ENCLOSURE 1

EXAMINATION REPORT - 50-338/OL-85-02

Facility Licensee: Virginia Electric and Power Company
Richmond, Virginia 23261

Facility Name: North Anna

Facility Docket Nos.: 50-338 and 50-339

Written, simulator and oral examinations were administered at North Anna near Mineral, Virginia.

Chief Examiner: Sandy Lawyer — Sandy Lawyer — 11/18/85 Date Signed

Approved by: Bruce A. Wilson — Bruce A. Wilson, Section Chief — 11/19/85 Date Signed

Summary:

Examinations on October 15-18, 1985

Oral examinations were administered to 12 candidates; all of whom passed. Simulator examinations were administered to 11 candidates; all of whom passed. Written examinations were administered to 11 candidates; ten of whom passed.

REPORT DETAILS

1. Facility Employees Contacted:

   D. C. Hawkins, Senior Instructor-Nuclear/SRO Lead Instructor
   W. Shura, Senior Instructor-Nuclear
   L. Edmonds, Superintendent, Nuclear Training
   L. R. Buck, Supervisor-Training (Power Station OPS)
   R. O. Enfinger, Superintendent, Operations
   B. L. Shriver, Director, Nuclear Training
   E. R. Smith, Jr., Assistant Station Manager (NS&L)
   M. D. Crist, Senior Instructor-Nuclear
   D. C. Fellows, Lead Simulator Instructor
   J. Ballard, Nuclear Training Auditor
   L. Johnson, Nuclear Traning Auditor

   (All Attended Exit Meeting)

2. Examiners:

   *Sandy Lawyer, NRC Examiner
   Tom Rogers, NRC Examiner
   Bill Dean, NRC Examiner
   Gerry Douglas, NRC Examiner

   *Chief Examiner

3. Examination Review

   Prior to the exit meeting, the examiners met with W. Shura, D. Hawkins, L. Edmonds, and R. Buck to review the written examination and facility comments. The following are the NRC responses to the facility comments and recommendations of Enclosure 4.

   a. RO Exam

      (1) Question - Section 2 General Comments

      The comment states that some of the examination questions, particularly on Category 2, are not operationally oriented and are not supported by the (North Anna) job task analysis (JTA). The NRC's responsibility, at the present time, is to ensure that written examinations comply with the requirements of 10 CFR 55, specifically paragraphs 55.21 and 55.22. NUREG-1021, Operator Licensing Examiner Standards, provides clarification concerning the scope and content of examinations. NUREG-1122, the recently issued K/A Catalog, provides additional guidance to the examiners on testable knowledges and abilities. Also included in NUREG-1122 are relative importance ratings that serve as guides

in determining which topics should or should not be included in examinations. Both of these NUREG's are intended to clarify and provide guidance, but not supercede the pertinent regulations. Since the North Anna JTA was not provided to the NRC with the other reference material, we are unable to determine if it encompasses the requirements and guidance of the above three mentioned documents. It is therefore, not inconceivable that some NRC questions may not be supported by the North Anna JTA, nor meet the North Anna definition of operationally oriented.

(2) Question 2.02 - We agree with the comment. Section (c) of Question 2.02 was deleted.

(3) Question 2.05 - Step 4.1.14 in 1-OP-14.1 refers to manipulation of CC from the RHR heat exchanger to ensure that CC return temp is <700°F while step 4.1.25 refers to positioning of RHR heat exchange outlet to control the cooldown rate. The question asked for the valve that the operator manually adjusts to reduce RCS temp. Only answer (b) is correct. No change required.

(4) Question 2.06 - This question was intended to determine the candidates knowledge of the pressurizer pressure control system and each distractor (including b) was specifically worded to test on misconceptions or concepts important to satisfactory job performance. Since the wording of distractor (b) was ambiguous, (i.e., improved wording would be " ... each bypass line is sized to permit 1 gpm flow rate ...") but is clearly incorrect, this choice cannot be accepted for credit. Due to the apparent confusion created, this question was deleted from the examination. We are concerned that no candidate was able to choose the correct answer, choice (c), which may indicate a training deficiency in this area.

(5) Question 2.09 - We disagree with the recommendation and in part with the comment. This question tests knowledge of the design of an operationally important feature of the loop isolation valves. No change required.

(6) Question 2.17 - We disagree with the comment in that the question does not "ask for obscure descriptions and words taken out of context." We agree that a variety of correct answers to fit the format of the blanks could result. All reasonable answers were accepted. No change required.

(7) Question 2.18 - We disagree with the comment and agree with the recommendation. Facility's recommended additional answers are essentially the same as the answers given in the answer key. Any reasonable answer similar to the answer key was accepted. No change required.

(8) Question 2.19 - We disagree with the comment but agree with the recommendation. Knowledge of rod internal pressure buildup is important operator knowledge due to its impact on fission product poison leakage and structural strength of the individual fuel rods. However, conflicting curves given in lesson plan NCRODP 88.1 and in NAPS FSAR Fig. 4.2-5 could lead to some confusion as to which curve is the correct choice. Therefore, the question was deleted.

(9) Question 2.20 - We disagree with the comment. Knowledge of design basis accidents, including assumed mitigating events, is required knowledge. No change required.

(10) Question 2.23 - The question states the Decontamination Factor (DF); it does not require the candidate to state it. The K/A Catalog, NUREG-1122, lists demineralizer performance and design attributes including isotopic control in two different areas of required RO knowledge. Although not asked for in this question, NUREG-1122 also lists DF's as an RO knowledge area; however, it is relatively low in importance. No change required.

(11) Question 2.25 - We disagree with the comment and recommendation in part. Design of important components such as the MSRs is necessary knowledge for the reactor operator (Ref: NUREG-1122, System 039). However, we agree that the candidates may have been confused by the wording of the second part of the question. The portion of the question dealing with the effect on the plant's thermal power capabilities was deleted.

(12) Question 2.26 - Answers listing correct physical or system location will be accepted. The recommended answers were added to the answer key.

(13) Question 2.31(1.c) - Tours of the facility indicated that (2) is the only correct answer. No change required.

(14) Question 3.03 - Lesson Plan 91.1 conflicts with TS on this setpoint and should be changed. RCS pressure ¶2000 psig was accepted. The answer key was changed accordingly.

(15) Question 3.06 - We do not concur that information asked for is beyond the scope of an RO's knowledge and abilities. The RO candidate must be familiar with the conditions that require the use of safety and emergency systems and why such protection is required with emphasis on areas where a malfunction will require immediate operator action. Malfunctions of instruments and controls taken credit for in accident analyses are prime examples. No change is required.

(16) Question 3.09 - We consider this a required knowledge area for RO's and SRO's. It is in fact specifically listed in NUREG-1021, ES-202, Paragraph B3; "The candidates should have sufficient knowledge of the nuclear instruments ... to answer questions concerning ... (the) diagrammatic representation of instrumentation systems."

It is also listed at a relatively high importance rating in NUREG-1122, System 015. No change to question or answer key.

(17) Question 3.16 - We agree that the first part of the required answer is more important operator knowledge. However, the second part of the answer does not require "knowledge of maintenance tasks performed by instrument technicians". It merely requires the operator to know that some tasks performed by the instrument tech are made more convenient - not what the tasks are or how they are performed. The answers were weighted (.75) for the first part and (.25) for the second part.

(18) Question 3.18 - During staff review, a typographical error was discovered in the answer to part (a). The correct answer is 7. The answer key was changed accordingly.

(19) Question 3.22 & 3.23 - It was discovered during the examination that these questions were testing the same candidate knowledge. Since question 3.23 was in the preferred wording, it was retained and a question substituted for 3.22. The candidates were informed and the master question and answer key have been changed accordingly.

(20) Question 3.27 - We agree. The FSAR reference supplied supports the comment and the recommendation is an appropriate resolution. The answer key was changed to accept either (b) or (c).

(21) Question 4.12 - We disagree. Recognition of a correct answer from among incorrect distractors does not require memorization of the correct answer. In addition, candidates are responsible for knowledge of the contents of normal, off-normal, and emergency procedures, regardless of whether the procedure contains immediate action steps. These steps which are not immediate actions need not be memorized, but the candidate must be able to describe, conceptually, the objectives and methods used to achieve those objectives for all emergency and off-normal procedures. No change required.

(22) Question 4.14 - We disagree. This question did not require verbatim recitation of the five items. This is an example of asking the candidate to describe, conceptually, the five methods used to achieve the objective of ensuring that certain Tech Spec requirements are met.

During staff post examination review, two additional answers were judged to be correct. These were added to the answer sheet. No other changes were required.

(23) Question 4.23 - It was discovered by the staff during onsite post examination review that a fifth answer was correct. This was a result of the utility supplied material being outdated. The answer key was changed accordingly.

(24) Question 4.38 - We agree with the comment and the utility recommendation. Any answer from the RWP was accepted.

b. SRO Exam

(1) Question 6.02 - Utility comment and NRC response same as Question 2.06.

(2) Question 6.06 - Utility comment and NRC response same as Question 3.27.

(3) Question 6.10 - Utility comment and NRC response same as Question 2.19.

(4) Question 6.19(c) - Utility comment and NRC response same as Question 2.31(c).

(5) Question 6.24 - Utility comment and NRC response same as Question 2.25.

(6) Question 6.28 - Knowledge of Fire Protection Systems (FPS) including fire, smoke and heat detectors and ability to monitor automatic operation of the FPS, including actuation of fire detectors is listed in NUREG-1122. Both areas are rated sufficiently high in Importance to Safety particularly for SRO's. Actuation setpoints must be known in order to properly monitor automatic operation. In this instance, the staff judged that ±10°F was an acceptable tolerance range for knowledge of setpoint actuation. The answer key was changed accordingly.

(7) Question 7.02 - Utility comment and NRC response same as Question 4.12.

(8) Question 7.30 - The Purpose of AP-52, "Loss of Refueling Cavity Level During Refueling" is the following:

This procedure provides the indication of, probable cause for, and the immediate and long term action to be taken in the event of a Rapid Decrease in Refueling Cavity Level.

The first response in this procedure is to "Immediately commence make-up to the refueling cavity." Question 7.30 asked for this immediate makeup lineup and the answer key stated simply:

"From RWST through LPSH into hot legs."

We believe the question and answer are entirely appropriate since:

1. Rapid loss of refueling cavity level is a distinct possibility.

2. North Anna has an approved procedure for this condition in which an immediate response is indicated.

3. Long term action memorization, Tech Spec operability requirements and detailed valve lineups were not required in the answer.

No change to question or answer key.

(9) Question 7.34 - Utility comment and NRC response same as Question 4.23.

(10) Question 7.38 - Utility comment and NRC response same as Question 4.38.

(11) Question 7.39 - Utility comment and NRC response same as Question 4.14.

(12) Question 8.01 - We agree with the latter part of the comment and with the recommendation. Based upon discussions with facility senior operations personnel, we have determined that a conflict exists between NA Administrative Instruction 16.5, which lists several functions of the SS when performing an Emergency Work Request and the facility's current policy of having others handle these administrative requirements. The facility Administrative Instruction 16.5 will be changed to reflect the current practice. The question was deleted.

(13) Question 8.05 - We agree with the comment and recommendation. The question was deleted.

(14) Question 8.07 - We agree in part. Review of the subject amendments reveal that only distractor (c) is affected. Both answers (b) and (c) were accepted.

(15) Question 8.33(b) - We agree with the comment and recommendation. Updating of training material should be accomplished as soon as possible. Both answers were accepted.

(16) Question 8.35 - We agree with the comment and recommendation. Updating of training material should be accomplished as soon as possible. Both answers were accepted.

4. Exit Meeting

At the conclusion of the site visit, the examiners met with representatives of the plant staff to discuss the results of the examination. Those individuals who clearly passed the oral and simulator examinations were identified.

There were no generic weaknesses (greater than 75 percent of candidates giving incorrect answers to one examination topic) noted during the oral examination.

The cooperation given to the examiners and the effort to ensure an atmosphere in the control room conducive to oral examinations were also noted and appreciated.

The licensee did not identify as proprietary any of the material provided to or reviewed by the examiners.

ENCLOSURE 3

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: NORTH ANNA 1&2

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 85/10/15

EXAMINER: TOM ROGERS

APPLICANT: MASTER FILE Copy

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
40.00	25.00	________	________	1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
38.0 ~~40.00~~	25.00	________	________	2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
40.00	25.00	________	________	3. INSTRUMENTS AND CONTROLS
40.00	25.00	________	________	4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
158.0 ~~160.00~~	100.00	________	________	TOTALS

FINAL GRADE ________________%

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

____
APPLICANT'S SIGNATURE

QUESTION 1.01 (1.00)

An ECP is calculated for a reactor startup 4 hours after a reactor trip from 100% equilibrium conditions. Which of the following conditions would cause the actual critical position to be lower than the ECP?

a. The startup is delayed until 8 hours after the trip.

b. Actual boron concentration is 10 ppm more than the predicted boron concentration.

c. A rod finger is separated from its spider assembly.

d. The steam dump pressure setpoint is lowered by 100 psi prior to reactor startup.

QUESTION 1.02 (1.00)

Which of the following will cause the Axial Flux Difference to become more positive (less negative)?

a. Power increase with power defect compensated for by dilution only.

b. Power increase with power defect compensated for by rod withdrawal only.

c. Buildup of xenon in top portion of core.

d. Burnup of xenon in bottom portion of core.

QUESTION 1.03 (1.00)

Which of the following will cause the Moderator Temperature Coefficient to become more negative?

a. Increasing boron concentration.

b. Insertion of control rods.

c. Decreasing moderator temperature.

d. Flux moving towards center of core.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.04 (1.00)

Which of the following will cause the Fuel Temperature Coefficient (pcm per degree) to become less negative?

a. Fuel temperature increase.

b. Boron concentration decrease.

c. Control rod insertion (at constant power).

d. Core age increase.

QUESTION 1.05 (2.00)

If steam goes through a throttling process, indicate whether the following parameters will INCREASE, DECREASE, or REMAIN THE SAME.

a. Enthalpy (0.5)

b. Pressure (0.5)

c. Entropy (0.5)

d. Temperature (0.5)

QUESTION 1.06 (2.00)

Answer TRUE or FALSE to the following.

a. The use of a ramped Tave program allows the plant to operate with a higher thermodynamic efficiency than does a constant Tave program. (0.5)

b. Increasing condensate depression (subcooling) will cause BOTH a decrease in plant efficiency AND an increase in condensate (hotwell) pump available NPSH. (0.5)

c. During a RCS heatup, as temperature gets higher, it will take a smaller letdown flow rate to maintain a constant pressurizer level. (0.5)

d. The difference between pump suction pressure and the saturation pressure of the fluid being pumped is referred to as net positive suction head. (0.5)

QUESTION 1.07 (1.00)

State whether you AGREE or DISAGREE with the following statements.

a. Starting with a reactor that is critical low in the intermediate range, an equal amount of positive or negative reactivity (50 pcm) insertion will produce stable SURs of equal magnitude. (0.5)

b. Whenever NR-45 (SRNI) is drawing a straight vertical line, the reactor is critical. (0.5)

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.08 (1.00)

The -1/3 DPM SUR following a reactor trip is caused by which of the following?

a. The decay constant of the longest-lived group of delayed neutrons.

b. The ability of U-235 to fission with source neutrons.

c. The amount of negative reactivity added on a trip being greater than the Shutdown Margin.

d. The doppler effect adding positive reactivity due to the temperature decrease following a trip.

QUESTION 1.09 (1.00)

The reactor is producing 100% rated thermal power at a core delta T of 60 degrees and a mass flow rate of 100% when a blackout occurs. Natural circulation is established and core delta T goes to 40 degrees. If decay heat is 2%, what is the core mass flow rate (in %)?

a. 1.3

b. 2.0

c. 3.0

d. 4.0

QUESTION 1.10 (1.00)

Which of the following statements concerning Xenon-135 production and removal is correct?

a. At full power, equilibrium conditions, about half of the xenon is produced by iodine decay and the other half is produced as a direct fission product.

b. Following a reactor trip from equilibrium conditions, xenon peaks because delayed neutron precursors continue to decay to xenon while neutron absorption (burnout) has ceased.

c. Xenon production and removal increases linearly as power level increases; i.e., the value of 100% equilibrium xenon is twice that of 50% equilibrium xenon.

d. At low power levels, xenon decay is the major removal method. At high power levels, burnout is the major removal method.

QUESTION 1.11 (1.00)

Which of the following is the units of heat flux?

a. Watts / cubic centimeter

b. BTU / (hr square ft)

c. Calories / gram

d. kW / ft

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.12 (2.00)

A motor driven centrifugal pump is operating at rated flow. You then start closing down on the discharge valve. How (INCREASE, DECREASE, or REMAIN THE SAME) will each of the following be affected?

a. Flow (0.5)

b. Discharge Presure (0.5)

c. Available NPSH (0.5)

d. Motor Amps (0.5)

QUESTION 1.13 (1.00)

The main condenser must remove more heat energy to condense which of the following?

a. One pound of steam at 0 psia.

b. One pound of steam at 300 psia.

c. Two pounds of steam at 600 psia.

d. Two pounds of steam at 1200 psia.

QUESTION 1.14 (1.00)

Which of the following describes the changes to the water that occur between the inlet and outlet of a steam generator?

a. Enthalpy DECREASES, Entropy DECREASES, Quality DECREASES

b. Enthalpy INCREASES, Entropy INCREASES, Quality INCREASES

c. Enthalpy CONSTANT, Entropy DECREASES, Quality INCREASES

d. Enthalpy DECREASES, Entropy INCREASES, Quality DECREASES

QUESTION 1.15 (1.00)

The reactor is critical at 10,000 cps when a S/G PORV fails open. Assuming BOL conditions, no rod motion, and no reactor trip, choose the answer below that best describes the values of Tavg and nuclear power for the resulting new steady state. (POAH = point of adding heat).

a. Final Tavg greater than initial Tavg, Final power above POAH.

b. Final Tavg greater than initial Tavg, Final power at POAH.

c. Final Tavg less than initial Tavg, Final power at POAH.

d. Final Tavg less than initial Tavg, Final power above POAH.

QUESTION 1.16 (1.00)

Which of the following statements concerning the use of water as the moderator is correct?

a. Water has a HIGH scattering cross-section, a LOW absorption cross-section, and a LARGE energy decrement per collision.

b. Water has a LOW scattering cross-section, a HIGH absorption cross-section, and a LARGE energy decrement per collision.

c. Water has a HIGH scattering cross-section, a LOW absorption cross-section, and a SMALL energy decrement per collision.

d. Water has a LOW scattering cross-section, a HIGH absorption cross section, and a SMALL energy decrement per collision.

QUESTION 1.17 (1.00)

Which of the following statements describes the relationship between integral and differential rod worth?

a. Integral rod worth (at any location) is the slope of the differential rod worth curve at that location.

b. Integral rod worth (at any location) is the total area under the differential rod worth curve from the end of the rod to that location.

c. Integral rod worth (at any location) is the square of the differential rod worth at that location.

d. There is no relationship between integral and differential rod worth.

QUESTION 1.18 (1.00)

The reactor is operating at 30% power when one RCP trips. Assuming no reactor trip or turbine load change occur, which of the following parameters will DECREASE?

a. Flow in operating reactor coolant loops

b. Core delta T

c. Reactor vessel delta P

d. Steam generator pressure in affected loop

QUESTION 1.19 (1.00)

Which of the following would cause an INCREASE in the conductive heat transfer rate across a slab?

a. Decrease heat transfer area of slab.

b. Decrease delta T across slab.

c. Decrease thickness of slab.

d. Decrease thermal conductivity of slab.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.20 (1.50)

Indicate whether the following statements concerning delayed neutrons are TRUE or FALSE.

a. If the reactor is supercritical, then the fraction of delayed neutrons shifts to the shorter lived precursors and the value of the effective decay constant (lambda) decreases. (0.5)

b. Due to the significant decrease in the percentage of fast fission occuring over core life, the value of the effective delayed neutron fraction decreases over core life. (0.5)

c. Delayed neutrons are produced at some time after fission as a result of the radioactive decay of fission products. (0.5)

QUESTION 1.21 (1.00)

With the reactor initially at a keff of 0.99, a certain reactivity change causes the count rate to double. If this same amount of reactivity is again added to the reactor, which of the following will be the status of the reactor?

a. Subcritical

b. Critical

c. Supercritical

d. Prompt Critical

QUESTION 1.22 (1.00)

A 1/M curve that predicts criticality early is referred to as which of the following?

a. Useless

b. Conservative

c. Non-conservative

d. Ideal

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.23 (1.00)

Which of the following can be defined as 'the number of neutrons causing fission that were originally born delayed divided by the total number of neutrons causing fission'?

a. Lambda effective

b. Rho

c. Beta effective

d. Tau

QUESTION 1.24 (1.00)

Which of the following is NOT a reason for pressurizing the fuel rods with helium?

a. Minimize clad creeping inwards toward fuel pellets.

b. Increase gap (pellet to clad) thermal conductivity.

c. Allow detection of clad failure by helium analysis of the coolant.

d. Maintain lower fuel centerline temperature.

QUESTION 1.25 (.50)

TRUE or FALSE?

For similar heat exchangers operating under the same inlet temperatures and flow rates, a counterflow heat exchanger will transfer more heat than a parallel flow heat exchanger.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.26 (1.00)

Which of the following statements concerning samarium reactivity effects is correct?

a. The equilibrium (at power) value of samarium is dependent upon power level. The peak value of samarium following a shutdown is dependent upon power level prior to shutdown.

b. The equilibrium (at power) valve of samarium is dependent upon power level. The peak value of samarium following a shutdown is independent of power level prior to shutdown.

c. The equilibrium (at power) value of samarium is independent of power level. The peak value of samarium following a shutdown is dependent upon power level prior to shutdown.

d. The equilibrium (at power) value of samarium is independent of power level. The peak value of samarium following a shutdown is independent of power level prior to shutdown.

QUESTION 1.27 (1.00)

Resctivity is defined as which of the following?

a. The ratio of the number of neutrons at some point in this generation to the number of neutrons at the same point in the previous generation.

b. The fractional change in neutron population per generation.

c. The factor by which neutron population changes per generation.

d. The rate of change of reactor power in neutrons per second.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.28 (1.00)

The following are the boiling phases associated with nucleate boiling and departure from nucleate boiling.

1) Transition Boiling
2) Bulk Boiling
3) Film Boiling
4) Sub-cooled Nucleate Boiling

Which of the following is the order in which they would occur in a channel with normal flow and high heat flux?

a. 2, 4, 3, 1

b. 2, 4, 1, 3

c. 4, 2, 3, 1

d. 4, 2, 1, 3

QUESTION 1.29 (1.00)

Which of the following statements best characterizes Natural Circulation?

a. It needs a pump to get started.

b. The elevation of the heat source must be above that of the heat sink.

c. The driving force is a difference in density.

d. Heat transfer is more efficient if steam is mixed with water.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.30 (1.00)

The reactor startup procedure requires that the critical rod position be taken at 10E-8 amps on the intermediate range. If, during a xenon free reactor startup at MOL, the operator 'overshoot' 10E-8 amps and instead leveled off at 10E-7 amps, which of the following statements is correct?

a. At 10E-7 amps, there are little or no effects from nuclear heat; but, since the reactor is a decade higher in power, the critical rod position would be higher than at 10E-6 amps.

b. At 10E-7 amps, there are little or no effects from nuclear heat; therefore, the critical rod position should be the same as at 10E-8 amps.

c. At 10E-7 amps, there are substanial effects from nuclear heat; therefore, the critical rod position will be higher than at 10E-8 amps.

d. At 10E-7 amps, nuclear heat, xenon, and the decade higher in power level will result in a higher critical rod position than at 10E-8 amps.

QUESTION 1.31 (1.00)

The reactor trips from full power, equilibrium xenon conditions. Six hours later the reactor is brought critical at 10E-8 amps on the intermediate range. If power level is maintained at 10E-8 amps which of the following statements concerning rod motion requirements for the next two hours is correct?

a. Rods will have to be withdrawn since xenon will closely follow its normal build-in rate following a trip.

b. Rods will have to be inserted since xenon will closely follow its normal decay rate following a trip.

c. Rods will have to be rapidly inserted since the critical reactor will cause a high rate of burnout.

d. Rods will have to be rapidly withdrawn since the critical, reactor will cause a higher than normal rate of build-in.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.32 (1.00)

The Technical Specifications allow operations for a 2-hour time period with a Quadrant Power Tilt Ratio (QPTR) of greater than 1.02. Which of the following is the reason for allowing these operations?

a. To allow time for corrective action in the event of xenon redistribution following power changes.

b. To allow time for correction of a dropped or misaligned control rod.

c. To allow time for boron concentration changes to restore the QPTR to less than 1.02.

d. To allow time for correction of RCS flow imbalances.

QUESTION 1.33 (2.00)

Indicate how (INCREASE, DECREASE, or REMAIN THE SAME) an increase in moderator temperature will affect the following parameters.

a. Resonance Escape Probability (0.5)

b. Thermal Utilization Factor (0.5)

c. Fast Non-Leakage Probability (0.5)

d. Fast Fission Factor (0.5)

QUESTION 1.34 (1.00)

Which of the following causes differential boron worth (pcm/ppm) to decrease (become less negative)?

a. Boron concentration decreases

b. Moderator temperature decreases

c. Fission product poison buildup

d. RCS pressure increases

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.35 (1.00)

Which of the following describes the flux shifts that occur to the critical low power flux profile over core life? Assume a cycle 1 core loading.

a. Flux shifts upward and towards the edges of the core.

b. Flux shifts upward and towards the center of the core.

c. Flux shifts downward and towards the edges of the core.

d. Flux shifts downward and towards the center of the core.

QUESTION 1.36 (1.00)

Which of the following is NOT one of the conditions necessary for brittle fracture to occur?

a. Plastic deformation at or below the yield point.

b. Temperature at or below the NDTT.

c. Nominal stress level.

d. Flaw such as a crack present.

(***** END OF CATEGORY 01 *****)

QUESTION 2.01 (1.00)

Which of the following describes the Service Water System automatic actions on a single unit SI?

a. All SW pumps start, that unit's spray header isolation MOVs receive an 'open' signal

b. Only the SW pumps supplied from that unit's emergency buses start, that unit's spray header isolation MOVs receive an 'open' signal

c. All SW pumps start, all spray header isolation MOVs receive an 'open' signal.

d. The SW pumps supplied from that unit's emergency buses start, all spray header isolation MOVs receive an 'open' signal.

