U.S. NUCLEAR REGULATORY COMMISSION
REGION I

OPERATOR LICENSING EXAMINATION REPORT

REPORT NO.:	92-21(OL)
DOCKET NOS.:	50-272 50-311
LICENSEE:	Public Service Electric and Gas Company 244 Chestnut Street Salem, N.J. 08079
FACILITY:	Salem Generating Station, Units 1 and 2
LOCATION:	Hancocks Bridge, New Jersey
DATES:	December 14 - 17, 1992
EXAMINERS:	T. Vehec, Pacific Northwest Laboratories (PNL) G. Benjamin, PNL

CHIEF EXAMINER: David M Silk 1/27/93

David M. Silk, Sr. Operations Engineer Date
PWR Section, Operations Branch

APPROVED BY: David H Jaffe for 1/27/93

Glenn W. Meyer, Chief Date
PWR Section, Operations Branch
Division of Reactor Safety

EXECUTIVE SUMMARY

During this examination, six instant senior reactor operator (SRO) candidates were evaluated. Written and operating examinations were administered during the week of December 14, 1992. All six candidates passed the examination and were issued licenses.

The training department staff was helpful and accommodating during the examination development, validation, and administration. The timely submittal of reference materials allowed the examination team to develop the written and operating examination despite time lost due to holidays. The availability of facility staff to review the written examination prior to administration and the availability of the simulator to validate scenarios and job performance measures (JPMs) was helpful to the examination process.

Analysis of the examination results yielded several noteworthy observations regarding candidate or training program weaknesses. From the review of written examination results, several weak knowledge areas were noted in component cooling water system lineup, reactor protection system alarms, and confirming a steam generator tube rupture in a faulted steam generator. The operating portion of the examination revealed weaknesses in locating information contained in control room materials for administrative questions and performing containment ventilation operations.

During the operating examination, the simulator generally functioned well. However, there were two model deficiencies noted. In one scenario, it was evident that there was minimal, if any, decay heat being modeled by the simulator. During a loss of all main and auxiliary feedwater, core exit thermocouples (CETs) were either remaining constant or decreasing slightly at times. The downward trend of the CETs could have caused an erroneous transition out of EOP-FRHS-1, Response to Loss of Secondary Heat Sink, without addressing the loss of feedwater condition. The other simulator deficiency was the unreliability of the availability of the P-250 computer used to provide information to the candidates during scenarios and JPMs.

Overall, the evaluation of the candidates' performance and the observation of facility staff and training facilities indicates that the training program is being implemented effectively.

DETAILS

1.0 INTRODUCTION

The NRC administered initial licensing examinations to six SRO instant candidates. The examinations were administered by the NRC chief examiner, with the assistance of two contractor examiners, in accordance with Revision 6 of NUREG-1021. All candidates passed the examination and were issued licenses. Overall, the candidates performed well in the operating portion of the examination, with several weaknesses noted after analyzing the examination results. The simulator performed well with only two noted deficiencies.

2.0 SUMMARY OF EXAMINATION RESULTS

2.1 Individual Examination Results

	SRO	RO
Written	6/0	N/A
Simulator	6/0	N/A
Walk-through	6/0	N/A
Overall	6/0	N/A

2.2 Generic Weaknesses

A weakness is an operating examination item performed unsatisfactorily or with difficulty. This item may not, by itself, result in an individual failure. A weakness may also be a written examination item when greater than 50% of the candidates incorrectly respond to the item. A weakness could indicate an area where the training program should be assessed to determine if increased emphasis or instruction is warranted.

Operating Examinations

The majority of the candidates demonstrated minor unfamiliarity with procedures examined in Section A of the operating examination by having difficulty locating the answers in the control room references. Candidates generally obtained the correct responses but were not quick to do so.

Two candidates did not properly implement OP-II-16.3.1, Containment Ventilation Operation, to reduce containment pressure. Both candidates skipped over step 5.6.1 and started at step 5.6.2. Step 5.6.1 directs operators to verify the status of containment ventilation radiation monitors prior to initiating a containment release.

Written Examinations

Question #	Area of Knowledge
29	Knowledge of component cooling water (CCW) system valve alignment to prevent overpressurization of a CCW heat exchanger that has had CCW flow secured to it.
43	Knowledge of the conditions causing the actuation of the Reactor Protection System "GENERAL WARNING ALARM."
76	Knowledge of the method to confirm the diagnosis of a steam generator tube rupture on a steam generator that has had a doubled end steam line break inside of containment.

3.0 RELATED ISSUES

During the operating examination, the simulator functioned well with the exception of a couple of problems. In one scenario, it was evident that there was minimal, if any, decay heat being modeled by the simulator because, during a loss of all main and auxiliary feedwater, the core exit thermocouples (CETs) were remaining constant or even decreasing at times. The downward trend of the CETs could have caused a transition out of EOP-FRHS-1, Response to Loss of Secondary Heat Sink, at step 14.2 back to EOP-TRIP-1, Reactor Trip or Safety Injection, without addressing the loss of feedwater condition. In this scenario, the operators were at step 14.2 of FRHS-1 when the CETs were trending downward. The candidate in the SRO position correctly chose to remain in FRHS-1 because no action had been taken to restore feed flow to the steam generators. The other simulator problem was the unreliability of the P-250 computer during the conduct of scenarios and job performance measures.

4.0 CONCLUSIONS

Based upon observation of candidate performance and reviews of the grading of the written and operating portions of the examination, the NRC concludes that the facility training program satisfactorily trains its licensed operator candidates to operate its nuclear facility. Several weaknesses were noted in candidate performance, but none were determined serious enough to deny issuance of licenses or to indicate inadequate training by the facility.

ATTACHMENT 1

SRO EXAMINATION AND ANSWER KEY

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

CANDIDATE'S NAME: Master

FACILITY: Salem 1 & 2

REACTOR TYPE: PWR-WEC4

DATE ADMINISTERED: 92/12/14

INSTRUCTIONS TO CANDIDATE:

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	CANDIDATE'S SCORE	%	
~~99.00~~ 95.00		%	TOTALS
	FINAL GRADE		

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

Candidate's Signature

Multiple Choice (Circle or X your choice)
If you change your answer, write your selection in the blank.

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

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(********** 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. Before you turn in your examination, consecutively number each answer sheet, including any additional pages inserted when writing your answers on the examination question page.

8. Use abbreviations only if they are commonly used in facility literature. Avoid using symbols such as < or > signs to avoid a simple transposition error resulting in an incorrect answer. Write it out.

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

10. Show all calculations, methods, or assumptions used to obtain an answer to any short answer questions.

11. Partial credit may be given except on multiple choice questions. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

12. Proportional grading will be applied. Any additional wrong information that is provided may count against you. For example, if a question is worth one point and asks for four responses, each of which is worth 0.25 points, and you give five responses, each of your responses will be worth 0.20 points. If one of your five responses is incorrect, 0.20 will be deducted and your total credit for that question will be 0.80 instead of 1.00 even though you got the four correct answers.

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

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

15. 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.

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

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

18. 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)

WHICH ONE (1) of the following is the MAXIMUM permissible circulating water thermal discharge Delta-T allowed by the New Jersey Pollutant Discharge Elimination System (NJPDES) for Salem?

a. 30.5 degrees F.

b. 27.5 degrees F.

c. 24.5 degrees F.

d. 21.5 degrees F.

QUESTION: 002 (1.00)

Given the following:

- An Abnormal Plant condition exists.
- SC.OP-DD.ZZ-AD46(Q) - "Troubleshooting Abnormal Plant Conditions" is in use.

WHICH ONE (1) of the following individuals is responsible for authorizing testing and/or troubleshooting identified by the procedure?

a. Maintenance Supervisor.

b. The System Engineer.

c. The Shift Supervisor.

d. The Operating Engineer.

QUESTION: 003 (1.00)

WHICH ONE (1) of the following MUST approve an EMERGENCY Technical Specification Interpretation (TSI) prior to implementation in the control room?

a. Station Operations Review Committee (SORC).

b. Licensing Department.

c. 2 On-Shift licensed Senior Reactor Operators.

d. Operations Manager AND an Operating Engineer.

QUESTION: 004 (1.00)

Given the following:

- Unit 1 is in Mode 6.
- RCP #1 Motor is uncoupled from the pump.
- Loop 1 is full.
- Maintenance is working on the #1 RCP pump.

WHICH ONE (1) of the following prevents leakage of reactor coolant up the RCP shaft?

a. Pump shaft mates with the top of the thermal barrier assembly.

b. Seal Leakoff collects any RCS leakage up the shaft and directs it back to the VCT.

c. Seal injection is maintained during this condition.

d. Nozzle dam installation prevents RCS water from entering the RCP shaft area.

QUESTION: 005 (1.00)

Given the following:

- A Site Blackout has occurred.
- ESF buses are de-energized and no emergency diesel generator is running.
- The STA reports the status of the CSF's as follows:

Heat Sink	-	RED
Shutdown Margin	-	GREEN
Containment	-	GREEN
Inventory	-	YELLOW
Core Cooling	-	RED
Thermal Shock	-	GREEN

WHICH ONE (1) of the following procedures should be used to mitigate this event following transition from EOP-TRIP-1, REACTOR TRIP OR SAFETY INJECTION?

a. EOP-FRCC-1 - Response to Inadequate Core Cooling

b. EOP-FRCI-2 - Response to Low RCS Inventory

c. EOP-FRHS-1 - Loss of Secondary Heat Sink

d. EOP-LOPA-1 - Loss of All AC Power

QUESTION: 006 (1.00)

WHICH ONE (1) of the following describes how to independently verify the position of a throttled valve during a system lineup in accordance with OD-7?

a. Operator #1 positions the valve and also performs the second verification by measuring valve stem height.

b. Operator #2 observes Operator #1 positioning the valve and does not operate the valve.

c. Operator #1 places the valve in the required position, Operator #2 independently places the valve in the required position.

d. Operator #1 places the valve in the required position and observes Operator #2 placing the valve in the required position.

QUESTION: 007 (1.00)

WHICH ONE (1) of the following requires actions to be taken per OD-6,"Circuit Breaker Reclosure Policy Following a Trip"?

a. Circuit breaker has opened due to a designed interlock.

b. Circuit breaker automatically opened as a consequence of a Reactor Trip/SI.

c. Circuit breaker was inadvertently opened during breaker alignment.

d. Circuit breaker trips open on overcurrent.

QUESTION: 008 (1.00)

WHICH ONE (1) of the following describes why the handwheel must be in place when performing alignments and verifications on Grinnel-Saunders diaphragm type valves?

a. For accurate measurement of throttle valve positioning.

b. The valve diaphragm and stem are free to ride up and down without the handwheel installed.

c. Independent verification is not possible without the handwheel in place.

d. Control Room position control or indication accuracy is not within acceptable tolerances.

QUESTION: 009 (1.00)

WHICH ONE (1) of the following individuals, by job title, can request the cancellation of a "Temporary Release"?

a. Job Supervisor.

b. Nuclear Control Operator.

c. Nuclear Shift Supervisor.

d. Senior Nuclear Shift Supervisor.

QUESTION: 010 (1.00)

WHICH ONE (1) of the following describes the process for positioning and tagging of valves under the cognizance of the Chemistry Department?

a. Valves are placed in the required position by Operations personnel and the tags are hung by Operations personnel.

b. Valves are placed in the required position by Chemistry personnel and the tags are hung by Chemistry personnel.

c. Valves are placed in the required position by Chemistry personnel and the tags are hung by Operations personnel.

d. Valves are placed in the required position by Operations personnel and the tags are hung by Chemistry personnel.

QUESTION: 011 (1.00)

Given the following:

- #21 Charging pump breaker has been tagged out for maintenance.
- Nuclear Shift Supervisor has approved a YELLOW tag request to allow a test of the pump breaker by the Relay Technician, who is the person named on the YELLOW tag.
- The Control Room operator prepared the tags and the Nuclear equipment operator hung the tags.

WHICH ONE (1) of the following named individuals, may grant permission to operate the charging pump breaker during normal operating conditions?

a. Relay Technician

b. Nuclear Control Room Operator

c. Nuclear Equipment Operator

d. Nuclear Shift Supervisor

QUESTION: 012 (1.00)

Given the following:

A loss of the "C" 125 VDC Bus has occurred.

WHICH ONE (1) of the following statements that describes the affect that this loss of power has on vital loads powered from this bus?

a. Load breakers may be locally closed with the "close" pushbutton at the breaker cubicle.

b. Load breakers may be operated from the control room after opening and closing springs are manually charged.

c. In the event of a SI, running SI loads will strip and not sequence back on due to a spring charging failure.

d. The "Spring Charging Failure" alarm will annunciate for all 4KV load breakers.

QUESTION: 013 (1.00)

Given the following:

- The oncoming Nuclear Control Operator informs the shift supervisor that he witnessed a drum of Hydrazine fall from the bed of an incoming truck into the marsh on the access road.

WHICH ONE (1) of the following MUST be notified by the Nuclear Shift Supervisor to perform IMMEDIATE actions for this loss of control of chemicals outside the protected area?

a. Hazardous Materials Manager

b. Nuclear Site Protection (Fire Department)

c. Radiation Protection Department

d. Nuclear Regulatory Commission

QUESTION: 014 (1.00)

Given the following:

- Unit 1 is in Mode 5.
- Unit 2 is Defueled.
- Shift manning is at minimum for the above conditions.

