ENCLOSURE 1

EXAMINATION REPORT - 50-338/OL-87-01

Facility Licensee: Virginia Electric and Power Company

Facility Name: North Anna Power Station

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

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

Chief Examiner: J H Moorman III 4-6-87
J. H. Moorman, III Date Signed

Approved by: John F Munro 4/7/87
John F. Munro, Section Chief Date Signed

Summary:

Examinations on February 9-12, 1987.

Written, oral, and simulator examinations were administered to eight candidates; all of whom passed. Three candidates were administered simulator re-examinations; all candidates passed.

Based on the results described above, five of five RO's passed and six of six SRO's passed.

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REPORT DETAILS

1. Facility Employees Contacted:

*R. Buck, Supervisor, Operations Training
*M. Crist, Training Staff
*L. Edmonds, Superintendent, Nuclear Training
*R. Enfinger, Superintendent, Operations
*D. Fellows, Training Staff
*W. Harrell, Station Manager
*W. Shura, Training Staff
*R. Stevens, Training Staff

*Attended Exit Meeting

2. Examiners:

J. Arildsen
N. Jensen, INEL
*J. Moorman
B. Picker, INEL

*Chief Examiner

3. Examination Review Meeting

At the conclusion of the written examinations, the examiners provided Mike Crist with a copy of the written examination and answer key for review. The comments made by the facility reviewers are included as Enclosure 3 to this report and the NRC Resolutions to these comments are listed below.

a. RO Examination - Analogous SRO questions in parenthesis

(1) Question 1.03 (5.17)

NRC Resolution:

Utility comment accepted. The answer has been modified to allow full credit if "95%" is not mentioned. However, the answer must provide assurance that the candidate understands that maintaining DNBR ≥ 1.3 will not always prevent DNB.

(2) Question 1.04(3) (5.10b)

NRC Resolution:

Utility comment accepted. RCS Temperature is a parameter and will be accepted for full credit. T_h will also be accepted; however, T_c will not be accepted since it remains essentially constant during operation.

(3) Question 1.05 (5.18)

NRC Resolution:

Utility comment not accepted. The non-existence of superheating in a system is not a condition that must be present for natural circulation to exist. Rather, the non-existence of superheating promotes natural circulation.

(4) Question 1.08b (5.07b)

NRC Resolution:

Utility comment accepted. The MTC becomes "more negative" with core age and "increase" will be an acceptable answer. The answer key has been modified.

(5) Question 1.11 (5.19)

NRC Resolution:

Utility comment not accepted. The utility comment is based on a dynamic situation (i.e., the difference between two points). The question clearly states the initial conditions of the problem.

(6) Question 1.19

NRC Resolution:

Utility comment not accepted. This type of knowledge should be basic to anyone who has completed an RO or SRO training program. Errors caused by miscalculation will not be compounded through the remainder of the question.

(7) Question 1.23a

NRC Resolution:

Utility comment not accepted. It is clearly stated in the question that all parameters are equal (i.e., source strength of each reactor is equal).

(8) Question 1.23b

NRC Resolution:

Utility comment not accepted. Knowledge of how rod worth varies over core life is supported by NUREG-1122, Knowledge and Abilities Catalog for Nuclear Power Plant Operators: Pressurized Water Reactors. The question was used to check operator knowledge of how rod worth varies over core life. To assume equivalent rod worth would imply a trivial answer.

(9) Question 2.06

NRC Resolution:

Utility comment not accepted. Knowledge of the power supplies for pressurizer heaters is supported by NUREG-1122. The point value of the question has been changed to 1.25 points.

(10) Question 2.10e

NRC Resolution:

Utility comment accepted. The utility's comment is an equivalent answer. This clarification will be added to the answer key.

(11) Question 2.11a

NRC Resolution:

Utility comment not accepted. Reference material provided states that the floating seal ring limits leakage to 50 gpm. Since no further reference material is available to support other limits, this is the value that will be used.

(12) Question 2.15

NRC Resolution:

Utility comment accepted. Discharge canal and bladder tank are equivalent answers and either will be accepted for full credit. The answer key has been modified.

(13) Question 2.17

NRC Resolution:

Utility comment accepted. The answer key was changed to require only the minimum values asked for in the question.

(14) Question 2.23 (6.07)

NRC Resolution:

Utility comment accepted. Due to the poor quality of the question, Part A has been deleted. Total value of the question is changed to .75 points.

(15) Question 3.06

Utility comment accepted. Due to poor wording of the question and the clarification given during the exam, both A and C will be accepted as correct answers.

(16) Question 3.12c (6.17c)

Utility comment accepted. Due to a typographical error in the question, this part of the question has been deleted. The question value is now 1.0 point.

(17) Question 3.16 (6.14)

Utility comment accepted. Recommended answers are equivalent to answers 3 and 1. The answer key has been modified to recognize these clarifications.

(18) Question 3.18

NRC Resolution:

Utility comment accepted. The answer key has been modified to accept this additional answer.

(19) Question 3.24a

NRC Resolution:

Utility comment accepted. The answer has been changed to include N44 as an input to the main feed bypass valve controller and to give .25 points for its inclusion in the answer. The total value of the question is now 1.25 points.

(20) 4.01 (7.01)

NRC Resolution:

Utility comment not accepted. It is acknowledged that the question is not specific as to whether seal injection flow or No. 1 seal leakoff flow is to be used. However, the reference material provided by the utility states that the minimum seal injection flow is six gpm and the minimum No. 1 seal leakoff flow is .2 gpm. If a candidate answered the question that there are no correct answers using the six gpm minimum seal injection flow as a premise, this answer will be accepted for full credit. If the candidate answered the question assuming .2 gpm for No. 1 seal leakoff flow and 15 psig minimum VCT pressure (answer 2), this will be accepted for full credit.

(21) Question 4.13 (7.13)

NRC Resolution:

Utility comment accepted. The answer key has been modified to distinguish between the separate conditions.

b. SRO Examination

(1) Question 5.05

NRC Resolution:

Utility comment accepted. The answer key has been revised to reflect the correct answer.

(2) Question 5.14b

NRC Resolution:

Utility comment not accepted. Although conductive heat transfer does occur in this instance, it is not the primary cause for the increase in heat transfer.

(3) Question 6.10a

NRC Resolution:

Utility comment accepted. The answer key has been modified to reflect all possible answers. The point value of the question has been revised to reflect the additional answers. Part C was deleted by a clarification during the exam.

(4) Question 7.09d

NRC Resolution:

Utility comment accepted. The answer key has been changed to reflect the correct answer.

(5) Question 7.14 (4.14)

NRC Resolution:

Utility comment accepted. The answer has been modified to accept "Inject the BIT" as a separate step. The point value of the question has been increased by .25 points.

(6) Question 7.20

NRC Resolution:

Utility comment accepted. The requirement for the flow paths to be placed in an order has been deleted and the value for each response has been raised to .365 points.

(7) Question 8.09

NRC Resolution:

Utility comment accepted. The answer key has been changed to reflect the requirements of the North Anna Power Station.

c. After further review of the examinations, the following changes were made.

(1) Question 1.18

NRC Resolution:

Distractors 'A' and 'C' are basically the same statement. Either answer will be accepted for full credit.

(2) Question 2.20

NRC Resolution:

This question only solicits answer (2). This answer will be accepted for .75 points. Part one of the answer has been deleted.

The parenthesis have been deleted to require "chloride stress corrosion" to be in the answer.

(3) Question 2.21 (6.03)

NRC Resolution:

The answer will be modified to accept "to aid in warming up the RHR system" and "for overpressure protection anytime." The answer "pressure equalization" is equivalent to "allow for expansion."

(4) Question 3.19

NRC Resolution:

The answer will be changed as follows:

Normal: 125 vdc vital bus (125 vdc/120/vac static inverter)
Alternate: 480 V ac emergency panel (1H1,1J1) - 480/120 ac transformer
Emergency: battery (to 125 v dc vital bus)

(5) Question 4.23 (7.24)

NRC Resolution:

Acceptable as a reason for using steam pressure mode will be "because Tavg mode can not be used to cooldown below 547°F (no load Tavg)." Reasonable wording will also be accepted.

(6) Question 3.17 (6.20)

NRC Resolution:

The answer key has been amended to clarify the "Breaker 86 and 87 protective relays" as overcurrent and phase-differential.

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.

There was one generic weakness noted during the simulator examination. The area of below normal performance was imprecise communications.

The cooperation given to the examiners and the effort to ensure an atmosphere in the control room conducive to oral examinations was also noted and appreciated. Additionally, the accommodations made by the facility for under-instruction examiners and examiner audit personnel were noted and appreciated.

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

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: NORTH ANNA 1&2

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 87/02/09

EXAMINER: MOORMAN, J.

CANDIDATE: MASTER

INSTRUCTIONS TO CANDIDATE:

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	CANDIDATE'S SCORE	% OF CATEGORY VALUE	CATEGORY
30.00	~~25.00~~ 25.6	________	________	1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
~~30.00~~ 28.25	~~25.00~~ 24.1	________	________	2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
~~30.00~~ 29.75	~~25.00~~ 25.4	________	________	3. INSTRUMENTS AND CONTROLS
~~30.00~~ ~~29.25~~ 30.25	~~25.00~~ 24.9	________	________	4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
~~120.00~~ ~~117.25~~ 118.25		________ Final Grade	________%	Totals

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

____
Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

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

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

3. Use black ink or dark pencil <u>only</u> to facilitate legible reproductions.

4. Print your name in the blank provided on the cover sheet of the examination.

5. Fill in the date on the cover sheet of the examination (if necessary).

6. Use only the paper provided for answers.

7. Print your name in the upper right-hand corner of the first page of <u>each</u> section of the answer sheet.

8. Consecutively number each answer sheet, write "End of Category __" as appropriate, start each category on a <u>new</u> page, write <u>only on one side</u> of the paper, and write "Last Page" on the last answer sheet.

9. Number each answer as to category and number, for example, 1.4, 6.3.

10. Skip at least <u>three</u> lines between each answer.

11. Separate answer sheets from pad and place finished answer sheets face down on your desk or table.

12. Use abbreviations only if they are commonly used in facility <u>literature</u>.

13. The point value for each question is indicated in parentheses after the question and can be used as a guide for the depth of answer required.

14. Show all calculations, methods, or assumptions used to obtain an answer to mathematical problems whether indicated in the question or not.

15. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

16. If parts of the examination are not clear as to intent, ask questions of the <u>examiner</u> only.

17. You must sign the statement on the cover sheet that indicates that the work is your own and you have not received or been given assistance in completing the examination. This must be done after the examination has been completed.

18. When you complete your examination, you shall:

    a. Assemble your examination as follows:

        (1) Exam questions on top.

        (2) Exam aids - figures, tables, etc.

        (3) Answer pages including figures which are part of the answer.

    b. Turn in your copy of the examination and all pages used to answer the examination questions.

    c. Turn in all scrap paper and the balance of the paper that you did not use for answering the questions.

    d. Leave the examination area, as defined by the examiner. If after leaving, you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION 1.01 (2.00)

Given two pumps of equivalent design, operating at the same, constant speed:

A. What will be the effect of placing the two pumps in series (with respect to flow and head)?

B. What will be the effect of placing the two pumps in parallel (with respect to flow and head)?

QUESTION 1.02 (1.00)

Given: Three reactor coolant (RCP) pumps operating in parallel, each with a flow rate "m" and a combined flow rate "M". Out of the four possibilities below, choose the one that best fits if one RCP is secured.

a. The resulting core flow (M) will increase.

b. The resulting core flow (M) will increase along with individual operating RCP flow (m).

c. The resulting core flow (M) will decrease as individual operating RCP flow (m) increases.

d. The resulting core flow (M) will not change due to decrease in RCP back pressure.

QUESTION 1.03 (1.00)

What is the design basis of having a DNBR > or = to 1.3?

QUESTION 1.04 (2.00)

List the four (4) plant parameters observed to insure that CHF or DNBR are not exceeded.

QUESTION 1.05 (2.00)

What are all the conditions that must be present in order for natural circulation to exist?

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

QUESTION 1.06 (1.00)

With respect to reactor thermal limits which of the following statements is NOT correct.

a. The ratio of the peak linear power density to the average linear power density in the core at a particular elevation is called the nuclear heat flux hot channel factor.

b. The average linear power density in the core is expressed in units of kw/ft and is the total thermal power divided by the active length of all the fuel rods.

c. The purpose of limiting the enthalpy rise hot channel factor is to prevent bulk boiling from taking place during normal operations.

d. The rod bow penalty (RBP) accounts for the bowing of fuel rods as their burnup increases.

e. The purpose of the limit on the heat flux hot channel factor is to insure that fuel clad temperature does not exceed 2200 deg F during normal operations.

QUESTION 1.07 (.50)

Consider a fuel pellet at 70 deg F. A 6.7ev neutron coming in will be absorbed. The 6.7ev neutron will be absorbed in the outer part of the fuel. The inner fuel will not even see the neutron (low flux). This phenomenon is called ____________.

QUESTION 1.08 (1.50)

Write on your answer sheet INCREASES , DECREASES or DOES NOT CHANGE for the following:

The magnitude of the fuel temperature coefficient (FTC):

A. INCREASES / DECREASES / DOES NOT CHANGE with increase in power.

B. INCREASES / DECREASES / DOES NOT CHANGE with core age.

C. INCREASES / DECREASES / DOES NOT CHANGE with decrease in moderator temperature coefficient (MTC).

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

QUESTION 1.09 (1.00)

The negative reactivity added when fuel temperature increases is primarily caused by ______.

a. self shielding of the fuel

b. doppler broadening

c. an increase in the GAMMA heating contribution

d. fuel pellet swell thus decreasing the gap

QUESTION 1.10 (1.00)

Which one of the following statements is correct?

At the beginning of the Xe transient on a power decrease following 100 hours at 100% power:
note: [Xe] denotes xenon concentration

a. Direct [Xe] increases, indirect [Xe] decreases, total [Xe] decreases.

b. Direct [Xe] increases, indirect [Xe] increases, total [Xe] increases.

c. Direct [Xe] decreases, indirect [Xe] decreases, total [Xe] decreases.

d. Direct [Xe] decreases, indirect [Xe] increases, total [Xe] increases.

e. Direct [Xe] decreases, indirect [Xe] increases, total [Xe] decreases.

QUESTION 1.11 (1.00)

GIVEN: Two identical control rods, each absorb an equal amount of neutrons. The neutron flux at the center of the core equals that at the edge of the core. Why do the control rods in the middle of the core (radially) have a greater effect on Keff than the control rods at the edge of the core (radially).

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

QUESTION 1.12 (1.00)

What effect does rod shadowing have on the worth of control rods?

QUESTION 1.13 (1.50)

On a reactor startup, what 3 conditions indicate the reactor is critical?

QUESTION 1.14 (1.00)

Give two reasons why 10 exp -8 amps is chosen as a standard reference for critical rod height data.
note: "standard reference" is NOT an acceptable answer

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

QUESTION 1.15 (2.00)

Match the term in column A with the correct definition in column B.

column A	column B
a) Specific Entropy	1) BTU/deg F
b) DNBR	2) Ratio of local $\dot{Q}$ to to CHF >1.30
c) Quality	3) Internal energy of a substance
d. Enthalpy	4) % steam mass to total steam & water mass
	5) BTU/ lbm-deg R
	6) Ratio of critical $\dot{Q}$ to local $\dot{Q}$
	7) Internal Energy plus Flow Energy of a substance
	8) % steam volume to total steam and water volume

QUESTION 1.16 (1.00)

What effect does adding an 800 ci source yielding 1 X 10 exp 8 neutrons/sec have on the magnitude of Keff in a subcritical reactor?
NOTE: For simplicity assume the microscopic TOTAL cross section of the source equals zero.

a. increase

b. decrease

c. no change

d. insufficient data

QUESTION 1.17 (1.00)

Given a SUR of 0.1 dpm, determine the final power P in terms of the initial power Po after 0.1 hr. Show all work.

QUESTION 1.18 (1.00)

Choose the best answer for the definition of subcritical multiplication.

a. The process of utilizing source neutrons to sustain the chain reaction for Keff < 1.

b. The phenomenon where by source neutrons are used to measure the fractional curvature change of the flux for Keff < 1

c. The manipulation of neutron sources to sustain the chain reaction until Keff = 1.

d. The phenomenon where by source neutrons are used to stabilize reactor period/startup rate thus ensuring reactivity (rho) is << Beff for Keff < 1.

QUESTION 1.19 (1.50)

Calculate the heat transferred across one U-tube of a steam generator. Show all work.

GIVEN: (for simplicity)
U-tube heat transfer coefficient: 1.565 BTU/(sq ft-deg F)
U-tube height: 25 ft
U-tube outer radius: 1/2 inch
primary coolant temperature: 550 deg F
secondary water temperature: 480 deg F

QUESTION 1.20 (1.50)

List three things, that in practice, prevent water hammers from occurring

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

QUESTION 2.01 (1.00)

Which one of the following is NOT a source of water to the PRT?

a) Letdown Relief Valve

b) RCP Seal Water return line relief valve

c) Excess letdown/loop drain header relief valve

d) Reactor Vessel Flange Leakoff Detection Drain

e) PDTT relief valve

QUESTION 2.02 (1.00)

Which one statement below regarding the Source Range Nuclear Instrumentation System is INCORRECT.

a) P-6 allows the source range high level reactor trip signal to be bypassed manually when one of the two intermediate range instruments is above 10 E-10 ion chamber amps.
b) Placing BOTH source range blocking switches to the BLOCK position de-energizes the high voltage supply to both source range instruments.
c) The source range high level trip is blocked when P-10 is present.
d) When P-6 is present and P-10 is not present, the source range high level trip is automatically reinstated and the source range high voltage re-energized when one of the two intermediate ranges is below P-6 reset.

