 2 has resulted in flooding of the main steam lines and a reactor pressure of 500 psig that is rapidly increasing (150 psig per minute).

Which one of the following will increase the SRV hydraulic discharge loads when the SRV(s) is/are opened?

a. Increased drywell temperature and pressure.

b. Reduced subcooling of the water in the main steam lines.

c. Prolonging the time that an SRV is open to reduce reactor pressure.

d. Opening the SRV at a reactor pressure of 1030 psig instead of 700 psig.

QUESTION: 043 (1.00)

Unit 2 is operating at power when a recirculation flow reduction event results in entry into Region 2 of the Power to Flow Map (attached for reference). Plant conditions prior to the event were as follows:

- Reactor power was 90%.
- APRMs indicated 90% +/- 1%.
- All LPRMs were above downscale alarms and below upscale alarms.
- LPRMs near center of core indicated 95% +/- 2%.
- Period meter indicated infinity.

After the flow reduction event and core flow first reaches its lowest flow rate, which ONE of the following neutron instrumentation responses indicates reactor instability?

a. APRMs are oscillating between 58% and 62% every second.

b. Period meter is oscillating between a -50 second period and a +50 second period every 2 seconds.

c. LPRMs near the center of the core are oscillating between 64% and 72% every 10 to 15 seconds.

d. LPRM downscale alarms occur briefly and then clear at 10 seconds, 25 seconds, and 60 seconds.

QUESTION: 044 (1.00)

Which one of the following describes the effect of losing both divisions of the 24 VDC power system?

a. Core Spray will inadvertently initiate.

b. Reactor feedwater pumps will lock up.

c. A main turbine trip occurs only if turbine first stage pressure indicates greater than 30% power.

d. A reactor scram occurs if the Mode Switch is in STARTUP.

QUESTION: 045 (1.00)

While at 100% power the Unit 2 operator observes the following plant conditions.

- "INVERTER TROUBLE" alarm.
- Loss of control rod position indication.

Which one of the following describes the expected plant response?

a. Turbine bypass valves will fail closed if the main turbine trips and its speed coasts below 1650 rpm.

b. Reactor recirc pumps will trip if pump speed is greater than 30%.

c. RFP "C" will lockup and all RFP minimum flow recirc valves will fail open.

d. The hydraulic jack for each running RFP will automatically take control of RFP speed and lock pump speed at its current value.

QUESTION: 046 (1.00)

Unit 3 is operating at 100% power when the operator observes the following plant conditions:

- CRD charging water header pressure is 1490 psig.
- Accumulator trouble indicating lights are lit on the following withdrawn control rods:

  Rod 10-19
  Rod 18-27
  Rod 42-49
  Rod 38-07

Which one of the following is the required action that must be directed by the SRO? (Figure T-LOT-0080-3, Full Core Display, attached for reference.)

a. Direct a reactor engineer to determine shutdown margin (SDM), if a SDM of 0.38% dk/k cannot be met, then place the reactor in COLD SHUTDOWN within 48 hours.

b. Exercise these four rods with normal CRD drive pressure, if the four rods cannot be moved then place the reactor in COLD SHUTDOWN within 48 hours.

c. Declare the control rods inoperable and place the reactor in COLD SHUTDOWN within 24 hours.

d. Exercise each partially or fully withdrawn operable control rod at least one notch within 24 hours.

QUESTION: 047 (1.00)

ON-106 "Stuck Control Rod" directs the operator to lower CRD drive pressure to 150 psid if attempts to move a stuck rod at elevated drive pressures was unsuccessful.

Which one of the following is the reason for this step?

a. To initiate rod movement by reducing CRDM seal swelling or binding.

b. To reduce the discharge head on the CRD pump to prevent CRD pump seal damage.

c. To reduce cooling water flow and allow the CRDM seals to heat up in an attempt to free any binding.

d. To allow the CRDM collet fingers to relax in the event they are binding.

QUESTION: 048 (1.00)

Unit 3 is operating at 100% power and experiencing a loss of instrument air pressure in the Turbine Building.

Which one of the following describes the expected plant response?

a. The hotwell reject valves will fail open, eventually causing a trip of the condensate pumps on low suction pressure.

b. The condensate short path recirculation valve fails open, causing a reduction in RFP suction pressure and discharge capacity.

c. The RFP minimum flow recirc valves will fail open, causing a reduction in reactor feedwater flow.

d. The condensate filter demin inlet and outlet valves will fail closed and the demin bypass valve will open, causing a trip of the RFPs.

QUESTION: 049 (1.00)

Unit 2 has been operating at 100% power for several weeks. When the SRO reviews the logs, he notes that nitrogen makeup to the drywell has been abnormally high. Every day for the past three days, nitrogen makeup has been required to maintain drywell pressure above 0.25 psig and oxygen concentration below 3.0%.

Which one of the following is the action that must be directed by the SRO?

a. Correct the cause of the daily de-inerting within 24 hours or place the reactor in COLD SHUTDOWN within following 24 hours.

b. Commence a shutdown per GP-3 to place the reactor in COLD SHUTDOWN within 24 hours.

c. Perform an integrated leak rate test of the drywell and the suppression chamber over the next 24 hours.

d. De-inert the primary containment within 24 hours and place the reactor in COLD SHUTDOWN within the following 24 hours.

QUESTION: 050 (1.00)

Which one of the following is the basis for the Technical Specification requirement to scram the reactor if Torus water temperature reaches 110 deg. F?

To ensure sufficient heat capacity ________.

a. to prevent boiling in the Torus if the plant must be shutdown from power with a stuck open SRV.

b. to prevent boiling in the Torus if a full power ATWS occurs and MSIVs cannot be opened.

c. to ensure complete condensation of the high energy steam escaping from the reactor during a LOCA.

d. to ensure SRV tailpipe hydraulic discharge loads are within design limits if an emergency depressurization must be performed.

QUESTION: 051 (1.00)

Unit 2 is operating at 100% power when a plant transient occurs. Plant conditions are as follows:

- APRMs decrease by 8%.
- Reactor total steam flow decreases by 1E6 lbm/hr..
- Main Generator output decreases by 70 MW.
- Core pressure drop decreases by 4 psid.
- Reactor recirculation loop "A" drive flow increases by 2,000 gpm.

Which one of the following is the cause of this transient?

a. Main steam line "A" SRV lifts and sticks partially open.

b. Reactor recirculation pump "B" shaft shear.

c. Reactor recirculation loop "A" flow instrumentation failure.

d. Jet pump failure in reactor recirculation loop "A".

QUESTION: 052 (1.00)

Entry into T-103, Secondary Containment Control, requires verification of the isolation of the Reactor Building and Refuel Floor Ventilation and the initiation of SBGTS for a Reactor Building Ventilation Exhaust radiation above the high alarm setpoint.

Which one of the following is the reason for these actions?

a. To maintain a negative Reactor Building-to-drywell differential pressure.

b. To maintain a negative Reactor Building-to-atmosphere differential pressure.

c. To rapidly reduce the airborne radioactivity in the Reactor Building for personnel access.

d. To direct the effluent to the vent stack to be monitored for offsite radiation releases.

QUESTION: 053 (1.00)

Unit 3 is operating at 25% power with irradiated fuel handling in progress on the Refuel Floor. Reactor Building and Refuel Floor Ventilation has been isolated for maintenance and SBGT train "A" have been placed in operation. The operator then observes the following plant conditions:

- SBGT train "A" flow rate is 11,300 cfm.
- Reactor Building to atmospheric dp is -0.05 inches of water.

Which one of the following is the action that must be directed by the SRO?

a. Place both trains of SBGT in operation and restore Reactor Building-to-atmosphere dp to less than -0.15 inches of water.

b. Stop SBGT train "A" and restore Reactor Building Ventilation and Reactor Building-to-atmospheric dp to less than -0.20 inches of water.

c. Place the unit in HOT SHUTDOWN within 12 hours and suspend irradiated fuel handling operations within 36 hours.

d. Immediately suspend irradiated fuel handling operations and place the unit in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within 36 hours.

QUESTION: 054 (1.00)

A reactor startup is in progress and the operator is withdrawing control rods to position 48 using the ROD NOTCH OVERRIDE switch when the following indications are received:

- "ROD DRIFT" alarm.
- "ROD OVERTRAVEL" alarm.
- No rod position is indicated.

Which one of the following will cause the above indications?

a. The rod has drifted beyond the last even numbered position and is still settling to position 48.

b. The operator provided a withdraw signal to the rod for an excessive period of time after reaching position 48.

c. The rod is uncoupled and its position in the core is unknown.

d. The Reactor Manual Control rod drive timer has malfunctioned.

QUESTION: 055 (1.00)

While executing ON-108 "Low CRD Scram Air Header Pressure" the operator is directed to scram the reactor and enter T-100 "Scram" if more than one control rod begins drifting into the core.

Which one of the following is the basis for this action?

a. To minimize the challenge to the fuel cladding integrity

b. To prevent the scram discharge instrument volume from filling

c. To ensure CRD HCU scram outlet valves open before the scram inlet valves to prevent damage to the CRDM

d. To ensure CRD HCU accumulator pressure is available to scram the rods

QUESTION: 056 (1.00)

Which one of the following describes a plant condition that could cause airborne plant release rates to exceed the Technical Specification limits?

a. Continuing to operate the SJAE when reactor power is less than 5% due to insufficient dilution flow.

b. Continuing to operate the SJAE when reactor power is less than 5% due to insufficient flow to cool the offgas adsorbers.

c. Starting the mechanical vacuum pump when reactor power is greater than 5% because it discharges directly to the offgas stack.

d. Starting the mechanical vacuum pump when reactor power is greater than 5% because it discharges directly to the Turbine Building Ventilation discharge.

QUESTION: 057 (1.00)

Which one of the following is the reason that ON-113, Loss of RBCCW, cautions the operator to maintain CRD seal purge to the reactor recirc pumps in service during a loss of RBCCW?

a. To reduce the chance of radioactive reactor coolant from leaking into the RBCCW pump coolers

b. To prevent hot reactor coolant from overheating the pump shaft and the lower motor bearing

c. To prevent recirc pump bearing failure due to thermal shock when RBCCW is restored

d. To restrict the flow of hot reactor coolant into the recirc pump seals

QU:STION: 058 (1.00)

OY-102 "Air Ejector Discharge High Radiation" directs the operator to maintain air ejector discharge radiation levels below 700 mr/hr.

Which one of the following alarms would indicate that the air ejector discharge radiation level has reached or exceeded 700 mr/hr?

a. AIR EJECTOR DISCHARGE HI RADIATION

b. AIR EJECTOR DISCHARGE HI HI RADIATION

c. VENT EXH STACK RAD MONITOR HI

d. VENT EXH STACK RAD MONITOR HI HI

QUESTION: 059 (1.00)

Unit 2 has scrammed on a main steam line high radiation condition. The operator observes the following plant conditions:

- Off Gas radiation is 4.1E+05 mr/hr and decreasing by 0.5E+05 mr/hr every minute.
- Reactor coolant sample taken 20 minutes ago has an activity of 500 uCi/gm dose equivalent I-131.
- Containment radiation indicates 4.5E+03 R/hr on two independent monitors but is decreasing by 0.25E+03 R/hr every minute.

Which one of the following is the emergency classification that must be declared by the SRO?

a. General Emergency

b. Site Area Emergency

c. Alert

d. Unusual Event

QUESTION: 060 (1.00)

Due to an unidentified primary boundary leak, Unit 2 was rapidly shutdown and is currently at 600 psig with a continued depressurization at the maximum allowed cooldown rate in progress. The operator has just completed the following actions:

- Reported Drywell Floor Drain Sump leak rate is currently 25 gpm.
- Determined that 10 minutes ago the leak rate was 65 gpm.

Which one of the following is the action that the Emergency Director must take, assuming the plant is currently in an Unusual Event?

a. Escalate the emergency classification to a Site Area Emergency and make the required off site agency reports.

b. Escalate and terminate the emergency classification in a single report to off site agencies.

c. Classify and report the previous leak rate to the required off site agencies but continue in the Unusual Event.

d. Record the previous leak rate but do not classify or report it to off site agencies and continue in the Unusual Event.

QUESTION: 061 (1.00)

Which one of the following is the reason for initiating a DC load shed as soon as possible per SE-11 (Sheet 5), "Loss of Off-Site Power with No Diesel Generators Available"?

a. To prevent spurious operation of ECCS components during restoration of AC power

b. To extend the amount of time that the DC loads required for adequate core cooling are available

c. To preserve the DC power that is necessary to initiate a Diesel Generator start to rated rpm and voltage

d. To reduce the buildup of hydrogen in the battery rooms to prevent a hydrogen combustion hazard

QUESTION: 062 (1.00)

Unit 2 Control Room has been evacuated and the unit is being cooled down from the HPCI ASD panel. The operator observes the following parameters:

TIME	PRESSURE	REACTOR LEVEL (LI 2-2-3-112)
2245	800 psig	-19 inches
2300	700 psig	-18 inches
2315	600 psig	-17 inches
2330	500 psig	-16 inches
2345	400 psig	-15 inches

Which one of the following describes the trend of ACTUAL reactor water level over the past hour and the current water level? (SE-10 "Shutdown Outside Control Room" Attachment 9, Figure 1, Actual Rx Level as a Function of Rx Press and Indicated Level, is attached.)

a. Level has been decreasing and is currently less than zero inches.

b. Level has been increasing and is currently less than zero inches.

c. Level has been decreasing and is currently between 0 and +40 inches.

d. Level has been increasing and is currently between 0 and +40 inches.

QUESTION: 063 (1.00)

Following a transient, Unit 2 conditions are as follows:

- RPV pressure is 1000 psig and steady.
- Torus temperature is 163 deg. F., increasing at 0.3 deg. F/min.
- Torus level is 11.0 feet and steady.
- Drywell temperature is 230 deg. F., increasing at 0.5 deg. F/min.
- Drywell pressure is 4 psig.
- Torus pressure is 3 psig.

Which one of the following is the action that must be taken per the Trip Procedures?

a. Perform a normal depressurization and initiate torus spray but NOT drywell spray.

b. Perform a normal depressurization and initiate both torus and drywell sprays.

c. Perform an emergency blowdown and initiate torus spray but NOT drywell spray.

d. Perform an emergency blowdown and initiate both torus and drywell sprays.

QUESTION: 064 (1.00)

T-102 "Primary Containment Control" provides the following caution:

Operation of HPCI or RCIC with Torus suction AND Torus temperature above 190 deg. F may result in equipment damage.

Which one of the following is the basis for this caution?

a. Insufficient pump cooling may result in impeller binding and mechanical seal failure.

b. Turbine exhaust check valve may chatter resulting in valve failure or steam hammer.

c. Inadequate NPSH may result in pump cavitation causing excessive vibration and pitting damage of pump components.

d. The torus is no longer capable of condensing all of the turbine exhaust resulting in increased torus pressures.

QUESTION: 065 (1.00)

Select the choice below that completes the following statements.

A LOCA is in progress, causing highly elevated secondary containment temperatures. The indicated reactor vessel water level will be ________.

a. lower than actual due to boiling in the reference legs.

b. higher than actual due to a rupture failure of the diaphragm in the dp cell.

c. inaccurate because boiling may occur in both the variable and reference legs.

d. inaccurate because the dp cell may experience steam binding

QUESTION: 066 (1.00)

An abnormal event is in progress on Unit 3. The unit has been successfully scrammed and plant conditions are as follows:

- Torus temperature is 135 deg. F.
- Torus level is 16.2 feet and slowly increasing.
- RPV pressure is 1000 psig and steady.
- RPV water level is 10 inches.

Which one of the following is the action that must be taken per the Trip Procedures and the basis for this action?

a. Perform an emergency blowdown to prevent breaking the SRV tailpipes.

b. Perform an emergency blowdown to prevent overfilling and overpressurizing the Torus.

c. Perform a RPV depressurization to prevent breaking the SRV tailpipes.

d. Perform a RPV depressurization to prevent overfilling and overpressurizing the Torus.

QUESTION: 067 (1.00)

An ATWS is in progress on Unit 2. After the insertion of the initial scram signal, the operator observes the following:

- CRD pumps are tripped.
- RPV pressure is 350 psig.
- About half of the blue scram lights are NOT lit.
- Control Rod Drive Scram Solenoid Group 1 and 3 in RPS Channel B are lit.
- SCRAM DISCH VOL HI WATER LEVEL TRIP 50 GAL alarm has been received.
- All ACCUMULATOR TROUBLE alarm lights on the full core display are lit.

Which one of the following alternate rod insertion methods is MOST likely to successfully insert any withdrawn rods?

a. Removal of the group scram solenoid fuses per T-213 "Scram Solenoid Deenergization".

b. Resetting the scram, draining the scram discharge volume, and manually scramming per T-216 "Rod Insertion by Manual Scram or Individual Scram Test Switches".

c. Placing the individual scram test switches for the withdrawn rods to the full down position per T-213 "Scram Solenoid Deenergization".

d. Manual venting of the CRD withdrawal lines per T-215 "Control Rod Insertion by Withdraw Line Venting".

QUESTION: 068 (1.00)

A LOCA is in progress on Unit 2. The plant has been successfully scrammed and plant conditions are as follows:

- RPV water level:
  Shutdown Range (LI-86) indicates +10 inches.
  Narrow Range (LI-94A,B,&C) indicate +5 to +8 inches.
  Wide Range (LI-85A&B) indicate -10 inches.
  Fuel Zone instruments indicate -25 inches.
- Drywell temperature (TI-2-501):
  Point 126 indicates 270 deg. F.
  Point 127 indicates 267 deg. F.
- Reactor Building temperature (TR-2-13-139):
  Point 22 indicates 155 deg. F.
- RPV pressure is 200 psig.

Which one of the following instruments is NOT AVAILABLE for RPV level indication?

a. Shutdown Range

b. Wide Range

c. Narrow Range

d. Fuel Zone

QUESTION: 069 (1.00)

Which one of the following reactor water level anomalies can ONLY be detected or observed during a rapid RPV depressurization below 450 psig?

a. Boiling in the variable leg.

b. Boiling in the reference leg.

c. Degassing in the reference leg.

d. Hydrogen buildup in the condensing chamber.

QUESTION: 070 (1.00)

Following a LOCA and a partial ATWS on Unit 3, containment conditions are as follows:

- Drywell pressure is 35 psig and steady.
- Drywell temperature is 350 deg. F. and slowly decreasing.
- Torus pressure is 15 psig and slowly increasing.
- Torus temperature is 210 deg. F. and steady.
- RPV pressure is 1000 psig and steady.
- Torus level is 14.7 feet but steadily decreasing due to an unisolable leak.

Which one of the following is the LOWEST Torus level at which RHR can continuously be operated at a maximum flow rate of 10,000 gpm per pump in each loop?

a. 12.7 feet

b. 9.6 feet

c. 8.0 feet

d. 6.5 feet

QUESTION: 071 (1.00)

An ATWS is in progress on Unit 2. Plant conditions are as follows:

- MSIVs are open.
- Main generator load is 300 MWe.
- Reactor recirc pumps are at 25% speed.
- APRMs indicate 25%.

Which one of the following is the reason that T-101, Step RC/Q-15, directs the operator to trip the reactor recirc pumps at least 10 seconds apart?

a. To ensure RPV water level swell is not enough to reach the reactor feed pump and main turbine trip setpoint.

b. To ensure RPV water level swell is not enough to undesireably flood the main steam lines and result in carryover to any operating turbines.

c. To ensure RPV water level shrinkage is not enough to undesireably initiate HPCI or RCIC.

d. To ensure RPV water level shrinkage is not enough to cause an overspeed trip condition on any running reactor feed pumps.

QUESTION: 072 (1.00)

An event has occurred on Unit 2 that has resulted in the lowering of Torus level. Plant conditions are as follows:

- Torus level is 10.0 feet and steady.
- Torus temperature is 135 deg. F.
- RPV pressure is 1000 psig.

Which one of the following describes the adverse consequences of plant equipment operation at this Torus level?

a. Operation of RCIC will increase Torus pressure causing RCIC to trip on high turbine exhaust pressure.

b. Operation of HPCI will increase Torus pressure and threaten containment integrity.

c. Opening SRVs will rapidly increase Torus pressure.

d. A LOCA will result in overpressurizing the containment.

QUESTION: 073 (1.00)

T-117 "Level Power Control" is being executed on Unit 2. Step LQ-20 directs the operator to restore and maintain RPV level above -172 inches with Condensate/Feedwater, CRD, RCIC, HPCI, and LPCI, regardless of ECCS suction requirements.

Which one of the following is the reason that Step LQ-20 EXCLUDES the use of Core Spray to recover RPV level?

a. To ensure one ECCS injection source is protected from damage due to vortexing or insufficient NPSH.

b. Because there are no heat exchangers in the Core Spray system to remove core decay heat.

c. To minimize the injection of cold water directly into the core which could add significant positive reactivity.

d. To ensure the natural circulation that has been established in the RPV is not interrupted.

QUESTION: 074 (1.00)

T-102 "Primary Containment Control" is being executed. Step PC/P-9 indicates:

IF Drywell pressure drops below 2 psig,
THEN Terminate drywell sprays.

Which one of the following is the basis for Step PC/P-9?

a. To prevent cycling the Drywell-to-Torus vacuum breakers.

b. To prevent opening the Reactor Building-to-Torus vacuum breakers.

c. To prevent developing an excessive differential pressure between the Drywell and the Torus.

d. To prevent developing an excessive differential between the water level inside the downcomer and the Torus.

QUESTION: 075 (1.00)

In T-101-2 "RPV Control", if SRVs are cycling, the operator is directed to manually open SRVs and control RPV pressure between 960 and 1060 psig.

