OCT 31 1984

ENCLOSURE 1

EXAMINATION REPORT NO. 50-413/OL-84-01

Facility Licensee: Duke Power Company
422 South Church Street
Charlotte, N.C. 28242

Facility Name: Catawba Nuclear Station

Facility Docket No. 50-413

Written and oral examinations were administered at the Catawba Nuclear Station near Clover, South Carolina.

Chief Examiner: Thomas Rogers — 10/26/84
Thomas Rogers — Date Signed

Approved by: Bruce A. Wilson — 10/26/84
Bruce A. Wilson, Section Chief — Date Signed

Summary:

Examinations on September 10-14, 1984

Oral examinations were administered to seventeen candidates, sixteen of whom passed. Seventeen candidates were administered written examinations of whom fourteen candidates passed.

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REPORT DETAILS

1. Persons Examined

SRO Candidates:

S. H. Ballenger
G. B. Ice
D. P. Kimball
R. E. Kimray
J. C. Leathers
L. B. Long
P. J. Loss
W. H. Miller
T. G. Poetzsch
G. E. Spurlin

RO Candidates

R. W. Ellingwood
M. T. Lee
T. Ramseur
W. H. Rhyne
L. R. Saunders
R. L. White
J. L. Yon

Other Facility Employees Contacted:

*J. W. Hampton, Plant Manager
*C. W. Graves, Superintendent of Operations
*W. H. Barron, Senior Instructor
*D. Tower, Operations Engineer
*J. Knuti, Operations Engineer

NRC Employees Contacted

*P. Skinner, Senior Resident Inspector of Operations
*K. VanDoorn, Senior Resident Inspector of Construction

2. Examiners:

#*T. Rogers
*A. Johnson
O. Burke
*M. Baldwin
R. Thornton

*Attended exit meeting
#Chief Examiner

3. Examination Review Meeting

At the conclusion of the written examinations, the examiners met with Wendel' Barron, Tommy Kiker, Devereux Tower and Bob Ferguson to review the written examination and answer key. The following comments were made by the facility reviewers:

a. SRO Exam

(1) Question 6.7

Facility Comment:

This question should be deleted because the Rad-Chem Group operates the waste gas system and not the Operations Group.

NRC Resolution:

10 CFR 55.22(i) requires procedures and equipment available for handling and disposal of radioactive materials and effluents be included in the content of a senior operator written examination, therefore, this question will remain as part of the examination as presented during the exam review.

(2) Question 6.12.

Facility Comment:

Choice "C", Nuclear Service Water System, should also be accepted as a correct answer.

NRC Resolution:

The reference material provided by the facility did not include the Reactor Coolant Pump motor cooler as a load on the Nuclear Service Water System in the Nuclear Service Water System or the Reactor Coolant Pump Plant Summary Manual Chapters. However, since the Nuclear Service Water system is indeed a backup cooling water supply, as supported by Piping and Instrument Drawing RN-15742.3, answer (b) or (c) will be credited as a correct answer. The answer key has been changed to reflect this.

(3) Question 6.14

Facility Comment:

There is no correct answer. CPCS set point is ≥ .3 psid as per Table 3.3-4 of Tech Specs.

NRC Resolution:

The reference material provided indicated a setpoint of 0.35 psid which is the only choice that is greater than the 0.3 psid requirement of Tech Specs. Since the reference material does not conflict with the Tech Spec requirement, this question and answer will remain as part of the examination as presented during the exam review.

(4) Question 7.8

Facility Comment:

Answer "C" should be accepted as correct as per the Nuclear Service Water normal operating alignment CN-SYS-RN-1.

NRC Resolution:

Since the question did not specify the plant conditions during the loss of the Nuclear Service Water pump, choice (c) or (d) will be credited as a correct answer. The answer key was changed to reflect this.

(5) Question 7.11

Facility Comment:

"Rod control urgent failure does not exist" should also be accepted as correct.

NRC Resolution:

Since the Rod Control urgent failure annunciator response procedure OP/1/B/6100/10C, page 12 cautions the operators in the immediate actions not to move rods with the alarm in, "Rod Control Urgent Failure does not exist" will be credited as a correct answer. The answer key was changed to reflect this.

(6) Question 7.13

Facility Comment:

EP-01 provides an additional answer that should be considered correct. That is "secondary heat sink available."

NRC Resolution:

"Secondary heat sink available" is acceptable as one of the correct responses in accordance with EP/1/A/5000/01 step 15.d. The answer key was changed to reflect this.

(7) Question 7.17

Facility Comment:

Two cases for containment conditions are in EP-2F3 and the conditions for the other case should be added to the answer key. The other case is containment without hydrogen and the additional conditions are:

(a) Head vent block valves are energized.
(b) VV (containment ventilation) is operating.

Reference EP-2F3, page 6 (Retype 0).

NRC Resolution:

"Head vent block valves are energized" and "containment ventilation is operating" will each be accepted as one of the required four responses in accordance with EP/1/A/5000/2F3, steps 12.b. and 12.c. The answer key was changed to reflect this.

b. RO Exam

(1) Question 1-4.

Facility Comment:

6.8% reactivity total from Xenon should also be considered correct because of the way he curve is laid out.

NRC Resolution:

6.8 + .3% total reactivity worth from Xenon will be the only correct answer based on the Catawba Unit Data Book curves 6.6.1 and 6.6.2. The answer key was changed to reflect this.

(2) Question 2-4

Facility Comment:

Accept containment pressure setpoint of ± 1.2 psig and steam line pressure of ≥725 psig as listed in Table 3.3-4 of Tech Specs.

NRC Resolution:

1.2 psig and 725 psig will be accepted as correct setpoints in accordance with the updated values specified in the Tech Specs. The answer key was changed to reflect this.

(3) Question 3-8

Facility Comment:

Answer key should be NV238, not NV236.

NRC Resolution:

The answer key has been changed as supported by CN-SYS-NV-1 system diagram. However, the valve number is not required for full credit.

(4) Question 3-9

Facility Comment:

Answer for part b should consider C-7A and C-7B as correct answers.

Answer for part c should accept vacuum system and circulating water system.

NRC Resolution:

C-7A and C-7B has been added as acceptable answers for part b as supported by CNS PSM CN-IC-IDE. The vacuum system and circulating water system are two systems that are required to satisfy the C-9 permissive as already stated for the answer to part c. These systems will be credited as correct responses but are not considered changes to the answer key.

(5) Question 3-12

Facility Comment:

The part of the answer on FWP ΔP program should not be required because this is part of the Feedwater Pump Turbine Speed Control System and not the Steam Generator Level Control System.

NRC Resolution:

The FWP ΔP program was deleted as part of the required answer since the facility comment was supported by the Catawba Plant Summary Manual Vol III, section IWE.

(6) Question 4-3

Facility Comment:

EP-01 provides an additional answer that should be considered correct. That is "secondary heat sink available."

NRC Resolution:

"Secondary heat sink available is acceptable as one of the correct responses in accordance with EP/1/A/5000/01 step 15d. The answer key was changed to reflect this.

(7) Question 4-9

Facility Comment:

Question should be deleted because item #1 in the stem of the question is NOT an objective of the Tube rupture (EP-IE).

NRC Resolution:

Although item #1 may not be an objective of the procedure, it is a step within the procedure. Item #1 was used as an additional distractor within the provided choices. It is not a required step of the procedure to have sequentially memorized in order for a candidate to select the correct answer. This question and answer will remain as presented during the exam review.

(8) Question 4.15

Facility Comment:

Answer "c" should also be accepted as correct.

NRC Resolution:

The reference material provided by the facility did not include the Reactor Coolant pump cooler as a load on the Nuclear Service Water System in the Nuclear Service Water System or Reactor Coolant pump plant summary manual chapters. However, since the Nuclear Water System is indeed a backup cooling water supply, as supported by Piping and Instrument Drawing RN-15742.3, answer (b) or (c) will be credited as a correct answer. The answer key has been changed to reflect this.

4. NRC Post Grading Review

Following the review of graded examinations in accordance with NUREG-1021, ES-108, Quality Assurance Program for Review of Graded Examinations, the following changes were made to the answer key with the effected examinations regraded accordingly.

The following changes were made to the Senior Operator Examination answer key.

Question 7.10 - "ND Heat Exchanger" and "NCDT Heat Exchanger" were accepted as correct responses in accordance with the Catawba Plant Summary Manual, Volume 1, CN-SYS-NC paragraph V.2.

Question 8.1 - (a) was accepted as a correct answer because counting penalty time can be interpreted as a required operator action. Counting penalty time is required when outside the target band at power levels greater than 15% per Technical Specification 3.2.1.

Question 8.3 - Partial credit was given as appropriate for the candidates showing the method of calculating the extention time.

Question 8.10 - The question was deleted from the examination based on evidence of confusion by the candidates between the Technical Specification definition of the unrestricted area per figure 5.1-4 and the definition of the unrestricted area per 10 CFR 20.3(17).

The following changes were made to the Reactor Operator Examination answer key.

Question 4-12 - Low charging line temperature and NV1 or NV2 shut was accepted as technically correct responses as supported by the Catawba Plant Summary Manual, Vol. I, Diagram CN-SYS-NV-1.

5. Exit Meeting

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

There were no generic weaknesses (greater than 75 percent of candidates giving incorrect answers to one examination topic) noted during the oral examination. However, improvements in operator knowledge in the areas of procedure performance outside of the main control room and Technical Specification limiting conditions for operation should be made. A request to assure complete and accurate plant reference material was made by the NRC in order to support the NRC written examination development. 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.

(Enclosure 3 1of 2)

U. S. NUCLEAR REGULATORY COMMISSION

REACTOR OPERATOR LICENSE EXAMINATION

Facility: Catawba

Reactor Type: PWR

Date Administered: 09/10/84

Examiner: O. W. Burke

Applicant: Burke's Master

INSTRUCTIONS TO APPLICANT:

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

Category Value	% of Total	Applicant's Score	% of Cat. Value	Category
25.0	25.0			1. Principles of Nuclear Power Plant Operations, Thermodynamics, Heat Transfer and Fluid Flow
25.0	25.0			2. Plant Design Including Safety and Emergency Systems
25.0	25.0			3. Instruments and Controls
25.0	25.0			4. Procedures -- Normal, Abnormal, Emergency & Radiological Control
100.0	100.0			TOTALS

Final Grade ______ %

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

____
Applicant's Signature

1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER, & FLUID FLOW (25.0)

1-1 Following a reactor trip from 100% power, how long would you expect it to take for the source range instrumentation to be energized? Show by calculation and also state any assumptions used. (1.0)

1-2 Assume a reactor is shutdown by 6.5% delta K/K and has a count rate of 250 cps. If control rods worth 4.2% delta K/K are withdrawn, what will the new count rate be? Show all work. (2.0)

1-3 The time required to reach equilibrium xenon concentration after start-up of a xenon free core is approximately _________ hours. (0.5)

1-4 The largest xenon concentration peak occurs after a trip from 100% power. The negative reactivity due to xenon in this worst case is ___% δK/K. (0.5)

1-5 A reactor whose reactivity change information is given in figures 1.1 through 1.3, attains criticality at 100 steps on rod group D (using a 100 step overlap). Assume that the boron concentration is held constant at 900 ppm and that the reactor is initially at BOL and HZP, with all rods bottomed. Use $\beta = 0.007$ and $\lambda = 0.08$.

a. What was the actual shutdown reactivity with all rods bottomed? (1.0)

b. What was the initial k effective (with all rods bottomed)? Show all work. (0.5)

c. What would be the D group position required to obtain a stable 0.8 DPM startup rate (starting from a stable, just critical position of 100 steps on D bank)? Show all work. (1.0)

d. Assuming that this reactor is similar to your reactor, how would the control rod positions at 10^{-8} amps compare with the rod positions at 10^{-10} amps (assume steady state conditions at both power levels)? Explain. (0.5)

e. If the reactor power is increased to 30% of full power using rod motion only (constant boron concentration) and disregarding any long term poison effects, what will be the group D position at 30% steady power? (1.0)

1-6 In order to prevent overheating the fuel, the reactor is operated such that the point of DNB is not reached (DNBR > 1.3). What four (4) primary system parameters are monitored to assure that DNBR is > 1.3? (1.0)

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U.S. NUCLEAR REGULATORY COMMISSION

INTEGRAL CONTROL BANK WORTH VERSUS STEPS WITHDRAWN WITHOUT OVERLAP (BOL AND HZP)

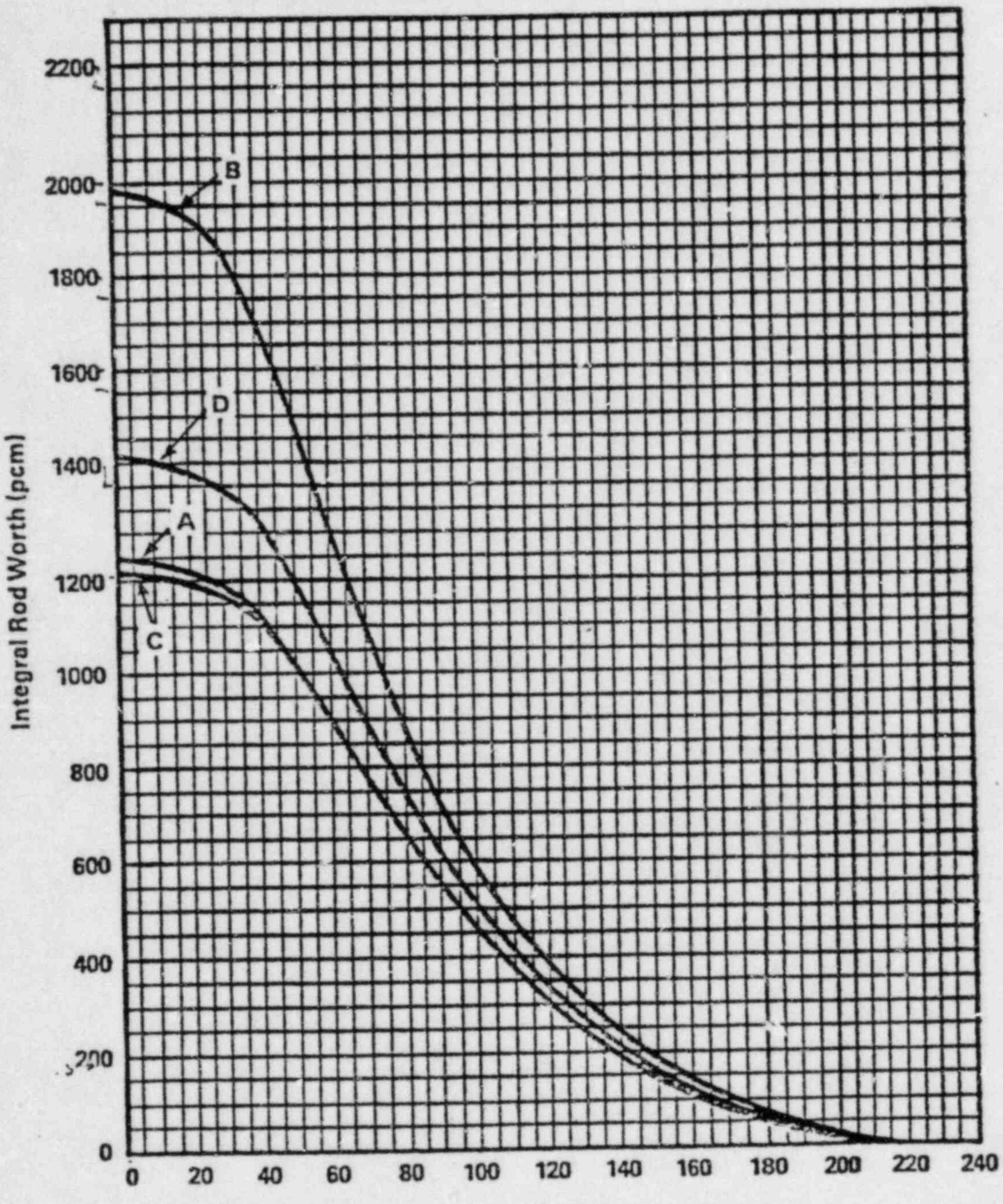


FIGURE I.1

U.S. NUCLEAR REGULATORY COMMISSION

INTEGRAL SHUTDOWN BANK WORTH VERSUS STEPS WITHDRAWN (BOL AND HZP)