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

QUESTION 1.21 (2.00)

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

a. Enthalpy

b. Pressure

c. Entropy

d. Temperature

QUESTION 1.22 (1.50)

A motor driven centrifugal pump is operating at a low flow condition. You then start opening the throttle valve on the discharge side. How will each of the following be affected? (INCREASE, DECREASE, or NO CHANGE)

a. Discharge Pressure

b. Available NPSH

c. Motor Amps

QUESTION 1.23 (1.00)

Unit A is at EOL while Unit B has just been started up after a refueling. Assuming a rod speed of 48 spm, both reactors are taken critical by pulling 50 steps at a time, waiting until counts stabilize then pulling again. Assuming all systems and parameters are identical at the commencement of the startup, and both units are initially shutdown by 2% (delta k/k):

a) Which Unit will have the highest source range counts when criticality is reached?

b) How will critical rod heights compare in the two Units?

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

QUESTION 2.03 (1.00)

Which one 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 discahrges the contents of the tank into the QS pump suction with a pre-determined flow rate set by a manual throttle valve.

QUESTION 2.04 (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

QUESTION 2.05 (1.00)

Which valve listed below is used to throttle auxiliary spray flow?

a) FCV-122 (Charging Flow Control Valve)
b) HCV-311 (Aux Spray Valve)
c) PCV-455B (Loop C Spray Valve)
d) PCV-455A (Loop A Spray Valve)
e) You cannot throttle auxiliary spray

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

QUESTION 2.06 (1.25) ~~(1.50)~~

Match the following Pressurizer heater banks in Column A with their proper MCC in Column B.

COLUMN A	COLUMN B
A) Back-up Heaters	
1) Group I (0.25)	a) 1A1
2) Group II (0.25)	b) 1B1
3) Group IV (0.25)	c) 1C1
4) Group V (0.25)	d) 1D1
B) Control bank heaters Group III (0.25)	e) 1G1
	f) 1H1
	g) 1J1

QUESTION 2.07 (1.00)

A "High Containment Pressure" Automatic Safety Injection signal will: (Choose one)

a) cause a main steam line isolation.

b) be initiated by 2/4 containment pressure instruments greater than 17 psig.

c) be blocked whenever the reactor trip breakers are open.

d) cause a feedwater isolation and a phase "A" isolation.

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

QUESTION 2.08 (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, per OP 14.1?

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 mini-flow recirculation valve.

QUESTION 2.09 (1.00)

Listed below are valves associated with the Recirculation Spray (RS) System. Indicate whether each of 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 suction valve)

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

QUESTION 2.10 (1.00)

In regards to the Chemical and Volume Control System (CVCS), state what position (OPEN, CLOSED, AS IS) the following valves fail upon a loss of air.

a) Letdown isolation valves LCV-1460 A/B

b) Orifice Isolation valves LCV 1200 A/B/C

c) Pure Grade water supply valve FCV-1114A

d) Boric Acid supply to blender FCV-1113A

e) Emergency Borate valve

QUESTION 2.11 (1.50)

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

a) The floating ring seal, will limit leakage to 50 gpm if the #1 seal fails.

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

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

QUESTION 2.12 (.50)

TRUE/FALSE
A RED urgent failure alarm light indicates that a major electrical failure has occurred in the logic cabinet.

QUESTION 2.13 (1.50)

The principle driving force for PZR normal spray flow is the differential pressure between _________and the _________.

QUESTION 2.14 (2.00)

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

QUESTION 2.15 (1.50)

List the 3 independent sources of water to the Fire Main System.

QUESTION 2.16 (2.00)

LIST 4 of the 5 Design bases for the ECCS Cooling Performance following a LOCA as stated in 10CFR50.46.

QUESTION 2.17 (2.00)

List 5 parameters associated with the RCP's which are monitored after starting a RCP as stated in OP 5.2 "RCP Operation". Provide the required minimum values which must exist if applicable.

QUESTION 2.18 (1.50)

List 6 Emergency loads supplied by the Service Water System for a period of up to 1 hour after a LOCA. Include those that require service water as a back up. (Sets are considered as one load)

QUESTION 2.19 (1.00)

Where is the source of power for the automatic field flash of the Emergency Diesel Generators generated ?

QUESTION 2.20 (0.75) ~~(1.50)~~

Sodium Hydorxide (NaOH) added during the injection phase after a LOCA will eventually be distributed by the Quench Spray System and raises the Containment sump pH approximately 8. What are the two (2) reasons for establishing the elevated pH in the containment?

QUESTION 2.21 (1.50)

State 3 reasons for having HCV-1142 (RHR letdown penetration from the RHR heat exchangers) kept about 10% open.

QUESTION 2.22 (1.50)

State two purposes for the interlock between the letdown isolation valves, LCV-1460A/B, and the orifice isolation valves, HCV-1200A/B/C.

QUESTION 2.23 ,75 ~~(1.50)~~

Concerning the Rod Control System:

deleted ~~a)~~ Place the following components in their proper flow path order. Start from the normal power supply and ending at the CRDM's

1) DC hold cabinet
2) Power Cabinet
3) Motor generator set
4) Reactor Trip breaker
5) Automatic Rod Control Unit
6) Rod Position Indication Cabinet
7) Logic Cabinet

b) For the components in Part a, above, STATE the number of each present in the system.

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

QUESTION 3.01 (1.00)

Which of the following is NOT a function of the P-4 permissive (trip and bypass breakers open)?

a) Allows bypassing a steam dump cooldown interlock.

b) Allows operator block of SI signal.

c) Causes feedwater isolation if low Tavg is also present.

d) Causes a turbine trip.

QUESTION 3.02 (1.00)

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

a) RWST Lo-Lo Level

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

c) SI signal present

d) SI Recirculation Mode signal present

QUESTION 3.03 (1.50)

Concerning the Overtemperature Delta Temperature Setpoint (OTSP) describe how (increases, decreases or remains the same) each of the following parameter changes will effect the OTSP.

a) Increase in Tavg

b) Decrease in Reactor Pressure

c) Increase in Delta Flux Penalty

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

QUESTION 3.04 (1.00)

Which statement below regarding the Main Generator Protection System is INCORRECT.

a) A generator trip always results in a turbine trip when the generator is loaded.

b) Once the generator is loaded, a turbine trip always results in a generator trip.

c) A turbine trip above the protection interlock P-7 (10% power) results in a Reactor trip.

d) A reactor trip always results in a turbine trip.

QUESTION 3.05 (1.25)

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

QUESTION 3.06 (1.00)

Which statement below regarding pressurizer control is CORRECT?

a) All ~~4~~ 3 channels provide input to the SI low pressure signal.

b) All ~~4~~ 3 channels can be utilized to control the operation of the spray valves.

c) All ~~4~~ 3 channels send their signals through an Isolation Amplifier after supplying input to their respective protective circuit.

d) All ~~4~~ 3 channels can supply input to PORV Interlock circuitry to prevent PORV's lifting at low pressures.

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

QUESTION 3.07 (1.00)

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

a) Power Range Nuclear Power

b) RCS pressure

c) Tavg

d) AFD

QUESTION 3.08 (1.00)

Which of one the following statements describes the two Delta T's 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 Th) and
(Average core thermocouple - Loop A Tc)

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

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

QUESTION 3.09 (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 one 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 3.10 (1.00)

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

QUESTION 3.11 (1.00)

Which of the following would be the INITIAL response of the feedwater flow due to the response of the S/G Water Level Control System if the steam pressure transmitter controlling the SGWLCS failed HIGH while at 50% power?

a.) Feed flow would INCREASE due to the maximum steam pressure input to the steam flow signal.

b.) Feed flow would INCREASE due to the level mismatch error between actual and programmed level caused by the pressure instrument failure

c.) Feed flow would DECREASE due to the mismatch between steam and feed flow signals caused by the pressure instrument failure.

d.) Feed flow would REMAIN THE SAME due to the dominace of the level error signal over the flow error signal.

e.) Feed flow would REMAIN THE SAME as the steam pressure will not affect the steam flow signal.

QUESTION 3.12 1.00 ~~(1.50)~~

Indicate whether each of the statements below regarding the High Head Safety Injection System (HHSI) is TRUE or FALSE.

a) The alternate power source, J Bus, is ONLY used for maintenance on the "B" charging pump, and this pump has no automatic pump start capability when connected to the J bus.

b) Normal lead pumps during a SI are the "A" and "B" HHSI pumps.

deleted ~~c)~~ All three pumps get a start signal from a SI signal, but the "A" pump is locked out to allow the "C" pump to start on its normal (H bus), if its breaker is racked out.

QUESTION 3.13 (.50)

TRUE or FALSE
Pulling the control power fuses when the Source Range level trip switch is in "Bypass" will cause a trip signal to occur.

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

QUESTION 3.14 (.50)

TRUE or FALSE
Following a loss of offsite power, the load shedding feature is actuated when the diesel output breaker closes.

QUESTION 3.15 (2.00)

List 5 protection logic signals generated by the Pressurizer Protection System. (Include in your answer set points, coincidence and associated interlocks, if any)

QUESTION 3.16 (1.50)

List 3 purposes of Rod Insertion Limits.

QUESTION 3.17 (1.00)

List the 4 requirements, control manipulations that will make up the logic to manually close the Diesel Generator output breakers (15H2).

QUESTION 3.18 (1.50)

List the 6 reactor trips which are enabled/blocked by the reactor trip system interlock P-7.

QUESTION 3.19 (1.50)

List the THREE power supplies to the Vital 120 VAC distribution system and identify them by their priority.

QUESTION 3.20 (1.00)

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

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

QUESTION 3.21 (1.00)

In regards to the PZR Pressure Control System, Unit 2 has an alarm, MANUAL NDT PROTECTION REQUIRED, which annunciates whenever temperature is <340 deg. F and pressure is > 550 psig. What action is required of the operator if such an alarm is received?

QUESTION 3.22 (1.00)

The Reactor breaker shunt trip coils have been modified to also energize upon any trip signal to the Undervoltage coils. What is the reason for this modification?

QUESTION 3.23 (1.00)

The Detector Current Comparator receives input from all 4 upper and lower power range detectors. How are these inputs compared, and what conditions are needed to auto bypass circuitry while at power?

QUESTION 3.24 1.25 ~~(1.00)~~

a) What are the inputs to the Main Feedwater Bypass Valve controllers?

b) To place the Main Feedwater Bypass Valves in AUTOMATIC control while at low power, what controller-related conditions must be established?

QUESTION 3.25 (1.75)

a) What consequences could be expected in the Rod Control System's DC Hold Cabinet if 2 or more groups of rod drive mechanisms were placed on hold power (excluding Control Bank D rods)? Explain you reasoning. (1.0)

b) Why is there both a 125 VDC and a 70 VDC power supply in the DC Hold Cabinet?

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

QUESTION 3.26 (1.50)

Sketch the rod speed program by indicating rod speed versus error signal.

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

QUESTION ~~4.01~~ deleted ~~(1.00)~~ 1.00

Prior to operating Reactor Coolant Pumps in accordance with OP-5.2, Reactor Coolant Pump Operations, the minimum seal flow should be ____ gpm and VCT pressure should be a minimum of ____ psig.

1. 0 , 10

2. 0.2 , 15

3. 2.0 , 30

4. 5.0 , 20

QUESTION 4.02 (1.00)

List FOUR indications of one dropped rod at 75% power.

QUESTION 4.03 (1.00)

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

QUESTION 4.04 (1.00)

Which of the following describes a temporary change which alters the INTENT of a procedure?

a. A change that corrects an incorrect valve lineup.

b. A change that modifies the criteria by which a system's operability is determined.

c. A change that allows partial use of a procedure to test a subtrain without affecting remaining equipment in that train.

d. A change that allows you to change incorrectly specified instruments for data taking.

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323

QUESTION 4.05 (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 4.06 (1.00)

If a "Rod Control Urgent Failure" alarm occurs due to a failure in the logic cabinet, the Tave/Tref mismatch is immediately maintained by which of the following?

a. controlling turbine load.

b. taking manual control of individual control rod banks.

c. taking manual control of individual control rod groups.

d. boration and dilution of the reactor coolant system.

QUESTION 4.07 (2.00)

Prior to a reactor startup, with the RCS at normal operating pressure and temperature, the following RCS leakages exist. For each leak listed below, indicate whether you could STARTUP or would have to remain SHUTDOWN. (Treat each leak below as an independent event)

a) A leak from an unknown source of 1.5 GPM.

b) 6.0 GPM from a manual valve packing gland.

c) 0.4 GPM from one S/G.

d) 0.1 GPH from the reactor vessel head INNER seal.

QUESTION 4.08 (.50)

You are releaseing radioactive liquid waste in accordance with 1-OP-22.11, Releasing Radioactive Liquid Waste, when one of the operating circulating water pumps trips. You may continue the release for up to 5 minutes while attempting to restart the pump. TRUE/FALSE

QUESTION 4.09 (1.00)

List all conditions that require the Control Rod Drive Mechanism Shroud Cooling Fans to be in operation.

QUESTION 4.10 (1.50)

Match the action listed in Column A with the approximate power level in Column B at which this action is taken on a unit startup to 100% power.

COLUMN A	COLUMN B
a. Place a second Main Feed pump in service	1) 15%
	2) 30%
b. Stop increasing power and check for a chemistry hold	3) 50%
	4) 60%
	5) 70%
c. Perform a calorimetric	6) 90%

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

QUESTION 4.11 (2.50)

Match the terms in column A to the values in column B for the radiation exposure guidelines. Assume whole body dose unless otherwise stated. CAUTION: Some answers could be used more than once. (0.5 ea)

COLUMN A	COLUMN B
a. NRC limits/qtr	1. 0.5 REM
b. Virginia Power limits/qtr	2. 1.25 REM
c. NRC pregnant woman limit/gestation	3. 1.0 REM
d. NRC general public limit/year	4. 0.75 REM
e. NRC quarterly limit with a Form 4	5. 5 REM
	6. 3 REM

QUESTION 4.12 (1.00)

List the 4 methods given in the S/G Tube Rupture EOP to identify which S/G is ruptured.

QUESTION 4.13 (1.50)

Following a valid reactor trip and safety injection, what are the Reactor Coolant Pump Trip Criteria? (Assume normal containment conditions)

QUESTION 4.14 ~~(1.00)~~ (1.25)

List the immediate operator actions to initiate emergency boration if it is required on an Anticipated Transient Without Trip condition. Assume Safety Injection has not accuated and is not desired.

QUESTION 4.15 (1.50)

List the SI termination criteria following a LOCA. (Include all appropriate values)

QUESTION 4.16 (1.00)

List the 4 DISTINCT hazards to which personnel are exposed when an entry into the reactor compartment is made during reactor operations.

QUESTION 4.17 (1.00)

List four of the critical conditions required to be recorded during a startup when 1 X 10E-8 amps is attained.

QUESTION 4.18 (1.00)

List ALL immediate operator actions required by 1-AP-14, Low Condensor Vacuum, if condensor vacuum lowers, but does not increase above 9.5" HG absolute.

QUESTION 4.19 (1.00)

List all of the immediate operator actions if a valid Reactor Coolant Pump Vibration DANGER Annunciator is received while at 30% power?

QUESTION 4.20 (2.50)

List FIVE indications of a loss of Component Cooling Water in accordance with AP-15, Loss of Component Cooling.

QUESTION 4.21 (3.00)

List ALL immediate actions required by 1-AP-14, Loss of Reactor Coolant System Pressure. List only those items that would bbe verified, not items from the "Response Not Obtained" category.

QUESTION 4.22 (1.00)

Briefly explain the effect that placing an "unsaturated" mixed bed demineralizer in service will have on the reactor coolant system and on control of the reactor.

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

QUESTION 4.23 (1.00)

During a natural circulation cooldown, it is desired to cooldown using the steam dumps. Which MODE is the steam dump system operated in and WHY?

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

ANSWER 1.01 (2.00)

A. It doubles (or increase) the head for a given mass flow rate.

B. It will double (or increase) the mass flow rate capacity for a given head.

REFERENCE
Surry lesson plan ND-83-LP-8, Rev 1, p8.18; NA NCRODP-83 191004; K1.09/1.10(2.4/2.4)

ANSWER 1.02 (1.00)

c

REFERENCE
Surry lesson plan ND-83-LP-8, Rev 1; NA NCRODP-83 191004; K1.14(2.4)

ANSWER 1.03 (1.00)

With a DNBR of 1.3, during normal operation and anticipated operational occurrences, there is (a 95%) confidence that DNB does not occur. When > 1.3 likelihood of DNB occurring decreases.