Which one of the following is the reason for establishing the minimum RPV pressure at 960 psig?

a. To ensure the turbine bypass valves do not have the opportunity to stick closed

b. To prevent MSIVs from closing on low main steam line pressure

c. To minimize the amount of steam that is sent to the suppression pool

d. To prevent excessive loss of reactor coolant inventory

QUESTION: 076 (1.00)

An accident is in progress on Unit 2. Torus sprays have been initiated but CANNOT maintain torus pressure below 9 psig.

Which one of the following is the reason for initiating drywell sprays when torus pressure cannot be maintained below 9 psig?

a. To prevent chugging and eventual fatigue failure at the junction of the downcomers and the vent header

b. To prevent excessive cyclic stresses on the SRV tailpipes

c. To prevent exceeding the torus pressure limit if a DBA LOCA occurs

d. To prevent exceeding the structural design limits of the torus if an emergency depressurization is required

QUESTION: 077 (1.00)

Following a reactor scram, T-100 "Scram", Step S-10, directs the following operator action:

VERIFY GEN LOCKOUT.

Which one of the following is the method that the operator will use to perform step S-10?

a. Check that the main generator disconnects indicate open.

b. Check that the 13 KV system has transferred from the unit to the startup transformers.

c. Check that the emergency transformers have transferred to the Startup Bus.

d. Check that the main generator output and field breakers indicate open.

QUESTION: 078 (1.00)

Unit 3 is being maintained in Cold Shutdown with the RPV head bolts tensioned per GP-12 "Core Cooling". One loop of RHR is in shutdown cooling with one pump operating at a flow rate of 8,000 gpm.

Which one of the following sets of indications should be used by the operator to ensure that shutdown cooling flow through the core is sufficient to prevent thermal stratification and remove decay heat? (Refer to GP-12, Figure 3, Vessel Level vs. Flow Through One Recirc Loop.)

	LI-86 Shutdown Range	LI-94 Narrow Range
a.	30 inches	30 inches
b.	35 inches	35 inches
c.	35 inches	38 inches
d.	38 inches	35 inches

QUESTION: 079 (1.00)

T-100 "Scram", Step S-8, directs the operator to VERIFY 13KV TRANSFER. Following a scram, the operating crew has reached this step and the main turbine is still on line and is not expected to trip on reverse power for several minutes.

Which one of the following is an advantage of manually transferring the 13KV system to the Startup Source?

a. To prevent tripping the reactor recirc pumps.

b. To prevent tripping the circulating pumps.

c. To prevent tripping the condensate and feedwater pumps.

d. To prevent tripping the station air compressors.

QUESTION: 080 (1.00)

Step SC/L-1 of T-103 "Secondary Containment Control" directs the following action:

MONITOR AND CONTROL SECONDARY CONTAINMENT WATER LEVELS.

Which one of the following is an appropriate operator action that would be taken to accomplish this step?

a. Isolate any leaking ECCS system even though it is currently the only system available to maintain RPV level.

b. Direct maintenance to immediately install sandbags around the outside of the room doors.

c. Plot the rate of increase in the sump water levels to determine whether sump capacity is sufficient.

d. Start all available sump pumps and operate them to remove water from the sumps.

QUESTION: 081 (1.00)

An accident has resulted in the following plant conditions:

- Drywell pressure is 30 psig.
- Drywell temperature is 325 deg. F.

Which one of the following is the reason for initiating drywell sprays within the safe region of the Drywell Spray Initiation Limit Curve?

a. To ensure that the Reactor Building-to-Torus Vacuum breakers can open in time to prevent implosion of the primary containment

b. To ensure that the Torus-to-Drywell Vacuum breakers will prevent implosion of the primary containment

c. To preclude the occurrence of a saturated drywell atmosphere to prevent an evaporative cooling pressure drop

d. To preclude the occurrence of a saturated drywell atmosphere to prevent a convective cooling pressure drop

QUESTION: 082 (1.00)

During a LOCA on Unit 3, plant conditions have degraded to the following:

- RPV pressure is 750 psig and decreasing at 5 psig/min.
- RPV level is -172 inches and decreasing at 2 inches/min.
- All high pressure, including condensate and ECCS injection systems are inoperable.
- The refuel water transfer system is the only alternate injection subsystem available, and it has been aligned with one pump running.

Which one of the following is the NEXT action that must be directed by the SRO?

a. Initiate injection with the refuel water transfer system.

b. Initiate steam cooling until level decreases to -210 inches.

c. Initiate steam cooling until RPV pressure decreases to 400 psig.

d. Initiate an emergency blowdown.

QUESTION: 083 (1.00)

Following an outage, fuel is being reloaded into the core. A problem has been discovered with the control rod drive mechanism for the peripheral rod 18-03. To repair the CRDM, the rod must be fully withdrawn from the core. Plant conditions are as follows:

- SRMs A, B, C, and D are reading 7, 6, 8, 5 cps respectively.
- No fuel has been loaded around rod 18-03.
- Mode Switch is locked in REFUEL.
- All other control rods are operable and fully inserted.
- A Shutdown Margin of 0.3% dk with the strongest rod fully withdrawn in addition to the withdrawal of rod 18-03 has been verified.

Which one of the following is an acceptable action that may be taken by the SRO to fully withdraw rod 18-03 for maintenance and then continue the fuel loading?

a. Declare rod 18-03 inoperable, bypass the one rod out interlock on rod 18-03, and allow the full withdrawal of the rod for repair.

b. Bypass the one rod out interlock for all control rods and then allow rod 18-03 to be fully withdrawn for repair.

c. Declare rod 18-03 inoperable and place the mode switch in STARTUP to allow the rod to be fully withdrawn for repair.

d. Place the mode switch in SHUTDOWN and bypass the one rod out interlock for rod 18-03 to allow the rod to be fully withdrawn for repair.

QUESTION: 084 (1.00)

T-116 "RPV Flooding" is being executed following a complete loss of level indication. Plant conditions are as follows:

- All control rods have been fully inserted for 5 hours.
- RPV level reference legs have been filled per T-260 "Filling RPV Level Reference Legs".
- RPV level instrumentation is available.
- TR-3-13-139 point 22 indicates 150 deg. F.
- TI-3501 points 126 and 127 indicate 199 and 208 deg. F., respectively.
- 5 SRVs have been continuously open and have maintained RPV pressure between 100 and 110 psig for 60 minutes.
- Torus pressure is 18 psig.

Which one of the following is the NEXT action to be directed by the SRO?

a. Continue injection to maintain current reactor pressure for at least 30 more minutes before terminating injection.

b. Increase injection rate to raise RPV pressure to greater than 230 psig for at least 90 minutes before terminating injection.

c. Terminate all injection to RPV for a maximum of 5 minutes to observe for the restoration of RPV level indication.

d. Terminate all injection to RPV for a maximum of 7 minutes to observe for the restoration of RPV level indication.

QUESTION: 085 (1.00)

Following a LOCA, which one of the following will result in the most rapid production of hydrogen in the drywell?

a. High temperature steam/nitric acid corrosion of stainless steel components

b. Radiolytic decomposition of water by fission product gammas in the torus

c. Boric acid corrosion of various metal components when SBLC is injected

d. Zirc water reaction of the fuel cladding when the core is uncovered

QUESTION: 086 (1.00)

A reactor operator has been off-shift for six months.

Which one of the following describes the MINIMUM actions which must be taken before the operator may return to the normal shift rotation?

a. Work the first 6 hours of a shift for 7 days under the supervision of a qualified RO to include a complete plant tour.

b. Work five 12-hour shifts under the supervision of a qualified RO to include participation in shift turnovers and pre- and post-ALARA job briefings.

c. Work 40 hours as an extra operator on-shift under the supervision of a qualified SRO to include pre- and post- job briefings.

d. Work five 8-hour shifts under the supervision of a qualified RO to include a complete plant tour and participation in shift turnovers.

QUESTION: 087 (1.00)

A welding cable will be run down a Reactor Building stairwell from one elevation to another and then run an additional 30 feet across the lower elevation for maintenance that will occur during a single day shift.

Which one of the following is a requirement for running this cable in accordance with A-30 "Plant Material Condition and Housekeeping Controls"?

a. Route and tie the cables in an elevated position.

b. Attach ONE "WORK INCOMPLETE TAG" at the source of the cable.

c. Ensure Fire Protection Group reviews the configuration to ensure the Fire Protection Plan is not violated.

d. Ensure a 10CFR50.59 review is completed to ensure the Class 1E separation criteria are not violated.

QUESTION: 088 (1.00)

Select the choice below that completes the following statement in accordance with A-14 "Plant Modifications".

The reactor coolant pressure boundary includes all of those pressure-containing components connected to the reactor coolant system (RCS), up to and including ________.

a. the second of two valves, that are normally closed during normal reactor operation, located in a vent line connected to the RCS which does NOT penetrate primary containment

b. the first of two valves, that are normally closed during normal reactor operation, located in a drain line connected to the RCS which does NOT penetrate primary containment

c. the RCIC turbine steam inboard isolation valve (MOV-15)

d. the inboard MSIVs

QUESTION: 089 (1.00)

In accordance with the C & T Manual, which one of the following describe operation of Special Condition Tag (SCT) equipment?

a. Only Shift Manager shall authorize operation of SCT equipment at the request of SCT clearance holder.

b. The plant reactor operator (PRO) shall authorize operation of SCT equipment at the request of the responsible work group.

c. The SCT clearance holder may authorize operation of SCT equipment at the request of the responsible work group.

d. The Shift Manager may delegate authorization to operate SCT equipment to the SCT clearance holder for the duration of the job.

QUESTION: 090 (1.00)

During the performance of a surveillance test, the operator encounters difficulty in performing a step and obtaining the required procedural response. The Shift Manager has initiated and authorized a Troubleshooting, Minor Rework, and Testing (TMT) form to troubleshoot the problem.

Which one of the following is a course of action that is allowed while using the TMT?

a. If minor equipment degradation is identified, minor maintenance may be performed under the TMT.

b. If any testing on environmentally qualified equipment will be performed, I&C Supervision approval is required.

c. The TMT can be used to accomplish the step if the step is typographically incorrect and the intent of the step is met.

d. The TMT can be used to accomplish a temporary plant modification to satisfy the prerequisites for proper completion of the step.

QUESTION: 091 (1.00)

Due to an accident, a reactor operator is temporarily unable to meet the medical requirements of his/her license as determined by Occupational Health and Safety.

Which one of the following is the required action in accordance with A-C-10 "Operator Licenses"?

a. The NRC must be notified in writing within 30 days.

b. The operator must be removed administratively from licensed duties.

c. A conditional license request must be submitted to the NRC within 30 days.

d. The operator's license must be surrendered to the NRC.

QUESTION: 092 (1.00)

Which one of the following components can be approved for MINOR MAINTENANCE in accordance with A-C-26 "Maintenance Work Process" if WELDING will be performed on the component?

a. Non-safety-related, Augmented QA class component

b. Non-safety-related, passive QA class component

c. Safety-related component for which detailed post-maintenance testing is not required

d. Safety-related component for which detailed planning is not required

QUESTION: 093 (1.00)

Which one of the following is indicated by a "*" when looking at the QA classification provided by PIMS?

a. An Active QA component which changes state in response to imposing a design basis load demand on the system.

b. A Passive QA component for which no mechanical motion occurs when imposing a design basis load demand on the system.

c. The component is non-safety related but the utility has made a regulatory commitment associated with the component.

d. The QA classification of the component may require an evaluation by the Engineering Department.

QUESTION: 094 (1.00)

A surveillance test is in progress.

Which one of the following conditions would require an operator to STOP a surveillance TESTING procedure IMMEDIATELY in accordance with A-C-43 "Surveillance Testing Program"?

a. Testing has placed the system outside of Technical Specification limits but the system can be restored to specifications within 1 hour after the test is completed.

b. A system parameter has been observed to be outside of the normal band, but the procedure step provides guidance for adjustment of the abnormal system parameter.

c. The procedure does not have a blank for the recording of a specified parameter required for satisfactory performance of the system being tested.

d. The procedure is taking much longer to perform than expected as explained in the pre-test briefing of the personnel involved in the test.

QUESTION: 095 (1.00)

Which one of the following describes a radiologically controlled area that is posted as a YELLOW ZONE?

a. A buffer area that is established around any accessible Red Zone.

b. An area that is established around any open bag containing hot particles.

c. Any area outside of the restricted area but inside the site boundary for which PECo restricts access.

d. An area in which an individual will inhale greater than 12 DAC-hours in a 40 hour work-week.

QUESTION: 096 (1.00)

An operator observes that an area is posted as a VERY HIGH RADIATION AREA and HP reports it is a Level I area.

Which one of the following radiation dose rates would be expected inside this area?

a. Greater than or equal to 1 rem/hr but less than 10 rem/hr at 30 cm.

b. Greater than or equal to 10 rem/hr but less than 500 rem/hr at 30 cm.

c. Greater than or equal to 1 rad/hr but less than 10 rad/hr at 1 meter.

d. Greater than or equal to 10 rad/hr but less than 500 rad/hr at 1 meter.

QUESTION: 097 (1.00)

Select the choice below that completes the following statement.

OM-P-3.2 indicates that, unless relieved, the SRO assigned to the control room to satisfy Technical Specification requirements ________.

a. must remain within sight of the annunciators for the unit to which he is assigned

b. must remain within audible range of the annunciators or operator at the controls

c. must limit the amount of time that he is beyond the line of sight of the operator at the controls

d. must limit the amount of time that he is outside of the control room

QUESTION: 098 (1.00)

Which one of the following describes the PRIMARY function of the STA during a plant transient?

a. To assist the CRS in identifying and interpreting plant procedures and alarm response cards.

b. To provide technical assistance and information to the shift reactor operators.

c. To determine if the transient is proceeding in accordance with the plant design and if significant parameters are within acceptable limits.

d. To assist the CRS and the Shift Manager in the review of the Technical Specifications and th Emergency Response classifications.

QUESTION: 099 (1.00)

The Shift Manager and the CRS have just determined that an LCO has been entered and are unable to notify the Senior Manager of Operations.

Which one of the following managers must be notified by the Shift Manager in accordance with OM-P-12.1 "Limiting Conditions for Operations"?

a. Operations Services Manager

b. Operations Support Group Manager

c. Plant Manager

d. Plant Maintenance Manager

QUESTION: 100 (1.00)

Select the choice below that completes the following statements in accordance with OM-P-7.6 "Fuses and Quality Parts".

An operator has just replaced a blown fuse in a circuit, and the circuit has been restored to normal. The operator must then provide the blown fuse to the ________ with a written description of the circumstances surrounding the failure.

a. Shift Technical Advisor

b. Electrical Maintenance Supervisor

c. Operations Support Group Manager

d. Instrument and Controls Supervisor

(********** END OF EXAMINATION **********)