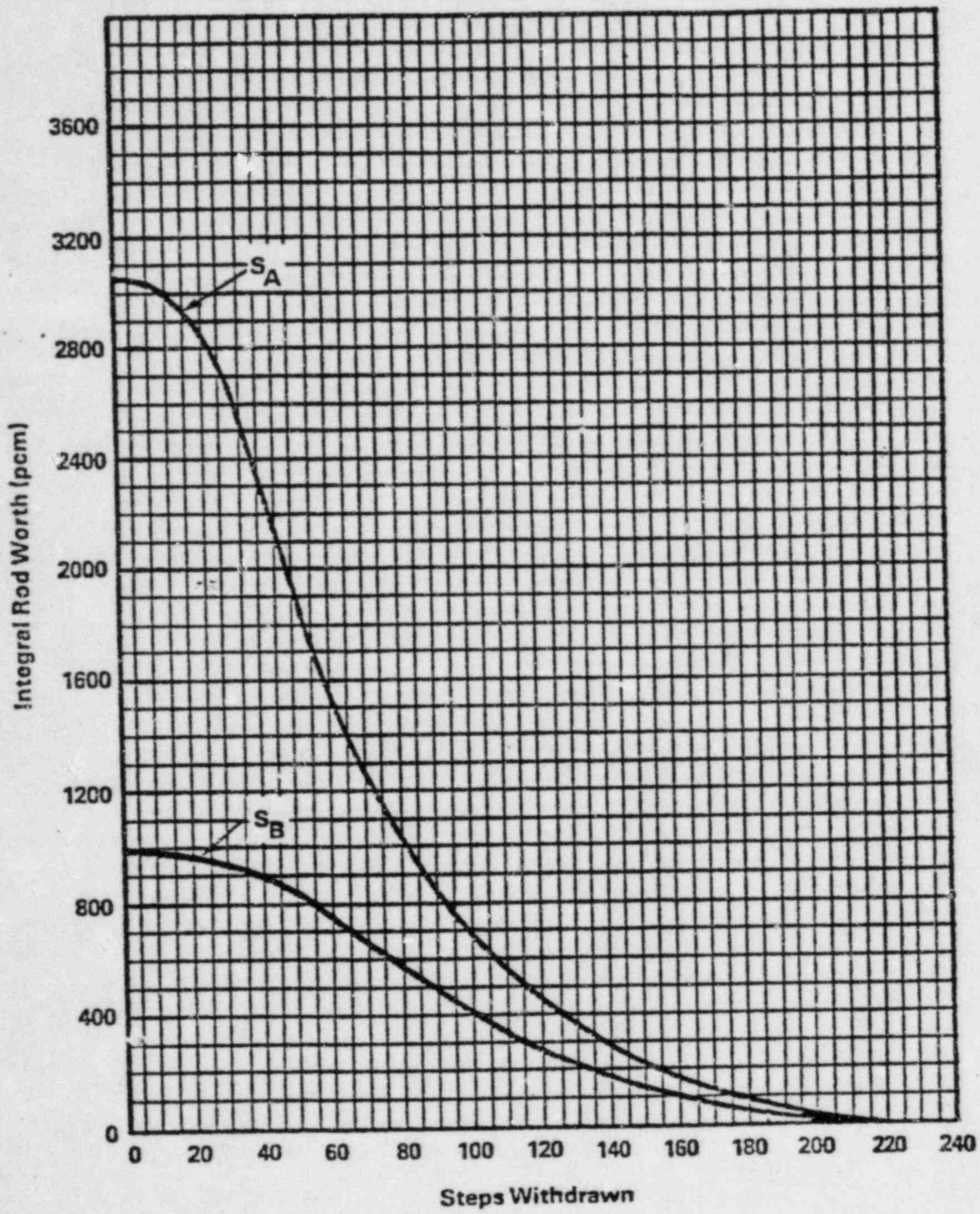
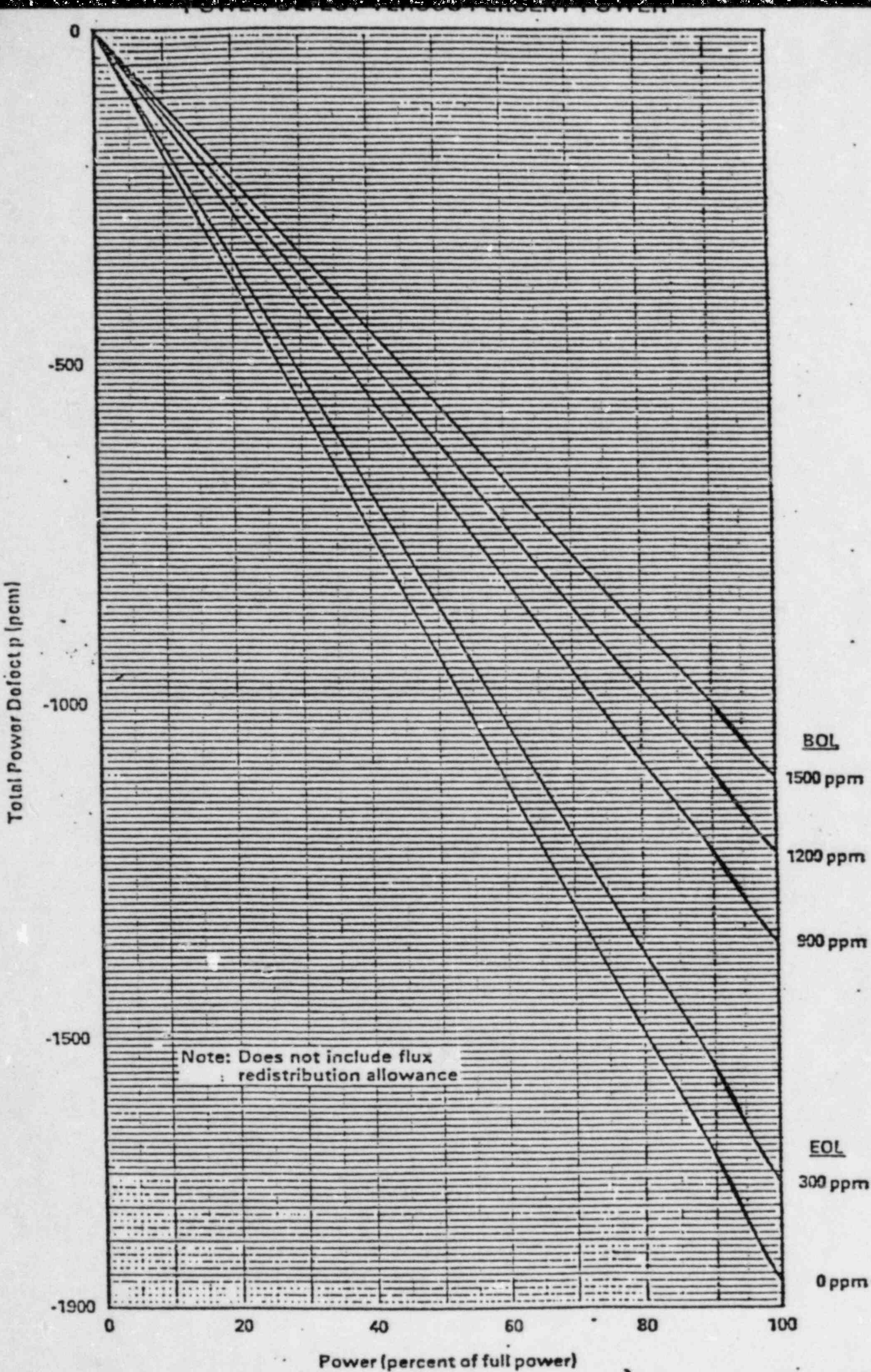


FIGURE I. 2



0
-500
-1000
-1500
-1900
Total Power Defect ρ (pcm)
0
20
40
60
80
100
Power (percent of full power)
BOL
1500 ppm
1200 ppm
900 ppm
EOL
300 ppm
0 ppm
Note: Does not include flux redistribution allowance

FIGURE 1.3

1-7 Assume that you are starting up your reactor and the reactor is critical at a steady power level of 10^{-8} amps on the intermediate range NI's. If you should step change the reactivity by +10 pcm and not have any subsequent rod motion or boron concentration changes, which of the following statements best describes the subsequent reactor behavior? (1.0)

a. K_{eff} will increase, neutron population and core power will continue to increase indefinitely until the operator inserts -10pcm by inserting rods or boron.

b. K_{eff} will increase, neutron population and core power will increase, which will add negative reactivity due to temperature changes, cancelling out the +10 pcm. The core power will level out at the power level having a power defect of 10 pcm.

c. K_{eff} will increase, neutron population and core power will increase, which will add negative reactivity due to temperature changes cancelling out the +10pcm. The core power will level out by itself at some power level above POAH.

d. K_{eff} will increase, neutron population and core power will increase, which will add negative reactivity due to temperature changes cancelling out the +10pcm. The core power will return to the original power level.

1-8 Fissionable PU^{239} in appreciable quantities is formed in the reactor by the fissioning of U^{238}. TRUE or FALSE. (0.5)

1-9 Most of the Xenon in a reactor is produced from the radioactive decay of fission products. TRUE or FALSE. (0.5)

1-10 A control rod is worth more in reactivity when it is adjacent to a withdrawn control rod as compared to its worth when adjacent to an inserted rod. TRUE or FALSE. (0.5)

1-11 The "secondary source" in a reactor refers to the neutrons released as a result of radioactive decay of fission products. TRUE or FALSE. (0.5)

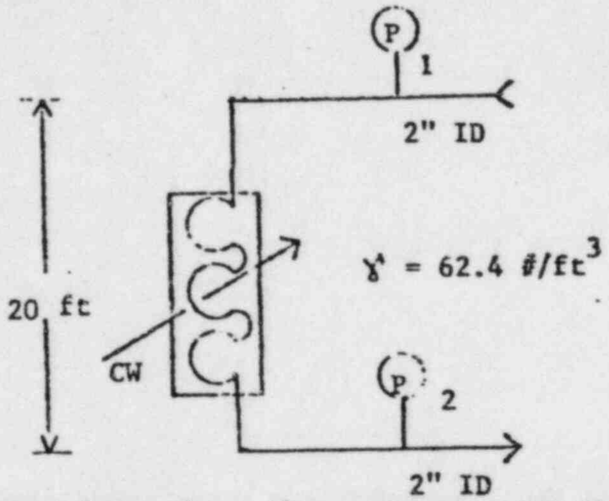
(continued on next page)

1-12 The major source of the heat produced in the core of a reactor 5 minutes after it has been shut down is: (1.0)

a. Subcritical multiplication
b. Spontaneous fission
c. Fission product decay
d. Fission by delayed neutrons

1-13 List three (3) bases for the control rod insertion limits. (1.5)

1-14 (2.0)



In the diagram shown above the pressure gauge at point one reads 45 psi and the pressure gauge at point two reads 36 psi. The total head loss due to friction in the pipes between points 1 and 2 (excluding the piping inside the Hx) is 5 ft. According to the design specifications the heat exchanger is fouled and should be cleaned when a pressure drop of greater than 9 psi exists across the heat exchanger. Is the heat exchanger fouled? Show calculations to substantiate your answer.

1-15 a. What is cavitation? (0.5)

b. What control room indications would alert the operator to cavitation of an ND pump? List at least three (3). (1.5)

1-16 Using a plant performance and heat-balance diagram for a simplified PWR, the following data were obtained:

1. Feedwater temperature = 450°F
2. Feedwater flow = steam flow = 14.925×10^6 lbs. mass/hr.
3. Steam pressure to turbine = 785 psig
4. Enthalpy of steam leaving turbine = 1007 BTU/lbm. mass
2.55×10^3 BTU/hr = 1 hp 3.41×10^3 BTU/hr = 1kw

Calculate the thermal power of the reactor, in MW. (2.0)

(continued on next page)

1-17 One way to increase a pump's net positive suction head (NPSH) is to raise the level in its suction tank. True or False (0.5)

1-18 a. Find the enthalpy change in an isentropic expansion of steam through a turbine into a condenser (NOTE: Pstm = 825 psia, saturated, Pcond = 2 psia). (1.0)

b. How would the enthalpy change in part a be affected by a less than ideal turbine (i.e., some degree of inefficiency)? Select one of the following answers. (0.5)
1. Higher
2. No change
3. Lower

1-19 In the NC System when forced convection cooling is lost, Natural Circulation will be established "naturally" if all of the systems respond correctly. You as an operator must be able to verify that this occurs by looking at your plant instrumentation. Give five (5) indications that natural circulation has been established. (2.5)

(End of Section 1)

2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS ((25.0)

2-1 a. Suppose a high activity alarm occurs in a Gaseous Radiation Monitor. If you reset the alarm and it comes back immediately, would you consider the alarm valid or invalid? Explain. (0.75)

b. If you PURGE the gaseous monitor after it alarms and the alarm does not clear, would you consider the alarm valid or invalid? Explain. (0.75)

2-2 Which of the following will cause the OPΔT setpoint calculator to reduce its setpoint. (1.0)

a. T(ave) above rated
b. Rate of change of T(ave) as T(ave) is decreasing
c. Pressure below 2235 psig
d. Delta flux exceeding the target band.

2-3 The input into the rod insertion limit calculator used as an indication of reactor <u>power</u> is: (1:0)

a. P(imp)
b. Auct. High T(ave)
c. Total steam flow
d. Auct. high ΔT

2-4 List the signals, setpoints, and coincidences for safety injection actuation (four required). (1.0)

2-5 Assume that you, as a control operator, need to startup and parallel a Diesel Generator. Describe the system response when shutting the D/G breaker for each of the following conditions:

a. Voltages are equal when you attempt to close the breaker with the sync needle at 12 o'clock and rotating rapidly in the fast direction. (1.0)

b. Voltages are equal when you attempt to close the breaker with the sync needle at 12 o'clock and rotating slowly in the fast direction. (1.0)

c. The D/G voltage is lower than the grid voltage when you attempt to close the breaker with the sync needle at 12 o'clock and rotating slowly in the slow direction. (1.0)

2-6 What occurs in the NV System on an emergency low level in the VCT? (List valve actuations, alarms, etc.). (1.5)

(continued next page)

2-7 a. For a LOCA resulting in a slow, steady depressurization, list the ECCS components that will inject water into the NCS. Include the operating pressure at which each component will begin to inject water. (2.0)

b. For a large LOCA coincident with a loss of all 6.9 KV buses, will there be any change in the order of injection as compared to that in part a. Explain. (1.0)

2-8 List two (2) indications in the Control Room that will alert the operator that a Pressurizer PORV did not completely reseat after opening. (1.0)

2-9 After drawing a steam bubble in the pressurizer, you find the Pressurizer Relief Tank (PRT) temperature to be excessively high. List two (2) ways available to the operator for cooling the PRT. (1.0)

2-10 Assume a loss of station power and a concurrent diesel failure occurs resulting in a loss of all station power except for batteries. Briefly describe three features of your facility that will provide reactor cooling. (1.5)

2-11 With respect to the electrical distribution system, explain how power is supplied to essential equipment under the following conditions: (3.0)

a. A safety injection signal occurs, then 30 seconds later a blackout occurs.

b. A blackout occurs, then 15 seconds later a safety injection occurs.

c. A safety injection occurs, then 30 minutes later the SI signal is reset in anticipation of entering the recirculation mode. One minute later, a blackout occurs.

2-12 What are the purposes of the NCP flywheel and the anti-rotation device? (1.0)

2-13 a. How does the operator identify the difference between an NC System leak in containment and a steam generator tube leak? (0.5)

b. Give three (3) indications of each event. (3.0)

(continued next page)

2-14 List the five (5) reactor trips associated with the Nuclear Instrumentation System. Include setpoints and interlocks provided if any. (2.0)

(End of Section 2)

3. INSTRUMENTS AND CONTROL (25.0)

3-1 List five (5) means provided to detect a coolant leak in the primary system in containment. (2.0)

3-2 Let us suppose that the (T_{ave}-T_{ref}) error signal input to the control rod drive controller has become disconnected so that the only input to the controller is the power mismatch signal. Turbine power is 100% and nuclear power is 95% and they have been at these values for 5 minutes. At the present time, the control rods will be (controller in "auto") (1.0)

a. inserting
b. withdrawing
c. stationary

3-3 What position (open, closed, or throttled) will the main feedwater flow control valve assume under the following conditions? (1.5)

a. Rx Trip with Tavg of 560°F
b. Increase in Load from 50% to 60%
c. Safety Injection
d. S/G Level of 82%
e. Loss of air
f. Loss of Instrument Electrical Power

3-4 If you were operating your plant at 80% full reactor power when a narrow range control channel $T_{(h)}$ RTD fails high, how would the following parameters be affected? (Answer with 1. increase, 2. decrease, or 3. remain the same. Assume that all control systems are in automatic and that there is no operator action.) (2.0)

a. steam dump valve position
b. charging flow
c. rod insertion limit setpoint
d. control rod bank position

3-5 List the four (4) measured or sensed system parameters used to generate the control rod drive controller demand signal. (1.0)

(continued next page)

3-6 During the course of normal 100% power operation, a loss of T(ref) signal to the steam dump controller will cause the steam dumps to open. TRUE or FALSE. (0.5)

3-7 If an RTD in the loop bypass manifold that is utilized for calculating Tave shorted, how could you determine from the main control board whether it was a Tc or Th RTD? (1.0)

3-8 What must the operator do to initiate "Emergency Boration" of the NCS? (1.0)

3-9 Pertaining to the steam dump system during normal, 100% power operation:

a. What would you as an operator check in the control room to determine the readiness of the steam dump system to operate in case of a turbine trip or load rejection? List 3. (1.0)

b. What indicators would you check to determine whether any of the steam dumps are open? List 2. (1.0)

c. List at least two (2) plant service systems whose failure would prevent the opening of any of the steam dump valves. (1.0)

Condenser

3-10 Will the power range nuclear instrumentation channels indicate power that is higher than or lower than actual reactor power if they are calibrated from results of a calorimetric which used a feedwater temperature that was lower than the actual feedwater temperature. (0.5)

3-11 Match the correct control rod position indicating instrumentation with the functions listed below: (2.0)

1. Individual Rod Position Indication Circuitry

2. Group Demand Circuitry

a. Measures actual rod position
b. Provides an input to the step counters on main control board
c. Provides an input to the rod insertion limit alarm circuit
d. Actuates rod bottom lights

3-12 a. List the secondary system parameters that are used for automatic control of the steam generator level. (1.0)

b. Describe the S/G level control system response to an increase in load. (1.0)

(continued next page)

3-13 During a normal reactor shutdown, you observe the current decay on an intermediate range channel. For the following condition, provide the reason and any operator actions required. (1.0)

The level decays to $2x10^{-10}$ amps and then remains constant.