QUESTION 2.02 (2.50)

Match the RCS penetrations in Column A with the appropriate RCS loop segment listed in Column B. (answers may be used more than once)

Column A	Column B
a. Normal Letdown	1) Loop A cold leg
b. Pzr Surge Line	2) Loop A hot leg
3 ~~c. Sample System~~ dele	3) Loop A intermediate leg
d. PZR Spray Line	4) Loop B intermediate leg
e. RHR Suction	5) Loop B hot leg
	6) Loop C cold leg
	7) Loop C hot leg

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.03 (1.00)

Which of the following statements concerning the operation of the letdown isolation valves (LCV-1460A and B) is correct?

a. Neither of the letdown isolation valves can be opened if the containment letdown isolation valve (TV 1204) is shut.

b. Shutting all orifice isolation valves will automatically shut the letdown isolation valves

c. All orifice isolation valves must be shut in order to open the letdown isolation valves.

d. High temperature on the letdown outlet of the regenerative heat exchanger will automatically shut the letdown isolation valves.

QUESTION 2.04 (1.00)

According to 10CFR50.46, which of the following is NOT a design criteria of the Emergency Core Cooling System subsystems.

a. The calculated peak centerline temperature shall not exceed 2000 degrees F.

b. The maximum cladding oxidation shall not exceed 17% of the total cladding thickness.

c. The calculated total amount of hydrogen generated from the cladding reaction with water shall not exceed 1% of the amount that would be generated if all cladding surrounding the fuel reacted.

d. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.05 (1.00)

Which of the following does the operator MANUALLY adjust to reduce the RCS temperature when the RHR system is in service for a normal plant cooldown?

a. Throttle open CCW from RHR Heat Exchanger outlet isolation valve.

b. Throttle open RHR Heat Exchanger outlet isolation valve.

c. Throttle closed RHR Heat Exchanger bypass valve.

d. Throttle closed RHR miniflow recirculation valve.

QUESTION 2.06 (1.00)

Delete - [illegible]

Which statement below regarding pressurizer spray is correct?

a. The position of the auxiliary spray valve can be adjusted from fully shut through the fully open position by the operator at the main control board.

b. Each of the normal spray valves has a 1 gpm bypass line to prevent cooldown of the spray lines.

c. The maximum spray flow is designed to prevent the PORVs from opening on a 10% step decrease in power.

d. Auxiliary spray utilizes a separate spray nozzle due to the large temperature differential that exists when auxiliary spray is used.

QUESTION 2.07 (1.00)

Which location below is the discharge point for the pressurizer head vent?

a. Containment fuel canal

b. Upper region of containment below quench spray rings

c. Pressurizer Relief Tank

d. Suction side of containment Hydrogen Recombiners

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.08 (1.50)

Indicate whether the following statements regarding RCP seals are TRUE or FALSE.

a) The floating ring seal, located between the pump radial bearing and the #1 seal, will limit leakage to 50 gpm if the #1 seal fails. (.5)

b) #3 seal is designed to withstand full RCS pressure. (.5)

c) Seal water injection from CVCS enters the RCP between the seal package and the pump radial bearing. (.5)

QUESTION 2.09 (1.50)

The RCS Loop Isolation Valves have two penetrations used for internal pressurization to minimize leakage during maintenance operations. List the two ways in which the valve internals can be pressurized and which one is the preferred method.

QUESTION 2.10 (1.50)

Indicate whether the following statements regarding the ECCS are TRUE or FALSE.

a) The discharge of ONLY UNIT 1's LHSI pumps can be lined up to supply or receive water from the discharge of the Outside RS pumps. (.5)

b) All three HHSI Pumps will receive a start signal on an SI. (.5)

c) The throttle valves located in each of the HHSI Cold Leg Injection lines will ensure a rupture in a loop will not cause all the charging flow to spill out that break. (.5)

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.11 (1.00)

Which of the following describes the method of NaOH solution addition to the Quench Spray System?

a. An eductor utilizing QS pump discharge draws NaOH solution from the Chemical Addition Tank (CAT) into the QS pump output.

b. Gravity feed from the CAT to the RWST near where the QS pumps take a suction.

c. Gravity feed from the CAT to the area between the QS pump inlet isolation valve and the suction side of the pump.

d. The CAT pump discharges the contents of the tank into the QS pump suction with a pre-determined flow rate set by a manual throttle valve.

QUESTION 2.12 (1.00)

Listed below are valves associated with the Recirculation Spray (RS) System. Indicate whether the valves listed are NORMALLY OPEN or CLOSED.

a) MOV-SW-102A and B (Service Water supply header X-connects)

b) MOV-SW-105A and B (Service Water B return header isolation valves)

c) MOV-RS-101A (Casing Cooling Pump A FIRST discharge valve)

d) MOV-RS-155B (Outside RS Pump B suction valve)

QUESTION 2.13 (1.00)

Which of the following signals, acting independently, will automatically CLOSE the Main Feedwater Regulating Valves (MOV-FW-1154A, B and C)?

a. Hi-Hi S/G level in any S/G (2 out of 3 detectors)

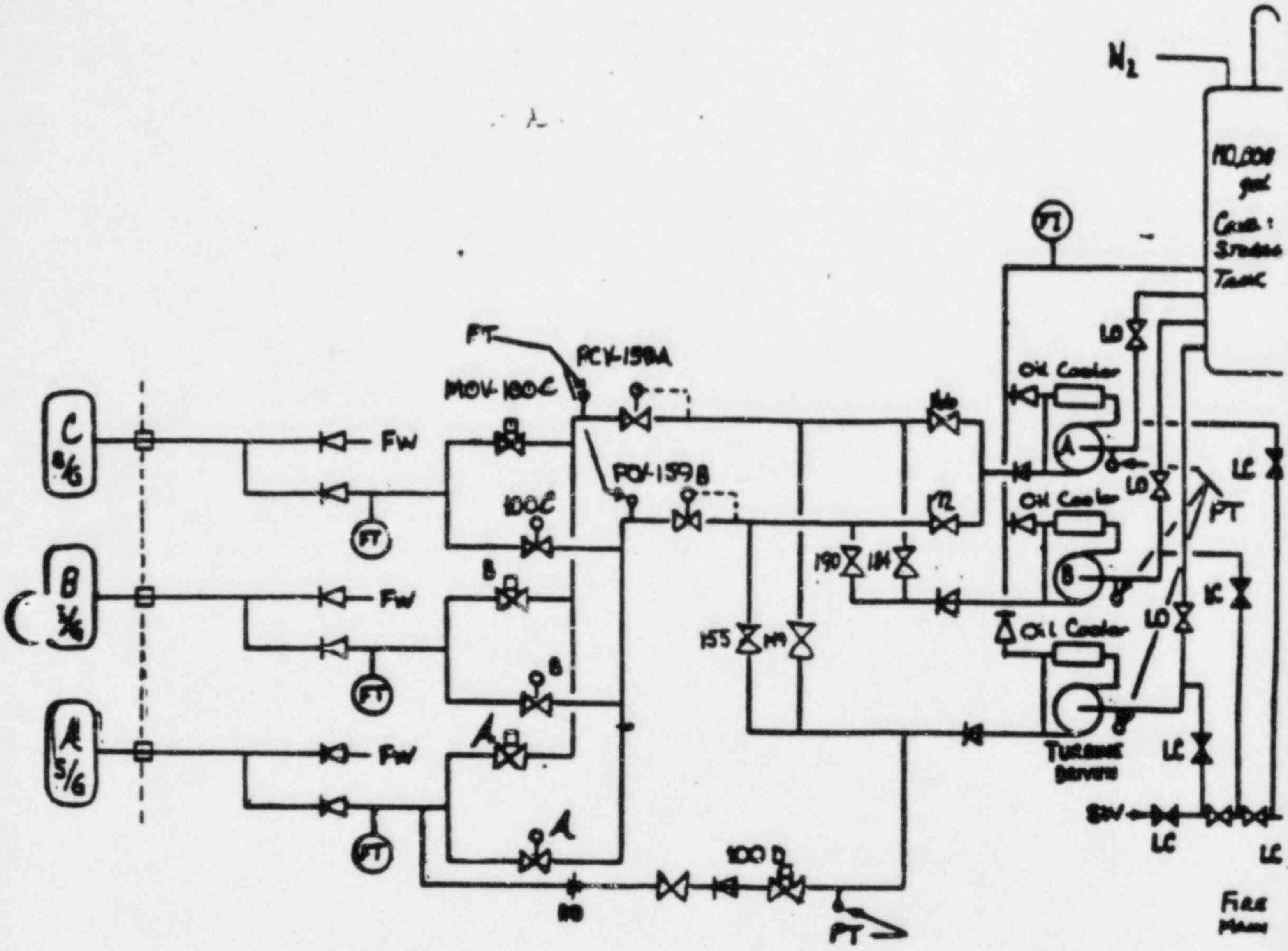
b. Low Tavg (2 out of 3)

c. Reactor Trip

d. Phase B Isolation

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

AUXILIARY FEEDWATER SYSTEM



ATTACHMENT FOR QUESTION 2.14

QUESTION 2.14 (2.00)

Attached is a drawing of the Auxiliary Feedwater System. List the valves downstream of the AFW pumps which are closed in a normal Mode 1 lineup.

QUESTION 2.15 (1.50)

The Fire Protection Water Main is an alternate source of water for the AFW System. List the 3 independent sources of water to this Fire Main System.

QUESTION 2.16 (1.00)

The steam driven AFW pump is a 6-stage centrifugal pump, nearly identical to the 6-stage centrifugal motor driven AFW pumps. Yet the steam driven AFW pump is rated at a much higher capacity. Why does this difference exist?

QUESTION 2.17 (1.50)

Fill in the blanks in the following statements about the Reactor Vessel Head Assembly.

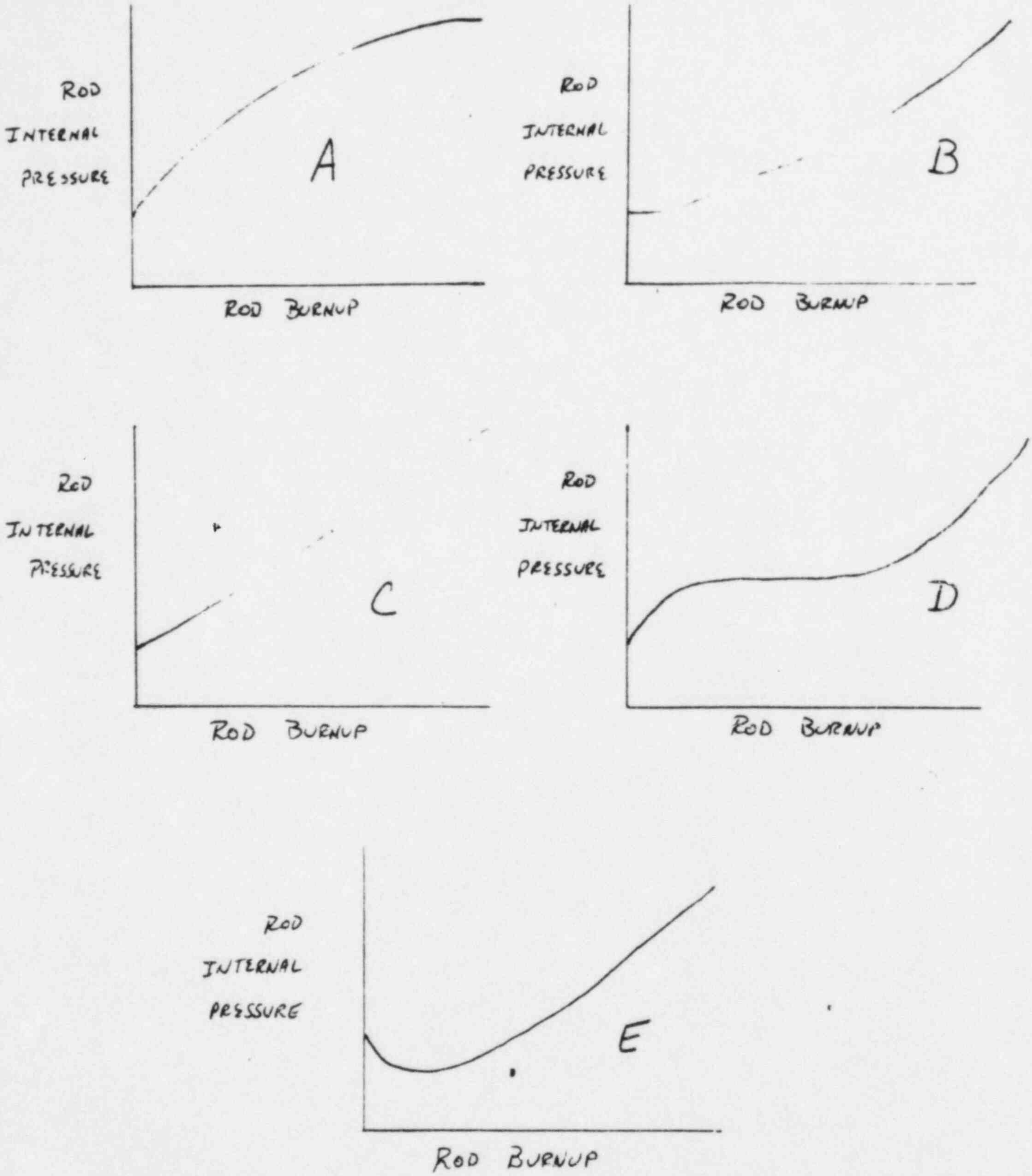
a) There are 2 ____ ____ between the upper head flange and the vessel flange that form an _______ and ________ _____. (1.0)

b) The center partial length control rod penetration is used for the ______ ______ ______ ______ System. (0.5)

QUESTION 2.18 (2.00)

List the 4 flow paths within the reactor vessel which BYPASS the fuel rods.

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)



ATTACHMENT FOR QUESTION 2.19

dele

QUESTION 2.19 (1.00)

Which graph on the attached page represents the change in the internal pressure of a fuel rod versus fuel burnup?

a. A
b. B
c. C
d. D
e. E

QUESTION 2.20 (1.00)

The pressurizer safety valves are designed to limit RCS pressure to within the safety limit of 2735 psig following a complete loss of turbine generator load while at rated thermal power. Which of the following mitigating events is assumed to have occurred in the analysis for this design basis?

a. Reactor trip due to turbine loss of load

b. PORV operation

c. Hi PZR pressure trip.

d. Both spray valves actuate.

e. Relief valves on all 3 S/Gs actuate.

QUESTION 2.21 (1.00)

Which of the following correctly describes the distribution of J-tubes around the Feedwater Distribution Ring in the S/Gs?

a. They are evenly distributed.

b. There is no pre-determined distribution pattern.

c. More J-tubes are located at the outlet chamber side of the downcomer than the inlet chamber side.

d. More J-tubes are located at the inlet chamber side of the downcomer than the outlet chamber side.

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.22 (1.50)

What are the three groups of RCS piping penetrations made BELOW the horizontal centerline of the piping? (penetrations with a similar purpose are to be treated as one group)

QUESTION 2.23 (2.00)

Fill in the blanks in the paragraph below regarding CVCS demineralizers.

The CVCS contains 5 demineralizers consisting of _____ mixed bed, _____ cation bed and _____ deborating deminerlizers. The mixed bed's resin will provide a minimum decontamination factor of 10 for all fission products except ________, ________ and _______. The cation demineralizer's purpose is to control the concentration of ________ and the buildup of ________.

QUESTION 2.24 (1.50)

Indicate whether the following CVCS valves will FAIL OPEN, CLOSED or AS IS on a Loss of Instrument Air.

a) Low Pressure Letdown Valve (PCV-1145)
b) Charging Flow Control Valve (FCV-1122)
c) Letdown Orifice Isolation Valves (HCV-1200A, B and C)

QUESTION 2.25 (1.00)

Unit 2's Moisture Separator Reheaters (MSRs) have a hexagonal tube bundle which has almost twice as many tubes as Unit 1's MSRs, but the Unit 2 MSRs are wider and shorter than Unit 1's. What effect, if any, does this difference in MSR construction have on the Differential Pressure across Unit 2's MSRs ~~and the plant's thermal power capabilities~~? (Indicate any changes by direction and approximate magnitude.)

dele 5

QUESTION 2.26 (1.00)

List the location of all the Radiation Monitors associated with the Service Water System. (If there are monitors on redundant components list as one location)

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.27 (1.00)

Which of the following is a load using Service Water as a BACKUP source of heat removal?

a. Service Air Compressors

b. Instrument Air Compressors

c. Control Room Air Conditioning Units

d. Containment Air Recirc Coolers

QUESTION 2.28 (1.00)

Which of the following describes the NORMAL operating range of containment pressure?

a. 1-4 psia

b. 5-8 psia

c. 9-12 psia

d. 13 psia-2 psig

e. 3 psig-6 psig

QUESTION 2.29 (1.00)

Which of the following functions is NOT performed by the hydraulic power system associated with the Containment Personnel Air Lock?

a. Operation of the Equalizing Valve

b. Door Locking Ring Rotation

c. Interior Door Operation

d. Exterior Door Operation

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.30 (1.50)

Indicate whether the following containment isolation valve combinations are ALLOWED or NOT.

a) One LOCKED CLOSED isolation valve INSIDE and a simple check valve used as an AUTOMATIC isolation valve OUTSIDE

b) An AUTOMATIC isolation valve INSIDE (not a simple check valve) and a LOCKED CLOSED isolation valve OUTSIDE containment.

c) AUTOMATIC isolation valves BOTH INSIDE and OUTSIDE containment with the INSIDE valve a simple check valve.

QUESTION 2.31 (1.50)

Match the Containment Atmospheric Cleanup System component in Column A with the correct location in Column B.

COLUMN A	COLUMN B
a) H2A-HC-200 H2 Analyzer	1) Aux Bldg near Drumming Area
b) H2A-HC-201 Remote Analyzer Panel	2) Below Rod Drive Room
c) 1-HC-HC-1 H2 Recombiner	3) Aux Bldg near Demin Alley
	4) Unit 2 Instrument Rack Room
	5) By High Rad Sampling Panels

(***** END OF CATEGORY 02 *****)

QUESTION 3.01 (1.00)

Which statement below regarding the pressurizer pressure control and protective system is NOT correct?

a. The master pressure controller (MPC) provides the control signal for only one of the PORVs

b. There is a lead/lag compensation circuit for pressure inputs to the low pressure reactor trip that varies the trip setpoint with the rate of pressure decrease

c. The two pressurizer spray valves are controlled by separate transmitters

d. To block SI actuation on a normal plant depressurization, the operator must operate TWO block switches to prevent inadvertant ECCS actuation

QUESTION 3.02 (1.00)

PT-402 and 403 measure RCS pressure upstream of RHR suction isolation valves MOV 1700 and 1701. Which of the following statements concerning the effect(s) of PT-402 failing upscale (high) is correct?

a. If both MOV 1700 and MOV 1701 are open they will shut.

b. If open, MOV 1700 will shut and if closed, MOV 1701 will not be able to be opened.

c. If both MOV 1700 and MOV 1701 are shut, they will not be able to be opened.

d. If open, MOV 1700 will shut, but MOV 1701 can be positioned as desired by the operator.

e. MOV 1700 can be positioned as desired by the operator, but MOV 1701 will not be able to be opened if it is shut.

QUESTION 3.03 (1.00)

List the TWO conditions that will provide signals to automatically open the ECCS accumulator discharge valves (1865A, B, and C).

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.04 (1.00)

Which of the following is NOT an input into the OT Delta T trip point calculator?

a. RCS Flow

b. RCS Pressure

c. Tavg

d. AFD

QUESTION 3.05 (1.00)

With the pressurizer level control selector switch in position I/II, a failure causes the following plant events. (Assume no operator actions taken.)

1. Charging flow reduced to minimum
2. Pressurizer level decreases
3. Letdown secured and heaters off
4. Level increases until high level trip

Which of the following failures occurred?

a. Level channel I failed high

b. Level channel I failed low

c. Level channel II failed high

d. Level channel II failed low

QUESTION 3.06 (1.00)

Which of the Reactor Trips listed below is taken credit for in accident analysis?

a. Over Power Delta T

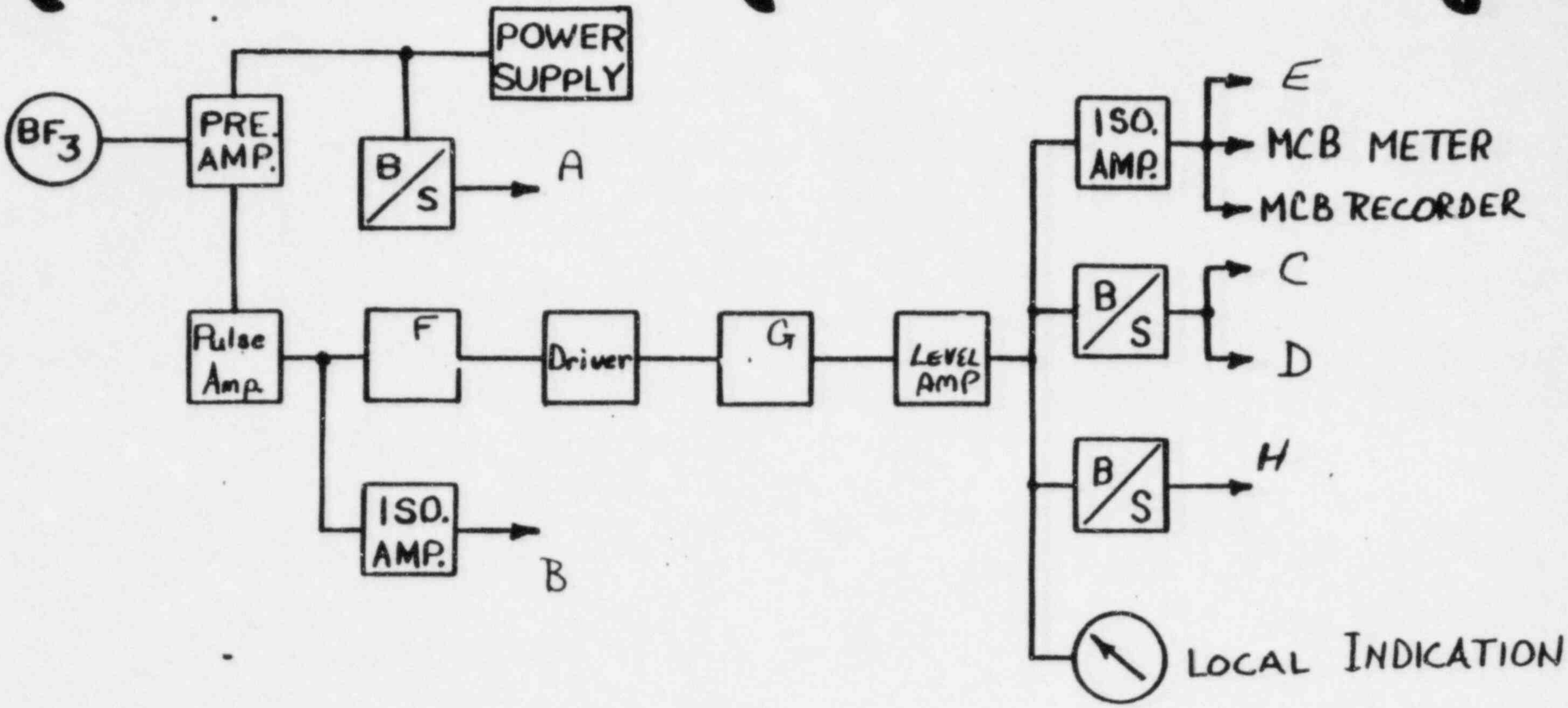
b. Power Range High Neutron Flux-Low Set Point

c. Steam Flow > Feed Flow, coincident with Low S/G Level

d. Pressurizer High Water Level

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

ATTACHMENT FOR QUESTION 3.09



SOURCE RANGE

N31 & N32

QUESTION 3.07 (1.00)

List the two conditions which necessitate the use of 2/4 reactor trip protection logic vice 2/3 logic.

QUESTION 3.08 (2.00)

For the following protection/control circuits, indicate whether the listed permissive function occurs as a result of MANUAL or AUTOMATIC Action.

a) Bypassing the Power Range 'Neutron Flux-Low' setpoint above P-10.

b) Reenergizing the Source Range instrument below the P-6 setpoint.

c) Bypassing S/G 'Lo-Lo Water Level Trip' when RCS Loop Stop Valves are closed.

d) Blocking High Steam Flow SI when Tavg < 543 degrees F (P-12).

QUESTION 3.09 (2.00)

Attached is a block diagram of the Source Range instrument. Indicate on your answer sheet the missing components/outputs labelled A-H on the diagram.

QUESTION 3.10 (1.50)

Answer the following questions regarding the Excore NIS TRUE or FALSE.

a) An Intermediate Range detector that is overcompensated could result in the P-6 setpoint never being reached on a shutdown.

b) Placing the Source Range instrument in 'Bypass' at the instrument drawer will prevent a reactor trip signal if either the Instrument or the Control Power Fuses are pulled.

c) Neither the Source Range or the Intermediate Range 'Hi Flux' trips are taken credit for in accident analysis.

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.11 (1.00)

Which of the following statements describes the two Delta Ts measured on the Core Cooling Monitor when the Loop 1 button is depressed?

a. (Loop A Th-Loop A Tc) and (Highest core thermocouple-Loop A Tc)

b. (Loop A Th-Loop A Tc) and (Highest core thermocouple-Loop A Th)

c. (Average core thermocouple-Loop A Th) and (Average core thermocouple-Loop A Tc)

d. (Loop A Th-Highest core thrmocouple) and (Highest core thermocouple-Loop A Tc)

e. (Loop A Th-Average core thermocouple) and (Loop A Th-Loop A Tc)

QUESTION 3.12 (2.00)

Indicate whether the following situations will cause the Steam Dump System to ARM ONLY, ARM AND ACTUATE or have NO EFFECT. (Assume all switches are in normal positions unless otherwise indicated)

a) 60% power, 3 %/min load reject, Tavg>Tref by 8 degrees F, Mode Selector Switch in Tavg.

b) Turbine trip from 25% power, Tavg=545 degrees F, Mode Selector Switch in Tavg.

c) Hot Zero Power, Tavg=550 degrees F, Mode Selector Switch in STM PRESS, 1020 psig setpoint selected.

d) 40% power, 5%/min load increase, Pimp CH-III fails low, Tavg=560 degrees F, Mode Selector Switch in Tavg

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.13 (1.00)

Which of the following describes the 3 inputs used in the S/G Bypass Water Level Control System?

a. S/G Level, Steam Flow, Feed Flow

b. S/G Level, Steam Flow, 1st Stage Turbine Pressure

c. Reactor Power, Feed Flow, 1st Stage Turbine Pressure

d. Reactor Power, Steam Flow, Feed Flow

e. Reactor Power, S/G Level, 1st Stage Turbine Pressure

QUESTION 3.14 (1.50)

With the reactor at 50% Power, S/G A's steam pressure input to the Steam Generator Water Level Control System fails high. Explain how this will affect the Level Control System for S/G A, and why this action occurs.