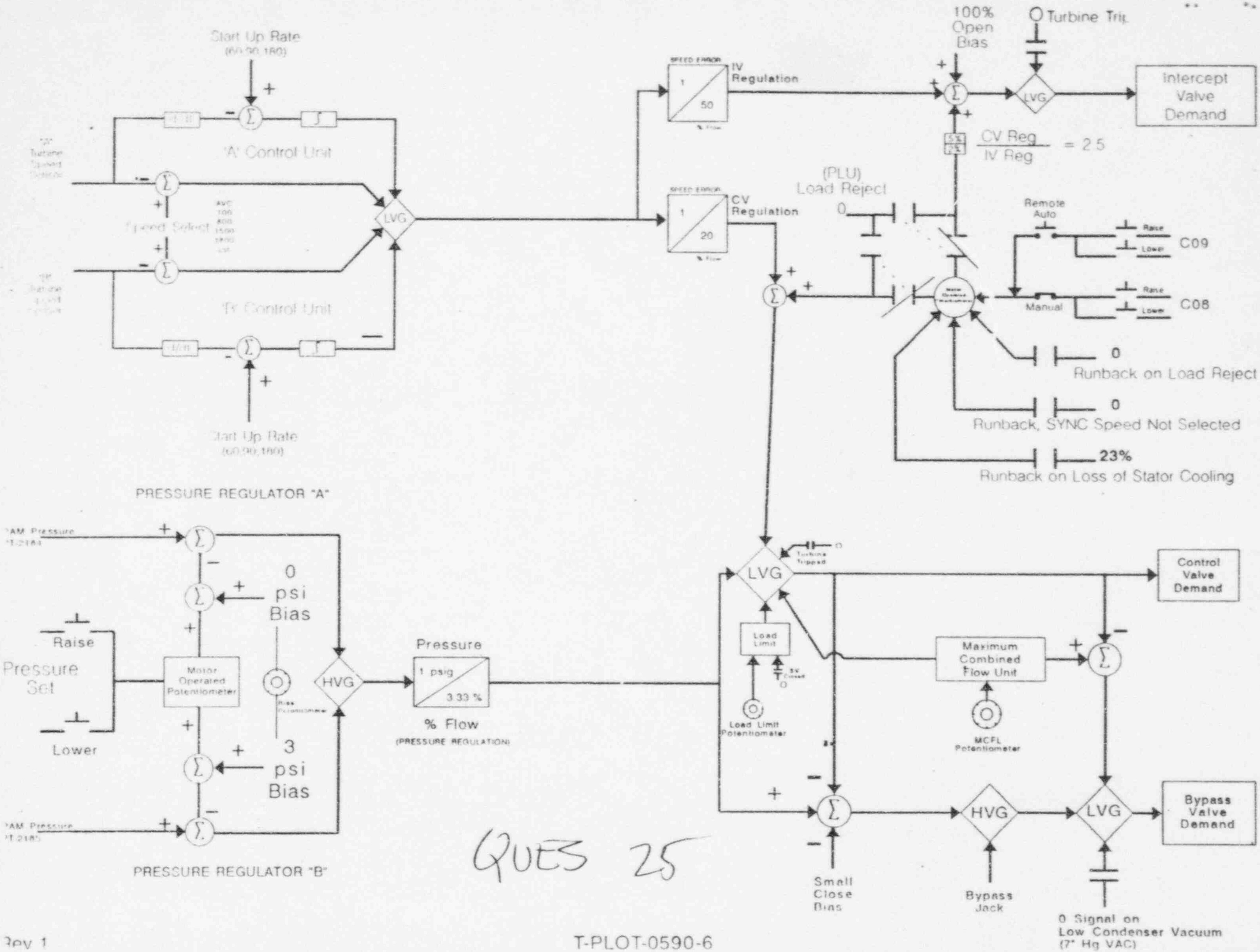

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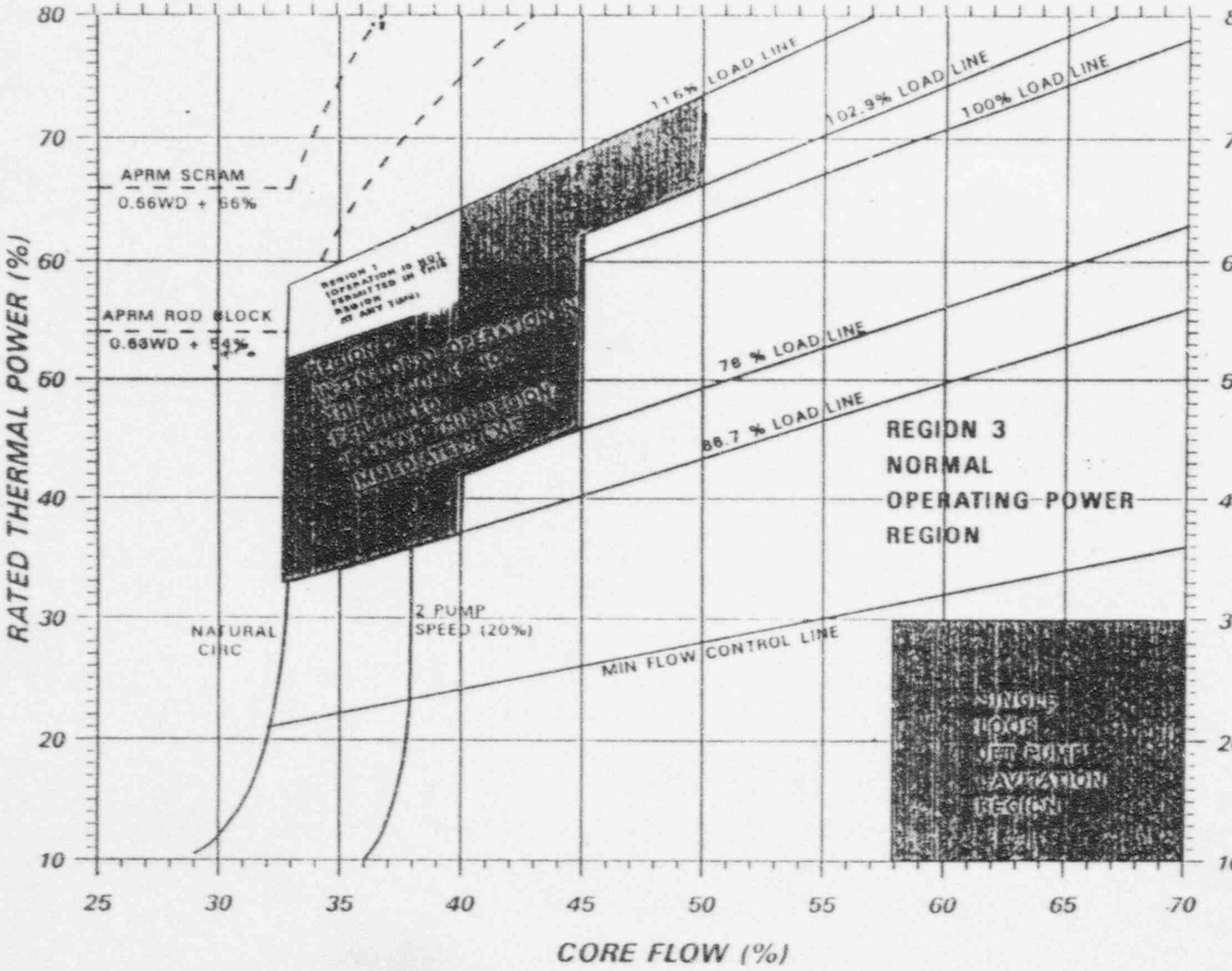
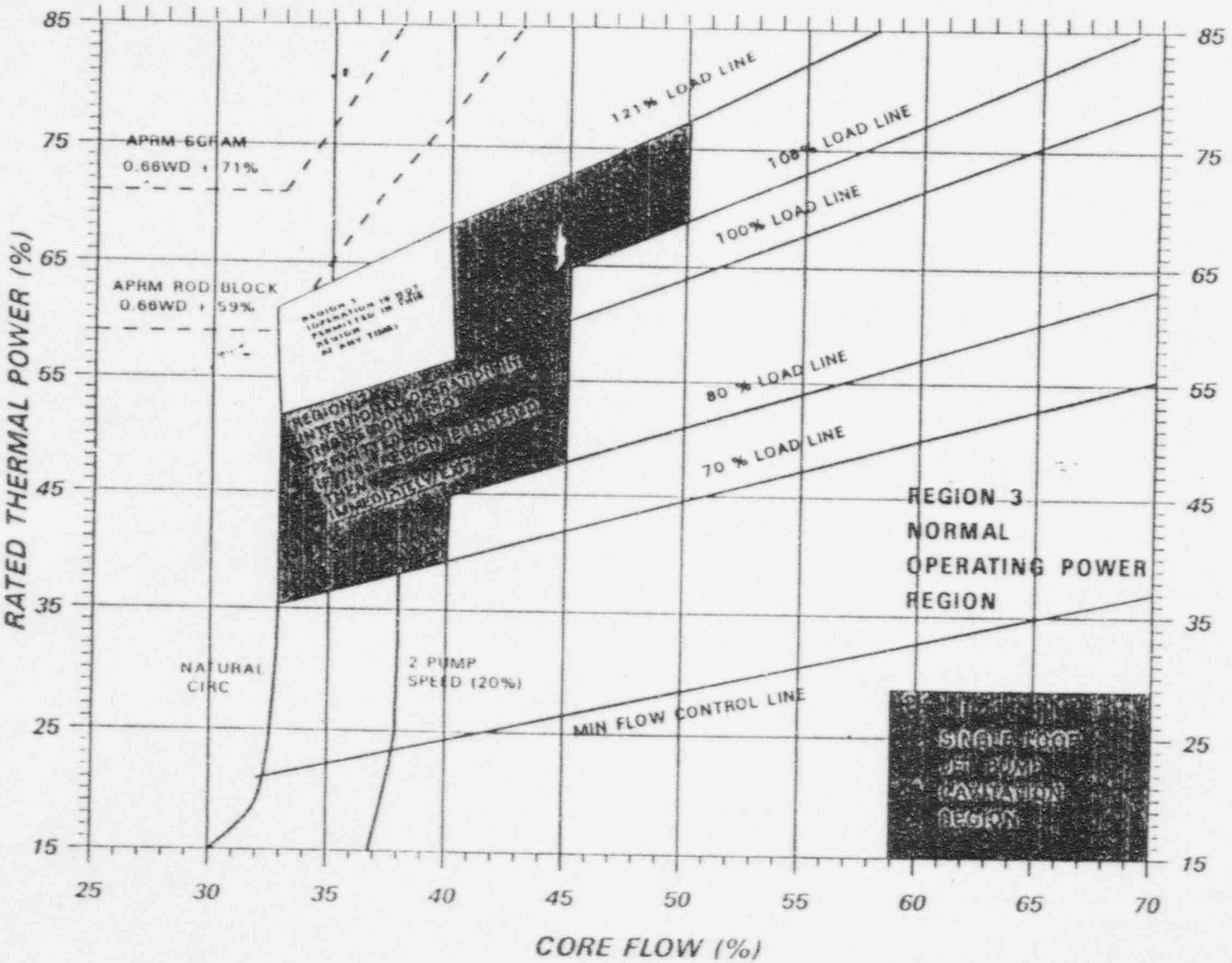
FIGURE A
UNIT 2 ONLY



QUES 33

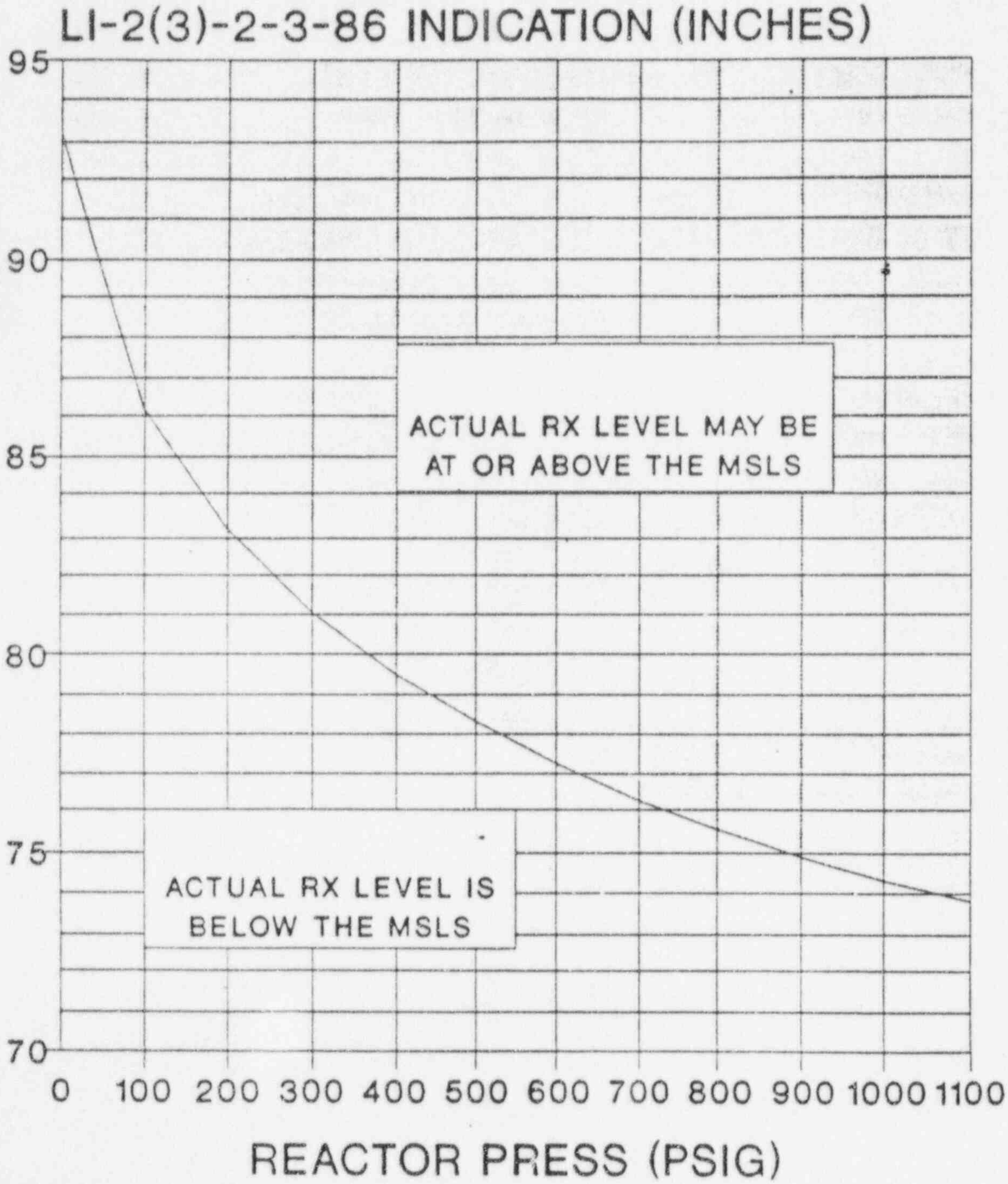
FIGURE B

UNIT 3 ONLY



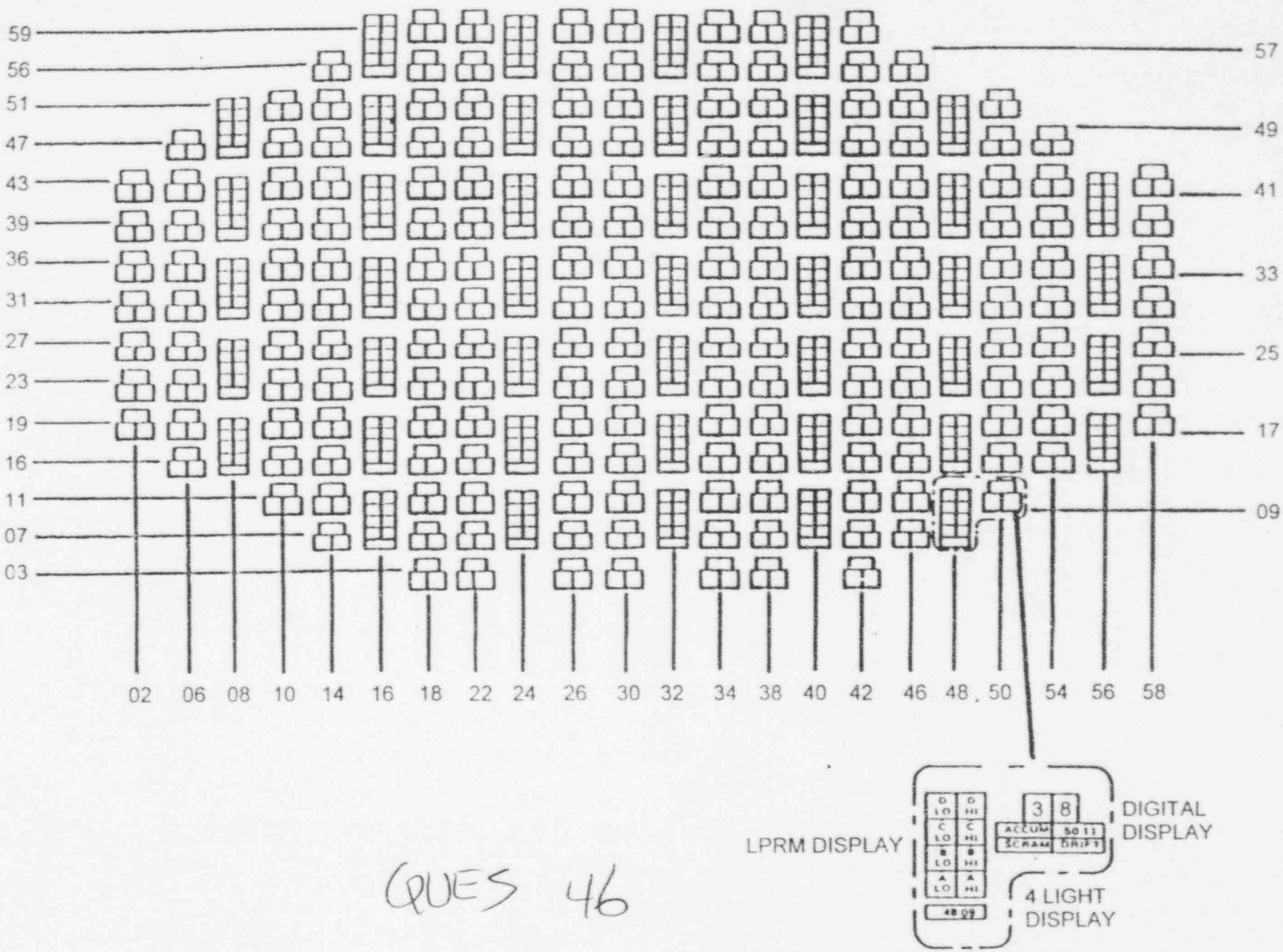
QUES 33

FIGURE 1



QUES 41

FULL CORE DISPLAY

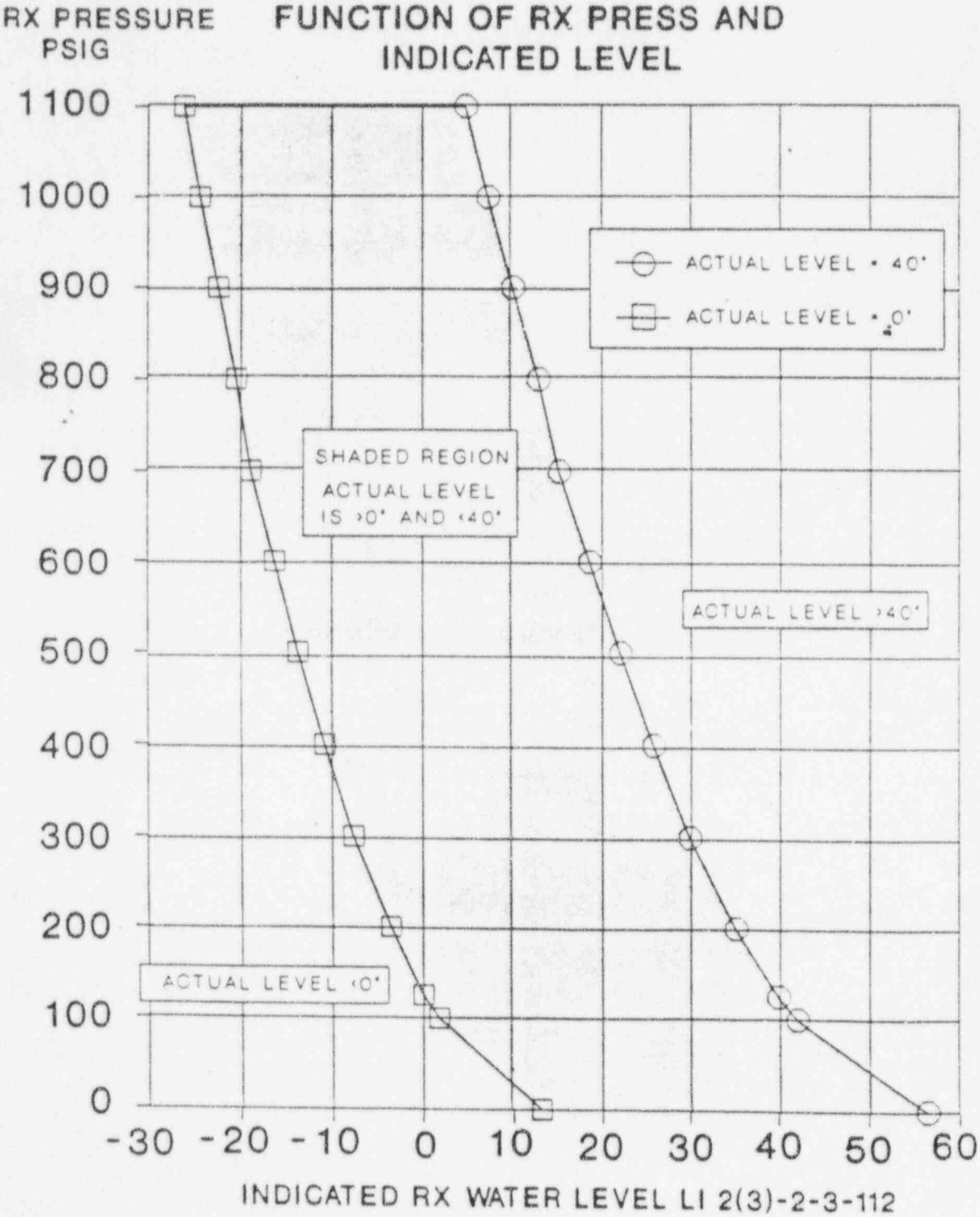


QUES 46

T-LOT-0080-3

SE-10 Attachment 9
Figure 1

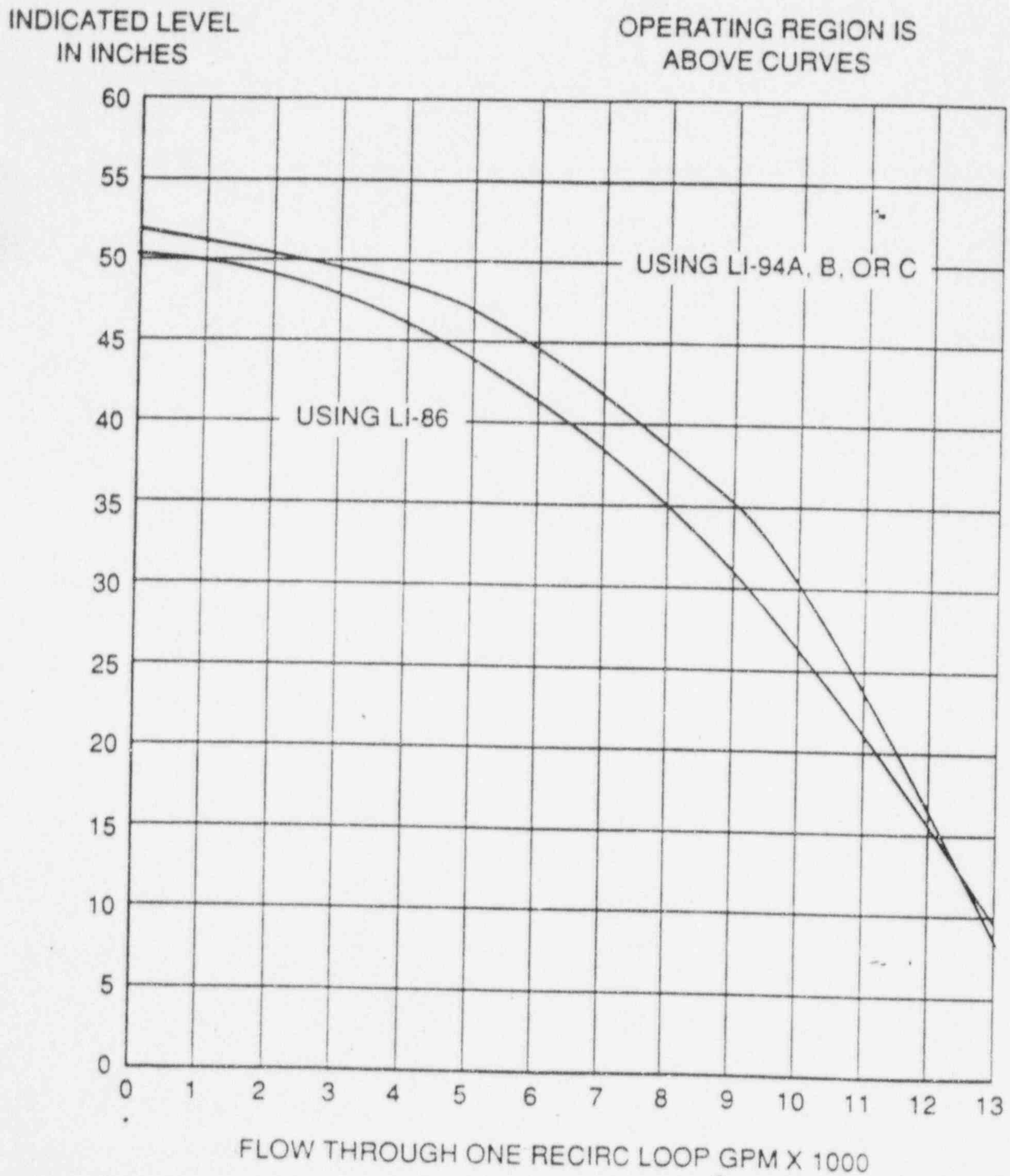
ACTUAL RX LEVEL AS A FUNCTION OF RX PRESS AND INDICATED LEVEL



QUES 62

FIGURE 3

Vessel Level Vs. Flow Through One Recirc Loop



QUES 78

ANSWER: 001 (1.00)

c.

REFERENCE:

Facility Question Bank No. 4578
LOT-0370

203000K301 [4.3/4.1]

203000A405 ..(KA's)

ANSWER: 002 (1.00)

d.

REFERENCE:

LOT-0380, SO 13.2.A-2
KA217000A401
(3.7/3.7)

217000A401 ..(KA's)

ANSWER: 003 (1.00)

b.

REFERENCE:

PLOT 0030-1, p. 28
Learning Objective 6.g

202001K502 [3.1/3.2]

202001K502 ..(KA's)

ANSWER: 004 (1.00)

d.

REFERENCE:

LOT 0070, system drawing
Learning objective 3.

201001A109 [2.9/2.8]

201001A109 ..(KA's)

ANSWER: 005 (1.00)

a.

REFERENCE:

Control Rod Hydraulic system, LOT 0070, p. 24.
Learning Objective 6.a
Facility Question Bank No. 3883

201001K101 [3.1/3.1]

201001K101 ..(KA's)

ANSWER: 006 (1.00)

a.

REFERENCE:

Reactor Core Isolation Cooling, LOT-380 p. 20-21.

217000K407 [3.6/3.6]

217000K407 ..(KA's)

ANSWER: 007 (1.00)

c.

REFERENCE:

Automatic Depressurization System, LOT 0330, p. 15, 16
Learning Objective 5.b

218000K501 [3.8/3.8]

218000K501 ..(KA's)

ANSWER: 008 (1.00)

c.

REFERENCE:

Automatic Depressurization System, LOT-0330 p. 8
Learning Objective 6.f

218000K606 [3.4/3.6]

218000K606 ..(KA's)

ANSWER: 009 (1.00)

c.