WHICH ONE (1) of the following represents the MINIMUM number of Equipment Operators who must be on shift for BOTH units for the above conditions? (See attached Tech Spec table.)

a. 2

b. 3

c. 4

d. 5

QUESTION: 015 (1.00)

Given the following:

- The reactor has tripped due to a steam line break and partial loss of feedwater has occurred.
- EOP-LOSC-1, "Loss of Secondary Coolant" is in use.
- The STA reports the critical safety function status as follows:

Shutdown Margin	- GREEN
Core Cooling	- YELLOW
Heat Sink	- PURPLE
Thermal Shock	- GREEN
Containment	- GREEN
Coolant Inventory	- YELLOW

WHICH ONE (1) of the following is the required MINIMUM frequency for monitoring the Critical Safety Functions?

a. Monitor continuously.

b. Monitor every 5 minutes.

c. Monitor every 10 minutes.

d. Monitor every 15 minutes.

QUESTION: 016 (1.00)

WHICH ONE (1) of the following represents the MAXIMUM dose limit that may be authorized in an emergency situation to save a life?

a. 25 Rem

b. 50 Rem

c. 75 Rem

d. 100 Rem

(REFERENCE PROVIDED)

Which one (1) of the following individuals is responsible for daily visual accountability of the security keys on the Unit 1 Primary key ring?

a. Primary Operator

b. Shift Clerk

c. Nuclear Control Operator

d. Nuclear Shift Supervisor

QUESTION: 018 (1.00)

Given the following:

- A Bank "C" Shutdown rod has fallen into the core.
- A dropped rod recovery is in progress per S2.OP-AB.ROD-0002(Q), "Dropped Rod".
- Rod recovery has commenced.
- An URGENT failure alarm is NOT present.

WHICH ONE (1) of the following explains why the URGENT failure alarm is NOT energized?

a. There is no master cycler irput for Shutdown Bank C.

b. Shutdown rods receive no input from the bank overlap unit.

c. Shutdown rods have no multiplexing thyristors.

d. There is only one group of rods in Shutdown Bank C.

QUESTION: 019 (1.00)

WHICH ONE (1) of the following is used as the reactor power input to the Rod Insertion Limit (RIL) computer?

a. Auctioneered High T-avg

b. Auctioneered High Power Range NI

c. Auctioneered High Delta T

d. Auctioneered High Tref

QUESTION: 020 (1.00)

Given the following:

- Salem Unit 1 is operating at 30% steady state reactor power.
- I&C technician receives permission to perform a calibration on PR N-41.
- The I&C technician pulls the fuses on N-42, then reinserts the fuses for N-42 and pulls the fuses for the correct channel, N-41, causing a reactor trip.

WHICH ONE (1) of the following describes the reason for the reactor trip?

a. PR neutron flux low setpoint trip

b. OP Delta T trip

c. PR neutron flux high setpoint trip

d. PR rate trip

QUESTION: 021 (1.00)

Given the following:

- N-41 is out of service.
- A shutdown is being performed IAW IOP-5, "Minimum Load to Hot Standby".
- Reactor power is at 12%.

WHICH ONE (1) of the following actions must be verified by the Nuclear Shift Supervisor to ensure that the low power trips are reactivated when power decreases below 10%?

a. Verify that the "Power Mismatch Bypass" switch is in the "N-41" position.

b. Verify that the "Comparator Channel Defeat" switch is in the "N-41" position.

c. Verify that the appropriate jumpers are installed in SSPS Train "A" and "B".

d. Verify that the "Upper" and "Lower Section Channel Defeat" switches are in the "N-41" position.

QUESTION: 022 (1.00)

WHICH ONE (1) of the following is NOT an input to the Subcooling Margin Monitor?

a. Core Exit Thermocouples

b. Wide Range T-hot

c. Containment Pressure

d. RCS Pressure

QUESTION: 023 (1.00)

WHICH ONE (1) of the following is removed PRIMARILY by the CATION bed demineralizer in the CVCS system?

a. Tritium

b. Boron

c. Cesium

d. Sodium Hydroxide

QUESTION: 024 (1.00)

WHICH ONE (1) of the following explains the requirement to have all letdown orifice valves OPEN during solid plant operation?

a. Provides additional relief protection.

b. Prevents loss of NPSH to the RHR pumps.

c. Provides maximum letdown purification flow.

d. Prevents lifting of RHR suction relief.

QUESTION: 025 (1.00)

WHICH ONE (1) of the following describes the operation of 2SJ-67, SI Mini-flow isolation valve, from panel RP4?

a. The "Power Lockout Switch" is overridden by a safety injection signal.

b. The "Power Lockout Switch" locks out "opening" power to the valve motor operator.

c. The "Recirc Override Switch" in the "NORMAL OPER" position can be used to control valve position.

d. The "Recirc Override Switch" in the "RECIRC OVERRIDE CLOSE" position directly operates contacts in the MOV control circuit.

QUESTION: 026 (1.00)

Given the following:

- Plant cooldown and depressurization is in progress.

At WHICH ONE (1) of the following RCS pressures would the accumulators START to inject if the operator failed to close 11-14SJ54, Accumulator Isolation valves when required?

a. 900 psig.

b. 800 psig.

c. 700 psig.

d. 600 psig.

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QUESTION: 027 (1.00)

Given the following:

- Unit 1 is at 100% power.
- PT-948A - Containment Pressure Channel I is failed high.
- ALL Bistables for PT-948A have been placed in the tripped condition.

WHICH ONE (1) of the following would occur if PT-948D Containment Pressure detector Channel IV failed high?

a. Main Steam Isolation and Containment Spray Actuation

b. Containment Spray Actuation and Phase A

c. Main Steam Isolation, Safety Injection and Phase A

d. Main Steam Isolation, Safety Injection, Containment Spray Actuation, and Phase A

Deleted

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QUESTION: 028 (1.00)

Given the following:

- Unit 1 is in Hot Shutdown.
- RCS Cooldown has commenced.

WHICH ONE (1) of the following is the MAXIMUM service water inlet temperature for performing an RCS cooldown at the normal rate?

a. 80 degrees F.

b. 82 degrees F.

c. 90 degrees F.

d. 92 degrees F.

Deleted

QUESTION: 029 (1.00)

Given the following:

- Unit 1 is in Mode 4.
- CCW Heat Exchangers #12A/B are in service.
- CCW flow has been secured to the #11 CCW heat exchanger.

WHICH ONE (1) of the following actions must be taken to the CCW valves to prevent overpressurization of the #11 CCW heat exchanger?

a. Close ONLY the heat exchanger inlet valve.

b. Close ONLY the heat exchanger outlet valve.

c. Open heat exchanger inlet AND outlet valves.

d. Close heat exchanger inlet AND outlet valves.

QUESTION: 030 (1.00)

WHICH ONE (1) of the following will cause an AUTOMATIC trip of the #11 Motor Driven Auxiliary Feedwater Pump following an AUTO start signal?

a. 1A Bus differential

b. Phase B overcurrent

c. Mode 1 SEC Operations

d. Auxiliary Feedwater Storage Tank low level

QUESTION: 031 (1.00)

WHICH ONE (1) of the following describes the setpoints and coincidence required for AMSAC to actuate?

a. 2/2 Turbine impulse pressures greater than 50%, 2/4 S/G's narrow range levels less than 10%.

b. 1/2 Turbine impulse pressures greater than 50%, 2/4 S/G's narrow range levels less than 10%.

c. 2/2 Turbine impulse pressures greater than 40%, 3/4 S/G's narrow range levels less than 5%.

d. 1/2 Turbine impulse pressures greater than 40%, 3/4 S/G's narrow range levels less than 5%.

QUESTION: 032 (1.00)

Given the following:

- Gas Decay Tank release is in progress per OP-II-12.3.1, "Gaseous Waste Disposal System Normal Operations"
- Gas Decay Tank radiation monitors 2R42A and B are in alarm.

WHICH ONE (1) of the following is the reason for de-energizing the Auxiliary Building elevator as a consequence of the radiation alarms?

a. To prevent an unmonitored release through the elevator shaft.

b. To prevent spreading of gaseous activity to other areas of the Auxiliary Building.

c. To prevent the ignition of combustible gas.

d. To prevent inadvertent personnel access to the area.

QUESTION: 033 (1.00)

WHICH ONE (1) of the following is the reason for the release rate limits specified in Technical Specification 3.11.2.1, "Gaseous Effluents Dose Rate"?

a. Ensures that the dose at any time inside the site boundary from radioactive effluents is within the dose limits of 10 CFR 20.

b. Prevents exceeding 10 CFR 20 limits at the site boundary during an accident condition.

c. Ensures that no member of the general public at or beyond the Site Boundary will receive a Beta/Gamma skin dose of 1500 mrem above background.

d. Prevents exceeding 1500 mrem/year thyroid dose rate above background to a child via the ingestion pathway.

QUESTION: 034 (1.00)

Given the following:

- Liquid radwaste release is in progress.
- NO high rad alarm is present.
- WL-51, Liquid Radwaste isolation valve closes.

WHICH ONE (1) of the following could have caused the closure of WL-51, Liquid Radwaste Discharge Valve?

a. Loss of control air to valve.

b. Loss of 115VAC supply to valve.

c. Control switch on Waste Disposal Panel CC-1 taken to "CLOSE".

d. Local control switch at Alternate S/D panel taken to "CLOSE".

QUESTION: 035 (1.00)

Given the following:

- The unit is operating at 35% power.
- Reactor Coolant Pump 12 trips.

WHICH ONE (1) of the following describes the IMMEDIATE unit response to the RCP trip? Assume NO operator action.

a. Steam generator water level will shrink but a reactor trip will NOT occur.

b. Steam Generator water level will swell but a reactor trip will NOT occur.

c. Steam generator water level will shrink and a reactor trip WILL occur.

d. Steam generator water level will swell and a reactor trip WILL occur.

QUESTION: 036 (1.00)

WHICH ONE (1) of the following Unit 1 Area Radiation Monitors has automatic actuations associated with a HIGH level alarm annunciating in the control room?

a. 1R2 - Containment Radiation Monitor.

b. 1R7 - In-Core Seal Table Radiation Monitor.

c. 1R32A - Fuel Handling Crane Radiation Monitor.

d. 1R5 - Fuel Handling Building Radiation Monitor.

QUESTION: 037 (1.00)

WHICH ONE (1) of the following is the basis for maintaining a minimum of 200 psid on the RCP seals during RCP startup and operation?

a. Ensures that adequate seal cooling flow from the RCS is available.

b. Prevents the #1 RCP seal from swapping from a face rubbing to a film riding seal.

c. Prevents the weight of the seal ring from limiting cooling flow through the seal gap.

d. Ensures adequate cooling flow to the pump radial bearing.

QUESTION: 038 (1.00)

WHICH ONE (1) of the following describes how the proximity of the Rod Position Indication Search Coil to the control rod shaft extension relates to the actual rod position?

a. Indicates that the rod is fully inserted.

b. Indicates that the rod is fully withdrawn.

c. Indicates rod position within +/-1^ steps of actual position.

d. Indicates rod position within +/-50 steps of actual position.

QUESTION: 039 (1.00)

Given the following:

- The Unit is in Mode 6 with the reactor head installed.
- OP-II-1.3.4, "Filling and Venting the RCS" is in use.
- A temporary tygon hose has been connected to Loop 23 for use as RCS level indication.
- The RCS is being drained to 97'6".

WHICH ONE (1) of the following explains why there is a concern about nitrogen pressure in the RCS?

a. Causes gas binding of the RHR pumps.

b. Causes a proportional increase in reactor vessel level indication on the tygon hose.

c. Causes invalid pressurizer level indication.

d. Causes difficulty establishing natural circulation flow if RHR flow were lost.

QUESTION: 040 (1.00)

Given the following:

- The blowdown phase of a large break LOCA is in progress.
- The pressurizer has gone empty after the reactor tripped.
- All RCPs are tripped.
- The NCO reports that all the RVLIS channels are displaying that there are no voids forming in the reactor vessel.

WHICH ONE (1) of the following statements describe the cause of the RVLIS indication?

a. RVLIS indication can be erratic during the blowdown phase due to the rapid insurge of subcooled ECCS flow.

b. RVLIS is not accurate during the blowdown phase of the LOCA due to high flows through the reactor core.

c. RVLIS indication was providing the correct level indication during the blowdown transient.

d. RVLIS indications were normal for this event due to the dynamic D/P cells being density compensated.

QUESTION: 041 (1.00)

WHICH ONE (1) of the following would result from a break on the common REFERENCE LEG for Channel I pressurizer pressure instrument PT-455 and Channel I pressurizer level instrument LT-459?

a. PT-455 will fail HIGH and LT-459 will fail HIGH.

b. PT-455 will fail LOW and LT-459 will fail HIGH.

c. PT-455 will fail HIGH and LT-459 will fail LOW.

d. PT-455 will fail LOW and LT-459 will fail LOW.

QUESTION: 042 (1.00)

Given the following:

- Pressurizer Overpressure Protection System (POPS) testing for 2PR1, PORV is ready to be performed IAW OP-II-2.3.4, "Pressurizer Overpressure Protection Operability Check and Arming of 2PR1 and 2PR2".