3-14 In the source and intermediate ranges of the ex-core nuclear instrumentation system, the gamma has to be removed from the neutron signal.

a. How are neutrons detected in each range? (0.5)

b. How is the gamma signal removed in each range? (1.0)

3-15 Your plant is operating with the Pressurizer Pressure Control Select switch in the 1-2 position when Channel 1 fails low.

a. List four (4) automatic actions that will occur as a result of the above failure. (2.0)

b. What action should you, as the control room operator, take? (0.5)

3-16 For the below listed radiation monitors, provide any automatic actions that occur if a high alarm is received.

a. Component cooling water monitor (0.5)
b. Condenser air ejector gas monitor (0.5)
c. Control room ventilation monitor (0.5)
d. Steam generator sample monitor (1.0)

(End of Section 3)

4-1 Explain the reason for each of the following precautions:

a. 1. Control rod banks should be withdrawn and inserted in the prescribed sequence. (0.5)

2. Overlap of control banks may not exceed the prescribed setpoint. (0.5)

b. If the source range count rate increases by a factor of two (2) or more during a change in boron concentration, the operation must be suspended. (0.5)

4-2 During your reactor startup an acceptable reactivity band around the ECP is established. If you should go critical below the lower reactivity limit with rods above the lo-lo insertion limit, you must: (1.0)

a. Emergency borate, recalculate the ECP, and restart after notifying the Reactor Group Unit Engineer.

b. Re-insert rods to at least the Lo-Lo insertion limit, evaluate the problem, restart after notifying the Reactor Group Unit Engineer.

c. Continue normal operation and notify the Reactor Group Unit Engineer (within 16 hours).

d. Borate a sufficient amount to attain criticality at the ECP and continue operation after notifying the Reactor Group Unit Engineer.

4-3 Give the SI termination criteria for a spurious SI actuation. (3 required). (1.5)

4-4 You are the control room operator and you receive a valid low low level S/G reactor trip signal. Looking at your rod position indicators and lights, you see that the rods have not tripped into the core.

a. What would be your first immediate action and why? (1.0)

b. What would you do if the action in (a) did not work? (Give 4 possibilities). (2.0)

(continued on next page)

4-5 During unit operation while increasing power from 70% power to 100% power at a rate of 3% per hour, the "Rod Control Urgent Failure" annunciator alarms and No Auto rod motion occurs. What three (3) immediate actions are required of the operator? (1.5)

4-6 List four (4) conditions which require the operator to trip a NCP. Use specific values. (2.0)

4-7 When can you, as a licensed operator, deviate from the sequence of steps on an operating procedure? (1.0)

4-8 List the five (5) immediate operator actions required in response to a loss of all AC power. (2.5)

4-9 The steam generator tube rupture procedure has a number of objectives, including the following: (1.0)

1. Check for main steam isolation and depress "CLOSE RESET" if MSIV"s are closed.

2. Identify ruptured S/G(s)

3. Check ruptured S/G(s) level

4. Initiate NC system cooldown

5. Isolate steam flow from ruptured S/G(s)

6. Depressurize NC system to minimize leakage

Select the letter corresponding to the proper order for initiating these actions in accordance with EP/1/A/5000/1E.

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

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

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

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

4-10 List the five (5) immediate actions required for Reactor trip or safety injection (EP/1/A/5000/01). (2.5)

(continued on next page)

~~2.8~~ 4-15 If RN pump 1A trips and cannot be restarted for an indefinite period of time during mode 1 operation, the appropriate action would be to: (1.0)

(a) trip the reactor.

(b) commence unit shutdown.

(c) start or verify running pump 1B and verify it is providing adequate flow.

either

(d) start or verify running RN pump 2A and verify it is providing adequate flow.

Ref: CNS AP/1/A/5500/20, p2.

~~8.16~~ 4-16 Safety tag stubs are not removed until the: (1.0)

(a) work supervisor signs for issue approval.

(b) shift supervisor signs for issue approval.

(c) safety tag is affixed to the required operating device.

(d) safety tag is removed from the operating device for clearance.

Ref; CNS Station Directive 3.1.1, P 6.11.1.

[redacted]

4-11 What two (2) individuals, by title, may fill the role of Emergency Co-ordinator? (1.0)

4-12 In accordance with AP/1/A/5500/12, "Loss of Charging or Letdown", list the symptoms of a loss of letdown (five [5] required). (1.5)

4-13 Give four (4) requirements necessary for entry into the incore instrument area (under the reactor vessel). (1.0)

4-14 Supply the missing numbers in the blanks (numbered 1 through 8) pertaining to radiation exposure. (2.0)

Conditions of Exposure	Basic Permissible Dose Limits	
1) Whole body-(gamma)	NRC:	(1) rems/quarter (2) rems/year
	DPC Admin:	(3) rems/quarter (4) rems/year
2) Skin of whole body (beta & gamma & neutron)	NRC: DPC Admin:	(5) rems/quarter (6 rems/quarter
3) Extremities (hands, forearms, feet, ankles)	NRC: DPC Admin:	(7) rems/quarter (8) rems/quarter per extremity

~~4-15 List eight (8) conditions that require the operator to trip a RCP. Use specific values where applicable. (2.0)~~

Substituted questions 4-15 and 4-16.

(End of Section 4)

$1 Ci = 3.7 \times 10^{10} d/s$

$\alpha_D = -1 \times 10^{-5} \frac{\Delta K}{K}/°F$

$\alpha_V = -1 \times 10^{-3} \frac{\Delta K}{K}/\%$ voids

$\alpha_M = -1.0 \times 10^{-4} \frac{\Delta K}{K}/\%°F$

$\alpha_P = -4.5 \times 10^{-4} \frac{\Delta K}{K}/\%$ power

$I(t) = Io\ e^{-\lambda t}$

$T1/2 = \ln(2)/\lambda$

$Cp = (CP_{base})(Ks)(K_A)$

$\frac{\rho_1}{\gamma} + Zi + \frac{V_1^2}{2g} + ha - hr - h_L = \frac{P^2}{\delta} + Z_2 + \frac{V_2^2}{2g}$

$\frac{V_2}{V^1} = \frac{(\Delta P_2)}{(\Delta p_1)}^{1/2}$

$\dot{Q} = MCp\ \Delta t$

$\Delta\rho = f \frac{L}{D} \frac{\rho V^2}{2g_c}$

$f = 64/Re$

$\rho = \frac{k(eff) - 1}{K(eff)}$

$\frac{1}{M} = \frac{CR1}{CR2} = \frac{1-K(eff)2}{1-K(eff)1}$

$\dot{Q} = M\Delta h$

$\dot{Q} = UA\Delta T$

$\lambda = 0.1$

$h_L = km\dot{V}^2$

$P = P_o\ 10^{SUR(t)}$

$SUR = \frac{26.06}{\tau}$

$P = P_o\ e^{t}/\tau$

$A = \lambda N$

$A(x) = A_o e^{-\mu x}$

$M = 1/(1-k) = \frac{CR_1}{CR_0}$

$N(t) = No\ e^{-\lambda T}$

$\alpha_r \propto (L_f + L_s) \frac{(\phi rod)^2}{(\phi avg)}$

$n = \nu/(1 + d)$

$P = \Sigma\ \phi\ \nu/(3.7 \times 10^{10})$

$\tau = (\beta - \rho)/\lambda\rho$

$\tau = \bar{\ell}/\rho + (\beta - \rho)/\lambda\rho$

$\tau = \ell/(\rho - \beta)$

$v = v_f + xv_{fg}$

$H = xh_g + (1-x)\ h_f$

$S = xS_g + (1-x)\ S_f$

1 in = 2.54 cm

1 gal. = 3.785 liters

1 kg = 2.205 lb

$N = \rho A_0/A$

17.58 watts = 1 BTU/min

1psi = 6.895 Pa

1psi = 2.036 " Hg (@ 0C)

1psi = 27.68 " H_2O (@ 4C)

$\bar{\beta} = .0071$

$\bar{\ell} = 2 \times 10^{-5}$ sec

$\ell d = 12.5$

$RR = \Sigma_f \phi_{th}$

$SCR = \frac{S}{1-Keff}$

$\rho = \frac{\beta}{\lambda\tau + 1}$

Reactor thermal power = $(h_2 - h_1)$ x steam flow rate

$K = \frac{1}{1 - \rho}$

Burke - Master

1-1 $P(d) = P_o e^{-t/T}$

where $T = \frac{56 \text{ sec}}{.693}$ approximately 80 sec. (0.5)

Assume that after prompt drop, $P_o = 10^{-6}$ amps and that P-6 is energized at 10^{-10} amps.

$10^{-10} = 10^{-6} e^{-t/T}$

$80 \ln \frac{10^{-10}}{10^{-6}} = -t$

737 sec = t

12.3 min. = t (0.5)

REF.: Questions and answers supplied by Catawba, Theory Section, page 57. Also, Catawba notes on Reactor Kinetics, page 16.

1-2

$-6.5\% = \rho_1$

$+4.2\%$
$-2.3\% \; \Delta k/k = \rho_2$

$C_1 (1-K_1) = C_2 (1-K_2)$ $\qquad K_1 = \frac{1}{1-\rho_1}$

$250 (1-.939) = C_2 (1-.977)$ $\qquad K_1 = \frac{1}{1-(-.065)} = 0.939$ (0.5)

$\frac{250 (.061)}{.023} = C_2$ $\qquad K_2 = \frac{1}{1-\rho_2}$

= 663 cps. (1.0) $\qquad K_2 = \frac{1}{1-(-.023)} = 0.977$ (0.5)

REF.: Questions & Answers supplied by Catawba, Theory Section, page 29 Also, Catawba no .s on subcritical multiplication, page 20.

(continued on next page)

1-3 40 (+5)

REF.: Catawba notes on Reactor Poisons, page 17, item 6. (0.5)

1-4 ~~3.9~~ 6.8 (cont. above 100% eq. value) + 0.3

REF.: Catawba unit 1 data book, curves ~~6.2~~ 6.6.1 and 6.6.2. (0.5)

1-5 a. Get total rod worths when all bottomed using figures I.1 and I.2: (0.5)

Rod Bank	integral rod worth on bottom, pcm
A	1250
B	2030
C	1215
D	1465
S_A	3050
S_B	1000
	10,010 pcm - total rod worth

From figure I.1, we see that the total rod worth is appx. 500 pcm (all rods except D bank fully withdrawn.) (0.5)
SDM - 10,010 - 500 = 9510 pcm or 9.51% $\delta k/k$

REF.: Catawba Theory Notes, Reactivity Control Section

b. (0.5)

$$P = \frac{K_{eff} - 1}{K_{eff}}$$

$$-.0951 = \frac{K_{eff}-1}{K_{eff}}$$

$$K_{eff} = 0.9132$$

NOTE: Full credit will be given if the correct method is used and an incorrect number from part A is used.

REF.: Catawba Theory Notes, Reactor Kinetics Section

(continued on next page)

c. $SUR = \frac{26}{\tau}$

$0.8 = \frac{26}{\tau}$; $.8\tau = 26$, $\tau = 32.5$ sec. (0.25)

$\tau = \frac{\beta-\rho}{\lambda\rho} = 32.5$

$\frac{0.007 - \rho}{.08\rho} = 32.5$

$\rho = .00194 = 194$ pcm (0.25)

Using figure I.1, this would put Group D at appx. 124 steps. (0.5)

REF.: Catawba Theory Notes, Reactor Kinetics Section

d. The rod positions should be essentially the <u>same</u> (below sensible heat range). (0.5)

e. Using figure 1.3 at 900 ppm boron conc., the power defect at 30% power is appx. 410 pcm. We went critical with D at 100 steps (total rod worth appx. 500 pcm). (0.25)
500 - 410 = 90 pcm total rod worth at 30% power. (0.25)
From figure 1.1, group D would be at appx. 165 steps. (0.5)

REF.: Catawba Theory Notes, Reactivity Control Section

1-6 Nuclear power, NCS temperature (Tave), NC loop flow, and NCS pressure (0.25/ea

(Acceptable Answer:)

1) Rods ± 12 steps, 2) AFD in target band, 3) rods proper seq. & overlap between banks 4) proper rod insertion limits

REF.: Catawba Question bank on theory, page 2

1-7 C d is also acceptable due to some "loose" wording. Ok (1.0)

1-8 F REF.: Catawba notes on the Fission Process, page 23 (0.5)

1-9 T REF.: Catawba notes on Reactor Poisons, page 14 (0.5)

1-10 T REF.: Catawba notes on Reactor Poisons, page 24 (0.5)

1-11 F REF.: Catawba notes on Subcritical Multiplication, page 15 (0.5)

1-12 C (0.5)

(continued on next page)

1-13 1. Maintaining an adequate shutdown margin throughout cycle life, while allowing as much manuvering band as possible. (0.5)

2. Maintaining $F^N_{\Delta H}$ below design specs. (0.5)

3. Rod ejection accident consequences acceptable with rods inserted to their limits. (0.5)

REF.: Catawba theory notes on <u>Reactivity Control</u>, page 21, item 2.10D Also, Catawba, Selected Lesson Plans, Vol. III, Section 5, IRE, page 36, item H.

1-14 General Energy Equation:

$$\frac{\rho_1}{\gamma} + Zi + \frac{V_1^2}{2g} + ha - hr - h_L = \frac{P^2}{\delta} + Z_2 + \frac{V_2^2}{2g}$$ (0.5)

V1 = V2 since diameters are equal and no pump. (0.25)

ha ε hr = 0 no pumps or other mechanical devices (0.25)

$hL = h_L\ (Hx) + h_L(F)$

Gen. energy eqn. becomes:

$$\frac{P_1 - P_2}{\gamma} + Z_1 - Z_2 - h_{L(F)} - h_{L(HX)} = 0$$

$$h_{L(Hx)} = \frac{45 - 36}{62.4} \times 144 + 20 - 5$$

$h_{L(HX)}$ = 20.77 ft. + 15 ft. = 35.77 ft. (0.5)

$h_{L(HX)}$ = 35.77 x .433 = 15.5 psi

The heat exchanger is fouled. (0.5)

REF.: Catawba notes on <u>Fluid Flow</u>, page 17, sample problem #2.