REFERENCE:

Residual Heat Removal, H-LOT 0370-1, p. 1,2

203000G009 [4.3/3.9]

203000G009 ..(KA's)

ANSWER: 010 (1.00)

b.

REFERENCE:

High Pressure Coolant Injection, LOT-0340, p. 10
Learning Objective 2c, 2d.

206000A217 [3.9/4.3]

206000A217 ..(KA's)

ANSWER: 011 (1.00)

d.

REFERENCE:

HPCI System Operating Procedure, SO 23.1, p. 2
LOT-0340, Learning Objective 5.k

206000G010 [3.9/3.8]

206000G010 ..(KA's)

ANSWER: 012 (1.00)

a. d c

REFERENCE:

P&IDs, E-26
LOT 0340, Learning Objective 3.a

206000K201 [3.2/3.3]

206000K201 ..(KA's)

ANSWER: 013 (1.00)

d.

REFERENCE:

Starting the first recirculation pump, SO 2.A.a.A-2
PLOT-0030-1, pg 30, Learning Objective 5.i

202002A401 [3.3/3.1]

202002A401 ..(KA's)

ANSWER: 014 (1.00)

c.

REFERENCE:

Condensate system, LOT-0520, p. 21
Learning Objective 7.f

256000K606 [3.3/3.3]

256000K606 ..(KA's)

ANSWER: 015 (1.00)

b.

REFERENCE:

Startup of Second or Third RFP, SO 6C.1.C-2, p. 5
Facility Exam Bank question 1659

259001G010 [3.2/3.3]

259001G010 ..(KA's)

ANSWER: 016 (1.00)

a.

REFERENCE:

PBAPS LOT-0530, pg 15
PBAPS LOT-1610, pg 4

From March, 1995 Examination

259001A204 [3.3/3.4]

259001A204 ..(KA's)

ANSWER: 017 (1.00)

b.

REFERENCE:

T-LOT-0670-21

From August 1994 Exam

264000A404 [3.7/3.7]

264000A404 ..(KA's)

ANSWER: 018 (1.00)

b.

REFERENCE:

Modified Examination Bank Q. 2771, E-1, E-188, SO 54.7.B
LOT-0660, pg 26, Figures 1 and 2, Learning Objective 4, 7, and 11
NOTE: No LPCI initiation signal exists so there is no start signal to the RHR or HPSW pumps.

262001A302 [3.2/3.3]

262001A302 ..(KA's)

ANSWER: 019 (1.00)

c.

REFERENCE:

SO 52A.1.A

264000A401 [3.3/3.4]

264000A401 ..(KA's)

ANSWER: 020 (1.00)

b.

REFERENCE:

Reactor Water Cleanup System, PLOT-0110 p. 7
Learning Objective 1.a

204000A304 [3.4/3.5]

204000G004 ..(KA's)

ANSWER: 021 (1.00)

a.

REFERENCE:

Reactor Water Cleanup, PLOT-0110, p. 11
Learning Objective 4.d

204000K404 [3.5/3.6]

204000K404 ..(KA's)

ANSWER: 022 (1.00)

a.

REFERENCE:

Off-Gas system, LOT-0510, p. 10
Learning Objective 1.b

271000K101 [3.1/3.1]

271000K101 ..(KA's)

ANSWER: 023 (1.00)

b.

REFERENCE:

Off-Gas system, LOT-510, pg 10, 17, 19
Learning Objective 6.c, 4.o

271000A211 [2.8/2.9]

271000A211 ..(KA's)

ANSWER: 024 (1.00)

d.

REFERENCE:

Feedwater Control system, LOT-0550, p. 37 and 40
Learning Objective 4.j
Facility Question Bank No. 2581
259002A401 [3.8/3.6]

259002A401 ..(KA's)

ANSWER: 025 (1.00)

c.

REFERENCE:

T-PLOT-0590-6
Electro-hydraulic Control Logic, p. 10

241000A101 [3.9/3.8]

241000A101 ..(KA's)

ANSWER: 026 (1.00)

b.

REFERENCE:

Significantly modified PB Bank #3885
LOT 0180, rev 008, pg 11 - 17, Learning Objective 5.b

223002A302 [3.5/3.5]

223002A302 ..(KA's)

ANSWER: 027 (1.00)

b.

REFERENCE:

Feedwater control system, LOT-0550, p. 22
Learning Objective 5.b

259002K603 [3.1/3.1]

259002K603 ..(KA's)

ANSWER: 028 (1.00)

b.

REFERENCE:

Rod Block Monitor, LOT-0280, p. 12
T-LOT-0280-6
Learning Objective 5.c

215002K403 [2.9/3.0]

215002K403 ..(KA's)

ANSWER: 029 (1.00)

c.

REFERENCE:

Rod Block Monitor, LOT-0280, p. 7
Learning Objective 2.b

215002K102 [3.1/3.2]

215002K102 ..(KA's)

ANSWER: 030 (1.00)

b.

REFERENCE:

Reactor Protective System, LOT-0300, p. 23
Learning Objective 2.1

212000K114 [3.6/3.7]

212000K114 ..(KA's)

ANSWER: 031 (1.00)

a.

REFERENCE:

Reactor Protective System, LOT-0300, p. 28
Learning Objective 5.j

211000K410 [3.3/3.6]

212000K410 ..(KA's)

ANSWER: 032 (1.00)

c.

REFERENCE:

OT-106, Rev 13, pg 1
LOT-1540, Learning Objective 2
modified PB exam bank # 2544
295002G010 [3.8/3.7]

295002G010 ..(KA's)

ANSWER: 033 (1.00)

d.

REFERENCE:

OT-112, Rev 16, pg 1
LOT-1540, Learning Objective 1, 2, and 3
modified pb exam bank #2547

NOTE: Include OT-112 Figures A and B in exam.

295001G010 [3.8/3.7]

295001G010 ..(KA's)

ANSWER: 034 (1.00)

a.

REFERENCE:

OT-114, Rev 6, and OT-116 Bases, pg 5
LOT-1540, Learning Objective 4

239002A205 [3.2/3.5]

239002A205 ..(KA's)

ANSWER: 035 (1.00)

b.

REFERENCE:

OT-111, Rev 2, OT-111 Bases, pg 2
LOT-1540, Learning Objective 4

241000A121 [3.4/3.4]

241000A121 ..(KA's)

ANSWER: 036 (1.00)

a.

REFERENCE:

OT-114, Rev 2, pg 1
OT-114 Bases

259002K603 [3.1/3.1]

259002K603 ..(KA's)

ANSWER: 037 (1.00)

b.

REFERENCE:

OT-102-3, Rev 0, pg 1
LOT-1540, Learning Objective 2

295007A105 [3.7/3.8]

295007A105 ..(KA's)

ANSWER: 038 (1.00)

c.

REFERENCE:

OT-101 Bases, Rev 8, pg 5
LOT-1540, Learning Objective 4 and 5 (Although SRO objectives, this function is performed by the RO, and as such the caution needs to be well understood by the RO.)
PB exam bank #4115

295010G006 [3.8/3.9]

295010G006 ..(KA's)

~~ANSWER: 039 (1.00)~~

~~b.~~ Deleted

REFERENCE:

OT-101 Bases, Rev 8, pg 9
T.S. 3.5.A.6

202001G006 [3.0/4.1]

202001G006 ..(KA's)

ANSWER: 040 (1.00)

b. 4 a

REFERENCE:

Modified PB Exam Bank # 4306

295007K201 [3.5/3.7]

295007K201 ..(KA's)

ANSWER: 041 (1.00)

c.

REFERENCE:

OT-110, Rev 2, Figure 1, pg 5
LOT-1540, Learning Objective 3

NOTE: Include figure 1 in exam.

295008A201 [3.9/3.9]

295008A201 ..(KA's)

ANSWER: 042 (1.00)

d.

REFERENCE:

OT-110 Bases, Rev 2, pg 7
LOT-1540, Learning Objective 4 and 5

295008K204 [3.1/3.3]

295008K204 ..(KA's)

ANSWER: 043 (1.00)

b. + a

REFERENCE:

OT-112, Rev 16, pg 1
OT-112 Bases, Rev 15

295001A106 [3.3/3.4]

295001A106 ..(KA's)

ANSWER: 044 (1.00)

d.

REFERENCE:

Modified PB Bank #4587
LOT-0690, Learning Objective 3.a (related objective)

295004K203 [3.3/3.3]

295004K203 ..(KA's)

ANSWER: 045 (1.00)

a.

REFERENCE:

ON-112-2, Rev 0, pg 1
LOT-1550, Learning Objective 1
PB exam bank #1698, modified for different answers and distractors.
262002K105 [2.7/2.9]

262002K105 ..(KA's)

ANSWER: 046 (1.00)

c.

REFERENCE:

ON-107, Rev 4
T.S. 3.3.A.2.f

295022G003 [3.1/3.8]

295022G003 ..(KA's)

ANSWER: 047 (1.00)

a.

REFERENCE:

On-106 Bases, Rev 3, pg 13
LOT-1550, Learning Objective 3

201003A201 [3.4/3.6]

201003A201 ..(KA's)

ANSWER: 048 (1.00)

c.

REFERENCE:

ON-119, Rev 11, Attachment 1
LOT-1550, Learning Objective 1

295019K203 [3.2/3.3]

295019K203 ..(KA's)

ANSWER: 049 (1.00)

a.

REFERENCE:

ON-110, Rev 1, pg 1
T.S. 3.7.A.3

223001G006 [3.0/4.0]

223001G006 ..(KA's)

ANSWER: 050 (1.00)

c.

REFERENCE:

T.S. 3.7.A Bases, pg 189 and 190.

295013G004 [3.0/4.1]

295013G004 ..(KA's)

ANSWER: 051 (1.00)

d.

REFERENCE:

ON-100, Rev 2, pg 1
LOT-1550, Learning Objective 1

290002K303 [3.3/3.4]

290002K303 ..(KA's)

ANSWER: 052 (1.00)

b.

REFERENCE:

T-103 Bases, Rev 6, pg 4

295035K202 [3.6/3.8]

295035K202 ..(KA's)

ANSWER: 053 (1.00)

d.

REFERENCE:

T.S. 3/4.7.C.1

295035G003 [2.8/3.9]

295035G003 ..(KA's)

ANSWER: 054 (1.00)

c.

REFERENCE:

ON-105 Bases, Rev 3, pg 1

201003A402 [3.5/3.5]

201003A402 ..(KA's)

ANSWER: 055 (1.00)

a.

REFERENCE:

ON-108 Bases, Rev 4, pg 1
LOT-1550, Learning Objective 3
PB Exam Bank #3777 modified

295015K103 [3.8/3.9]

295015K103 ..(KA's)

ANSWER: 056 (1.00)

c.

REFERENCE:

OT-106 Bases, Rev 11, pg 3
LOT-0500, Fig. 5, Learning Objective 3b and 5d
LOT-1550, Learning Objective 3

295017G007 [3.2/3.6]

295017G007 ..(KA's)

ANSWER: 057 (1.00)

d.

REFERENCE:

ON-113 Bases, Rev 8, pg 3
LOT 1550, Learning Objective 3

295018G007 [3.2/3.4]

295018G007 ..(KA's)

ANSWER: 058 (1.00)

b.

REFERENCE:

ON-102, Rev 4, pg 1
LOT-1550, Learning Objective 1 and 2
LOT-0720, pg 28, Learning Objective 3a and 3c

295038A203 [3.5/4.3]

295038A203 ..(KA's)

ANSWER: 059 (1.00)

b.

REFERENCE:

ERP-101, Rev 15, pg 4 and Table 11

294001A116 [2.9/4.7]

294001A116 ..(KA's)

ANSWER: 060 (1.00)

c.

REFERENCE:

EP-101, Rev 15, pg 1 (final paragraph) and Table 15

294001A116 [2.9/4.7]

294001A116 ..(KA's)

ANSWER: 061 (1.00)

b.

REFERENCE:

SE-113 Bases, Rev 5, pg 45
LOT-1555, Learning Objective 13a

295003K303 [3.5/3.6]

295003K303 ..(KA's)

ANSWER: 062 (1.00)

a.

REFERENCE:

SE-10 Attachment 9 Figure 1, Rev 0, pg 2
LOT-1555, Learning Objective 2a and 2b

295016A202 [4.2/4.3]

295016A202 ..(KA's)

ANSWER: 063 (1.00)

c.

REFERENCE:

T-102, Rev 9, Step T/L-8

295026G012 [3.8/4.5]

295026G012 ..(KA's)

ANSWER: 064 (1.00)

c.

REFERENCE:

LOT-1560, Learning Objective 6
modified PB exam bank #1694

295026K101 [3.0/3.4]

295026K101 ..(KA's)

ANSWER: 065 (1.00)

c.

REFERENCE:

LOT-1560, Learning Objective 3

295032G007 [3.3/3.5]

295032G007 ..(KA's)

ANSWER: 066 (1.00)

c.

REFERENCE:

T-102, Curve T/L-3, Step T/L-19
LOT-1560, Learning Objective 6 and 8

295029K301 [3.5/3.9]

295029K301 ..(KA's)

ANSWER: 067 (1.00)

d.

REFERENCE:

T-215, Rev 0, pg 1
T-216, pg 4 Note; T-213, pg 2 Note 3

295015K201 [3.8/3.9]

295015K201 ..(KA's)

ANSWER: 068 (1.00)

a.

REFERENCE:

T-102, Rev 9, Table DW/T-1
T-103, Rev 6, Table SC/T-4
LOT 1560, Learning Objective 8

295028A203 [3.7/3.9]

295028A203 ..(KA's)

ANSWER: 069 (1.00)

c.

REFERENCE:

LOT-0050-5, Rev 10, pg 1 of 2., Learning Objective 6f

295031K202 [3.8/3.9]

295031K202 ..(KA's)

ANSWER: 070 (1.00)

a.

REFERENCE:

T-102 sh 3 of 3, Rev 9, ECCS suction requirements tables
LOT-1560, Learning Objective 8

295030A101 [3.6/3.8]

295030A101 ..(KA's)

ANSWER: 071 (1.00)

a.

REFERENCE:

T-101 Bases, Rev 14, step RC/Q-15, pg 7
LOT-1560, Learning Objective 3

295037K301 [4.1/4.2]

295037K301 ..(KA's)

ANSWER: 072 (1.00)

d. + c

REFERENCE:

T-102 Bases, Rev 10, pg 7 and T-102 T/L-8 (Bases for trip procedure curves not located.)
NOTE: Verify with facility that downcomers are uncovered at 10 feet but SRV tailpipes are still covered.

295030K207 [3.5/3.8]

295030K207 ..(KA's)

ANSWER: 073 (1.00)

c.

REFERENCE:

T-117 Bases, Rev 9, pg 9
LOT 1560, Learning Objective 3
Modified PB Bank #2889

295037K106 [4.0/4.2]

295037K106 ..(KA's)

ANSWER: 074 (1.00)

b. & C

REFERENCE:

T-102 Bases, pg 12 and 13, PC/P-5 and -9
LOT-1560, Learning Objective 6

295010G007 [3.6/3.8]

295010G007 ..(KA's)

ANSWER: 075 (1.00)

c.

REFERENCE:

T-101 Bases, Rev 14, Step RC/P-3, pg 20

295025K301 [4.2/4.3]

295025K301 ..(KA's)

ANSWER: 076 (1.00)

a.

REFERENCE:

T-102 Bases, Rev 10, Step PC/P-5,-6,-7,-8, pg 12 and 13

295024K301 [3.6/4.0]

295024K301 ..(KA's)

ANSWER: 077 (1.00)

d.

REFERENCE:

T-100 Bases, Rev 7, Step S-10, pg 4
LOT-1560, Learning Objective 1

295005A208 [3.2/3.3]

295005A208 ..(KA's)

ANSWER: 078 (1.00)

c.

REFERENCE:

GP-12, Rev 17, pg 5 and Figure 3
OM-C-7.1, Rev 0, Step 3.0, pg 2 and 3
295021A203 [3.5/3.5]

295021A203 ..(KA's)

ANSWER: 079 (1.00)

a.

REFERENCE:

T-100, Rev 7, Step S-8, pg 4
LOT-1560, Learning Objective 12

295006A104 [3.1/3.2]

295006A104 ..(KA's)

ANSWER: 080 (1.00)

d.

REFERENCE: Deleted

T-103 Bases, Rev 6, Step SC/L-1, pg 5
LOT-1560, Learning Objective 8
295036A101 [3.0/3.2]

295036A101 ..(KA's)

ANSWER: 081 (1.00)

b.

REFERENCE:

T-102 Bases, Steps DW/T-13 and PC/P-9, pg 13, 14, 20, and 21
LOT-1560, Learning Objective 3

295024G007 [3.6/3.9]

295024G007 ..(KA's)

ANSWER: 082 (1.00)

d.

REFERENCE:

T-111, Rev 8, Step LR-18 and LR-20
LOT-1560, Learning Objective 8

295009G012 [3.8/4.4]

295009G012 ..(KA's)

ANSWER: 083 (1.00)

a.

REFERENCE:

T.S. 3.10.A.

295023G003 [2.9/3.8]

295023G003 ..(KA's)

ANSWER: 084 (1.00)

a.

REFERENCE:

T-116, Rev 8, Steps RF-22 and RF-23

295031G012 [3.9/4.5]

295031G012 ..(KA's)

ANSWER: 085 (1.00)

d.

REFERENCE:

LOT-0160, Rev 6, pg 6 and 7, Learning Objective 7

294001K115 [3.4/3.8]

294001K115 ..(KA's)

ANSWER: 086 (1.00)

d.

REFERENCE:

A-C-10, Rev 10, Step 7.5.2.1.c, pg 14
LOT-0005, Learning Objective 2
294001A103 [2.7/3.7]

294001A103 ..(KA's)

ANSWER: 087 (1.00)

a.

REFERENCE:

A-30, Rev 13, step 7.1.6, pg 8
LOT-1570, Learning Objective 1.h

294001A102 [4.2/4.2]

294001A102 ..(KA's)

ANSWER: 088 (1.00)

a.

REFERENCE:

A-14, Rev 21, step 4.12, pg 6
LOT-1570, Learning Objective 1.f

294001A107 [3.0/3.7]

294001A107 ..(KA's)

ANSWER: 089 (1.00)

a.

REFERENCE:

C & T Manual pg 21 of 142
LOT-1570, Learning Objective 1.k

294001K103 [3.9/4.5]

294001K103 ..(KA's)

ANSWER: 090 (1.00)

a.

REFERENCE:

A-42.1, Rev 5, Step 2.0, pg 1
LOT-1570, Learning Objective 1.m

294001A112 [3.5/4.2]

294001A112 ..(KA's)

ANSWER: 091 (1.00)

b.

REFERENCE:

A-C-10, Rev 0, step 7.4.7, pg 13
LOT-0005, Learning Objective 2
294001A103 [2.7/3.7]

294001A103 ..(KA's)

ANSWER: 092 (1.00)

b.

REFERENCE:

A-C-26, Rev 0, Step 4.5, pg 3
LOT-1570, Learning Objective 1.g

294001A110 [3.6/4.2]

294001A110 ..(KA's)

ANSWER: 093 (1.00)

d.

REFERENCE:

A-C-26, Rev 0, step 4.9, pg 4
LOT-1570, Learning Objective 1.g

294001A108 [3.1/3.6]

294001A108 ..(KA's)

ANSWER: 094 (1.00)

a.

REFERENCE:

A-C-43, Rev 0, Step 7.4.3 and 7.4.4, pg 7
LOT-1570, Learning Objective 1.n

294001A113 [4.5/4.3]

294001A113 ..(KA's)

ANSWER: 095 (1.00)

a.

REFERENCE:

HP-C-215, Rev 1, pg 9

294001K103 [3.3/3.8]

294001K103 ..(KA's)

ANSWER: 096 (1.00)

a.

REFERENCE:

HP-C-215, Rev 1, pg 9

294001K103 [3.3/3.8]

294001K103 ..(KA's)

ANSWER: 097 (1.00)

b.

REFERENCE:

LOT-0005
USNRC Regulatory Guide 1.114, Section B.2

294001A109 [3.3/4.2]

294001A109 ..(KA's)

ANSWER. 098 (1.00)

c.

REFERENCE:

OM-P-16.1:5, Rev 1, Step 2.2, pg 1
LOT-lesson plan, none located.

294001A111 [3.3/4.3]

294001A111 ..(KA's)

ANSWER: 099 (1.00)

a.

REFERENCE:

OM-P-12.1, Rev 0, Step 3.1, pg 2
LOT-0008, Learning Objective 2

294001A110 [3.6/4.2]

294001A110 ..(KA's)

ANSWER: 100 (1.00)

a.