REFERENCE
Surry lesson plan ND-86.3-LP-2, p2.10; NA NCRODP-83, ARR-13 193008; K1.10(2.9)

ANSWER 1.04 (2.00)

1. reactor power

2. coolant flow rate

3. RCS ~~cold~~ temperature ~~(Tc)~~ Tavg or T_H

4 RCS pressure.

REFERENCE
Surry lesson plan ND-86.3-LP-2, p2.10; NA TS 2.1
193008; K1.05(3.4)

ANSWER 1.05 (2.00)

1. Density difference (or DELTA T) created by heat addition by the heat source and heat removal by the heat sink.

2. The heat sink must be elevated physically above the heat source.

REFERENCE
Surry lesson plan ND-86.3-LP-4, p4.5; NA NCRODP-83, ARR-12
193008; K1.21(3.9)

ANSWER 1.06 (1.00)

e

REFERENCE
Surry lesson plan ND-86.3-LP-3, pp3.4, 3.5, 3.7, 3.10, 3.12
193009; K1.05(3.1)

REFERENCE
NA TS 3.1

PWG-5(2.9/3.9)

ANSWER 1.07 (.50)

Self shielding / Self shielding of the fuel pellet.

REFERENCE
Surry lesson plan ND-86.2-LP-1, p1.7; NA NCRODP-86.1
192001; K1.08(2.3)

ANSWER 1.08 (1.50)

a. DECREASES

b. ~~DECREASES~~ INCREASES

c. DOES NOT CHANGE

REFERENCE
Surry lesson plan ND-86.2-LP-1, p1.4, 1.11; NA NCRODP-86.1
192004; K1.07(2.9)

ANSWER 1.09 (1.00)

b

REFERENCE
Surry lesson plan ND-86.2-LP-1, p1.16; NA NCRODP-86.1
192004; K1.05(2.3)

ANSWER 1.10 (1.00)

d

REFERENCE
Surry lesson plan ND-86.2-LP-4, p4.4, 4.8; NA NCRODP-86.1
192006; K1.06(3.4)

ANSWER 1.11 (1.00)

Neutrons at or near the edge of the core have a higher probability of leaking out than the ones at the center which have a higher probability of causing fission. (Hence: DRW at center is > than at edge).

REFERENCE
Surry lesson plan ND-86.2-LP-6, p6.12; NA NCRODP-86.1
192005; K1.14(3.2)

ANSWER 1.12 (1.00)

The presence of adjacent control rods may cause a significant change in an individual control rod worth.

REFERENCE
Surry lesson plan ND-86.2-LP-6, p6.19; NA NCRODP-86.1
001/000; K5.05(3.5)

ANSWER 1.13 (1.50)

Start up rate is positive and constant, reactor power is increasing, and there is no outward rod motion.

REFERENCE
Surry lesson plan ND-86.2-LP-7, p7.51; NA NCRODP-86.1
192008;K1.11(3.8)

ANSWER 1.14 (1.00)

(two of the three answers below required)

1. Neutron production is relatively high, so power is constant when the reactor is critical.

2. Below 10 exp-8 amps the output of the intermediate range may not be directly proportional to the neutron population.

3. Reactivity has not yet been changed by the moderator or fuel temperature.

REFERENCE
Surry lesson plan ND-86.2-LP-7, p7.57; NA NCRODP-86.1
192008; K1.12(3.5)

ANSWER 1.15 (2.00)

a) 5
b) 6
c) 4
d) 7

REFERENCE
Surry lesson plan ND-83-LP-(1-10); NA NCRODP-83
193008; K1.10/K1.06(2.9/2.8)

ANSWER 1.16 (1.00)

c

REFERENCE
Surry lesson plan ND-86.1-LP-6, p6.35; NA NCRODP-86.1
000/015; K5.06(3.4)

ANSWER 1.17 (1.00)

0.1hr = 6 min

P = Po 10 exp SUR(t) (+.5)

P = Po 10 exp 0.1 dpm(6 min) (+.25)

P = Po 10 exp 0.6 (+.25)

P = 3.98 Po

REFERENCE
Surry lesson plan ND-86.1-LP-8, p8.12; NA NCRODP-86.1
192003; K1.09(2.3)

ANSWER 1.18 (1.00)

a or c

REFERENCE
Glasstone & Sesonske. Nuclear reactor engineering
third ed. New York: Van Nostrand Reinhold Co., 1981.
192003; K1.01(2.7)

ANSWER 1.19 (1.50)

Q = UA DELTA T (+.5)
A = 25' (2 pi r) (+.25)
DELTA T = 70 deg F (+.25)
0.5" = .042' (+.25)
Q = 1.565BTU/(sq ft-deg F) x 25ft x 2 pi r x 70 deg F (+.25)
Q = 723 BTU

REFERENCE
Surry lesson plan ND-83-LP-1, p1.27; NA NCRODP-83
193007; K1.08(3.1)

ANSWER 1.20 (1.50)

1. Gradual warm up of steam lines

2. Proper venting of tanks and components during warm up and operation.

3. Steam traps

4. Lines kept full

(Others as appropriate)

REFERENCE
Surry lesson plan ND-83-LP-8, p8-36; NA NCRODP-83
193006; K1.04/1.10(3.4/3.3)

ANSWER 1.21 (2.00)

a. REMAIN THE SAME

b. DECREASE

c. INCREASE

d. DECREASE

REFERENCE
Surry lesson plan ND-83-LP-(1-10); steam tables; NA NCRODP-83
193004; K1.15(2.8)

ANSWER 1.22 (1.50)

a. Decrease

b. Decrease

c. Increase

REFERENCE
Surry lesson plan ND-83-LP-8; NA NCRODP-83
191004; K1.15(2.6)

ANSWER 1.23 (1.00)

a) Will be the same (+.5 ea)
b) Unit B will be higher

REFERENCE
Westinghouse Reactor Core Control, pp 6-23/26
Westinghouse Fundamentals of Nuclear Reactor Theory, pp 8-48/60

001/010; K5.08(2.9/3.2) & 001/000: K1.05(4.5/4.4)

ANSWER 2.01 (1.00)

d (1.0)

REFERENCE
NA NCORDP 88.1 Reactor Coolant p 2.24
007/000; A3.01 (2.7/2.9)

ANSWER 2.02 (1.00)

d (1.0)

REFERENCE
NA NCROPD-77 RPS p 39
015/000; K4.01 (3.1/3.3)

ANSWER 2.03 (1.00)

b 1.00

REFERENCE
NA NCORDP 91.1 ESF-QSS
026/000; K4.01 (4.2/4.3)

ANSWER 2.04 (1.00)

a aaaaaaaaaaaaaaa

REFERENCE
NA NCRODP 88.1 "RCS-PZR and Press. Relief"
002/000; K4.03 (2.9/3.2)

ANSWER 2.05 (1.00)

a (1.0)

REFERENCE
NA NCRODP 93.8 PZR Press. Control and Protect.
010/000; A4.01 (3.7/3.5)

ANSWER 2.06 ~~(1.50)~~ 1.25

A) 1) 1J1 (a) (0.25)
   2) 1B1 (b) (0.25)
   3) 1H1 (d) (0.25)
   4) 1C1 (c) (0.25)
B) 1A1 (e) (0.25)

REFERENCE
NA NCRODP 93.8
010/000 K2.01 (3.0/3.4)

ANSWER 2.07 (1.00)

d

REFERENCE
NA NCRODP 91.1 ESF p.2.29
013/000 K1.01 (4.2/4.4)

ANSWER 2.08 (1.00)

b bbbbbbbbbbbbbbb

REFERENCE
NA NCRODP 88.2 RHR System
1-OP-14.1

ANSWER 2.09 (1.00)

a) Open (0.25 ea)
b) Closed
c) Open
d) Open

REFERENCE
NA NCRODP 91.1, "ESF-RSS"
026/000; K1.02 (4.1/4.1)

ANSWER 2.10 (1.00)

a) Fails closed (0.2 ea)
b) Fails closed
c) Fails closed
d) Fails open
e) Fails as is (or this valve is a motor operated valve and not affected by instrument air.

REFERENCE
NA NCRODP 88.3 Chemical and Volume Control
004/000; A2.04 (3.6/4.2)

ANSWER 2.11 (1.50)

a) True (0.5 ea)
b) False
c) False

REFERENCE
NA NCORDP 88.1 "RCS-RCP"
002/000; K1.13 (4.1/4.2)

ANSWER 2.12 (.50)

TRUE (0.5)

REFERENCE
NA NCRODP 93.5 Rod Control
001/000; K4.03 (3.5/3.8)

ANSWER 2.13 (1.50)

RCP discharge and the PZR or DP across the core, and water level in the PZR (0.75 ea)

REFERENCE
NA NCRODP 88.1 RCS p. 2.9
002/000; K1.09 (4.1/4.1)

ANSWER 2.14 (2.00)

1) To upper head plenum via nozzles in core barrel flange. (0.5)
2) Between hot leg discharge nozzles and upper core barrel outlets. (0.5)
3) Between baffle plates and core barrel. (0.5)
4) Around inserts in guide thimble tubes in the fuel assemblies. (0.5)

REFERENCE
ND 88.1 LP-2 p. 2.32
002/000; A1.05 (3.4/3.7)

ANSWER 2.15 (1.50)

-Service Water Reservoir (0.5 ea)
-Lake Anna
-Discharge Canal or Bladder Tank

REFERENCE
NA NCRODP 89.4, Feedwater Systems-AFW
086/000; K1.03 (3.4/3.5)

ANSWER 2.16 (2.00)

(any 4 of 5 at 0.5 ea)
1) Max. Fuel Element Cladding Temp. < 2200 Deg. F
2) Cladding Oxidation < 17% thickness
3) Hydrogen generated by Zirc-Water reaction <1% of max. possible.
4) Core remains in a coolable geometry
5) Provides for long term decay heat removal

REFERENCE
10CFR50.46
NA NCRODP 91.9 ESF p.2.6/2.7
006/050; PWG 4 (4.2/4.3)

ANSWER 2.17 (2.00)

1) Motor current (0.25 ea)
2) Bearing temperature
3) Seal injection flow, 6-~~9~~ gpm
4) Seal leak off flow, 0.2-~~3~~ gpm
5) Seal differential pressure > or = 200 psid

REFERENCE
NA NCROPD 88.1 p. 3.22
OP 5.2 p. 8/9
003/000; PWG-7 (3.5/3.9)

ANSWER 2.18 (1.50)

1) 4 Recirculation spray heat exchangers (0.25 ea)
2) Back-up for containment Recirculation air coolers (if needed)
3) Charging pumps oil
4) Charging pumps water coolers
5) One compressor per unit
6) One control room air conditioner system per unit

REFERENCE
NA NCRODP 92.2 Service Water
076/000; K1.19 (3.6/3.7)

ANSWER 2.19 (1.00)

From its own (0.5) 125 VDC Distribution system (0.5)

REFERENCE
NA NCRODP 90.4 Print ESIC 11C sh 6
064/000; K1.04 (3.6/3.9)

ANSWER 2.20 (0.75) ~~(1.50)~~

1) ~~To remove radioactive iodine from the containment atmosphere.~~ ~~(0.75)~~
2) To control the pH of the water that collects in the containment sump. ~~A basic pH helps to prevent Chloride stress corrosion.~~ (0.75)

REFERENCE
NA NCRODP 91.1 ESF-Quench Spray System
026/000; K4.02 (3.1/3.6)

ANSWER 2.21 (1.50)

To provide a path to keep the RHR system full (0.50) and to allow for expansion of the system during heat up of the RCS (0.5) and thus ambiently heating up RHR (0.50). or to aid in warming up the RHR system (.5) or " for overpressure protection anytime (.5)

REFERENCE
NA NCRODP 88.2 RHR
004/000; K1.01 (3.4/3.9)

ANSWER 2.22 (1.50)

1) Prevents shocking the regenerative heat exchanger and the orifices (0.75)
2) Keeps the regenerative heat exchanger and associated piping pressurized to prevent flashing (0.75)

REFERENCE
NA NCRODP 88.3 CVCS p. 6
004/020; K4.03 (3.0/3.4)
004/020; K6.12 (2.9/3.1)

ANSWER 2.23 ~~(1.50)~~ .75

~~a)~~ deleted (3)	motor generator set	b)	2
(4)	reactor trip breaker		2
(2)	power cabinet		4
(7)	logic cabinet		1
(6)	rod position indication cabinet		4
(5)	automatic rod control unit		1
(1)	DC hold cabinet		1

~~(0.75 for~~ a) fully correct, 0.75 for b) fully correct,
~~-0.1 for each switch needed to place a component in proper order)~~

REFERENCE
NA NCRODP 93.5 Rod Control System
001/000 K4.01 (3.5/3.8)

ANSWER 3.01 (1.00)

a

REFERENCE
NA NCRODP 93.1
012/000; K6.10 (3.3/3.5)

ANSWER 3.02 (1.00)

c cccccccccccccccc

REFERENCE
NA NCRODP 91.1 "ESF"
005/000; K4.11 (3.5/3.9)

ANSWER 3.03 (1.50)

a) STSP decreases (0.5 EA)
b) STSP decreases
c) STSP decreases

REFERENCE
NA NCRODP 77 RPS p 25
012/000; A1.01 (2.9/3.4)

ANSWER 3.04 (1.00)

a (1.0)

REFERENCE
NA NCRODP 93.9 Main Generator Control & Protection
045/010; K1.11 (3.6/3.7)

ANSWER 3.05 (1.25)

40% setpoint from 0-20% (0.5) Turbine power (0.25)
and linearly from 40-110% as Turbine Power goes from 20-100%
(0.5)

REFERENCE
NA NCRODP 91.1 "ESF-SI of ECCS"
013/000; K1.01 (4.2/4.4)

ANSWER 3.06 (1.00)

a or C

REFERENCE
Westinghouse PWR Systems Manual, Sect 10.2 PZR Pressure Control
010/000; K4.03 (3.8/4.1) & K6.01 (2.7/3.1) & PWG-4 (3.6/3.7)

ANSWER 3.07 (1.00)

a (1.0)

REFERENCE
NA NCRODP 93.10 RPS
012/000;K6.11 (2.9/2.9) & A2.05 (3.1/3.2)

ANSWER 3.08 (1.00)

c (1.0)

REFERENCE
NA NCRODP 93.4 "CORE COOLING MONITOR"
NA NCRODP 93.4 Learning Objective; Section I, 2.4

ANSWER 3.09 (1.00)

a (1.0)

REFERENCE
NA NCRODP 93.8 PZR Level Protection and Control
011/000 A2.10 (3.4/3.6)

ANSWER 3.10 (1.00)

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

REFERENCE
NA NCRODP 93.10 RPS
012/000; K4.09 (2.8/3.1)

ANSWER 3.11 (1.00)

a

REFERENCE
Westinghouse PWR System Manual, "SGWLC"

035/010; A2.03 (3.4/3.6)

ANSWER 3.12 1.00 ~~(1.50)~~

a) TRUE (0.5 EA)
b) FALSE
deleted ~~c) TRUE~~

REFERENCE
NA NCRODP 91.1 p.2.18
013/000; K1.11 (3.3/3.8)

ANSWER 3.13 (.50)

TRUE

REFERENCE
NA NCRODP 93.2 Excore Instrumentation System
015/000 K1.01 (4.1/4.2)

ANSWER 3.14 (.50)

FALSE

REFERENCE
NA NCRODP 90.4 p2.45
062/000 K1.02 (4.1/4.4)

ANSWER 3.15 (2.00)

1) PZR Hi Press. Trip(0.2) 2385 psig(0.1), 2/3(0.1)
2) PZR Lo Press. Trip(0.2) 1870 psig(0.1), 2/3(0.1)
3) PZR Lo-Lo Press. SI(0.2) 1765 psig and not blocked(0.1), 2/3(0.1)
4) P-11(0.2) <2000 psig(0.1), 2/3(0.1)
5) Press. input to the OT Delta T(0.4)

REFERENCE
NA NCRODP 93.8
010/000; K1.01 (3.9/4.1)

ANSWER 3.16 (1.50)

1) Compensate for power defect or to maintain minimum shutdown margin (0.5 ea)
2) To minimize the amount of positive reactivity inserted during a rod ejection accident, and
3) To minimize radial flux tilt (peaking) or to maintain acceptable power distribution limits.