1-15 a. Cavitation is the formation of vapor bubbles in the low pressure region of a pump or piping system followed by their collapse in a higher pressure region of the pump or system. (0.5)

b. Erratic running current, flow reduction, Low Discharge Pressure. (0.5/e
flow oscillations

REF.: Catawba Question bank on theory, page 12

(continued on next page)

1-16 Assume that the feedwater is saturated liquid and that the S/G steam has a quality of 100%.
From steam tables: h_1 @ 450°F = 430.2 BTU/# (0.5)
h_2 @ 785 psig = 1199.4 BTU/# (0.5)

Reactor thermal power = (h_2-h_1) x steam flow rate = (1199.4-430.2) BTU/# x 14.925 x 10^6#/hr = 1.148 x 10^{10} BTU/hr. = (1.0)

$$\frac{1.148 \times 10^{10} \text{BTU/hr}}{3.41 \times 10^3 \text{ BTU/hrMW} \times 10^3 \text{KW/MW}} = \underline{3.37 \times 10^3 \text{ MW}}$$

REF.: Catawba Theory Notes, Thermodynamics Section, Item 8

1-17 T (0.5)

REF.: Catawba Theory Notes, Fluid Flow Section

1-18 a. 378 BTU/LBM (used Mollier diagram) (1.0)

b. 3 (0.5)

REF.: Catawba Theory Notes, Thermodynamics Section

1-19 1. Incore thermocouples < 600°F and steady or decreasing (0.5/e

2. T_h wide range steady or decreasing

3. Core ΔT close to but less than full power ΔT

4. S/G pressure constant (or decreasing with T_h)

5. S/G level constant with constant feed flow

NOTE: Additional ones acceptable from procedures

REF.: Catawba notes on thermodynamics, page 24

2-1 a. Alarm is valid. (0.25) RESET opens the electronic circuits but does not offset the detector. A high activity will alarm immediately when a path is re-established. (0.5)

b. Alarm is invalid. (0.25) PURGE removes the activity from the sample volume. If the alarm does not clear, then it cannot be due to high activity. (0.5)

REF.: Catawba, Selected Lesson Plans, Vol. III, IEC Section 13, EMF

2-2 a. (0.5)

REF.: Catawba, Selected Lesson plans, Vol. III, IEC Section 2, IPE, page 23

2-3 d. (0.5)

2-4

SIGNAL	COINCIDENCE	SETPOINT
Manual (.15)	1/2 (.05)	Switch depressed (.05)
Hi Cont Pressure (.15)	2/3 (.05)	1.~~1~~ 2 psig (.05)
Low Steamline Pressure(.15)	2/3 on 1/4 steamline (.05)	~~710~~ psig (.05) 725
Low Pressurizer Pressure(.15)	2/4 (.05)	1845 psig (.05)

REF: Catawba, Selected Lesson Plans, Vol. III, Section IEC 3, ISE, pages 13-16. also T.S. table 3.3-4, page 3/4 3-27 for new values.

2-5 a. Breaker will trip open or breaker will not close because of the 20° sync check relay. The problem is that the two power sources will probably be out of phase. (1.0)

b. Properly done. (1.0)

c. Trying to motorize the generator and will trip on reverse power trip. (1.0)

REF: Catawba, Procedure OP/1/A/C350/02, Diesel Generator Op., page 2

2-6 1. On an emergency low level in the VCT, 1NV-252A and 1NV-253B will open to supply water to the suction of the charging pumps from the FWST. (0.5)

2. 1NV-188A and 1NV-189B in the VCT outlet line will close. (0.5) An emergency low level signal actuates an alarm in the Control Room. (0.5)

REF.: Catawba Question and Answer Bank, page 97

(continued next page)

2-7 a. (1) Centrifugal charging pumps - immediately on pump start (or 1845 when S_S signal is received) (0.4)
(2) NI pumps - approximately 1500 psig (1520 psig) (0.4)
(3) UHI - approximately 1250 psig (1240 psig) (0.4)
(4) C.L. accumulators - approximately 450 psig (0.4)
(5) ND pumps - approximately 200 psig (0.4)

b. Yes: Faster depressurization will cause the pressurized accumulators to inject sooner than the pumps due to time delay for D/G to start and sequence ECCS pumps on. (1.0)

REF: Catawba, Selected Lesson Plan Vol. 1, Sec. 4, NI, page 14

2-8 1. Temperature indication downstream of PORV *any 2* (0.5/ea.)
2. High discharge temperature alarm
3. Increasing PRT level and temperature (alarms)

REF: Catawba, question and answer bank, page 145.

2-9 1. Spray with reactor makeup water from RMWST (0.5)
2. Cooldown using NCDT heat exchanger (0.5)

REF: Catawba, Selected Lesson Plans, Vol. I, Sec. 1, NC, pages 17 & 18.
also question 147.

2-10 a. Steam driven turbine aux. feedwater pump to supply cooling to "B" and "C" S/Gs. (0.5)

b. A main steam PORV which can be throttled locally to control steam release rate from S/G, or S/G Safeties. (0.5)

c. Plant is designed to have natural circulation on a loss of NCPs because the Heat Sink is higher than the Heat Source. (0.5)

REF: Catawba, question and answer bank, page 186.

(continued next page)

2-11 a. The sequencer would load shed the essential bus. Then the ECCS equipment needed for safe shutdown would be sequenced back on relative to the SI signal. (1.0)

b. Blackout loads on the Bus when SI occurs will trip. The sequencer will block all non-LOCA loads and then sequence on all loads required for SI. (1.0)

c. Due to blackout, the Bus will be stripped of all loads and the blackout loads will be sequenced on. (1.0)

The Operator must respond and reset the sequencer, then load back on essential equipment required.

REF: Catawba, question and answer bank, page 120.

2-12 Flywheel - Ensures adequate flow during coastdown after loss of pump - DNB protection until heat flux has a chance to decay off. (0.5)

Anti-rotation device - Reduces starting currents by ensuring pump is not rotating backwards upon start and minimize core bypass flow through pumps. (0.5)

REF: Catawba, question and answer bank, page 89.

2-13 a. Abnormal conditions will be indicated in containment for an NC system leak to containment while the containment indications will be normal for an S/G tube leak. (0.5)

b. LOCA - Increasing indications exist for: (any 0.5

a. Containment pressure
b. Containment radiation
c. Containment sump level
d. Cont. Temp increasing
e. Cont. Humidity increasing
f. VUCDT Level increasing

S/G tube leak: (any 3 0.5

a. No containment indication changes and
b. High radiation in air ejector off-gas (EMF-33) or
c. High radiation in steam generator sample effluent (EMF-34)
d. S/G level abnormal high in 1/4 S/Gs (SGWLC will try to maintain level)

REF: Catawba, procedure EP/1/A/5000/1C, High energy line break inside containment, and EP/1/A/5000/1E, Steam Generator Tube Rupture.

also question 109.

(continued next page)

2-14	1.	S.R. NI High	10^5 cps	P-6, P-10	(0.4)
	2.	I.R. NI High	amp = 25% pwr	P-10	(0.4)
	3.	P.R. NI Lo	25% pwr	P-10	(0.4)
	4.	P.R. NI High	109% pwr		(0.4)
	5.	P.R. Pos/Neg Trip	±5%/2 sec.		(0.4)

REF.: Catawba, Selected Lesson Plans, Vol. III, ENB, pages 32 & 33.

also question 189

3-1 Any 5 @ 0.4 each

1. Containment sump level and flow monitoring
2. Containment radiation monitoring (EMF-38, 39, & 40)
3. Containment temperature
4. Containment humidity
5. Containment pressure
6. Containment ventilation condensate drain tank level

REF.: Catawba, Question and Answer Bank, Page 86

3-2 c. (1.0)

REF: Catawba, Selected Lesson Plans, Vol. III, Section 4, IRE, page 4. (1.0)

3-3
a. Closed (0.25)
b. Throttled (0.25)
c. Closed (0.25)
d. Closed (0.25)
e. Closed (0.25)
f. Closed (0.25)

REF: Catawba Selected Lesson Plans, Vol. III, Section 8.0, SGL, pages 12-14. Also, Catawba question and answer bank, page 180.

3-4
a. 3 (0.5)
b. 1 (0.5)
c. 1 (0.5)
d. 2 (0.5)

REF.: Catawba Selected Lesson Plans, Vol. III, IEC.

3-5
1. T_h (0.25)
2. T_c (0.25)
3. Turbine first stage (impulse) pressure (0.25)
4. Nuclear flux or nuclear power (0.25)

REF: Catawba Selected Lesson Plans, Vol. III, Section 4, IRE, pages 5-8.

3-6 False (0.5)

REF: Catawba Selected Lesson Plans, Vol. III, Section 10, IDE, page 13.

3-7 If an RTD shorts, it would provide zero resistance or a low temperature indication. (1.0)
If Tc RTD shorts, Tave would fail low and ΔT would fail high.
If Th RTD shorts, Tave would fail low and ΔT would fail low.

REF.: Catawba Selected Lesson Plans, Vol. 1, NC

3-8 Start a boric acid pump and open 1NV-236B (1.0)
OR
Start a boric acid pump and open 1NV-238A and manually unlock and open 1NV-240.

REF.: Catawba, Question and Answer Bank, page 92

(continued next page)

3-9 a. (1) Indicator light C-9 lit (condenser available) (0.33)

(2) Steam dump selector switch in Tavg position (0.33)

(3) Both steam dump channel switches in the ON position (0.33)

b. (1) Any dump valve OPEN indicating lights lit (0.5)

(2) Dump valve demand signal exceeding its deadband (0.5)

(3) arming signals present.

c. (1) Air supply system to the valves (any 2 @ 0.5 ea)

(2) Loss of electrical power to the steam dump controller or solenoid valves.

(3) Loss of any system that would prevent the C-9 permissive from becoming energized

may split up into 2.

REF: Catawba Selected Lesson Plans, Vol. III, Section 10, IDE.

3-10 FW temperature too low → indicated feed. to steam Δh too large → indicated S/G heat transfer rate too large → power calculated too high → PRNI indications are higher than actual power. (1.0)

REF.: Catawba Procedure PT/0/A/4220/01, Hand Heat Balance and NC flow calculation.

3-11 a. 1 (0.5)
b. 2 (0.5)
c. 2 (0.5)
d. 1 (0.5)

REF: Catawba Selected Lesson Plans, Vol. III, Section 12, EDA.

3-12 a. S/G Level Steam Flow (4 @ 0.25 ea)
Feed Flow Steam Pressure

b. Steam flow and steam pressure change and a signal is sent to the feed/steam mismatch amp, this signal will initially open the FEED valve. (0.~~25~~ 33) As nuclear power catches up and the program level setpoint increases, a mismatch of S/G level and program level will exist. This will send a signal to open the FEED valve more. (0.~~25~~ 33) As level approaches the new program level, the signals will decrease and opening of the valve will stop. (0.~~25~~ 33) ~~The FWP DP Program will increase the DP causing the valve to close back to its optimum throttle range. (0.25)~~

REF: Catawba, Selected Lesson Plans, Vol. III, Section 8, SGL.

also question 210.

(continued next page)

3-13 The circuit is undercompensated and the signal is too high because of additional gamma signal. (0.5)

The P-6 permissive will not be removed and therefore the source ranges will have to be manually re-energized, by the operator. (0.5)

REF: Catawba, Selected Lesson Plans, Vol. III, Section I, ENB, page 12
also question 113.

3-14 a. Both ranges use the same method of neutron detection:

Boron-10 is used to detect neutrons by undergoing a reaction wich (0.5)

$$ {}^{10}_{5}B + {}^{1}_{0}n \rightarrow [{}^{11}_{5}B] \rightarrow {}^{7}_{3}Li + {}^{4}_{2}He $$

The products of Li-7 and He-4 are ions which will cause additional ionizations inside the detector.

b. In the source range, the large pulses caused by the Li-7 and He-4 ions and the smaller pulses caused by the gamma ionizations are passed through a pulse height discriminator circuit which removes the smaller gamma pulses. (0.5)

In the intermediate range, two signals are produced: one is a combination of neutrons and gamma ionizations, and the second is a gamma signal only. Electronically, these signals oppose each other which causes the gamma signal to be cancelled out. (0.5)

REF: Catawba, Selected Lesson Plans, Vol. III, Section 1, ENB.
also question 213.

3-15 a. 1. all heaters come on (0.5)
2. PORV NC34 blocked (0.5)
3. PORV's NC32 and NC36 open (0.5)
4. Pressure modulates between 2235 and 2315 psig (0.5)
question doesn't no. 4. 3 may be 2 ans.

b. operator should select 3-2 switch position (0.5)

REF: Catawba Selected Lesson Plans, Vol. III, Section 6, ILE, page 25.

3-16 a. component cooling water surge tank vents are closed (0.5)
b. no control action from this channel (0.5)
c. closes air intakes to control room (0.5)
d. closes the following valves:

(1) Flow Control Valves BB65, 69, 24, 73 (0.25)
(2) Blowoff Tank Drain BB48 (0.25)
(3) Blowoff Tank Vent BB27 (0.25)
(4) S/G Sample Header Isolation Valves NM 267, 269, 270, 271, 272 (0.25)

REF: Catawba Selected Lesson Plans, Vol. III, Section 13.0, EMF, pages 9-13.

4-1 a. 1. To ensure uniform reactivity changes during rod movements. (0.5)

2. Proper bank overlap is one method that the operator has to ensure that the hot channel factors are within prescribed limits. (0.5)

b. Increasing the count by a factor of two will reduce the amount the reactor is shut down by one-half. Further dilution could cause criticality. (0.5)

REF.: Catawba, Question and Answer Bank, page 214

4-2 C (1.0)

REF.: Catawba "Controlling procedure for unit startup", ID NO. OP/1/A/6100/01, page 10

4-3 1. NCS pressure is stable or increasing, AND, (0.5)

2. Pressurizer level is greater than 5%, AND, any 3 (0.5)

3. RCS subcooling is greater than 50°F, AND, (0.5)

4. Secondary heat sink available, EP/01/A/5000/01 page 17.

REF.: Catawba, EP/1/A/5000/1B, page 2, SI Termination Following Spurious SI.

also question 220.

4-4 a. Try to trip the reactor manually because this is an ATWS event. (1.0)

b. Reactor failed to trip

1. Manually drive rods in (0.5)
2. Attempt to open Reactor trip breakers locally (0.5)
3. Trip M-G Sets (0.5)
If this fails--
4. Open supply breaker to M-G sets (0.5)

REF.: Catawba procedure EP/1/A/5000/2AI, Nuclear Power Generation/ATWS.

also question 215.

4-5 1. Go to manual on rod control (0.5)
2. Terminate load increase (0.5)
3. Terminate any boron change (0.5)

REF.: Catawba, procedure AP/1/A/5500/15, Rod Control malfunctions, page 2

also question 222.

(continued on next page)

4-6 (4/@0.5/

a. Pump Radial Bearing Temperature 225°
b. NCP Motor Bearing Temperature 195°
c. NCP Motor Stator Temperature 311°
d. #1 Seal Leakoff Temperature 235°
e. Hi-Hi Vibration -
   Shaft 20 mil
   Frame 5 mils
f. Hi/Lo Upper and Lower Bearing Oil Reservoir Levels
g. NC System at Saturation Conditions W/SI flow verified
h. Loss of KC flow to motors

REF.: Catawba, question and answer bank, page 228

4-7 The sequence of steps may be deviated from as long as the procedure and the intent of the instruction is not changed, and agreement of two (2) operators with one being a supervisor and holding an SRO license or is an SRO Cold license candidate. (1.0)

REF.: Catawba, question and answer bank, page 246

4-8
1. Manually trip the reactor (0.5)
2. Verify reactor trip (0.5)
3. Verify turbine trip (0.5)
4. Check DG status (are they running?) (0.5)
5. Obtain key & dispatch two (2) operators to establish NC pump seal injection. (0.5)

REF.: Catawba Nuclear Station Procedure ID NO. EP/1/A/5000/03, Loss of all AC power.

4-9 c. (1.0)

REF.: Catawba procedure EP/1/A/5000/1E, Steam generator tube rupture.

(continued on next page)

4-10
1. Manually trip the reactor (0.5)
2. Verify reactor trip (0.5)
3. Verify turbine trip (0.5)
4. Verify 4160V essential power buses energized (0.5)
5. Check if S/I is actuated (0.5)

REF.: Catawba procedure EP/1/A/5000/01, "Reactor Trip and Safety Injection."

4-11
1. Station Manager (0.5)
2. Shift Supervisor until Station Manager arrives. (0.5)

REF.: Catawba, Nuclear Station Directive 3.8.4, Onsite Emergency Organization

also question 233.

4-12 (0.3/ea.
1. •Letdown flow: Decreasing or Zero
2. •Pzr Level: Increasing (if No NC Leak Present)
3. •"PZR Hi Level Dev Control" Annunciator
4. •VCT Level: DECREASING
5. •Indications of Letdown Relief Valve(s) Lifting (To PRT or VCT)
6. Low [illegible] line temperature

REF.: Catawba procedure AP/1/A/5500/12, Loss of charging or Letdown, page 1

7. NV [illegible] NV2 SHUT

4-13 (0.25/ea
a. confined space entry permit
b. RWP
c. HP escort with high range detector
d. incore detectors in storage and tagged

REF.: Catawba question and answer bank, page 249

4-14 (8@0.25/

1. 1.25
2. 5.0 (no longer NRC requirement - count everything correct)
3. 1.0
4. 4.5
5. 7.5
6. 6.0
7. 18.75
8. 15.0

REF.: Catawba, Health Physics Program Requirements and General Information page 9

~~4-15 1. Pump radial bearing temperature >225°F~~ (0.25)
~~2. NCP motor bearing temperature >195°F~~ (0.25)
3. NCP motor stator temperature >311°F (0.25)
4. #1 seal leakoff temperature >235°F (0.25)
5. Hi-Hi vibration
shaft >20 mils (0.12
frame > 5 mils (0.12
6. Hi/Lo upper and lower bearing oil (0.25)
7. NC system at saturated conditions W/SI flow verified (0.25)
8. Loss of KC flow to motors (0.25)

~~REF.: Catawba Question and Answer Bank, page 228~~

Substituted question 4-15 and 4-16. answers on question sheet.