QUESTION 3.15 (1.00)

Which of the following describes the 3 Reactor Vessel Water Level ranges measured by RVLIS?

a. Lower, Full, Dynamic

b. Lower, Full, Upper

c. Dynamic, Full, Upper

d. Dynamic, Lower, Upper

QUESTION 3.16 (1.00)

Describe the reasons why the D/P cells used for RVLIS are located outside of containment.

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.17 (1.00)

Which of the following rod withdrawal blocking permissives is ONLY effective when Rod Control is in AUTO?

a. Intermediate Range Overpower (C-1)

b. Power Range Overpower (C-2)

c. Overtemperature Delta T (C-3)

d. Turbine Load Interlock (C-5)

QUESTION 3.18 (2.00)

Match the function in Column A with the appropriate component in the Rod Control System listed in Column B.

Column A	Column B
a) Minimizes rod motion at higher power levels to limit transients	1) Bank Overlap
	2) Pulse Amplifier
	3) Non-Linear Gain Unit
b) Indicates to power cabinets how to move the rods	4) Pulser
	5) Master Cycler
	6) Slave Cycler
c) Automatically selects proper control bank	7) Variable Gain Unit
d) Selects which group within a shutdown bank is to be moved	

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.19 (1.00)

Which of the following describes the operation of the search coil associated with Individual Rod Position Indication (IRPI)?

a. Located about 1 foot from the Detector Bottom position, it allows the operator to check rods on the bottom.

b. Located about 6 inches from the Detector Bottom position, it provides input to the Rod Bottom Light circuitry.

c. It runs the length of the rod housing and has a variable tap that allows the operator to approximate rod position when IRPI is in doubt.

d. No such coil is associated with IRPI.

QUESTION 3.20 (1.00)

Which statement below is correct regarding implementation of the Reduced Pressure Operations Protective System as the plant cools down?

a. PORV reduced setpoints are automatically inserted when Tc<320 degrees F and a second pair of PORV setpoints must be manually 'keyed' in when Tc<140 degrees F.

b. PORV reduced setpoints are automatically inserted when Tc<320 degrees F and also when Tc<140 degrees F.

c. PORV reduced setpoints must be manually 'keyed' in when Tc<320 degrees F and again when Tc<140 degrees F.

d. PORV reduced setpoints are manually 'keyed' in when Tc<320 degrees F and are automatically inserted when Tc<140 degrees F.

QUESTION 3.21 (1.00)

State the reason for having backup heaters automatically energize when actual pressurizer level > reference level by 5% or more.

QUESTION 3.22 (1.50)

Indicate whether the following controls for 1H EDG are transferred to/from the 1H diesel room using the 'Control Room Emergency' (CRE) switch or the 'Diesel Control Isolation' (DCI) switch.

a) Control and synchronization for the 4.16 KV Emergency Bus emergency supply breaker.

b) Alternate governor controls by means of the Droop/Isochronous Selector switch.

c) Diesel Start/Stop control

QUESTION 3.23 (1.50)

Describe the improvements to the flexibility and reliability of the electrical distrubution system after a Unit 1 trip due to the installation of the Unit 1 main generator breaker G12.

QUESTION 3.24 (2.00)

Answer the following questions regarding a DEGRADED voltage condition on an Emergency 4160 VAC Bus.

a) A degraded voltage condition occurs when ____ out of ____ relays detect <_____% nominal voltage and initiates a ____ timer. (Fill in the blanks) (.75)

b) Describe the sequence of events from the timing out of the timer until all required loads are reenergized on the Emergency Bus. (Be sure to reference any loads NOT shed) (1.25)

QUESTION 3.25 (1.00)

Which of the following conditions is NOT required for automatic swapover of the LHSI pumps to the Recirculation Mode following an SI?

a. RWST Lo-Lo Level

b. A LHSI pump recirc isolation MOV closed for each pump

c. SI signal present

d. SI Recirc Mode signal present

QUESTION 3.26 (1.00)

Following a Quench Spray system actuation, what 2 conditions must be met in order to shut the suction valve for a Quench Spray pump?

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.27 (1.00)

Which of the following describes the purpose of the time delay (195 sec) in starting the Inside Recirculation Spray (IRS) pumps on a CDA signal?

a. Prevents overloading of the emergency buses as loads are sequentially energized during accident conditions.

b. Allows time for the containment sumps to collect sufficient fluid to provide NPSH for the pumps.

c. Enhances core cooling by increasing reflood rate after a LOCA as the pressure drop between core exit and the break is reduced with a higher containment pressure.

d. Allows time for the fluid collecting in the containment sumps to cool, to avoid flashing in the RS heat exchangers as the fluid is cooled by Service Water.

QUESTION 3.28 (1.00)

Which of the following describes the normal path of power to Unit 1's Charging Pump 1C?

a. RSS A--4160 VAC Xfer Bus D--4160 VAC Emerg Bus J

b. RSS C--4160 VAC Xfer Bus F--4160 VAC Emerg Bus J

c. RSS A--4160 VAC Xfer Bus D--4160 VAC Emerg Bus H

d. RSS C--4160 VAC Xfer Bus F--4160 VAC Emerg Bus H

e. RSS A--4160 VAC Xfer Bus F--4160 VAC Emerg Bus H

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.29 (2.00)

Indicate whether the following statements regarding Containment Isolation are TRUE of FALSE.

a) The Hi-Hi containment pressure input to the CDA logic matrix must ENERGIZE relays to initiate a CDA signal.

b) Phase B isolation can be manually initiated by turning one of two switches on the back of the console in the control room.

c) If Phase A is reset following signal initiation, without having the initiating signal disappear, than the phase A isolation signal will reoccur when the reset buttons are released.

d) S/G Blowdown is isolated by both Phase A and Phase B isolation signals.

QUESTION 3.30 (1.00)

List all the inputs to the electrical starting ci:cuit that must be satisfied in order to start a Reactor Coolant Pump.

QUESTION 3.31 (1.00)

Which of the following conditions will NOT result in 'C' Charging Pump receiving a 'Start-Permissive Trip' signal when it is powered from the H bus? (all breakers listed below are open unless stated as closed)

a. Breaker 15J7 racked in

b. Breaker 15H6 racked in

c. Breaker 15J6 racked out

d. Breaker 15J7 closed

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.32 (1.00)

Which type of **sensor** below is used to indicate when pressurizer spray flow is too low?

a. Temperature

b. Pressure

c. Flow

d. Differential Pressure

(***** END OF CATEGORY 03 *****)

QUESTION 4.01 (1.00)

Given the following procedural steps, select the choice that provides the proper sequence for paralleling a diesel generator to a live bus.

1. Turn the synchronizer switch on.
2. Start the emergency diesel generator.
3. Turn off the synchronizer switch.
4. Operate the governor control switch so the synchroscope moves slowly in the fast direction.
5. Close the generator output breaker.
6. Wait for the synchroscope needle to pass the 11 o'clock position.
7. Operate the exciter control switch to match incoming voltage to running voltage.

a. 1,2,4,6,5,7,3.

b. 2,1,7,4,6,5,3.

c. 2,1,4,6,5,3,7.

d. 1,2,7,6,5,4,3.

QUESTION 4.02 (1.00)

When placing the RCS in a solid water condition per OP-3.4, the criterion used to verify the RCS is solid is

a. a high pressure spike occuring on the pressurizer pressure instruments.

b. when spraying the pressurizer no longer will decrease RCS pressure.

c. when indication of flow to the PRT through the pressurizer PORVs is verified.

d. when an increase in letdown flow has been verified.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.03 (1.00)

Which of the following is the preferred method of providing a means of decay heat removal during a refueling outage with the reactor vessel head removed if all RHR pumps are lost?

a. Using natural circulation by feeding the SGs with AFW and bleeding them using SG blowdown.

b. By lining up the LHSI pumps to recirculate flow through the RWST and the reactor vessel.

c. By lining up the refueling purification system to recirculate water from the reactor cavity to the spent fuel pit using the spent fuel coolers for cooldown.

d. By feeding the RCS using a LHSI pump and bleeding it from a SG primary manway.

QUESTION 4.04 (1.00)

Prior to transferring rod control from MANUAL to AUTO during a reactor startup, Tave and Tref should be within

a. 1 degree F.

b. 3.5 degrees F.

c. 5 degrees F.

d. 10 degrees F.

QUESTION 4.05 (1.00)

List four of the critical conditions required to be recorded during a startup when 1 X E-8 amps is attained.

QUESTION 4.06 (.50)

For a reactor startup, it is required by the procedure to go critical while the power indication is still in the source range. TRUE or FALSE?

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.07 (1.00)

If criticality is not acheived before the upper limit of the ECP is reached during a reactor startup, the operator should

a. insert Bank D control rods to the bottom.

b. stop control bank withdrawal and approach criticality via boron dilution.

c. proceed with the startup utilizing an Inverse Count Rate Plot from the time the ECP upper limit is reached until the time criticality is reached.

d. proceed with the startup utilizing a pull-and-wait approach to criticality.

QUESTION 4.08 (1.00)

Control rod cooling air is required to be operating

a. only if the control rod mechanism is energized when the plant temperature is above 350 F.

b. whenever the plant temperature is above 350 F, regardless of the rod mechanism status.

c. during any plant cooldown evolution below 350 F.

d. whenever the individual rod position indication system or any rod drive mechanism is energized.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.09 (1.00)

The boron concentration is equalized between the pressurizer and the rest of the reactor coolant system by

a. charging pure water or borated water to the PZR through the aux. spray line as required.

b. charging a blended flow equal to the Reactor Coolant System boron concentration into the Reactor Coolant System to force more of the RCS inventory up the surge line and into the PZR.

c. taking MANUAL control of the Pressurizer Level Control System and then increasing and decreasing the pressurizer level as required in order to get an exchange between the RCS and PZR water inventory.

d. operation to the pressurizer heaters and spray valves as required.

QUESTION 4.10 (1.00)

The administrative temperature limit between the pressurizer and spray delta T is

a. 150 F.

b. 200 F.

c. 260 F.

d. 320 F.

QUESTION 4.11 (1.00)

Which of the following is NOT required operator action upon loss of power range channel N-41 during Mode 1 operations?

a. Place the test switch for the OT delta T to the TEST position for the failed channel.

b. Place or verify the control rods are in MANUAL control.

c. Place the test switch for the OP delta T to the TEST position for the failed channel.

d. Remove the 'Control Power' fuses for N-41 at the power range drawers.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.12 (1.00)

Which of the following is an INAPPROPRIATE operator action if, during refueling, the containment gas and particulate radiation monitors alarm?

a. Stop the containment purge supply and exhaust fans.

b. Close the purge and exhaust fan MOVs.

c. Stop the containment air recirculation fans.

d. Start the containment iodine filtration fans.

QUESTION 4.13 (1.00)

The trip bistables of a failed power range detector are placed in the trip condition by

a. placing the applicable bistable test switch in the 'Test' position in the Reactor Protection Cabinet.

b. removing the applicable control and instrument power fuses on the power range drawers.

c. placing the applicable Power Mismatch Bypass switch to the failed position at the Miscellaneous Control and Indication Panel.

d. placing the applicable Comparator Channel Defeat switch to the failed channel position at the Detector Current Comparator Panel.

QUESTION 4.14 (1.25)

List five surveillance items that must be done by operators to ensure Tech Spec requirements are met in the event of a loss of both units PRODAC-250 Computers.

QUESTION 4.15 (.50)

A rod control logic cabinet internal failure can be verified in the control room. TRUE or FALSE?

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.16 (1.00)

If instrument air is lost outside the containment and the plant is solid, what is your required immediate actions?

QUESTION 4.17 (1.00)

What is the required operator actions if both source range instruments fail when >P-6 and <P-10 during a power increase to 100% power?

QUESTION 4.18 (1.00)

List the immediate operator actions for one dropped rod at 75% power.

QUESTION 4.19 (1.00)

List the immediate operator actions for a rod insertion limit low-low alarm.

QUESTION 4.20 (1.00)

List the immediate operator actions for a failure of the controlling first stage impulse pressure instrument when at 50% power.

QUESTION 4.21 (1.00)

List the immediate operator actions for an uncontrollable increase in main condenser pressure with the reactor at 50% power.

QUESTION 4.22 (1.00)

What operator actions are required upon evacuating the control room if the reactor could not be tripped before exiting the control room?

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.23 (1.00)

List the immediate operator actions for an Anticipated Transient Without Trip.

QUESTION 4.24 (3.25)

List the immediate operator actions for the Reactor Trip or Safety Injection procedure.

QUESTION 4.25 (1.00)

Which of the following indications require SI reinitiation following a spurious SI that has been secured?

a. RCS pressure at 1950 psig.

b. RCS subcooling at 35 degrees F.

c. Pressurizer level at 17%.

d. All steam generator levels at 15%.

QUESTION 4.26 (1.00)

During a natural circulation cooldown, the preferred method of reactor coolant system depressurization is by

a. opening the normal spray valve.

b. opening the auxiliary spray valve.

c. opening the pressurizer PORVs.

d. opening the reactor vessel vent valves.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.27 (1.00)

A hydrogen bubble formed in the reactor vessel is eliminated by

a. increasing pressurizer temperature above core thermocouple readings.

b. injecting oxygen into the reactor coolant system via the chemical and volume control system.

c. maximizing coolant flow by running all reactor coolant pumps, increasing letdown flow to 120 gpm, and placing the cation bed demineralizer in service in parrallel with the mixed bed demineralizer.

d. venting the reactor vessel head.

QUESTION 4.28 (1.00)

A fire found in the 4160 switchgear is best extinguished with a fire extinguisher marked with the symbol

a. A

b. B

c. C

d. D

QUESTION 4.29 (1.00)

The Gai-Tronics channel reserved for emergencies is

a. channel 1.

b. channel 3.

c. channel 5.

d. channel 7.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.30 (1.00)

What should the operator do about an SI signal if it exists during a loss of all AC power?

QUESTION 4.31 (1.00)

What two conditions constitutes adverse containment conditions?

QUESTION 4.32 (.50)

Will adverse containment conditions cause affected control room indications to indicate higher or lower than actual conditions?

QUESTION 4.33 (1.00)

List the whole body administrative limits per calender quarter that can be achieved (a) without any additional approval and (b) with the highest level of approval.

QUESTION 4.34 (1.00)

Which of the following is a 10 CFR 20 exposure limit?

a. 5 rem/year-whole body.

b. 1 rem/quarter-whole body.

c. 18.75 rem/quarter-hands.

d. 7 rem/quarter-skin of whole body.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.35 (1.00)

If you are in a 100 mRad/hour gamma field for 45 minutes, what is your dose in mREM after 45 minutes?

a. 45

b. 75

c. 450

d. 750

QUESTION 4.36 (1.00)

Which of the following radiation exposures would inflict the greatest biological damage to man?

a. 1 Rem of GAMMA.

b. 1 Rem of ALPHA.

c. 1 Rem of NEUTRON.

d. NONE of the above; they are all equivalent.

QUESTION 4.37 (1.00)

It is required to exit a radiation area and report to HP whenever your 0-200 mr self-reading dosimeter reaches

a. 100 mr.

b. 125 mr.

c. 150 mr.

d. 175 mr.

QUESTION 4.38 (1.00)

List five pieces of information an RWP will tell you that you will need to know in order to work in the applicable restricted area.

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.39 (1.00)

An RWP number is necessary to obtain

a. a dose card.

b. a TLD.

c. a self-reading dosimeter.

d. your current quarterly dose.

(***** END OF CATEGORY 04 *****)
(************* END OF EXAMINATION ***************)

ANSWER 1.01 (1.00)

d

REFERENCE
NUS, Nuclear Energy Training - Reactor Operation
VEGP, Training Text, Vol. 9, Ch. 21
DPC, Fundamentals of Nuclear Reactor Engineering

001/010-K5.13

ANSWER 1.02 (1.00)

b

REFERENCE
NUS, Nuclear Energy Training - Reactor Operation

ANSWER 1.03 (1.00)

b

REFERENCE
NUS, Nuclear Energy Training - Reactor Operation, p. 9.2-5
Westinghouse Reactor Physics, pp. I-5.5, 10, & 11

ANSWER 1.04 (1.00)

a

REFERENCE
NUS, Nuclear Energy Training - Reactor Operation, p. 9.2-5

ANSWER 1.05 (2.00)

a. REMAIN THE SAME (0.5)
b. DECREASE (0.5)
c. INCREASE (0.5)
d. DECREASE (0.5)

REFERENCE
Steam Tables

010/000-K5.02 (2.6/3.0)

ANSWER 1.06 (2.00)

a. TRUE (0.5)
b. TRUE (0.5)
c. FALSE (0.5)
d. TRUE (0.5)

REFERENCE
NUS, Nuclear Energy Training - Plant Performance, p. 5-3.2
General Physics, HT & FF, pp. 155, 319, and 320
VEGP, Question Bank, #752-a and # 336

ANSWER 1.07 (1.00)

a. DISAGREE (0.5)
b. DISAGREE (0.5)

REFERENCE
VEGP, Training Text, Vol. 9, pp. 21-25 & 49 and VEGP Question Bank #60

ANSWER 1.08 (1.00)

a

REFERENCE
VEGP, Training Text, Vol. 9, p. 21-47
Westinghouse Reactor Physics, pp. I-3.17 & 19
DPC, Fundamentals of Nuclear Reactor Engineering, p. 106

001/000-K5.49 (2.9/3.4)

ANSWER 1.09 (1.00)

c

REFERENCE
General Physics, HT & FF, Section 3.2
WBN, HT & FF, p. 13

002/000-K5.01 (3.1/3.4)

ANSWER 1.10 (1.00)

d

REFERENCE
Westinghouse Reactor Physics, pp. I-5.63 - 76
HBR, Reactor Theory, Sessions 38 and 39
DPC, Fundamentals of Nuclear Reactor Engineering, Section VI

001/000-K5.39 (3.5/4.1)

ANSWER 1.11 (1.00)

b

REFERENCE
General Physics, HT & FF, p. 229

002/000-K5.01 (3.1/3.4)

ANSWER 1.12 (2.00)

a. DECREASE (0.5)
b. INCREASE (0.5)
c. INCREASE (0.5)
d. DECREASE (0.5)

REFERENCE
General Physics, HT & FF, pp. 319 - 334

ANSWER 1.13 (1.00)

c

REFERENCE
Steam Tables

ANSWER 1.14 (1.00)

b

REFERENCE
General Physics, Heat Transfer Thermodynamics and Fluid Flow,
pp. 68, 82, & 156

ANSWER 1.15 (1.00)

d

REFERENCE
Westinghouse Reactor Physics, Section I-5, MTC and Power Defect
DPC, Fundamentals of Nuclear Reactor Engineering

002/000-K5.02 (3.3/3.6)

ANSWER 1.16 (1.00)

a

REFERENCE
Westinghouse Reactor Physics, pp. I-2.19 - 21
HBR, Reactor Theory, Session 14, p. 3
DPC, Fundamentals of Nuclear Reactor Engineering, p. 53

001/000-K5.57 (3.0/3.2)

ANSWER 1.17 (1.00)

b

REFERENCE
Westinghouse Reactor Physics, p. I-5.40
HBR, Reactor Theory, Session 36, p. 2

ANSWER 1.18 (1.00)

c

REFERENCE
General Physics, HTFF - Fluid Flow Applications for Systems
and Components

ANSWER 1.19 (1.00)

c

REFERENCE
WBN, HT & FF, p. 19
General Physics, HT&FF, p. 105

ANSWER 1.20 (1.50)

a. FALSE (0.5)
b. FALSE (0.5)
c. TRUE (0.5)

REFERENCE
Westinghouse Reactor Physics, pp. I-3.9 and I-3.4
HBR, Reactor Theory, Sessions 22 and 23
DPC, Fundamentals of Nuclear Reactor Engineering

ANSWER 1.21 (1.00)

c

REFERENCE
HBR, Reactor Theory, Session 42, pp. 3 & 4
DPC, Fundamentals of Nuclear Reactor Engineering

004/000-K5.08 (2.6/3.2)

ANSWER 1.22 (1.00)

b

REFERENCE
NUS, Reactor Theory

ANSWER 1.23 (1.00)

c

REFERENCE
NUS, Reactor Theory

ANSWER 1.24 (1.00)

c

REFERENCE
General Physics, HT & FF, pp. 239 and 240

ANSWER 1.25 (.50)

TRUE

REFERENCE
General Physics, HT & FF, p. 176

002/000-K5.01

ANSWER 1.26 (1.00)

c

REFERENCE
DPC, Fundamentals of Nuclear Reactor Engineering, p. 170

001/000-K5.13 (3.7/4.0)

ANSWER 1.27 (1.00)

b

REFERENCE
DPC, Fundamentals of Nuclear Reactor Engineering, p. 96

001/000-K5.56 (2.8/3.1)

ANSWER 1.28 (1.00)

d

REFERENCE
NUS, Vol 4, pp 3.3-2

ANSWER 1.29 (1.00)

c

REFERENCE
General Physics, HT&FF, pp. 355 - 358

ANSWER 1.30 (1.00)

b

REFERENCE
Westinghouse Reactor Physics, Sects 3 and 5
NUS, Nuclear Energy Training, Units 6 and 8

ANSWER 1.31 (1.00)

a

ANSWER 1.32 (1.00)

b

ANSWER 1.33 (2.00)

a. DECREASE (0.5)
b. INCREASE (0.5)
c. DECREASE (0.5)
d. INCREASE (0.5)

REFERENCE
Westinghouse Nuclear Training Operations, pp. I-2.31 - 36

ANSWER 1.34 (1.00)

c

REFERENCE
Westinghouse Nuclear Training Operations, p. I-5.31

ANSWER 1.35 (1.00)

a

REFERENCE
Westinghouse Nuclear Training Operations, pp. I-5.53 and 54

ANSWER 1.36 (1.00)

a

REFERENCE
DPC, FNRE, pp. 200 and 221

ANSWER 2.01 (1.00)

a

REFERENCE
NA NCRODP 92.2, 'Service Water System'

ANSWER 2.02 (2.50) ~~Deleted~~ .5

a) 1
b) 7
~~c) 5~~ Deleted .5
d) 1,6
e) 2

REFERENCE
Farley SD, 'RCS', Fig 7
NA NCRODP, 'RCS'; 'ESF-ECCS'; 'CVCS'; 'RHR'

ANSWER 2.03 (1.00)

c

REFERENCE
Farley SD, 'CVCS', pp 13
NA NCRODP 88.3, 'CVCS'

ANSWER 2.04 (1.00)

a

REFERENCE
10CFR50.46(b)
FNP, SD, 'ECCS', pp 5
NA NCRODP 91.1, 'ESF-ECCS'

006/050-PWG#35 (4.2/4.3)

ANSWER 2.05 (1.00)

b

REFERENCE
FNP, SD, 'RHR System', pp 17
NA NCRODP 88.2, 'RHR System'; NA OP 14.1

PWG K/A 29 (3.6/3.9)

ANSWER 2.06 (1.00)

~~c~~ Delete - BW

~~REFERENCE~~
~~NA NCRODP 88.1, 'RCS-PZR and Press. Relief'~~

ANSWER 2.07 (1.00)

a

REFERENCE
NA NCRODP 88.1, 'RCS-PZR and Press Relief'

ANSWER 2.08 (1.50)

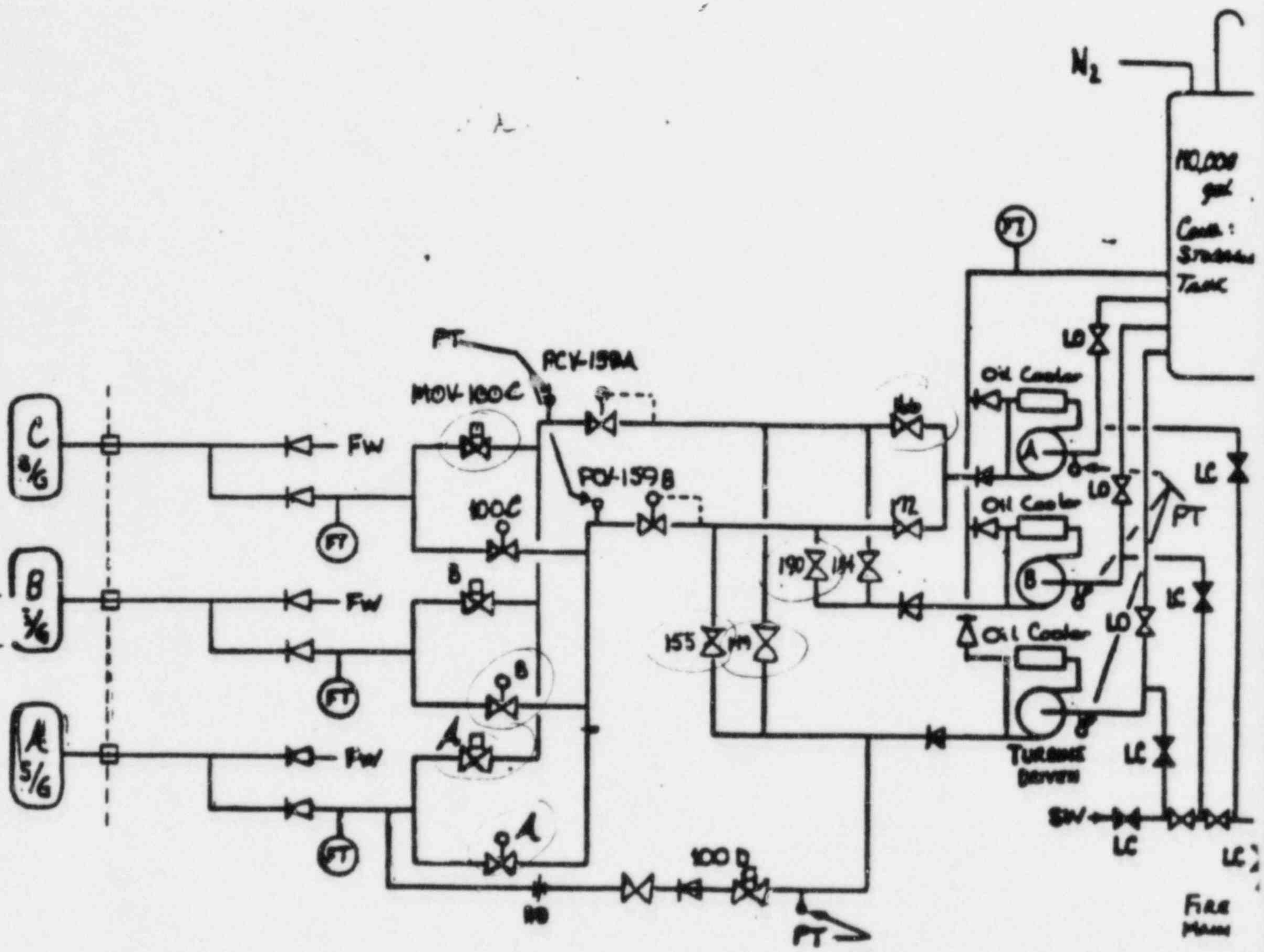
a) True (+.5 ea)
b) False
c) False

REFERENCE
NA NCRODP 88.1, 'RCS-RCP'