REFERENCE
NA NCRODP 77 RPS
001/000; K5.04 (4.3/4.7)

ANSWER 3.17 (1.00)

1) Control switch to close (0.25)
2) Synchronizing selector switch is ON (0.25)
3) DG terminal voltage is 95% (0.25)
4) Breaker 86 and 87 protective relay's are reset (0.25)
(86 -Breaker Overcurrent, 87- Phase differential)

REFERENCE
NA NCRODP 90.4 EDG p2.57
064/000; A4.01 (4.0/4.3)

ANSWER 3.18 (1.50)

-PZR high water level (0.25 ea)
-PZR lo pressure
-Lo primary coolant flow
-RCP breakers open (two pumps)
-Under voltage on both 4 KV buses
-Turbine trip
- Underfrequency on both 4KV buses

REFERENCE
NA NCRODP 93.10
012/000; K4.06 (3.2/3.5)

ANSWER 3.19 (1.50)

-Normal: ~~480 VAC Vital~~ (0.5 ea) 125 v dc vital bus (→ 125 vdc/120 vac static inverter)
alternate ~~Standby:~~ ~~125 VDC Battery~~ 480 v ac emergency panel (1H1, 1J1) → 480/120 vac transformer
emergency ~~Maintenance:~~ ~~120 VAC~~ battery (to 125 v dc vital) bus

REFERENCE
NA NCRODP 90.3 Vital and Emergency Distribution
062/000; K4.09 (2.4/2.9)

ANSWER 3.20 (1.00)

1) RCS pressure >2000 psig
2) SI

REFERENCE
NA NCRODP 91.1 p2.15
006/000 A3.01 (4.0/39)

ANSWER 3.21 (1.00)

The operator must manually (0.5) initiate over pressure protection (0.5). (Manually open PZR PORV's)

REFERENCE
NA NCRODP 93.8 PZR Press. Control and Protection
010/000; K4.03 (3.8/4.0)

ANSWER 3.22 (1.00)

The design change resulted because of experiences where the undervoltage trip signal alone was not sufficient to trip the breaker. (1.0)

REFERENCE
NA NCRODP 77 RPS p.35
012/000; K6.03 (3.1/3.5)

ANSWER 3.23 (1.00)

The highest reading upper/lower detector is compared to the average of the upper/lower detectors (0.5). The circuit auto defeats below 50% power on ALL channels (0.5).

REFERENCE
NA NCRODP 93.2 Excore Instrumentation Sys.
015/000; K6.04 (3.1/3.2) & A1.04 (3.5/3.7)

ANSWER 3.24 ~~(1.00)~~ 1.25

a) Inputs are the S/G level instruments (0.25), Turbine first stage pressure (0.25) Nuclear Instruments (N44) (0.25)

b) The output and demand signals must be approximately 0 (0.5)

REFERENCE
NA NCRODP 93.12 SGWLC
OP 31.0 MAIN FEEDWATER
059/000; A4.08 (3.0/2.9)

ANSWER 3.25 (1.75)

a) Cabinet has the capacity to support up to 6 stationary gripper coils simultaneously (0.5). So with 2 groups or more, would overload/heat the cabinet (0.5).

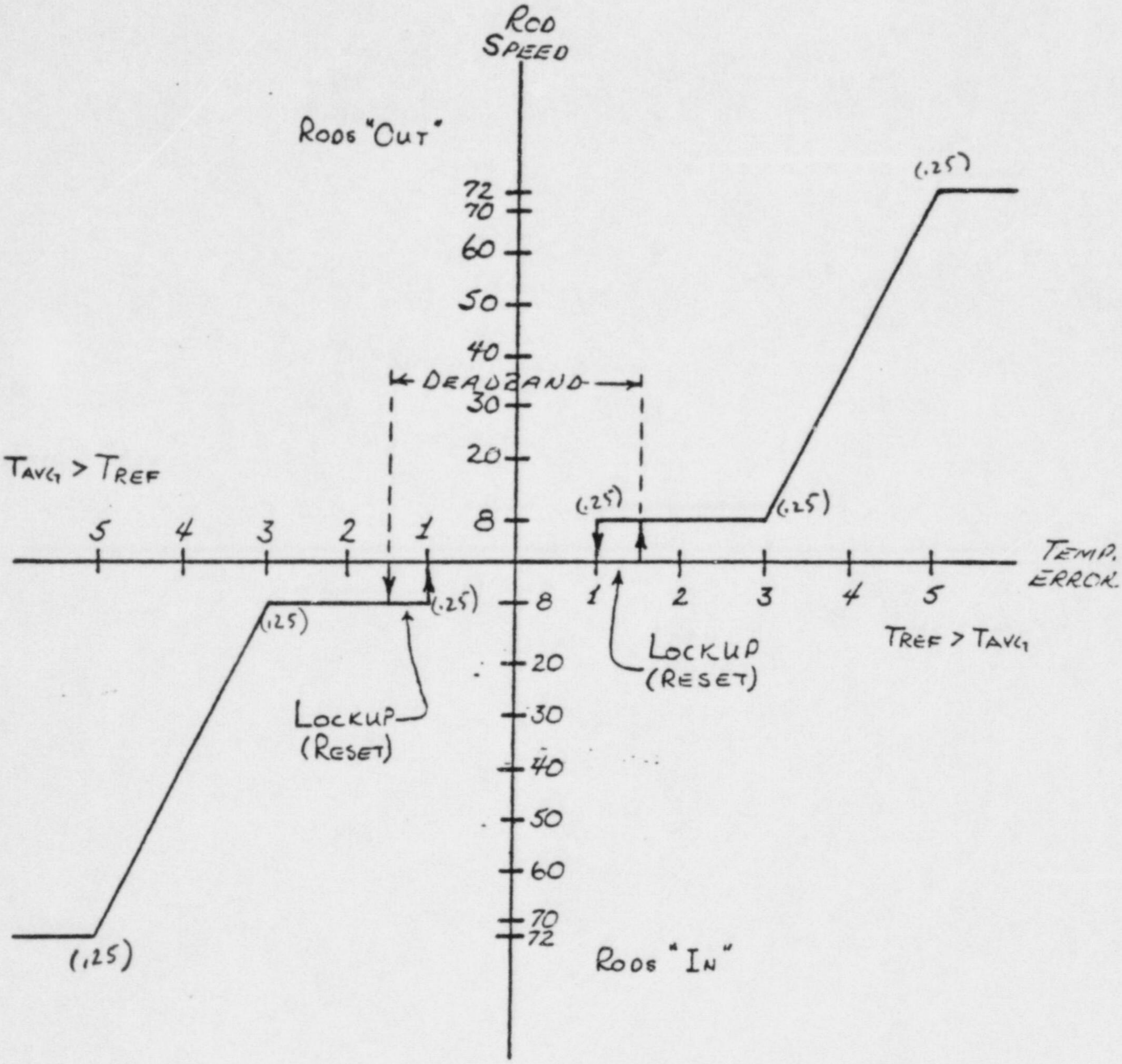
b) 125 VDC-Latching Rods
70 VDC-Holding Rods (0.5 for reasons, 0.25 for correctly associating voltages)

REFERENCE
NA NCRODP 93.5 Rod Control
001/050; PWG-1(3.6/4.1)

ANSWER 3.26 (1.50)

See attached sketch

REFERENCE
NA NCRODP 93.5
001/000; K4.03 (3.5/3.8)



CONTROL ROD SPEED VS TEMPERATURE ERROR
(AUTOMATIC ROD CONTROL)

ANSWER ~~4.01~~ deleted ~~(1.00)~~ 1.00

2 or no correct answer since minimum seal injection flow is 6gpm

REFERENCE
VCS, SOP-101 p1
NA OP-5.2 p 4
SUR OP-5.2 p 2,4

ANSWER 4.02 (1.00)

four @ 0.25 points each:

1. Rod bottom light
2. Computer alarm, power range tilt, rod deviation/sequence
3. Flux deviation alarm(s)
4. Rapid drop in Tavg and power level
5. Rapid drop in przr level and pressure
6. Power range negative rate alarm

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

001/050 PWG-10 4.3/4.5

ANSWER 4.03 (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.
SUR 1-AP-20, p 5

ANSWER 4.04 (1.00)

b

REFERENCE
NA ADM 5.8, pp 2/3
Sur SUADM-ADM-21 p 21
PWG-23: Plant Staffing and Activities (2.8/3.5)

ANSWER 4.05 (1.00)

b

QF=1 for gamma

100(45/60)(1)=75

REFERENCE
10 CFR 20.

PWG-15: Radcon Knowledge (3.4/3.9)

ANSWER 4.06 (1.00)

a

REFERENCE
MNS, AP/2/A/5500/14, Case I, p.2.
CAT, AP/1/A/5500/15, Case I, p.2.
Surry AP-1.00 pp 2,3
NA, AP-1.0 p 4
001/050; PWG-11(4.4/4.4)

ANSWER 4.07 (2.00)

a) Shutdown (+.5 ea)
b) Startup
c) Shutdown
d) Startup

REFERENCE
SQN TS 3.4.6.2
NA TS 3.4.6.2
SUR TS 3.1-13 002/020; PWG-8 (3.5/4.4)

ANSWER 4.08 (.50)

FALSE

REFERENCE
NA 1-OP-22.11, p 4

ANSWER 4.09 (1.00)

Whenever CRDM's are energized.
or if CRDM's are energized when primary plant temp is 100F to 350F

REFERENCE
NA OP-21.1 p4 NA 1-OP-1.2 p 5
SUR OP-21.3 p 9

ANSWER 4.10 (1.50)

a) 4 (+.5 ea)
b) 2
c) 6 (5 for Surry)

REFERENCE
NA OP-2.1, pp 9-13
Surry OP-2.1.1 pp 14-19

PWG-12: Perform Integrated Plant ops (3.5/3.4)

ANSWER 4.11 (2.50)

a 2
b 4
c 1
d 1
e 6

REFERENCE
Virginai Power GET pp 14-17

PWG-15 3.4/3.9

ANSWER 4.12 (1.00)

-Unexpected rise in S/G level (+.25 ea)
-High radiation on a S/G blowdown line
-High radiation on an MS line monitor
-High radiation as determined by sampling and analysis

REFERENCE
Surry EP-4.00, pp 2
NA 2-EP-3, pp 2

EPE-038; EA2.03 (4.4/4.6)

ANSWER 4.13 (1.50)

1) Verify Charging/SI flow (+.5 ea) (1,2 or 3)
2) RCS Pressure < 1230 psig
3) If component cooling water to any pump is lost

REFERENCE
SQNP Foldout Page
NA Foldout page for 2-EP-0
Surry Foldout page for EP-1.00

003/000; PWG-10 (4.1/4.4)

ANSWER 4.14 (1.25) ~~(1.00)~~

Surry (+.25 ea)
Verify SI/CHG pumps running/flow
Check RCS pressure <2335 psig
Switch BATP to fast speed
Open MOV-( )350

North Anna (+.25 ea)
1. Verify 2 SI/CHG pumps running/flow
2. Switch BATP to fast speed
3. Open MOV 2350 ~~or Inject the BIT~~
4. Check pzr press <2335
5. Inject the BIT

REFERENCE
Surry FRP-S.1 p 3
NA FRP-S.1 p 4

EPE-029; PWG-11 (4.5/4.7)

ANSWER 4.15 (1.50)

NORTH ANNA (+.3 ea)

- RCS Press > 2000 psig & increasing
- RCS Subcooling > 50 Deg F
- PZR Level > 50%
- SG Level > 10% or > 30% Advrs Cntm
  OR
- AFW Flow > 730 GPM

REFERENCE

NA EP-0 Foldout page

006/050; PWG-7 (3.8/4.2)

ANSWER 4.16 (1.00)

ioinizing radiation; heat stress; differential pressure; O2 deficiency (+.25 ea)

REFERENCE
Surry SUADMO-19 p 3
NA ADM 20.9, pp 1

PWG-18: Knowledge of Safety Procedures (3.0/3.1)

ANSWER 4.17 (1.00)

Any 4 @ 0.25 points each:

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

Surry
1. Date Critical
2. Time Critical
3. Average RCS temp.
4. RSC Boron concentration
5. Bank C Control rod position
6. Bank D Control rod position
7. Actual critical position within admin. requirements

REFERENCE
NAPS 1.OP-1.5, p.12.
SUR 1-OP-1C App. A p 10 of 10
001.010; K5.08 (2.9/3.3)

ANSWER 4.18 (1.00)

1. Reduce generator load until vacuum stabilizes
2. Check vacuum breaker (MOV-AS-100) closed (.25) and a water seal present (.25)

REFERENCE
NA 1- AP-14, p 3

ANSWER 4.19 (1.00)

Manually trip the reactor (.50) and the affected RCP (.50)

REFERENCE
NA AP-29, p 2

000/015 PWG-10 4.2/4.5

ANSWER 4.20 (2.50)

CC surge tank low level alarm
CC pump auto trip alarm
CCW low flow discharge header alarm
CCW low pressure discharge header alarm
Reactor coolant pump low flow/high temp alarm
Excess letdown HX low flow/high temp
Non-regenerative HX high temp

REFERENCE
NA AP-15, p 2

008/030 PWG-10 3.8/4.2

ANSWER 4.21 (3.00)

1. Verify pzr porv's closed
2. Verify master controller PC-1-444J not failed
3. Verify pzr spray valves closed
4. Verify aux spray valve closed
5. Verify all pzr heaters on
6. Verify RCS pressure stable or increasing

REFERENCE
NA AP-44 p 3

ANSWER 4.22 (1.00)

An unsaturated mixed bed demineralizer will remove boron from the reactor coolant system (.50) and add positive reactivity(.5)
(Reasonable wording accepted)

REFERENCE
VCS, sop-102 p 1
NA OP-8.2, p 5
SUR OP 8.2, p 2

004/000 K6.02 2.5/2.1

ANSWER 4.23 (1.00)

Steam pressure mode [0.25]

Tavg input to the steam dump control is not valid without forced flow in the loops. [.75] or Tavg mode cannot be used to cooldown below 547F (no load Tavg setpoint)

REFERENCE

NA 1-AP-10, Att. 2, p 2
SUR AP-39, p 4

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

FACILITY: NORTH ANNA 1&2

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 87/02/09

EXAMINER: MOORMAN, J.

CANDIDATE: MASTER

INSTRUCTIONS TO CANDIDATE:

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	CANDIDATE'S SCORE	% OF CATEGORY VALUE	CATEGORY
30.00	25.4 ~~25.00~~			5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
29.0 ~~30.00~~	24.5 ~~25.00~~			6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
30.25 ~~29.25~~ ~~30.00~~	24.7 ~~25.00~~			7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
30.00	25.4 ~~25.00~~			8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
119.25 ~~118.25~~ ~~120.00~~		Final Grade	%	Totals

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

Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

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

2. Restroom trips are to be limited and only c   andidate at a time may leave. You must avoid all contacts with an   outside the examination room to avoid even the appearance or possibility of cheating.

3. Use black ink or dark pencil <u>only</u> to facilitate legible reproductions.

4. Print your name in the blank provided on the cover sheet of the examination.

5. Fill in the date on the cover sheet of the examination (if necessary).

6. Use only the paper provided for answers.

7. Print your name in the upper right-hand corner of the first page of <u>each</u> section of the answer sheet.

8. Consecutively number each answer sheet, write "End of Category __" as appropriate, start each category on a <u>new</u> page, write <u>only on one side</u> of the paper, and write "Last Page" on the last answer sheet.

9. Number each answer as to category and number, for example, 1.4, 6.3.

10. Skip at least <u>three</u> lines between each answer.

11. Separate answer sheets from pad and place finished answer sheets face down on your desk or table.

12. Use abbreviations only if they are commonly used in facility <u>literature</u>.

13. The point value for each question is indicated in parentheses after the question and can be used as a guide for the depth of answer required.

14. Show all calculations, methods, or assumptions used to obtain an answer to mathematical problems whether indicated in the question or not.

15. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

16. If parts of the examination are not clear as to intent, ask questions of the <u>examiner</u> only.

17. You must sign the statement on the cover sheet that indicates that the work is your own and you have not received or been given assistance in completing the examination. This must be done after the examination has been completed.

18. When you complete your examination, you shall:

a. Assemble your examination as follows:

(1) Exam questions on top.

(2) Exam aids - figures, tables, etc.

(3) Answer pages including figures which are part of the answer.

b. Turn in your copy of the examination and all pages used to answer the examination questions.

c. Turn in all scrap paper and the balance of the paper that you did not use for answering the questions.

d. Leave the examination area, as defined by the examiner. If after leaving, you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION 5.01 (1.50)

With respect to reactor thermal limits, indicate whether each of the following statements are TRUE or FALSE.

a. The average linear power density in the core is expressed in units of kw/ft and is the total thermal power divided by the active length of all the fuel rods.

b. The purpose of limiting the enthalpy rise hot channel factor is to prevent bulk boiling from taking place during a LOCA.

c. The purpose of the limit on the heat flux hot channel factor is to insure that fuel clad temperature does not exceed 2200 deg F during normal operations.

QUESTION 5.02 (1.00)

Concerning subcritical multiplication, which one of the following statements is NOT correct?

a. The neutron behavior per generation can be stated mathematically.

b. The neutron population will reach and maintain an equilibrium value.

c. The fuel in the core effectively multiplies the source neutrons.

d. As the source strength is increased, the magnitude of Keff is increased.

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

QUESTION 5.03 (1.00)

Given: Three reactor coolant (RCP) pumps operating in parallel, each with a flow rate "m" and a combined flow rate "M". Out of the four possibilities below, choose the one that best fits if one RCP is secured.

a. The resulting core flow (M) will increase.

b. The resulting core flow (M) will increase along with individual operating RCP flow (m).

c. The resulting core flow (M) will decrease as individual operating RCP flow (m) increases.

d. The resulting core flow (M) will not change due to decrease in RCP back pressure.

QUESTION 5.04 (1.00)

The negative reactivity added when fuel temperature increases is primarily caused by ______.

a. depletion of U-238

b. doppler broadening

c. depletion of U-235

d. fuel pellet swell thus decreasing the gap

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

QUESTION 5.05 (1.00)

Which one of the following statements below is NOT correct regarding xenon behavior following a power increase?
note: [Xe] denotes xenon 135 concentration

a. The minimum [Xe] reached is independent of the magnitude of the power level increase and initial power level.

b. The time to reach equilibrium is also dependent on the magnitude of the power change and final power level.

c. The time to reach the minimum [Xe] is always < 11 hours.

d. The time to reach equilibrium is approximately 40-50 hours.