WHICH ONE (1) of the following statements reflects a requirement/interlock that must be met before 2PR1 can be successfully tested?

a. Pressurizer pressure must be less than 375 psig.

b. PT-403 must be OPERABLE.

c. 2PR6, PORV block valve, must be closed.

d. 2PR2, PORV, must be isolated.

QUESTION: 043 (1.00)

WHICH ONE (1) of the following will NOT cause the actuation of the Reactor Protection System "GENERAL WARNING ALARM"?

a. Loss of either 15 VDC power supply in RPS logic cabinet.

b. Reactor trip bypass breaker racked in.

c. Loose or removed circuit card in RPS logic cabinet.

d. Loss of 115 VAC to or in SSPS Output cabinet.

QUESTION: 044 (1.00)

WHICH ONE (1) of the following is a "DEMAND" signal that a Reactor Trip is required?

a. IRPI indicates all rods on bottom.

b. OHA F-40, "REACTOR TRIP" energized.

c. OHA F-36, "TURBINE TRIP AND P-9" energized and associated bistable coincidence on RP-4.

d. Reactor trip and associated bypass breakers OPEN.

QUESTION: 045 (1.00)

WHICH ONE (1) of the following will cause an AUTOMATIC trip of a Containment Iodine Removal System Fan?

a. Phase A

b. Safety Injection signal

c. Roughing Filter HIGH differential pressure

d. Charcoal Filter HIGH temperature

4/7/93

QUESTION: 046 (1.00)

WHICH ONE (1) of the following could produce enough hydrogen to exceed flammability limits inside the containment following a large break LOCA? (Assume all ECCS Systems function as designed.)

a. Radiolysis of water.

b. Zirconium-Water reaction.

c. Corrosion of aluminum structural components.

d. Corrosion of structural components painted with zinc based paint.

Deleted

QUESTION: 047 (1.00)

Given the following:

- A large break LOCA has occurred.
- Prior to the LOCA, containment temperature was 70 degrees F.
- Following the LOCA, containment pressure is 15.7 psia.
- The EOP's require that the Hydrogen recombiner be placed in service.

WHICH ONE (1) of the following values should be set on the Hydrogen Recombiner potentiometer based on the above conditions? (Use II-15.3.1, "Hydrogen Recombiners - Normal Operation" Attached)

a. 46.2 Kw.

b. 48.4 Kw.

c. 52.8 Kw.

d. 57.2 Kw.

QUESTION: 048 (1.00)

WHICH ONE (1) of the following is the basis for maintaining greater than 16% narrow range level in the steam generator with a tube leak?

a. Reduces the release by "filtering the release" through a layer of water prior to it escaping to the atmosphere.

b. Reduces the probability of a greater tube rupture by avoiding thermal stresses associated with tube uncovery.

c. Maintains the thermal stratification layer in the steam generator to assist in RCS/SG pressure equalization.

d. Assures that an adequate heat sink exists for the reactor core by maintaining a minimum mass in the steam generators.

DB 1/7/93

Deleted

QUESTION: 049 (1.00)

Given the following:

- Reactor power is at 25%.
- All plant controls are in automatic.
- Loop 21 MSIV inadvertently shuts.

WHICH ONE (1) of the following parameters would show an INITIAL INCREASE following this event? (Assume NO operator actions are taken).

a. Steam generator #21 level.

b. Steam Generator #22 pressure.

c. Loop #24 Cold leg temperature.

d. Steam Generator #23 level.

QUESTION: 050 (1.00)

Given the following:

- A large break LOCA has occurred on Unit 2.
- Radiation levels are increasing rapidly inside the Unit 2 Containment.

At WHICH ONE (1) of the following radiation levels will 2R44A insert adverse containment correction levels into the Subcooling Monitor?

a. 1.0 E+2 R/HR

b. 1.0 E+3 R/HR

c. 1.0 E+4 R/HR

d. 1.0 E+5 R/HR

QUESTION: 051 (1.00)

Given the following:

- Loss of offsite power has occurred.
- Diesel Generator #1A has had an actuation of its fire protection system due to high operating temperatures.
- Alarms in the control room actuate indicating that there is a fire in the #1B Diesel Generator room.
- The "First in with Lockout" signal is preventing CO2 deluge actuation in the #1B Diesel Room.

WHICH ONE (1) of the following describes what must occur to actuate the #1B Diesel Generator CO2 deluge system?

a. Wait for the timed discharge sequence circuitry on the Fire Protection system to cycle and pick up the new signal on #1B Diesel room.

b. Manually isolate the deluge system for the #1A Diesel.

c. Energize the shunt trip coil for the #1B Diesel actuation system at the fire protection panel using the toggle switch provided.

d. Bypass the electrical actuation circuitry by manually operating the pilot cabinet operating lever for the #1B Diesel actuation system.

QUESTION: 052 (1.00)

Given the following:

- Unit 2 is at 20% power during a startup.
- The Main Generator has just been synched to the grid.
- The steam dumps are closed.
- Alarms are received in the control room indicating that the 21A, 22B and 23A Circulating Water Pumps have tripped.

WHICH ONE (1) of the following could cause the simultaneous trip of the three circulating water pumps?

a. Opening the 500KV Keeney Line (5015).

b. Opening the 500KV Hope Creek Tie Line (5037).

c. Opening the 13KV supply from Hope Creek switchyard.

d. A Phase to Ground fault on the Salem 2E 4KV bus.

QUESTION: 053 (1.00)

Given the following:

- Transfer from Station Power Transformer (SPT), to the Auxiliary Power Transformer (APT), is in progress on Unit 1.
- Actions are being taken to transfer the first Group Bus.

WHICH ONE (1) of the following actions must be taken to ensure that the SPT and APT breakers do not trip and lock out when the first APT Infeed breaker to the Group Bus "CLOSE" pushbutton is depressed?

a. Verify running and incoming voltages matched.

b. Ensure synchroscope is moving slowly in the fast direction.

c. Ensure that the Unit Isolation Trip Multi-Trip relay is reset.

d. Ensure that the SPT Infeed breaker Anti-Pump relay is reset.

QUESTION: 054 (1.00)

Given the following:

- Diesel Generator 2A is in the process of being started on Unit 2 to parallel it to the grid.

WHICH ONE (1) of the following describes the operation of the diesel generator voltage control switch during this evolution?

a. Lowering the voltage control switch manually has no effect on the generator if selected to the AUTO (Isochronous) mode of operation.

b. Raising the voltage control switch to a higher value, will cause the generator to pick up a larger share of the reactive load after breaker closure.

c. Raising the voltage control switch will correct a synchroscope which is traveling slowly in the SLOW direction.

d. Lowering the voltage control switch raises the speed of the generator when operating in the MANUAL mode prior to paralleling with offsite source.

QUESTION: 055 (1.00)

WHICH ONE (1) of the following is an interlock that must be met in order to close a diesel generator output breaker from the Control Room?

a. Both Vital Bus infeed breakers must be OPEN.

b. Loading switch in MANUAL (Droop).

c. Voltage Regulator in AUTO.

d. Diesel speed at 800 rpm.

QUESTION: 056 (1.00)

WHICH ONE (1) of the following is normally used to fill the refueling cavity from the RWST during preparations for refueling?

a. Gravity feed

b. Spent Fuel Pit Cooling pump

c. RHR pump

d. Boric Acid transfer pumps

QUESTION: 057 (1.00)

Given the following:

- Main Turbine overspeed testing is in progress on Unit 2 IAW S2-OP-ST.TRB-0003(Q), "Turbine Mechanical Overspeed Test".
- Generator output breakers are open.
- You are the SRO at the Turbine Front Standard.

WHICH ONE (1) of the following procedural/automatic trips provides overspeed protection for this evolution?

a. Mechanical overspeed trip at 1850 rpm.

b. Manual trip at control room frequency counter indication of 1870 rpm.

c. Manual trip at turbine front standard speed indication of 1980 rpm.

d. Electrical overspeed trip at 2025 rpm.

QUESTION: 058 (1.00)

Given the following:

- Plant is in Mode 5.
- RHR pump 21 is in service.
- RCP 21 is in service.
- You are directed to start RHR pump 22.

WHICH ONE (1) of the following describes why 2CV18, Letdown Heat Exchanger Outlet valve, must be immediately adjusted following the start of RHR pump 22?

a. Prevent RHR pump cavitation.

b. Prevent cooldown of the RCS.

c. Prevent lifting the letdown relief.

d. Prevent RCS pressure from dropping below the minimum required for RCP operation.

QUESTION: 059 (1.00)

Given the following:

- Unit 1 has been shutdown for 3 days following a 6 month full power run.
- The RCS temperature is 120 degrees F.
- The RCS is at midloop.
- A total loss of RHR occurs.
- No core cooling is re-established.

WHICH ONE (1) of the following is the MINIMUM time required for the RCS to reach saturation?

a. 15 minutes

b. 45 minutes

c. 120 minutes

d. 180 minutes

QUESTION: 060 (1.00)

Given the following:

- Reactor power is at 85%
- A dropped rod recovery is in progress IAW S2.OP-AB.ROD-0002(Q), "Dropped Rod".

WHICH ONE (1) of the following describes the reason for ensuring that T-ave is within 1.5 degrees F of program prior to operating the Rod Bank Selector Switch through the AUTO position?

a. Prevents misalignment of the Bank Overlap Unit.

b. Prevents misalignment of the Pulse to Analog converter.

c. Prevents dropping rods with disconnected lift coils.

d. Prevents misalignment of rod insertion limit computer.

QUESTION: 061 (1.00)

Given the following:

- All systems are in automatic.
- Reactor power is at 75% and increasing slowly.
- Pressurizer pressure is steady.
- Pressurizer level is increasing.
- T-ave is increasing.
- Containment parameters are normal.
- No operator actions have been taken.

WHICH ONE (1) of the following could cause the above symptoms to occur?

a. Steam leak outside containment.

b. PORV leaking to PRT.

c. Loop T-ave drifting high.

d. Continuous rod withdrawal.

QUESTION: 062 (1.00)

Given the following:

- The reactor has tripped.
- 2 control rods have failed to insert on the trip.

WHICH ONE (1) of the following is the MINIMUM volume that must be added to the RCS from the RWST as a result of the stuck rods?

a. 1960 gallons

b. 3920 gallons

c. 5600 gallons

d. 11200 gallons

QUESTION: 063 (1.00)

Given the following:

- A large break LOCA has occurred.
- CET's are 1300 degrees F.
- Operators are performing actions of 2-EOP-FRCC-1, "Response to Inadequate Core Cooling".

WHICH ONE (1) of the following is the reason for maintaining intact S/G levels above the top of the U-tubes during performance of this procedure?

a. To ensure S/G inventory is adequate to avoid thermal shocking the tubesheet.

b. To prevent depressurization of the S/G steam space.

c. To ensure maximum heat transfer capability.

d. To maintain reflux boiling capability in the event that SI cannot be restored.

QUESTION: 064 (1.00)

WHICH ONE (1) of the following explains why it is preferable to leave the RCP's running during a small break LOCA if the RCS pressure RCP trip criteria on the conditional action summary are met but there is no SI flow?

a. To maintain two phase mixture level above the break longer.

b. To provide heat removal through the break and the S/G's.

c. To limit single phase inventory loss out of the break.

d. To decrease loop transit time for boron delivery to the core.

QUESTION: 065 (1.00)

Given the following:

- Small break LOCA mitigation is in progress IAW 2-EOP-LOCA-1, "Loss of Reactor Coolant".
- RHR pumps have been secured IAW Step 13 of that procedure.
- Containment pressure is 3 psig.

WHICH ONE (1) of the following would require the manual start of the RHR pumps?

a. RCS subcooling margin is 10 degrees F

b. RCS pressure is 200 psig

c. Charging pump at runout flow conditions

d. SI pump at runout flow conditions

QUESTION: 066 (1.00)

WHICH ONE (1) of the following indicates that the PRT rupture disk is ruptured following a pressurizer PORV failing OPEN?

a. PRT temperature decreasing

b. Relief line temperatures increasing

c. PRT level low

d. Pressurizer level decreasing

QUESTION: 067 (1.00)

WHICH ONE (1) of the following would require that maintenance install jumpers on 2A East Valves and Misc. 230V Control Center terminals for 21SJ49, RHR discharge to cold leg when S2.OP-AB.CR-0001(Q), "Control Room Evacuation," is in progress ?

a. Automatic initiation of Safety Injection

b. Manual Safety Injection alignment

c. Boration to cold shutdown conditions

d 2RP4 lockout switches not in valve operate position prior to evacuation.

QUESTION: 068 (1.00)

WHICH ONE (1) of the following is the reason that the RCP's are tripped as a subsequent action of S2.OP-AB.CR-0002(Q), "Control Room Evacuation due to Fire in the Control Room"?

a. Limits heat input by RCP's to the RCS during cooldown.

b. Prevent damage to the RCP's if a LOCA were to occur later in the event.

c. Prevent damage to the core if a partial loss of flow were to occur later in the event.

d. No local control of RCP auxiliaries is available.

QUESTION: 069 (1.00)

Given the following:

- Reactor power is 3 %
- Reactor/Plant Startup in progress.
- NI-35 has failed LOW.
- NI-36 is operable.
- I&C has been called to investigate.

WHICH ONE (1) of the following is the effect on the startup of this failure?

a. Verify/actuate reactor trip.

b. Power must be maintained less than 5%.

c. Power must be reduced to less than the POAH.

d. Rods must be driven in manually to shutdown the reactor.