ANSWER 2.09 (1.50)

1) Portable system connecting to blank flanges (+.5)
2) Temporary piping connection from accumulators via blank flanges (+.5)
----#2 is the preferred method (+.5)

AUXILIARY FEEDWATER SYSTEM



REFERENCE
NA NCRODP 88.1, 'RCS Piping and Instrumentation'

ANSWER 2.10 (1.50)

a) True (+.5 ea)
b) True
c) True

ANSWER 2.11 (1.00)

b

REFERENCE
NA NCRODP 91.1, 'ESF-QSS'

ANSWER 2.12 (1.00)

a) Open (+.25 ea)
b) Closed
c) Open
d) Open

REFERENCE
NA NCRODP 91.1, 'ESF-RSS'

ANSWER 2.13 (1.00)

a

REFERENCE
NA NCRODP 89.4, 'Feedwater Systems-Main Feed'

ANSWER 2.14 (2.00)

MOV-100C (+.25 ea)
MOV-100A
H PCV-100B
H PCV-100A
155
149
166
190
See Attached Drawing

REFERENCE
NA NCRODP 89.4, 'Feedwater Systems-AFW'

ANSWER 2.15 (1.50)

-Service Water Resevoir (+.5 ea)
-Lake Anna
-Discharge Canal WH-5

REFERENCE
NA NCRODP 89.4, 'Feedwater Systems-AFW'

ANSWER 2.16 (1.00)

The Steam driven pump is operated at a higher speed. (+1.0)

REFERENCE
NA NCRODP 89.4, 'Feedwater Systems-AFW'

ANSWER 2.17 (1.50)

a) Hollow O-rings (+.5)
Inner and outer seal (+.5)

b) Reactor Vessel Head Vent System (RVHVS) (+.5)

REFERENCE
NA NCRODP 88.1, 'RCS-Reactor Vessl and Core Construction'

ANSWER 2.18 (2.00)

1) To upper head plenum via nozzles in core barrel flange (.5 ea)
2) Between hot leg discharge nozzles and upper core barrel outlets
3) Between baffle plates and core barrel
4) Around inserts in guide thimble tubes in the fuel assemblies

REFERENCE
NA NCRODP 88.1, 'RCS-Reactor Vessel/Core Construction'

ANSWER 2.19 (1.00)

~~c~~ Deleted

REFERENCE
NA NCRODP 88.1, 'RCS-Reactor Vessel/Core Construction'

ANSWER 2.20 (1.00)

c

REFERENCE
NA NCRODP 88.1, 'RCS-PZR and Press Relief'

ANSWER 2.21 (1.00)

d (improves circulation ratio and velocity across tube sheet)

REFERENCE
NA NCRODP 88.1, 'RCS-S/G'

ANSWER 2.22 (1.50)

-RHR System Inlet (+.5 ea)
-Loop Drain Lines
-Differential Pressure taps for RCS flow indication

REFERENCE
NA NCRODP 88.1, 'RCS-Piping and Instrumentation'

ANSWER 2.23 (2.00)

2; 1; 2; cesium; yttrium; molybendum; Li-7; Cs-137 (+.25 ea)

REFERENCE
NA NCRODP 88.3, 'CVCS'

ANSWER 2.24 (1.50)

a) Open (+.5 ea)
b) Open
c) Closed

REFERENCE
NA NCRODP 88.3, 'CVCS'

ANSWER 2.25 (1.00) Deleted

-Lower Delta P (8' vs 12-14' on Unit 1) (+.5 ea)
~~-approx. 3 MW gain for Unit 2~~

REFERENCE
NA NCRODP 89.1, 'Main Stam System'

ANSWER 2.26 (1.00)

CC Heat Xer Dschg (+.25 ea)
Dschg to Unit 2 Discharge Tunnel
Dschg to Resevoir
Recirc Spray Ht Xer

REFERENCE
NA NCRODP 92.2, 'Service Water System'

ANSWER 2.27 (1.00)

d

REFERENCE
NA NCRODP 92.2, 'Service Water System'

add:

CC Heat Exchanger/SW side. Basement of Aux Bldg., East end of CC Heat Exchangers on floor.

Discharge to tunnel - Turbine Bldg. SW pit.

Discharge to reservoir - Southeast corner of SW pump house.

Recirc Spray Hx's - Basement of Quench Spray pump house.

ANSWER 2.28 (1.00)

c

REFERENCE
NA NCRODP 91.2, 'Containment and Containment Systems'

ANSWER 2.29 (1.00)

d

REFERENCE
NA NCRODP 91.2, 'Containment and Containment Systems'

ANSWER 2.30 (1.50)

a) Not allowed (+.5 ea)
b) Allowed
c) Allowed

REFERENCE
NA NCRODP 91.2, 'Containment Isolation'

ANSWER 2.31 (1.50)

a) 3 (+.5 ea)
b) 4
c) 2

REFERENCE
NA NCRODP 91.2, 'Containment Atmospheric Cleanup System'

ANSWER 3.01 (1.00)

c

REFERENCE
Farley SD, 'PZR Press/Level Control', pp 6, 14, 15, Fig 3
NAPS Lesson plan, RCS Press. Instr.

ANSWER 3.02 (1.00)

d

REFERENCE
FNP, RHR Lesson Plan, pp. 8 & 9
NA NCRODP 88.2, 'RHR System'

005/000-K4.07 (3.2/3.5)

ANSWER 3.03 (1.00)

1. RCS pressure > 2010 psig (+.5 ea)
2. SI > 2000 psig

REFERENCE
FNP, ECCS Lesson Plan, Figure 9
NA NCRODP 91.1, 'ESF-ECCS'

006/000-A3.01 (4.0/3.9)

ANSWER 3.04 (1.00)

a

REFERENCE
Farley SD, 'Reactor Protection System', pp 33/34
NA Tech Spec pg 2-8

ANSWER 3.05 (1.00)

a

REFERENCE
CAT, PSM, Figures CN-IC-ILE-11 & 22
FNP, SD, 'PZR Pressure and Level Control', Fig 7
NA Lesson Plan, 'PZR Level Control and Protection'

011/000-A2.10 (3.4/3.6)

ANSWER 3.06 (1.00) ~~Deleted~~

b

REFERENCE
FNP, TS pp B 2-3\B 2-6
NA TS pp B2-2/6

ANSWER 3.07 (1.00)

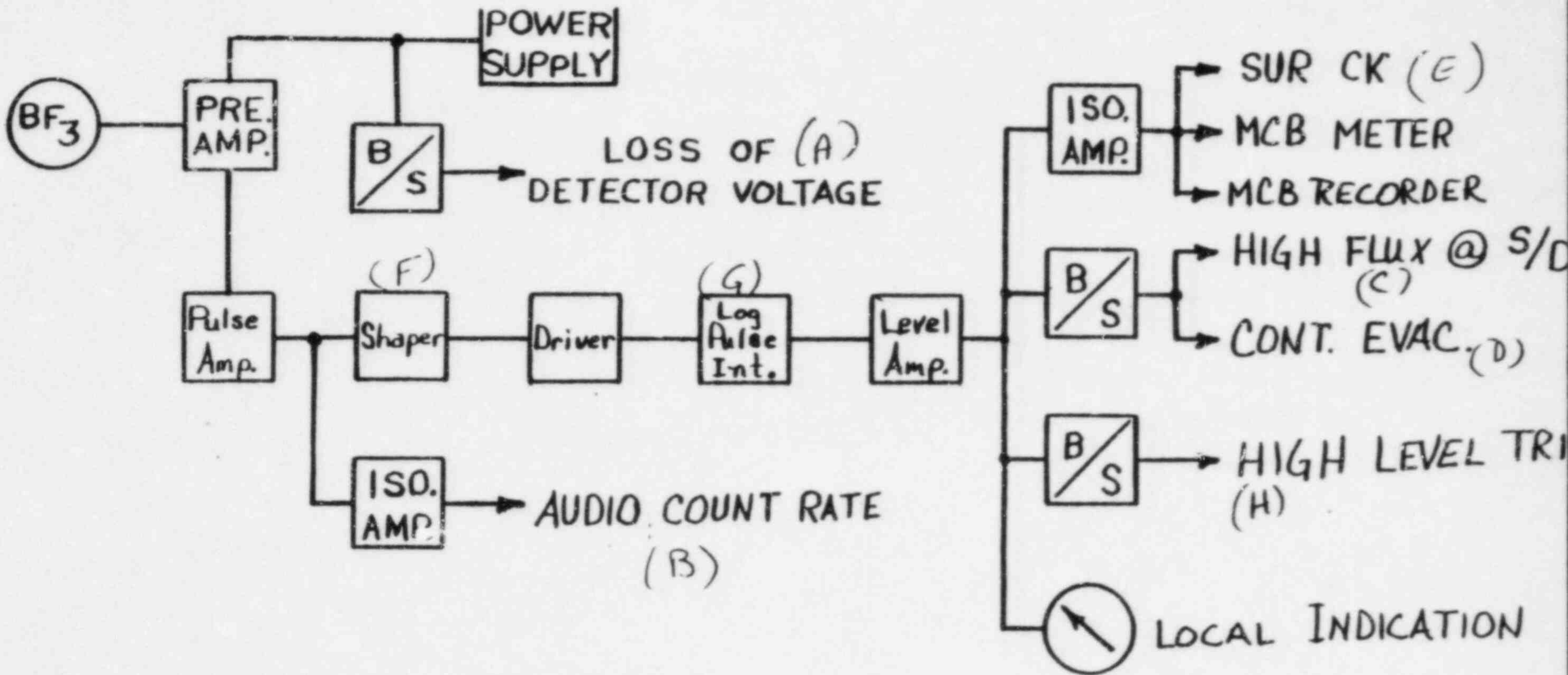
1) Trips not backed up by another protection circuit (+.5 ea)
2) The channel is also being used for control purposes

REFERENCE
NA NCRODP 93.10, 'RPS'

ANSWER 3.08 (2.00)

a) Manual (+.5 ea)
b) Auto
c) Auto
d) Manual

REFERENCE
NA NCRODP 93.10, 'RPS'



SOURCE RANGE

N31 & N32

ANSWER 3.09 (2.00)

See attached dwg

REFERENCE
NA NCRODP 93.2, 'Excore NIS'

ANSWER 3.10 (1.50)

a) False
b) False
c) True

REFERENCE
NA NCRODP 93.2, 'Excore NIS' pp 14-22

ANSWER 3.11 (1.00)

c

REFERENCE
NA NCRODP 93.4, 'Core Cooling Monitor'

ANSWER 3.12 (2.00)

a) No effect (+.5 ea)
b) Arm only
c) Arm and Actuate
d) No effect

REFERENCE
NA NCRODP 93.11, 'Steam Dumps'

ANSWER 3.13 (1.00)

e

REFERENCE
NA NCRODP 93.12, 'S/G Water Level Control and Protection'

ANSWER 3.14 (1.50)

-S/G Press used for density compensation of steam flow, so if pressure fails high, steam flow input to the SGWLCS will increase (+1.0)

-Ws>Wf will cause FWRV to open, level will increase until level error counteracts the steam flow/feed flow mismatch (+.5)

REFERENCE
NA NCRODP 93.12, 'SGWLC and Protection'

ANSWER 3.15 (1.00)

c

REFERENCE
NA NCRODP 93.13, 'RVLIS'

ANSWER 3.16 (1.00)

1) Eliminate accuracy reductions which would be caused by adverse environment in the containment during an accident. (+.75)
2) Allow easier accomplishment of system operations (calibrations, cell replacements, etc.) (+.25)

REFERENCE
NA NCRODP 93.13, 'RVLIS'

ANSWER 3.17 (1.00)

d

REFERENCE
NA Lesson Plan, 'Rod Control System'

ANSWER 3.18 (2.00)

a) 3 7
b) 6
c) 1
d) 5

REFERENCE
NA Lesson Plan, 'Rod Control System'

ANSWER 3.19 (1.00)

a

REFERENCE
NA Lesson Plan, 'Rod Position Indication'

ANSWER 3.20 (1.00)

d

REFERENCE
NA Lesson Plan, 'RCS Pressure Instrumentation'

ANSWER 3.21 (1.00)

-Assume a cold insurge (+.25)
-Which will cause a pressure drop (+.5)
-So heaters energized to anticipate this (+.25)

REFERENCE
NA Lesson Plan, 'PZR Level Control and Protection'

ANSWER ~~3.22~~ 3 23 (1.50)

-Allows Unit 1 to have its normal station service busses supplied from its normal station service transformers backfed from 500 KV switchyard for most Unit 1 trips. (+1.0)

-This reduces probability of combined loading from both Units normal and emergency buses on the reserve station transformer (+.5)

REFERENCE
NA NCRODP 90.1, 'Basic Electrical Distribution'

ANSWER ~~3.23~~ 3.22 ~~(1.50)~~

~~1) Low Lube Oil Pressure (+.3 ea)~~
~~2) High Crankcase Pressure~~
~~3) High Lube Oil Temperature~~
~~4) High Jacket Coolant Temperature~~
~~5) Start Failure~~

a) DCI (+.5 ea)
b) DCI
c) CRE Ref.
NCRODP 90.4, "EDG"
Handout H-2.4

ANSWER 3.24 (2.00)

a) 2 out of 3; 90%; 56 seconds (+.25 ea)

b) -diesel starts (+.25 ea)
-emergency bus isolated from RS xfrmr
-all loads shedded except Chg pumps, LHSI pumps and 480 VAC loads
-output bkr shuts when diesel up to speed and voltage
-loads sequenced onto bus

REFERENCE
NA NCRODP 90.4, 'Emerg Diesel Generator'

ANSWER 3.25 (1.00)

c

REFERENCE
NA NCRODP 91.1, 'ESF'

ANSWER 3.26 (1.00)

1) QS pump breaker open (+.5 ea)
2) CDA signal reset

REFERENCE
NA NCRODP 91.1, 'ESF-QSS'

ANSWER 3.27 (1.00)

c or b

REFERENCE
NA NCRODP 91.1, 'ESF-Recirc Spray System'

ANSWER 3.28 (1.00)

d

REFERENCE
NA NCRODP 91.1, 'ESF-ECCS'; 90.1, 'Basic Elect Distribution'

ANSWER 3.29 (2.00)

a) True (+.5 ea)
b) False
c) False
d) True

REFERENCE
NA NCRODP 91.2, 'Containment Isolation'

ANSWER 3.30 (1.00)

-Loop stop/Bypass isolation valves positioned to allow flow (+.25 ea)
-Bearing lift pump running with >700 psig discharge
-no electrical faults on RCP motor
-locked rotor protection speed switch satisfied

REFERENCE
NA NCRODP 88.1, 'RCS-RCPs'

ANSWER 3.31 (1.00)

b

REFERENCE
NA NCRODP 88.3, 'CVCS'

ANSWER 3.32 (1.00)

a

REFERENCE
NA NCRODP 88.1, 'RCS-PZR and Press Relief'

ANSWERS -- NORTH ANNA 1&2 -85/10/15-TOM ROGERS

ANSWER 4.01 (1.00)

b.

REFERENCE
NAPS, 1-OP-6.2, p. 7.

ANSWER 4.02 (1.00)

d.

REFERENCE
NAPS, 1-OP-3.4, p. 13.

ANSWER 4.03 (1.00)

c.

REFERENCE
NAPS, 1-AP-11, p.9.

ANSWER 4.04 (1.00)

a

REFERENCE
MNS OP/1/A/6100/01 p. 19.
CNS OP/1/A/6100/01, Encl. 4.1, 2.95.10.
NAPS 1-OP-2.1, p.8.

ANSWER 4.05 (1.00)

Any 4 @ 0.25 points each:

1. Bank C position.
2. Bank D position.
3. Auct. High Tavg.
4. IR N35.
5. IR N36.
6. RCS boron concentration.

REFERENCE
NAPS 1.OP-1.5, p.12.

ANSWER 4.06 (.50)

FALSE.

REFERENCE
NAPS 1-OP-1.5, p.10.

ANSWER 4.07 (1.00)

a

REFERENCE
NAPS, 1-OP-1.5, p.10.

ANSWER 4.08 (1.00)

b

REFERENCE
NAPS 1-OP-1.1.

ANSWER 4.09 (1.00)

d

ANSWERS -- NORTH ANNA 1&2 -85/10/15-TOM ROGERS

REFERENCE
NAPS 1-OP-8.3, p.5.

ANSWER 4.10 (1.00)

c

REFERENCE
NAPS 1-OP-1.4, Attachment 1.

ANSWER 4.11 (1.00)

b.

REFERENCE
NAPS 1-AP-4, pp. 3&8.

ANSWER 4.12 (1.00)

c.

REFERENCE
NAPS 1-AP-5.1, pp.8-9.

ANSWER 4.13 (1.00)

b

REFERENCE
MNS AP/2/A/5500/16, Case IV, p.9.
CNS AP/1/A/5500/16, Case IV, p.14.
NAPS 1-OP-4, p.8.

ANSWER 4.14 (1.25)

Any 5 @ 0.25 points each:

1. IRPI within demand position limit.
2. Each IRPI operable.
3. AFD within limits.
4. Containment temperature.
5. Control room temperature.
6. Circ. water temperature.
7. QPTR
8. HEAT BALANCE

REFERENCE
NAPS 1-AP-42.

ANSWER 4.15 (.50)

FALSE. (urgent failure alarm could be from logic or power cabinet failure)

REFERENCE
NAPS 1-AR-1, 1A-4.

ANSWER 4.16 (1.00)

@ 0.25 points each:
1. Start all available service air compressors.
2. Start all available instrument air compressors.
3. Secure reactor coolant pumps.
4. Secure charging pump.

REFERENCE
NAPS 1-AP-28.1, p.3.

ANSWER 4.17 (1.00)

Place both level trip switches in BYPASS. (Ops may continue.)

REFERENCE
NAPS 1-AP-4, p.4.

ANSWER 4.18 (1.00)

@ 0.25 points each:

1. Shift rod control to manual.
2. Control turbine to match Tavg and Tref.
3. Ensure Tavg is > 541 or get >541 within 15 min or trip Rx.
4. Notify Shift Supervisor.

REFERENCE
NAPS 1-AP-1.4, p.3.

ANSWER 4.19 (1.00)

@ 0.25 points each:

1. Go to BORATE on Blender Mode Switch.
2. Adjust FC-113A to allow 10 or more gpm flow rate.
3. Turn Blender Control Switch to START.
4. Verify flow.

REFERENCE
NAPS 1-AP-2.0, p.3.

ANSWER 4.20 (1.00)

@ 0.5 points each:

1. Place or verify rods in MANUAL.
2. Place or verify steam dumps in steam pressure mode.

REFERENCE
NAPS 1-AP-3, p.4.

ANSWER 4.21 (1.00)

@ 0.25 points each:

1. Reduce generator load in attempt to stabilize vacuum.
2. Check vacuum breaker closed and water seal present.
3. Manually trip turbine at 9.5" HG abs if it has not automatically done so.
4. Trip reactor.

REFERENCE
NAPS 1-AP-14, p.3.

ANSWER 4.22 (1.00)

@ 0.5 points each:

1. Trip turbine locally.
2. Manually open reactor trip breakers or the rod drive MG output breakers.

REFERENCE
NAPS 1-AP-20, p.3.

ANSWER 4.23 (1.00)

For 0.20 points each.
1. Attempt a reactor trip using both manual trip switches.
2. Attempt a turbine trip using both manual trip buttons.
3. Verify all AFW pumps are running.
4. Verify AFW flow to all SGs.
5. Emergency Borate

REFERENCE
NAPS 1-ECA-1, pp. 2&3.

ANSWER 4.24 (3.25)

For 0.25 points each.
1. Verify reactor trip.
2. Verify turbine trip.
3. Verify AC emergency busses energized.
4. Check if SI actuated.
5. Check charging/SI pump valve lineup.
6. Verify FW isolation.
7. Verify AFW flow.
8. Verify charging/SI flow.
9. Verify LHSI pumps running.
10. Verify containment phase A isolation.
11. Verify service water pumps running.
12. Verify RCS heat removal.
13. Check containment pressure.

REFERENCE
NAPS 1-EP-0, pp. 1-5.

ANSWER 4.25 (1.00)

b.

REFERENCE
NAPS 1-ES-0.2, Foldout page.

ANSWER 4.26 (1.00)

b

REFERENCE
MNS EP/2/A/5000/1.1, p.5.
CNS EP/1/A/5000/1A1, p.4.
NAPS 1-ES-0.3, p.5.

ANSWER 4.27 (1.00)

d

REFERENCE
MNS EP/2/A/5000/16.C
CNS EP/1//A/5000/2F3, p.7.
NAPS 1-FRP-I.3A, p.3.

ANSWER 4.28 (1.00)

c

REFERENCE
Vepco GET, p.49.

ANSWER 4.29 (1.00)

c

REFERENCE
Vepco GET, p.49.

ANSWER 4.30 (1.00)

Reset it.

REFERENCE
NAPS 1-ECA-0.1, p.2.

ANSWER 4.31 (1.00)

@ 0.5 points each:
1. Containment pressure at 19.7 psia.
2. Containment rad. levels at 40,000 R/hr.

REFERENCE
NAPS 1-EP-0, p.2.

ANSWER 4.32 (.50)

Higher.

REFERENCE
NAPS 1-EP-0 Foldout page.

ANSWER 4.33 (1.00)

For 0.5 points each.
(a) 750 mr/qtr
(b) 2750 mr/qtr

REFERENCE
NAPS Radiation Protection Manual, p.2.3-7.

ANSWER 4.34 (1.00)

c

REFERENCE
10 CFR 20.101

ANSWER 4.35 (1.00)

b

QF=1 for gamma

100(45/60)(1)=75

REFERENCE
10 CFR 20.

ANSWER 4.36 (1.00)

a

REFERENCE
10 CFR 20.

ANSWER 4.37 (1.00)

c

REFERENCE
Vepco GET, p.21.

ANSWER 4.38 (1.00)

Any 5 @ 0.2 points each:

1. Job location.
2. Brief job description.
3. Radiological conditions at job site.
4. Anti C requirements.
5. Respiratory requirements.
6. Dosimetry requirements.
7. Necessary HP coverage.
8. Any special instructions.

REFERENCE
Vepco GET, p.41

ANSWER 4.39 (1.00)

c

REFERENCE
Vepco GET, p.41.

EXAMINER:

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: NORTH ANNA 1&2

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 85/10/15

EXAMINER: DEAN, W M

APPLICANT: MASTER FILE COPY

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
40.00	25.00	________	________	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
38.5 ~~40.00~~	25.00	________	________	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
40.00	25.00	________	________	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
38.0 ~~40.00~~	25.00	________	________	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
156.5 160.00	100.00	________	________	TOTALS

FINAL GRADE ________________%

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

____
APPLICANT'S SIGNATURE

QUESTION 5.01 (1.00)

The reactor is producing 100% rated thermal power at a core delta T of 60 degrees and a mass flow rate of 100% when a blackout occurs. Natural circulation is established and core delta T goes to 40 degrees. If decay heat is 2%, what is the core mass flow rate (in %)?

a. 1.3

b. 2.0

c. 3.0

d. 4.0

QUESTION 5.02 (1.00)

Which of the following does NOT contribute to assuring that the enthalpy rise hot channel limits are maintained?

a. Axial power distribution is maintained within limits.

b. Control rod banks are sequenced with proper overlap.

c. Control rod insertion limits are maintained.

d. The MTC is within its analyzed temperature range.

QUESTION 5.03 (1.00)

Which of the following is NOT a reason for pressurizing the fuel rods with helium?

a. Minimize clad creeping inwards toward fuel pellets.

b. Increase gap (pellet to clad) thermal conductivity.

c. Allow detection of clad failure by helium analysis of the coolant.

d. Maintain lower fuel centerline temperature.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.04 (1.00)

Which of the following is NOT one of the conditions necessary for brittle fracture to occur?

a. Plastic deformation at or below the yield point.

b. Temperature at or below the NDTT.

c. Nominal stress level.

d. Flaw such as a crack present.

QUESTION 5.05 (1.00)

Which statement below describes centrifugal pump runout conditions?

a. High pressure, low flow, high power demand
b. High pressure, low flow, low power demand
c. Low pressure, high flow, high power demand
d. Low pressure, high flow, low power demand
e. Low pressure, low flow, high power demand

QUESTION 5.06 (1.00)

An ECP is calculated for a reactor startup 4 hours after a reactor trip from 100% power, equilibrium Xe conditions. Which of the following would cause the actual critical position to be lower than the ECP?

a. The startup is delayed until 8 hours after the trip

b. Actual boron concentration is 10 ppm more than the predicted boron concentration.

c. A rod finger is separated from its spider assembly.

d. The steam dump pressure setpoint is lowered by 100 psi prior to reactor startup.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.07 (1.00)

Which of the following will cause the Axial Flux Difference to become more positive (less negative)?

a. Power increase with power defect compensated for by dilution only.

b. Power increase with power defect compensated for by rod withdrawal only.

c. Buildup of xenon in top portion of core.

d. Burnup of xenon in bottom portion of core.

QUESTION 5.08 (1.00)

Which of the following will cause the Fuel Temperature Coefficient (pcm per degree) to become LESS negative?

a. Fuel temperature increase.

b. Boron concentration decrease.

c. Control rod withdrawal (at constant power).

d. Core age increase.