REFERENCE:

OM-P-7.6, Rev 0, step 3.2, pg 4

294001K107 [3.3/3.6]

294001K107 ..(KA's)

(********** END OF EXAMINATION **********)

TEST CROSS REFERENCE

SRO Exam BWR Reactor
Organized by Question Number

QUESTION	VALUE	REFERENCE	
001	1.00	27094	1
002	1.00	9000001	2
003	1.00	9000003	4
004	1.00	9000004	5
005	1.00	9000005	6
006	1.00	9000006	7
007	1.00	9000007	8
008	1.00	9000008	9
009	1.00	9000009	10
010	1.00	9000010	11
011	1.00	9000011	12
012	1.00	9000012	13
013	1.00	9000013	14
014	1.00	9000016	17
015	1.00	9000017	18
016	1.00	9000018	19
017	1.00	9000019	20
018	1.00	9000020	21
019	1.00	9000021	22
020	1.00	9000022	23
021	1.00	9000023	24
022	1.00	9000024	25
023	1.00	9000025	26
024	1.00	9000026	27
025	1.00	9000027	28
026	1.00	9000028	29
027	1.00	9000029	30
028	1.00	9000030	31
029	1.00	9000031	32
030	1.00	9000032	33
031	1.00	9000033	34
032	1.00	9000044	45
033	1.00	9000045	46
034	1.00	9000046	
035	1.00	9000047	
036	1.00	9000049	48
037	1.00	9000050	49
038	1.00	9000051	50
039	1.00	9000052	
040	1.00	9000053	51
041	1.00	9000054	52
042	1.00	9000055	53
043	1.00	9000056	54
044	1.00	9000057	55
045	1.00	9000058	56
046	1.00	9000060	
047	1.00	9000062	
048	1.00	9000063	59
049	1.00	9000064	

SRO Exam BWR Reactor
Organized by Question Number

QUESTION	VALUE	REFERENCE	
050	1.00	9000065	
051	1.00	9000067	61
052	1.00	9000068	62
053	1.00	9000069	
054	1.00	9000070	63
055	1.00	9000071	
056	1.00	9000072	64
057	1.00	9000073	65
058	1.00	9000074	66
059	1.00	9000075	
060	1.00	9000076	
061	1.00	9000077	67
062	1.00	9000078	69
063	1.00	9000080	
064	1.00	9000081	70
065	1.00	9000082	71
066	1.00	9000083	
067	1.00	9000084	72
068	1.00	9000085	73
069	1.00	9000086	74
070	1.00	9000087	
071	1.00	9000088	75
072	1.00	9000089	76
073	1.00	9000090	77
074	1.00	9000091	78
075	1.00	9000092	79
076	1.00	9000093	80
077	1.00	9000094	81
078	1.00	9000095	82
079	1.00	9000096	
080	1.00	9000097	83
081	1.00	9000098	
082	1.00	9000099	
083	1.00	9000100	
084	1.00	9000101	
085	1.00	9000102	84
086	1.00	9000103	85
087	1.00	9000106	
088	1.00	9000107	
089	1.00	9000108	
090	1.00	9000109	
091	1.00	9000110	
092	1.00	9000111	
093	1.00	9000112	
094	1.00	9000114	89
095	1.00	9000116	91
096	1.00	9000117	92
097	1.00	9000118	
098	1.00	9000120	

SRO Exam BWR Reactor
Organized by Question Numb r

QUESTION	VALUE	REFERENCE	
099	1.00	9000121	
100	1.00	9000123	95

	100.00		

	100.00		

SRO Exam BWR Reactor
Organized by KA Group

PLANT WIDE GENERICS

QUESTION	VALUE	KA
087	1.00	294001A102
086	1.00	294001A103
091	1.00	294001A103
088	1.00	294001A107
093	1.00	294001A108
097	1.00	294001A109
092	1.00	294001A110
099	1.00	294001A110
098	1.00	294001A111
090	1.00	294001A112
094	1.00	294001A113
060	1.00	294001A116
059	1.00	294001A116
089	1.00	294001K103
095	1.00	294001K103
096	1.00	294001K103
100	1.00	294001K107
085	1.00	294001K115
PWG Total	18.00	

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
013	1.00	202002A401
001	1.00	203000A405
009	1.00	203000G009
010	1.00	206000A217
011	1.00	206000G010
012	1.00	206000K201
030	1.00	212000K114
031	1.00	212000K410
002	1.00	217000A401
006	1.00	217000K407
007	1.00	218000K501
008	1.00	218000K606
049	1.00	223001G006
026	1.00	223002A302
034	1.00	239002A205
025	1.00	241000A101
035	1.00	241000A121
024	1.00	259002A401
027	1.00	259002K603

S R O E x a m B W R R e a c t o r
O r g a n i z e d b y K A G r o u p

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
036	1.00	259002K603
018	1.00	262001A302
019	1.00	264000A401
017	1.00	264000A404

PS-I Total	23.00	

Group II

QUESTION	VALUE	KA
004	1.00	201001A109
005	1.00	201001K101
039	1.00	202001G006
003	1.00	202001K502
020	1.00	204000G004
021	1.00	204000K404
029	1.00	215002K102
028	1.00	215002K403
016	1.00	259001A204
015	1.00	259001G010
045	1.00	262002K105
023	1.00	271000A211
022	1.00	271000K101

PS-II Total	13.00	

Group III

QUESTION	VALUE	KA
047	1.00	201003A201
054	1.00	201003A402
014	1.00	256000K606
051	1.00	290002K303

PS-III Total	4.00	

PS Total	40.00	

EMERGENCY PLANT EVOLUTIONS

Group I

SRO Exam BWR Reactor
Organized by KA Group

EMERGENCY PLANT EVOLUTIONS

Group I

QUESTION	VALUE	KA
061	1.00	295003K303
079	1.00	295006A104
037	1.00	295007A105
040	1.00	295007K201
082	1.00	295009G012
038	1.00	295010G006
074	1.00	295010G007
050	1.00	295013G004
055	1.00	295015K103
067	1.00	295015K201
062	1.00	295016A202
056	1.00	295017G007
083	1.00	295023G003
081	1.00	295024G007
076	1.00	295024K301
075	1.00	295025K301
063	1.00	295026G012
064	1.00	295026K101
070	1.00	295030A101
072	1.00	295030K207
084	1.00	295031G012
069	1.00	295031K202
073	1.00	295037K106
071	1.00	295037K301
058	1.00	295038A203

EPE-I Total	25.00	

Group II

QUESTION	VALUE	KA
043	1.00	295001A106
033	1.00	295001G010
032	1.00	295002G010
044	1.00	295004K203
077	1.00	295005A208
041	1.00	295008A201
042	1.00	295008K204
057	1.00	295018G007
048	1.00	295019K203
078	1.00	295021A203
046	1.00	295022G003
068	1.00	295028A203
066	1.00	295029K301

SRO Exam BWR Reactor
Organized by KA Group

EMERGENCY PLANT EVOLUTIONS

Group II

QUESTION	VALUE	KA
065	1.00	295032G007
053	1.00	295035G003
052	1.00	295035K202
080	1.00	295036A101

EPE-II Total	17.00	

EPE Total	42.00	

Test Total	100.00	

ANSWER KEY

MULTIPLE CHOICE

001	c	023	b
002	d	024	d
003	b	025	c
004	d	026	b
005	a	027	b
006	a	028	b
007	c	029	c
008	c	030	b
009	c	031	a
010	b	032	c
011	d	033	d
012	a + c	034	a
013	d	035	b
014	c	036	a
015	b	037	b
016	a	038	c
017	b	~~039 b~~ Deleted	
018	b	040	b + a
019	c	041	c
020	b	042	d
021	a	043	b + a
022	a	044	d
		045	a

ANSWER KEY

046 c

047 a

048 c

049 a

050 c

051 d

052 b

053 d

054 c

055 a

056 c

057 d

058 b

059 b

060 c

061 b

062 a

063 c

064 c

065 c

066 c

067 d

068 a

069 c

070 a

071 a

072 d & c

073 c

074 b & c

075 c

076 a

077 d

078 c

079 a

~~080 d~~ deleted

081 b

082 d

083 a

084 a

085 d

086 d

087 a

088 a

089 a

090 a

091 b

ANSWER KEY

092 b

093 d

094 a

095 a

096 a

097 b

098 c

099 a

100 a

(********** END OF EXAMINATION **********)

ATTACHMENT 3

FACILITY COMMENTS ON WRITTEN EXAMINATIONS

Gerald R. Rainey
Vice President
Peach Bottom Atomic Power Station

PECO ENERGY

PECO Energy Company
RD 1, Box 208
Delta, PA 17314-9739
717 456 7014

10CFR55 Subpart E

August 31, 1995

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

Mr. Thomas T. Martin
Regional Administrator
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

SUBJECT: Peach Bottom Atomic Power Station RO/SRO Written Exam Comments for NRC Combined Inspection 50-277/95-20 and 50-278/95-20

Dear Mr. Martin:

Attached is our response to selected questions and answers associated with the NRC RO/SRO written examination administered on August 17, 1995 at the Peach Bottom Atomic Power Station. These comments are being submitted using the guidance of NUREG 1021 to clarify areas where alternate correct answers should be considered. References have been indicated and attached to justify the comments made.

If there are any questions concerning these comments, please contact Dennis W. McClellan, PBAPS Manager - Operations Training, at 717-456-3204.

Sincerely,

GRRainey

Gerald R. Rainey
Vice President, PBAPS

GRR/~~JJS~~/DWM/PEN:clg

Attachment

cc: G. W. Meyer, Chief, BWR and PWR Section, USNRC, Region 1
W. L. Schmidt, Senior Resident Inspection, USNRC
D. Florek, USNRC, Region 1
Document Control Desk, USNRC, Washington, DC

CCN 95-14075

bcc:	Correspondence Control Program	61B-7, Chesterbrook
	G. D. Edwards	A4-1S, Peach Bottom
	G. H. Gellrich	A4-1S, Peach Bottom
	Nuclear Records	Peach Bottom

PEACH BOTTOM ATOMIC POWER STATION

NRC RO/SRO EXAM COMMENT ATTACHMENT

This attachment contains comments on the following NRC Exam Questions:

SRO 12 / RO 13
RO 39
SRO 39
SRO 40
SRO 43 / RO 54
SRO 72 / RO 76
SRO 74 / RO 78
SRO 80 / RO 83
SRO 91
SRO 99

NRC QUESTION: SRO 12/RO 13

The HPCI system is in its normal standby readiness lineup.

Which one of the following power supplies, if deenergized, would prevent HPCI from being either manually or automatically initiated?

A. 250 VDC bus (20D11)

B. 480 VAC MCC (20B38)

C. 125 VDC bus (20D22)

D. 480 VAC MCC (30B37)

NRC ANSWER:

A

NRC REFERENCE:

P&ID, E-26

206000K201 [3.2/3.3]

UTILITY COMMENT:

Answer A is correct using E-26, Sheet 1 and SE-13, Attachment 3, Part 3.

Answer C is also correct using E-26, Sheet 1 and SE-13, Attachment 3, Part 1.

UTILITY F ƆMMENDATION:

Accept both A and C.

UTILITY REFERENCE:

SE-13, Attachment 3 (Attached)
E-26, Sheet 1

ATTACHMENT 3

Part 1

> **NOTE**
>
> Annunciator 220 H-3 "2B DC POWER PANEL LO VOLTAGE" alarms from undervoltage at 20D22 or 2BD306. Determine if one or both panels have low voltage and then refer to the appropriate tables below.

20D22 125 VDC Power Distribution Panel Low Voltage
(References below are for Unit 2 unless otherwise specified)

	Plant Effect		Operator Action
1.	Multiple ECCS inoperable. (i.e.: HPCI, Core Spray, E-2 and E-4 Diesels, E-22 and E-23 buses).	1a. b.	Commence Unit 2 Shutdown. Commence Unit 3 Shutdown.
2.	E-22 bus differential protection relays are inoperable.	2.	Remove bus from service when possible.
3.	E-23 bus differential protection relays are inoperable.	3.	Remove bus from service when possible.
4.	#1 Aux Bus fast transfer is inoperable.	4.	Manually transfer #1 Aux Bus to offsite source.
5.	#2 Aux Bus Control and Protection is inoperable.	5a. b. c.	Operate breakers locally. Crosstie load centers fed from #2 Aux Bus in accordance with SO 53.6.B. Remove bus from service when possible.
★6.	HPCI is inoperable.	6a. b. c.	Place HPCI Aux Oil Pump in PULL TO LOCK. Use RCIC. Use ADS and LPCI A Loop.
7.	Core Spray Loop "B": o Initiation Logic is inoperable.	7a. b.	Use Core Spray A Loop. Use Loop B manually.
8.	Emergency Diesels: o E-2 and E-4 LOCA initiation is inoperable for Unit 2. o E-2 protective relays are inoperable.	8a. b. c.	Use E-1 and E-3 Diesels. Initiate E-4 from the MCR. Place E-2 in PULL TO LOCK. Use only in an emergency.

(Continued)

SRO 12/RO 13

ATTACHMENT 3

Part 3

2DB-R-B 250 VDC Motor Control Center Low Voltage
(References below are for Unit 2 unless otherwise specified)

Plant Effect	Operator Action
1. HPCI DC motor operated valves are inoperable.	1a. Use RCIC if necessary. b. Use ADS and LPCI if necessary.
2. The following PCIS valves are inoperable: o MO-2-10-17 RHR Shutdown Cooling Suction Valve o MO-2-10-33 RHR Reactor Head Spray Valve o MO-2-12-18 RWCU Suction Isolation Valve o MO-2-2-77 Main Steam Line Drain Valve o MO-2-14-71 Torus Water Filter Pump Suction Valve	2a. IF PCIS is required to be operable, THEN manually isolate the affected systems. b. Refer to GP-12 for Core Cooling.

NRC QUESTION: RO 39

Which one of the following statements describes the closing operation of the Main Steam Isolation Valves (MSIVs)?

A. Pneumatic pressure is NOT used to close the MSIVs.

B. Instrument air supplies pneumatic pressure to the inboard MSIVs during the valve closure operation.

C. Both the AC and DC solenoids must de-energize to close the MSIVs.

D. During test operation the MSIVs close under spring pressure alone.

NRC ANSWER:

C

NRC REFERENCE:

LOT-0120, page 20

239001K201 [3.2/3.3]

UTILITY COMMENT:

Answer C is correct.

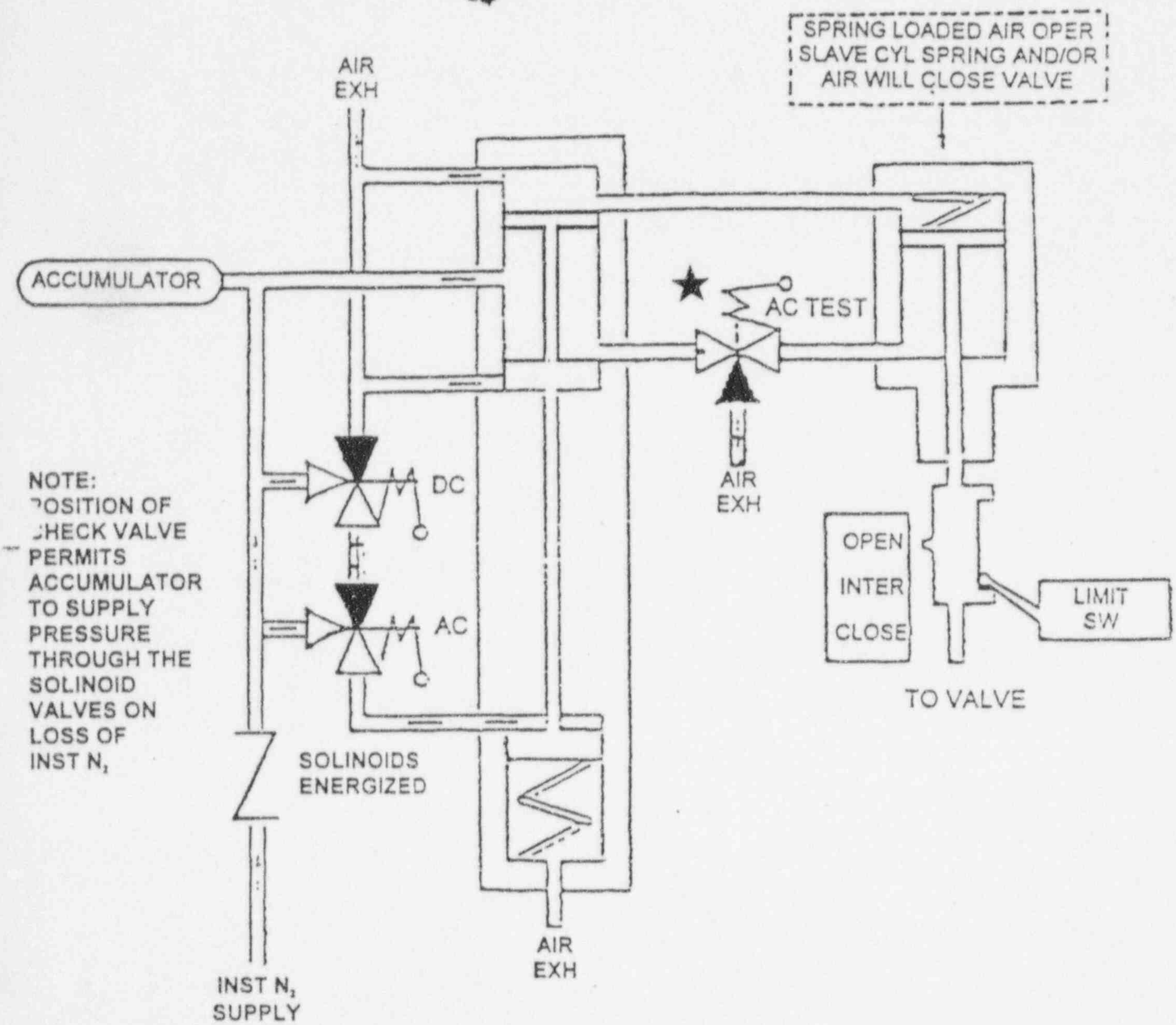
Answer D is also correct since test operation requires pushing the test pushbutton which energizes the AC test solenoid. This solenoid will vent off the operating air permitting the spring pressure alone to close the valve. (Reference attached drawing.)

UTILITY RECOMMENDATION:

Accept both C and D.

UTILITY REFERENCE:

T-LOT-0120-6 (Attached)



NOTE:
POSITION OF CHECK VALVE PERMITS ACCUMULATOR TO SUPPLY PRESSURE THROUGH THE SOLINOID VALVES ON LOSS OF INST N,

OPEN INBOARD MSIV OPERATING SYSTEM
T-LOT-0120-6

NRC QUESTION: SRO 39

Select the choice below that completes the following statements.

During execution of OT-101 "High Drywell Pressure", the shift suspects that reactor recirc loop A suction and discharge valve stems are leaking. Per OT-101, the recirc pump suction and discharge valves are NOT to be backseated because ______________.

A. operation of the LPCI system relies on the automatic closure of the recirc discharge valve, thus one LPCI subsystem must be declared inoperable and a 7 day LCO must be initiated

B. operation of the LPCI system relies on the automatic closure of the recirc discharge valve, thus requiring the reactor to be in COLD SHUTDOWN within 48 hours

C. both valves are required to isolate a recirc pump seal failure, thus requiring the reactor to be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN in the following 24 hours

D. both valves are required to isolate a recirc pump seal failure, thus the valves must be declared inoperable and the reactor must be in COLD SHUTDOWN within 24 hours

NRC ANSWER:

B

NRC REFERENCE:

OT-101 Bases, Rev. 8, page 9
Tech Specs 3.5.A.6

202001G006 [3.0/4.1]

UTILITY COMMENT:

Per OT-101, the Recirc valves are not to be backseated because breaking either one in the open direction removes the ability to isolate the pump, and, for the discharge valve, the LPCI system relies on automatic closure of the valve at 225 psig reactor pressure. When a valve is backseated it is considered inoperable.

Answers A and B both address applicable LCOs in the event of an inoperable Recirc discharge valve. Although B requires the most immediate action, A is still applicable to the situation. The Recirc valves are not backseated to avoid entering either shutdown LCO.

UTILITY RECOMMENDATION:

Accept both A and B.

UTILITY REFERENCE:

OT-101 Bases, Rev. 8, page 9 (Attached)
Tech Specs 3.5.A.5, 6, 7 (Attached)

The HPCI and RCIC steam isolation valves are treated differently because leaving the valves closed for an extended period could cause the downstream steam lines to cool down. If either valve is found to be the source of the drywell pressure increase, then the valve is to be deenergized closed to prevent an automatic initiation from admitting a large amount of steam to the cool piping.

The list of valves was selected to exclude those valves that would have an inordinately large transient effect on the reactor or the operability of core cooling systems.

10. IF the drywell pressure rise persists AND both Drywell Instrument N2 headers are in service, THEN isolate one Instrument N2 header at a time to check for N2 leaks using the following valves:

- "A" header - AO-2(3)969A, "A DRYWELL" at Panel 2(3)0C003-03
- "B" header - AO-2(3)969B, "B DRYWELL" at Panel 2(3)0C003-03

Any gas leaking into a constant volume will cause the pressure in that volume to increase. Isolation of that gas leak will stop the pressure rise.

CAUTION

Backseating a motor operated valve can cause valve damage and can make the valve inoperable.

11. IF the drywell pressure rise persists, THEN consider backseating the following valves in accordance with AO 56.2-2(3), "Backseating AC Motor Operated Valves from the MCC".

VALVE	NAME	U/2 FEED	U/3 FEED
MO-2(3)-13-015	RCIC INBD ISOL	E224-R-B (3721)	E234-R-B (3721)
MO-2(3)-23-015	HPCI INDB ISOL	E124-R-C (3614)	E134-W-A (3614)
MO-2(3)-12-015	RWCU INLET INBD ISOL	E124-R-C (3633)	E134-W-A (3633)
MO-2(3)-02-074	MSL DRAIN INBD ISOL	E124-R-C (3653)	E134-W-A (3653)
MO-2(3)-06-29A	FEED WTR STOP A	E224-R-B (3743)	E234-R-B (3743)
MO-2(3)-06-29B	FEED WTR STOP B	E124-R-C (3644)	E134-W-A (3644)

Backseating MOVs may reduce or eliminate steam packing leaks in the drywell. As stated in the CAUTION, backseating MOVs carries a certain amount of risk for valve and motor damage. For this reason, backseating is performed only after other actions are attempted and only for select valves. The recirc pump suction and discharge valves are NOT to be backseated because breaking either one in the open direction removes the ability to isolate the pump, and, for the discharge valve, the LPCI system relies on automatic closure of the valve at 225 psig reactor pressure.