QUESTION: 070 (1.00)

Given the following:

- 2-EOP-FRSM-1, "Response to Nuclear Power Generation" has been entered.
- Charging flow is greater than 100 gpm.
- SJ1 and SJ2, RWST to Charging pump valves are NOT open.
- Boric Acid pumps cannot be started.

WHICH ONE (1) of the following actions must be taken to commence rapid boration flow per 2-EOP-FRSM-1, "Response to Nuclear Power Generation"?

a. Open CV-175,(Rapid boration valve).

b. Start safety injection equipment manually.

c. Open SJ1 and SJ2 and establish boration from RWST through the BIT.

d. Initiate safety injection with actuation switches.

QUESTION: 071 (1.00)

WHICH ONE (1) of the following requires an immediate stop of the RCP's?

a. Component cooling thermal barrier return valve CC131 fails closed.

b. Component cooling lost to all RCPs.

c. Loss of seal injection flow to all RCPs.

d. RCP motor bearing indicated temperature of 180 degrees F for 2 minutes.

QUESTION: 072 (1.00)

WHICH ONE (1) of the following is the basis for reducing T-ave to less than 500 degrees F following a shutdown required by a Dose Equivalent I-131 level greater than Technical Specification limit?

a. Slows coolant/fuel reaction rate, immediately reducing the source term of the activity.

b. Prevents the release of activity following a steam generator tube rupture.

c. Minimizes the temperature related degradation of the CVCS demineralizers while the RCS clean-up is in progress.

d. Minimizes of the iodine spiking phenomena which occurs due to the large change in THERMAL POWER level caused by the unit shutdown.

QUESTION: 073 (1.00)

WHICH ONE (1) of the following describes why the main turbine is tripped during all ATWS conditions?

a. To initiate reactor shutdown from Doppler defect.

b. To mitigate the consequences of a loss of feed ATWS.

c. To prevent excessive cooldown of the RCS.

d. To prevent exceeding the DNBR/LPD limits on the core.

QUESTION: 074 (1.00)

WHICH ONE (1) of the following conditions is checked to verify that the reactor is subcritical during performance of 2-EOP-FRSM-1, "Response to Nuclear Power Generation"?

a. IRPI all indicate zero.

b. Rcd bottom lights energized.

c. Negative startup rate on Intermediate Range Indicators.

d. Power range channels indicate power less than 10%.

QUESTION: 075 (1.00)

Given the following:

- S/G level is decreasing
- Loop Delta-T is constant.
- Containment pressure is increasing.
- Steam Generator pressure is decreasing

WHICH ONE (1) of the following transients is indicated by the above conditions?

a. Steam Break inside containment

b. Feed Break inside containment

c Steam Generator Tube Rupture

d. LOCA inside containment

QUESTION: 076 (1.00)

Given the following:

- A double ended main steam line break inside containment has occurred.
- Operators are responding to the event per 2-EOP-LOSC-1, "Loss of Secondary Coolant."

WHICH ONE (1) of the following will be available to the operator to confirm the diagnosis of a SGTR in the FAULTED S/G?

a. S/G sample results.

b. R-19 Blowdown radiation monitors.

c. R-15 Condenser Air Ejector radiation monitor.

d. Uncontrolled S/G level increase.

QUESTION: 077 (1.00)

Given the following:

- Vacuum in the main turbine condenser is decreasing.
- No cause has yet been identified.
- A power reduction has commenced IAW directions in S2.OP-AB.COND-0001(Q), "Loss of Condenser Vacuum".

WHICH ONE (1) of the following is the reason that S2.OP-AB.COND-0001(Q), "Loss of Condenser Vacuum" limits the load decrease to less than 5%/minute?

a. Minimize turbine blade "flutter."

b. Prevent turbine rupture disc failure.

c. Maintain adequate vacuum for operation of condenser steam dumps.

d. Prevent operation of the main condenser steam dumps.

QUESTION: 078 (1.00)

Given the following:

- Loss of All AC power has occurred.
- 2-EOP-LOPA, "Loss of All AC Power and Recovery" is in use.

WHICH ONE (1) of the following is the reason for isolating the VCT?

a. CV-40 and 41, Charging Discharge valves, fail CLOSED on loss of power.

b. CV-40 and 41, Charging Discharge valves, fail OPEN on loss of power.

c. Prevent nitrogen and other non condensibles from interrupting natural circulation.

d. Auto makeup and RWST swapover are not available due to loss of power.

QUESTION: 079 (1.00)

Given the following:

- Loss of All AC power has occurred.
- 2-EOP-LOPA, "Loss of All AC Power and Recovery" is in use.

WHICH ONE (1) of the following parameter trends verify that natural circulation has been established?

a. RCS subcooling greater than 5 degrees.

b. RCS Narrow Range cold leg temperatures at saturation temperature for the pressurizer.

c. RCS Wide Range Hot leg temperatures stable or decreasing.

d. Steam generator pressure slowly increasing.

QUESTION: 080 (1.00)

WHICH ONE of the following sets of 115V Vital Instrument Buses contains the TWO buses which would require an immediate trip of the reactor if EITHER were de-energized?

a. 2A and 2B.

b. 2A and 2D.

c. 2B and 2C.

d. 2C and 2D

QUESTION: 081 (1.00)

Given the following:

- Loss of 2A 115V Vital Instrument Bus has occurred.
- All immediate action steps have been completed.
- Pressurizer heaters are being operated in the "LOCAL-MANUAL" mode for RCS pressure control.

WHICH ONE (1) of the following automatic pressurizer features is overridden by "LOCAL-MANUAL" operation of the pressurizer heaters?

a. Safety Injection heater cutoff.

b. Pressurizer low level heater cutoff.

c. Control group heaters "ON/OFF" setpoint.

d. Pressurizer program/actual level deviation heater energization.

QUESTION: 082 (1.00)

Given the following:

- Unit 1 is in Mode 6 with all circulators tagged out due to main condenser maintenance.
- Unit 2 is operating at 100% power.
- A release of radioactive liquid waste is necessary from the #21 CVCS Monitor Tank.

WHICH ONE (1) of the following actions must occur in order to release the #21 CVCS monitor tank to Unit 1.

a. Component Cooling water flow through the Unit 2 CCW HX selected must be greater than the required flow for the maximum release rate on the permit.

b. Service water flow through the Unit 2 CCW HX selected must be greater than the required flow for the maximum release rate on the permit.

c. Total service water flow through both Unit 1 CCW HXs must be greater than the required flow for the maximum release rate on the permit.

d. Discharge must go directly to the Unit 2 Circulators, then through the sluice cross-connect to Unit 1 discharge.

QUESTION: 083 (1.00)

Given the following:

- 2-EOP-FRCC-1, "Response to Inadequate Core Cooling" is in use.
- RWST Low Level alarm has just energized.
- Red Paths exist on Core Cooling and Heat Sink.

WHICH ONE (1) of the following actions should be taken by the operating crew?

a. Transition to 2-EOP-LOCA-3, "Transfer to Cold Leg Recirculation" immediately.

b. Transition to 2-EOP-LOCA-3, "Transfer to Cold Leg Recirculation" when T-hot and CET's show a definite decrease.

c. Transition immediately to 2-EOP-FRHS-1, "Response to Loss of Secondary Heat Sink".

d. Remain in FRCC-1 until Core Cooling Purple Path established, then transition to 2-EOP-FRHS-1, "Response to Loss of Secondary Heat Sink".

QUESTION: 084 (1.00)

Given the following:

- A small break LOCA has occurred.
- Operators are responding IAW 2-EOP-LOCA-2, "Post LOCA Cooldown and Depressurization".

WHICH ONE (1) of the following describes an appropriate action which would prevent an Inadequate Core Cooling Condition?

a. Maintain PZR level greater than 20% and Pressure greater than 1000 psig to ensure subcooling can be maintained while steaming the S/G's.

b. Reduce ECCS flow and lower RCS pressure to increase heat removal through the break while maintaining a constant S/G pressure.

c. Increase heat removal via the S/G's to increase cooldown and depressurization of the RCS allowing increased ECCS flow.

d. Stabilize S/G pressure and level, increase ECCS injection and RCS pressure to increase heat removal through the break.

QUESTION: 085 (1.00)

Given the following:

- A loss of RHR cooling has occurred.
- Operator actions are being taken IAW S2.OP-AB.RHR-0001(Q), Loss of RHR.
- The RHR pumps can not be restored prior to the RCS reaching the projected boiling point.
- The following cooling methods are available:

  1. Cooling the RCS with S/G's.
  2. Cooling the RCS with Spent Fuel Pool.
  3. Hot leg injection.
  4. Cold leg injection.

WHICH ONE (1) of the following reflects the preferred order of implementation of the four (4) available cooling methods?

a. 1, 2, 3, 4

b. 2, 3, 4, 1

c. 3, 4, 2, 1

d. 4, 1, 2, 3

QUESTION: 086 (1.00)

Given the following:

- RHR is in service at reduced inventory conditions.
- S2.OP-AB.RHR-0001, "Loss of RHR" has been implemented due to pump cavitation on the running pump.

WHICH ONE (1) of the following could cause an observed increase in RCS level during performance of this procedure?

a. Venting the RHR system.

b. Opening SJ69, RHR suction from RWST.

c. Any opening in the RCS boundary.

d. High RHR flow rate when restarting pump.

QUESTION: 087 (1.00)

Given the following:

- Unit 1 is at 100% power.
- PT-474, Pressurizer pressure channel IV is failed LOW.
- All actions of OP.IV-10.3.1, "Removing, Returning to Service and Loss of a Protective System Channel" are complete.

WHICH ONE (1) of the following describes the plant response to a failure of PT-455, Pressurizer pressure channel I, LOW?

a. Reactor trip.

b. Reactor trip AND safety injection.

c. All heaters energize, pressure increases to PORV setpoint, 1PR2 opens and 1PR1 does not, pressure cycles around PR2 setpoint.

d. All heaters energize, pressure increases to PORV setpoint, 1PR1 opens, 1PR2 does not, pressure cycles around PR1 setpoint.

QUESTION: 088 (1.00)

WHICH ONE (1) of the following describes the reason that selection of the same Auxiliary Building Ventilation exhaust unit on more than one control bezel is not advisable?

a. Could prevent the ventilation system from maintaining negative pressure on the Auxiliary Building.

b. Could cause isolation of one exhaust header.

c. Could prevent alignment of Post Accident ventilation for the Auxiliary Building.

d. Could override the thermostatic controls for the coolers.

QUESTION: 089 (1.00)

WHICH ONE (1) of the following determines the temperature at which the RCS cooldown is TERMINATED while responding to a steam generator tube rupture per 2-EOP-SGTR-1, "Steam Generator Tube Rupture"?

a. A temperature limit of 450 degrees F.

b. Temperature at which RHR can be placed in service.

c. Ruptured steam generator pressure.

d. 50 degrees F subcooling in the RCS available.

QUESTION: 090 (1.00)

Given the following:

- Reactor trip and SI have occurred.
- Operators are using 2-EOP-SGTR-1, "Steam Generator Tube Rupture" to mitigate the event.

WHICH ONE (1) of the following describes the applicable RCP trip criteria when responding to the SGTR.

a. RCPs should be tripped anytime during 2EOP-SGTR-1 when the RCP trip criteria are met.

b. RCPs should be tripped during 2-EOP-SGTR-1 ONLY if the RCP trip criteria are met at Step 1 when the operator is specifically required to check the criteria.

c. RCPs should be tripped during 2-EOP-SGTR-1 ONLY if the RCP trip criteria are met BEFORE beginning the cooldown and depressurization.

d. RCPs should be tripped during 2-EOP-SGTR-1 ONLY if the RCP trip criteria are met AFTER beginning the cooldown and depressurization.

QUESTION: 091 (1.00)

WHICH ONE (1) of the following Critical Safety Function conditions has the highest priority?

a. ECCS is not in service, NO RCP's are running, and RVLIS full range is 96%

b. RCS subcooling is 35 degrees F with one RCP running and RVLIS is 40% dynamic range

c. All S/G NR levels are less than 8% and total feed flow to the S/G's is 2E04 lb/hr.

d. Containment pressure is 25 psig

QUESTION: 092 (1.00)

WHICH ONE (1) of the following occurs on a loss of Control Air?

a. CV18, Letdown Heat Exchanger outlet valve fails OPEN.

b. #23 Charging pump speed controller fails AS IS.

c. Feed regulating valves fail OPEN.

d. RCP seal return valves fail AS IS.

QUESTION: 093 (1.00)

Given the following:

- Unit 1 is at 100% power.
- S2.OP-AB.CA-0001(Q), "Loss of Control Air" is in use by the operators.

WHICH ONE (1) of the following requires a reactor trip during performance of this procedure?

a. Station Air header pressure of 90 psig.

b. Control Air header pressure of 75 psig.

c. Redundant Air panel fails to transfer to 2A header.

d. 2 or more Rod Drive Vent Fan low flow alarms annunciate.

QUESTION: 094 (1.00)

WHICH ONE (1) of the following is NOT monitored by the Critical Safety Function Status Trees to determine loss of containment integrity?

a. Containment hydrogen concentration.

b. Containment temperature.

c. Containment sump level.

d. Containment pressure.