QUESTION 5.09 (1.00)

Which of the following describes the changes that steam in a REAL turbine undergoes comparing conditions at the inlet to those at the outlet?

a. Enthalpy DECREASES, Entropy DECREASES, Quality DECREASES

b. Enthalpy INCREASES, Entropy INCREASES, Quality INCREASES

c. Enthalpy CONSTANT, Entropy DECREASES, Quality DECREASES

d. Enthalpy DECREASES, Entropy INCREASES, Quality DECREASES

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.10 (1.00)

Which of the following tensile stresses is higher on the outer wall than on the inner wall of the reactor pressure vessel?

a. Pressure stress.

b. Composite (Total) stress during cooldown.

c. Cooldown stress (due to delta T only).

d. Heatup stress (due to delta T only).

QUESTION 5.11 (1.00)

Which of the following statements concerning Xenon-135 production and removal is correct?

a. At full power, equilibrium conditions, about half of the xenon is produced by iodine decay and the other half is produced as a direct fission product.

b. Following a reactor trip from equilibrium conditions, xenon peaks because delayed neutron precursors continue to decay to xenon while neutron absorption (burnout) has ceased.

c. Xenon production and removal increases linearly as power level increases; i.e., the value of 100% equilibrium xenon is twice that of 50% equilibrium xenon.

d. At low power levels, xenon decay is the major removal method. At high power levels, burnout is the major removal method.

QUESTION 5.12 (1.00)

Which of the following is the units of heat flux?

a. Watts / cubic centimeter

b. BTU / (hr square ft)

c. Calories / gram

d. kW / ft

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.13 (1.00)

Which of the following statements concerning the power defect is correct?

a. The power defect is the difference between the measured power coefficient and the predicted power coefficient.

b. The power defect increases the rod worth requirements necessary to maintain the desired shutdown margin following a reactor trip.

c. Because of the higher boron concentration, the power defect is more negative at beginning of core life.

d. The power defect necessitates the use of a ramped Tavg program to maintain an adequate Reactor Coolant System subcooling margin.

QUESTION 5.14 (1.00)

The reactor is critical at 10,000 cps when a S/G PORV fails open. Assuming BOL conditions, no rod motion, and no reactor trip, choose the answer below that best describes the values of Tavg and nuclear power for the resulting new steady state. (POAH = point of adding heat).

a. Final Tavg greater than initial Tavg, Final power above POAH.

b. Final Tavg greater than initial Tavg, Final power at POAH.

c. Final Tavg less than initial Tavg, Final power at POAH.

d. Final Tavg less than initial Tavg, Final power above POAH.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.15 (1.00)

The Moderator Temperature Coefficient (MTC) varies with certain plant conditions. Concerning these variations, which of the following is correct?

a. The MTC becomes more negative as boron concentration is increased.

b. The MTC causes axial flux distribution to be tilted towards the top of the core at BOL.

c. The MTC varies as temperature changes because of the non-linear density changes of water as temperature changes.

d. The MTC is not permitted by Technical Specifications to be positive in any plant operating modes.

QUESTION 5.16 (1.00)

Which of the following statements concerning Shutdown Margin (SDM) is correct?

a. The maximum SDM requirement occurs at EOL and is based on a rod ejection accident.

b. The maximum SDM requirement occurs at EOL and is based on a steam line break accident.

c. The maximum SDM requirement occurs at BOL and is based on having a positive moderator temperature coefficient.

d. The maximum SDM requirement occurs at BOL and is based on a rod withdrawal accident while in the source range.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.17 (1.00)

The reactor is operating at 30% power when one RCP trips. Assuming the reactor does not trip and no turbine load changes occur, which of the following parameters will DECREASE?

a. Flow in operating reactor coolant loops

b. Core delta T

c. Reactor vessel delta P

d. Steam generator pressure in affected loop

QUESTION 5.18 (1.00)

The Technical Specifications allow operations for a 2-hour time period with a Quadrant Power Tilt Ratio (QPTR) of greater than 1.02. Which of the following is the reason for allowing operations for these 2 hours?

a. To allow time for corrective action in the event of xenon redistribution following power changes.

b. To allow time for correction of a dropped or misaligned control rod.

c. To allow time for boron concentration changes to restore the QPTR to less than 1.02.

d. To allow time for correction of RCS flow imbalances.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.19 (1.00)

During a reactor trip recovery, the initial 1/M data point was 1.0. After a 1-hour delay, rod withdrawal was commenced. Upon stopping rod withdrawal to take 1/M data, the RO reported that the second 1/M data point was 1.1. Which of the following would explain this increase in the 1/M value?

a. This is NOT possible, the RO must have made an error when taking count rate data.

b. The buildup of xenon during the 1-hour delay added more negative reactivity than the rod withdrawal added positive.

c. Compensating voltage on the IR instruments fails low.

d. An inadvertent dilution is in progress.

QUESTION 5.20 (1.00)

During a reactor startup in a subcritical reactor, the first reactivity addition caused count rate to increase from 10 cps to 16 cps. The second reactivity addition caused count rate to increase from 16 cps to 32 cps. Which of the following statements describing the relationship between the first and second reactivity additions is correct?

a. The first reactivity addition was larger.

b. The second reactivity addition was larger.

c. The first and second reactivity additions were equal.

d. There is not enough data given to determine relationship between reactivity values.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.21 (1.00)

Which of the following is defined as 'the fractional change in neutron population per generation'?

a. keff

b. Delta k

c. Reactivity

d. Delta reactivity

QUESTION 5.22 (1.00)

Which of the following can be defined as 'the number of neutrons causing fission that were originally born delayed divided by the total number of neutrons causing fission'?

a. Lambda effective

b. Rho

c. Beta effective

d. Tau

QUESTION 5.23 (1.00)

Which of the following expresses the relationship between differential rod worth (DRW) and integral rod worth (IRW)?

a. DRW is the slope of the IRW curve at that location.

b. DRW is the area under the IRW curve at that location.

c. DRW is the square root of the IRW at that location.

d. There is no relationship between DRW and IRW.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.24 (1.00)

Which of the following conditions would result in the highest available net positive suction head?

a. Pump with heat source downstream and heat sink upstream with surge tank on its discharge line.

b. Pump with heat source downstream and heat sink upstream with surge tank on its suction line.

c. Pump with heat sink downstream and heat source upstream with surge tank on its discharge line.

d. Pump with heat sink downstream and heat source upstream with surge tank on its suction line.

QUESTION 5.25 (1.00)

Which of the following statements concerning samarium reactivity effects is correct?

a. The equilibrium (at power) value of samarium is dependent upon power level. The peak value of samarium following a shutdown is dependent upon power level prior to shutdown.

b. The equilibrium (at power) valve of samarium is dependent upon power level. The peak value of samarium following a shutdown is independent of power level prior to shutdown.

c. The equilibrium (at power) value of samarium is independent of power level. The peak value of samarium following a shutdown is dependent upon power level prior to shutdown.

d. The equilibrium (at power) value of samarium is independent of power level. The peak value of samarium following a shutdown is independent of power level prior to shutdown.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.26 (1.00)

The reactor startup procedure requires that the critical rod position be taken at 10E-8 amps on the intermediate range. If, during a xenon free reactor startup at MOL, the operator 'overshoots' 10E-8 amps and instead leveled off at 10E-7 amps, which of the following statements is correct?

a. At 10E-7 amps, there are little or no effects from nuclear heat; but, since the reactor is a decade higher in power, the critical rod position would be higher than at 10E-8 amps.

b. At 10E-7 amps, there are little or no effects from nuclear heat; therefore, the critical rod position should be the same as at 10E-8 amps.

c. At 10E-7 amps, there are substantial effects from nuclear heat; therefore, the critical rod position will be higher than at 10E-8 amps.

d. At 10E-7 amps, nuclear heat, xenon, and being a decade higher in power level will result in a higher critical rod position than at 10E-8 amps.

QUESTION 5.27 (1.00)

The reactor trips from full power, equilibrium xenon conditions. Six hours later the reactor is brought critical at 10E-8 amps on the intermediate range. If power level is maintained at 10E-8 amps which of the following statements concerning rod motion requirements for the next two hours is correct?

a. Rods will have to be withdrawn since xenon will closely follow its normal build-in rate following a trip.

b. Rods will have to be inserted since xenon will closely follow its normal decay rate following a trip.

c. Rods will have to be rapidly inserted since the critical reactor will cause a high rate of burnout.

d. Rods will have to be rapidly withdrawn since the critical reactor will cause a higher than normal rate of build-in.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.28 (1.00)

Which of the following is a reason for shifting the SI mode from cold leg recirculation to hot leg recirculation approximately 24 hours after a LOCA?

a. Increase recirculation flow through the core.

b. Ensure boron precipitation does not cause long-term cooling problems.

c. Allow sampling RCS to determine amount of core damage.

d. Minimize the hydrogen concentration in the coolant.

QUESTION 5.29 (1.00)

Which of the following would be an indication of a loss of natural circulation flow?

a. RCS delta T less than full power delta T.

b. S/G pressure decreasing rapidly as S/G level is increased.

c. RCS subcooling decreasing.

d. Cold leg at saturation temperature for S/G pressure.

QUESTION 5.30 (1.50)

Indicate whether the following statements are TRUE or FALSE.

a. A positive 100 pcm reactivity addition from criticality low in the intermediate range will produce the same constant startup rate throughout core life. (0.5)

b. A delayed neutron has a higher probability of causing fission than does a prompt neutron. (0.5)

c. If a reactor is supercritical, the fraction of delayed neutrons shifts to the shorter lived precursors and the value of the average decay constant (lambda) decreases. (0.5)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.31 (1.00)

When performing a reactor S/U to full power that commenced five hours after a trip from full power equilibrium conditions, a 2%/min ramp was used. How would the resulting xenon transient vary if instead a 0.5%/min ramp was used?

a. The xenon dip for the 0.5%/min ramp would occur sooner and be smaller.

b. The xenon dip for the 0.5%/min ramp would occur later and be smaller.

c. The xenon dip for the 0.5%/min ramp would occur sooner and be larger.

d. The xenon dip for the 0.5%/min ramp would occur later and be larger.

QUESTION 5.32 (1.00)

Which of the following conditions would cause a 1/M curve to predict criticality earlier than it will actually occur?

a. Source located too near detector.

b. Fuel assemblies loaded too far from detector.

c. Highest worth fuel assemblies loaded first.

d. Control rod located between fuel assemblies loaded and detector.

QUESTION 5.33 (1.00)

Which of the following will result in the largest INCREASE in the concentration of dissolved gases in a quantity of water? (Assume the changes in temperature and pressure below are of equal magnitude)

a. Increasing the pressure and lowering the temperature.

b. Decreasing the pressure and lowering the temperature.

c. Increasing the pressure and raising the temperature.

d. Decreasing the pressure and raising the temperature.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.34 (2.00)

If steam goes through a throttling process, indicate whether the following parameters will INCREASE, DECREASE, or REMAIN THE SAME.

a. Enthalpy (0.5)

b. Pressure (0.5)

c. Entropy (0.5)

d. Temperature (0.5)

QUESTION 5.35 (2.00)

A motor driven centrifugal pump is operating at rated flow. You then start closing down on the discharge valve. How (INCREASE, DECREASE, or REMAIN THE SAME) will each of the following be affected?

a. Flow (0.5)

b. Discharge Pressure (0.5)

c. Available NPSH (0.5)

d. Motor Amps (0.5)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.36 (2.00)

Answer TRUE or FALSE to the following.

a. The use of a ramped Tave program allows the plant to operate with a higher thermodynamic efficiency than does a constant Tave program. (0.5)

b. Increasing condensate depression (subcooling) will cause BOTH a decrease in plant efficiency AND an increase in condensate (hotwell) pump available NPSH. (0.5)

c. During a RCS heatup, as temperature gets higher, it will take a smaller letdown flow rate to maintain a constant pressurizer level. (0.5)

d. The difference between pump su tion pressure and the saturation pressure of the fluid being pumped is referred t~ as net positive suction head. (0.5)

QUESTION 5.37 (.50)

TRUE or FALSE?

For heat exchangers with the same heat transfer area, the same mass flow rates and with incoming fluid temperatures the same for both the cooling and cooled mediums, a counterflow heat exchanger will transfer more heat than a parallel flow heat exchanger.

(***** END OF CATEGORY 05 *****)

QUESTION 6.01 (1.00)

Which of the following conditions is NOT required for automatic swapover of the LHSI pumps to the Recirculation Mode following an SI?

a. RWST Lo-Lo Level
b. A LHSI pump recirc isolation MOV closed for each pump
c. SI signal present
d. SI Recirc Mode signal present

QUESTION 6.02 (1.00)

Which statement below regarding pressurizer spray is correct?

a. The position of the auxiliary spray valve can be adjusted from fully shut through the fully open position by the operator at the main control board.

b. Each of the normal spray valves has a 1 gpm bypass line to prevent cooldown of the spray lines.

c. The maximum spray flow rate is designed to prevent PORVs from opening during a 10% step DECREASE in power.

d. Auxiliary spray utilizes a separate spray nozzle due to the large temperature differential that exists when auxiliary spray is used.

QUESTION 6.03 (1.00)

Which of the following functions is NOT performed by the hydraulic power system associated with the Containment Personnel Air Lock?

a. Operation of the Equalizing Valve
b. Door Locking Ring Rotation
c. Interior Door Operation
d. Exterior Door Operation

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.04 (1.00)

PT-402 and 403 measure RCS pressure upstream of RHR suction isolation valves MOV 1700 and 1701. Which of the following statements concerning the effect(s) of PT-402 failing upscale (high) is correct?

a. If both MOV 1700 and MOV 1701 are open they will shut.

b. If open, MOV 1700 will shut and if closed, MOV 1701 will not be able to be opened.

c. If both MOV 1700 and MOV 1701 are shut, they will not be able to be opened.

d. If open, MOV 1700 will shut, but MOV 1701 can be positioned as desired by the operator.

e. MOV 1700 can be positioned as desired by the operator, but MOV 1701 will not be able to be opened if it is shut.

QUESTION 6.05 (1.00)

Which of the following statements concerning the operation of the letdown isolation valves (LCV-1460A and B) is correct?

a. Neither of the letdown isolation valves can be opened if the containment letdown isolation valve (TV 1204) is shut.

b. Shutting all orifice isolation valves will automatically shut the letdown isolation valves

c. All orifice isolation valves must be shut in order to open the letdown isolation valves

d. High temperature on the letdown outlet of the regenerative heat exchanger will automatically shut the letdown isolation valves

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.06 (1.00)

Which of the following describes the purpose of the time delay (195 sec) in starting the Inside Recirculation Spray (IRS) pumps on a CDA signal?

a. Prevents overloading of the emergency buses as loads are sequentially energized during accident conditions.

b. Allows time for the containment sumps to collect sufficient fluid to provide NPSH for the pumps.

c. Enhances core cooling by increasing reflood rate after a LOCA as the pressure drop between core exit and the break is reduced with a higher containment pressure.

d. Allows time for the fluid collecting in the containment sumps to cool, to avoid flashing in the RS heat exchangers as the fluid is cooled by Service Water.

QUESTION 6.07 (1.00)

Which of the following describes the Service Water System automatic actions on a single unit SI?

a. All SW pumps start, that unit's spray header isolation MOVs receive an 'open' signal

b. Only the SW pumps supplied from that unit's emergency buses start, that unit's spray header isolation MOVs receive an 'open' signal

c. All SW pumps start, all spray header isolation MOVs receive an 'open' signal

d. The SW pumps supplied from that unit's emergency buses start, all spray header isolation MOVs receive an 'open' signal

QUESTION 6.08 (1.00)

Which location below is the discharge point for the pressurizer head vent?

a. Containment fuel canal
b. Upper region of containment below quench spray rings
c. Pressurizer Relief Tank
d. Suction side of containment Hydrogen Recombiners

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.09 (1.00)

Which of the following signals, acting independently, will automatically CLOSE the Main Feedwater Regulating Valves (MOV-FW-1154A, B and C)?

a. Hi-Hi S/G level in any S/G (2 out of 3 detectors)
b. Low Tavg (2 out of 3)
c. Reactor Trip
d. Phase B Isolation

QUESTION 6.10 (1.00)

Which graph on the attached page represents the change in the internal pressure of a fuel rod versus fuel burnup?

a. A
b. B
c. C
d. D
e. E

QUESTION 6.11 (1.00)

The pressurizer safety valves are designed to limit RCS pressure to within the safety limit of 2735 psig following a complete loss of turbine generator load while at rated thermal power. Which of the following mitigating events is assumed to have occurred in the analysis for this design basis?

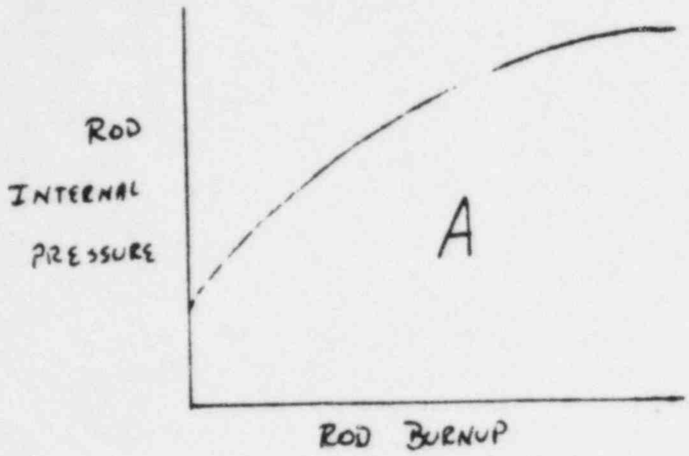
a. Reactor trip due to turbine loss of load
b. PORV operation
c. Hi PZR pressure trip
d. Both spray valves actuate
e. Relief valves on all 3 S/Gs actuate

QUESTION 6.12 (1.00)

Which of the following is a load using Service Water as a BACKUP source of heat removal?

a. Service Air Compressors
b. Instrument Air Compressors
c. Control Room Air Conditioning Units
d. Containment Air Recirc Coolers

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)


ROD
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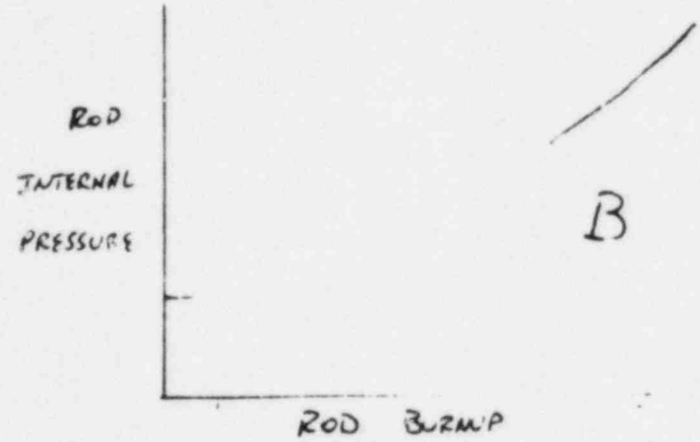
PRESSURE
A
ROD BURNUP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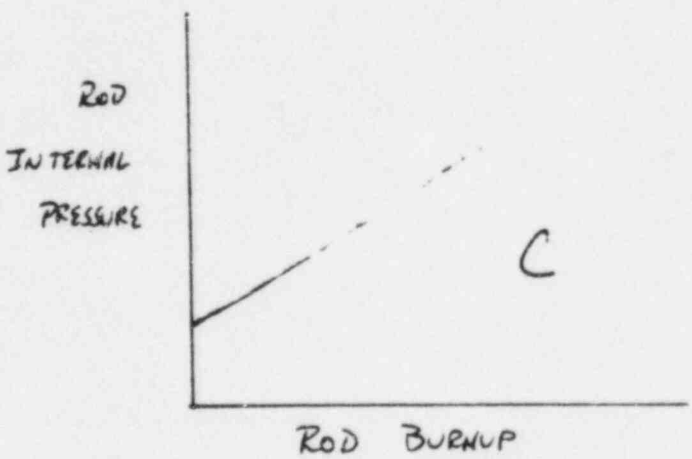
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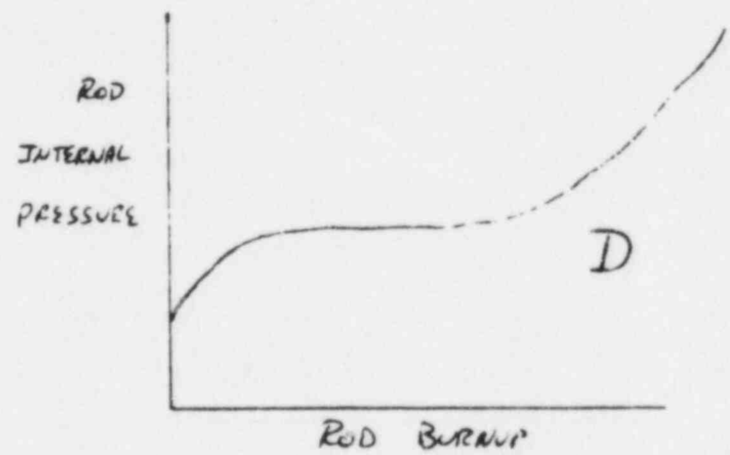
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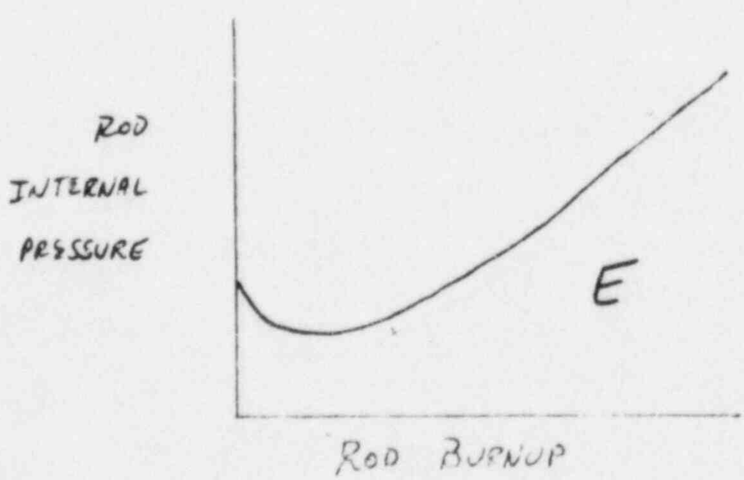
ROD
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PRESSURE
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ROD BURNUP

QUESTION 6.13 (1.00)

Which statement below regarding the North Anna offsite power system is correct?

a. 22 KV from offsite can be used to power BOTH the Reserve Station and Station Service Transformers.

b. A Reserve Station Transformer consists of a bank of three single-phase transformers.

c. All three Reserve Station Transformers can be supplied, via switching circuits, from EITHER of two 500/34.5 KV transformers.

d. North Anna Power Station is connected to the VEPCO grid by three 500 KV lines of which a MINIMUM of two are required to carry the output of BOTH units.

QUESTION 6.14 (2.00)

Indicate whether the following situations will cause the Steam Dump System to ARM ONLY, ARM AND ACTUATE or have NO EFFECT. (Assume all switches are in normal positions unless otherwise indicated)

a) 60% power, 3 %/min load reject, Tavg>Tref by 8 degrees F, Mode Selector Switch in Tavg.

b) Turbine trip from 25% power, Tavg=545 degrees F, Mode Selector Switch in Tavg.

c) Hot Zero Power, Tavg=550 degrees F, Mode Selector Switch in STM PRESS, 1020 psig setpoint selected.

d) 40% power, 5%/min load increase, Pimp CH-III fails low, Tavg=560 degrees F, Mode Selector Switch in Tavg

QUESTION 6.15 (1.50)

Indicate whether the following statements apply to Fire Pump 1-FP-P-1, Fire Pump 1-FP-P-2 or to BOTH.

a) 2500 GPM, diesel driven pump

b) Recirculates its discharge to the Unit 1 A-well

c) The pump can only be stopped locally

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.16 (1.50)

Answer the following questions regarding the Excore NIS TRUE or FALSE.

a) An Intermediate Range detector that is overcompensated could result in the P-6 setpoint never being reached on a shutdown.

b) Placing the Source Range instrument in 'Bypass' at the instrument drawer will prevent a reactor trip signal if either the Instrument or the Control Power Fuses are pulled.

c) Neither the Source Range or the Intermediate Range 'Hi Flux' trips are taken credit for in accident analysis.

QUESTION 6.17 (1.50)

Indicate whether the following containment isolation valve combinations are ALLOWED or NOT.

a) One LOCKED CLOSED isolation valve INSIDE and a simple check valve used as an AUTOMATIC isolation valve OUTSIDE

b) An AUTOMATIC isolation valve INSIDE (not a simple check valve) and a LOCKED CLOSED isolation valve OUTSIDE containment.

c) AUTOMATIC isolation valves BOTH INSIDE and OUTSIDE containment with the INSIDE valve a simple check valve.

QUESTION 6.18 (1.50)

Indicate whether the following controls for 1H EDG are transferred to/from the 1H diesel room using the 'Control Room Emergency' (CRE) switch or the 'Diesel Control Isolation' (DCI) switch.

a) Control and synchronization for the 4.16 KV Emergency Bus emergency supply breaker.

b) Alternate governor controls by means of the Droop/Isochronous Selector switch.

c) Diesel Start/Stop control

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.19 (1.50)

Match the Containment Atmospheric Cleanup System component in Column A with the correct location in Column B.

COLUMN A	COLUMN B
a) H2A-HC-200 H2 Analyzer	1) Aux Bldg near Drumming Area
b) H2A-HC-201 Remote Analyzer Panel	2) Below Rod Drive Room
c) 1-HC-HC-1 H2 Recombiner	3) Aux Bldg near Demin Alley
	4) Unit 2 Instrument Rack Room
	5) By High Rad Sampling Panels

QUESTION 6.20 (1.50)

Match the Main Steam Valve description listed in Column A with the appropriate valve listed in Column B.