(Enclosure 3 2of2)

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

Facility: Catawba
Reactor Type: Westinghouse 4 Loop
Date Administered: September 10, 1984
Examiner: T. Rogers
Candidate: ____________

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
25.0	25.0	______	______	5. Theory of Nuclear Power Plant Operation, Fluids, and Thermodynamics
25.0	25.0	______	______	6. Plant Systems Design, Control, and Instrumentation
25.0	25.0	______	______	7. Procedures - Normal, Abnormal, Emergency, and Radiological Control
24.0	25.0	______	______	8. Administrative Procedures, Conditions, and Limitations
99.0	100.0	______	______	TOTALS

Final Grade ________%

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

____
Applicant Signature

5.0 - Theory of Nuclear Power Plant Operations, Fluids, and Thermodynamics (25.0)

5.1 The majority of heat generated from fission comes from: (1.0)

(a) Kinetic energy of fission fragments.

(b) Kinetic energy of prompt neutrons.

(c) Kinetic energy of delayed neutrons.

(d) Instantaneous gamma-ray energy.

5.2 The moderator temperature coefficient: (1.0)

(a) makes up part of the power defect.

(b) tends to become more positive (less negative) over core life.

(c) tends to become more positive (less negative) as Tave increases.

(d) tends to become larger (more negative) as the boron concentration is increased.

5.3 Which of the following will increase the equilibrium subcritical multiplication level? (1.0)

(a) An increase in Beff.

(b) Increasing the pressurizer level using the refueling water storage tank for makeup.

(c) Increasing the spacing between core assemblies.

(d) Increasing the pressurizer level using the reactor makeup water storage tank for makeup.

5.4 The equilibrium value of subcritical multiplication takes longer to reach following equal additions of reactivity as Keff approaches unity. The reason for this is: (1.0)

(a) More neutron generations occur from fuel fission neutrons.

(b) The fuel temperature rises allowing the thermal neutrons to be at a higher energy and thus more thermal neutrons.

(c) The moderator temperature rises decreasing the moderator density causing the neutrons to take longer to thermalize in order for the nuclear instruments to detect them.

(d) The ratio of source neutrons to fuel neutrons becomes so small that the source neutrons no longer contribute to the count rate causing the fuel to furnish additional neutrons.

5.5 Which of the following is NOT a true statement concerning Xenon poisoning? (1.0)

(a) The concentration will buildup and insert negative reactivity immediately following a reactor trip.

(b) The time after trip that Xenon peaks is independent of neutron flux before the trip.

(c) Equilibrium Xenon reactivity worth at 50% power is not half of the equilibrium Xenon reactivity worth at 100% power.

(d) Xenon reactivity worth will increase over core life.

5.6 Which of the following is a true statement concerning control rods? (1.0)

(a) The rod reactivity worth is highest at its axial centerline.

(b) Two adjacent rods will reduce each others reactivity worth.

(c) Rod reactivity worth is independent of boron concentration at a constant position.

(d) The rod reactivity worth is the primary control to compensate for fuel burn-up.

5.7 It takes less reactivity to go prompt critical at: (1.0)

(a) BOL

(b) EOL

(c) MOL

(d) the point of adding heat regardless of core life.

5.8 The minimum acceptable shutdown margin does not: (1.0)

(a) ensure the reactor can be made subcritical from all normal operating conditions.

(b) ensure the reactivity transients associated with postulated accident conditions in the FSAR are controllable within acceptable limits.

(c) ensure the reactor will be maintained sufficiently subcritical in the shutdown condition at EOL with a double ended main steam line break with Tavg at a no load operating temperature.

(d) remain constant if the boron concentration changes, Tavg changes, or as fuel depletion occurs over core life.

5.9 Which of the following does NOT contribute to brittle fracture occurring? (1.0)

(a) Tavg outside of the ductile region.

(b) Compressive thermal stresses.

(c) Increased fast neutron fluence.

(d) Increasing the copper content of the base metal of the reactor vessel.

5.10 Following a reactor trip from 100% power, how long would you expect it to take for the source range instruments to be energized? Show calculations and state assumptions. (1.0)

5.11 During fuel loading, which of the following will have no effect on the shape of a 1/M plot? (1.0)

(a) The location of the neutron sources in the core.

(b) The strength of the neutron sources in the core.

(c) The location of the neutron detectors around the core.

(d) The order of placement of fuel assemblies provided the proper enrichments are placed in their proper location.

5.12 A reactor start-up is in progress and it is necessary to dilute 200 ppm boron to get to the critical boron concentration before pulling the control banks. Prior to beginning the dilution, the source range instruments read 30 and 37 cps. After diluting 100 ppm of boron, the same instruments show 62 and 70 cps. Should you as reactor operator go ahead with the planned dilution of another 100 ppm? Explain why or why not. (1.0)

5.13 The design criteria for emergency core cooling systems to limit the maximum cladding oxidation to 17% refers to: (1.0)

(a) limiting the degradation of the cladding thickness to less than 17%.

(b) limiting the total metal in its cladding from reacting by oxidation to less than 17%.

(c) limiting the oxide to cladding metal ratio to less than 17% by volume.

(d) limiting its oxide to cladding metal ratio to less than 17% by weight.

5.14 Which of the following post accident containment hydrogen contributors is dependent on the radiation field intensity inside containment for the amount of hydrogen released? (1.0)

(a) $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$

(b) $2Al + 3H_2O \rightarrow Al_2O_3 + 3H_2$

(c) $Zn + 2H_2O \rightarrow ZN(OH)_2 + H_2$

(d) $2H_2O \rightarrow 2H_2 + O_2$

5.15 Which one of the following assures the operator that the heat flux hot channel factor upper bound normalized axial peaking factor is not exceeded in the event of Xenon redistribution following power changes? (1.0)

(a) maintaining all control rods in a group within ±12 steps of the group demand position.

(b) maintaining control rods in specified sequence and overlap.

(c) maintaining specified rod insertion limits.

(d) maintaining axial flux difference within specified limits.

5.16 Which of the following will contribute to a lower fuel centerline temperature over core life? (1.0)

(a) Fuel densification.

(b) Crud buildup on clad.

(c) Fission gases released to the gap between fuel and cladding.

(d) Clad creep.

5.17 During a plant heatup from cold shutdown, the heatup rate will naturally slowdown when establishing the steam volume in the steam generators. This phenomenon is a result of the energy change from pump heat in the reactor coolant system to: (1.0)

(a) specific heat in the steam generator.

(b) sensible heat on the steam generator.

(c) latent heat of vaporization in the steam generator.

(d) superheat in the steam generator.

5.18 With the pressurizer at 1000 psia and a PORV leaking to indicate a downstream pressure of 20 psia, what would be the downstream temperature? (1.0)

(a) 545°F

(b) 608°F

(c) 303°F

(d) 228°F

5.19 If pressure gauges are installed at the bottom of a cylindrical tank with a radius of four feet and at the bottom of a similar tank with a radius of two feet, which of the following is true of the gauge readings if the water level is the same in both tanks? (1.0)

(a) They would read the same.

(b) The larger tank reading would be twice the smaller tank reading.

(c) The larger tank reading would be four times the smaller tank reading.

(d) The larger tank reading would be eight times the smaller tank reading.

5.20 If a centrifugal pump is operating at 1800 rpm to give 400 gal/min at a discharge head of 20 psi, what would be the discharge head if the speed is increased in order to deliver 800 gpm? (1.0)

(a) 40 psi

(b) 60 psi

(c) 80 psi

(d) 160 psi

5.21 Which of the following is <u>NOT</u> true concerning heat exchangers? (1.0)

(a) Heat transfer is by both the conductive and convective methods of heat transfer.

(b) The heat transfer rate for a parallel flow heat exchanger is higher than that of a counter flow heat exchanger under the same temperatures.

(c) The heat transfer rate is directly proportional to the heat transfer coefficient associated with material the tubes are made of.

(d) Higher thermal stresses across the tubes will accompany higher tube thickness.

5.22 The condensate subcooling in a condenser operating at 1 psia with a condensate temperature of 95°F is approximately: (1.0)

(a) 7.7°F

(b) 196.7°F

(c) 1.07°F

(d) 25.3°F

5.23 The break horsepower of a pump refers to : (1.0)

(a) the actual power required to stop the pump when running at rated capacity.

(b) the actual power input to the pump.

(c) the useful power delivered by the pump to the fluid being pumped.

(d) the amount of power delivered as heat to the fluid being pumped.

5.24 Which of the following is NOT advantageous to the promotion of natural circulation? (1.0)

(a) A means of continuously removing heat from the system.

(b) A means of continuously adding heat to the system.

(c) A complete flow path.

(d) Increasing the velocity of the fluid.

5.25 Which of the following boiling regimes is acceptable during normal reactor operations? (1.0)

(a) Transition boiling

(b) Film boiling

(c) Partial film boiling

(d) Nucleate boiling

Write "END OF SECTION 5.0" on your answer sheet.

6.0 Plant Systems Design, Control, and Instrumentation. (25.0)

6.1 Which of the following is a true statement concerning the reactor protection system's response for protecting DNBR? (1.0)

(a) If (i) Loop Flow Channel I of Loop 1 indicates 80% flow and (ii) Loop Flow Channel II of Loop 2 indicates 75% flow and (iii) the PR nuclear instruments indicate 30% the reactor will trip.

(b) If reactor coolant pump busses 1 and 2 drop to 55 Hz all four reactor coolant pump breakers will open regardless of power level.

(c) If reactor coolant pump busses 1 and 4 drop to 4600 volts, reactor coolant pump breakers 1 and 4 will open and the reactor will trip if operating at 30% power.

(d) If the reactor coolant pump breakers are open, the under voltage trip signals will also be present.

6.2 If Loop A Th RTD has an open circuit, the rod control system will: (1.0)

(a) not respond because its input is from Tave.

(b) not respond because the temperature inputs are auctioneered to allow only the highest loop temperature input to effect the rod control system response.

(c) will drive rods in until the temperature mismatch circuitry is satisfied.

(d) Will drive rods out until the temperature mismatch circuitry is satisfied.

6.3 A high containment pressure Automatic Safety injection signal will: (1.0)

(a) cause main steamline isolation.

(b) be initiated by 1/3 containment pressure instruments greater than 1.2 psig.

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

(d) always cause feedwater isolation, phase "A" isolation, and containment ventilation isolation signals.

6.4 Which of the following statements is true concerning the upper head injection system? (1.0)

(a) Accumulator isolation valves will auto close on low nitrogen surge tank pressure.

(b) Accumulator isolation valves will auto gag if a safety injection signal is present when they're shut by low accumulator level.

(c) The motor operated gag cannot be engaged unless the isolation valve is fully open.

(d) The standpipe and level switch arrangement of the UHI membrane leak detector is connected to the water side of the rupture disc membrane.

6.5 Which of the following describes a condensate booster pump interlock? (1.0)

(a) The discharge valve must be opened before and closed after the associated suction valve.

(b) If two out of three condensate booster pumps trip one feedwater pump turbine will auto trip.

(c) If main feedwater pump discharge header pressure is low coincident with low feedwater pump bearing oil pressures, opening a feedwater pump suction valve will trip all running condensate booster pumps.

(d) The condensate booster pumps cannot be started when the associated discharge valve is open.

6.6 The #3 reactor coolant pump seal leakoff is normally collected in the: (1.0)

(a) containment sump

(b) pressurizer relief tank

(c) volume control tank

(d) reactor coolant drain tank

6.7 Which of the following is not a waste gas compressor trip? (1.0)

(a) High moisture separator pressure.

(b) Low moisture separator pressure.

(c) High suction pressure.

(d) Low suction pressure.

6.8 The Reactor building manipulator crane bridge and trolley drives are operable only when the : (1.0)

(a) fuel gripper is fully up or down disengaged and the rod gripper is fully up or up first stage.

(b) fuel gripper is fully up or up disengaged and rod gripper is fully down.

(c) fuel gripper is fully up or up disengaged and rod gripper is fully up or up first stage.

(d) fuel gripper is fully up or up disengaged or the rod gripper is fully up or up first stage.

6.9 If turbine impulse pressure channel 1 fails high, the steam dump system will: (1.0)

(a) have no response even if C7A or C7B is armed.

(b) causes steam dumps to open if only C7A is armed until the Tave/Tref mismatch is satisfied.

(c) cause steam dumps to open if only C7B is armed until the Tave/Tref mismatch is satisfied.

(d) cause steam dumps to open only if C7A and C7B are armed until the Tave/Tref mismatch is satisfied.

6.10 Which of the following will Not cause a turbine runback? (1.0)

(a) Five psig stator cooling water pressure at 30% power.

(b) Loss of all cooling power to transformer 1A at 50% power.

(c) Feedwater pump turbine 1A trips at 85% power

(d) OPDT is within 2% of setpoint at 20% power.

6.11 When establishing a cooldown rate on ND, the operator will manually adjust: (1.0)

(a) the heat exchanger outlet valves (ND 26 & 60).

(b) the heat exchanger outlet valves and bypass valves (ND 26, 27, 60 & 61).

(c) the heat exchanger bypass valves (ND 27 & 61).

(d) the component cooling water outlet isolation valves to the heat exchangers (VC 57A & 82B).

6.12 The reactor coolant pump motor air cooler is cooled by the: (1.0)

(a) component cooling water system.

(b) containment cooling water system.

(c) nuclear service water system.

(d) low pressure service water system.

6.13 If Power range channel 41 fails high when selected for S/G level control with the plant at 75% power, the SG water level control system will: (1.0)

(a) cause SG A and D levels to increase to 50%.

(b) cause SG B and C levels to increase to 50%.

(c) cause a SG A and D levels to increase to 66%.

(d) cause SG B and C levels to increase to 66%.

6.14 If containment spray initiates due to Hi-Hi containment pressure, spray will be secured from the containment: (1.0)

(a) automatically when the containment pressure control signal reaches 0.35 psig.

(b) automatically when the containment pressure control signal reaches 0.25 psig.

(c) by manually securing the spray pumps after the spray signal has been reset and containment pressure is less than 1.2 psig.

(d) by depressing the spray signal reset button whenever the containment pressure is less than 1.2 psig.

6.15 If the digital rod position indication system suffers a data cabinet A failure, the operator will be aware of this by: (1.0)

(a) an urgent annunciator and a general warning light.

(b) an urgent annunciator a general warning light and a rod bottom light.

(c) a non-urgent annunciator and a general warning light.

(d) a non-urgent annunciator, a general warning light and a rod bottom light.

6.16 Which statement is true concerning the source range audio count rate circuit? (1.0)

(a) If the control room audio count rate is lost due to a detector failure, then the containment audio count rate is lost at the same time.

(b) If the control room audio count rate is lost due to a detector failure, the audio count rate can be restored by a selector switch behind the audio count rate instrument cabinet back panel.

(c) If the control room audio count rate is lost due to a detector failure, the audio count rate can be restored by a selector switch on the front of the audio count rate panel but then the containment audio count rate will be lost.

(d) If the control room audio count rate is lost due to an amplifier failure, the control room audio count rate will be lost until the amplifier is repaired or replaced.

6.17 If pressurizer level channel I is selected for control when pressurizer level channel II fails low, which of the following will NOT occur? (1.0)

(a) Regenerative heat exchanger's isolation valve NV2A will shut.

(b) Letdown orifice isolation valves NV-10, NV-11, and NV13 will shut.

(c) Pressurizer heater groups A, B, C, and D will de-energized.

(d) A -5% alarm will occur.

6.18 The reactor coolant system low range pressure instruments tap into (1.0)

(a) loop A

(b) loop B

(c) loop C

(d) loop D

6.19 A major difference between an ion chamber and a G-M detector is: (1.0)

(a) the G-M detector has a photo-multiplier tube to increase it's sensitivity.

(b) the ion chamber is filled with a gas and the G-M detector operates under a vacuum.

(c) the ion chamber operates at such a low voltage that a significant number of ion pairs are lost by recombination thereby decreasing it's sensitivity.

(d) the G-M detector operates at a much higher voltage causing gas multiplication to increase the charge collected to a value independent of the ionization initiating it.

6.20 Which of the following radiation monitors do NOT have any control function? (1.0)

(a) EMF-15 (refueling bridge-spent fuel building area monitor).