12. Consider any valves backseated in the previous step inoperable AND log them in ST-O-007-400-2(3), "Inoperable Isolation Valve Position Daily Log" as appropriate for Tech Spec 3.7.D.2 and 4.7.D.2.

    As explained in the bases for the previous step, backseating can render a valve inoperable - and each backseated valve shall be considered as such. ST-O-007-400-2(3) is used as required by Tech Spec 3.7.D.2 and 4.7.D.2.

13. IF the Recirc Pump(s) were tripped and isolated in step 7c, THEN shutdown the Recirc Pump(s) in accordance with SO 2A.2.A-2(3), "Recirc Pump Shutdown".

    The recirc pumps may have been quickly isolated in step 7c to stop the drywell pressure excursion. It is prudent to perform the remaining steps of SO 2A.2.A-2(3) to fully remove the pumps from service prior to exiting OT-101.

14. IF the Drywell Chiller(s) were tripped in step 7c, THEN shutdown the Drywell Chilled Water System in accordance with SO 44A.2.A-2(3), "Chilled Water System Shutdown".

    The Drywell Chilled Water System may have been quickly isolated in step 7c to stop the drywell pressure excursion. It is prudent to perform the remaining steps of SO 44A.2.A-2(3) to fully remove the system from service prior to exiting OT-101.

COMMITMENTS:

CM-1, INPO SER 12-83 (T00819)

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.A <u>Core Spray and LPCI Subsystem</u> (cont'd)	4.5.A <u>Core Spray and LPCI Subsystem</u> (cont'd)

3.5.A Core Spray and LPCI Subsystem (cont'd)

3. Two independent Low Pressure Coolant Injection (LPCI) subsystems will be operable with each subsystem comprised of:

   a. (Two 33-1/3%) capacity pumps,

   b. An operable flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel, and

   c. During power operation the LPCI system cross-tie valve closed and the associated valve motor operator circuit breaker locked in the off position.

Both LPCI subsystems shall be operable whenever irradiated fuel is in the reactor vessel, and prior to reactor startup from the Cold Shutdown Condition, except as specified in 3.5.A.4, 3.5.A.5, and 3.5.F below.

4. From and after the date that one of the four LPCI pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days provided that during such seven days the remaining active components of the LPCI subsystems, and all active components of both core spray subsystems are operable.

★ 5. From and after the date that one LPCI subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless it is sooner made operable, provided that during such 7 days all active components of both core spray subsystems and the remaining LPCI subsystem are operable.

4.5.A Core Spray and LPCI Subsystem (cont'd)

Item		Frequency
(c)	Motor Operated valve operability	Once/month
(d)	Pump Flow Rate	Once/3 months

Each LPCI pump shall deliver 10,900 gpm against a system head corresponding to a vessel pressure of 20 psig based on individual pump tests.

(e) DELETED

4. DELETED

5. DELETED

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

3.5.A Core Spray and LPCI Subsystem (cont'd)

★ 6. All recirculation pump discharge valves shall be operable prior to reactor startup (or closed if permitted elsewhere in these specifications).

★ 7. If the requirements of 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 48 hours.

B. Containment Cooling System (HPSW, Torus Cooling, Drywell Spray, and Torus Spray)

1. Except as specified in 3.5.B.2, 3.5.B.3, 3.5.B.4, 3.5.B.5, and 3.5.B.6 below, the containment cooling system shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212 degrees F, and prior to reactor startup from a Cold Shutdown Condition.

4.5.A Core Spray and LPCI Subsystem (cont'd)

6. All recirculation pump discharge valves shall be tested for operability during any period of reactor cold shutdown exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.

B. Containment Cooling System (HPSW, Torus Cooling, Drywell Spray, and Torus Spray)

1. Containment Cooling System components shall be tested as follows:

Item	Frequency
(a) Each HPSW Pump Operability.	Once/month
(b) Each HPSW motor operated valve operability.	Once/month
(c) HPSW Pump Capacity Test. Each HPSW pump shall deliver 4500 gpm at 233 psig.	After pump maintenance and every 3 months.
(d) Each Torus Cooling motor operated valve operability.	Once/month
(e) Each Drywell Spray motor operated valve operability.	Once/month
(f) Each Torus Spray motor operated valve operability.	Once/month
(g) Air test on drywell and torus headers and nozzles.	Once/5 years

Amendment No. 22, 32, 47, 5
148, 195

SEP 15 1994

NRC QUESTION: SRO 40

Unit 3 is operating at 85% pow r at 1020 psig when the following plant conditions occur:

- Reactor pressure spikes to 1043 psig and then stabilizes at 1030 psig.
- Reactor power increases to 91% and then stabilizes at 85%.

Which one of the following failures would cause this plant response?

A. The on-line EHC regulator's setpoint has failed high and the backup regulator is in control.

B. One MSIV disk has separated from its stem and has failed fully closed.

C. One SRV has inadvertently lifted and has failed to fully reseat upon re-closing.

D. The extraction steam to the fifth-point feedwater heater has isolated.

NRC ANSWER:

B

NRC REFERENCE:

Modified PB Exam Bank #4306

295007K201 [3.5/3.7]

UTILITY COMMENT:

B is a correct response.

A is also correct. When the on-line pressure setpoint fails high, the input summer output drops resulting in the backup pressure regulator output taking control by being the highest value into the high value gate. Pressure would be controlled at a higher reactor pressure.

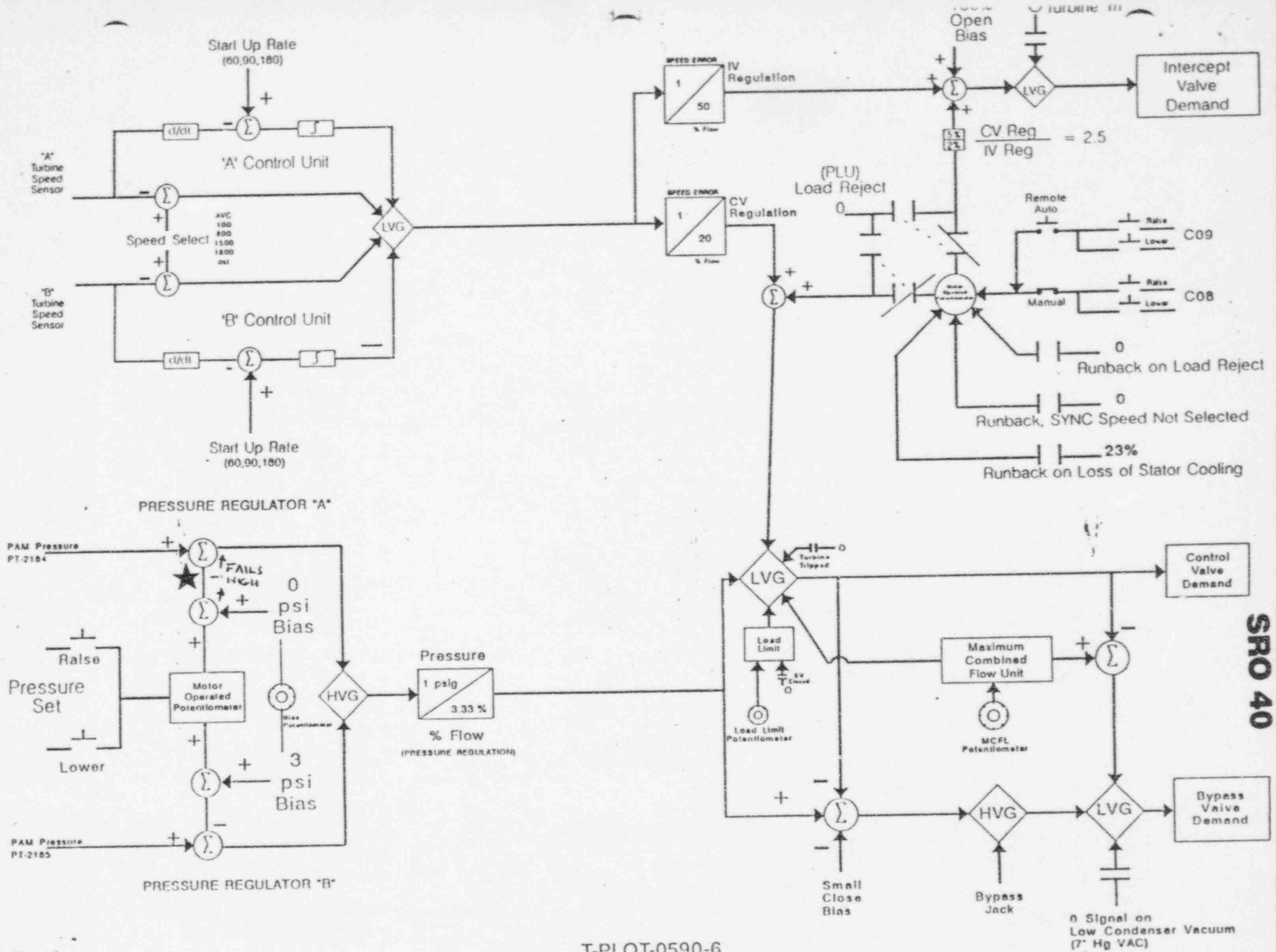
The attached Time Trends from the PBAPS Simulator show that both transients A and B result in pressure and power changes similar to those described in question NRC SRO 40.

UTILITY RECOMMENDATION:

Accept both A and B.

UTILITY REFERENCE:

PLOT-0590, Rev. 007, EHC Logic, page 10 (Attached)
T-PLOT-0590-6 (Attached)
Time Trend SG4 from the PBAPS Simulator (Attached)



SRO 40

Rev 1

T-PLOT-0590-6

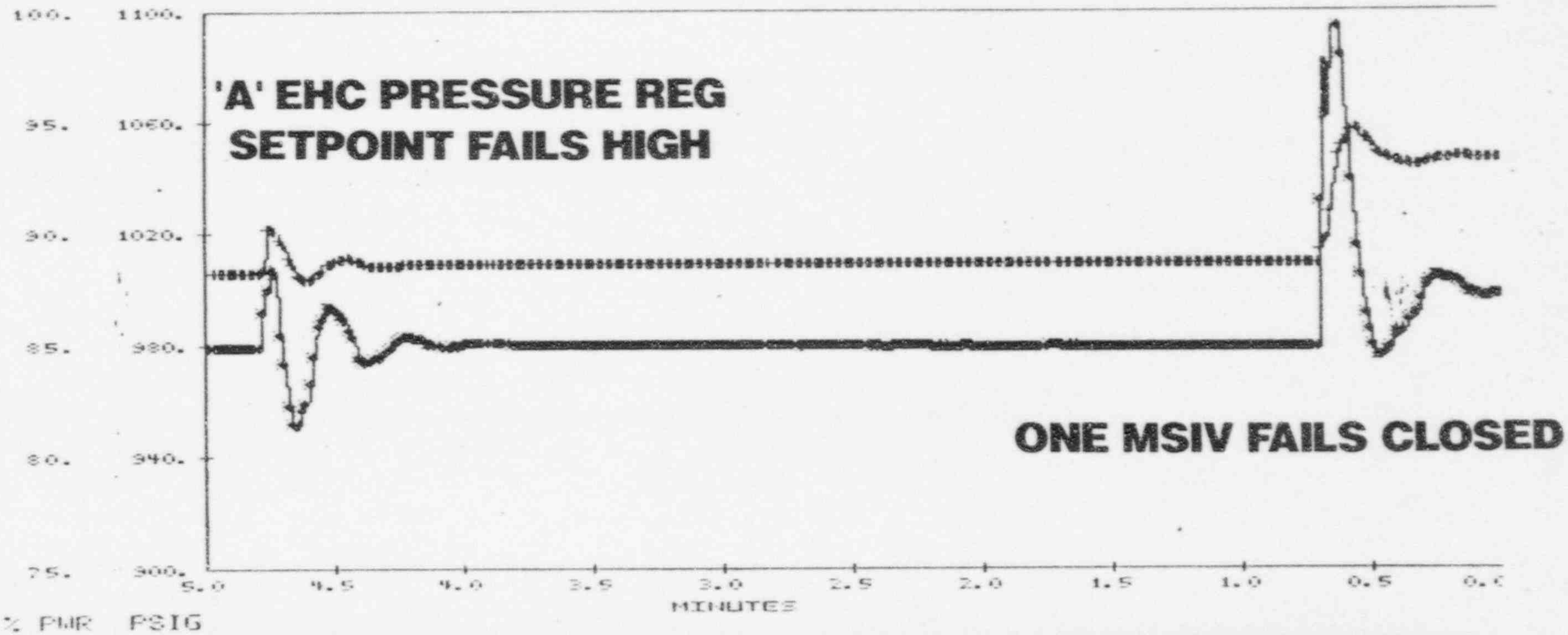
SUBJECT MATTER OUTLINE	SUPPORT INFORMATION
b. Range of 150 to 1100 psig.	
4. Input Summers (A and B)	
a. Inputs	
1) Pressure setpoints from motor-operated potentiometer.	
2) Main Steam Pressure from the averaging manifold.	PT 2(3)184, 5
3) Pressure Bias-Normally +3 is added to the "B" setpoint.	Controlled at adjustment panel in Cable Spreading Room. Permits regulator transfer.
This insures that the "A" summer is controlling and protects against low failure of the governing regulator. ★	If governing regulator fails low CVs would decrease until the non-operating regulator took control.
b. Input Summers sum the inputs algebraically and develop an output which is the pressure error signal.	
c. The output of both summers is sent to the High Value Gate (HVG).	
d. Swapping EHC pressure regulators.	AO 1D.1.2-(3)
5. High Value Gate (HVG)	
a. HVG receives inputs from both input summers.	Normally "A" due to the bias on "B"
b. These signals are compared and the largest signal is passed on to the Gain Unit.	
6. Gain Unit	1 psig is equivalent to 3.33% steam flow
a. Converts input pressure error signal to a steam flow signal which is directly proportional to reactor power.	Adjustable potentiometer in Cable Spreading Room.

SELECT FUNC. KEY OR TURN-ON CODE TT: 8/25/95 14:47:55

SRO 40

TIME TREND FOR SG4

POINT ID	SYMB	DESCRIPTION	QUAL	CFIT	CURRENT	MAXIMUM	MINIMUM
B013	++	RX PRESS	INHB	PROP	1047.	1058.	1004.
B000	**	A APRM PWR	GOOD	PROP	87.	99.	81.



F1=NEW PLOT F2=RESCALE X F3=RESCALE Y F4=QUICK PLT F5=NEXT PLOT F6=PLOT DEF.
PREV CANC PBAPS UNIT=3 CPU=A CONSOLE=PRIMARY MODE=RUN EVENT=

NRC QUESTION: SRO 43/RO 54

Unit 2 is operating at power when a recirculation flow reduction event results in entry into Region 2 of the Power to Flow Map (attached for reference). Plant conditions prior to the event were as follows:

- Reactor power was 90%.
- APRMs indicated 90% +/- 1%.
- All LPRMs were above downscale alarms and below upscale alarms.
- LPRMs near center of core indicated ?5% +/- 2%.
- Period meter indicated infinity.

After the flow reduction event and core flow first reaches its lowest flow rate, which ONE of the following neutron instrumentation responses indicates reactor instability?

A. APRMs are oscillating between 58% and 62% every second.

B. Period meter is oscillating between a -50 second period and a +50 second period every 2 seconds.

C. LPRMs near the center of the core are oscillating between 64% and 72% every 10 to 15 seconds.

D. LPRM downscale alarms occur briefly and then clear at 10 seconds, 25 seconds, and 60 seconds.

NRC ANSWER:

B

NRC REFERENCE:

OT-112, Rev. 16, pg. 1
OT-112 Bases, Rev. 15

295001A106 [3.3/3.4]

UTILITY COMMENT:

See Cautions on Page 4 of GP-9-2 and Notes on Page 1.

Answer B is correct according to Caution #4 and the third Note.

Answer A is correct according to Caution #3 and the second Note.

These Notes and Cautions are consistent with those in OT-112.

UTILITY RECOMMENDATION:

Accept both A and B.

UTILITY REFERENCE:

GP-9-2, Rev. 17, Fast Power Reduction, Cautions on page 4 and Notes on page 1 (Attached)
OT-112, Rev. 18, Notes on page 1 (Attached)

PECO Energy Company
Peach Bottom Unit 2

GP-9-2 FAST REACTOR POWER REDUCTION

1.0 PURPOSE

This procedure provides instructions necessary to rapidly reduce reactor power when required by plant conditions,

2.0 PREREQUISITES

2.1 Plant conditions require a fast reduction in power.

3.0 PERFORMANCE STEPS

3.1 IF rapid power reduction to hot shutdown is required, THEN perform step 3.2. OTHERWISE, proceed to step 3.3.

3.2 Rapid Power Reduction to Hot Shutdown: **CM-6**

3.2.1 As plant conditions permit, transfer house loads.

> **NOTE**
>
> Core thermal hydraulic instability may be occurring if any of the following conditions exist:
>
> - APRM oscillations of greater than OR equal to 10 percent peak-to-peak,
> - ★ LPRM OR APRM oscillations change from random to regular with a period of approximately 1 to 2 seconds, OR
> - ★ SRM Period meters display strong positive-to-negative swings with an oscillation period of approximately 1 to 2 seconds.

3.2.2 IF thermal hydraulic instability is present, THEN place the reactor mode switch in SHUTDOWN AND enter T-100 "SCRAM", AND exit this procedure.

CAUTIONS

Reactor power operation at high power/low flow may result in core thermal hydraulic instability. Core thermal hydraulic instability may be occurring if the following conditions exist:

1. Any LPRM OR APRM noise signal grows by two OR more times its initial noise level.

2. APRM oscillations increase to greater than OR equal to 10 percent (peak to peak).

3. ★ LPRM OR APRM oscillations change from random to regular (with approximately 1 to 2 seconds oscillation period).

4. ★ SRM period meters display strong positive to negative swings (with approximately 1 to 2 seconds oscillation period).

WHEN near Region 1 of Figure 1 (stability exclusion region), THEN control rod withdrawal, core flow reduction AND especially a reduction in Feed Water temperature increases the likelihood of instability. **CM-4, CM-5**

Intentional operation is NOT permitted in REGION 1 OR 2 on Figure 1. These are the regions of the power/flow map where core thermal hydraulic instability is most likely to occur; however, core thermal hydraulic instability can occur at any time. **CM-4**

3.7 IF at any time evidence of core thermal hydraulic instability exists OR entry is made into the stability exclusion region (Region 1) of the Power/Flow map, THEN inform Shift Management THEN immediately scram the reactor AND enter T-100. **CM-5**

3.8 IF Region 2 of Figure 1 is unintentionally entered, THEN inform Shift Management and insert GP-3-2 Appendix 1, "Unit 2 Shutdown Rod Insertion Sequence Instructions", Table 2 rods as required to exit Region 2. **CM-4, CM-5**

3.9 Reduce recirculation flow to minimum.

3.10 Verify Power/Flow conditions are outside Region 2 of Figure 1. **CM-3**

3.11 IF power drops below 40%, THEN verify reactor pressure is less than 1000 psig.

3.12 IF further power reduction is anticipated, THEN exit this procedure and enter GP-3, "Normal Plant Shutdown".

SRO 43/RO 54

PECO Energy Company
Peach Bottom Units 2 and 3

OT-112 - RECIRCULATION PUMP TRIP - PROCEDURE

1.0 ENTRY CONDITIONS

An unplanned trip of ONE or BOTH recirculation pumps while in the STARTUP or RUN modes.

2.0 IMMEDIATE OPERATOR ACTIONS

2.1 IF NO Recirculation Pumps are in operation, THEN Scram AND enter T-100, "Scram", AND execute concurrently with this procedure.

2.2 IF Region 1 of Figure A(B) is entered at any time, THEN Scram AND enter T-100, "Scram", AND execute concurrently with this procedure.

> **NOTE**
>
> Core thermal hydraulic instability may be occurring if any of the following conditions exist:
>
> - APRM oscillations of greater than OR equal to 10 percent peak-to-peak,
> - ★ LPRM OR APRM oscillations change from random to regular with a period of approximately 1 to 2 seconds, OR
> - ★ SRM Period meters display strong positive-to-negative swings with an oscillation period of approximately 1 to 2 seconds.

2.3 IF only ONE Recirculation Pump is in operation, THEN perform the following steps simultaneously:

2.3.1 Fully drive in ALL GP-9-2(3) Appendix 1, Table 1 rods.

2.3.2 Continually monitor for thermal hydraulic instability.

2.3.3 IF thermal hydraulic instability is present, THEN manually scram AND enter T-100, AND execute concurrently with this procedure.

2.3.4 Determine location on Power vs. Flow map using Attachment 1.

NRC QUESTION: SRO 72/RO 76

An event has occurred on Unit 2 that has resulted in the lowering of Torus level. Plant conditions are as follows:

- Torus level is 10.0 feet and steady.
- Torus temperature is 135 deg. F.
- RPV pressure is 1000 psig.

Which one of the following describes the adverse consequences of plant equipment operation at this Torus level?

A. Operation of RCIC will increase Torus pressure causing RCIC to trip on high turbine exhaust pressure.

B. Operation of HPCI will increase Torus pressure and threaten containment integrity.

C. Opening SRVs will rapidly increase Torus pressure.

D. A LOCA will result in overpressurizing the containment.

NRC ANSWER:

D

NRC REFERENCE:

T-102 Bases, Rev. 10, pg. 7
T-102 T/L-8

295030K07 [3.5/3.8]

UTILITY COMMENT:

Answers A and B are Incorrect.