Column A	Column B
a) Valve disc is independent of stem and floats on steam flow	1) Main Steam Trip Valve (TV-MS-101 A,B,C)
b) Valve disc is held out of position by use of air pistons	2) Equalizing Trip Valve (TV-MS-113 A,B,C)
c) Air is supplied to both the top and bottom of valve actuator	3) Non-Return Valve (NRV-MS-101 A,B,C)
	4) AFW Pump Steam Supply Valve (TV-MS-111 A,B)
	5) Equalizing Non-Return Valve (NRV-MS-103 A,B,C)

QUESTION 6.21 (1.50)

With the reactor at 50% Power, S/G A's steam pressure input to the Steam Generator Water Level Control System fails high. Explain how this will affect the Level Control System for S/G A, and why this action occurs.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.22 (2.00)

Attached is a drawing of the Auxiliary Feedwater System. List the valves downstream of the AFW pumps which are closed in a normal MODE 1 lineup.

QUESTION 6.23 (1.50)

What are the three groups of RCS piping penetrations made BELOW the horizontal centerline of the piping? (penetrations with a similar purpose are to be treated as one group)

QUESTION 6.24 (1.00)

Unit 2's Moisture Separator Reheaters (MSRs) have a hexagonal tube bundle which has almost twice as many tubes as Unit 1's MSRs, but the Unit 2 MSRs are wider and shorter than Unit 1's. What effect, if any, does this difference in MSR construction have on the Differential Pressure across Unit 2's MSRs and the plant's thermal power capabilities? (Indicate both direction and approximate magnitude of any differences noted)

QUESTION 6.25 (1.75)

With the exception of indication, list all the protective, control and permissive circuits which receive an input from PT-1446 and PT-1447 (turbine first stage pressure channels III and IV). Identify circuits by nomenclature where applicable. (eg P-6)

QUESTION 6.26 (1.00)

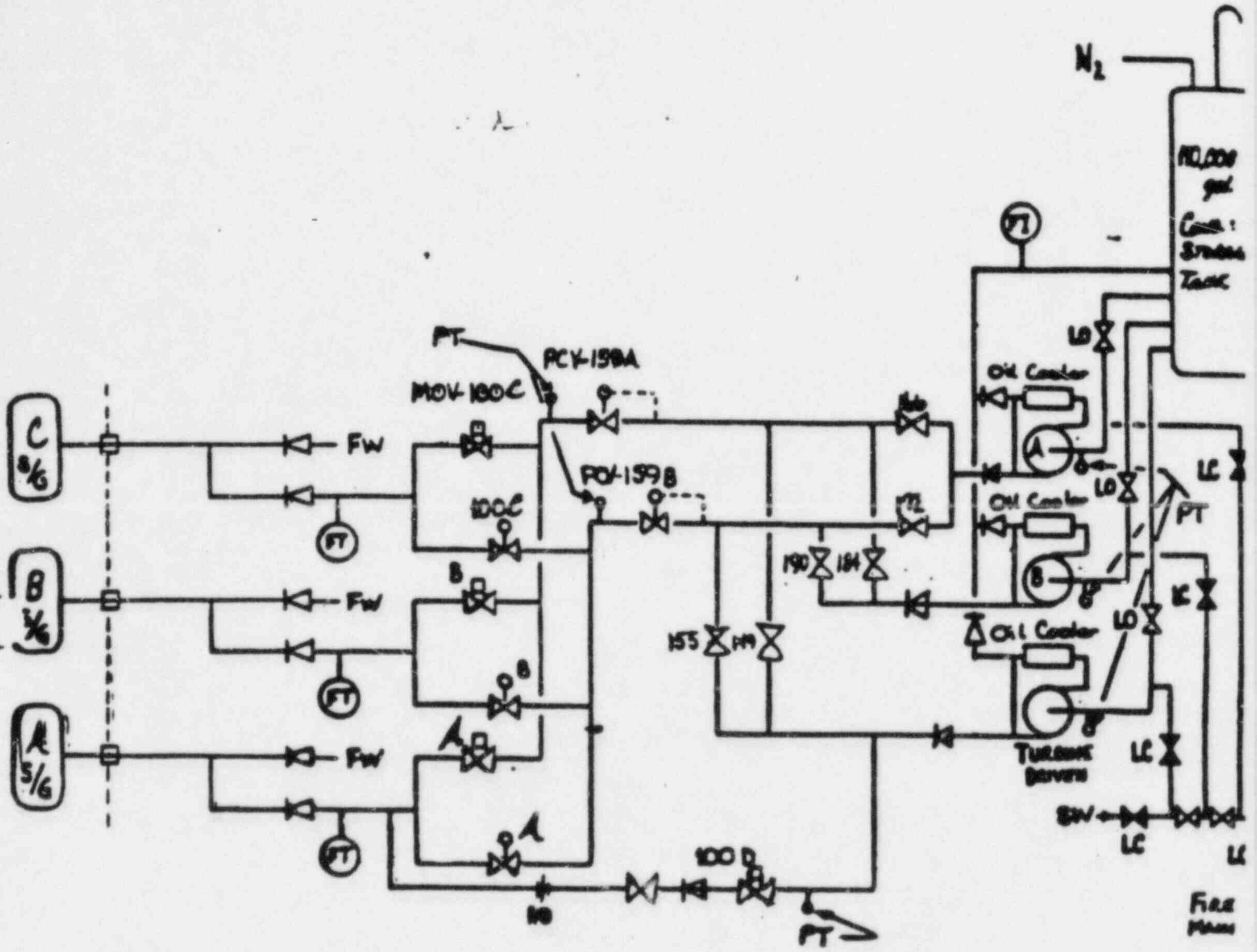
Explain why the load shedding feature associated with a degraded or undervoltage condition is defeated after an Emergency 4160 VAC Bus is isolated from the Reserve Station Transformer and the diesel output breaker closes.

QUESTION 6.27 (1.25)

Describe how the High Steam Line Flow SI input varies and the parameter on which this program is based.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

AUXILIARY FEEDWATER SYSTEM



QUESTION 6.28 (1.50)

List all the methods, including the actuating device, by which the contents of the Lube Oil/EHC Pump Facility 'Deluge' system can be dumped. Include any appropriate setpoints.

QUESTION 6.29 (1.00)

List all the electrical inputs to the starting circuitry that must be satisfied to successfully start a Reactor Coolant Pump.

QUESTION 6.30 (1.50)

Describe the improvements [illegible] and reliability of the electrical distribution system following a Unit 1 Trip due to the installation of Unit 1 Main Generator Breaker G12.

QUESTION 6.31 (2.00)

Answer the following questions regarding a DEGRADED voltage condition on an Emergency 4160 VAC Bus.

a) A degraded voltage condition occurs when ____ out of ____ relays detect <_____% nominal voltage and initiates a ____ second timer. (Fill in the blanks) (.75)

b) Describe the sequence of events from the timing out of the timer until all required loads are reenergized on the Emergency Bus. (Be sure to list any loads NOT shed) (1.25)

(***** END OF CATEGORY 06 *****)

QUESTION 7.01 (1.00)

Which of the following is NOT required operator action upon loss of power range channel N-41 during Mode 1 operations?

a. Place the test switch for the OT delta T to the TEST position for the failed channel.

b. Place or verify the control rods are in MANUAL control.

c. Place the test switch for the OP delta T to the TEST position for the failed channel.

d. Remove the 'Control Power' fuses for N-41 at the power range drawers.

QUESTION 7.02 (1.00)

Which of the following is an INAPPROPRIATE operator action if, during refueling, the containment gas and particulate radiation monitors alarm?

a. Stop the containment purge supply and exhaust fans.

b. Close the purge and exhaust fan MOVs.

c. Stop the containment air recirculation fans.

d. Start the containment iodine filtration fans.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.03 (1.00)

Given the following procedural steps, select the choice that provides the proper sequence for paralleling a diesel generator to a live bus.

1. Turn the synchronizer switch on.
2. Start the emergency diesel generator.
3. Turn off the synchronizer switch.
4. Operate the governor control switch so the synchroscope moves slowly in the fast direction.
5. Close the generator output breaker when the synchroscope needle passes the 11 o'clock position.
6. Operate the exciter control switch to match incoming voltage to running voltage.

   a. 1,2,4,5,6,3

   b. 2,1,6,4,5,3

   c. 2,6,1,4,5,3

   d. 1,2,6,5,4,3

   e. 2,1,4,5,6,3

QUESTION 7.04 (1.00)

What is the criterion used to verify the RCS is solid when placing the RCS in a solid water condition per OP-3.4?

   a. A high pressure spike occuring on the pressurizer pressure instruments.

   b. When spraying the pressurizer no longer will decrease RCS pressure.

   c. When indication of flow to the PRT through the pressurizer PORVs is verified.

   d. When an increase in letdown flow has been verified.

QUESTION 7.05 (1.00)

Which of the following is the PREFERRED method of providing a means of decay heat removal during a refueling outage with the reactor vessel head removed if all RHR pumps are lost?

a. Using natural circulation by feeding the SGs with AFW and bleeding them using SG blowdown.

b. By lining up the LHSI pumps to recirculate flow through the RWST and the reactor vessel.

c. By lining up the refueling purification system to recirculate water from the reactor cavity to the spent fuel pit using the spent fuel coolers for cooldown.

d. By feeding the RCS using a LHSI pump and bleeding it from a SG primary manway.

QUESTION 7.06 (1.00)

Which of the following indications require SI reinitiation following a spurious SI that has been secured?

a. RCS pressure at 1950 psig.

b. RCS subcooling at 35 degrees F.

c. Pressurizer level at 17%.

d. All steam generator levels at 15%.

QUESTION 7.07 (1.00)

Prior to transferring rod control from MANUAL to AUTO during a reactor startup, Tave and Tref should be within ________ degree(s) F.

a. 1

b. 3.5

c. 5

d. 10

QUESTION 7.08 (1.00)

How are the trip bistables of a failed power range detector placed in the trip condition?

a. Place the applicable bistable test switch in the 'Test' position in the Reactor Protection Cabinet.

b. Remove the applicable control and instrument power fuses on the power range drawers.

c. Place the applicable Power Mismatch Bypass switch to the failed position at the Miscellaneous Control and Indication Panel.

d. Place the applicable Comparator Channel Defeat switch to the failed channel position at the Detector Current Comparator Panel.

QUESTION 7.09 (1.00)

During a natural circulation cooldown, what is the preferred method of reactor coolant system depressurization?

a. Open the normal spray valve.

b. Open the auxiliary spray valve.

c. Open the pressurizer PORVs.

d. Open the reactor vessel vent valves.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.10 (1.00)

How do you eliminate a Hydrogen buble that has formed in the reactor vessel?

a. Increase pressurizer temperature above core thermocouple readings.

b. Inject oxygen into the reactor coolant system via the chemical and volume control system.

c. Maximize coolant flow by running all reactor coolant pumps, increase letdown flow to 120 gpm, and place the cation bed demineralizer in service in parallel with the mixed bed demineralizer.

d. Vent the reactor vessel head.

QUESTION 7.11 (1.00)

Which of the following is a 10 CFR 20 exposure limit?

a. 5 rem/year-whole body.

b. 1 rem/quarter-whole body.

c. 18.75 rem/quarter-hands.

d. 7 rem/quarter-skin of whole body.

QUESTION 7.12 (1.00)

If you are in a 100 mRad/hour gamma field for 45 minutes, what is your dose in mREM after 45 minutes?

a. 45

b. 75

c. 450

d. 750

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.13 (1.00)

Which of the following radiation exposures would inflict the greatest biological damage to man?

a. 1 Rem of GAMMA.

b. 1 Rem of ALPHA.

c. 1 Rem of NEUTRON.

d. NONE of the above; they are all equivalent.

QUESTION 7.14 (1.00)

Which action below should the operator perform if criticality is NOT achieved before the upper limit of the ECP is reached on a reactor startup?

a. Insert Bank D control rods to the bottom.

b. Stop control bank withdrawal and approach criticality via boron dilution.

c. Proceed with the startup utilizing an Inverse Count Rate Plot from the time the ECP upper limit is reached until the time criticality is reached.

d. Proceed with the startup utilizing a pull-and-wait approach to criticality.

QUESTION 7.15 (1.00)

When is control rod cooling air required to be operating?

a. Only if the control rod mechanism is energized when the plant temperature is above 350 F.

b. Whenever the plant temperature is above 350 F, regardless of the rod mechanism status.

c. During any plant cooldown evolution below 350 F.

d. Whenever the individual rod position indication system or any rod drive mechanism is energized.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.16 (1.00)

How is the boron concentration equalized between the pressurizer and the rest of the reactor coolant system?

a. Charging pure water or borated water to the PZR through the aux. spray line as required.

b. Charging a blended flow equal to the Reactor Coolant System boron concentration into the Reactor Coolant System to force more of the RCS inventory up the surge line and into the PZR.

c. Taking MANUAL control of the Pressurizer Level Control System and then increasing and decreasing the pressurizer level as required in order to get an exchange between the RCS and PZR water inventory.

d. Operating the pressurizer heaters and spray valves as required.

QUESTION 7.17 (1.00)

The administrative differential temperature limit between the pressurizer and its spray is ______ degrees F delta T.

a. 150

b. 200

c. 260

d. 320

QUESTION 7.18 (1.00)

It is required to exit a radiation area and report to HP whenever your 0-200 mr self-reading dosimeter reaches ______ mr.

a. 100

b. 125

c. 150

d. 175

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.19 (1.00)

An RWP number is necessary to obtain which of the following?

a. A dose card.

b. A TLD.

c. A self-reading dosimeter.

d. Your current quarterly dose.

QUESTION 7.20 (1.00)

During recovery from a refueling outage, what tool is utilized to relatch the RCC elements to their drive shafts?

a. The same tool used for unlatching the drive shafts.

b. A special drive shaft latching tool.

c. A new fuel handling tool.

d. A spent fuel handling tool.

QUESTION 7.21 (1.00)

Which of the following actions constitutes entry into Mode 6?

a. Increasing the boron concentration to 2000 ppm.

b. Decreasing reactor coolant temperature to less than 140 F.

c. Detensioning the first reactor vessel head stud.

d. Decreasing keff to less than 0.95.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.22 (1.00)

If one of the circulating water pumps trips during a radioactive liquid waste release, which action below is required?

a. The release must be terminated immediately.

b. No action is required if there is at least one other circulating water pump running.

c. Attempt to restart another circulating water pump. If this fails, then terminate the release.

d. Two circulating water pumps must be restarted for dilution in order to continue the release.

QUESTION 7.23 (1.00)

A fire found in the 4160 switchgear is best extinguished with a fire extinguisher marked with which symbol below?

a. A

b. B

c. C

d. D

QUESTION 7.24 (1.00)

Which Gai-Tronics channel is reserved for emergencies?

a. channel 1.

b. channel 3.

c. channel 5.

d. channel 7.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.25 (.50)

For a reactor startup, it is required by the procedure to go critical while the power indication is still in the source range. TRUE or FALSE?

QUESTION 7.26 (.50)

All radioactive gaseous releases must be made by a Licensed Senior Reactor Operator. TRUE or FALSE?

QUESTION 7.27 (.50)

Either the Health Physics Department OR the Operations Department may initiate a radioactive waste discharge record for a gaseous radwaste release. TRUE or FALSE?

QUESTION 7.28 (.50)

Will adverse containment conditions cause affected control room indications to indicate HIGHER or LOWER than actual conditions?

QUESTION 7.29 (.50)

A rod control logic cabinet internal failure can be verified in the control room. TRUE or FALSE?

QUESTION 7.30 (1.00)

What is the immediate make-up lineup upon loss of refueling cavity level during refueling?

QUESTION 7.31 (1.00)

What should the operator do about an SI signal if it exists during a loss of all AC power?

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.32 (1.00)

What two conditions constitute adverse containment conditions?

QUESTION 7.33 (1.00)

What are the required operator actions if both source range instruments fail when power is >P-6 and <P-10 during a power increase to 100% power?

QUESTION 7.34 (1.00)

List the immediate operator actions for an Anticipated Transient Without Trip.

QUESTION 7.35 (3.25)

List the immediate operator actions for the Reactor Trip or Safety Injection procedure.

QUESTION 7.36 (1.00)

List the whole body administrative limits per calender quarter that can be achieved (a) without any additional approval and
(b) with the highest level of approval.

QUESTION 7.37 (1.00)

List four of the critical conditions required to be recorded during a startup when 1 X 10E-8 amps is attained.

QUESTION 7.38 (1.00)

List five pieces of information an RWP will tell you what you will need to know in order to work in the applicable restricted area.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.39 (1.25)

List five surveillance items that must be done by operators to ensure Tech Spec requirements are met in the event of a loss of both units PRODAC-250 Computers.

QUESTION 7.40 (1.00)

If instrument air is lost outside the containment and the plant is solid, what are your required immediate actions?

(***** END OF CATEGORY 07 *****)

QUESTION 8.01 (1.00)

In performing an Emergency Work Request outside r rmal work hours, which one of the following is NOT a Shift Supervisor function?

a. Deciding who will perform the emergency maintenance based on exposure records, work requests and manpower availability.

b. Attaching an Equipment History/Failure Analysis form to the Emergency Work Order.

c. Identifying the emergency item.

d. Entering the emergency item into the Work Planning and Tracking System.

e. Notification of the SRO on call and the craft foreman on shift.

QUESTION 8.02 (1.00)

While in mode 3 on Unit 2 a planned boron dilution is completed at 0200. At 0230 boron dilution valves 2-CH-140 and 2-CH-160 are found to have been closed but not locked (not secured). Which one of the following is NOT a required action?

a. Suspend all operations involving positive reactivity changes.

b. Lock, seal or otherwise secure the valves in the closed position.

c. Verify that the shutdown margin is greater than or equal to 1.77% delta k/k.

d. Initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boron solution.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.03 (1.00)

Which one (1) of the following is NOT a 'Limiting Safety System Setting' trip setpoint?

a. Reactor coolant pump bus underfrequency '>' or '=' to 56.1 Hz.

b. Turbine trip - low trip system pressure '>' or '=' to 45 psig.

c. Power range, neutron flux, high negative rate '<' or '=' to 5% of rated thermal power with a time constant '>' or '=' to 2 seconds.

d. Low pressurizer water level '>' or '=' to 18% of instrument span.

QUESTION 8.04 (1.00)

Tech Spec 3.8.1.1 requires two separate and independent diesel generators. It further specifies certain diesel support equipment as a limiting condition for operation. Which of the following is NOT part of that LCO?

a. Each with a separate 125-volt battery bank, charger and inverter.

b. Each with a separate day tank containing a minimum of 750 gallons of fuel.

c. A fuel storage system containing a minimum of 45,000 gallons of fuel.

d. A separate fuel transfer pump.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.05 (1.00)

If one overtemperature delta T instrument channel is INOPERABLE and placed in the trip condition, which statement below is correct?(In Mode 4,3 Loops)

a. Startup may proceed but NO mode change is permitted.

b. Startup may proceed but no mode changes beyond mode 2 are allowed.

c. Power operation may proceed until performance of the next required channel functional test.

d. Power operation may proceed as long as the Quadrant Power Tilt Ratio is monitored at least once per 12 hours.

e. Startup and power operation may proceed if thermal power is restricted to less than or equal to 75% of rated thermal power.

QUESTION 8.06 (1.00)

With one Unit 2 rod position indicator inoperable in mode 1, the non-indicating rod is moved from its last known position more than 24 steps. As an alternate to reducing power, the operator may determine that rods' position using the movable incore detectors if done within a certain time interval after the last rod movement. What is that time interval?

a. immediately

b. 15 minutes

c. 30 minutes

d. 1 hour

e. 8 hours

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.07 (1.00)

The Unit 1 boron injection tank will be INOPERABLE under which of the following conditions?

a. If it contains 950 gallons of appropriate water.

b. If it is borated to 25,000 ppm.

c. If its temperature is 160 deg F.

d. If Hydrogen overpressure is 13 psig.

QUESTION 8.08 (1.00)

Controlled leakage as defined by Unit 1 Technical Specifications refers to:

a. Letdown flow.

b. Liquid radwaste release flow.

c. Reactor coolant pump seal water flow.

d. Excess letdown flow.

e. S/G tube leakage.

QUESTION 8.09 (1.00)

The Unit 1 reactor coolant system pressure exceeds 2735 psig when in mode 3. According to technical specifications, pressure must be restored within acceptable limits within what time frame given below?

a. 5 minutes.

b. 15 minutes.

c. 30 minutes.

d. one hour.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.10 (1.00)

The technical specification requiring the Unit 1 accumulator isolation valves be opened applies in which mode(s) listed below?

a. In mode 1 only.

b. In mode 1 only and only above 50% power.

c. In modes 1 and 2 only.

d. In modes 1,2 and 3 if pressurizer pressure is above 1000 psig.

QUESTION 8.11 (1.00)

If control power is lost to a Unit 2 pressurizer power operated relief valve while in mode 1, which statement below is correct?

a. Tech specs require no action provided another PORV is operable and all pressurizer code safety valves are operable.

b. Tech specs require the power supply to be removed from the associated block valve after verifying it to be open, if the PORV is not operable within 1 hour and continuous operation is desired.

c. Tech specs require the associated block valve to be shut and its power removed if the PORV is not made operable within one hour and continuous operation is desirable.

d. Tech specs require action to be initiated within one hour to place the plant in at least hot standby within the following hour if the PORV is not made operable.

QUESTION 8.12 (1.00)

What are the technical specification maximum heatup rates in any one hour period for the reactor coolant system (including the pressurizer)?

a. 100 degrees F for units 1 & 2.

b. 100 degrees F for unit 1 and 200 degrees F for unit 2.

c. 200 degrees F for unit 1 and 100 degrees F for unit 2.

d. 200 degrees F for units 1 & 2.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.13 (1.00)

If the lowest operating loop Tave drops below ______ in mode 1, Unit 1 & 2 tech specs allow ______ to restore Tave within the limit before proceeding to place the unit in hot standby.

a. 541 degrees F, 30 minutes

b. 541 degrees F, 15 minutes

c. 547 degrees F, 30 minutes

d. 547 degrees F, 15 minutes

QUESTION 8.14 (1.00)

Turbine overspeed protection is required by technical specifications because of which situation below?

a. Turbine components become missles which may penetrate the turbine casing & allow higher than allowed off-site doses with an assumed maximum allowable fuel cladding and primary-to-secondary leak.

b. Turbine components become missles, which may damage safety-related equipment.

c. The reactor thermal power exceeds limits in the unit's license.

d. The main steam trip valves exceed the closure time assumed in the accident analyses.

QUESTION 8.15 (1.00)

Which action below is required when the fire supression spray/sprinkler system for the auxiliary building, component cooling pump area, elev. 244'6 is declared INOPERABLE?

a. Commence a unit shutdown for the applicable unit within one hour.

b. Establish an hourly fire watch patrol for the affected area with backup water fire supression equipment within one hour.

c. Establish a continuous fire watch for the affected area with backup water fire suppression equipment within one hour.

d. Log ambient temperature readings for the affected area hourly and route an equivalent capacity fire hose to the area.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.16 (1.00)

Which one of the following is CORRECT concerning Unit 1 tech spec treatment of Fq(z)?

a. Since Fq(z) is not measureable, limits are placed on AFD.

b. Measureable limits are placed on Fq(z) only above 50% rated thermal power.

c. The limit placed on Fq(z) by tech specs varies as a function of core height.

d. The limit placed on Fq(z) by tech specs varies as a function of region average burnup.

QUESTION 8.17 (1.00)

Unit 2 tech spec 3.1.3.5 requires all shutdown rods to be fully withdrawn. Which one of the following is CORRECT concerning this specification?

a. It is applicable in modes 1,2 and 3 only.

b. Fully withdrawing all but one of the shutdown rods that were partially inserted will satisfy the action statement.

c. If one rod is not and cannot be fully withdrawn it must be declared inoperable.

d. The actions required by the action statement must be satisfied in 15 minutes or less.

QUESTION 8.18 (1.00)

A Unit 2 control rod is determined to be INOPERABLE in mode 2 as a result of excessive friction. Tech Specs require which action below in 1 hour?

a. Be in hot standby.

b. Restore the rod to operable status.

c. Position the remainder of the rods in that group to within '+' or '-' 12 steps of the inoperable rod.

d. Determine that the tech spec shutdown margin requirement is satisfied.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.19 (1.00)

A quarterly tech spec surveillance requirement may be extended up to ______ days without declaring the component inoperable due to the surveillance testing not being performed.

a. 9

b. 23

c. 32

d. 41

QUESTION 8.20 (1.00)

ADM 5.7 entitled 'Correcting Data on Completed Procedures' makes which of the following provisions?

a. It is permissible to make corrections to data which has been recorded. This must be performed in the following manner: draw a line through the erroneous data and record correct data. Initial and date the correction.

b. The cognizant supervisor shall list any available substantiating information.

c. The individual who conducted the test and the cognizant supervisor shall attest to the corrected data by affixing their signature and the date.

d. The station nuclear safety and operating committee shall review the circumstances and the chairman shall sign and date the form when their review is completed.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.21 (1.00)

According to Unit 2 tech specs, which of the following statements regarding the site fire brigade is correct?

a. Shall be composed of at least 10 members.

b. Shall not include any of the minimum shift crew required by table 6.2-1 (minimum shift manning).

c. Shall be responsible for the control room command function until the fire emergency is secured.

d. May NOT be less than the minimum requirement even to accomodate unexpected absences.

QUESTION 8.22 (1.00)

Which of the following is true of tagging of systems and/or components according to ADM 14.0?

a. Special order tags, form 625.6 are grey tags.

b. The operator shall be responsible for placing all tags.

c. Maintenance personnel to whom the tagging record is issued will independently review the tag selection and authorize the operator to place the tags.

d. When more than one department is to work on the same equipment, a separate tag will be placed for each department unless a fifteen minute headway tag has been authorized.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.23 (1.00)

Which one of the following statements is true of jumpers as controlled by ADM 14.1, 'Jumpers (temporary modifications)'?

a. Temporary hose connections necessary to perform a test are defined as constituting a jumper if their use is described in the text or drawings of the FSAR.

b. In order to control their use, ADM 14.1 only permits the use of jumpers as described in either an MOP or an MMP.

c. Jumpers not controlled by any approved procedure shall only be used with the shift supervisor's prior knowledge and approval.

d. For those jumpers that are installed by an approved procedure, a jumper log form shall be initiated and all pertinent data associated with the installation recorded.