(b) EMF-37 (unit vent iodine monitor).

(c) EMF-43 (control room ventilation monitor).

(d) EMF-45 (nuclear service water monitor).

6.21 Which of the following is NOT a true statement concerning turbine eccentricity? (1.0)

(a) Eccentricity is a measurement to indicate the degree of turbine shaft straightness.

(b) The eccentricity instrument is used at low turbine speeds and the bearing vibration instrument is used as an indication of eccentricity at high turbine speeds.

(c) If eccentricity exceeds 10% of the alarm setpoint, the turbine will trip.

(d) The eccentricity detector is mounted on the turbine front standard.

6.22 The emergency diesel generator load sequencer: (1.0)

(a) will sequence on the same loads for a LOCA and Blackout but the order will be different.

(b) will not be effected when sequencing on loads due to a blackout if the reset button is pushed.

(c) will not be prevented from loading when auxiliary shutdown panel selector switch is in LOCAL if the actuation signal is due to a LOCA.

(d) gives LOCA loads first priority and blackout loads second priority.

6.23 Which of the following would NOT prevent the emergency diesel generator from automatically starting? (1.0)

(a) Starting air pressure at 175 psig.

(b) Engine overspeed levers tripped.

(c) Maintenance Mode selected.

(d) Generator differential relay not reset.

6.24 The emergency diesel generator field flash power is provided by: (1.0)

(a) a permanent magnet generator on the diesel.

(b) 125 VDC Diesel Generator control power.

(c) 120 VAC vital instrument and control power.

(d) 120 VAC diesel generator auxiliary power.

"Section Continued On Next Page".

6.25 In the event that auxiliary AC power is lost, which of the following cannot be considered a condensate water supply for the auxiliary feedwater system? (1.0)

(a) UST

(b) CST

(c) Hotwell

(d) CACST

Write "END OF SECTION 6.0" on your answer sheet.

7.0 Procedures - Normal, Abnormal, Emergency, and Radiological Control (25.0)

7.1 When forming a Pressurizer steam bubble in accordance with OP/1/A/6100/01, nitrogen venting from the Pressurizer is considered complete: (1.0)

(a) by meeting the venting time requirements per the unit data graph based on the initial nitrogen concentration.

(b) by lowering the nitrogen concentration to less than 25cc/kg based on sample analysis by the Primary Chemist.

(c) by reducing PZR level to less than 92% while continuously venting.

(d) by verifying the PRT pressure does not increase with corresponding PRT level increase.

7.2 When performing a unit startup from a cold condition per OP/1/A/6100/01, why does the procedure caution the operator not to exceed 1955 psig NC pressure prior to S/G pressure reaching 710 psig? (1.0)

7.3 While performing a cooldown per OP/1/A/6100/02 via the ND system following a unit shutdown, there are four items that must be verified prior to securing all NC pumps. List any three. (1.5)

7.4 Steam generators are placed in wet layup recirculation whenever a cold shutdown duration is greater than: (1.0)

(a) 12 hours

(b) 24 hours

(c) 72 hours

(d) 7 days

7.5 How is NC pressure maintained at 75 psig when on a PZR steam bubble? (1.0)

7.6 During a unit startup with the reactor power just above 10%, what three permissives must be verified? Indicate whether the permissive light is on or off. (1.5)

7.7 List two indications of a boron dilution accident when the plant is shutdown per AP/1/A/5500/13. (1.0)

7.8 If RN pump 1A trips and cannot be restarted for an indefinite period of time during mode 1 operation, the appropriate action would be to: (1.0)

(a) trip the reactor.

(b) commence unit shutdown.

(c) start or verify running pump 1B and verify it is providing adequate flow.

(d) start or verify running RN pump 2A and verify it is providing adequate flow.

7.9 What are your immediate actions to a loss of normal power to an essential train? (1.5)

7.10 If an NC to KC leak has been confirmed by KC surge tank level increasing and by KC activity analysis, what three components are checked to identify the source of the leak? (1.5)

7.11 If control bank D fails to insert while in automatic control during a large down power transient, what should the operator verify prior to an attempt to move the rods manually? (.5)

7.12 What are the immediate actions if a fuel assembly is dropped in the reactor building during refueling and EMF-38, EMF-39, and EMF-17 are in alarm? (1.5)

7.13 What are the three plant conditions that must be met in order to terminate a spurious safety injection? (1.5)

7.14 When an operator is dispatched to verify P-4 operation during a safety injection, (1.0)

(a) to where does he go?

(b) and what should he see to verify P-4 operation?

7.15 List three criteria that must be satisfied to determine that a natural circulation cooldown is complete. (1.5)

7.16 List four different items you are required to check in preparing the containment for reactor vessel venting due to a void in the reactor vessel. (1.0)

7.17 The SGTR ALTERNATE COOLDOWN USING BACKFILLING procedure uses normal PZR spray to reduce NC system pressure: (1.0)

(a) to less than the SG with the ruptured tube in order to use the SG water to increase PZR level.

(b) to less than the UHI tank pressure in order to use the UHI water to increase PZR level.

(c) to less than the cold leg accumulator pressure in order to use the CLA water to increase PZR level.

(d) to less than the SI pump shut off head in order to use the SI pumps to increase PZR level.

7.18 Which of the following is a 10 CFR 20 occupational dose limit that does not require Form NRC-4 to be kept on record? (1.0)

(a) 3 rems per quarter - whole body.

(b) 1250 mrems per quarter - whole body.

(c) 5 rems per year - whole body.

(d) 7500 mrems per quarter - hands and forearms.

7.19 Goggles and face shields are used to shield __________ radiation because it is considered as a whole - body dose when exposed to the lens of the eye. (1.0)

(a) alpha

(b) beta

(c) gamma

(d) neutron

7.20 Contaminated trash should be placed in a __________ drum. (1.0)

(a) white

(b) yellow

(c) magenta

(d) black

7.21 At what whole body projected dose does RP/0/A/5000/05- General Emergency, recommend mandatory evacuation of the population in the affected area? (1.0)

(a) Greater than 5 mrem.

(b) Greater than 10 mrem.

(c) Greater than 5 rem.

(d) Greater than 10 rem.

Section 7 Continued on next page

7.22 The basic permissible Duke Power Company dose limits are: (1.0)

(a) 1.0 rem/quarter - whole body
6.0 rem/quarter - skin
15.0 rems/quarter - extremities

(b) 1.25 rem/quarter - whole body
6.0 rem/quarter - skin
15.0 rem/quarter - extremities

(c) 1.25 rem/quarter - whole body
7.5 rem/quarter - skin
15.0 rem/quarter - extremities

(d) 1.0 rem/quarter - whole body
7.5 rem/quarter - skin
14.75 rem/quarter - extremities

Write "END OF SECTION 7.0" on your answer sheet.

8.0 Administrative Procedures, Conditions, and Limitations (25.0)

8.1 Which of the following is NOT true concerning AFD Limits? (1.0)

(a) Required Operator action due to penalty deviation time is only applicable when operating above 50% of rated thermal power.

(b) The target band will change over core life and again at the next core cycle.

(c) If operating at 75% of rated thermal power and 1 hour of cumulative penalty deviation time has been calculated for the previous 24 hours, thermal power must be reduced to less than 50% within 30 minutes.

(d) If operating at 100% at rated thermal power and the indicated AFD is outside of the required target band, the AFD target band must be restored to within limits or reduce power to less than 90% within 15 minutes.

8.2 If the shutdown margin is less than 1.3% Δ K/K, then the required action would be to ________________, during mode 1 operations : (1.0)

(a) immediately initiate a shutdown and emergency borate at greater than or equal to 30 gpm using the boric acid tank until a new shutdown margin calculation has been done to verify the shutdown margin is 1.3% Δ K/K.

(b) immediately initiate and continue boration at greater than or equal to 30 gpm using the refueling water storage tank to borate until the shutdown margin is 1.3% Δ K/K.

(c) immediately initiate and continue boration at greater than or equal to 30 gpm using the boric acid tank to borate until the shutdown margin is 1.3% Δ K/K.

(d) immediately initiate a shutdown and initiate and continue boration of greater than or equal to 30 gpm using the boric acid tank to borate until the rod insertion limits are satisfied.

8.3 The pressurizer pressure high reactor trip channels has an analog channel operational test to be performed monthly, a review of the logs indicates that the test is normally due on the tenth of each month. The logs show however that it was done on June 10, July 13, and August 15. What would be the date of the maximum allowable extension of the surveillance test this month without declaring the channels inoperable? (1.0)

(a) September 16

(b) September 18

(c) September 22

(d) September 23

8.4 Which of the following conditions does Technical Specifications require placing the plant in a hot shutdown condition from mode 1 within 30 minutes if the LCO cannot be satisfied? (1.0)

(a) The lowest operating loop Tavg dropping to less than 551°F.

(b) One charging pump becomes inoperable.

(c) The nitrogen cover gas pressure is 350 psig on a cold leg accumulator.

(d) One component cooling water pump becomes inoperable.

8.5 Which of the following would cause an emergency diesel generator to be declared inoperable during mode 1? (1.0)

(a) Only one independent circuit between the offsite transmission network and the Onsite Essential Auxiliary Power System is available.

(b) One EDG day tank has 525 gallons of fuel and the other has 550 gallons of fuel.

(c) One fuel storage system contains 83,000 gallons of fuel and the other has 85,000 gallons of fuel.

(d) One EDG 125 VDC auxiliary power battery is undergoing an equalizing charge and the other is lined up to the float voltage of it's charger.

8.6 In order to perform surveillance testing on an instrument channel that is required to meet the Technical Specifications minimum operability requirements for ESF automatic actuation: (1.0)

(a) the plant must be placed in a mode in which the specification does not apply.

(b) the testing may be performed provided the other channel is operable and the channel to be tested is not bypassed for more than two hours.

(c) the testing may be performed because the channel may still be credited towards the minimum operability requirements during surveillance testing unless the test demonstrates the channel's inoperability.

(d) the testing may be performed provided the channel undergoing testing is placed in its tripped condition.

8.7 Which of the following statements is true concerning Technical Specifications limiting conditions for operation of pressurizer code safety relief valves? (1.0)

(a) All three code safety valves must be operable for modes 1, 2, and 3 and two code safety, valves must be operable for modes 4, 5, and 6.

(b) All three code safety valves or two code safety valves and two pressurizer power operated relief valves must be operable for modes 1, 2, and 3 and one code safety valve must be operable for modes 4 and 5.

(c) All three code safety valves must be operable for modes 1, 2, 3, and 4 and one code safety valve must be operable for modes 5 and 6.

(d) All three code safety valves must be operable for Modes 1, 2, and 3 and one must be operable for modes 4 and 5.

8.8 During modes 1, 2, 3, and 4, Technical Specifications allow the containment ventilation unit condensate drain tank level monitoring subsystem to be inoperable for the reactor coolant system leakage detection systems provided the containment: (1.0)

(a) floor and equipment sump level is operable.

(b) floor and equipment sump flow monitoring subsystem is operable.

(c) atmosphere particulate radioactivity monitoring system is operable.

(d) atmosphere gaseous radioactivity monitoring system is operable.

8.9 The refueling water storage tank is inoperable per Technical Specifications for modes 1, 2, 3, and 4 if (1.0)

(a) the water volume is 400,000 gallons.

(b) the boron concentration is 2050 ppm boron.

(c) the solution temperature is 75°F.

(d) the solution temperature is 105°F.

8.10 Question Deleted

8.11 Which of the following would eliminate the STA from being allowed to assume the control room command function and serve as the SRO in the case when both the shift supervisor and the SRO are absent from the control room? Base your choice only on the information provided. Assume all other requirements are satisfied. (1.0)

(a) The shift supervisor can be back in the control room in eight minutes if needed.

(b) The SRO will be back to assume control room duties in twelve minutes with this relief being the third time during the shift that he'll be relieved for twelve minutes.

(c) The SRO will be back to assume control room duties in ten minutes with this relief being the fifth time during the shift that he'll be relieved for ten minutes.

(d) The STA has a reactor operator license on the unit.

8.12 A delay to begin a shutdown of one hour may be granted by the Superintendent of Operations if the plant is immediately placed and continues to operate in a safe condition after exceeding: (1.0)

(a) a safety limit, a trip set point allowable value, or a Limiting Condition for Operations.

(b) a trip set point allowable value or a limiting condition for operation but not a safety limit.

(c) a limiting condition for operation but not a safety limit or a trip setpoint allowable value.

(d) a trip set point allowable value but not a safety limit or a limiting condition for operation.

8.13 Temporary Station Modifications are approved by the: (1.0)

(a) Projects Engineer.

(b) Shift Supervisor.

(c) Superintendent of Operations.

(d) Shift Technical Advisor.

8.14 After normal working hours, the ______________ will determine if a work area is a confined space or an enclosed space. (1.0)

(a) work supervisor

(b) shift supervisor

(c) safety section

(d) chemistry supervisor

8.15 If you are unable to reach your assembly point during a site assembly, you must call in to be accounted for within ____________ of the announcement. (1.0)

(a) five minutes

(b) ten minutes

(c) thirty minutes

(d) one hour

8.16 Safety tag stubs are not removed until the: (1.0)

(a) work supervisor signs for issue approval.

(b) shift supervisor signs for issue approval.

(c) safety tag is affixed to the required operating device.

(d) safety tag is removed from the operating device for clearance.

8.17 Before entering the reactor building when the reactor is critical, permission to enter must be received from the: (1.0)

(a) Plant or Assistant Plant Managers.

(b) Superintendent of Operations.

(c) Unit Supervisor.

(d) Operator at the Controls.

8.18 Fuel handling interlocks may be bypassed when not directed by an approved written procedure if approval is given by (indicate minimum level required): (1.0)

(a) any two licensed individuals provided one is an SRO.

(b) any two licensed individuals provided one is the fuel handling supervisor.

(c) any two operations group supervisors provided one is the shift supervisor.

(d) any operations group supervisor and the superintendent of operations.

8.19 Which of the following logbooks are not normally found in use in the control room? (1.0)

(a) Work request tagout logbook

(b) Diesel generator logbook

(c) Technical Specification Action Items logbook

(d) Test logbook

8.20 The fuel maneuvering limits are a set of instructions limiting: (1.0)

(a) the change of axial fuel reactivity worth as the fuel burns from the bottom to the top of the core.

(b) the reactor power changes to specified ramp rates which are only applicable for the initial core.

(c) the reactor power changes to specified ramp rates and is applicable after any core alterations.

(d) the ratio of high enrichment to low enrichment fuel assemblies loaded in the core during core alterations.

8.21 White shrouds on the main control boards are used around: (1.0)

(a) pump motor switches.

(b) motor-operated throttle valve switches.

(c) motor-operated gates valves.

(d) switchyard motor-operated disconnects.

8.22 When a deviation from the sequence of steps in an approved procedure is necessary, it is documented so by: (1.0)

(a) the individual performing the procedure on the control copy of the procedure.

(b) the individual performing the procedure on the controlled copy of the procedure along with the initials of the SRO approving the change.

(c) the individual performing the procedure on the working copy of the procedure along with the initials of the SRO approving the change.

(d) the individual performing the procedure on the working copy of the procedure along with a shift log entry by the SRO approving the change.

8.23 A key audit is required to be performed: (1.0)

(a) Every shift

(b) Daily

(c) Weekly

(d) Monthly

8.24 The "Buddy System" must be used when making a reactor building entry whenever the reactor coolant system temperature is: (select minimum temperature requirement) (1.0)

(a) Greater than 140° F

(b) Greater than 200° F

(c) Greater than 350° F

(d) Greater than 450° F

8.25 A 500mr scale pocket dosimeter should be rezeroed when it reads: (indicate minimum reading for required rezeroing) (1.0)

(a) 200mr

(b) 250mr

(c) 300mr

(d) 350mr

Write "END OF SECTION 8.0" on your answer sheet.