QUESTION: 095 (1.00)

WHICH ONE (1) of the following is an immediate action associated with the trip of a Steam Generator Feed Pump?

a. Bypass Feedwater heaters.

b. Bypass condensate polishers.

c. Start AFW system and take manual control of feed.

d. Start additional condensate pump.

ESTION: 096 (1.00)

Given the following:

A spent fuel assembly is being raised from its slot in the storage pool for return to the reactor.
Gas bubbles are now coming to the surface of the pool.
- Level in the spent fuel pool is decreasing.
- Radiation in the Spent Fuel area is increasing steadily.

WHICH ONE (1) of the following reflects the required response of personnel in the Fuel handling building?

a. Immediately evacuate all personnel.

b. Evacuate non-essential personnel when radiation levels exceed 250 mr/hr.

c. Evacuate all personnel when radiation levels exceed 1 R/hr

d. Complete fuel transfer to reactor cavity prior to evacuation of all personnel.

QUESTION: 097 (1.00)

Given the following:

- Reactor is at 100% steady state power BOL.
- All control systems are in automatic.

WHICH ONE (1) of the following will occur as a result of the controlling pressurizer level instrument failing low? (Assume no operator actions are taken.)

a. The reactor will trip on high pressurizer level.

b. The reactor will trip on high pressurizer pressure.

c. The level in the pressurizer will stabilize at the letdown isolation setpoint.

d. The reactor will trip on low pressurizer pressure.

QUESTION: 098 (1.00)

WHICH ONE (1) of the following actions of 2-EOP-LOPA-1, "Loss of All AC Power" is performed in order to limit the potential for a radioactive release inside the Auxiliary Building?

a. Isolation of the Seal Return line.

b. Isolation of the RCP seal injection lines.

c. Isolation of the thermal barrier CCW return line.

d. Isolation of the PORV stop valves

QUESTION: 099 (1.00)

Given the following:

- A LOCA outside of containment has occurred on Unit 1.
- Operators are using 1-EOP-LOCA-5, "Loss of Emergency Recirculation", due to the inability to isolate the leak.
- RWST level is less than the low level alarm setpoint.

WHICH ONE (1) of the following is the reason for depressurizing the S/G's to less than 675 psig in 1-EOP-LOCA-5, "Loss of Emergency Recirculation"?

a. To allow the S/G's to be fed from the condensate pumps.

b. To ensure adequate subcooling for restart of the RCP's.

c. To set up conditions for controlled injection of the accumulators.

d. To ensure that the RHR pumps are injecting at maximum flow for as long as possible while attempting to makeup to the RWST.

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

ANSWER: 001 (1.00)

*ANSWER
b. [+1.0]

REFERENCE:

*REFERENCE

1. SALEM: AD-17, Requirements of the New Jersey Pollutant Discharge Elimination System (NJPDES) Permit. Directives Pg. 1.
2. KA 194001K108 (3.5/3.4)
   194001K108 ..(KA's)

ANSWER: 002 (1.00)

d. [+1.0]

REFERENCE:

1. SALEM: SC.OP-DD.ZZ-AD46(Q) - "Troubleshooting Abnormal Plant Conditions", Section 3.1.
2. KA 19400A103 (2.5/3.4)
   194001A103 ..(KA's)

ANSWER: 003 (1.00)

d. [+1.0]

REFERENCE:

1. SALEM: AD-45, Technical Specification Interpretation Program Sec 3.1.13
2. KA 194001A102 (4.1/3.9)
   194001A102 ..(KA's)

ANSWER: 004 (1.00)

a. [+1.0]

REFERENCE:

1. SALEM: LP No. 300S-000.00S-RCPUMP-04, ELO 2, Pg. 17
2. KA 003000K407 (3.2/3.4)
   003000K407 ..(KA's)

ANSWER: 005 (1.00)

d. [+1.0]

REFERENCE:

1. SALEM: AD-44, Emergency/Abnormal Procedures Program, Pg. 24 of 45 Step 6.3.3.D
2. SALEM L/P - 300S-EOPINT-02, Introduction to the Use of the EOP's, Obj. 10, Pg 13.
3 KA 194001A102 (4.1/3.9)
194001A102 ..(KA's)

ANSWER: 006 (1.00)

b. [+1.0]

REFERENCE:

1. SALEM: OD-7, System Alignment, Pg 2 Sec 2.3.1
2. KA 194001K101 (3.6/3.7)
194001K101 ..(KA's)

ANSWER: 007 (1.00)

d. [+1.0]

REFERENCE:

1. SALEM: OD-6, "Circuit Breaker Reclosure Policy Following a Trip", Pg 1.
2. KA 194001K107 (3.6/3.7)
194001K107 ..(KA's)

ANSWER: 008 (1.00)

b. [+1.0]

REFERENCE:

1. SALEM: OD-7, System Alignments, Pg 5, Sec. 3.1.5
2. L/P 300S-000,00S-AR1A00-08, Administrative Requirements, Obj 4, Pg 26-27.
3 KA 194001K101 (3.6/3.7)
194001K101 ..(KA's)

ANSWER: 009 (1.00)

a. [+1.0]

REFERENCE:

1. SALEM: OD-8, TRIS Tagging Operations, Sec. 3.0, Pg. 9.
2. KA 194001K102 (3.7/4.1)
   194001K102 ..(KA's)

ANSWER: 010 (1.00)

c. [+1.0]

REFERENCE:

1. SALEM: NC.NA-AP.ZZ-0015, Safety Tagging Program, Sec.5.1.26, pg 13
2 KA 194001K102 (3.7/4.1)
   194001K102 ..(KA's)

ANSWER: 011 (1.00)

a. [+1.0]

REFERENCE:

1 SALEM: NC.NA-AP.ZZ-0015, Safety Tagging Program, Section 6.11.1 Pg. 52.
2. KA 194001K107 (3.7/4.1)
   194001K107 ..(KA's)

ANSWER: 012 (1.00)

a. [1.00]

REFERENCE:

1. SD-42 "Electric Distribution"
2. KA 000058A203 (3.5/3.9)
   000058A203 ..(KA's)

ANSWER: 013 (1.00)

b. [+1.0]

REFERENCE:

1. SALEM: NC.NA-AP.ZZ-0038, Chemical Control Program, Sec. 5.5.1-5.5.2, Pg. 14.
2. KA 194001K111 (3.4/3.5)
   194001K111 ..(KA's)

ANSWER: 014 (1.00)

b. [+1.0]

REFERENCE:

1. SALEM: Technical Specifications, Section 6.2.2e, Table 6.2
2. KA 194001K116 (3.5/4.2)
   194001K116 ..(KA's)

ANSWER: 015 (1.00)

a. [+1.0]

REFERENCE:

1. SALEM: AD-44 - Emergency/Abnormal Procedures Program, Sec 6.3.2, Pg 23.
2. KA 194001A111 (2.8/4.1)
   194001A111 ..(KA's)

ANSWER: 016 (1.00)

c. [1.0]

REFERENCE:

1. SALEM: NC.NA-AP.ZZ-0024, Radiation Protection Program, Pg 8.
2. 10 CFR 50.20, Emergency Dose Rate Allowances.
3. SALEM: L/P # 0215-000.00B-000002-01, EP training, SNSS,NSS,NSTA,NCO Duties/Responsibilities, Obj. 44, Pg.45.
4. KA 194001K103 (2.8/3.4)
   194001K103 ..(KA's)

ANSWER: 017 (1.00)

c. [+1.0]

REFERENCE:

1. SALEM: AD-37 - Key Control, Sec. 3.4.2, Pg 3.
2 KA 194001A110 (2.9/3.9)
194001A110 ..(KA's)

ANSWER: 018 (1.00)

d. [+1.0]

REFERENCE:

1. SALEM: LP No. 300S-000.00S-ROC000-01, ELO 2.i, Pg. 31.
2. SALEM: Proc. S2.OP-AB.ROD-0002(Q), NOTE Pg. 6.
3 KA 001050A201 (3.7/3.9)
001050A201 ..(KA's)

ANSWER: 019 (1.00)

c. [+1.0]

REFERENCE:

1. SALEM: LP No. 300-000.00S-ROC000-01, ELO 3d, Pg. 48.
2. KA 014000A103 (3.6/3.8)
014000A103 ..(KA's)

ANSWER: 020 (1.00)

d. [+1.0]

REFERENCE:

1. SALEM: 300-000.00S-EXCNIS-07, ELO 3.i, Pg 57.
2. KA 015000K405 (4.3/4.5)
015000K405 ..(KA's)

ANSWER: 021 (1.00)

c. [+1.0]

REFERENCE:

1. SALEM: Proc. No IV-10.3.1, Removing, Returning to Service and Loss of a Protective System Channel, Table 1
2. SALEM: LP No. 300S-000.00S-IOP500-01, Minimum Load to Hot Standby, ELO 4, Pg 6.
3. KA 015000K407 (3.7/3.8)
   015000K407 ..(KA's)

ANSWER: 022 (1.00)

b. [+1.0]

REFERENCE:

1. Salem: L/P 300S-000.00S-INCORE-04, ELO 2-3, Pg 44.
2. KA 017020A401 (3.8/4.1)
   017020A401 ..(KA's)

ANSWER: 023 (1.00)

c. [+1.0]

REFERENCE:

1. SALEM: 300S-000-00S-CVCS00-02, ELO 2a, Pg 29.
2. KA 004020K406 (2.3/2.7)
   004000K406 ..(KA's)

ANSWER: 024 (1.00)

a. [+1.0]

REFERENCE:

1. Salem Unit-2 Procedure S2.OP-SO.CVC-0001(Q), Precaution 3.8, Pg 3.
2. KA 004010K403 (3.1/3.6)
   004010K403 ..(KA's)

ANSWER: 025 (1.00)

d. [+1.0]

REFERENCE:

1. Salem L/P - 300S-000.00S-ECS00-07, ELO 2d, Pg. 32-33.
2. Salem P&ID 205234
3. KA 013000K201 (3.6/3.8)
   013000K201 ..(KA's)

ANSWER: 026 (1.00)

d. [+1.0]

REFERENCE:

1. Salem L/P - 300S-000.00S-ECS00-07, ELO 2f Pg. 79.
2. Salem Technical Specifications 3.5.1, Accumulators.
3. KA 006000A301 (4.0/3.9)
   006000A301 ..(KA's)

ANSWER: 027 (1.00)

Deleted DS 1/7/97

a. [+1.0]

REFERENCE:

1. Salem L/P - 300S-000.00S-CSPRAY-06, ELO 4, Pg. 15-16.
2. Salem Logic Diagram 221057
3. KA 026000A301 (4.3/4.5)
   026000K201 ..(KA's)

ANSWER: 028 (1.00)

DS 1/7/97

c. [+1.0]

Deleted

REFERENCE:

1. Salem L/P - 300S-000.00S-CCW000-01, ELO 11, Pg. 14.
2. KA 008000A102 (2.9/3.1)
   008000A102 ..(KA's)

ANSWER: 029 (1.00)

b. [+1.0]

REFERENCE:

1. Salem L/P - 300S-000.00S-CCW000-01, ELO 17a, Pg. 46.
2. Salem Proc. II-7.3.2, "Component Cooling Water System - Normal Operations"Precaution 3.3.
3. KA 008000G010 (3.1/3.2)
   008000G010 ..(KA's)

ANSWER: 030 (1.00)

a. [+1.0]

REFERENCE:

1. Salem L/P - 300S-000.00S-AFEEDW-10, ELO 7c, Pg. 31.
2. KA 061000K406 (4.0/4.2)
   061000K406 ..(KA's)

ANSWER: 031 (1.00)

c. [1.0]

REFERENCE:

1. Salem L/P - 300S-000.00S-AFEEDW-10, ELO 7a/b, Pg. 32.
2. KA 061000K402 (4.5/4.6)
   061000K402 ..(KA's)

ANSWER: 032 (1.00)

c. [+1.0]

REFERENCE:

1. Salem L/P - 300S-000.00S-WASGAS-02, ELO 12a, Pg. 21.
2. Salem Proc. OP-II-12.3.1, "Gaseous Waste Disposal System Normal Operations" Precaution 3.4.
3. KA 071000A409 (3.3/3.5)
   071000A409 ..(KA's)

ANSWER: 033 (1.00)

d. [+1.0] or C.