QUESTION 8.24 (1.00)

Assume both Units are in cold shutdown for refueling and maintenance. Incore instruments have been withdrawn to their storage position and the incore instrumentation thimble retraction has just been completed. All access doors to the reactor cavity are locked. Flooding of the refueling cavity in preparation for refueling was started six hours ago. It has just been determined that the water level in the refueling cavity is decreasing slowly. Assume you are the senior licensed SRO on shift. You have decided to send an auxilaiary operator into the reactor cavity in an effort to locate the source of leakage. Which of the following would be the anticipated dose rate range in the cavity?

a. < 30 mr/hr.

b. 30-300 mr/hr.

c. 3-30 R/hr.

d. 300-3,000 R/hr.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.25 (1.00)

Which one of the following is a condition requiring stoppage of all work and immediate evacuation of containment according to the precautions and limitations in 1-OP-4.1, 'Controlling Procedure for Refueling'?

a. The 'hi flux at shutdown' alarm is actuated during fuel movement.

b. Loss of audible neutron count rate (< two tones per minute) after offloading 3/4 of the reactor core.

c. The station evacuation alarm sounds.

d. Declaration of an 'alert'.

QUESTION 8.26 (1.00)

Which of the following conditions for refueling must be met if refueling is to continue?

a. The flow rate of reactor coolant through the reactor coolant system shall be '>' or '=' to 3000 gpm whenever a reduction in RCS boron concentration is being made.

b. Valve 1-CH-217 (PG to blender isolation) shall be locked in the open position to assure an adequate supply of borated water in the event emergency boration is required.

c. The following boron injection flow paths shall be operable:
   (1) a flow path from the boric acid tanks via a boric acid transfer pump through a charging pump to the RCS, and
   (2) the flow path from the refueling water storage tank via a charging pump to the RCS.

d. Two boric acid transfer pumps shall be operable if neither charging pump is operable.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.27 (1.00)

Which of the following is the principal candidate for relief of the Interim Station Emergency Manager in the event of an emergency at the station.

a. Assistant station manager (ops)

b. Superintendent of operations

c. Station manager

d. Manager nuclear operations and maintenance.

QUESTION 8.28 (1.00)

Which of the following require activation of both the TSC and OSC?

a. Either an unusual event, alert, site area emergency or general emergency.

b. Only an alert, site area emergency, or general emergency.

c. Only a site area emergency or general emergency.

d. Only a general emergency.

QUESTION 8.29 (1.00)

If an operator is returning to shift after a three week vacation, he is required to read and initial the logs for the previous ______. (choose one)

a. 1 day

b. 7 days

c. 14 days

d. 21 days

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.30 (1.00)

Following a reactor trip where the reason for the trip is clearly understood and corrected and no significant malfunctions of safety-related or important equipment occurred, who is responsible for making the decision to restart the reactor?

a. Shift supervisor.

b. Superintendent of operations (or SRO on call).

c. Assistant station manager (S & L).

d. Station manager.

QUESTION 8.31 (1.00)

Match the following tech spec defined leakage with the associated leak rate limit. Assume normal operating reactor coolant system temperature and pressure. The limits may be used more than once.

	Leakage		Leak rate limit
a)	Pressure boundary leakage.	1)	0 gpm
b)	Controlled leakage.	2)	1 gpm
c)	Identified leakage.	3)	5 gpm
d)	Unidentified leakage.	4)	10 gpm
e)	Primary-to-secondary leakage through one SG.	5)	30 gpm
		6)	50 gph
		7)	500 gpd

QUESTION 8.32 (1.00)

To prevent entering a technical specification action statement on Unit 1, the quadrant power tilt ratio shall not exceed ______ when reactor power is above 50%. If this limit is exceeded, then an extended temporary QPTR limit of ______ is allowed during efforts to restore QPTR to normal levels.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.33 (1.00)

Complete the following statements concerning refueling operations by filling in the correct number.

a) At least ______ feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

b) During movement of irradiated fuel in the reactor pressure vessel the reactor shall have been subcritical for at least ______ hours.

QUESTION 8.34 (1.00)

Provide the minimum number of individuals required by tech specs for the following positions to operate both units at full power.

a) ______ shift supervisor(s)

b) ______ SRO(s).(NOT including the shift supervisor(s))

c) ______ RO(s)

d) ______ AO(s)

e) ______ STA(s)

QUESTION 8.35 (2.00)

Fill in the blanks below:
With the shutdown margin less than ______%, immediately initiate and continue boration at ____gpm or greater of ____ppm boric acid solution or equivalent until ______________________.

QUESTION 8.36 (2.00)

ADM 19.3 (shift conduct, relieving the shift) states, 'the shift supervisor shall perform the following:'. (Supply the six items listed in the procedure for the unit 1 SRO to REVIEW and SIGN at the beginning of the shift)

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.37 (1.00)

When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable limiting condition for operation, provided two conditions are met. List these TWO conditions.

QUESTION 8.38 (1.00)

ADM 20.9 indicates that entry into containment during reactor operation exposes personnel to four distinct hazards. List these FOUR hazards.

(***** END OF CATEGORY 08 *****)
(************* END OF EXAMINATION ***************)

ANSWERS -- NORTH ANNA 1&2 -85/10/15-DEAN, W M

ANSWER 5.01 (1.00)

c

REFERENCE
General Physics, HT & FF, Section 3.2
WBN, HT & FF, p. 13

002/000-K5.01 (3.1/3.4)

ANSWER 5.02 (1.00)

d

REFERENCE
Surry, TS 3.12-16 and 17
HBR, TS, p. 3.10-13
NAPS, TS, p. B 3/4 2-4

ANSWER 5.03 (1.00)

c

REFERENCE
General Physics, HT & FF, pp. 239 and 240

ANSWER 5.04 (1.00)

a

REFERENCE
DPC, FNRE, pp. 200 and 221

ANSWER 5.05 (1.00)

c

REFERENCE
NUS, Vol 4, pp G-8

ANSWER 5.06 (1.00)

d

REFERENCE
NUS, Nuclear Energy Training - Reactor Operation
VEGP, Training Text, Vol. 9, Ch. 21
DPC, Fundamentals of Nuclear Reactor Engineering

001/010-K5.13

ANSWER 5.07 (1.00)

b

REFERENCE
NUS, Nuclear Energy Training - Reactor Operation

ANSWER 5.08 (1.00)

a

REFERENCE
NUS, Nuclear Energy Training - Reactor Operation, p. 9.2-5

ANSWER 5.09 (1.00)

d

REFERENCE
General Physics, Heat Transfer Thermodynamics and Fluid Flow,
pp. 145 - 148

ANSWER 5.10 (1.00)

d

REFERENCE
NUS, Nuclear Energy Training - Plant Performance, pp. 10-1.8 & 9

ANSWER 5.11 (1.00)

d

REFERENCE
Westinghouse Reactor Physics, pp. I-5.63 - 76
HBR, Reactor Theory, Sessions 38 and 39
DPC, Fundamentals of Nuclear Reactor Engineering, Section VI

001/000-K5.39 (3.5/4.1)

ANSWER 5.12 (1.00)

b

REFERENCE
General Physics, HT & FF, p. 229

002/000-K5.01 (3.1/3.4)

ANSWER 5.13 (1.00)

b

REFERENCE
Westinghouse Reactor Physics, pp. I-5.26 & 27

ANSWER 5.14 (1.00)

d

REFERENCE
Westinghouse Reactor Physics, Section I-5, MTC and Power Defect
DPC, Fundamentals of Nuclear Reactor Engineering

002/000-K5.02 (3.3/3.6)

ANSWER 5.15 (1.00)

c

REFERENCE
Westinghouse Reactor Physics, pp. I-5.2 - 16

ANSWER 5.16 (1.00)

b

REFERENCE
WBN, TS, p. B 3/4 1-1
HBR, TS, p. 3.10-10
CAT, TS, p. B 3/4 1-1
NAPS, TS, p. B 3/4 1-1

ANSWER 5.17 (1.00)

c

REFERENCE
General Physics, HTFF - Fluid Flow Applications for Systems and Components

ANSWER 5.18 (1.00)

b

REFERENCE
WBN, TS, p. B 3/4 2-1
HBR, TS, p. 3.10-17
NAPS, TS, p. B 3/4 2-5

ANSWER 5.19 (1.00)

b

ANSWERS -- NORTH ANNA 1&2 -85/10/15-DEAN, W M

REFERENCE
HBR, Reactor Theory, Session 42, pp. 4 & 10

ANSWER 5.20 (1.00)

a

REFERENCE
HBR, Reactor Theory, Sessions 41 and 42
DPC, Fundamentals of Nuclear Reactor Engineering, pp. 121 and 122

004/000-K5.08 (2.6/3.2)

ANSWER 5.21 (1.00)

c

REFERENCE
NUS, Reactor Theory

ANSWER 5.22 (1.00)

c

REFERENCE
NUS, Reactor Theory

ANSWER 5.23 (1.00)

a

REFERENCE
DPC, Fundamentals of Nuclear Reactor Engineering, p. 138

001/000-K5.02 (2.9/3.4)

ANSWER 5.24 (1.00)

b

ANSWERS -- NORTH ANNA 1&2 -85/10/15-DEAN, W M

REFERENCE
General Physics, HT & FF, p. 320

ANSWER 5.25 (1.00)

c

REFERENCE
DPC, Fundamentals of Nuclear Reactor Engineering, p. 170

001/000-K5.13 (3.7/4.0)

ANSWER 5.26 (1.00)

b

REFERENCE
Westinghouse Reactor Physics, Sects 3 and 5
NUS, Nuclear Energy Training, Units 6 and 8
NA OP 1.4

ANSWER 5.27 (1.00)

a

REFERENCE
Westinghouse Nuclear Training Operations

ANSWER 5.28 (1.00)

b

REFERENCE
NAPS, Instructor Guide NCRODP-95.2, Mitigating Core Damage

ANSWER 5.29 (1.00)

b

REFERENCE
NAPS, Instructor Guide NCRODP-95.2, Mitigating Core Damage

ANSWER 5.30 (1.50)

a. FALSE (0.5)
b. FALSE (0.5)
c. FALSE (0.5)

REFERENCE
Westinghouse Nuclear Training Operations, pp. I-3.9 - 15

ANSWER 5.31 (1.00)

d

REFERENCE
Westinghouse Nuclear Training Operations, p. I-5.76

ANSWER 5.32 (1.00)

c

REFERENCE
Westinghouse Nuclear Training Operations, pp. I-4.19 - 21

ANSWER 5.33 (1.00)

a

REFERENCE
General Physics, HT & FF, Chapter 1

ANSWER 5.34 (2.00)

a. REMAIN THE SAME (0.5)
b. DECREASE (0.5)
c. INCREASE (0.5)
d. DECREASE (0.5)

REFERENCE
Steam Tables

010/000-K5.02 (2.6/3.0)

ANSWER 5.35 (2.00)

a. DECREASE (0.5)
b. INCREASE (0.5)
c. INCREASE (0.5)
d. DECREASE (0.5)

REFERENCE
General Physics, HT & FF, pp. 319 - 334

ANSWER 5.36 (2.00)

a. TRUE (0.5)
b. TRUE (0.5)
c. FALSE (0.5)
d. TRUE (0.5)

REFERENCE
NUS, Nuclear Energy Training - Plant Performance, p. 5-3.2
General Physics, HT & FF, pp. 155, 319, and 320
VEGP, Question Bank, #752-a and # 336

ANSWER 5.37 (.50)

TRUE

REFERENCE
General Physics, HT & FF, p. 176

002/000-K5.01

ANSWER 6.01 (1.00)

c

REFERENCE
NA NCRODP 91.1, 'ESF'

ANSWER 6.02 (1.00)

c

REFERENCE
NA NCRODP 88.1, 'RCS-PZR and Press. Relief'

ANSWER 6.03 (1.00)

d

REFERENCE
NA NCRODP 91.2, 'Containment and Containment Systems'

ANSWER 6.04 (1.00)

d

REFERENCE
FNP, RHR Lesson Plan, pp. 8 & 9
NA NCRODP 88.2, 'RHR System'

005/000-K4.07 (3.2/3.5)

ANSWER 6.05 (1.00)

c

REFERENCE
Farley SD, 'CVCS', pp 13
NA NCRODP 88.3, 'CVCS'

ANSWERS -- NORTH ANNA 1&2 -85/10/15-DEAN, W M

ANSWER 6.06 (1.00)

c and b

REFERENCE
NA NCRODP 91.1, 'ESF-Recirc Spray System'

ANSWER 6.07 (1.00)

a

REFERENCE
NA NCRODP 92.2, 'Service Water System'

ANSWER 6.08 (1.00)

a

REFERENCE
NA NCRODP 88.1, 'RCS-PZR and Press Relief'

ANSWER 6.09 (1.00)

a

REFERENCE
NA NCRODP 89.4, 'Feedwater Systems-Main Feed'

~~ANSWER 6.10 (1.00)~~

~~e~~

deleted

~~REFERENCE~~
~~NA NCRODP 88.1, 'RCS-Reactor Vessel/Core Construction'~~

ANSWER 6.11 (1.00)

c

REFERENCE
NA NCRODP 88.1, 'RCS-PZR and Press Relief'

ANSWER 6.12 (1.00)

d

REFERENCE
NA NCRODP 92.2, 'Service Water System'

ANSWER 6.13 (1.00)

c

REFERENCE
NCRODP 90.1, 'Basic Electrical Distribution'

ANSWER 6.14 (2.00)

a) No effect (+.5 ea)
b) Arm only
c) Arm and Actuate
d) No effect

REFERENCE
NA NCRODP 93.11, 'Steam Dumps'

ANSWER 6.15 (1.50)

a) 1-FP-P-2 (+.5 ea)
b) 1-FP-P-1
c) Both

REFERENCE
NCRODP 92.1, 'Fire Protection'

ANSWER 6.16 (1.50)

a) False
b) False
c) True

REFERENCE
NA NCRODP 93.2, 'Excore NIS' pp 14-22

ANSWER 6.17 (1.50)

a) Not allowed (+.5 ea)
b) Allowed
c) Allowed

REFERENCE
NA NCRODP 91.2, 'Containment Isolation'

ANSWER 6.18 (1.50)

a) DCI (+.5 ea)
b) DCI
c) CRE

REFERENCE
NCRODP 90.4, 'EDG' Handout H-2.4

ANSWER 6.19 (1.50)

a) 3 (+.5 ea)
b) 4
c) 2

REFERENCE
NA NCRODP 91.2, 'Containment Atmospheric Cleanup System'

ANSWER 6.20 (1.50)

a) 3 (+.5 ea)
b) 1
c) 4

REFERENCE
NCRODP 89.1, 'Main Steam System'

ANSWER 6.21 (1.50)

-S/G Press used for density compensation of steam flow, so if pressure fails high, steam flow input to the SGWLCS will increase (+1.0)

-Ws>Wf will cause FWRV to open, level will increase until level error counteracts the steam flow/feed flow mismatch (+.5)

REFERENCE
NA NCRODP 93.12, 'SGWLC and Protection'

ANSWER 6.22 (2.00)

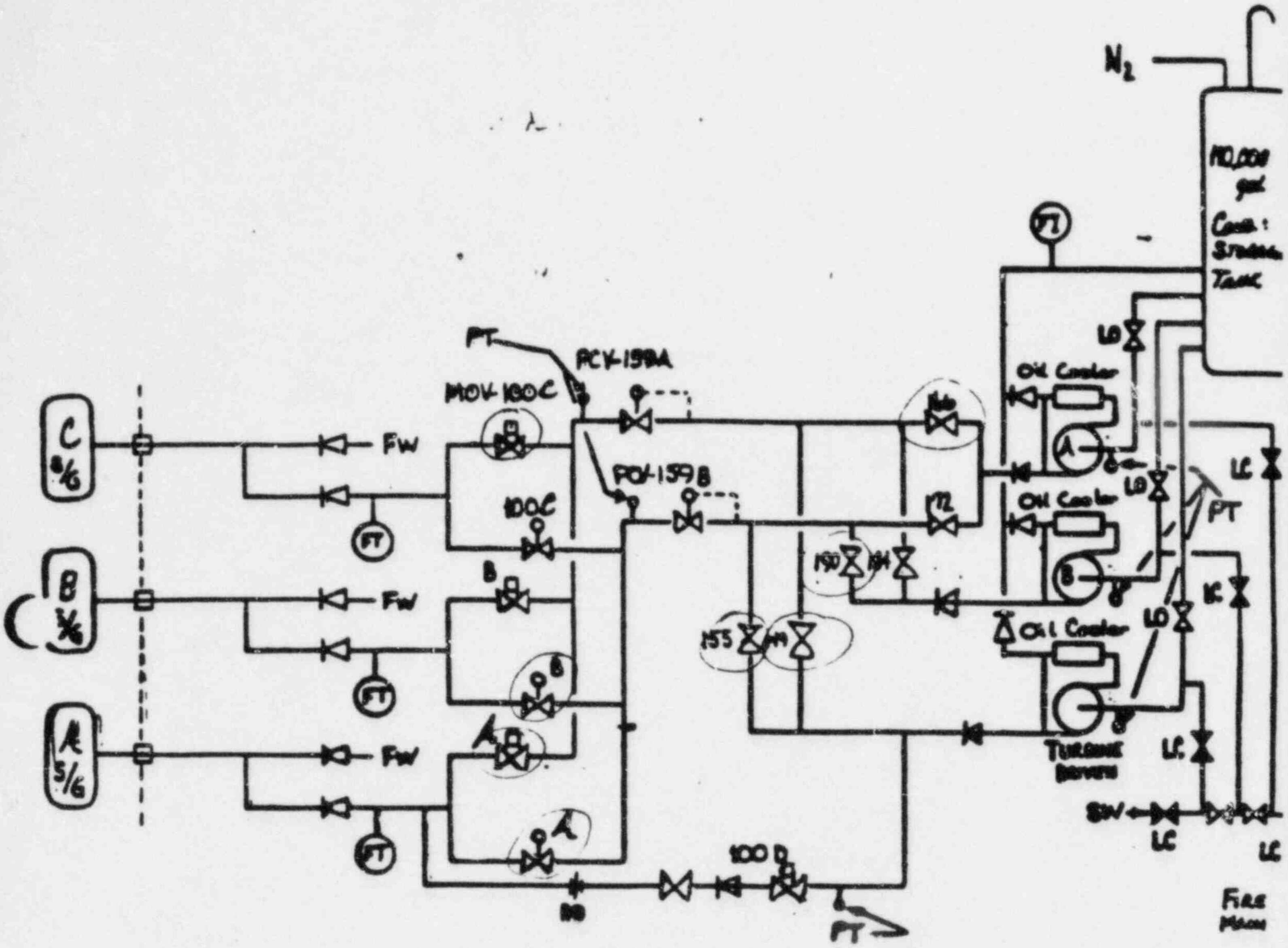
MOV-100C (+.25 ea)
MOV-100A
PCV-100B
PCV-100A
155
149
166
190
See Attached Drawing

REFERENCE
NA NCRODP 89.4, 'Feedwater Systems-AFW'

ANSWER 6.23 (1.50)

-RHR System Inlet (+.5 ea)
-Loop Drain Lines
-Differential Pressure taps for RCS flow indication

AUXILIARY FEEDWATER SYSTEM



REFERENCE
NA NCRODP 88.1, 'RCS-Piping and Instrumentation'

ANSWER 6.24 (1.00)

-Lower Delta P (8' vs 12-14' on Unit 1) (+.5 ~~ea~~
~~approx. 3 MW gain for Unit 2~~ deleted

REFERENCE
NA NCRODP 89.1, 'Main Stam System'

ANSWER 6.25 (1.75)

1) Reference Signal for comparison with actual steam flow (+.25 ea)
2) P-13
3) C-7
4) Feed water reg valve
5) C-5
6) Rod Control
7) Tref for comparison w/Tavg

REFERENCE
NCRODP 89.1, 'Main Steam System'

ANSWER 6.26 (1.00)

Prevents voltage drops during the diesel load sequence (+.5) from causing subsequent load shedding (+.5) negating the loading of the diesel.

REFERENCE
NCRODP 90.4, 'EDG'

ANSWER 6.27 (1.25)

40% setpoint from 0-20% (+.5) Turbine Power (+.25)
and linearly from 40-110% as Turbine Power goes from 20-100% (+.5)

REFERENCE
NCRODP 91.1, 'ESF-SI or ECCS'

ANSWER 6.28 (1.50)

±10°F

1) Auto Heat Actuated >190 degrees F (+.5)
2) Manual release on Control Room Fire Panel (+.25 ea)
3) " " " Deluge Panel in Track Bay
4) " " " Break Glass and Pull station next to the facility
5) " " " Deluge Valve

REFERENCE
NCRODP 92.1, 'Fire Protection'

ANSWER 6.29 (1.00)

-Loop stop/Bypass isolation valves positioned to allow flow (+.25 ea)
-Bearing lift pump running with >700 psig discharge
-no electrical faults on RCP motor
-locked rotor protection speed switch satisfied

REFERENCE
NA NCRODP 88.1, 'RCS-RCPs'

ANSWER 6.30 (1.50)

-Allows Unit 1 to have its normal station service busses supplied from its normal station service transformers backfed from 500 KV switchyard for most Unit 1 trips. (+1.0)

-This reduces probability of combined loading from both Units normal and emergency buses on the reserve station transformer (+.5)

REFERENCE
NA NCRODP 90.1, 'Basic Electrical Distribution'

ANSWER 6.31 (2.00)

a) 2 out of 3; 90%; 56 seconds (+.25 ea)

b) -diesel starts (+.15 ea correct response, .1 ea for correct order)
-emergency bus isolated from RS xfrmr
-all loads shedded except Chg pumps, LHSI pumps and 480 VAC loads
-output bkr shuts when diesel up to speed and voltage
-loads sequenced onto bus

REFERENCE
NA NCRODP 90.4, 'Emerg Diesel Generator'

ANSWER 7.01 (1.00)

b.

REFERENCE
NAPS 1-AP-4, pp. 3&8.

ANSWER 7.02 (1.00)

c.

REFERENCE
NAPS 1-AP-5.1, pp.8-9.

ANSWER 7.03 (1.00)

b.

REFERENCE
NAPS, 1-OP-6.2, p. 7.

ANSWER 7.04 (1.00)

d.

REFERENCE
NAPS, 1-OP-3.4, p. 13.

ANSWER 7.05 (1.00)

c.

REFERENCE
NAPS, 1-AP-11, p.9.

ANSWER 7.06 (1.00)

b.

REFERENCE
NAPS 1-ES-0.2, Foldout page.

ANSWER 7.07 (1.00)

a

REFERENCE
MNS OP/1/A/6100/01 p. 19.
CNS OP/1/A/6100/01, Encl. 4.1, 2.95.10.
NAPS 1-OP-2.1, p.8.

ANSWER 7.08 (1.00)

b

REFERENCE
MNS AP/2/A/5500/16, Case IV, p.9.
CNS AP/1/A/5500/16, Case IV, p.14.
NAPS 1-OP-4, p.8.

ANSWER 7.09 (1.00)

b

REFERENCE
MNS EP/2/A/5000/1.1, p.5.
CNS EP/1/A/5000/1A1, p.4.
NAPS 1-ES-0.3, p.5.

ANSWER 7.10 (1.00)

d

REFERENCE
MNS EP/2/A/5000/16.3
CNS EP/1//A/5000/2F3, p.7.
NAPS 1-FRP-I.3A, p.3.

ANSWER 7.11 (1.00)

c

REFERENCE
10 CFR 20.101

ANSWER 7.12 (1.00)

b

QF=1 for gamma

100(45/60)(1)=75

REFERENCE
10 CFR 20.

ANSWER 7.13 (1.00)

d

REFERENCE
10 CFR 20.

ANSWER 7.14 (1.00)

a

REFERENCE
NAPS, 1-OP-1.5, p.10.

ANSWERS -- NORTH ANNA 1&2 -85/10/15-DEAN, W M

ANSWER 7.15 (1.00)

b

REFERENCE
NAPS 1-OP-1.1.

ANSWER 7.16 (1.00)

d

REFERENCE
NAPS 1-OP-8.3, p.5.

ANSWER 7.17 (1.00)

c

REFERENCE
NAPS 1-OP-1.4, Attachment 1.

ANSWER 7.18 (1.00)

c

REFERENCE
Vepco GET, p.21.

ANSWER 7.19 (1.00)

c

REFERENCE
Vepco GET, p.41.

ANSWER 7.20 (1.00)

a

REFERENCE
NAPS 1-OP-4.1, p.41.

ANSWER 7.21 (1.00)

c

REFERENCE
NAPS 1-OP-4.1, p.23.

ANSWER 7.22 (1.00)

a

REFERENCE
NAPS 1-OP-22.11, p.4.

ANSWER 7.23 (1.00)

c

REFERENCE
Vepco GET, p.49.

ANSWER 7.24 (1.00)

c

REFERENCE
Vepco GET, p.49.

ANSWER 7.25 (.50)

FALSE.

REFERENCE
NAPS 1-OP-1.5, p.10.

ANSWER 7.26 (.50)

FALSE.

REFERENCE
NAPS 1-OP-23.2, p.4.

ANSWER 7.27 (.50)

TRUE.

REFERENCE
NAPS 1-OP-23.2, p.3.

ANSWER 7.28 (.50)

Higher.

REFERENCE
NAPS 1-EP-0 Foldout page.

ANSWER 7.29 (.50)

FALSE. (urgent failure alarm could be from logic or power cabinet failure)

REFERENCE
NAPS 1-AR-1, 1A-4.

ANSWERS -- NORTH ANNA 1&2 -85/10/15-DEAN, W M

ANSWER 7.30 (1.00)

From RWST through LPSH into hot legs.

REFERENCE
NAPS 1-AP-52, p.3.

ANSWER 7.31 (1.00)

Reset it.

REFERENCE
NAPS 1-ECA-0.1, p.2.

ANSWER 7.32 (1.00)

@ 0.5 points each:
1. Containment pressure at 19.7 psia.
2. Containment rad. levels at 40,000 R/hr.

REFERENCE
NAPS 1-EP-0, p.2.

ANSWER 7.33 (1.00)

Place both level trip switches in BYPASS. (Ops may continue.)

REFERENCE
NAPS 1-AP-4, p.4.

ANSWER 7.34 (1.00)

For 0.25 points each.
1. Attempt a reactor trip using both manual trip switches.
2. Attempt a turbine trip using both manual trip buttons.
3. Verify all AFW pumps are running.
4. Verify AFW flow to all SGs.

emergency borate

REFERENCE
NAPS 1-ECA-1, pp. 2&3.

ANSWER 7.35 (3.25)

For 0.25 points each.
1. Verify reactor trip.
2. Verify turbine trip.
3. Verify AC emergency busses energized.
4. Check if SI actuated.
5. Check charging/SI pump valve lineup.
6. Verify FW isolation.
7. Verify AFW flow.
8. Verify charging/SI flow.
9. Verify LHSI pumps running.
10. Verify containment phase A isolation.
11. Verify service water pumps running.
12. Verify RCS heat removal.
13. Check containment pressure.