TABLE D-1a*
Properties of Dry Saturated Steam +
Pressure

Abs press., psia	Temp., °F	Specific volume		Enthalpy			Entropy		
		Sat liquid	Sat vapor	Sat liquid	Evap	Sat vapor	Sat liquid	Evap	Sat vapor
p	t	v_f	v_g	h_f	h_{fg}	h_g	s_f	s_{fg}	s_g
1.0	101.74	0.01614	333.6	69.70	1036.3	1106.0	0.1326	1.8456	1.9782
2.0	126.08	0.01623	173.73	93.99	1022.2	1116.2	0.1749	1.7451	1.9200
3.0	141.48	0.01630	118.71	109.37	1013.2	1122.6	0.2008	1.6855	1.8863
4.0	152.97	0.01636	90.63	120.86	1006.4	1127.3	0.2198	1.6427	1.8625
5.0	162.24	0.01640	73.52	130.13	1001.0	1131.1	0.2347	2.6094	1.8441
6.0	170.06	0.01645	61.98	137.96	996.2	1134.2	0.2472	1.5820	1.8292
7.0	176.85	0.01649	53.64	144.76	992.1	1136.9	0.2581	1.5586	1.8167
8.0	182.86	0.01653	47.34	150.79	988.5	1139.3	0.2674	1.5383	1.8057
9.0	188.28	0.01656	42.40	156.22	985.2	1141.4	0.2759	1.5203	1.7962
10	193.21	0.01659	38.42	161.17	982.1	1143.3	0.2835	1.5041	1.7876
14.696	212.00	0.01672	26.80	180.07	970.3	1150.4	0.3120	1.4446	1.7566
15	213.03	0.01672	26.29	181.11	969.7	1150.8	0.3135	1.4415	1.7549
20	227.96	0.01683	20.089	196.16	960.1	1156.3	0.3356	1.3962	1.7319
25	240.07	0.01692	16.303	208.42	952.1	1160.6	0.3533	1.3606	1.7139
30	250.33	0.01701	13.746	218.82	945.3	1164.1	0.3680	1.3313	1.6993
35	259.28	0.01708	11.898	227.91	939.2	1167.1	0.3807	1.3063	1.6870
40	267.25	0.01715	10.498	236.03	933.7	1169.7	0.3919	1.2844	1.6763
45	274.44	0.01721	9.401	243.36	928.6	1172.0	0.4019	1.2650	1.6669
50	281.01	0.01727	8.515	250.09	924.0	1174.1	0.4110	1.2474	1.6585
55	287.07	0.01732	7.787	256.30	919.6	1175.9	0.4193	1.2316	1.6509
60	292.71	0.01738	7.175	262.09	915.5	1177.6	0.4270	1.2168	1.6438
65	297.97	0.01743	6.655	267.50	911.6	1179.1	0.4342	1.2032	1.6374
70	302.92	0.01748	6.206	272.61	907.9	1180.6	0.4409	1.1906	1.6315
75	307.60	0.01753	5.816	277.43	904.5	1181.9	0.4472	1.1787	1.6259
80	312.03	0.01757	5.472	282.02	901.1	1183.1	0.4531	1.1676	1.6207
85	316.25	0.01761	5.168	286.39	897.8	1184.2	0.4587	1.1571	1.6158
90	320.27	0.01766	4.896	290.56	894.7	1185.3	0.4641	1.1471	1.6112
95	324.12	0.01770	4.652	294.56	891.7	1186.2	0.4692	1.1376	1.6068
100	327.81	0.01774	4.432	298.40	888.8	1187.2	0.4740	1.1286	1.6026
110	334.77	0.01782	4.049	305.66	883.2	1188.9	0.4832	1.1117	1.5948

TABLE D-1a
Properties of Dry Saturated Steam *(continued)*
Pressure

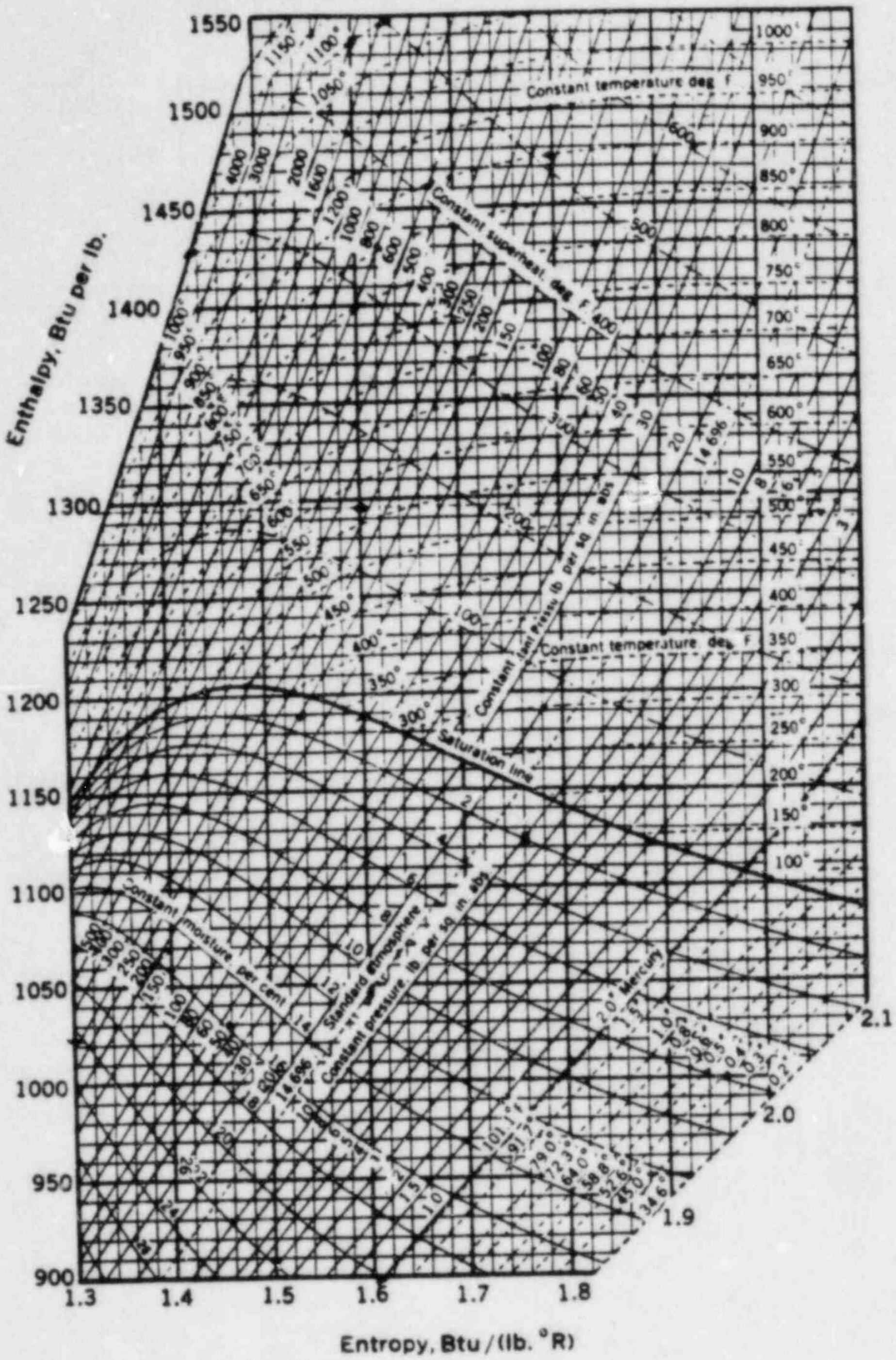
Abs. press., psia	Temp., °F	Specific volume		Enthalpy			Entropy		
		Sat liquid	Sat vapor	Sat liquid	Evap	Sat vapor	Sat liquid	Evap	Sat vapor
p	t	v_f	v_g	h_f	h_{fg}	h_g	s_f	s_{fg}	s_g
120	341.25	0.01789	3.728	312.44	877.9	1190.4	0.4916	1.0962	1.5878
130	347.32	0.01796	3.455	318.81	872.9	1191.7	0.4995	1.0817	1.5812
140	353.02	0.01802	3.220	324.82	868.2	1193.0	0.5069	1.0682	1.5751
150	358.42	0.01809	3.015	330.51	863.6	1194.1	0.5138	1.0556	1.5694
160	363.53	0.01815	2.834	335.93	859.2	1195.1	0.5204	1.0436	1.5640
170	368.41	0.01822	2.675	341.09	854.9	1196.0	0.5266	1.0324	1.5590
180	373.06	0.01827	2.532	346.03	850.8	1196.9	0.5325	1.0217	1.5542
190	377.51	0.01833	2.404	350.79	846.8	1197.6	0.5381	1.0116	1.5497
200	381.79	0.01839	2.288	355.36	843.0	1198.4	0.5435	1.0018	1.5453
250	400.95	0.01865	1.8438	376.00	825.1	1201.1	0.5675	0.9588	1.5263
300	417.33	0.01890	1.5433	393.84	809.0	1202.8	0.5879	0.9225	1.5104
350	431.72	0.01913	1.3260	409.69	794.2	1203.9	0.6056	0.8910	1.4966
400	444.59	0.0193	1.1613	424.0	780.5	1204.5	0.6214	0.8630	1.4844
450	456.28	0.0195	1.0320	437.2	767.4	1204.6	0.6356	0.8378	1.4734
500	467.01	0.0197	0.9278	449.4	755.0	1204.4	0.6487	0.8147	1.4634
550	476.94	0.0199	0.8424	460.8	743.1	1203.9	0.6608	0.7934	1.4542
600	486.21	0.0201	0.7698	471.6	731.6	1203.2	0.6720	0.7734	1.4454
650	494.90	0.0203	0.7083	481.8	720.5	1202.3	0.6826	0.7548	1.4374
700	503.10	0.0205	0.6554	491.5	709.7	1201.2	0.6925	0.7371	1.4296
750	510.86	0.0207	0.6092	500.8	699.2	1200.0	0.7019	0.7204	1.4223
800	518.23	0.0209	0.5687	509.7	688.9	1198.6	0.7108	0.7045	1.4153
850	525.26	0.0210	0.5327	518.3	678.8	1197.1	0.7194	0.6891	1.4085
900	531.98	0.0212	0.5006	526.6	668.8	1195.4	0.7275	0.6744	1.4020
950	538.43	0.0214	0.4717	534.6	659.1	1193.7	0.7355	0.6602	1.3957
1000	544.61	0.0216	0.4456	542.4	649.4	1191.8	0.7430	0.6467	1.3897
1100	556.31	0.0220	0.4001	557.4	630.4	1187.7	0.7575	0.6205	1.3780
1200	567.22	0.0223	0.3619	571.7	611.7	1183.4	0.7711	0.5956	1.3667
1300	577.46	0.0227	0.3293	585.4	593.2	1178.6	0.7840	0.5719	1.3559
1400	587.10	0.0231	0.3012	598.7	574.7	1173.4	0.7963	0.5491	1.3454
1500	596.23	0.0235	0.2765	611.6	556.3	1167.9	0.8082	0.5269	1.3351
2000	635.82	0.0257	0.1878	671.7	463.4	1135.1	0.8619	0.4230	1.2849
2500	668.13	0.0287	0.1307	730.6	360.5	1091.1	0.9126	0.3197	1.2322
3000	695.36	0.0346	0.0858	802.5	217.8	1020.3	0.9731	0.1885	1.1615
3206.2	705.40	0.0503	0.0503	902.7	0	902.7	1.0580	0	1.0580

TABLE D-1b
Properties of Dry Saturated Steam (*continued*)
Temperature

Temp., °F	Abs press., psia	Specific volume		Enthalpy			Entropy		
		Sat liquid	Sat vapor	Sat liquid	Evap	Sat vapor	Sat liquid	Evap	Sat vapor
t	p	v_f	v_g	h_f	h_{fg}	h_g	s_f	s_{fg}	s_g
32	0.08854	0.01602	3306	0.00	1075.8	1075.8	0.0000	2.1877	2.1877
35	0.09995	0.01602	2947	3.02	1074.1	1077.1	0.0061	2.1709	2.1770
40	0.12170	0.01602	2444	8.05	1071.3	1079.3	0.0162	2.1435	2.1597
45	0.14752	0.01602	2036.4	13.06	1068.4	1081.5	0.0262	2.1167	2.1429
50	0.17811	0.01603	1703.2	18.07	1065.6	1083.7	0.0361	2.0903	2.1264
60	0.2563	0.01604	1206.7	28.06	1059.9	1088.0	0.0555	2.0393	2.0948
70	0.3631	0.01606	867.9	38.04	1054.3	1092.3	0.0745	1.9902	2.0647
80	0.5069	0.01608	633.1	48.02	1048.6	1096.6	0.0932	1.9428	2.0360
90	0.6982	0.01610	468.0	57.99	1042.9	1100.9	0.1115	1.8972	2.0087
100	0.9492	0.01613	350.4	67.97	1037.2	1105.2	0.1295	1.8531	1.9826
110	1.2748	0.01617	265.4	77.94	1031.6	1109.5	0.1417	1.8106	1.9577
120	1.6924	0.01620	203.27	87.92	1025.8	1113.7	0.1645	1.7694	1.9339
130	2.2225	0.01625	157.34	97.90	1020.0	1117.9	0.1816	1.7296	1.9112
140	2.8886	0.01629	123.01	107.89	1014.1	1122.0	0.1984	1.6910	1.8894
150	3.718	0.01634	97.07	117.89	1008.2	1126.1	0.2149	1.6537	1.8685
160	4.741	0.01639	77.29	127.89	1002.3	1130.2	0.2311	1.6174	1.8485
170	5.992	0.01645	62.06	137.90	996.3	1134.2	0.2472	1.5822	1.8293
180	7.510	0.01651	50.23	147.92	990.2	1138.1	0.2630	1.5480	1.8109
190	9.339	0.01657	40.96	157.95	984.1	1142.0	0.2785	1.5147	1.7932
200	11.526	0.01663	33.64	167.99	977.9	1145.9	0.2938	1.4824	1.7762
210	14.123	0.01670	27.82	178.05	971.6	1149.7	0.3090	1.4508	1.7598
212	14.696	0.01672	26.80	180.07	970.3	1150.4	0.3120	1.4446	1.7566
220	17.186	0.01677	23.15	188.13	965.2	1153.4	0.3239	1.4201	1.7440
230	20.780	0.01684	19.382	198.23	958.8	1157.0	0.3387	1.3901	1.7288
240	24.969	0.01692	16.322	208.34	952.2	1160.5	0.3531	1.3609	1.7140
250	29.825	0.01700	13.821	216.48	945.5	1164.0	0.3675	1.3323	1.6998
260	35.429	0.01709	11.763	228.64	938.7	1167.3	0.3817	1.3043	1.6860
270	41.858	0.01717	10.061	238.84	931.8	1170.6	0.3958	1.2769	1.6727
280	49.203	0.01726	8.645	249.06	924.7	1173.8	0.4096	1.2501	1.6597
290	57.556	0.01735	7.461	259.31	917.5	1176.8	0.4234	1.2238	1.6472
300	67.013	0.01745	6.466	269.59	910.1	1179.7	0.4369	1.1980	1.6350
310	77.68	0.01755	5.626	279.92	902.6	1182.5	0.4504	1.1727	1.6231
320	89.66	0.01765	4.914	290.28	894.9	1185.2	0.4637	1.1478	1.6115
330	103.06	0.01776	4.307	300.68	887.0	1187.7	0.4769	1.1233	1.6002
340	118.01	0.01787	3.788	311.13	879.0	1190.1	0.4900	1.0992	1.5891

TABLE D-1b
Properties of Dry Saturated Steam (continued)
Temperature

Temp., °F	Abs press., psia	Specific volume		Enthalpy			Entropy		
		Sat liquid	Sat vapor	Sat liquid	Evap	Sat vapor	Sat liquid	Evap	Sat vapor
t	p	v_f	v_g	h_f	h_{fg}	h_g	s_f	s_{fg}	s_g
350	134.63	0.01799	3.342	321.63	870.7	1192.3	0.5029	1.0754	1.5783
360	153.04	0.01811	2.957	332.18	852.2	1194.4	0.5158	1.0519	1.5677
370	173.37	0.01823	2.625	342.79	853.5	1196.3	0.5286	1.0287	1.5573
380	195.77	0.01836	2.335	353.45	844.6	1198.1	0.5413	1.0059	1.5471
390	220.37	0.01850	2.0836	364.17	835.4	1199.6	0.5539	0.9832	1.5371
400	247.31	0.01864	1.8633	374.97	826.0	1201.0	0.5664	0.9608	1.5272
410	276.75	0.01878	1.6700	385.83	816.3	1202.1	0.5788	0.9386	1.5174
420	308.83	0.01894	1.5000	396.77	806.3	1203.1	0.5912	0.9166	1.5078
430	343.72	0.01910	1.3499	407.79	796.0	1203.8	0.6035	0.8947	1.4982
440	381.59	0.01926	1.2171	418.90	785.4	1204.3	0.6158	0.8730	1.4887
450	422.6	0.0194	1.0993	430.1	774.5	1204.6	0.6280	0.8513	1.4793
460	466.9	0.0196	0.9944	441.4	763.2	1204.6	0.6402	0.8298	1.4700
470	514.7	0.0198	0.9009	452.8	751.5	1204.3	0.6523	0.8083	1.4606
480	566.1	0.0200	0.8172	464.4	739.4	1203.7	0.6645	0.7868	1.4513
490	621.4	0.0202	0.7423	476.0	726.8	1202.8	0.6766	0.7653	1.4419
500	680.8	0.0204	0.6749	487.8	713.9	1201.7	0.6887	0.7438	1.4325
520	812.4	0.0209	0.5594	511.9	686.4	1198.2	0.7130	0.7006	1.4136
540	962.5	0.0215	0.4649	536.6	656.6	1193.2	0.7374	0.6568	1.3942
560	1133.1	0.0221	0.3868	562.2	624.2	1186.4	0.7621	0.6121	1.3742
580	1325.8	0.0228	0.3217	588.9	588.4	1177.3	0.7872	0.5659	1.3532
600	1542.9	0.0236	0.2668	610.0	548.5	1165.5	0.8131	0.5176	1.3307
620	1786.6	0.0247	0.2201	646.7	503.6	1150.3	0.8398	0.4664	1.3062
640	2059.7	0.0260	0.1798	678.6	452.0	1130.5	0.8679	0.4110	1.2789
660	2365.4	0.0278	0.1442	714.2	390.2	1104.4	0.8987	0.3485	1.2472
680	2708.1	0.0305	0.1115	757.3	309.9	1067.2	0.9351	0.2719	1.2071
700	3093.7	0.0369	0.0761	823.3	172.1	995.4	0.9905	0.1484	1.1389
705.4	3206.2	0.0503	0.0503	902.7	0	902.7	1.0580	0	1.0580