REFERENCE: & 1/7/93

1. Salem L/P - 300S-000.00S-WASGAS-02, ELO 10, Pg. 20.
2. Salem Tech Specs, 3.11.2.1, Bases, B 3/4.11-3
3. KA 071000G006 (1.9/2.9)
   071000G006 ..(KA's)

ANSWER: 034 (1.00)

a. [+1.0]

REFERENCE:

1. Salem L/P - 300S-000.00S-WASLIQ-02, ELO 6, Pg. 19.
2. Salem Proc. II-11.3.2b, Pg 6, 5.2.11.b.
3. KA 000059A101 (3.5/3.5)
   000059K301 ..(KA's)

ANSWER: 035 (1.00)

a. [+1.0]

REFERENCE:

1. Salem L/P, 300S-000.00S-EXCNIS-07, Obj. 3b, Pg 55.
2. Salem L/P, 300S-000.00S-SGWLCS-03, Obj. 1, Pg 13.
3. KA 000015K104 (2.9/3.1)
   000015K104 ..(KA's)

ANSWER: 036 (1.00)

d. [+1.0]

REFERENCE:

1. Salem L/P, 300S-000.00S-RMS000-00, Obj. 1d/e, Pg 70-73.
2. Salem Proc. IV-11.3.1, Table 2
3. KA 072000K402 (3.2/3.4)
   072000K402 ..(KA's)

ANSWER: 037 (1.00)

c. [+1.0]

REFERENCE:

1. Salem S/D, 302S-SD/1:19, Pg 15.
2. Salem L/P 300S-000.00S-RCPUMP-04, Obj. 13, Pg. 28.
3. KA 003000K407 (3.2/3.4)
   003000K407 ..(KA's)

ANSWER: 038 (1.00)

a. [+1.0]

REFERENCE:

1. Salem S/D, 302S-SD/1:25, Pg 2
2. Salem L/P 302/304-145.12-RPI-04, Obj. 1/2.1.3, Pg 8.
3. KA 014000A202 (3.1/3.6)
   014000A202 ..(KA's)

ANSWER: 039 (1.00)

b. [+1.0]

REFERENCE:

1. Salem Proc. OP-II-1.3.4 "Filling and Venting the RCS", Precaution 3.
2. Salem L/P 300S-000.00S-RCS000-05 Obj. 12, Pg 27.
3. KA 002000G010 (3.4/3.9)
   002000G010 ..(KA's)

ANSWER: 040 (1.00)

b. [+1.0]

REFERENCE:

1. Salem S/D Chapter 24, Reactor Vessel Level Monitoring System
2. Salem L/P 300S-000.00S-RVLIS0-05 Obj. 2
3. KA 002000K107 (3.5/3.7)
   002000K107 ..(KA's)

ANSWER: 041 (1.00)

b. [+1.0]

REFERENCE:

1. Salem P&ID 205201, D-2
2. Salem L/P 300S-000.00S-PZRP/L-00 Obj. 2 Pg 35.
3. KA 010000K601 (2.7/3.1)
   010000K601 ..(KA's)

ANSWER: 042 (1.00)

c. [+1.0]

REFERENCE:

1. Salem Tech Spec 3.10.3, Pg 3/4 4-31.
2. Salem L/P 300S-000.00S-PZRP/L-00 Obj. 12, Pg 37.
3. OP-II-2.3.4, "Pressurizer Overpressure Protection Operability Check and Arming of 2PR1 and 2PR2".
4. KA 010000K403 (3.8/4.1)
   010000K403 ..(KA's)

ANSWER: 043 (1.00)

d. [+1.0]

REFERENCE:

1. Salem S2.OP-SO.RCP-0002(Q), Reactor Trip or Bypass Breaker Operation, Precaution 3.1.
2. Salem L/P 300S-000.00S-RXPROT-08 Obj. 2o Pg 57-58.
3. KA 012000A202 (3.6/3.9)
   012000A202 ..(KA's)

ANSWER: 044 (1.00)

c. [+1.0]

REFERENCE:

1. Salem 2-EOP-TRIP-1, Entry conditions.
2. Salem L/P 300S-000.00S-RXPROT-08 Obj. 15 Pg 62.
3. KA 012000A306 (3.7/3.7)
   012000A306 ..(KA's)

ANSWER. 045 (1.00)

b. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-CONTMT-02 Obj. 12 Pg 60.
2. KA 027000A403 (3.3/3.2)
   027000A403 ..(KA's)

ANSWER: 046 (1.00) Deleted DS 1/7/93

a. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-CONTMT-02 Obj. 8 Pg 106.
2. KA 028000K503 (2.9/3.6)
   028000K503 ..(KA's)

ANSWER: 047 (1.00)

c. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-CONTMT-02 Obj. 15 Pg 111.
2. Salem Proc II-15.3.1, "Hydrogen Recombiners - Normal Operation" Steps 5.1.6-5.1.8, Fig 1.
3. KA 028000A201 (2.9/3.6)
   028000A201 ..(KA's)

ANSWER: 048 (1.00) Deleted DS 1/7/93

c. (+1.0)

REFERENCE:

1. S2.OP-AB.SG-0001, Steam Generator Tube Leak, AOP Basis p. 7.
2. 300S-000.00S-ABSG01-00 Steam Generator Tube Leak, Obj. 3, p. 7
3. KA 000037K307 (4.2/4.4)
   000037K307 ..(KA's)

ANSWER: 049 (1.00)

d. [+1.0]

REFERENCE:

1. Salem S/D Chapter 25 - Steam Generator Water Level Control, Pg. 7.
2. KA 035010K601 (3.2/3.6)
   035010K601 ..(KA's)

ANSWER: 050 (1.00)

d. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S- Obj. 2, Pg 13.
2. Salem S2.OP-AB.RAD-0001 (Q), Area Radiation Monitors, Att.3.
3. KA 000061A101 (3.6/3.6)
   000061A101 ..(KA's)

ANSWER: 051 (1.00)

d. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-FIRPRO-03, Obj. 7b, Pg 71-72.
2. KA 086000A204 (3.3/3.9)
   086000A204 ..(KA's)

ANSWER: 052 (1.00)

c. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-CWATER-03, Obj. 2.
2. S2.OP-SO.CW-0001, "Circulating Water System Operation
3. KA 062000A201 (3.4/3.9)
   062000A201 ..(KA's)

ANSWER: 053 (1.00)

c. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-4KV000-05, Obj. 12, Pg 56.
2. KA 062000K403 (2.8/3.1)
   062000K403 ..(KA's)

ANSWER: 054 (1.00)

b. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-DIESEL-05, Obj. 4, Pg 38.
2. S2.OP-SO.DG-0001(Q), Pg. 15
3. KA 064000A401 (4.0/4.3)
   064000A401 ..(KA's)

ANSWER: 055 (1.00)

a. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-DIESEL-05, Obj. 7 Pg 31.
2. S2.OP-SO.DG-0001(Q), Pg. 14
3. KA 064000K401 (3.8/4.1)
   064000K401 ..(KA's)

ANSWER: 056 (1.00)

c. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-RO-05, Refueling Operations, Obj 36, Pg 85
2. OP-II-8.3.3, Filling the Reactor Refueling Cavity, Pg 1
3. KA 034000A102 (2.9/3.7)
   034000A102 ..(KA's)

ANSWER: 057 (1.00)

a [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-MNTURB-00, Obj. 12b, Pg 54, (Old reference, contradicts new procedure setpoint).
2. S2-OP-OT.TRB-0001(Q), "Turbine Auto Trip Mechanism Operational Test"., Precaution 3.8, Pg 3
3. KA 045000G014 (2.8/2.9)
   045000G014 ..(KA's)

ANSWER: 058 (1.00)

d. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S.RHR000-06, Obj. 16, Pg.36.
2. S2.OP-SO.RHR-0001(Q) - Initiating RHR, Precaution 3.9.2, Pg 4.
3. KA 005000A102 (3.3/3.4)
   005000A102 ..(KA's)

ANSWER: 059 (1.00)

a. [+1.0]

REFERENCE:

1. S2.OP-AB.RHR-001 (Q), "Loss of RHR", Attachment 1, p.15
   Generic Letter 88-17
   Note: This question reveals if the candidate is sufficiently sensitive to the issue of loss of RHR, and the very short time frame available to respond to same. Industry events have occurred where RHR has been lost at reduced inventory, and one key issue is the operators were often not aware how little time was available until saturation was reached in the core. The question does not require detailed knowledge of the saturation vs time curve due to the very large time frame of the incorrect distractors....
2. KA 000025K101 (3.9/4.3)
   000025K101 ..(KA's)

ANSWER: 060 (1.00)

a. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-ABROD02-00 Obj 3 Pg 11
2. S2.OP-AB.ROD-0002(Q), Dropped Rod, Step 3.34, Caution.
3. KA 000003G007 (3.4/3.6)
   000003G007 ..(KA's)

ANSWER: 061 (1.00)

d. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-ABROD03-00 Obj 3 Pg 8.
2. S2.OP-AB.ROD-0003(Q), Continuous Rod Motion Bases Document, Pg 5-6.
3. KA 000001A205 (4.4/4.6)
   000001A205 ..(KA's)

ANSWER: 062 (1.00)

d. [+1.0]

RFFERENCE:

1. Salem L/P 300S-000.00S-TRIP02-03 Obj 2a Pg. 7.
2. S2.EOP-ES-0.1, TRIP2, Reactor Trip Response Bases Document, Pg 12.
3. KA 000005A203 (3.5/4.4)
   000005A203 ..(KA's)

ANSWER: 063 (1.00)

c. [+1.0]

REFERFNCE:

1. Salem L/P 300S-000.00S-FRCC00-02 Obj 2.2, Pg 11.
2. 2-EOP-FRCC-1, "Response to Inadequate Core Cooling", Sheet 4, Step 34.
3. KA 000074K308 (4.1/4.2)
   000074K308 ..(KA's)

ANSWER: 064 (1.00)

b. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-RCPTAA-01 Obj 1, Pg 11.
2. 2-EOP-LOCA-1, "Loss of Reactor Coolant", Sheet 1, CAS.
3. KA 000009K323 (4.2/4.3)
   000009K323 ..(KA's)

ANSWER: 065 (1.00)

b. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-LOCA02-02 Obj 2d, Pg 8.
2. 2-EOP-LOCA-1, ""Loss of Reactor Coolant",Step 13, Continuous Action Step.
3. KA 000009K321 (4.2/4.[illegible])
   000009K321 ..(KA's)

ANSWER: 066 (1.00)

a. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-SBLOCA-00 Obj 2, Pg 9.
2. 2-EOP-TRIP-1, "Reactor Trip or Safety Injection", Step 27-28
3. KA 000008A108 (3.8/3.8)
   000008A108 ..(KA's)

ANSWER: 067 (1.00)

d. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-AOPEVA-00 Obj 2, Pg 6
2. S2.OP-AB.CR-0001(Q), "Control Room Evacuation", Step 3.18.
3. KA 000068K318 (4.2/4.5)
   000068K318 ..(KA's)

ANSWER: 068 (1.00)

a. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-AOPFIR-00 Obj 2, Pg 6
2. S2.OP-AB.CR-0002(Q), "Control Room Evacuation Due To Fire in Control Room, Relay Room, or Ceiling of the 460/230V Switchgear Room ", Subsequent action I, Step 3.9.
3. S2.OP-AB.CR-0002(Q), "Control Room Evacuation Due To Fire in Control Room, Relay Room, or Ceiling of the 460/230V Switchgear Room Technical Basis Document", Pg 8.
4. KA 000068A112 (4.4/4.4)
   000068A112 ..(KA's)

ANSWER: 069 (1.00)

b. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-ABNIS1-00 Obj 3A, Pg. 8
2. Salem T.S. 3.3.1, Table 3.3-1, Item 5, Action 3
3. KA 015000K301 (3.9/4.2)
   015000K301 ..(KA's)

ANSWER: 070 (1.00)

c. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-FRSM01-01 Obj 6, Pg 9.
2. Salem 2-EOP-FRSM-1, Step 2.
3. KA 000029G010 (4.5/4.5)
   000029G010 ..(KA's)

ANSWER: 071 (1.00)

b. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-ABRCP1-00 Obj 1, Pg 12
2. S2.OP-AB.CC-0001(Q), "Component Cooling Abnormality", RCP trip criteria.
3 KA 000026G010 (3.6/3.5)
   000026G010 ..(KA's)

ANSWER: 072 (1.00)

b. (+1.0)

REFERENCE:

1. Salem Technical Specifications Bases 4.4.9.
2. KA 000076G004 (2.1/3.7)
   000076G004 ..(KA's)

ANSWER: 073 (1.00)

b. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-FRSM01-01 Obj 2, Pg 8.
2. 2-EOP-FRSM-1, Response to Nuclear Power Generation" Basis, Pg. 2.
3 KA 000029K312 (4.4/4.7)
   000029K312 ..(KA's)

ANSWER: 074 (1.00)

c. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-FRSM01-01 Obj 2, Pg 8.
2. 2-EOP-FRSM-1, Response to Nuclear Power Generation" Step 11.
3. KA 000029A201 (4.4./4.7)
   000029A201 ..(KA's)

ANSWER: 075 (1.00)

b. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-LOSCTA-02, Loss of Secondary Coolant Transients Description, Obj. 1.2, Pg 13.
2. KA 000040A201 (4.2/4.7)
   000040A201 ..(KA's)

ANSWER: 076 (1.00)

a. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-LOSC01-02, Loss of Secondary Coolant Obj. 5 Pg 10.
2. Salem 2-EOP-LOSC-1, "Loss of Secondary Coolant", Step 11.
3. KA 000040K304 (4.5/4.7)
   000040K304 ..(KA's)

ANSWER: 077 (1.00)

d. [+1.0]

REFERENCE:

1. Salem S2.OP-AB.COND-0001(Q), "Loss of Condenser Vacuum", Pg 1 Note.
2. Salem L/P - 300S-ABCOND-00, Loss of Condenser Vacuum, Obj. 3, Pg. 5.
3. KA 000051G007 (2.3/2.5)
   000051A202 ..(KA's)

ANSWER: 078 (1.00)

d. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-LOPA00-01, "Loss of ALL AC and Recovery", Obj. 3, Pg. 13.
2. Salem EOP-LOPA-1, "Loss of ALL AC and Recovery", Step 16 Bases.
3. KA 000055K302 (4.3/4.6)
   000055K302 ..(KA's)