QUESTION 5.06 (1.50)

Indicate whether each of the following will make the moderator temperature coefficient less negative, more negative, or have no effect.

a. increase temperature

b. decrease boron concentration

c. increase core age

QUESTION 5.07 (1.50)

Write on your answer sheet INCREASES , DECREASES or DOES NOT CHANGE for the following:

The magnitude of the fuel temperature coefficient (FTC):

A. INCREASES / DECREASES / DOES NOT CHANGE with increase in power.

B. INCREASES / DECREASES / DOES NOT CHANGE with core age.

C. INCREASES / DECREASES / DOES NOT CHANGE with decrease in moderator temperature coefficient (MTC).

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

QUESTION 5.08 (1.00)

A centrifugal pump is started up with its discharge valve open. How would the following parameters differ (INCREASE, DECREASE, or REMAIN THE SAME) if the pump was started with its discharge valve shut?

a. Motor current

b. Discharge pressure

QUESTION 5.09 (1.50)

Nuclear reactors are initially loaded with more fuel than is required to bring the reactor critical. The additional fissile material in the core is said to represent built in or excess reactivity. List 3 things that excess reactivity is designed to overcome.

QUESTION 5.10 (2.00)

List the four (4) plant parameters observed to insure that CHF or DNBR are not exceeded.

QUESTION 5.11 (1.50)

On a reactor startup, what 3 conditions indicate the reactor is critical?

QUESTION 5.12 (1.50)

List three things, in practice, that prevent water hammers from occurring

QUESTION 5.13 (1.00)

What effect does rod shadowing have on the worth of control rods?

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

QUESTION 5.14 (1.50)

Attached is a typical boiling curve for water as it approaches, then exceeds, the DNB point. What are the thermodynamic conditions that cause:

a) The decrease in heat transfer rate in Region III?

b) The increase in heat transfer rate in Region IV?

QUESTION 5.15 (2.00)

A. How does Beff vary over the life of the core?

B. How is Beff affected as plutonium isotopes are produced over the life of the core?

C. How is reactor response affected by a lower delayed neutron fraction?

QUESTION 5.16 (2.00)

Given two pumps of equivalent design, operating at the same, constant speed:

A. What will be the effect of placing the two pumps in series (with respect to flow and head)?

B. What will be the effect of placing the two pumps in parallel (with respect to flow and head)?

QUESTION 5.17 (1.00)

What is the design basis of having a DNBR > or = to 1.3?

QUESTION 5.18 (2.00)

What are all the conditions that must be present in order for natural circulation to exist?

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

QUESTION 5.19 (1.00)

GIVEN: Two identical control rods, each absorb an equal amount of neutrons. The neutron flux at the center of the core equals that at the edge of the core. Why do the control rods in the middle of the core (radially) have a greater effect on Keff then the control rods at the edge of the core (radially).

QUESTION 5.20 (1.00)

Give two reasons why 10 exp -8 amps is chosen as a standard reference for critical rod height data.
note: "standard reference" is NOT an acceptable answer

QUESTION 5.21 (2.50)

What are the purposes of each of the following reactor thermal limits? If a specific accident or condition applies, state this in your answer.

a. Reactor safety limits (1.0)

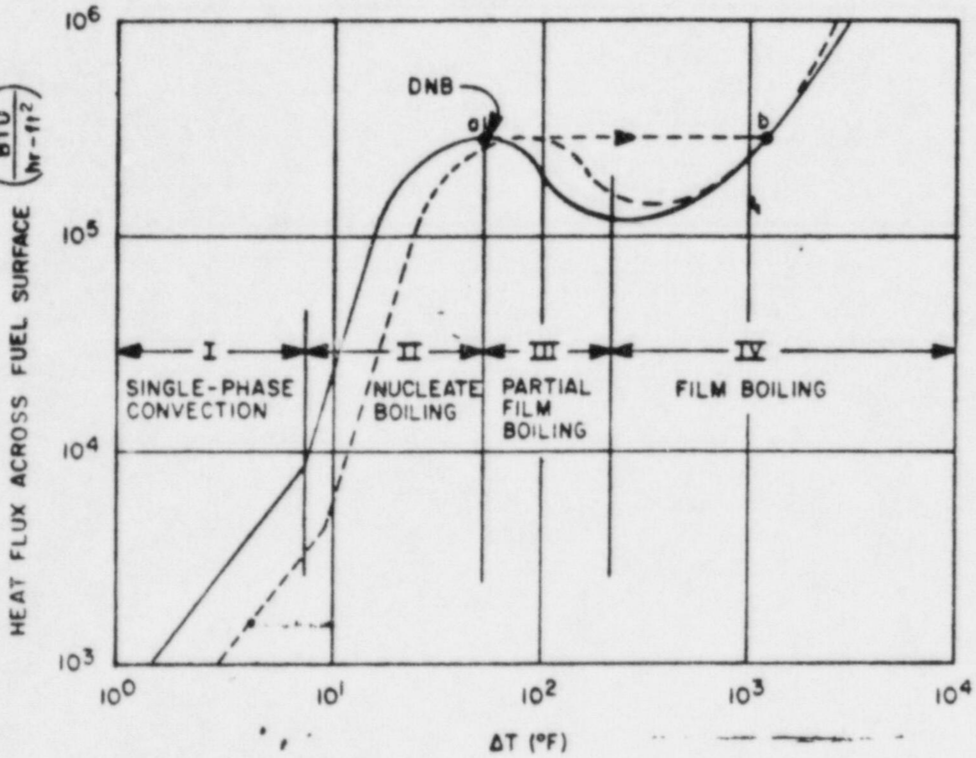
b. Enthalpy rise hot channel factor (Fn(delta H)) (0.5)

c. Nuclear flux hot channel factor (Fq(z)) (1.0)

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

QUESTION 5.14

p 8



(TEMPERATURE DIFFERENCE BETWEEN FUEL ROD SURFACE AND SATURATION TEMPERATURE OF THE COOLANT)

FIGURE FND-HT-102: BOILING CURVE AND DNB AT VARIOUS PRESSURES (REV. 1)

QUESTION 5.14

QUESTION 6.01 (2.00)

LIST 4 of the 5 Design bases for the ECCS Cooling Performance following a LOCA as stated in 10CFR50.46.

QUESTION 6.02 (1.00)

Which valve listed below is used to throttle auxiliary spray flow?

a) FCV-122 (Charging Flow Control Valve)
b) HCV-311 (Aux Spray Valve)
c) PCV-455B (Loop C Spray Valve)
d) PCV-455A (Loop A Spray Valve)
e) You cannot throttle auxiliary spray

QUESTION 6.03 (1.50)

State 3 reasons for having HCV-1142 (RHR letdown penetration from the RHR heat exchangers) kept about 10% open?

QUESTION 6.04 (1.50)

State two purposes for the interlock between the letdown isolation valves, LCV-1460A/B, and the orifice isolation valves, HCV-1200A/B/C.

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

QUESTION 6.05 (1.00)

Which statement below regarding the Source Range Nuclear Instrumentation System is INCORRECT.

a) P-6 allows the source range high level reactor trip signal to be bypassed manually when one of the two intermediate range instruments is above 10 E-10 ion chamber amps.

b) Placing BOTH source range blocking switches to the BLOCK position de-energizes the high voltage supply to both source range instruments.

c) The source range high level trip is blocked when P-10 is present.

d) When P-6 is present and P-10 is not present, the source range high level trip is automatically reinstated and the source range high voltage re-energized when one of the two intermediate ranges is below P-6 reset.

QUESTION 6.06 (.50)

TRUE/FALSE
A RED urgent failure alarm light indicates that a major electrical failure has occurred in the logic cabinet.

QUESTION 6.07 (~~1.50~~) .75

Concerning the Rod Control System:

deleted ~~a)~~ Place the following components in their proper flow path order. Start from the normal power supply and ending at the CRDM's

1) DC hold cabinet
2) Power cabinet
3) Motor generator set
4) Reactor Trip Breaker
5) Automatic Rod Control Unit
6) Rod Position Indication Cabinet
7) Logic Cabinet

b) For the components in Part a, above, STATE the number of each
present in the system.

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

QUESTION 6.08 (1.50)

List the design bases for the minimum level requirements of the Emergency Condensate Storage Tank.

QUESTION 6.09 (1.00)

Which of the following is NOT a design basis of the Steam Dump System?

a) Accommodate ramp load increases greater than 10%/minute.

b) Pass 40% steam flow on a 50% turbine step rejection without a reactor trip occurring.

c) Allow a turbine trip and a subsequent reactor trip from 100% power without lifting the S/G code safety valves.

d) Allow for a smooth shift of plant steam load from the steam dump system to the turbine on a plant startup.

QUESTION 6.10 (1.75) ~~(1.50)~~

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) Excess Letdown	1) Loop A cold leg
b) PZR Surge Line	2) Loop A hot leg
deleted ~~c)~~ CVCS Alternate Charging	3) Loop A intermediate leg
d) PZR Spray Line	4) Loop B intermediate leg
e) RHR Suction	5) Loop C hot leg
	6) Loop C cold leg
	7) Loop C intermediate leg

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

QUESTION 6.11 (2.00)

While performing maintenance, it has been determined that the B charging pump must be tagged out, and the control power fuses for the B charging pump must be removed.

What two manipulations must be done to prevent the letdown orifice isolation valves from closing?

QUESTION 6.12 (2.00)

List 5 protection logic signals generated by the Pressurizer Protection System. (Include in your answer set points, coincidence and associated interlocks, if any)

QUESTION 6.13 (1.00)

The Reactor trip breaker shunt trip coils have been modified to also energize upon any trip signal to the Undervoltage coils. What is the reason for this modification?

QUESTION 6.14 (1.50)

List 3 purposes of Rod Insertion Limits.

QUESTION 6.15 (1.50)

Concerning the Overtemperature Delta Temperature Setpoint (OTSP) describe how (increases, decreases or remains the same) each of the following parameter changes will effect the OTSP.

a) Increase in Tave

b) Decrease in Reactor Pressure

c) Increase in Delta Flux Penalty

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

QUESTION 6.16 (1.00)

Which statement below regarding the Main Generator Protection System is INCORRECT.

a) To prevent a turbine overspeed event, a generator trip always results in a turbine trip when the generator is loaded.

b) Once the generator is loaded, a turbine trip always results in a generator trip.

c) A turbine trip above the protection interlock P-7 (10% power) results in a Reactor trip.

d) A reactor trip always results in a turbine trip.

QUESTION 6.17 ~~(1.50)~~ 1.00

Indicate whether each of the statements below regarding the High Head Safety Injection System (HHSI) is TRUE or FALSE.

a) The alternate power source, J Bus, is ONLY used for maintenance on the "B" char g pump, and this pump has no automatic pump start capability when connected to the J Bus.

b) Normal lead pumps during a SI are the "A" and "B" HHSI pumps.

deleted ~~c)~~ All three pumps get a start signal from a SI signal, but the "A" pump is locked out to allow the "C" pump to start on its normal (H bus), if its breaker is racked out.

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

QUESTION 6.18 (1.50)

Indicate whether each of the statements below regarding permissive
functions associated with the Excore Nuclear Instrumentation is TRUE or FALSE.

a) In order for the P-7 permissive (At Power Trips) to be DISABLED, both reactor power permissive P-10 and turbine power permissive P-13 must clear.

b) The single loop loss-of-flow reactor trip is one of the trips ENABLED by the P-7 permissive.

c) When actuated, the P-10 permissive will automatically DE-ENERGIZE the high voltage to the Source Range Instrument, but it will NOT RE-ENERGIZE the SR Instrument high voltage when P-10 clears.

QUESTION 6.19 (1.25)

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

QUESTION 6.20 (1.00)

List the 4 requirements, control manipulations that will make up the logic to manually close the Diesel Generator output breakers (15H2).

QUESTION 6.21 (1.00)

The Detector Current Comparator receives input from all 4 upper and lower power range detectors. How are these inputs compared, and what conditions are needed to auto bypass circuitry while at power?

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

QUESTION 6.22 (1.75)

a) What consequences could be expected in the Rod Control System's DC Hold Cabinet if 2 or more groups of rod drive mechanisms were placed on hold power (excluding control Bank D rods)?

b) Why is there both a 125 VDC and a 70 VDC power supply in the DC Hold Cabinet?

(***** END OF CATEGORY 06 *****)

QUESTION ~~7.01~~ deleted ~~(1.00)~~ (1.00)

Prior to operating Reactor Coolant Pumps in accordance with OP-5.2, Reactor Coolant Pump Operations, the minimum seal flow should be ____ gpm and VCT pressure should be a minimum of ____ psig.

1. 0 , 10

2. 0.2 , 15

3. 2.0 , 30

4. 5.0 , 20

QUESTION 7.02 (1.00)

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

QUESTION 7.03 (1.00)

Which of the following describes a temporary change which alters the INTENT of a procedure?

a. A change that corrects an incorrect valve lineup.

b. A change that modifies the criteria by which a system's operability is determined.

c. A change that allows partial use of a procedure to test a subtrain without affecting remaining equipment in that train.

d. A change that allows you to change incorrectly specified instruments for data taking.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.04 (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 7.05 (1.00)

One of the source range channels fails on a reactor startup just above the point where P-6 is actuated. Which one statement below describes the correct action(s) that should be taken by the operator?

a. Insert the control banks to the fully inserted position and repair the source range instrument before increasing power above P-6 again.

b. Continue with the reactor startup.

c. Insert control banks until below P-6 , then repair the malfunctioning source range channel before continueing with the startup.

d. Borate the RCS to the shutdown margin requirements of the applicable Technical Specifications section.

QUESTION 7.06 (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 7.07 (1.50)

For each of the following, indicate YES or NO if the conditions violate critical safety function (CSF) red path criteria.

a) Pressurizer level of 5% and RVLIS upper head 80%

b) Total AFW flow 400 gpm with all S/G levels < 6%

c) Containment pressure 65 psig

QUESTION 7.08 (1.50)

Answer the following questions regarding EOP usage TRUE or FALSE:

a) If a Function Restoration Procedure (FRP) is entered due to an ORANGE Critical Safety Function (CSF) condition, and a HIGHER priority ORANGE condition is encountered, the original FRP must be completed prior to proceeding to the newly identified FRP.

b) Unless specified, a task need not be fully completed before proceeding to a subsequent step as long as that task is progressing satisfactorily

c) If a procedure transition occurs, any tasks still in progress from the procedure which was in effect need not be completed.

QUESTION 7.09 (2.00)

Prior to a reactor startup, with the RCS at normal operating pressure and temperature, the following RCS leakages exist. For each leak listed below, indicate whether you could STARTUP or would have to remain SHUTDOWN. (Treat each leak below as an independent event)

a) A leak from an unknown source of 1.5 GPM.

b) 6.0 GPM from a manual valve packing gland.

c) 0.4 GPM from one S/G.

d) 0.1 GPH from the reactor vessel head INNER seal.

QUESTION 7.10 (.50)

You are releaseing radioactive liquid waste in accordance with 1-OP-22.11, Releasing Radioactive Liquid Waste, when one of the operating circulating water pumps trips. You may continue the release for up to 5 minutes while attempting to restart the pump. TRUE/FALSE

QUESTION 7.11 (2.50)

Match the terms in column A to the values in column B for the radiation exposure guidelines. Assume whole body dose unless otherwise stated. CAUTION: Some answers could be used more than once. (0.5 ea)

COLUMN A	COLUMN B
a. NRC limits/qtr	1. 0.5 REM
b. Virginia Power limits/qtr	2. 1.25 REM
c. NRC pregnant woman limit/gestation	3. 1.0 REM
d. NRC general public limit/year	4. 0.75 REM
e. NRC quarterly limit with a Form 4	5. 5 REM
	6. 3 REM

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.12 (1.00)

List the 4 methods given in the S/G Tube Rupture EOP to identify which S/G is ruptured.

QUESTION 7.13 (1.50)

Following a valid reactor trip and safety injection, what are the Reactor Coolant Pump Trip Criteria? (Assume normal containment conditions)

QUESTION 7.14 (1.25) ~~(1.00)~~

List the immediate operator actions to initiate emergency boration if it is required on an Anticipated Transient Without Trip condition. Assume Safety Injection has not accuated and is not desired.

QUESTION 7.15 (1.50)

List the SI termination criteria following a LOCA.(Include all appropriate values)

QUESTION 7.16 (1.00)

List the 4 DISTINCT hazards to which personnel are exposed when an entry into the reactor compartment is made during reactor operations.

QUESTION 7.17 (1.00)

List four of the critical conditions required to be recorded during a startup when 1 X 10E-8 amps is attained.

QUESTION 7.18 (1.00)

List ALL immediate operator actions required by 1-AP-14, Low Condensor Vacuum, if condensor vacuum lowers, but does not increase above 9.5" HG absolute.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.19 (1.00)

List all of the immediate operator actions if a valid Reactor Coolant Pump Vibration DANGER Annunciator is received while at 30% power?