REFERENCE
NAPS 1-EP-0,pp. 1-5.

ANSWER 7.36 (1.00)

For 0.5 points each.
(a) 750 mr/qtr
(b) 2750 mr/qtr

REFERENCE
NAPS Radiation Protection Manual, p.2.3-7.

ANSWER 7.37 (1.00)

Any 4 @ 0.25 points each:

1. Bank C position.
2. Bank D position.
3. Auct. High Tavg.
4. IR N35.
5. IR N36.
6. RCS boron concentration.

REFERENCE
NAPS 1.OP-1.5, p.12.

ANSWER 7.38 (1.00)

Any 5 @ 0.2 points each:

1. Job location.
2. Brief job description.
3. Radiological conditions at job site.
4. Anti C requirements.
5. Respiratory requirements.
6. Dosimetry requirements.
7. Necessary HP coverage.
8. Any special instructions.

REFERENCE
Vepco GET, p.41

ANSWER 7.39 (1.25)

Any 5 @ 0.25 points each:

1. IRPI within demand position limit.
2. Each IRPI operable.
3. AFD within limits.
4. Containment temperature.
5. Control room temperature.
6. Circ. water temperature.

7. QPTR
8. Heat Balance

REFERENCE
NAPS 1-AP-42.

ANSWER 7.40 (1.00)

@ 0.25 points each:
1. Start all available service air compressors.
2. Start all available instrument air compressors.
3. Secure reactor coolant pumps.
4. Secure charging pump.

REFERENCE
NAPS 1-AP-28.1, p.3.

~~ANSWER 8.01 (1.00)~~

~~(a)~~

Deleted

~~REFERENCE~~
~~NA ADM 16.5, pp 5~~

~~PWG-1:Responsibilities during Maintenance (3.5/3.9)~~

ANSWER 8.02 (1.00)

(d)

REFERENCE
NA U2 TS 3.1.1.3

004.020; PWG-5 (2.9/4.1)

ANSWER 8.03 (1.00)

(d)

REFERENCE
NA U2 TS table 2.2-1.

PWG-5: Generic TS Knowledge (2.9/3.9)

ANSWER 8.04 (1.00)

(a)

REFERENCE
NA U2 TS 3.8.1.1

064/050; PWG-5 (3.1/4.1)

~~ANSWER 8.05 (1.00)~~

~~(c)~~

~~REFERENCE~~
~~NA TS 3.3-1.~~

~~012/000; PWG-5 (3.2/4.0)~~

Deleted

ANSWER 8.06 (1.00)

(a)

REFERENCE
NA U2 TS 3.1.3.2

014/000; PWG-5 (2.9/3.9)

ANSWER 8.07 (1.00)

(b) + (c)

REFERENCE
NA U1 TS 3.5.4.1

006/050; PWG-5 (3.2/4.3)

ANSWER 8.08 (1.00)

(c)

REFERENCE
NA U1 TS 1.7

004/020; PWG-5 (2.9/4.1)

ANSWER 8.09 (1.00)

(a)

REFERENCE
NA U1 TS 2.1.2

010/000; PWG-5 (2.9/4.1)

ANSWER 8.10 (1.00)

(d)

REFERENCE
NA U1 TS 3.5.1

006/020; K4.04 (3.8/4.2)

ANSWER 8.11 (1.00)

(c)

REFERENCE
NA U2 TS 3.4.4

010/000; A2.03 (4.1/4.2)

ANSWER 8.12 (1.00)

(a)

REFERENCE
NA U1&2 TS 3.4.9.1

002/020; PWG-5 (2.9/4.1)

ANSWER 8.13 (1.00)

(b)

REFERENCE
NA U1 & 2 TS 3.1.1.5

002/020; PWG-5 (2.9/4.1)

ANSWER 8.14 (1.00)

(b)

REFERENCE
NA TS bases 3/4.7.1.6 and 7

045/050; PWG-5 (2.7/3.5)

ANSWER 8.15 (1.00)

(c)

REFERENCE
NA TS 3.7.14.6

086/000; PWG-36 (2.8/3.7)

ANSWER 8.16 (1.00)

(c)

REFERENCE
NA U1 TS 3.2.2

001/000; K5.46 (2.3/3.6)

ANSWER 8.17 (1.00)

(c)

REFERENCE
NA U2 TS 3.1.3.5

001/050; PWG-5 (2.9/4.3)

ANSWER 8.18 (1.00)

(d)

REFERENCE
NA U2 TS 3.1.3.1

001/050; PWG-5 (2.9/4.3)

ANSWER 8.19 (1.00)

(b)

REFERENCE
NA TS 4.0.2 & 4.0.5

PWG-5: Generic TS Knowledge (2.9/3.9)

ANSWER 8.20 (1.00)

(a)

REFERENCE
ADM 5.7, p 1.

PWG-26: Logs/Records (3.3/3.6)

ANSWER 8.21 (1.00)

(b)

REFERENCE
NA U2 TS 6.2.2

PWG-19: Fire Brigade (3.4/4.2)

ANSWER 8.22 (1.00)

(b)

REFERENCE
ADM 14.0, p 1.

PWG-14: Tagging (3.6/4.0)

ANSWER 8.23 (1.00)

(c)

REFERENCE
ADM 14.1, p 3.

PWG-14: Tagging (3.6/4.0)

ANSWER 8.24 (1.00)

(d)

REFERENCE
1. IE information notice 82-51 'Overexposures in PWR cavities'.
2. Surry 2 event, april 1979.

034/000; A1.02 (2.9/3.7)

ANSWER 8.25 (1.00)

(c)

REFERENCE
1-OP-4.1, p 15.

EPE-036; EK3.01 (3.1/3.7)

ANSWER 8.26 (1.00)

(a)

REFERENCE
1-OP-4.1, p 5.

004/020; PWG-7 (3.4/4.1)

ANSWER 8.27 (1.00)

(c)

REFERENCE
NA EP 5.2.1.1

PWG-36: E-Plan (2.9/4.7)

ANSWER 8.28 (1.00)

(b)

REFERENCE
EPIP 3.02 and 3.03.

PWG-36: E-Plan (2.9/4.7)

ANSWER 8.29 (1.00)

(b)

REFERENCE
ADM 19.3, p 1.

PWG-26: Logs (3.3/3.6)

ANSWER 8.30 (1.00)

(b)

REFERENCE
ADM 19.18, p 2.

PWG-2: Notification of Plant Supv. and Off-Plant Personnel (2.7/3.8)

ANSWER 8.31 (1.00)

a. 1
b. 5
c. 4
d. 2
e. 7

REFERENCE
NA U1 TS 3.4.6.2

002/020; PWG-5 (2.9/4.1)

ANSWER 8.32 (1.00)

1.02 (+.5 ea)
1.09

REFERENCE
NA U1 TS 3.2.4

001/050; PWG-5 (2.9/4.3)

ANSWER 8.33 (1.00)

a. 23
b. 100 or 150

REFERENCE
a. NA U2 TS 3.9.10
b. NA U2 TS 3.9.3

034/000; PWG-5 (2.8/3.7)

ANSWER 8.34 (1.00)

a. 1
b. 1
c. 3
d. 3
e. 1

REFERENCE
NA U1&2 TS table 6.2-1.

PWG-23: Staffing/Activities (2.8/3.5)

ANSWER 8.35 (2.00)

1.77
10
20,000 or 12,950
the required shutdown margin is restored

REFERENCE
NA TS 3.1.1.1

004/010; AZ.07 (3.8/3.9)

ANSWER 8.36 (2.00)

1) Review and sign the Unit 1 CRO shift turnover checklist.
2) Review and sign the Unit 1 Turbine Building shift turnover checklist.
3) Review and sign the outside shift turnover checklist.
4) Review and sign the auxiliary building shift turnover checklist.
5) Review and sign the backboards shift turnover checklist.
6) Review and sign the boilers and diesels shift turnover checklist.

REFERENCE
ADM 19.3, p 1.

PWG-23: Staffing/Activities (2.8/3.5)

ANSWER 8.37 (1.00)

1. Its corresponding normal or emergency source is operable, and
2. All its redundant components are operable.

REFERENCE
NA TS 3.0.5

062/000; PWG-5 (3.0/4.0)

ANSWER 8.38 (1.00)

1. Ionizing radiation.
2. Heat stress.
3. Differential pressure.
4. Potential oxygen deficiency.

REFERENCE
NA ADM 20.9, p 1.

103/000; K3.02 (3.8/4.2)

TEST CROSS REFERENCE

QUESTION	VALUE	REFERENCE
05.01	1.00	WMD0000260
05.02	1.00	WMD0000267
05.03	1.00	WMD0000277
05.04	1.00	WMD0000289
05.05	1.00	WMD0000130
05.06	1.00	WMD0000253
05.07	1.00	WMD0000254
05.08	1.00	WMD0000255
05.09	1.00	WMD0000258
05.10	1.00	WMD0000259
05.11	1.00	WMD0000261
05.12	1.00	WMD0000262
05.13	1.00	WMD0000264
05.14	1.00	WMD0000265
05.15	1.00	WMD0000266
05.16	1.00	WMD0000268
05.17	1.00	WMD0000269
05.18	1.00	WMD0000270
05.19	1.00	WMD0000272
05.20	1.00	WMD0000273
05.21	1.00	WMD0000274
05.22	1.00	WMD0000275
05.23	1.00	WMD0000276
05.24	1.00	WMD0000278
05.25	1.00	WMD0000280
05.26	1.00	WMD0000281
05.27	1.00	WMD0000282
05.28	1.00	WMD0000283
05.29	1.00	WMD0000284
05.30	1.50	WMD0000285
05.31	1.00	WMD0000286
05.32	1.00	WMD0000287
05.33	1.00	WMD0000288
05.34	2.00	WMD0000256
05.35	2.00	WMD0000263
05.36	2.00	WMD0000257
05.37	.50	WMD0000279

	40.00	
06.01	1.00	WMD0000308
06.02	1.00	WMD0000316
06.03	1.00	WMD0000341
06.04	1.00	WMD0000160
06.05	1.00	WMD0000168
06.06	1.00	WMD0000310
06.07	1.00	WMD0000313
06.08	1.00	WMD0000317
06.09	1.00	WMD0000323
06.10	1.00	WMD0000329

QUESTION	VALUE	REFERENCE
06.11	1.00	WMD0000330
06.12	1.00	WMD0000338
06.13	1.00	WMD0000347
06.14	2.00	WMD0000295
06.15	1.50	WMD0000351
06.16	1.50	WMD0000293
06.17	1.50	WMD0000342
06.18	1.50	WMD0000349
06.19	1.50	WMD0000343
06.20	1.50	WMD0000346
06.21	1.50	WMD0000297
06.22	2.00	WMD0000324
06.23	1.50	WMD0000333
06.24	1.00	WMD0000336
06.25	1.75	WMD0000344
06.26	1.00	WMD0000348
06.27	1.25	WMD0000350
06.28	1.50	WMD0000352
06.29	1.00	WMD0000314
06.30	1.50	WMD0000305
06.31	2.00	WMD0000307

	40.00	
07.01	1.00	WMD0000436
07.02	1.00	WMD0000440
07.03	1.00	WMD0000433
07.04	1.00	WMD0000434
07.05	1.00	WMD0000435
07.06	1.00	WMD0000439
07.07	1.00	WMD0000442
07.08	1.00	WMD0000443
07.09	1.00	WMD0000444
07.10	1.00	WMD0000445
07.11	1.00	WMD0000446
07.12	1.00	WMD0000447
07.13	1.00	WMD0000448
07.14	1.00	WMD0000451
07.15	1.00	WMD0000452
07.16	1.00	WMD0000453
07.17	1.00	WMD0000454
07.18	1.00	WMD0000455
07.19	1.00	WMD0000457
07.20	1.00	WMD0000459
07.21	1.00	WMD0000460
07.22	1.00	WMD0000461
07.23	1.00	WMD0000464
07.24	1.00	WMD0000465
07.25	.50	WMD0000450
07.26	.50	WMD0000462
07.27	.50	WMD0000463

QUESTION	VALUE	REFERENCE
07.28	.50	WMD0000469
07.29	.50	WMD0000470
07.30	1.00	WMD0000458
07.31	1.00	WMD0000467
07.32	1.00	WMD0000468
07.33	1.00	WMD0000472
07.34	1.00	WMD0000437
07.35	3.25	WMD0000438
07.36	1.00	WMD0000441
07.37	1.00	WMD0000449
07.38	1.00	WMD0000456
07.39	1.25	WMD0000466
07.40	1.00	WMD0000471
	40.00	
08.01	1.00	WMD0000395
08.02	1.00	WMD0000411
08.03	1.00	WMD0000413
08.04	1.00	WMD0000414
08.05	1.00	WMD0000396
08.06	1.00	WMD0000397
08.07	1.00	WMD0000398
08.08	1.00	WMD0000399
08.09	1.00	WMD0000401
08.10	1.00	WMD0000402
08.11	1.00	WMD0000403
08.12	1.00	WMD0000404
08.13	1.00	WMD0000405
08.14	1.00	WMD0000406
08.15	1.00	WMD0000407
08.16	1.00	WMD0000408
08.17	1.00	WMD0000409
08.18	1.00	WMD0000410
08.19	1.00	WMD0000412
08.20	1.00	WMD0000415
08.21	1.00	WMD0000416
08.22	1.00	WMD0000417
08.23	1.00	WMD0000418
08.24	1.00	WMD0000419
08.25	1.00	WMD0000420
08.26	1.00	WMD0000421
08.27	1.00	WMD0000422
08.28	1.00	WMD0000423
08.29	1.00	WMD0000424
08.30	1.00	WMD0000425
08.31	1.00	WMD0000428
08.32	1.00	WMD0000400
08.33	1.00	WMD0000426
08.34	1.00	WMD0000427
08.35	2.00	WMD0000429

QUESTION	VALUE	REFERENCE
08.36	2.00	WMD0000430
08.37	1.00	WMD0000431
08.38	1.00	WMD0000432

	40.00	

	160.00	

ENCLOSURE 4

COMMENTS ON WRITTEN NRC EXAMINATIONS
NORTH ANNA POWER STATION

Reactor Operator Exam

Section 1: No Comments

Section 2:

General Comments on Section 2

While many questions in section 2 relate to the operation of the power station, some questions do not. For example questions 2.04, 2.09, 2.16, 2.18, 2.19, 2.20, 2.21 and 2.25 ask for detailed knowledge concerning the specific design of the plant with little emphasis on the operational aspects of the plant. The knowledge required by these questions is not supported by the job task analysis.

Recommendations

Evaluate operator performance on section 2 to ensure that a lack of knowledge of these detailed design questions will not result in the failure of operators who have demonstrated knowledge and skills on other sections of the examinations.

Question 2.02

Comment: Part c of Column A, answer key has (5) as correct answer. RCS samples come off of Hot and Cold legs of all loops thus all answer on key are correct.

Recommendation: Change answer key to indicate all selections in Column B are correct.

Reference: FM-93A and 89D.

Question 2.05

Comment: Answer key has (b) as correct answer, however, 1-OP-14.1 Step 4.1.14 throttles CC outlet MOV's. Step 4.1.25 mentions HCV-1758. Both (a) and (b) are correct in that throttling of CC MOV's determines position of HCV 1758 thus either or both can be throttled to control C/D rate.

Recommendation: Change answer key to accept both (a) and (b) as correct answers.

Reference: 1-OP-14.1

Question 2.06

Comment: The distractor b is a trick answer in that one word was chanted from the lesson plan (i.e., a total of 1 gpm vs. 1 gpm per line). The difference is not significant from an operational viewpoint. Thus both answer (b) and (c) are correct.

Recommendation: Accept both b and c as correct answers.

Reference: NCRODP-88.1, RCS Pressurizer and Pressurizer Relief, page 2.12.

Question 2.09

Comment: Question not operationally oriented. Asks for memorized portions of operational procedure for plant cooldown. These types of knowledges are specifically not required to be memorized IAW NUREG 1021 ES202. The operation in question is only performed ≈ once/18 months and the maintenance procedure is used which clearly provides the knowledge required by this question.

Recommendation: Delete the question.

Reference: MOP 5.04.

Question 2.17

Comment: Answer key has "Hollow O-Ring and Inner and outer seal". These rings are commonly referred to at North Anna as "Seal Rings". Question asks for obscure descriptions and words taken out of context. We feel, upon examination, that all candidates can fully explain the entire vessel head seal system and leak detection system.

Completion items should minimize the number of blanks. Although some multiple blank statements seem to measure rather complex reasoning abilities, such responses are more appropriate measures of intelligence than achievement.

Recommendation: Delete question.

Reference: Gronlund, N. E. Measurement and Evaluation in Teaching 2nd Ed.

Question 2.18

Comment: Answer key is not complete.

Recommendation: Add the following to the answer key:

1. T_c up through holes to cool upper head.

2. T_c to T_h via discharge nozzle/core panel outlet interface.

Reference: NCRODP 88.1.

Question 2.19

Comment: The exact shape of the rod pressure vs. life curve is not important to operators since the introduction of high density fuel pellets and use of pressurization. Operators know that in general the rod pressure increases through life but the presence of an initial drop or the changes in rate of pressure increase following this is not an operational concern.

Recommendation: Delete the question.

Reference: RO Job/Task Analysis.

Question 2.20

Comment: Question asks the operator to distinguish between what usually occurs on loss of turbine load while at rated thermal to what is assumed to have occurred in the analysis for this design basis. Whether or not the operator is able to successfully answer this question correctly has not consequence to his safe operation of the plant.

Recommendation: Note recommendation for general comment on Section 2.

Question 2.23

Comment: Knowledge of number of demineralizers is O.K. however, knowledge of D.F.'s not required by operator. D.F.'s are not considered by operators when taking demins in/out of service. This is based on recommendation from Chemistry department. This knowledge not identified in RO Task list.

Recommendation: Delete portion of question: "The mixed Bed's resin will provide a minimum decontamination factor of 10 for all fission products except ______, ______ and ______.

Reference: This knowledge not referenced in RO JTA/KSA.

Question 2.25

Comment: May be nice to know information but not information which will impact performance as reactor operator. Also second part of question is confusing where asking for effect on plant "thermal power capabilities". Answer key indicates examiner was looking for effect on generator output ("3 MW gain for Unit 2").

Recommendation: Delete question.

Reference: No reference for this knowledge requirement in RO Task list or KSA catalog.

Question 2.26

Comment: Question asks for location - Not clear as to whether examiner is looking for physical location or location within system creating confusion for candidate.

Recommendation: Add the following to the answer key as acceptable answers.

1. CC Heat Exchanger/SW side. Basement of Aux Bldg., East end of CC Heat Exchangers on floor.

2. Discharge to tunnel - Turbine Bldg. SW pit.

3. Discharge to reservoir - Southeast corner of SW pump house.

4. Recirc Spray Hx's - Basement of Quench Spray pump house.

Reference: Plant locations/drawings (FB).

Question 2.31 c

Comment: Location is not under rod drive room as answer key indicates. It is close to this, however, candidate may choose (1) since it is outside of the drumming area. The correct description is - in the alley west of Aux. Buld. by primary plant gas house.

Recommendation: Change answer key to accept (1) or (2) as correct answers.

Reference: Actual plant locations.

Section 3:

Question 3.03

Comment: The answer key states that accum. discharge valves receive a signal to open on an SI signal and when RCS pressure is '2010 psig. RCS pressure '2010 psig is incorrect. The Accum. discharge MOV's received signal to open on actuation of permissive (P-11) which occurs at RCS pressure '2000 psig.

Recommendation: Change answer key to reflect correct pressure.

Reference: T.S. 33.2.1, Page 3/4 3-23.

Question 3.06

Comment: Question asks operator to distinguish between reactor trips designed into the reactor protection system and reactor trips taken credit for in the accident analysis. This type of information is beyond the scope of a reactor operator's knowledge and his ability to successfully answer the question has no consequence to plant operation.

Recommendation: Delete the question.

Reference: JTA/KSA.

Question 3.09

Comment: Knowing the internals of the source range instrumentation has no bearing on the ability of the operator to safely operate the plant. (e.g: items B, F, G)

Recommendation: Note recommendation for general comment on Section 2.

Question 3.16

Comment: Answer b. calls for knowledge of maintenance tasks performed by instrumentation technicians. This is not a required operator knowledge and is not supported by the JTAs and KSA for this station.

Recommendation: Delete portion 3.16 b. answer requirement.

Reference: North Anna Operator JTA (KSA).

Question 3.27

Comment: Answers (b) and (c) are both correct.

Recommendation: Change answwer key to accept either (b) or (c) as correct.

Reference: FSAR Vol. IX, Pages 6-2-90 and 6-2-100 (6-2-100 deals with NPSH).

Section 4

Question 4.12

Comment: The question requires candidates to commit subsequent actions to memory (step 5.9.1 of AP-5.1) which is not required by NUREG 1021, ES 202, Section B-4.

Recommendation: Delete the question.

Reference: NUREG 1021, ES 202, Section B-4.

Question 4.14

Comment: The questions requires candidate to commit to memory long terms actions of AP-42. This is not IAW NUREG 1021, ES 202, Section B-4.

Recommendation: Delete question.

Reference: NUREG 1021, ES 202, Section B-4.

Question 4.38

Comment: Answer key does not include all possible answer to the question.

Recommendation: Additional answers that can be accepted include:

1. time and date of expiration
2. general instructions
3. respiratory protection requirements
4. radiation levels
5. contamination levels
6. airborne radioactivity levels

Reference: NAPS Radiation Work Permit Form. Section 2.1 Radiation Protection Manual.

Senior Reactor Operator Exam

Section 5: No Comments.

Section 6:

Question 6.02

Comment: Answers (b) and (c) are both correct.

Recommendation: Change answer key to accept both (b) and (c) as correct answers.

Reference: NCRODP-88.1, RCS Pressurizer and Pzr Relief, Page 2.12.

Question 6.06

Comment: Answers (b) and (c) are both correct.

Recommendation: Change answer key to accept both (b) and (c) as correct answers.

References: FSAR Vol. IX, Pages 6-2-90 and 6-2-100 (6-2-100 deals with NPSH).

Question 6.10

Comment: The exact shape of the rod pressure vs. life curve is not important to operators since the introduction of high density fuel pellets and use of pressurization. Operators know that in general the rod pressure increases through life but the presence of an initial drop or the changes in rate of pressure increase following this is not an operational concern.

Question 6.19 c

Comment: Location is not under Rod Drive Room as answer key indicates - This is closest answer but candidate could just as easily choose (1) since it is outside the drumming area. The correct answer which is not one of the choices s "In alley west of Aux. Bldg. by primary plant gas house".

Recommendation: Change answer key to accept either (1) or (2) as correct answers.

Reference: Actual Plant Locations.

Question 6.24

Comment: May be nice to know information but not information which will impact performance as reactor operator. Also second part of question is confusing where asking for effect on plant "thermal power capabilities". Answer key indicates examiner was looking for effect on generator output ("3 MW gain for Unit 2").

Recommendation: Delete question.

Reference: No reference for this knowledge requirement in RO Task list or KSA catalog.

Question 6.28

Comment: Setpoint should not be required especially with a (.25) point value assigned to it. The heat sensors are set/tested/adjusted by the Fire Marshall, not operations department. Thus, knowing that there is an automatic actuation by a heat sensor is acceptable.

Recommendation: Delete requirement for setpoints - see comment section. Setpoint knowledge of minimium importance compared to knowledge of locations and methods.

Reference: None

Section 7

Question 7.02

Comment: The question requires candidates to commit subsequent actions to memory (step 5.9.1 of AP-5.1) which is not required by NUREG 1021, ES 202, Section B-4.

Recommendation: Delete the question.

Reference: NUREG 1021, ES 202, Section B-4.

Question 7.30

Comment: This question asks for long term action memorization. LHSI pumps are not required to be operable in mode 6 by Tech. Specs. 1-OP-4.1 Page 13 of 49, Step 3.1.30 says "must have one LHSI or RP pump operable" thus LHSI may not be operable so question is not as straight forward as 1-AP-52 leads you to believe.

Recommendation: Delete question.

Reference: Tech. Specs, 1-AP-51. NUREG 1021 ES-202 page 3 of 6 (E.2).

Question 7.38

Comment: Answer key does not include all possible answers to the question.

Recommendation: Additional answers that can be accepted include:

1. time and date of expiration
2. general instructions
3. respiratory protection requirements
4. radiation levels
5. contamination levels
6. airborne radioactivity levels

Reference: NAPS Radiation Work Permit Form. Section 2.1 Radiation Protection Manual.

Question 7.39

Comment: Question asks candidate to repeat long term actions. Not required by NUREG 1021.

Recommendation: Delete question.

Reference: NUREG 1021 ES-202.

Section 8

Question 8.01

Comment: Question requires candidate to recall from memory four names that are attached to an Emergency Work Order. Since the reference is available to the SRO, memorization of form names is not appropriate. Also to reduce the workload on the SRO on shift the Operations Maintenance Coordinator process the "Equipment History/failure Analysis" form after the work has been completed and then attaches it to the work package the next working day.

Recommendation: Delete the question.

Reference: Admin 16.5.

Question 8.05

Comment: A difference exists in the action statements between Units 1 and 2. Unit 1 T.S. - Table 3.3 - 1 requires answers (d) or (e). Unit 2 T.S. Table 3.3-1 requires answer (c). Since unit was not specified, question is invalid.

Recommendation: Delete the question.

Reference: Unit 1 and 2 T.S. - Table 3.3-1.

Question 8.07

Comment: All 3 distractors and "correct" answer are incorrect due to design change.

Recommendation: Delete question.

Reference: Ammendment to License #58 and 64.

Question 8.33 b

Comment: A recent Tech. Spec. change has been received that changes the shutdown requirement to 150 hours. The contorl room copy of T.S. has been revised but training copies have not been updated as of this date resulting in a conflict of candidate references.

Recommendation: Change answer key to accept either 100 hours or 150 hours as correct answer.

References: Unit 1 and 2 Tech. Specs. 3.93.

Question 8.35

Comment: A recent T.S. change has been received that changes the required boron concentration to 12,950 ppm instead of 20,000. The control room copy of this T.S. has been revised but training copies have not been revised as of this date resulting in a conflict of candidate references.

Recommendation: Change answer key to accept within 20,000 ppm or 12,950 as correct answers.

Reference: Unit 1 and 2 Tech. Specs. 3.1.1.1.