Answer D would result in the most adverse consequences at this Torus level but a LOCA is not usually referred to as a plant equipment operation. Some candidates ruled out this correct answer due to the term "plant equipment operation".

Answer C is the best choice of the three plant equipment operations. Opening SRVs under these plant conditions results in increasing Torus temperature to saturation and then rapidly increasing Torus pressure. The attached simulator Time Trend demonstrates this response which occurs over a few minutes.

UTILITY RECOMMENDATION:

Accept C or D.

UTILITY REFERENCE:

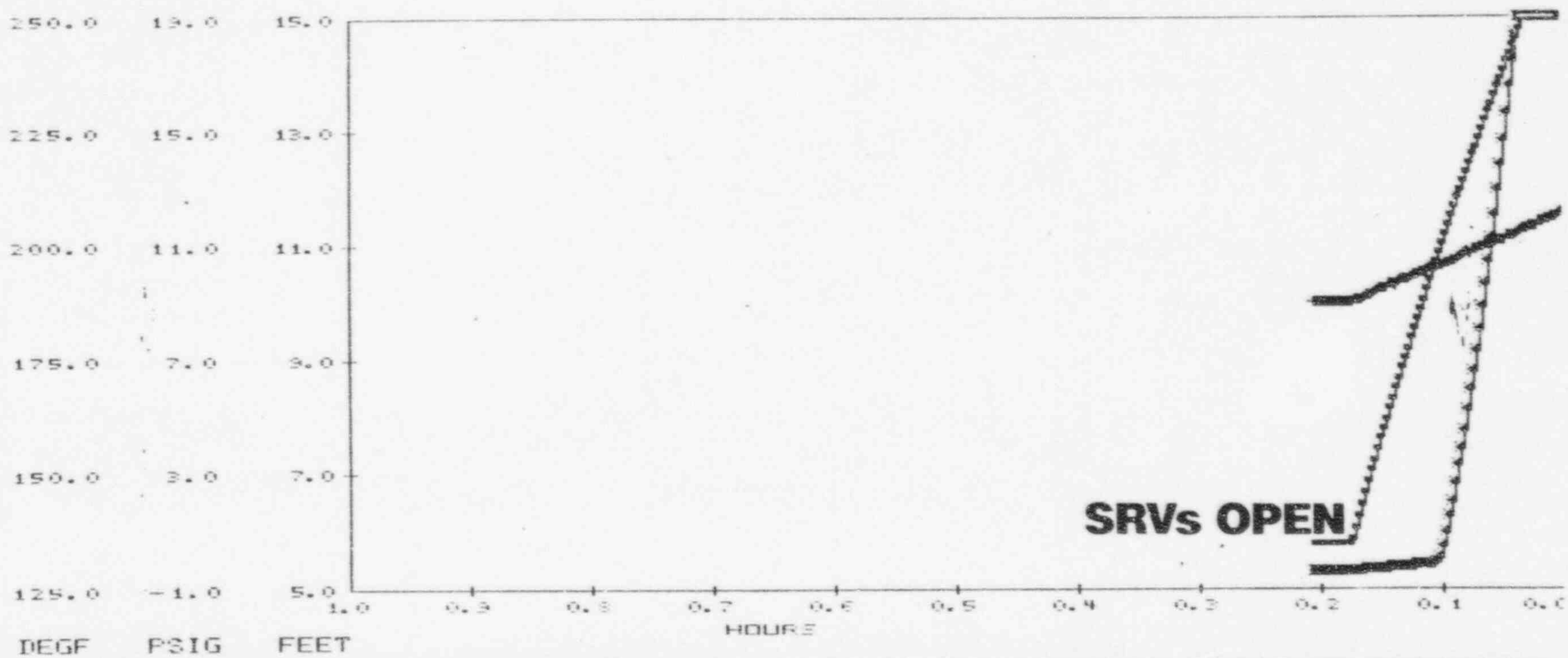
Time Trend SG1 from the PBAPS Simulator (Attached)

SELECT FUNC. KEY OR TURN-ON CODE TT:

8/30/95
11:34:55 [redacted]

TIME TREND FOR SG1

POINT ID	SYMB	DESCRIPTION	QUAL	CFIT	CURRENT	MAXIMUM	MINIMUM
SPDS0037	++	AVG WIDE RNG TORUS LVL	GOOD	AVG	11.7	11.7	9.9
SPDS0041	**	AVG TORUS PRESS	REDU	AVG	34.8	34.8	-0.3
H012	--	A TORUS BULK AVG TEMP	HRL	AVG	271.3	271.3	135.3



SRO 72/RO 76

[redacted]

F1=NEW PLOT F2=RESCALE X F3=RESCALE Y F4=QUICK PLT F5=NEXT PLOT F6=PLOT DEF.
PREV CANC PBAPS UNIT=3 CPU=A CONSOLE=PRIMARY MODE=RUN EVENT=

NRC QUESTION: SRO 74/RO 78

T-102 "Primary Containment Control" is being executed. Step PC/P-9 indicates:

IF Drywell pressure drops below 2 psig,
THEN terminate drywell sprays.

Which one of the following is the basis for Step PC/P-9?

A. To prevent cycling the Drywell-to-Torus vacuum breakers.

B. To prevent opening the Reactor Building-to-Torus vacuum breakers.

C. To prevent developing an excessive differential pressure between the Drywell and the Torus.

D. To prevent developing an excessive differential between the water level inside the downcomer and the Torus.

NRC ANSWER:

B

NRC REFERENCE:

T-102 Bases, pages 12 and 13, PC/P-5 and 9

295010G007 [3.6/3.8]

UTILITY COMMENT:

Students were provided with T-102 which indicated that Step PC/P-9 is actually:

IF Torus level exceeds 18 ft. OR DW press. drops below 2 psig,
THEN terminate DW sprays.

The question asks for the basis for Step PC/P-9.

According to the attached PC/P-9 Bases from T-102, both B and C are correct bases for this step.

UTILITY RECOMMENDATION:

Accept both B and C.

UTILITY REFERENCE:

T-102 Bases for Steps PC/P-5 and 9 (Attached)
OM-P-16.1:1, Rev. 0, pgs. 1 and 2 (Attached)

PC/P PRIMARY CONTAINMENT PRESSURE

PC/P-1 MONITOR AND CONTROL DW AND TORUS PRESSURE

Even though entry to T-102 may not have been from the drywell pressure entry condition, it is still appropriate to monitor drywell and torus pressure. Because changes that may be occurring in other containment parameters can potentially change containment pressure it is appropriate to monitor the pressure so any increases can be corrected in a timely manner.

PC/P-2 CONTROL PRI CTMT PRESS BELOW 2 PSIG USING OT-101 IF NECESSARY

OT-101 gives direction for immediate and follow up actions to respond to increasing drywell pressure for pressures below 2 psig.

PC/P-3 IS PRI CTMT PRESS BELOW 2 PSIG

This question is asked to determined if the steps that follow must be executed.

PC/P-4 VERIFY AS APPROPRIATE: MANUAL ISOL OF RBCCW AND DW CHILLED WATER USING GP-8B **CM-3**

The DW chilled water and RBCCW containment isolation valves do not automatically close on a containment isolation. Leakage from the drywell via these systems is possible if drywell pressure is greater than system pressure and system integrity is lost. The operator is directed to GP-8B for specific guidance on identifying and correcting conditions that can lead to leakage. The step is an override because of the possibility of changing conditions through the event.

PC/P-5 IF TORUS PRESS DROPS BELOW 2 PSIG, THEN TERMINATE TORUS SPRAYS

This step is an override that applies through the rest of the PC/P leg. Terminating torus sprays when torus pressure drops below 2 psig prevents a possible excessive negative pressure in the torus. Negative pressure in the containment is undesirable because Rx ★ building-to-torus vacuum breaker operation can add air and thus deinert the atmosphere. Additionally, if the negative pressure is too great, the containment could collapse. It is permitted to terminate sprays before 2 psig if it is desired to maintain pressure above the 2 psig setpoint for LOCA initiation. This is an important point because reinitiation of automatic actions that could occur if the signal clears and subsequently recurs could complicate event response. The decision to terminate sprays at 2 psig or just above 2 psig should be made by Shift Management based on current expectations of containment pressure trends.

PC/P-6 BEFORE TORUS PRESSURE REACHES 9 PSIG SPRAY THE TORUS PER T-203
USE ONLY THOSE RHR PUMPS NOT CONTINUOUSLY REQUIRED TO ASSURE ADEQUATE CORE COOLING

9 psig is the Torus Spray Initiation Limit. The limit is explained in the Curves, Tables and Limits Bases and basically is designed to give a threshold for taking actions to reduce the pressure in the torus before conditions are met for chugging in the downcomers. Chugging refers to a problem that can occur when steam from a LOCA break in the drywell passes to the exit of the downcomers and condenses. The pressure of the steam initially displaces water down and out of the downcomer, but as the steam condenses, water pushes back up into the space originally occupied by the steam bubbles. This phenomenon occurs cyclically and causes stresses that could eventually lead to failure at the junction of the downcomer and vent header. Torus sprays may not prevent chugging, but if the pressure problem is solved, the eventual use of drywell sprays may not be required.

PC/P-7 IS TORUS PRESS BELOW 9 PSIG

PC/P-8 MAINTAIN TORUS PRESS BELOW 9 PSIG

These steps direct the operator to continue with actions taken up to this point if torus pressure is being maintained below 9 psig. If torus sprays were unsuccessful in reducing pressure then further response is required to prevent chugging.

PC/P-9	<u>IF</u>	TORUS LEVEL EXCEEDS 18 FT <u>OR</u> DW PRESS DROPS BELOW 2 PSIG,
	<u>THEN</u>	TERMINATE DW SPRAYS

Terminating drywell sprays when drywell pressure drops below 2 psig has the same basis as PC/P-5. Torus levels below 18 ft assure that no part of the drywell side of the torus to drywell vacuum breakers will be submerged when the vacuum breakers need to be operable. The vacuum break capability is important during the use of drywell sprays because condensing of steam in the drywell can produce negative differential pressures sufficient to collapse the containment.

The same flexibility regarding termination of drywell sprays at 2 psig or at some pressure just above 2 psig that was discussed for step PC/P-5 also is permitted in this case.

PC/P-10	<u>IF</u>	WHILE EXECUTING THE FOLLOWING STEPS: TORUS LEVEL IS BELOW 18 FT <u>AND</u> THE COMBINATION OF DW BULK AVG TEMP AND DW PRESS IS BELOW CURVE PC/H-1,

Sprays produce a temperature and pressure reduction through evaporative cooling and convective cooling. Evaporative cooling occurs when water is sprayed into an atmosphere <u>hotter</u> then saturation temperature. Heat energy is absorbed as the spray droplets flash to steam (evaporate). This causes a <u>very</u> rapid reduction in the pressure because this is a very efficient means of heat transfer. Convective cooling occurs when water is sprayed into a saturated atmosphere. Heat is transferred as the spray water droplets increase in temperature as they pass through the steam. This produces a slower, more controlled pressure decrease. The drywell spray initiation limit is designed to assure that the pressure drop that occurs when drywell

NRC QUESTION: SRO 80/RO 83

Step SC/L-1 of T-103 "Secondary Containment Control" directs the following action:

MONITOR AND CONTROL SECONDARY CONTAINMENT WATER LEVELS.

Which one of the following is an appropriate operator action that would be taken to accomplish this step?

A. Isolate any leaking ECCS system even though it is currently the only system available to maintain RPV level.

B. Direct maintenance to immediately install sandbags around the outside of the room doors.

C. Plot the rate of increase in the sump water levels to determine whether sump capacity is sufficient.

D. Start all available sump pumps and operate them to remove water from the sumps.

NRC ANSWER:

D

NRC REFERENCE:

T-103 Bases, Rev. 6, Step SC/L-1, pg. 5

295036A101 [3.0/3.2]

UTILITY COMMENT:

This question does not specifically indicate the existence of a problem with secondary containment water levels; T-103 may have been entered due to containment temperature or radiation problems.

Step SC/L-1 directs the operator to monitor and control secondary containment water levels. The bases for Step SC/L-1 indicates that the operator should monitor the level and it permits action (e.g., operation of sump pumps) to keep water levels from exceeding their alarm setpoints. (Step SC/L-6 specifically directs operation of sump pumps to restore and maintain sump levels.)

Answer C is correct as it indicates monitoring of sump levels. OM-P-16.1:1 directs SROs to write parameter values on the TRIP flowcharts for "monitor and control steps". These values are recorded to maintain the CRS or other SRO's awareness of parameter status. This expectation is reinforced during simulator training.

Answer D is correct as it indicates the use of sump pumps.

UTILITY RECOMMENDATION:

Accept both C and D.

UTILITY REFERENCE:

T-103 Bases for Step SC/L-1 and SC/L-6.
OM-P-16.1:1, Control Room Supervisor Direction and Control During Unexpected Conditions and Transients

SC/R

SC/R-1 MONITOR AND CONTROL SECONDARY CONTAINMENT RAD LEVELS

Since the operator is directed to execute all three flow paths concurrently, he/she must be aware of the respective parameter values. This step directs the operator to monitor radiation levels within secondary containment in order to determine if additional actions will be required. Actions to prevent area radiation levels from exceeding their alarm levels, such as placing equipment cell exhaust ventilation on SBGTS, are permitted by this step.

SC/R-2 IS ANY AREA RAD LEVEL ABOVE AN ALARM LEVEL LISTED IN TABLE SC/R-1

The high alarm setpoint for the Area Radiation Monitors (ARMs) are set to correspond to the Maximum Normal Operating (MNO) radiation levels for the particular area. Typically, the setpoints are 3 times the normal full power background levels or 5 mr/hr, whichever is lower. Exact values are not listed on this table since the setpoints can be changed every quarter by Chemistry. The setpoints for the ARMs are posted on the 2(3)0C014 Panels.

The question in SC/R-2 is asked to determine if further operator action is required due to elevated radiation levels. A YES response (rad levels > ARM alarm setpoint) will lead to additional direction in the SC/R leg. A NO response (rad levels < ARM alarm setpoint) will direct the operator back to step SC/R-1 and continued monitoring.

SC/R-3 PERFORM A LOCAL EVACUATION IF NECESSARY USING GP-15

If area radiation levels are above the ARM setpoints, any personnel in the effected area should be evacuated to prevent unnecessary overexposure and possible contamination. Procedure GP-15 is used to perform local plant evacuations.

NOTE

Following this step, the SC/R leg joins the SC/T and SC/L legs and the procedure continues at step SCC-3.

SC/L

SC/L-1 MONITOR AND CONTROL SECONDARY CONTAINMENT WATER LEVELS

As previously mentioned, all three secondary containment parameters must be monitored while performing T-103. This step directs the operator to monitor reactor building water ★ levels. Action to prevent water levels from exceeding their alarm setpoints (e.g. operation of sump pumps) is also permitted by this step.

SC/L-2 IS ANY AREA WATER LEVEL ABOVE AN ALARM LEVEL LISTED IN TABLE SC/L-2

The alarm levels listed in table SC/L-2 are the setpoints for the room flood alarms. Indication of water level above these values can be determined by the respective alarms, or from local inspection. A YES response (level > alarm) will lead to step SC/L-4 and additional action. A NO response (level < alarm) will lead to step SC/L-3 and a question on reactor building sump levels.

SC/L-3 IS THE RX BLDG FLOOR DRAIN SUMP LEVEL ABOVE THE HI HI ALARM LEVEL

Any water system breach on elevations above 91'6" will eventually reach the reactor building floor drain sump via the drain system. This question is asked in conjunction with step SC/L-2 to determine if a water level problem exists within secondary containment. A YES response (sump >hi hi alarm) will lead to step SC/L-4 and additional action. A NO response (sump < hi hi alarm) will direct the operator back to step SC/L-1 and continued monitoring.

SC/L-4 PERFORM A LOCAL EVACUATION IF NECESSARY USING GP-15

Consideration should be given to evacuating personnel in areas where water levels have exceeded the alarm levels. This is done for personnel safety, as well as to prevent possible contamination. Procedure GP-15 provides the necessary direction for a local evacuation.

SC/L-5 REFER TO SE-9 FOR ADDITIONAL INSTRUCTIONS

Procedure SE-9, "Radioactive Spill," provides direction to contain any radioactive spill within secondary containment. This procedure is referenced to aid the operator in minimizing a radioactive release outside of secondary containment.

SC/L-6 OPERATE AVAILABLE SUMP PUMPS TO RESTORE AND MAINTAIN WATER LEVEL BELOW THE ALARM LEVELS

★

The most efficient method to reduce reactor building area water levels is through the operation of the installed sump pumps. This step also allows the use of any portable pumps which may be available and can be temporarily installed.

NOTE

At this point the SC/L leg joins the SC/R and SC/T flow paths, and the procedure continues at step SCC-3.

REVIEW	PB
PORC	NO
SQR	YES
NQA	NO
50.59	NO

SRO 74/RO 78

CONTROLLED BY
DS

2-10-95

OSPS: CONTROL ROOM SUPERVISOR USE OF TRIP PROCEDURES - GENERAL

1.0 PURPOSE

1.1 Provide expectations for the method of using the T-100 TRIP and other flowcharted procedures. This OSPS does not apply to T-200 or T-300 series TRIP procedures.

2.0 COMPLIANCE WITH TRIP PROCEDURES

2.1 The TRIP procedures are to be followed exactly. The leg which the CRS progresses through is dictated by the specific transient. The CRS is expected to progress through the applicable flowcharts until exit requirements are met. Fully completing a step may not be required prior to moving to the next step. Annotating those steps where orders have been given but not completed will be annotated as shown in Attachment 1. This ensures other crew members conducting overview of the execution of the TRIPs can clearly see what steps have been ordered, executed, or not completed. Under no circumstances is it acceptable to skip steps in the TRIPs.

3.0 SATISFACTORY PERFORMANCE

3.1 Satisfactory performance includes compliance with TRIP procedures as described above and:

1. Entry conditions, entry paths, and exit paths are annotated as displayed on Attachment 1.

2. The flowchart paths are marked as the CRS progresses through the flowchart, showing decision paths taken. It is not necessary to erase and reinitialize TRIP flowcharts when reentry conditions occur. It is expected that the CRS clearly annotate the steps each time the "procedure leg" is reentered.

3. Each step and caution is checked when performed. Steps ordered by the CRS to be completed, but which are not complete, should be circled and then checked following completion. Check and recheck steps should be checked each time the step is read.

4. Where multiple actions are listed together in a step box, each action is annotated as the action is taken or determined to be inappropriate.

5. Where multiple methods are listed together in a step box, the method(s) used are annotated.

6. When plant conditions do not allow the step to be performed, each action is annotated with an "X".

7. Calculations are completed and locations with respect to "SAFE/UNSAFE" curves are annotated.

★ 8. Data, such as parameter values associated with entry conditions, "monitor and control" steps and limits contained in step boxes, are written on the flowchart as needed by the CRS or other SRO to maintain an awareness of parameter status.

4.0 UNSATISFACTORY PERFORMANCE

4.1 Unsatisfactory performance consists of one or more of the following:

1. Inability to progress through the TRIP flowchart(s).

2. Failure to progress through the TRIP flowchart(s) at the pace dictated by the transient.

3. Noncompliance with a TRIP, for example:

   a. Making an incorrect decision.
   b. Making a decision to not adhere to the TRIP flowchart.
   c. Improper prioritization of "procedure leg" that results in an adverse transient response.
   d. Skipping procedure step(s) that results in an adverse transient response or degraded plant condition.
   e. Annotation does not provide adequate information to determine decisions made by the CRS.

5.0 REMEDIAL TRAINING

5.1 Remedial training will be administered in accordance with OM-P-16.1, Section 5.0.

NRC QUESTION: SRO 91

Due to an accident, a reactor operator is temporarily unable to meet the medical requirements of his/her license as determined by Occupational Health and Safety.

Which one of the following is the required action in accordance with A-C-10 "Operator Licenses"?

A. The NRC must be notified in writing within 30 days.

B. The operator must be removed administratively from licensed duties.

C. A conditional license request must be submitted to the NRC within 30 days.

D. The operator's license must be surrendered to the NRC.

NRC ANSWER:

B

NRC REFERENCE:

A-C-10, Rev. 0, Step 7.4.7, pg. 13

294001A103 [2.7/3.7]

UTILITY COMMENT:

This question does not identify the duration of the inability to meet medical requirements.

Given a short duration, inability to meet medical requirements, A-C-10 gives the guidance that the operator be removed administratively from licensed duties. This makes answer B correct.

10CFR55.25 and 10CFR50.74 require that the NRC be notified within 30 days when a Licensed Operator becomes incapacitated due to disability or illness. This is also directed by the PECO Reportability Reference Manual, Section 3.1, Page 12, NP-36. Since the duration of the inability to meet medical requirements is not specifically identified, answer A is a conservatively correct action as identified in these references.

For an actual occurrence, verbal discussions between the utility and the NRC would clarify the appropriate response for the specific case.

UTILITY RECOMMENDATION:

Accept both A and B.