QUESTION 7.20 (1.75)

List the make up flow paths to the Refueling Cavity, in the order of preference, for a loss of refueling cavity level per AP-52, Loss of Refueling Cavity Level During Refueling.
(ie. Containment sump via LHSI pump to refueling cavity)
.35ea, ~~.25 for correct order~~

QUESTION 7.21 (2.50)

List FIVE indications of a loss of Component Cooling Water in accordance with AP-15, Loss of Component Cooling.

QUESTION 7.22 (.75)

What constitutes a Class II reactor trip?

QUESTION 7.23 (1.00)

During normal operations, why is Oxygen concentration in the VCT limited to less than 5% by volume?

QUESTION 7.24 (1.00)

During a natural circulation cooldown, it is desired to cooldown using the steam dumps. Which MODE is the steam dump system operated in and WHY?

(***** END OF CATEGORY 07 *****)

QUESTION 8.01 (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.

QUESTION 8.02 (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 contnuous 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.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.03 (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.

QUESTION 8.04 (1.00)

According to Tech Specs, which of the following is the correct action to be taken if the Radwaste Effluent Monitoring Line Process Monitor is out of service?

a. Effluent releases cannot be performed until the Monitor is back in service.

b. Effluent releases may be performed if Grab Samples are analyzed every twelve hours during the release.

c. Effluent releases may be performed provided two samples taken prior to the release are analyzed and do not exceed 10CFR20 limits and two qualified staff members verify the release rate calculations and the discharge valve lineup.

d. The effluent release may be performed provided a sample prior to the release indicates that the Lower Limit of Detection(LLD) is not exceeded for all the analyses required and subsequent hourly samples during the release confirm this condition continues to exist.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.05 (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 8.06 (1.00)

Which one of the following statements is correct regarding the control and issuance of Special Order Tags (Blue tags)?

a. These tags may be used by all departments except Health Physics

b. These tags may be used in lieu of a mechanical danger tag.

c. The Control Room Operator may authorize tag removal.

d. The tag indicates who must be contacted to operate the equipment.

QUESTION 8.07 (1.00)

Answer TRUE or FALSE to the following:

a) IF a component's emergency power supply is INOPERABLE but all other supporting equipment for that component is OPERABLE, than surveillance requirements on that component must still be performed within the proper time frame.

b) If it is required by an LCO Action Statement to be in HOT STANDBY in 6 hours and then HOT SHUTDOWN in the next 6 hours, it is permissable to be in HOT STANDBY in 3 hours then use the next 9 to be in HOT SHUTDOWN.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.08 (2.50)

Use the attached Technical Specifications to determine the correct response to the questions below regarding Nuclear Instrumentation.

a) What is the MAXIMUM # of each NI that can be out of service at any time without requiring action to reduce the plant operating mode? Assume you are in Mode 1 at 75% power.

b) You are at the minimum # of operable Power Range NIs, when an IC tech requests permission to put an operable PR NI in test for a channel functional test. Can this be done? Refer to applicable TS in answer.

QUESTION 8.09 (1.50)

List 3 additional administrative precautions that must be met to enter a Locked High Radiation Area (> 1r/hr) that are not required for entry into a High Radiation Area (< 1000 mr/hr).

QUESTION 8.10 (1.00)

a) During a non-emergency situation, who must authorize a temporary change to a operating procedure which does not change the procedural intent?

b) What 2 forms are temporary changes and permanent changes to procedures documented on?

QUESTION 8.11 (1.00)

List 3 of the 4 pieces of information that the Shift Supervisor must obtain for transmittal to the appropriate medical facility prior to transporting a contaminated, injured worker off-site, IAW EPIP 5.01.

QUESTION 8.12 (1.00)

While in a refueling mode of operation, with A RHR pump in operation circulating reactor coolant, and the normal power supply to the J Bus out of service due to maintenance, it is discovered that the EDG supply to the H bus has a malfunctioning air distributor making it INOPERABLE and that the water level above the reactor vessel flange has dropped below 23 feet. What action, if any, is required? Use the attached Technical Specifications and identify those which are applicable.

QUESTION 8.13 (1.50)

List the support equipment in TS 3.8.1, "AC Sources", required for a Diesel Generator to be considered operable (there are 3 different criteria that must be met).

QUESTION 8.14 (1.50)

List five hard copy sources of information that are referred to when performing a post trip review, following an unplanned reactor trip.

QUESTION 8.15 (1.00)

List the four conditions, as stated in ADM-20.9, "Containment Egress and Ingress", that will initiate containment evacuation.

QUESTION 8.16 (2.00)

What are the 4 conditions listed in the EPIPs that dictate when updates should be given to offsite authorities regarding an emergency, subsequent to the initial 15 minute notification?

QUESTION 8.17 (1.50)

a) What are the two emergency exposure limits addressed in EPIP 5.06, "Emergency Radiation Exposure Authorization", for damage control considerations. (0.5)

b) List 4 criteria that should be considered by the Station Emergency Manager in selecting personnel for emergency radiation exposure (1.0)

QUESTION 8.18 (1.00)

What is meant by the statement in Technical Specifications that says "The provisions of Specification 3.0.4 are not applicable"?

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.19 (1.00)

Why, in the attached Tech Spec 3/4.4.8, is it a requirement to cooldown to less than 500 degrees as an action if the RCS activity limits are exceeded?

QUESTION 8.20 (1.00)

As stated in 10CFR50.54, under what conditions may actions be taken that depart from a license condition or a technical specification, and who, as a minimum, must approve such action?

QUESTION 8.21 (2.00)

a) State one of the two accidents specified in the bases for TS 3.4.9.3 "Overpressure Protection Systems" that having two operable PORVs when less than 340 deg F in a cold leg is supposed to sufficiently protect against.

b) Explain why the TS for the OPMS require that Pressurizer level be no more than 457 cubic feet when Tavg is between 320 and 340 degrees F.

QUESTION 8.22 (1.50)

a) What are the three criteria which define an Excluded Radiation Worker? Include any applicable dose limits.

b) Assuming an Excluded Radiation Worker is a station employee, what entry conditions, if any, are required to enter a Radiation Area or a Restricted Control Area.

QUESTION 8.23 (1.00)

What is different in the forms and procedures that are required when performing a simple jumper installation (e.g. lifted lead) as opposed to a jumper installation requiring multiple steps? Assume that no approved procedure exists in either case initially.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.24 (1.00)

If is is determined, OUTSIDE of normal working hours, that an Emergency Work Order is necessary, what two personnel shall be notified by the Shift Supervisor and what two forms need to be processed/initiated prior to commencement of work?

(***** END OF CATEGORY 08 *****)

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

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Limiting Conditions for Operation and ACTION requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for each Specification.

3.0.2 Adherence to the requirements of the Limiting Condition for Operation and/or associated ACTION within the specified time interval shall constitute compliance with the Specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour ACTION shall be initiated to place the unit in a MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within 6 hours,
2. At least HOT SHUTDOWN within the next 6 hours, and
3. At least COLD SHUTDOWN within the following 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications. This specification is not applicable in MODES 5 or 6.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceeding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3.1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	12
2. Power Range, Neutron Flux	4	2	3	1, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2##	4
B. Shutdown	2	1	2	3*, 4* and 5*	15
C. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature ΔT					
Three Loop Operation	3	2	2	1, 2	7#
Two Loop Operation	3	1**	2	1, 2	9

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

**The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped condition.

***With the Reactor Trip Breaker open for surveillance testing in accordance with Specification Table 4.3-1 (item 21A).

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above P-6.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1 provided the other channel is operable.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of the redundant channel(s) per Specification 4.3.1.1.1.

c. Either, THERMAL POWER is restricted to ≤ 75% of RATED THERMAL POWER and the Power Range, Neutron Flux trip setpoint is reduced to ≤ 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours.

d. The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75 percent of RATED THERMAL POWER with one Power Range Channel inoperable by using the moveable incore detectors to confirm that the normalized symmetric power distribution, obtained from 2 sets of 4 symmetric thimble locations or a full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

TABLE 3.3-1 (Continued)

a. Below P-6, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above P-6 but below 5% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

c. Above 5% of RATED THERMAL POWER, POWER OPERATION may continue.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below P-6, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above P-6, operation may continue.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - Not applicable.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and POWER OPERATION may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 8 - Not applicable

REFUELING OPERATIONS

RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION 28

ALL WATER LEVELS

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal loop shall be in operation. 28

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 A residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 4 hours. 28

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible. 28

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per Specification 4.0.5.

*The normal or emergency power source may be inoperable for each RHR loop.

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. $\leq$ 1.0 µCi/gram DOSE EQUIVALENT I-131, and

b. $\leq 100/\bar{E}$ µCi/gram.

APPLICABILITY: MODES 1, 2, 3, 4 and 5

ACTION:

MODES 1, 2 and 3*

a. With the specific activity of the primary coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} < 500°F within 6 hours.

c. With the specific activity of the primary coolant > $100/\bar{E}$ µCi/gram, be in at least HOT STANDBY with T_{avg} < 500°F within 6 hours.

MODES 1, 2, 3, 4 and 5

a. With the specific activity of the primary coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131 or > $100/\bar{E}$ µCi/gram, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A special report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2. This report shall contain the results of the specific activity analyses together with the following information:

*With $T_{avg} \geq$ 500°F.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a lift setting of: 1) less than or equal to 420 psig whenever any RCS cold leg temperature is less than or equal to 375°F, and 2) less than or equal to 350 psig whenever any RCS cold leg temperature is less than 185°F, or

b. A reactor coolant system vent of greater than or equal to 2.07 square inches, or

c. A maximum pressurizer water volume of 457 cubic feet with all RCS cold leg temperatures greater than or equal to 320°F.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 375°F, except when the reactor vessel head is removed.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through 2.07 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

b. With both PORVs inoperable, depressurize and vent the RCS through a 2.07 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

ANSWER 5.01 (1.50)

a. true
b. false
c. false

REFERENCE
surry LP-ND-86.3; NA TS 2.1 bases

193009; K1.05(3.5)

ANSWER 5.02 (1.00)

d

REFERENCE
Surry lesson plan ND-86.2-LP-7, p7.28-7.34; NA NCRODP-86.1
192003; K1.01(2.8)

ANSWER 5.03 (1.00)

c

REFERENCE
Surry lesson plan ND-83-LP-8, Rev 1; NA NCRODP-83
191004; K1.14(2.5)

ANSWER 5.04 (1.00)

b

REFERENCE
Surry lesson plan ND-86.2-LP-1, p1.16; NA NCRODP-86.1
192004; K1.05(2.4)

ANSWER 5.05 (1.00)

~~b~~ a

REFERENCE
Surry lesson plan ND-86.2-LP-4, p4.11; NA NCRODP-86.1
192006; K1.02(3.1)

ANSWER 5.06 (1.50)

a. more negative

b. more negative

c. more negative

REFERENCE
Surry lesson plan ND-86.2-LP-2, p2.11-2.17; NA NCRODP-86.1
192004; K1.06(3.1)

ANSWER 5.07 (1.50)

a. DECREASES

b. ~~DECREASES~~ INCREASES

c. DOES NOT CHANGE

REFERENCE
Surry lesson plan ND-86.2-LP-1, p1.4, 1.11; NA NCRODP-86.1
192004; K1.07(2.9)

ANSWER 5.08 (1.00)

a. Lower

b. Higher

REFERENCE
Surry lesson plan ND-83-LP-8, p8.9/10; NA NCRODP-83
191004;K1.04(3.4)

ANSWER 5.09 (1.50)

(acceptable answers, 3 of any 4)

a. fuel burnup

b. fission product poison buildup

c. power defect

d. heat up

REFERENCE
Surry lesson plan ND-86.2-LP-5, p5.5; NA NCRODP-86.1
192002; K1.09(2.7)

ANSWER 5.10 (2.00)

1. reactor power

2. coolant flow rate

3. RCS ~~cold~~ Tavg or TH temperature ~~(Tc)~~

4 RCS pressure.

REFERENCE
Surry lesson plan ND-86.3-LP-2, p2.10; NA TS 2.1
193008; K1.05(3.6)

ANSWER 5.11 (1.50)

Start up rate is positive and constant, reactor power is increasing, and there is no outward rod motion.

REFERENCE
Surry lesson plan ND-86.2-LP-7, p7.51; NA NCRODP-86.2
192008;K1.11(3.8)

ANSWER 5.12 (1.50)

1. Gradual warm up of steam lines

2. Proper venting of tanks and components during warm up and operation.

3. Steam traps

4. Lines kept full

(others as appropriate)

REFERENCE
Surry lesson plan ND-83-LP-8, p8-36; NA NCRODP-83
193006; K1.04/1.10(3.6/3.4)

ANSWER 5.13 (1.00)

The presence of adjacent control rods may cause a significant change in an individual control rod worth.

REFERENCE
Surry lesson plan ND-86.2-LP-6, p6.19; NA NCRODP-86.2
001/000; K5.05(3.9)

ANSWER 5.14 (1.50)

a) > DNB, have partial film boiling, where the fuel rod is alternately covered with steam and water (+.25). Steam has poor thermal conductivity capabilities (+.25), so heat transfer rate drops and Delta T rises (+.25)

b) As fuel surface temperatures rise, stable steam layer forms (+.25) causing a further increase in fuel rod temperatures (+.25). Eventually, significant radiative heat transfer occurs causing heat xfer rate to incarease (+.25)

REFERENCE
Westinghouse Thermal/Hydraulic Principles II, pp 13-18/20

EPE-074; EK1.02(4.6/4.8)

ANSWER 5.15 (2.00)

A. (As the U-235/U-238 isotopes are depleted their fraction of fissions decreases thus) Beff decreases.

B. Production of plutonium isotopes with smaller delayed neutron fractions decreases the average delayed neutron fraction over core life.

C. As the delayed neutron fraction decreases, one is likely to see a quicker response to change in power (i.e. more of a prompt jump/prompt drop)

REFERENCE
Surry lesson plan ND-86.1-LP-7, p7.10; NA NCRODP-86.1
192003; K1.07/K1.08(3.0/2.9)

ANSWER 5.16 (2.00)

A. It doubles (or increases) the head for a given mass flow rate.

B. It will double (or increases) the mass flow rate capacity for a given head.

REFERENCE
Surry lesson plan ND-83-LP-8, Rev 1, p8.18; NA NCRODP-83
191004; K1.09/1.10(2.5/2.4)

ANSWER 5.17 (1.00)

With a DNBR of 1.3, during normal operation and anticipated operational occurrences, there is (a 95%) confidence that DNB does not occur. When > 1.3, the likelyhood of DNB occurring decreases.

REFERENCE
Surry lesson plan ND-86.3-LP-2, p2.10; NA NCRODP-86.1
193008; K1.10(3.1)

ANSWER 5.18 (2.00)

1. Density difference (or DELTA T) created by heat addition by the heat source and heat removal by the heat sink.

2. The heat sink must be elevated physically above the heat source.

REFERENCE
Surry lesson plan ND-86.3-LP-4, p4.5; NA NCRODP-83, ARR-12
193008; K1.21(4.2)

ANSWER 5.19 (1.00)

Neutrons at or near the edge of the core have a higher probability of leaking out than the ones at the center which have a higher probability of causing fission. (Hence: DRW at center is > than at edge).

REFERENCE
Surry lesson plan ND-86.2-LP-6, p6.12; NA NCRODP-86.1
192005; K1.14(3.5)

ANSWER 5.20 (1.00)

(any 2 of the the 3)

1. Neutron production is relatively high, so power is constant when the reactor is critical.

2. Below 10 exp -8 amps the output of the intermediate range may not be directly proportional to the neutron population.

3. Reactivity has not yet been changed by the moderator or fuel temperature.

REFERENCE
Surry lesson plan ND-86.2-LP-7, p7.57; NA NCRODP-86.2
192008; K1.12(3.6)

ANSWER 5.21 (2.50)

a. Maintain DNBR < 1.3 and core exit enthalpy < saturated(+1.0)

b. Prevent bulk boiling during normal operations(+0.5)

c. Ensure fuel clad temperature < 2200 deg F during a LOCA(+1.0)

REFERENCE
Surry lesson plan ND-86.3-LP-3, p3.12; NA NCRODP-86.3
193009;K1.07(3.3)

ANSWER 6.01 (2.00)

(any 4 of 5 at 0.5 ea)

1) Max. Fuel Element Cladding Temp. < 2200 Deg. F
2) Cladding Oxidation < 17% thickness
3) Hydrogen generated by Zirc-Water reaction <1% of max. possible.
4) Core remains in a coolable geometry
5) Provides for long term decay heat removal

REFERENCE
10CFR50.46
NA NCRODP 91.9 ESF p.2.6/2.7
006/050; PWG 4 (4.2/4.3)

ANSWER 6.02 (1.00)

a (1.0)

REFERENCE
NA NCRODP 93.8 PZR Press. Control and Protect.
010/000; A4.01 (3.7/3.5)

ANSWER 6.03 (1.50)

To provide a path to keep the RHR system full (0.50) and to allow for expansion of the system during heat up of the RCS (0.50) and thus ambiently heating up RHR (0.50). or "to aid in warming up the RHR system (.5) or "for overpressure protection anytime" (.5)

REFERENCE
NA NCRODP 88.2 RHR
004/000; K1.01 (3.4/3.9)

ANSWER 6.04 (1.50)

1) Prevents shocking the regenerative heat exchanger and the orifices (0.75)
2) Keeps the regenerative heat exchanger and associated piping pressurized to prevent flashing (0.75)

REFERENCE
NA NCRODP 88.3 CVCS p. 6
004/020; K4.03 (3.0/3.4)

004/020; K6.12 (2.9/3.1)

ANSWER 6.05 (1.00)

d (1.0)

REFERENCE
NA NCROPD-77 RPS p 39
015/000; K4.01 (3.1/3.3)

ANSWER 6.06 (.50)

TRUE (0.5)

REFERENCE
NA NCRODP 93.5 Rod Control
001/000; K4.03 (3.5/3.8)

ANSWER 6.07 ~~(1.50)~~ .75

~~a)~~ (3)	motor generator set	b)	2
(4)	reactor trip breaker		2
(2)	power cabinet		4
(7)	logic cabinet		1
(6)	rod position indication cabinet		4
(5)	automatic rod control unit		1
(1)	DC hold cabinet		1

~~(0.75 for a) fully correct~~, 0.75 for b) fully correct
~~-0.1 for each switch needed to place a component in proper order)~~

REFERENCE
NA NCRODP 93.5 Rod Control System
001/000 K4.01 (3.5/3.8)

ANSWER 6.08 (1.50)

Sufficient water available to maintain the RCS at Hot Standby for 8 hours (1.0) with a steam discharge to the atmosphere (0.25) with a total loss of offsite power (0.25).