ANSWER: 079 (1.00)

c. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-LOPA00-01, "Loss of ALL AC and Recovery", Obj. 5, Pg. 14.
2. Salem EOP-LOPA-1, "Loss of ALL AC and Recovery", Step 34.
3. KA 000055A202 (4.4/4.6)
   000055A202 ..(KA's)

ANSWER: 080 (1.00)

d. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-115VAC-05, "115 VAC Control Power System",Obj. 5, Pg. 24.
2. Salem S2.OP-AB.115-0001-4, "Loss of 2A-2D Vital Instrument Buses, Immediate Actions.
3. KA 000057G010 (3.6/3.7)
   000057G010 ..(KA's)

ANSWER: 081 (1.00)

b. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-PZRPRT-05, "Pressurizer and Pressurizer Relief Tank, Obj. 2f, Pg 14,24.
2. Salem S2.OP-AB.115-0001-4, "Loss of 2A-2D Vital Instrument Buses, Note prior to Step 3.3.
3. KA 000057A106 (3.5/3.5)
   000057A106 ..(KA's)

ANSWER: 082 (1.00)

b. [+1.0]

REFERENCE:

1. Salem S2.OP-SO.CW-0001(Q), "Circulating Water Pump Operation", precautions and limitations.
2. Salem II-11.3.2b, "Release of Radioactive Liquid to the Circulating Water System from 21 or 22 Monitor Tanks", Note, Pg 3.
3. KA 000059A204 (3.2/3.5)
   000059A204 ..(KA's)

ANSWER: 083 (1.00)

a. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-FRCC00-02, "MCD/EOP FRCC1,2,3, Obj. 6 Pg 8.
2. Salem 2-EOP-FRCC-1, "Response to Inadequate Core Cooling", Caution prior to Step 1.
3. KA 000074G010 (4.5/4.6)
   000074G010 ..(KA's)

ANSWER: 084 (1.00)

c. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-LOCA2C-05, Post LOCA Cooldown and Depressurization, Obj. 2d, Pg 8
2. Salem question bank #000284
3. KA 000011A113 (4.1/4.2)
   000011A113 ..(KA's)

ANSWER: 085 (1.00)

c. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-ABRHR0-00, Loss of RHR, Obj. 2, Pg 6.
2. Salem S2.OP-AB.RHR-0001, "Loss of RHR, Step 3.31 Attachment list.
3. KA 000025K301 (3.1/3.4)
   000025K301 ..(KA's)

ANSWER: 086 (1.00)

b. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-ABRHR0-00, Loss of RHR, Obj. 3a, Pg 6.
2. Salem S2.OP-AB.RHR-0001, "Loss of RHR, Step 3.14 Cautions.
3. KA 000025A102 (3.8/3.9)
   000025A102 ..(KA's)

ANSWER: 087 (1.00)

a. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-ICF000-03, I&C FAILURE ANALYSIS, Obj. 2e, Pg. 50.
2. Salem Logic Diagram 221055, Reactor Protection System Pressurized Trip Signals.
3. KA 000027A216 (3.6/3.9)
   000027A216 ..(KA's)

ANSWER: 088 (1.00)

b. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-AOPVNT-00, Unit 1&2 Ventilation Malfunctions, Obj. 2, Pg. 6.
2. Salem II-17.3.1 "Auxiliary Building Ventilation Operation, Precautions.
3. KA 000060K202 (2.7/3.1)
   000060K202 ..(KA's)

ANSWER: 089 (1.00)

c. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-SGTR01-04, Steam Generator, Obj. 6c, Pg. 9
2. Salem 2-EOP-SGTR-1, "Steam Generator Tube Rupture, Step 18.
3. KA 000038A136 (4.3/4.5)
   000038A136 ..(KA's)

ANSWER: 090 (1.00)

c. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-SGTR01-04, Steam Generator, Obj. 6b, Pg. 9
2. Salem 2-EOP-SGTR-1, "Steam Generator Tube Rupture, Step 15.
3. KA 000038K308 (4.1/4.2)
   000038K308 ..(KA's)

ANSWER: 091 (1.00)

c. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-EOPINT-02, Introduction to the EOP's, Obj. 9, Pg. 12.
2. Salem 2-EOP-CSFT-1, "Critical Safety Function Status Tress, Fig's 1-6.
3. KA 000054G011 (3.4/3.3)
   000054G011 ..(KA's)

ANSWER: 092 (1.00)

a. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-ABCA01-00, Loss of Control Air, Obj. 3, Pg. 6.
2. Salem S2.OP-AB.CA-0001(Q), "Loss of Control Air", Att. 1.
3. KA 000065A208 (2.9/3.3)
   000065A208 ..(KA's)

ANSWER: 093 (1.00)

b. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-ABCA01-00, Loss of Control Air, Obj. 3, Pg. 6.
2. Salem S2.OP-AB.CA-0001(Q), "Loss of Control Air", Reactor trip requirements.
3. KA 000065A206 (3.6/4.2)
   000065A206 ..(KA's)

ANSWER: 094 (1.00) DD 11/7/93

b. [+1.0] or a.

REFERENCE:

1. Salem 300S-000.00S-EOPINT-02, Introduction to the EOP's, Obj. 9, Pg. 12.
2. Salem 2-EOP-CSFT-1, "Critical Safety Function Status Tress, Fig's 1-3.1.
3. KA 194001A116 (3.1/4.4)
   194001A116 ..(KA's)

ANSWER: 095 (1.00)

b. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-ABCN01-00, Main Feedwater/Condensate System Abnormality, Obj. 1, Pg 5.
2. Salem S2.OP-AB.CN-0001(Q), "Main Feedwater/Condensate System Abnormality", Immediate Actions.
3. KA 000054G010 (3.2/3.2)
   000054G010 ..(KA's)

ANSWER: 096 (1.00)

c. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-AFUEL2-00, Fuel handling Incident, Obj. 3, Pg 8.
2. Salem S2.OP-AB.FUEL-0002(Q) "Fuel Handling Incident", Subsequent Actions.
3. KA 000[illegible]G011 (3.5/3.9)
   000[illegible]G011 ..(KA's)

ANSWER: 097 (1.00)

a. [+1.0]

REFERENCE:

1. Salem SD 25, Pressurizer Pressure and Level Control.
2. Salem L/P 300S-000-00S-PZRP/L-00, Pressurizer Pressure Level Control, Obj. 5, Pg. 28-29.
3. KA 000028A101 (3.8/3.9)
   000028A101 ..(KA's)

ANSWER: 098 (1.00)

a. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-LOPA00-01, Loss of AC power and Recovery Obj. 3 Pg 15
2. Salem 2-EOP-LOPA-1, "Loss of All AC Power and Recovery" Step 40.
3. KA 000056K302 (4.4/4.7)
   000056K302 ..(KA's)

ANSWER: 099 (1.00)

c. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-LOCA05-03 Obj 2, Pg 9.
2. 2-EOP-LOCA-5, ""Loss of Emergency Recirculation",Sheet 4.
3. KA 000011K312 (4.4/4.6)
   000011K312 ..(KA's)

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

001 b
002 d
003 d
004 a
005 d
006 b
007 d
008 b
009 a
010 c
011 a
012 a
013 b
014 b
015 a
016 c
017 c
018 d
019 c
020 d
021 c
022 b
023 c
024 a
025 d
026 d
027 ~~a~~ c Deleted 1/7/93
028 ~~c~~ c Deleted 1/7/93
029 b
030 a
031 c
032 c
033 A or C 1/7/93
034 a
035 a
036 d
037 c
038 a
039 b
040 b
041 b
042 c
043 d
044 c
045 b
046 ~~a~~ c Deleted 1/7/93
047 c
048 ~~c~~ c Deleted 1/7/93
049 d
050 d

051	d	076	a
052	c	077	d
053	c	078	d
054	b	079	c
055	a	080	d
056	c	081	b
057	~~c~~ a 1/7/93	082	b
058	d	083	a
059	a	084	c
060	a	085	c
061	d	086	b
062	d	087	a
063	c	088	b
064	b	089	c
065	b	090	c
066	a	091	c
067	d	092	a
068	a	093	b
069	b	094	b or a Q2 1/7/93
070	c	095	b
071	b	096	c
072	b	097	a
073	b	098	a
074	c	099	c
075	b		

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

Organized by Question Number

QUESTION	VALUE	REFERENCE
001	1.00	8000001
002	1.00	8000002
003	1.00	8000003
004	1.00	8000004
005	1.00	8000005
006	1.00	8000006
007	1.00	8000007
008	1.00	8000008
009	1.00	8000009
010	1.00	8000010
011	1.00	8000011
012	1.00	8000012
013	1.00	8000014
014	1.00	8000015
015	1.00	8000016
016	1.00	8000017
017	1.00	8000018
018	1.00	8000019
019	1.00	8000020
020	1.00	8000021
021	1.00	8000022
022	1.00	8000023
023	1.00	8000024
024	1.00	8000025
025	1.00	8000026
026	1.00	8000027
~~027~~	~~1.00~~	~~8000028~~
~~028~~	~~1.00~~	~~8000029~~
029	1.00	8000030
030	1.00	8000031
031	1.00	8000032
032	1.00	8000033
033	1.00	8000034
034	1.00	8000035
035	1.00	8000036
036	1.00	8000037
037	1.00	8000038
038	1.00	8000039
039	1.00	8000040
040	1.00	8000041
041	1.00	8000042
042	1.00	8000043
043	1.00	8000044
044	1.00	8000045
045	1.00	8000046
~~046~~	~~1.00~~	~~8000047~~
047	1.00	8000048
~~048~~	~~1.00~~	~~8000049~~
049	1.00	8000050
050	1.00	8000051

DA 11/7/93

DA 11/7/93

Organized by Question Number

QUESTION	VALUE	REFERENCE
051	1.00	8000052
052	1.00	8000053
053	1.00	8000054
054	1.00	8000055
055	1.00	8000056
056	1.00	8000057
057	1.00	8000058
058	1.00	8000059
059	1.00	8000060
060	1.00	8000061
061	1.00	8000062
062	1.00	8000063
063	1.00	8000064
064	1.00	8000065
065	1.00	8000067
066	1.00	8000068
067	1.00	8000069
068	1.00	8000070
069	1.00	8000071
070	1.00	8000072
071	1.00	8000073
072	1.00	8000074
073	1.00	8000075
074	1.00	8000076
075	1.00	8000077
076	1.00	8000078
077	1.00	8000079
078	1.00	8000080
079	1.00	8000081
080	1.00	8000082
081	1.00	8000083
082	1.00	8000084
083	1.00	8000085
084	1.00	8000086
085	1.00	8000087
086	1.00	8000088
087	1.00	8000089
088	1.00	8000090
089	1.00	8000091
090	1.00	8000092
091	1.00	8000093
092	1.00	8000094
093	1.00	8000095
094	1.00	8000096
095	1.00	8000097
096	1.00	8000098
097	1.00	8000099
098	1.00	8000100
099	1.00	8000066
	~~99.00~~	
	95.00	1/7/97

SRO Exam PWR Reactor
Organized by KA Group

PLANT WIDE GENERICS

QUESTION	VALUE	KA
005	1.00	194001A102
003	1.00	194001A102
002	1.00	194001A103
017	1.00	194001A110
015	1.00	194001A111
094	1.00	194001A116
006	1.00	194001K101
008	1.00	194001K101
010	1.00	194001K102
009	1.00	194001K102
016	1.00	194001K103
011	1.00	194001K107
007	1.00	194001K107
001	1.00	194001K108
013	1.00	194001K111
014	1.00	194001K116
PWG Total	16.00	

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
018	1.00	001050A201
004	1.00	003000K407
037	1.00	003000K407
023	1.00	004000K406
024	1.00	004010K403
025	1.00	013000K201
019	1.00	014000A103
038	1.00	014000A202
069	1.00	015000K301
020	1.00	015000K405
021	1.00	015000K407
022	1.00	017020A401
~~037~~	~~1.00~~	~~026000K201~~ c Q1 1/7/93
031	1.00	061000K402
030	1.00	061000K406
032	1.00	071000A409
033	1.00	071000G006
036	1.00	072000K402
PS-I Total	~~18.00~~ Q1 17.00 1/7/93	

S R O E x a m P 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 II

QUESTION	VALUE	KA
039	1.00	002000G010
040	1.00	002000K107
026	1.00	006000A301
042	1.00	010000K403
041	1.00	010000K601
043	1.00	012000A202
044	1.00	012000A306
045	1.00	027000A403
047	1.00	028000A201
~~046~~	~~1.00~~	~~028000K503~~ c 01 1/2/93
056	1.00	034000A102
049	1.00	035010K601
052	1.00	062000A201
053	1.00	062000K403
054	1.00	064000A401
055	1.00	064000K401
051	1.00	086000A204
PS-II Total	~~17.00~~ 16.00	02 1/2/93

Group III

QUESTION	VALUE	KA
058	1.00	005000A102
~~028~~	~~1.00~~	~~008000A102~~ c 02 1/2/93
029	1.00	008000G010
057	1.00	045000G014
PS-III Total	~~4.00~~ 3.00	03 1/2/93
PS Total	~~39.00~~ 36.00	01 1/4/93