Mollier diagram for steam

EQUATION SHEET

$$Q = m\Delta h$$

$$Q = UA\Delta T$$

$$Q = mcp\Delta T$$

$$DNBR = \frac{Q_c}{Q_x}$$

$$P = P_o 10\ SUR(t)$$

$$P = P_o e^{t/T}$$

$$SUR = \frac{26.06}{T}$$

$$T = \frac{\beta - p}{\lambda p}$$

$$T = \frac{1^*}{p} + \frac{\beta - p}{\lambda p}$$

$$p = \frac{Keff-1}{Keff}$$

$$p = \frac{K_2 - K_1}{K_2 K_1}$$

$$\frac{CR1}{CR2} = \frac{1 - Keff_2}{1 - Keff_1}$$

$$RR = \Sigma f \phi th$$

$$SCR = \frac{S}{1-Keff}$$

$$M = \frac{CR_1}{Cr_o}$$

$$\ell^* = 10^{-5} \text{ sec}$$

$$A = \lambda N$$

$$\lambda = \frac{\ln 2}{t_{1/2}}$$

$$N = N_o e^{(-\lambda t)}$$

$$t_{1/2} = \frac{0.693}{\lambda}$$

$$R/_{hr} = \frac{6CEn}{d^2}$$

$$\lambda = 0.1 \text{ sec}^{-1}$$

$$q\ 1\text{-}2 = h_2 - h_1$$

$$q = h\, a \Delta t$$

Answer Sheet

5.0 - Theory of Nuclear Power Plant Operations, Fluids, and Thermodynamics.

5.1 (a) (1.0)
Ref: Fundamentals of Nuclear Reactor Engineering, Duke Power Company p. 19.

5.2 (a) (1.0)
Ref: Fundamentals of Nuclear - Reactor Engineering, Duke Power Company p. 156.

5.3 (d) (1.0)
Ref: Fundamentals of Nuclear Reactor Reactor Engineering, Duke Power Company, pp. 62, 65, 71, 117 and CNS Lesson Plans, NV System p. 23.

5.4 (a) (1.0)
Ref: Fundamentals of Nuclear Reactor Engineering, Duke Power Company, p. 117.

5.5 (b) (1.0)
Ref: Fundamentals of Nuclear Reactor Engineering, Duke Power Company, pp. 162, 163, 168.

5.6 (b) (1.0)
Ref: Fundamentals of Nuclear Reactor Engineering, Duke Power Company, pp. 136, 140.

5.7 (b) (1.0)
Ref: Fundamentals of Nuclear Reactor Engineering, Duke Power Company, p. 99.

5.8 (d) (1.0)
Ref: CNS Tech. Specs, Bases 3/4.1.1.

5.9 (b) (1.0)
Ref: CNS Tech Specs, Bases 3/4.4.9.

5.10 $P = Po\ e^{-t/\tau} \sim 80$ sec. (.5) (1.0)

Assumptions, Prompt drop to 10^{-6} amps P-6 is energized at 10^{-10} amps

$$10^{-10} = 10^{-6}\ e^{-t/80}$$

$$80 \ln \frac{10^{-10}}{10^{-6}} = -t \qquad [.5]$$

737 sec = t
12.3 min = t

Ref. CNS Exam Question bank, Theory p. 57.

5.11 (b) (1.0)

Ref. Reactor Operation Nuclear Energy Training, NVS Corporation, Section 12.3

5.12 Answer: <u>No</u> (.5) (1.0)

$$\frac{CR^0}{CR^1} = \frac{1-K^1}{1-k^0} = \frac{P^1}{P^0} \qquad \frac{30}{62} = .4839$$

$$\frac{37}{70} = .5286$$

Thumb Rule: Double counts; SDM halfed. (.5)

By looking at the inverse count rate ratio, it is obvious that the shutdown reactivity was decreased by approximately 50%.

By adding the same amount of reactivity again, the reactor would be critical.

Ref. CNS Exam Question Bank, Theory Section p. 70.

5.13 (a) (1.0)

Ref. 10 CFR 50.46 (b) (2)

5.14 (d) (1.0)

Ref. CNS FSAR Ch. 15.

5.15 (d) (1.0)

Ref. CNS Tech. Specs, Bases 3/4.2.1

5.16 (d) (1.0)

Ref. Heat Transfer, Thermo, & Fluid Flow Fundamentals, General Physics Corp. pp. 235-239.

5.17 (c) (1.0)
Ref. Thermodynamics, Kenneth Wark, p. 118.

5.18 (c) (1.0)
1000 psia, satvap h=1192.9
$h^1 = h^2$
20 psia α 300°F => 1191.4
20 psia α ? => 1191.4
20 psia α 350°F => 1215.4

$\frac{x}{50} = \frac{1.5}{24}$ => x = 3.125
=> 300°F + 3°F = 303°F

Ref. Thermo. Fluid Flow, & Heat Transfer for NPP, Duke Power Co., p D-16.

5.19 (a) (1.0)
Ref. Thermo, Fluid Flow, & Heat Transfer for NPP, Duke Power Company, p. 134

5.20 (c) Double speed => $(2)^2$ x head (1.0)
Ref. Heat Transfer, Thermo, & Fluid Flow Fundamentals, General Physics Corporation, pp. 322-324

5.21 (b) (1.0)
Ref. Heat Transfer, Thermo & Fluid Flow Fundamentals, General Physics Corporation pp. 165-176

5.22 (a) (1.0)
Ref. Heat Transfer, Thermo, & Fluid Flow Fundamentals, General Physics Corporation, p.183

5.23 (b) (1.0)
Ref. Heat Transfer, Thermo, & Fluid Flow Fundamentals, General Physics Corporation, p.331

5.24 (d) (1.0)
Ref. Thermo, Fluid Flow, Heat Transfer for Nuclear Power Plants, Duke Power Company, p.197

5.25 (d) (1.0)
Ref: Thermo, Fluid Flow, & Heat Transfer for NPP, Duke Power Company, p. 202.

6.0 Plant Systems Design, Control, and Instrumentation, Answers.

6.1 (b) (1.0)
Ref: CNS PSM, CN-IC-IPE-10, -17

6.2 (c) (1.0)
Ref: CNS PSM, CN-IC-IRE-22

6.3 (d) (1.0)
Ref: CNS PSM, CN-IC-ISE-6

6.4 (b) (1.0)
Ref: CNS PSM, UHI System, pp.2-3

6.5 (c) (1.0)
Ref: CNS PSM, CN-SYS-CM, pp.4-5.

6.6 (d) (1.0)
Ref: CNS PSM, CN-WS-WL-3.

6.7 (c) (1.0)
Ref: CNS PSM CN-SYS-WG, p.4

6.8 (c) (1.0)
Ref: CNS PSM, CN-SYS-FC-12

6.9 (a) (1.0)
Ref: CNS PSM, CN-IL-IDE-15

6.10 (b) (1.0)
Ref: CNS PSM, CN-EHC Runbook Logic Diagram

6.11 (a) (1.0)
Ref: CNS PSM, CN-SYS-ND p. 4

6.12 (b) or (c) (1.0)
Ref: CNS PSM, CN-CMP-NCP, p.2.

6.13 (c) (1.0)
Ref: CNS PSM, CN-IC-IFE-4

6.14 (a) (1.0)
Ref: CNS PSM, CN-IC-ISE

6.15 (c) (1.0)
Ref: CNS PSM, CN-IC-EDA-8

6.16 (a) (1.0)
Ref: CNS PSM CN-IC-ENB-4

6.17 (d) (1.0)
Ref: CNS PSM, CN-IC-ILE-22

6.18 (c) (1.0)
Ref: CNS PSM, CN-SYS-NC-1

6.19 (d) (1.0)
Ref: Nuclear Radiation Detection, 2nd Ed.,
W. J. Price, pp. 42-44.

6.20 (d) (1.0)
Ref: CNS PSM CN-IC-EMF p 5, -8, -9.

6.21 (c) (1.0)
Ref: CNS Lesson Plans, TSI, pp.4-5.

6.22 (d) (1.0)
Ref: CNS Lesson Plan, EQB p. 5 & CNS PSM CN-SYS-EQB.

6.23 (a) (1.0)
Ref: CNS PSM, Cn-SYS-EQC, p.41

6.24 (b) (1.0)
Ref: CNS Lesson Plans EQP, p 8.

6.25 (b) (1.0)
Ref: CNS PSM, CNS-SYS-CA, p 41.

7.0 Procedures - Normal, Abnormal, Emergency, and Radiological Control . Answers.

7.1 (a) (1.0)
Ref: CNS OP/1/A/6100/01, Enclosure 4.1. ¶ 2.12

7.2 To prevent SI (.5) on Lo steam pressure (.5) (1.0)
Ref: CNS OP/1/A/6100/01, Enclosure 4.1, ¶ 2.54.2

7.3 A: any 3 of (1.5)
1) all TC <200°F (.5)
2) all NC loop temperatures <160°F (.5)
3) all SG depress via shell temperature <200°F (.5)
4) NC H_2 degas is complete (.5)

Ref: CNS OP/1/A/6100/02 Enclosure 4.1, ¶ 2.68 - 2 69.

7.4 (b) (1.0)
Ref: CNS OP/1/A/6100/02, Enclosure 4.1, ¶ 2.63.3

7.5 By adjusting INV-857 (PZR aux spray ctrl) (.5) (1.0)
Cycling PZR Heaters as required (.5)
Ref: CNS OP/1/A/6100/02, Enclosure 4.1 ¶ 2.72.

7.6 1) P-10; (.25) (Nuclear at power) On (.25) (1.5)
2) P-13; (.25) (Turbine not at power), Off (.25)
3) P-7; (.25) (Lo power Rx Trips Blocked) Off (.25)

Ref: CNS OP/1/A/6100/01, Enclosure 4.1, ¶ 2.85.

7.7 Unanticipated neutron flux level increase (.5) (1.0)
S/R Hi flux level at shutdown annunciator (.5)

Ref: CNS AP/1/A/5500/13 p.1.

7.8 (d) or (c) (1.0)
Ref: CNS AP/1/A/5500/20, p 2.

7.9 1) Verify affected train D/G Running (.5) (1.5)
2) Verify B/O load sequencer actuated (.5)
3) Ensure CA pump #1 running (steam driven) (.5)

Ref: CNS AP/1/A/5500/07 p 2.

7.10 1) NC Pump thermal barrier (.5) (1.5)
2) NV Letdown heat exchanger (.5)
3) NV Excess letdown heat exchanger (.5)
4) ND Heat Exchanger (.5)
5) NCDT Heat Exchanger (.5)

Ref: AP/1/A/5500/10 ¶ C. and CNS PSM, Vol. 1, CNS-SYS-KC, ¶ V.Z.

7.11 Verify Control Rod Urgent Failure does not exist or Verify that a logic cabinet failure does not exist. (.5)

Ref: CNS AP/1/A/5500/15 p 2., CNS OP/1/B/6100/10C p.2.

7.12
1) Sound containment evacuation alarm (.5) (1.5)
2) Verify waste gas release terminated (.5)
3) Ensure containment purge stopped (.5)

Ref: CNS AP/1/A/5500/25

7.13
1) PZR level >5% (.5) (1.5)
2) NC subcooling >50°F (.5)
3) NC pressure stable or increasing (.5)
4) Secondary heat sink available

Ref: CNS EP/1/A/5000/1B p 2., EP/1/A/5000/01 ¶ 15.d.

7.14
(a) Rx trip switchgear (pent. Room 594, cc-51) (.5) (1.0)
(b) Train A and Train B meter reading in P-4 position: 0 VDC (.5)

Ref: CNS EP/1/A/5000/01, Enclosure 2.

7.15 Any three of: (1.5)
1) W/R NC T-Hot <200°F (.5)
2) S/G Pressure : zero (.5)
3) All CRD vent fans running (.5)
4) Rx vessel upper range level >97% and stable (.5)

Ref: CNS EP/1/A/5000/1A1

7.16 (1.0)
1) VQ isolation valves shut (cont air pressure)
2) VP isolation valves shut (cont purge)
3) Containment air return is operation (vx)
4) Hydrogen skimmer in operation (ux)
5) Hydrogen ignitor available
6) Hydrogen recombiner available
7) YV system in operation (containment ventilation)
8) Head vent and block valves energized

Any 4 α .25 each.

Ref: CNS EP/1/A/5000/2F3.

7.17 (a) (1.0)
Ref: CNS EP/1/A/5000/1E2

7.18 (b) (1.0)
Ref: 10 CFR 20, ¶ 20.101

7.19 (b) (1.0)
Ref: CNS 4p Procedural Guide II-5 3.j.

7.20 (b) (1.0)
Ref: CNS HP Procedural Guide II-25.

7.21 (c) (1.0)
Ref: CNS RP/0/A/5000/15, ¶ 3.2.2.

7.22 (a) (1.0)
Ref: CNS HP Manual, Section I ¶ E. H.

8.0 Administrative Procedures, Conditions, and Limitations. Answers.

8.1 (d) or (a) (1.0)
Ref: CNS Technical Specifications ¶ 3.2.1 & 4.2.1

8.2 (c) (1.0)
Ref: CNS Technical Specifications ¶ 3.1.1.1.

8.3 (b) (Partial Credit given for calculation) (1.0)
Ref: CNS Technical Specifications, 4.0.2

8.4 (a) (1.0)
Ref: CNS Technical Specifications ¶ 3.1.1.4, 3.1.2.4, 3.5.1.1, 3.7.3.

8.5 (d) (1.0)
Ref: CNS Technical Specifications, 3.8.1.1 & CNS PSM CN-SYS-EPQ p.1

8.6 (b) (1.0)
Ref: CNS Technical Specifications, Table 3.3-3, Action 14.

8.7 (d) (1.0)
Ref: CNS Technical Specifications, ¶ 3.4.2.1 & 3.4.2.2

8.8 (c) (1.0)
Ref: CNS Technical Specifications, ¶ 3.4.6.1

8.9 (d) (1.0)
Ref: CNS Technical Specifications, 3.5.4

8.10 Question Deleted.

8.11 (d) (1.0)
Ref: CNS Technical Specifications, ¶ 6.2.2.

8.12 (b) (1.0)
Ref: CNS Station Directive 3.1.19, ¶ 5.1 & 5.2

8.13 (a) (1.0)
Ref: CNS Station Directive 4.4.3, ¶ 5.1

8.14 (c) (1.0)
Ref: CNS Station Directive, ¶ 2.11.4 & ¶ 5.1

8.15 (c) (1.0)
Ref: CNS Station Directive, ¶ 2.9.

8.16 (c) (1.0)
Ref: CNS Station Directive 3.1.1 ¶ 6.11.1.

8.17 (c) (1.0)
Ref: CNS Station Directive 3.1.2, ¶ 5.1

8.18 (c) (1.0)
Ref: CNS Station Directive 3.1.17.

8.19 (a) (1.0)
Ref: CNS OMP 2-28, 2-29, 2-30

8.20 (c) (1.0)
Ref: CNS OP/1/A/6700/01, Section 1.3.

8.21 (a) (1.0)
Ref: CNS OMP 2-20, ¶ 3.1.B.

8.22 (c) (1.0)
Ref: CNS OMP 1-4, p. 7.

8.23 (d) (1.0)
Ref: CNS OMP 2-9, p.5

8.24 (c) (1.0)
Ref: CNS Directive 3.1.2 p. 5.3.1

8.25 (c) (1.0)
Ref: CNS Directive 3.8.6, ¶ 2.8.