REFERENCE
NA Technical Specifications Bases 3/4 7.1.3
026/000; PWG-5 (3.3/4.1)

ANSWER 6.09 (1.00)

a (1.0)

REFERENCE
NA NCRODP 93.11
041/020; K4.17 (3.7/3.9); K4.18 (3.4/3.6)

ANSWER 6.10 1.75 ~~(1.50)~~

a) 3,4,7
b) 5 (0.25 ea)
deleted ~~c) 6~~
d) 1,6
e) 2

REFERENCE
NA NCRODP 88.1 RCS
002/000; K1.09 (4.1/4.1), K1.06 (3.7/4.0)

ANSWER 6.11 (2.00)

a) The C charging pump must be put on the alternate bus. (1.0)

b) A jumper must be installed (to provide a signal that a J bus (C) charging pumps are running). (1.0)

REFERENCE
NA NCRODP 88.3
004/000 K2.02 (3.3/3.5)
004/020 PWG-1 (3.6/4.1)

ANSWER 6.12 (2.00)

1) PZR Hi Press. Trip(0.2) 2385 psig(0.1), 2/3(0.1)
2) PZR Lo Press. Trip(0.2) 1870 psig(0.1), 2/3(0.1)
3) PZR Lo-Lo Press. SI(0.2) 1765 psig and not blocked(0.1), 2/3(0.1)
4) P-11(0.2) <2000 psig(0.1) on 2/3 (0.1)
5) Press. input to the OT Delta T (0.4)

REFERENCE
NA NCRODP 93.8
010/000; K1.01 (3.9/4.1)

ANSWER 6.13 (1.00)

The design change resulted because of experiences where the undervoltage trip signal alone was not sufficient to trip the breaker. (1.0)

REFERENCE
NA NCRODP 77 RPS p.35
012/000; K6.03 (3.1/3.5)

ANSWER 6.14 (1.50)

1) Compensate for power defect or to maintain minimum shutdown margin (0.5 ea)
2) To minimize the amount of positive reactivity inserted during a rod ejection accident, and
3) To minimize radial flux tilt (peaking) or to maintain acceptable power distribution limits.

REFERENCE
NA NCRODP 77 RPS
001/000; K5.04 (4.3/4.7)

ANSWER 6.15 (1.50)

a) STSP decreases (0.5 EA)
b) STSP decreases
c) STSP decreases

REFERENCE
NA NCRODP 77 RPS p 25
012/000; A1.01 (2.9/3.4)

ANSWER 6.16 (1.00)

a (1.0)

REFERENCE
NA NCRODP 93.9 Main Generator Control & Protection
045/010; K1.11 (3.6/3.7)

ANSWER 6.17 1.00 ~~(1.50)~~

a) TRUE (0.5 EA)
b) FALSE
~~c) TRUE~~ deleted

REFERENCE
NA NCRODP 91.1 p.2.18
013/000; K1.11 (3.3/3.8)

ANSWER 6.18 (1.50)

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

REFERENCE
NA NCRODP 93.2 Excore Instrumentation Sys.
015/000; K4.07 (3.7/3.8)

ANSWER 6.19 (1.25)

40% setpoint from 0-20% (0.5) Turbine power (0.25)
and linearly from 40-110% as Turbine Power goes from 20-100%
(0.5)

REFERENCE
NA NCRODP 91.1 "ESF-SI of ECCS"
013/000; K1.01 (4.2/4.4)

ANSWER 6.20 (1.00)

1) Control switch to close (0.25)
2) Synchronizing selector switch is ON (0.25)
3) DG terminal voltage is 95% (0.25)
4) Breaker 86 and 87 protective relay's are reset (0.25)
(86-Breaker overcurrent, 87-phase differential

REFERENCE
NA NCRODP 90.4 EDG p2.57
064/000; A4.01 (4.0/4.3)

ANSWER 6.21 (1.00)

The highest reading upper/lower detector is compared to the average of the upper/lower detectors (0.5). The circuit auto defeats below 50% power on ALL channels (0.5).

REFERENCE
NA NCRODP 93.2 Excore Instrumentation Sys.
015/000; K6.04 (3.1/3.2) & A1.04 (3.5/3.7)

ANSWER 6.22 (1.75)

a) Cabinet has the capacity to support up to 6 stationary gripper coils simultaneously (0.5). So with 2 groups or more, would overload/heat the cabinet(0.5).
b) 125 VDC-Latching Rods
70 VDC-Holding Rods (0.5 for reasons, 0.25 for correctly associating voltages).

ANSWERS -- NORTH ANNA 1&2 -87/02/09-MOORMAN, J

ANSWER ~~7.01~~ ~~deleted~~ ~~(1.00)~~ (1.00)

2 or no correct answer since minimum seal injection flow is 6 gpm

REFERENCE
VCS, SOP-101 p1
NA OP-5.2 p 4
SUR OP-5.2 p 2,4

ANSWER 7.02 (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.
SUR 1-AP-20, p 5

ANSWER 7.03 (1.00)

b

REFERENCE
NA ADM 5.8, pp 2/3
Sur SUADM-ADM-21 p 21
PWG-23: Plant Staffing and Activities (2.8/3.5)

ANSWER 7.04 (1.00)

b

QF=1 for gamma

100(45/60)(1)=75

REFERENCE
10 CFR 20.

PWG-15: Radcon Knowledge (3.4/3.9)

ANSWERS -- NORTH ANNA 1&2 -87/02/09-MOORMAN, J

ANSWER 7.05 (1.00)

b

REFERENCE
VCS, T/S p 3/4 3-2, 3/4 3-6
NA T/S Table 3.3-1
SUR T/S Table 3.7-1

ANSWER 7.06 (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.
SUR FRP-I.3, p 9

ANSWER 7.07 (1.50)

a) No (+.5 ea)
b) No
c) Yes

REFERENCE
NA CSF F-0.4, F-0.5, F-0.6
Sur CSF F-3, F-5, F-6
PWG-10: Recognize abnormal indications for EOPs (4.1/4.5)

ANSWER 7.08 (1.50)

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

REFERENCE
Westinghous. User's Guide for EOPs, pp 5-12

PWG-22(4.3/4.3)

ANSWER 7.09 (2.00)

a) Shutdown (+.5 ea)
b) Startup
c) Shutdown
d) ~~Shutdown~~ Startup

REFERENCE
SQN TS 3.4.6.2
NA TS 3.4.6.2
SUR TS 3.1-13 002/020; PWG-8 (3.5/4.4)

ANSWER 7.10 (.50)

FALSE

REFERENCE
NA 1-OP-22.11, p 4

ANSWER 7.11 (2.50)

a 2
b 4
c 1
d 1
e 6

REFERENCE
Virginai Power GET pp 14-17

PWG-15 3.4/3.9

ANSWER 7.12 (1.00)

-Unexpected rise in S/G level (+.25 ea)
-High radiation on a S/G blowdown line
-High radiation on an MS line monitor
-High radiation as determined by sampling and analysis

REFERENCE
Surry EP-4.00, pp 2
NA 2-EP-3, pp 2

EPE-038; EA2.03 (4.4/4.6)

ANSWER 7.13 (1.50)

1) Verify Charging/SI flow (+.5 ea) (1,2 or 3)
2) RCS Pressure < 1230 psig
3) If component cooling water to any pump is lost

REFERENCE
SQNP Foldout Page
NA Foldout page for 2-EP-0
Surry Foldout page for EP-1.00

003/000; PWG-10 (4.1/4.4)

ANSWER 7.14 ~~(1.00)~~ (1.25)

Surry (+.25 ea)
Verify SI/CHG pumps running/flow
Check RCS pressure <2335 psig
Switch BATP to fast speed
Open MOV-( )350

North Anna (+.25 ea)
1. Verify 2 SI/CHG pumps running/flow
2. Switch BATP to fast speed
3. Open MOV 2350 ~~or Inject the BIT~~
4. Check pzr press <2335
5. Inject the BIT

REFERENCE
Surry FRP-S.1 p 3
NA FRP-S.1 p 4

EPE-029; PWG-11 (4.5/4.7)

ANSWER 7.15 (1.50)

NORTH ANNA (+.3 ea)

- RCS Press > 2000 psig & increasing
- RCS Subcooling > 50 Deg F
- PZR Level > 50%
- SG Level > 10% or > 30% Advrs Cntm
  OR
- AFW Flow > 730 GPM

REFERENCE

NA EP-0 Foldout page

ANSWERS -- NORTH ANNA 1&2 -87/02/09-MOORMAN, J

006/050; PWG-7 (3.8/4.2)

ANSWER 7.16 (1.00)

ioinizing radiation; heat stress; differential pressure; O2 deficiency (+.25 ea)

REFERENCE
Surry SUADMO-19 p 3
NA ADM 20.9, pp 1

PWG-18: Knowledge of Safety Procedures (3.0/3.1)

ANSWER 7.17 (1.00)

Any 4 @ 0.25 points each:

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

Surry
1. Date Critical
2. Time Critical
3. Average RCS temp.
4. RSC Boron concentration
5. Bank C Control rod position
6. Bank D Control rod position
7. Actual critical position within admin. requirements

REFERENCE
NAPS 1.OP-1.5, p.12.
SUR 1-OP-1C App. A p 10 of 10
001.010; K5.08 (2.9/3.3)

ANSWER 7.18 (1.00)

1. Reduce generator load until vacuum stabilizes
2. Check vacuum breaker (MOV-AS-100) closed (.25) and a water seal present (.25)

REFERENCE
NA 1- AP-14, p 3

ANSWER 7.19 (1.00)

Manually trip the reactor (.50) and the affected RCP (.50)

REFERENCE
NA AP-29, p 2

000/015 PWG-10 4.2/4.5

ANSWER 7.20 (1.75)

(.35 ea)
~~.3 ea, .25 for correct order~~

1. RWST to hot legs via LHSI
2. RWST to cold legs via LHSI
3. RWST via HHSI to cold leg
4. RWST via HHSI to hot leg
5. RWST via RP system to reactor cavity

REFERENCE
NA AP-52, p 4,5

034/000 PWG-11 2.8/4.1

ANSWER 7.21 (2.50)

CC surge tank low level alarm
CC pump auto trip alarm
CCW low flow discharge header alarm
CCW low pressure discharge header alarm
Reactor coolant pump low flow/high temp alarm
Excess letdown HX low flow/high temp
Non-regenerative HX high temp

REFERENCE
NA AP-15, p 2

008/030 PWG-10 3.8/4.2

ANSWER 7.22 (.75)

Cause not clearly understood (+.25) or safety related/important equipment operated in an abnormal ar degraded manner (+.5)

REFERENCE
Surry SUADM-0-02 p 3
NA ADM 19.18, pp 1

PWG-10: Recognizing abnormal indications (4.1/4.5)

ANSWER 7.23 (1.00)

An oxygen concentration of <5% by volume must be maintained in the VCT to avoid explosive mixtures in the gas space

REFERENCE
VCS, SOP-102 p 1
SUR OP-8.6 p 2
NA OP-8.6 p 4

004/000 K5.04 2.8/3.2

ANSWER 7.24 (1.00)

Steam pressure mode [0.25]

Tavg input to the steam dump control is not valid without forced flow in the loops. [.75] or Tavg mode cannot be used to cooldown below 547F (no load Tavg setpoint)

REFERENCE

NA 1-AP-10, Att. 2, p 2
SUR AP-39, p 4

ANSWER 8.01 (1.00)

(a)

REFERENCE
NA U1 TS 2.1.2

010/000; PWG-5 (2.9/4.1)

ANSWER 8.02 (1.00)

(c)

REFERENCE
NA U2 TS 3.4.3.2
TPT TS 3.1-1a
Surry TS 3.1-5.6

010/000; A2.03 (4.1/4.2)

ANSWER 8.03 (1.00)

(d)

REFERENCE
NA U2 TS 3.1.3.1

001/050; PWG-5 (2.9/4.3)

ANSWER 8.04 (1.00)

b

REFERENCE
TPT TS 3.9
NA TS 3.3-12
Surry TS table 3.7-5a
073/000; PWG-5(3.0/3.8)

ANSWER 8.05 (1.00)

(b)

REFERENCE
EPIP 3.02 and 3.03.

PWG-36: E-Plan (2.9/4.7)

ANSWER 8.06 (1.00)

d

REFERENCE
NA ADM 14.0, pp 6/7
SU-ADM-O-13

PWG-14(3.6/4.0)

ANSWER 8.07 (1.00)

a) TRUE (+.5 ea)
b) TRUE

REFERENCE
TPT TS B3.0.1, B3.0.5
NA TS B3.0.3/B3.0.5/B4.0.3
PWG-5: TS Knowledge(2.9/3.9)

ANSWER 8.08 (2.50)

a) SR-2 (+.5 ea)
IR-1
PR-1
b) Yes (+.5) TS 3.3.1 action 2.b applies (+.5)

REFERENCE
NA TS 3.3.1

015/020;PWG-5(2.8/3.9)

ANSWER 8.09 (1.50)

1) Use of buddy system is required(two people in constant contact or communication) (+. 5 ea)
2) The entrance is guarded while area is occupied
3) ~~Two personnel~~ must sign for key

REFERENCE
NA HP Manual

PWG-15(3.4/3.9)

ANSWER 8.10 (1.00)

a) 2 SROs (+.25)of which one must be the shift supvsr. or Ops Supt.(+.25)
b) Temp: Procedure Deviation (+.25 ea)
Perm: Request to Change Procedure

REFERENCE
Surry SDM-60, pp 19/21
NA ADM 5.8, pp 4,5

PWG-23(2.8/3.5) Change Procedure

ANSWER 8.11 (1.00)

1) Time of accident (+.33 ea for any 3)
2) Severity of injuries
3) Dose received by victim
4) Is victim neutron irradiated

REFERENCE
NA EPIP 5.01, pp 2/3
Surry EPIP 5.01, pp 3/4

PWG-36(2.9/4.7)

ANSWER 8.12 (1.00)

No action is required

REFERENCE
NA TS 3.9.8.1/3.9.8.2

064/050; PWG-5(3.1/4.1)

ANSWER 8.13 (1.50)

1) Day tank (+.3) level of at least 750 gal (+.2)
2) On-site supply of fuel (+.3) of greater than 45,000 gallons (+.2)
3) Separate operable fuel transfer pump (+.5)

REFERENCE
NA TS 3/4.8.1

064/050; PWG-5(3.1/4.1)

ANSWER 8.14 (1.50)

1) Sequence of events recorder (+.3 ea for any 5)
2) P-250 Alarm Typewriter
3) Strip Charts
4) Logs
5) Completed Procedures
6) Post trip review printout

REFERENCE
SU ADM 0-02, Attachment B
NA ADM-19.18, Attachment B

PWG-28(2.9/3.5)

ANSWER 8.15 (1.00)

1) Loss of source range audible counts (+.25 ea)
2) High flux at shutdown alarm
3) Station evacuation alarm
4) Announcement of containment evacuation

REFERENCE
NA ADM-20.9, pp 9

103/000; A2.04(3.5/3.6)

ANSWER 8.16 (2.00)

1) approximately 30 minute intervals (+.5 ea)
2) Significant change to meteorlogical data
3) " " " " plant status
4) " " " " radiological data

REFERENCE
Surry/NA EPIP Note following State/County Notification Step

PWG-36(2.9/4.7)

ANSWER 8.17 (1.50)

a) Whole Body- 25 Rem (+.25 ea) Thyroid- 125 Rem
b) --volunteers (+.25 ea for any 4)
--professional rescue personnel
--good physical health
--above the age of 45
--should not be a woman capable of reproduction
--familiar with consequences of exposure

REFERENCE
Surry/NA EPIP 5.06, pp 3/4

PWG-36(2.9/4.7)

ANSWER 8.18 (1.00)

Entry into an Operational Mode may be made (+.5) even if the conditions for an LCO are not met (+.5)

REFERENCE
TS 3.0.4

PWG-5: T/S Knowledge(2.9/3.9)

ANSWER 8.19 (1.00)

With Temperature < 500 degrees, the release of activity to the environment due to a SGTR is precluded (+.7) since Psat is < SG PORV lift setpoint(+.3)

REFERENCE
TPT TS B3.1-6
NA TS B 3/4.4.8
Surry TS 3.1-17

002/020; PWG-5(2.9/4.1)

ANSWER 8.20 (1.00)

In an emergency when the action is needed to protect the health and safety of the public (+.75), approved by at least a licensed SRO (+.25).