SRO Exam PWR Reactor
Organized by KA Group

EMERGENCY PLANT EVOLUTIONS

Group I

QUESTION	VALUE	KA
061	1.00	000001A205
060	1.00	000003G007
062	1.00	000005A203
084	1.00	000011A113

EMERGENCY PLANT EVOLUTIONS

Group I

QUESTION	VALUE	KA
099	1.00	000011K312
035	1.00	000015K104
071	1.00	000026G010
074	1.00	000029A201
070	1.00	000029G010
073	1.00	000029K312
075	1.00	000040A201
076	1.00	000040K304
077	1.00	000051A202
079	1.00	000055A202
078	1.00	000055K302
081	1.00	000057A106
080	1.00	000057G010
012	1.00	000058A203
082	1.00	000059A204
034	1.00	000059K301
068	1.00	000068A112
067	1.00	000068K318
083	1.00	000074G010
063	1.00	000074K308
072	1.00	000076G004

EPE-I Total	25.00	

S R O E x a m P 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

Group II

QUESTION	VALUE	KA
066	1.00	000008A108
065	1.00	000009K321
064	1.00	000009K323
086	1.00	000025A102
059	1.00	000025K101
085	1.00	000025K301
087	1.00	000027A216
~~048~~	~~1.00~~	~~000037K307~~ Ad 1/7/93
089	1.00	000038A136
090	1.00	000038K308
095	1.00	000054G010
091	1.00	000054G011
088	1.00	000060K202
050	1.00	000061A101
093	1.00	000065A206
092	1.00	000065A208

EPE-II Total	~~13.00~~ Ad 1/7/93 15.00	

EMERGENCY PLANT EVOLUTIONS

Group III

QUESTION	VALUE	KA
097	1.00	000028A101
096	1.00	000036G011
098	1.00	000056K302

EPE-III Total	3.00	

EPE Total	~~44.00~~ Ad 1/7/93 ~~32.00~~	

Test Total	~~99.00~~ Ad 1/7/93 95.00	

ATTACHMENT 2

POST-EXAMINATION FACILITY COMMENTS

Public Service
Electric and Gas
Company

Stanley LaBruna

Public Service Electric and Gas Company P.O. Box 236, Hancocks Bridge, NJ 08038 609-339-1200

Vice President - Nuclear Operations

December 17, 1992

NTC-92-3220

Mr. T. Timothy Martin
Regional Administrator
U. S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406-1415

Dear Mr. Martin:

WRITTEN EXAMINATION REVIEW COMMENTS - SALEM SENIOR REACTOR OPERATOR EXAMINATION

On November 30, 1992, and again on December 11, 1992, our examination review team had an opportunity to review and comment on the proposed Salem SRO Written Examination. There were numerous comments and recommendations, most of which were incorporated into the examination. We appreciate the efforts of David Silk, Chief Examiner, who was both patient and open-minded concerning our comments and recommendations. The pre-examination review continues to be an important part of conducting these examinations in a manner fair to both the evaluators and candidates.

The post-examination review with the candidates and other instructors raised issues that were not identified in the pre-examination review or were not solved by the changes that were incorporated. There are four such questions (Numbers 027, 076, 087, 094) for which we have enclosed comments. The format for the comments is:

- Question
- Key Answer and Reference Data
- Facility Comment
- Facility Recommendation
- Supporting Documentation (attached)

There were four other questions (Numbers 029, 057, 082, 096) that we are not formally challenging but, because they dealt with information in procedures that would be used in-hand, were less than desirable and one question (003) has resulted in a review of the related reference to determine if the step/action is appropriate or necessary.

If you have any questions, comments, or need additional information, please contact A. Orticelle (609) 339-3800, or G. Mecchi (609) 339-3857. They will provide the requested information or will see that you are contacted by the appropriate person.

Sincerely,

S. LaBruna

Attachments (4)

C Mr. Lee H. Bettenhausen
Chief, Operator Licensing Branch
U.S. NRC, Region I

Mr. David Silk
Chief Examiner
U.S. NRC, Region I

QUESTION: 027 (1.00)

Given the following:

- Unit 1 is at 100% power.
- PT-948A - Containment Pressure Channel I is failed high.
- ALL Bistables for PT-948A have been placed in the tripped condition.

WHICH ONE (1) of the following would occur if PT-948D Containment Pressure detector Channel IV failed high?

a. Main Steam Isolation and Containment Spray Actuation

b. Containment Spray Actuation and Phase A

c. Main Steam Isolation, Safety Injection and Phase A

d. Main Steam Isolation, Safety Injection, Containment Spray Actuation, and Phase A

ANSWER: 027 (1.00)

a. [+1.0]

REFERENCE:

1. Salem L/P - 300S-000.00S-CSPRAY-06, ELO 4, Pg. 15-16.
2. Salem Logic Diagram 221057
3. KA 026000A301 (4.3/4.5)
   026000K201 ..(KA's)

Facility Comment:

There is no correct answer because the channel numbers and bistable numbers are inconsistent with the actual system logic. Both of the indicated bistables are fed from Containment Pressure, Channel IV. In addition, the failure of a containment pressure signal would result in a bypass of that input to the Hi-Hi Containment Pressure circuit so that the logic changes from 2/4 to 2/3. The end result of the actions described in the question would be 1/3 signals necessary for an SI signal and Containment Hi-Hi Press logic changed to 2/3.

In addition, this question is fundamentally the same as #87 in that they both deal with four channels of protection that utilize all four channels to generate one protection signal and three of those same channels to generate another through numerous bistables. Having two questions on an examination that test the same knowledge presents a "double jeopardy situation" for the candidate who has not memorized the logic diagrams. While our training does require a recall understanding of the number of channels, coincidence, setpoint, etc., it is not intended that candidates memorize specific channel and/or bistable functions unless a failure of that channel would cause an immediate control response. Otherwise, they are expected to utilize available references to answer the questions.

Recommended Action: Delete this question based on there being no correct answer and an unnecessary required recall level of knowledge.

QUESTION: 076 (1.00)

Given the following:

- A double ended main steam line break inside containment has occurred.
- Operators are responding to the event per 2-EOP-LOSC-1, "Loss of Secondary Coolant."

WHICH ONE (1) of the following will be available to the operator to confirm the diagnosis of a SGTR in the FAULTED S/G?

a. S/G sample results.

b. R-19 Blowdown radiation monitors.

c. R-15 Condenser Air Ejector radiation monitor.

d. Uncontrolled S/G level increase.

ANSWER: 076 (1.00)

a. [+1.0]

REFERENCE:

1. Salem L/P 300S-000.00S-LOSC01-02, Loss of Secondary Coolant Obj. 5 Pg 10.
2. Salem 2-EOP-LOSC-1, "Loss of Secondary Coolant", Step 11.
3. KA 000040K304 (4.5/4.7)
   000040K304 ..(KA's)

Facility Comment

This question was the subject of much discussion during the examination review. The primary means utilized to identify a SGTR is an uncontrolled SG level increase or certain process radiation alarms. Given the coincident LOSC in the question, the process radiation alarms are not a factor because both Phase A and Steamline Isolation have occurred. If a SGTR exists and an uncontrolled SG level increase cannot be confirmed at Step 11 - LOSC, then at Step 14 - LOSC (LOCA Evaluation), or as a result of a Continuous Action Summary (CAS), the Containment Radiation Monitors might require a transition to LOCA-1. In both LOSC-1 and LOCA-1 there are steps requiring that a sample be drawn after certain related interlocks are cleared. However, that step in both procedures comes after the primary means (uncontrolled level increase, process radiation) have been evaluated and the interlocks are cleared. Even though the uncontrolled level increase may not be a good indicator of a SGTR for this situation, the term "available" implies an on-line function. These factors create a problem for the candidate responding to this question on a written examination because the situation is much more complex than the question. In retrospect, a better question would have been the basis for requiring a SG sample.

Recommended Action: Given the technical complexity of the situation presented and the wording of the question (available), either a. or d. should be considered to be correct.

QUESTION: 087 (1.00)

Given the following:

- Unit 1 is at 100% power.
- PT-474, Pressurizer pressure channel IV is failed LOW.
- All actions of OP.IV-10.3.1, "Removing, Returning to Service and Loss of a Protective System Channel" are complete.

WHICH ONE (1) of the following describes the plant response to a failure of PT-455, Pressurizer pressure channel I, LOW?

a. Reactor trip.

b. Reactor trip AND safety injection.

c. All heaters energize, pressure increases to PORV setpoint, 1PR2 opens and 1PR1 does not, pressure cycles around PR2 setpoint.

d. All heaters energize, pressure increases to PORV setpoint, 1PR1 opens, 1PR2 does not, pressure cycles around PR1 setpoint.

ANSWER: 087 (1.00)

a. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-ICF000-03, I&C FAILURE ANALYSIS, Obj. 2e, Pg. 50.
2. Salem Logic Diagram 221055, Reactor Protection System Pressurized Trip Signals.
3. KA 000027A216 (3.6/3.9)
   000027A216 ..(KA's)

Facility Comment:

This question is fundamentally the same as #27 in that they both deal with four channels of protection that utilize all four channels to generate one protection signal and three of those same channels to generate another through numerous bistables. Having two questions on an examination that test the same knowledge presents a "double jeopardy situation" for the candidate who has not memorized the logic diagrams. While our training does require a recall understanding of the number of channels, coincidence, setpoint, etc., it is not intended that candidates memorize specific channel and/or bistable functions unless a failure of that channel would cause an immediate control response. Otherwise they are expected to utilize available references to answer the questions, i.e., reactor protection status panel indication.

Recommended Action: Delete this question based on an unnecessary required recall level of knowledge.

QUESTION: 094 (1.00)

WHICH ONE (1) of the following is NOT monitored by the Critical Safety Function Status Trees to determine loss of containment integrity?

a. Containment hydrogen concentration.

b. Containment temperature.

c. Containment sump level.

d. Containment pressure.

ANSWER: 094 (1.00)

b. [+1.0]

REFERENCE:

1. Salem 300S-000.00S-EOPINT-02, Introduction to the EOP's, Obj. 9, Pg. 12.
2. Salem 2-EOP-CSFT-1, "Critical Safety Function Status Tress, Fig's 1-6.1.
3. KA 194001A116 (3.1/4.4)
   194001A116 ..(KA's)

Facility Comment

A review of the applicable CSFST indicates that both a. and b. are correct responses. Reference attached.

Recommended Action: Consider both a. and b. as correct responses.

ATTACHMENT 3

NRC POST-EXAMINATION RESOLUTIONS

A. NRC Response to Facility Comments

Question No.	Resolution
27	This question was reworded at the request of the facility. The most notable change to the question was the addition of channel numbers (I-IV) to the stem. The changes requested were added verbatim. The channel numbers provided by the facility for the bistables were incorrect with the result of there being no correct answer for the question. The question was deleted.
76	This question dealt with the diagnostic ability of the candidate and his familiarity with plant conditions during the response to a faulted-ruptured SG. The facility recommends that answers A and D be accepted as correct, even though their argument supports the designated answer. Comment not accepted. A will remain the only acceptable answer.
87	The question is supported by the facility learning objectives. No objections were raised by the facility during the preexamination review. Comment not accepted. The question will remain in the examination.
94	Because hydrogen is not monitored by the CSFSTs, A will be accepted as a correct response also.

B. Additional NRC Examination Changes

Question No.	Resolution
28	This question was testing for knowledge of the service water temperature limit. The question was deleted because of ambiguity caused by the wording "for performing an RCS cooldown at the normal rate." The possible emphasis on the cooldown rate could have been misleading.

33 The NRC will accept C as an additional correct answer because this distractor was not totally incorrect. The wording did not specify the 1500 mrem/year as a maximum exposure limit. 500 mrem/year was the correct maximum limit. If the public is limited to less than 500 mrem/year it will also not exceed the 1500 mrem/year as stated in distractor C.

46 This question was deleted, based upon further review of the reference material. To exceed the hydrogen flammability limit several mechanisms of hydrogen production would be needed. The question implied that only one mechanism would produce the necessary hydrogen; therefore, the question was not valid.

48 This question was deleted due to ambiguity in regard to the size of the SG tube leak. If one assumes that the leak is small and the AOP was being implemented then only one choice is correct. If one assumes a rupture occurred and the EOPs were being implemented then two of the distractors would also be correct according to the EOP bases. Therefore, the question was deleted.

57 Due to a typographic error in the answer key, the correct answer is A instead of C.

ATTACHMENT 4

SIMULATOR FIDELITY REPORT

Facility Licensee: Public Service Electric and Gas Company
244 Chestnut Street
Salem, N.J. 08079

Facility Docket Nos.: 50-272 and 50-311

Initial Examination Administered on: December 14 - 17, 1992

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 which may be used in future evaluations. No licensee action is required in response to these observations. During the examination the following items were observed:

The NRC ran three scenarios on two different crews on the same day. Running the same scenarios permitted the exam team to observe the consistency of the simulator to replicate malfunctions. In all scenarios, the simulator was consistent and appeared to correctly model RCS and secondary responses expect in one instance. The exception was the absence of core decay heat. Specifically, on a loss of all main and auxiliary feedwater, the CETs were stable and in some instances were slightly decreasing. Besides the decay heat model the only other deficiency was the unreliability of the plant computer (P-250) in the simulator.

Overall, the simulator performed well. During the three days that the simulator was used for scenarios and JPMs, it modeled a variety of conditions and was useful in the licensing decision process.