UTILITY REFERENCE:

A-C-10, Rev. 0
10CFR50.74, Notification of Change in Operator or Senior Operator Status
10CFR55.25, Incapacity because of Disability or Illness
PECO Reportability Reference Manual, Section 3.1, Page 12, NP-36.

d. The request for a conditional license may be included with the notification required by Section 7.4.5.3 of this procedure if the decision is made within the 30 day time period required by 10CFR50.74 and 10CFR55.25.

e. The licensed operator may be returned to licensed duties by the Senior Manager Operations/Manager Reactor Services prior to issuance of a conditional license by the NRC provided Occupational Health and Safety, based on the guidance provided in ANSI/ANS 3.4-1983, has recommended to the Senior Manager Operations/Manager Reactor Services that the operator can return to licensed duties with the accommodation (e.g., the use of corrective lenses). This recommendation of medical qualification shall be documented and retained for future NRC inspection.

7.4.6 If a medical condition requiring accommodation, as determined by Occupational Health and Safety, exists prior to an individual applying for an operator license, then the appropriate medical certification and supporting medical evidence shall be included in the initial license application package prepared by the Training Director.

7.4.7 ★ If a licensed operator is temporarily unable to meet the medical requirements as determined by Occupational Health and Safety due to a temporary illness or disability, the Senior Manager Operations/Manager Reactor Services shall administratively remove the operator from licensed duties until the operator is once again certified by Occupational Health and Safety to meet all medical requirements. In this case, no notification to the NRC nor request for a conditional license for the temporary disability is required.

7.5 **LICENSE STATUS [10CFR55.53]**

7.5.1 Maintaining an Active Status License

1. The Senior Manager Operations/Manager Reactor Services shall ensure that all on-shift licensed operators maintain an active status license.

2. To maintain an active status license, each licensed operator shall actively perform the functions of a licensed operator or senior operator for a minimum of seven 8-hour or five 12-hour shifts per calendar quarter. The purpose of this requirement is to maintain operator proficiency relative to licensed duties.

(1) A research or training reactor as part of the individual's training as a student, or

(2) A facility as a part of the individual's training in a facility licensee's training program as approved by the Commission to qualify for an operator license under this part.

(b) Under the direction and in the presence of a licensed senior operator, manipulates the controls of a facility to load or unload the fuel into, out of, or within the reactor vessel.

Subpart C—Medical Requirements

§ 55.21 Medical examination.

An applicant for a license shall have a medical examination by a physician. A licensee shall have a medical examination by a physician every two years. The physician shall determine that the applicant or licensee meets the requirements of § 55.33(a)(1).

§ 55.23 Certification.

To certify the medical fitness of the applicant, an authorized representative of the facility licensee shall complete and sign Form NRC-396, "Certification of Medical Examination by Facility Licensee," available from Records and Reports Management Branch, Division of Information Support Services, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

(a) Form NRC-396 must certify that a physician has conducted the medical examination of the applicant as required in § 55.21.

(b) When the certification requests a conditional license based on medical evidence, the medical evidence must be submitted on NRC Form 396 to the Commission and the Commission then makes a determination in accordance with § 55.33.

[52 FR 9460, Mar. 25, 1987, as amended at 53 FR 43421, Oct. 27, 1988]

★ § 55.25 Incapacitation because of disability or illness.

If, during the term of the license, the licensee develops a physical or mental condition that causes the licensee to fail to meet the requirements of § 55.21 of this part, the facility licensee shall notify the Commission within 30 days of learning of the diagnosis. For conditions for which a conditional license (as describing in § 55.33(b) of this part) is requested, the facility licensee shall provide medical certification on Form NRC 396 to the Commission (as described in § 55.23 of this part).

§ 55.27 Documentation.

The facility licensee shall document and maintain the results of medical qualifications data, test results, and each operator's or senior operator's medical history for the current license period and provide the documentation to the Commission upon request. The facility licensee shall retain this documentation while an individual performs the functions of an operator or senior operator.

Subpart D—Applications

§ 55.31 How to apply.

(a) The applicant shall:

(1) Complete Form NRC-398, "Personal Qualification Statement—Licensee," available from Records and Reports Management Branch, Division of Information Support Services, U.S. Nuclear Regulatory Commission, Washington, DC 20555;

(2) File an original and two copies of Form NRC-398, together with the information required in paragraphs (a)(3), (4), (5) and (6) of this section, with the appropriate Regional Administrator;

(3) Submit a written request from an authorized representative of the facility licensee by which the applicant will be employed that the written examination and operating test be administered to the applicant;

(4) Provide evidence that the applicant has successfully completed the facility licensee's requirements to be licensed as an operator or senior operator and of the facility licensee's need for an operator or a senior operator to perform assigned duties. An authorized representative of the facility licensee shall certify this evidence on Form NRC-398. This certification must include details of the applicant's qualifications, and details on courses of instruction administered by the fa-

SRO 91

submit to the Commission under the provisions of this section must be of sufficient quality to permit legible reproduction and micrographic processing.

(f) *Exemptions.* Upon written request from a licensee including adequate justification or at the initiation of the NRC staff, the NRC Executive Director for Operations may, by a letter to the licensee, grant exemptions to the reporting requirements under this section.

(g) *Reportable occurrences.* The requirements contained in this section replace all existing requirements for licensees to report "Reportable Occurrences" as defined in individual plant Technical Specifications.

[48 FR 33858, July 26, 1983, as amended at 49 FR 47824, Dec. 7, 1984; 51 FR 40310, Nov. 6, 1986; 56 FR 23473, May 21, 1991; 56 FR 61352, Dec. 3, 1991]

§ 50.74 Notification of change in operator or senior operator status.

Each licensee shall notify the Commission in accordance with § 50.4 within 30 days of the following in regard to a licensed operator or senior operator:

(a) Permanent reassignment from the position for which the licensee has certified the need for a licensed operator or senior operator under § 55.31(a)(3) of this chapter;

(b) Termination of any operator or senior operator;

★ (c) Disability or illness as descrided in § 55.25 of this chapter.

[52 FR 9469, Mar. 25, 1987]

§ 50.75 Reporting and recordkeeping for decommissioning planning.

(a) This section establishes requirements for indicating to NRC how reasonable assurance will be provided that funds will be available for decommissioning. For electric utilities it consists of a step-wise procedure as provided in paragraphs (b), (c), (e), and (f) of this section. Funding for decommissioning of electric utilities is also subject to the regulation of agencies (e.g., Federal Energy Regulatory Commission (FERC) and State Public Utility Commissions) having jurisdiction over rate regulation. The requirements of this section, in particular paragraph (c), are in addition to, and not substitution for, other requirements, and are not intended to be used, by themselves, by other agencies to establish rates.

(b) Each electric utility applicant for or holder of an operating license for a production or utilization facility of the type and power level specified in paragraph (c) of this section shall submit a decommissioning report, as required by § 50.33(k) of this part containing a certification that financial assurance for decommissioning will be provided in an amount which may be more but not less than the amount stated in the table in paragraph (c)(1) of this section, adjusted annually using a rate at least equal to that stated in paragraph (c)(2) of this section, by one or more of the methods described in paragraph (e) of this section as acceptable to the Commission. The amount stated in the applicant's or licensee's certification may be based on a cost estimate for decommissioning the facility. As part of the certification, a copy of the financial instrument obtained to satisfy the requirements of paragraph (e) of this section is to be submitted to NRC.

(c) Table of minimum amounts (January 1986 dollars) required to demonstrate reasonable assurance of funds for decommissioning by reactor type and power level, P (in MWt); adjustment factor.[1]

	Millions
(1)(i) For a PWR:	
greater than or equal to 3400 MWt	$105
between 1200 MWt and 3400 MWt (For a PWR of less than 1200 MWt, use P=1200 MWt)	$(75+0.0088P)
(ii) For a BWR:	
greater than or equal to 3400 MWt	$135

[1] Amounts are based on activities related to the definition of "Decommission" in § 50.2 of this part and do not include the cost of removal and disposal of spent fuel or of nonradioactive structures and materials beyond that necessary to terminate the license.

ID#	REPORT/SUBJECT	REQUIREMENTS	RECIPIENT	DATE DUE & METHOD OF REPORTING	CONTENT	RESP IND
NP-27	Renewal of financial protection liability insurance.	10CFR140.17(b)	Director, NRR	Written notification of renewal at least 30 days prior to termination of policy, or file other proof of financial protection.	Notification of policy renewal or other proof of financial protection	VP, SSD
NP-28	Conviction for a felony of any licensed Reactor Operator (RO), Senior Reactor Operator (SRO) or Limited Senior Reactor Operator (LSRO).	10CFR55.53(g)	NRC Regional Administrator	Within 30 days of conviction.	Notification of felony conviction.	Site VP
NP-29	Permanent reassignment from the position for which the licensee has certified the need for a licensed Reactor Operator (RO), Senior Reactor Operator (SRO) or Limited Senior Reactor Operator (LSRO).	10CFR50.74(a)	NRC Document Control Desk, NRC Regional Administrator, NRC Resident Inspector	Within 30 days of reassignment.	Notification of permanent reassignment of a licensed operator or senior operator.	Site VP
NP-30	Termination of any licensed Reactor Operator (RO), Senior Reactor Operator (SRO) or Limited Senior Reactor Operator (LSRO).	10CFR50.74(b)	NRC Document Control Desk, NRC Regional Administrator, NRC Resident Inspector	Within 30 days of termination.	Notification of termination of a licensed operator or senior operator.	Site VP
NP-33	The occurrence of routine or planned impairments to fire protection equipment for simple one-day or less impairments (e.g., routine maintenance and surveillance work, adding a few sprinkler heads, changing diesel fire pump batteries), and for complex, large-scale, or long-term impairments (e.g., re-routing fire protection underground, draining and painting fire protection suction tank or reservoir).	ANI Information Bulletin 5A(81)	ANI	Advance notification of actual shutdown should not be less than: 24 hours for simple one-day or less impairments; one-week, or as much in advance as possible, for complex, large-scale, or long-term impairments.	See ANI Information Bulletin 5A(81).	Site VP
★ NP-36	Disability or illness as described in 10CFR55.25 of any licensed Reactor Operator (RO), Senior Reactor Operator (SRO) or Limited Senior Reactor Operator (LSRO).	10CFR50.74(c)	NRC Document Control Desk, NRC Regional Administrator, NRC Resident Inspector	Within 30 days of diagnosis.	Notification of disability or illness of a licensed operator or senior operator.	Site VP
NP-38	Identification of IGSCC cracks.	GL 88-01	Director, NRR	Verbal notification followed by written notification and request for approval. NRC approval of flaw evaluations and/or repairs is required prior to resumption of operation.		Site VP

SRO 91

NRC QUESTION: SRO 99

The Shift Manager and the CRS have just determined that an LCO has been entered and are unable to notify the Senior Manager of Operations.

Which one of the following managers must be notified by the Shift Manager in accordance with OM-P-12.1 "Limiting Conditions for Operations"?

A. Operations Services Manager

B. Operations Support Group Manager

C. Plant Manager

D. Plant Maintenance Manager

NRC ANSWER:

A

NRC REFERENCE:

OM-P-12.1, Rev. 0, Step 3.1, pg. 2

294001A110 [3.6/4.2]

UTILITY COMMENT:

Technical Specification 6.1.1 indicates that the Plant Manager is responsible for overall facility operation. Portions of his authority may be delegated to his direct reports. The receipt of information regarding LCO entries has been delegated to the Senior Manager - Operations. Informing either the Plant Manager or the Senior Manager - Operations is appropriate. Answer "C" is correct in accordance with Tech Spec 6.1.1.

OM-P-12.1 "Limiting Conditions for Operations" documents the Senior Manager - Operations' further delegation of this responsibility in his absence. Answer "A" is correct in accordance with OM-P-12.1.

Surveillance Tests are used to ensure LCO compliance. When a surveillance is unsatisfactory, Shift Management are expected to use their discretion whether to notify the Plant Manager or others (as appropriate). See the attached example of an ST Coversheet which shows this UNSAT notification.

UTILITY RECOMMENDATION:

Accept both A and C.

UTILITY REFERENCE:

OM-P-12.1 (Attached)
Technical Specification 6.1.1 (Attached)
Sample ST Coversheet (Attached)

6.0 ADMINISTRATIVE CONTROLS

6.1 Responsibility

★ 6.1.1 The Plant Manager shall be responsible for overall facility operation. In the absence of the Plant Manager, the Superintendent - Operations (or any other person that the Plant Manager may designate in writing) shall assume the Manager's responsibility for overall facility operation.

6.1.2 The Shift Supervisor (or, during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President, Peach Bottom Atomic Power Station shall be reissued to all station personnel on an annual basis.

6.2 Organization

6.2.1 Offsite and Onsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Peach Bottom Quality Assurance Plan.

★ b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President, Peach Bottom Atomic Power Station shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

3.0 NOTIFICATION

3.1 The Shift Manager shall be responsible for notifying the Senior Manager of Operations or the Operations Services Manager:

1. Prior to intentionally entering a situation in which the LCO requirements will not be satisfied (unless in a situation where immediate action is required).

★

2. As soon as possible after unplanned events create a situation in which the LCO requirements are not satisfied.

3. When it becomes apparent that an LCO might not be cleared as anticipated.

4. When it is determined that the plant or any plant activity is not in compliance with Technical Specifications or applicable rules and regulations.

5. Any time that an unsafe condition exists or is suspected to exist.

3.2 If an LCO is entered <u>AND</u> the Tech Spec action statement requires a written report to the NRC (either immediately or if the LCO is not satisfied within a certain time period), Shift Management shall ensure a PEP Issue is initiated in accordance with procedure LR-C-10, Performance Enhancement Program.

4.0 LCO ENTRY CONDITIONS <<T01075>>

4.1 The decision to enter an LCO should be based on reasonable evidence that the parameter in question is at the limit. "Reasonable evidence" includes, but is not limited to, recorder traces, indicators, alarms, automatic actions associated with Limiting Safety System Settings, and general knowledge of current plant conditions and events.

4.2 When a parameter is displayed by instruments having different ranges, the instrument with the narrowest range should be used in determining whether a Tech Spec limit is not met. In the situation where a parameter is displayed by more than one instrument of similar accuracies, enter the LCO when any single, reliable instrument reaches the Tech Spec limit.

5.0 TRACKING LIMITING CONDITIONS FOR OPERATION <<T00220>>

5.1 Limiting Conditions for Operation Logs are provided in the computerized Unified Control Room Log (UCRL), one each for Units 2 and 3, and a third for common equipment. Exhibit OM-P-12.1:1, Limiting Conditions for Operation Log, shall be used in the event of computer malfunctions and the data entered into the UCRL at a later time.

5.2 When a condition causes entry into an LCO, the applicable LCO Log shall be used to record the condition <u>AND</u> the actions required under present plant operating conditions. Exhibit OM-P-12.1:2, Example LCO Log, provides example LCO Log entries.

ST-I-023-100-2 HPCI LOGIC SYSTEM FUNCTIONAL TEST

5-15-95
Approval

TEST FREQUENCY: Once/Operating Cycle
TECH SPEC: 4.2.B, Table 4.2.B.1.10, 4.7.2
Table 4.2.B.2.4, Table 4.2.B.2.5, 4.5.C.1.(a)

1 **CHECK** why this procedure is being performed:

☐ Schedule ☐ OVF ☐ Retest Due To Unsat Test

☐ Other Reason: ______________________________

Approved By SMgt: ______________ ______ __/__/__ ______
Printed Name Time Date Initials

2 **INITIAL** one of the following Test Results:

A: All * steps are **SATISFACTORY** ______

B: One or More * steps are **UNSATISFACTORY** ______
Refer to Section 9.0 for Tech Spec LCO's

Performed By: ______________ ______ __/__/__ ______
Printed Name Time Date Initials

RO/CO Informed of Test Completion: ______________ ______ __/__/__ ______

SMgt Informed of Test Results: ______________ ______ __/__/__ ______

★ UNSAT Notification: ______________________________
SMgt Discretion: Plant Mgr or Others

Notified By: ______________ ______ __/__/__ ______

3 **IF** other portions of the test did **NOT** function properly, **OR** other discrepancies were noted **THEN COMPLETE** the following:

DESCRIBE discrepancies/actions taken: A/R or ETT #: ______

____

____

____

4 Reviewed/Approved Plant Staff: ______________ ______ __/__/__ ______
Printed Name Time Date Initials

3.0 NOTIFICATION

3.1 The Shift Manager shall be responsible for notifying the Senior Manager of Operations or the Operations Services Manager:

1. Prior to intentionally entering a situation in which the LCO requirements will not be satisfied (unless in a situation where immediate action is required).

2. As soon as possible after unplanned events create a situation in which the LCO requirements are not satisfied.

3. When it becomes apparent that an LCO might not be cleared as anticipated.

4. When it is determined that the plant or any plant activity is not in compliance with Technical Specifications or applicable rules and regulations.

5. Any time that an unsafe condition exists or is suspected to exist.

3.2 If an LCO is entered <u>AND</u> the Tech Spec action statement requires a written report to the NRC (either immediately or if the LCO is not satisfied within a certain time period), Shift Management shall ensure a PEP Issue is initiated in accordance with procedure LR-C-10, Performance Enhancement Program.

4.0 LCO ENTRY CONDITIONS <<T01075>>

4.1 The decision to enter an LCO should be based on reasonable evidence that the parameter in question is at the limit. "Reasonable evidence" includes, but is not limited to, recorder traces, indicators, alarms, automatic actions associated with Limiting Safety System Settings, and general knowledge of current plant conditions and events.

4.2 When a parameter is displayed by instruments having different ranges, the instrument with the narrowest range should be used in determining whether a Tech Spec limit is not met. In the situation where a parameter is displayed by more than one instrument of similar accuracies, enter the LCO when any single, reliable instrument reaches the Tech Spec limit.

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5.1 Limiting Conditions for Operation Logs are provided in the computerized Unified Control Room Log (UCRL), one each for Units 2 and 3, and a third for common equipment. Exhibit OM-P-12.1:1, Limiting Conditions for Operation Log, shall be used in the event of computer malfunctions and the data entered into the UCRL at a later time.

5.2 When a condition causes entry into an LCO, the applicable LCO Log shall be used to record the condition <u>AND</u> the actions required under present plant operating conditions. Exhibit OM-P-12.1:2, Example LCO Log, provides example LCO Log entries.

ST-I-023-100-2 HPCI LOGIC SYSTEM FUNCTIONAL TEST

5-15-95
Approval

TEST FREQUENCY: Once/Operating Cycle
TECH SPEC: 4.2.B, Table 4.2.B.1.10, 4.7.2
Table 4.2.B.2.4, Table 4.2.B.2.5, 4.5.C.1.(a)

1 CHECK why this procedure is being performed:

☐ Schedule ☐ OVF ☐ Retest Due To Unsat Test

☐ Other Reason: ______________________

Approved By SMgt: ____________ ______ __/__/__ ______
Printed Name Time Date Initials

2 INITIAL one of the following Test Results:

A: All * steps are SATISFACTORY ______

B: One or More * steps are UNSATISFACTORY ☐
Refer to Section 9.0 for Tech Spec LCO's

Performed By: ____________ ______ __/__/__ ______
Printed Name Time Date Initials

RO/CO Informed of
Test Completion: ____________ ______ __/__/__ ______

SMgt Informed of
Test Results: ____________ ______ __/__/__ ______

★ UNSAT Notification: ______________________
SMgt Discretion: Plant Mgr or Others

Notified By: ____________ ______ __/__/__ ______

3 IF other portions of the test did NOT function properly,
OR other discrepancies were noted THEN COMPLETE the following:

DESCRIBE discrepancies/actions taken: A/R or ETT #: ______

____

____

____

4 Reviewed/Approved
Plant Staff: ____________ ______ __/__/__ ______
Printed Name Time Date Initials

ATTACHMENT 4

NRC RESOLUTION OF FACILITY COMMENTS ON WRITTEN EXAMINATION

SRO-12/RO-13	Facility comment accepted. The answer key was revised to accept both A and C as correct answers.
RO-39	Facility comment accepted. The answer key was revised to accept both C and D as correct answers.
SRO-39	Facility comment was not accepted. The stem of the question asked for the basis for not backseating both of the recirculation pump suction and discharge valves. None of answers provided a correct statement for both of the valves. Therefore, since there were no correct answers, the question was deleted from the examination.
SRO-40	Facility comment accepted. The answer key was revised to accept both A and B as correct answers.
SRO-43/RO-54	Facility comment accepted. The answer key was revised to accept both A and B as correct answers.
SRO-72/RO-76	Facility comment accepted. The answer key was revised to accept both C and D as correct answers.
SRO-74/RO-78	Facility comment accepted. The answer key was revised to accept both B and C as correct answers.
SRO-80/RO-83	Facility comment was accepted. In addition the answer to install sandbags was also determined to be an acceptable response per the T-103 basis step SC/L-5 reference to SE-9. Therefore, in accordance with NRC Headquarters Operator Licensing Branch guidance, since there were three correct answers, the question was deleted from the examination.
SRO-91	Facility comment not accepted. Procedure A-C-10 clearly indicated that, if an operator was temporarily unable to meet the medical requirements, NRC notification was not required.
SRO-99	Facility comment not accepted. Procedure OM-P-12.1 does not indicate that the Plant Manager must be notified by the Shift Manager, but does indicate that the Operations Services Manager must be notified if the Senior Manager of Operations is not available.

ATTACHMENT 5

SIMULATION FACILITY REPORT

Facility License: DPR-63

Facility Docket No: 50-277

Operating Test Administration: August 28-September 1, 1995

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, 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.

ITEM	DESCRIPTION
1.	During Scenario 95-2, the simulator operator was not able to remotely increase the indication of Secondary Area Temperature Point Number 22. An instructor, performing as a STA during the scenario, was able to compensate and provide the needed information to the applicants.
2.	During last run of Scenario 95-4, the simulator starting behaving erratically without apparent reason. The scenario was halted. After a computer reboot, the simulator response was as expected.