REFERENCE
10CFR50.54

PWG-36(2.9/4.7)

ANSWER 8.21 (2.00)

a) CHG pump starts and injects into a solid RCS (+1.0 for either) or start of idle RCP with secondary temp within 50 deg of RCS cold leg
b) Gives sufficient time for an operator (approx 10 minutes) to respond in case a malfunction resulting in max charging flow from one Chg pump. (+1.0)

REFERENCE
NA TS Bases B 3/4-4-16

010/000; PWG-5(2.9/4.1)

ANSWER 8.22 (1.50)

a) Radiation worker with an accumulated whole body quarterly dose of 2750 mrem, calender year dose of 4800 mrem (+.5) or a Limited Radiation worker with a similar dose exceeding 1000 mrem. (+.25)
b) Cannot enter a radiation area (+.375 ea)
Need HP approval and instructions to enter a Restricted Control Area

REFERENCE
NA HP Manual 2.3-7/8

PWG-15(3.4/3.9)

ANSWER 8.23 (1.00)

Simple jumpers require a Jumper Log Form (+.5) whereas a more complex jumper operation would also require a controlling procedure (+.5)

REFERENCE
NA ADM-14.1, pp 5
SU-ADM-O-11, pp 14

PWG-14(3.6/4.0)

ANSWER 8.24 (1.00)

Notify: SRO on Call (+.25 ea)
Craft foreman on shift
Forms: Emergency Work Order
Equipment History/Failure Analysis

REFERENCE
NA ADM 16.5, pp 5/6

PWG-23(2.8/3.5)

NORTH ANNA POWER STATION
COMMENTS ON WRITTEN NRC EXAMINATIONS
ADMINISTERED ON FEBRUARY 9, 1987

A. Reactor Operator Examination

1. Question 1.03:

Comment: According to ES-202 Section E, General Guidance p. 3 of 6 "Technical Specification questions for reactor operators should be conceptual in nature (e.g., recognition of limiting conditions for operation and Technical Specifications that exist for a given area)." The 95% confidence level is a detail from the Technical Specifications Bases 2.1.1 Reactor Core that is beyond the required knowledge of a reactor operator.

Recommendation: Do not remove any credit for not mentioning 95% confidence.

Reference: ES-202 Section E, General Guidance p. 3 of 6 and North Anna Technical Specifications 2.1 Safety Limits Bases 2.1.1 Reactor Core p. B2-1. Refer to Attachment 1.

2. Question 1.04 (3):

Comment: Tech Specs Figure 2.1.1 specifies Tavg, not T_c. The negative slope of these curves is based on the actual thermal limit being the most restrictive on T_H. T_c is also acceptable because of the programming relationship between T_c and T_{ave}.

Recommendation: Accept either RCS temperature, T_{ave}, T_H or T_c as a correct answer.

Reference: North Anna Tech Specs 2.1 Figure 2.1.1 Refer to Attachment 2.

3. Question 1.05:

Comment: Our students may also mention that no superheating exists. Superheating would create vapor spaces (i.e. voids) that would interrupt natural circulation flow.

Recommendation: Accept above answer as correct in addition to reasons given on answer key. Suggest 2/3 grading criteria.

Reference: North Anna Lesson Plan: Reactor Energy Removal, NCRODP-86.3, Section IV: Natural Circulation, p. 4.8 and 4.12. Refer to Attachment 3.

4. Question 1.08b:

Comment: FTC magnitude increases with core age due to lower fuel temperatures and buildup of Pu-240.

Recommendation: Change answer key to INCREASE.

Reference: North Anna Lesson Plan: Reactor Operating Principles, NCRODP-86.2, Section I: Fuel Temperature Coefficient and Defect and Attachment 4.

5. Question 1.11:

Comment: Another acceptable answer would be that an inserted rod at the center of the core would cause a greater curvature of the neutron flux, than a rod inserted at the edge (increased buckling). This effect would increase neutron leakage and make the center rod worth more.

Recommendation: Accept above answer in addition to reason given on answer key.

Reference: Westinghouse Nuclear Training Operations: Station Nuclear Engineer's Text: Section 5: PWR Core Physics p. I-5.36 to 5.38. Refer to Attachment 5.

6. Question 1.19:

Comment: In order to solve problem, students needed formula for Right Circular Cylinder lateral surface area. It was not provided. Our students are not required to have this geometric formula memorized, but only to solve problems once given formula.

Recommendation: No credit be removed for incorrectly calculating U-tube surface area.

Reference: North Anna Lesson Plan: Mathematics, NCRODP-79, Section VII: Geometry. Refer to Attachment 6.

7. Question 1.23a:

Comment: Unit A could have a higher source range count. Unit A's source strength could be greater. Unit A's secondary source (Sb-Be) and intrinsic source (γ- ${}^{2}_{1}H$) will be stronger if Unit A has not been shutdown as long as the refueled unit.

Recommendation: Accept either A higher or both the same as correct.

Reference: North Anna Lesson Plan: Reactor Operating Principles, NCRODP-86.2, Section VII: Reactor Startup, p. 7.19. Refer to Attachment 7.

8. Question 1.23b:

Comment: The amount of reactivity added by the control rods will be the same for both units since they are shutdown by the same amount of reactivity. It is true that rod worth is greater at EOL, and as a consequence Unit B at BOL will have a higher critical rod height. However, North Anna lesson plans do not require control room operators to know how rod worth varies with core age without referring to the North Anna Station Curve Books. In other words, the question is beyond the knowledge and skills of a CRO, unless the curve books are supplied. Consequently, CRO's may very well answer the critical rod heights are the same, since the amount of rod reactivity added would be the same. In addition, the question says that "all systems and parameters are identical at the commencement of the startup." From this the student could infer that the rod worths are assumed to be the same.

The overall problem with this question is that sutdents do not know what "parameters" are assumed to be the same and what "parameters" are assumed not to be the same.

Recommendation: Accept either B higher or both units the same as correct.

Reference: North Anna lesson plans: Reactor Operating Principles, NCRODP-86.2, Section VII: Reactor Startup p. 7.3. Refer to Attachment 8.

9. Question 2.06:

Comment: Total point value does not match sum of individual point values. Total is 1.50. Sum is 1.25.

An RO is responsible for which heater banks are powered off the emergency bus and which are off the station service buses. Strict memorization of which MCC implies RO should have all MCC loads memorized which would be a vast undertaking.

Recommendation: Accept identification of which banks are powered off emergency bus MCC's and which banks powered off station service MCC's.

Reference: Candidates have been trained to use load lists for MCC loads. Only the power supplies for major components are to be memorized (i.e, MG sets, major pumps).

10. Question 2.10e:

Comment: Emergency borate valve is an MOV which will be unaffected by loss of IA. Answer states fails as is, agree remains as is but does not fail.

Recommendation: Accept failed as is or recognition valve is an MOV not affected by IA.

Reference: ESK 6EA (attached)

11. Question 2.11a:

Comment: Lesson plan states floating ring seals limit leakage to 50 gpm. Further analysis from W has indicated floating ring seals will limit leakage to < 100 gpm. The report of this analysis came from RCP Seal group in Pittsburgh via phone conversation thus lesson plan has not been updated until official documentation is received.

Recommendation: Since candidates have been exposed to the 100 gpm analysis as well as the 50 gpm, delete question.

Reference: N/A

12. Question 2.15:

Comment: Discharge canal is a correct source of water to fire main system. Water from discharge canal is pumped into a bladder tank which the warehouse 5 fire pumps take suction from.

Recommendation: Accept either the bladder or discharge canal as third source of water.

Reference: Fire Protection lesson plan, NCRODP-92.1

13. Question 2.17:

Comment: Answer key states range of associated parameters, question asked for minimum values.

Recommendation: Delete upper limit from answer key.

Reference: N/A

14. Question 2.23:

Comment: Question asked to place components in proper flow path order, starting with normal power supply ending at CRDM's. Question was written in reference to attached section of the lesson plan. This step, step 4, is strictly the overview stating components of the Rod Control system. Looking at the attached block diagram, step 4 is not stating components in any order by power supply or flow path, but merely in what order the components are to be taught in the lesson plan.

Recommendation: Use attached block diagram of Rod Control system to evaluate candidates knowledge of system. If candidate tried to answer the question without adding drawing/explanation, etc., request the poor quality of the question will be taken into consideration/or delete the question.

Reference: Rod Control lesson plan

15. Question 3.06:

Comment: Question addresses pressurizer control. The answer key addresses pressurizer protection. North Anna has the following:

2 pressurizer pressure control channels
3 pressurizer pressure protection channels
3 pressurizer level/control-protection channels.

Pzr. level control does not use an isolation amplifier per se but a card which serves the same purpose. To be qualified to distinguish between the two, candidate would require Inst. tech knowledge which is not required as per ES 202 B.3. Thus "C" is a correct answer. Also, clarification during exam defining "pressurizer control" as the pzr. Instrumentation as a whole makes "A" a correct answer.

Recommendation: Accept "A" or "C" as correct answers.

Reference: Attached

16. Question 3.12c:

Comment: Question as written with the clarification provided during the exam does not have an answer. A charging pump that is racked out cannot be "locked out" because the 86 relay is deenergized. Question was written in reference to attached lesson plan which refers to 1-CH-P-1C breaker when it states "its" breaker, not 1-CH-P-1A as clarified during the exam.

Recommendation: Delete question.

Reference: ESF Lesson Plan, p. 2.18.

17. Question 3.16:

Comment: Answer key states: Compensate for power defect and minimize radial peaking as 2 of 3 purposes.

Recommendation: Accept as stated in answer key or

1. Acceptable power distribution limits
2. Maintain minimum shutdown margin

Reference: Tech Spec Bases 3/4 1.4 attached. Page B 3/4 1-4.

18. Question 3.18:

Comment: Underfrequency on 4KV buses reactor trip is blocked by P-7.

Recommendation: Include underfrequency as a correct answer.

Reference: Westinghouse logic 5655D33 Sheet 5 attached.

19. Question 3.24a:

Comment: Inputs to main feedwater bypass valve controllers included N-44.

Recommendation: Include N-44 as an acceptable answer.

Reference: SGWLC Lesson Plan attached.

20. Question 4.01:

Comment: Question asked for minimum seal flow. Answer key correct answer is for minimum #1 seal leakoff flow. Term seal flow can imply:

seal injection flow, or #1 seal leakoff flow

Recommendation: Delete question

Reference: OP 5.2 attached

21. Question 4.13:

Comment: Clarification of answer key. Answers 1 and 2 must be present or 3.

Recommendation: Correct answer key to reflect 1 and 2 as trip criteria with both present, not independent of one another.

Reference: Foldout page EP-0 attached.

B. Senior Reactor Operator Exam

1. Question 5.05:

Comment: The [X_e] dip is not independent of the magnitude of the power increase.

Recommendation: Change correct answer to a.

Reference: North Anna lesson plan: Reactor Operating Principles, NCRODP-86.2, Section IV: Fission Product Poisons p. 4.11. Refer to Attachment 9.

✓2. Question 5.07b:

Comment: FTC magnitude increases with core age due to lower fuel temperatures and buildup of Pu-240.

Recommendation: Change answer key to INCREASE.

Reference: North Anna Lesson Plan: Reactor Operating Principles, NCRODP-86.2, Section I: Fuel Temperature Coefficient and Defect and Attachment 4.

✓3. Question 5.10b:

Comment: Tech Specs Figure 2.1.1 specifies Tavg, not T_c. The negative slope of these curves is based on the actual thermal limit being the most restrictive on T_H. T_c is also acceptable because of the programming relationship between T_c and T_{ave}.

Recommendation: Accept either RCS temperature, T_{ave}, T_H or T_c as a correct answer.

Reference: North Anna Tech Specs 2.1 Figure 2.1.1 Refer to Attachment 2.

4. Question 5.14b:

Comment: It is not only radiative heat transfer that causes heat transfer rate to increase. Conduction also increases heat transfer rate. $Q = UA\Delta T$. Since U is approximately constant after the stable film is established, the increased ΔT implies greater conductive heat transfer.

Recommendation: Accept either radiative or conductive heat transfer as reason for increased heat transfer in Region IV.

Reference: North Anna lesson plan NCRODP-83: Thermodynamics, Fluid Flow and Heat Transfer, Section X: Heat Exchangers pp. 10.10 and 10.11. Refer to Attachment 10.

5. Question 5.17:

Comment: According to ES-202 Section E, General Guidance p. 3 of 6 "Technical Specification questions for reactor operators should be conceptual in nature (e.g., recognition of limiting conditions for operation and Technical Specifications that exist for a given area)." The 95% confidence level is a detail from the Technical Specifications Bases 2.1.1 Reactor Core that is beyond the required knowledge of a reactor operator.

Recommendation: Do not remove any credit for not mentioning 95% confidence.

Reference: ES-202 Section E. General Guidance p. 3 of 6 and North Anna Technical Specifications 2.1 Safety Limits Bases 2.1.1 Reactor Core p. B2-1. Refer to Attachment 1.

6. Question 5.18:

Comment: Our students may also mention that no superheating exists. Superheating would create vapor spaces (i.e. voids) that would interrupt natural circulation flow.

Recommendation: Accept above answer as correct in addition to reasons given on answer key. Suggest 2/3 grading criteria.

Reference: North Anna Lesson Plan: Reactor Energy Removal, NCRODP-86.3, Section IV: Natural Circulation, p. 4.8 and 4.12. Refer to Attachment 3.

7. Question 5.19:

Comment: Another acceptable answer would be that an inserted rod at the center of the core would cause a greater curvature of the neutron flux, than a rod inserted at the edge (increased buckling). This effect would increase neutron leakage and make the center rod worth more.

Recommendation: Accept above answer in addition to reason given on answer key.

Reference: Westinghouse Nuclear Training Operations: Station Nuclear Engineer's Text: Section 5: PWR Core Physics p. I-5.36 to 5.38. Refer to Attachment 5.

8. Question 6.07:

Comment: Question asked to place components in proper flow path order, starting with normal power supply ending at CRDM's. Question was written in reference to attached section of the lesson plan. This step, step 4, is strictly the overview stating components in any order by power supply or flow path but merely in what order the components are to be taught in the lesson plan.

Recommendation: Use attached block diagram of Rod Control system to evaluate candidates knowledge of system. If candidate tried to answer the question without adding drawing/explanation, etc., request the poor quality of the question will be taken into consideration/or delete the question.

Reference: Rod Control lesson plan

9. Question 6.10A:

Comment: Excess letdown can be aligned to any of the three intermediate legs.

Recommendation: Change answer key to reflect correct answer.
A. 3, 4, 7

Reference: FM 93A attached

10. Question 6.14:

Comment: Answer key states: Compensate for power defect and minimize radial peaking as 2 of 3 purposes.

Recommendation: Accepted as stated in answer key or:
1. Acceptable power distribution limits
2. Maintain minimum shutdown margin

Reference: Tech Spec Bases 3/4 1.3 attached. Page B 3/4 1-4

11. Question 7.01:

Comment: Question asked for minimum seal flow. Answer key correct answer is for minimum #1 seal leakoff flow. Term seal flow can imply:
seal injection flow or #1 seal leakoff flow

Recommendation: Delete question

Reference: OP-5.2 attached

12. Question 7.09D:

Comment: Refer to RO 4.07 answer key. Answer is "startup" .1 gph inner seal leakage is collected in PDTT which is identified.

Recommendation: Change answer D to startup

Reference: RO 4.07 answer key and attached Annunciator Response

13. Question 7.13:

Comment: Clarification of answer key. Answers 1 and 2 must be present or 3.

Recommendation: Correct answer key to reflect 1 and 2 as trip criteria with both present, not independent of one another.

Reference: Foldout page EP-0 attached to RO 4.13

14. Question 7.14:

Comment: 'Inject the BIT' is a separate means of emergency boration thus a separate step not combined with MOV 2350 (MOV1350 Unit 1) as stated in answer key.

Recommendation: Accept 'Inject the BIT' as a separate step.

Reference: 1-FRP-S.1 attached

15. Question 7.20:

Comment: In order to obtain full credit, examinee must memorize longterm actions. Response as well as RNO's. Step 5.6 states use one or more of following methods, thus does not specify a specific order.

Recommendation: Accept answers in any order.

Reference: AP 5.2 attached

16. Question 8.09:

Comment: Answer key states "two personnel must sign for key". This applies to Surry not North Anna

Recommendation: Accept "must sign for key".

Reference: HP Manual pp. 2.3 - 10, step 2.3.5 attached

References are attached as appropriate.