ENCLOSURE 1

EXAMINATION REPORT - 50-327/OL-85-01

Facility Licensee: Tennessee Valley Authority
500A Chestnut Street
Chattanooga, TN 37401

Facility Name: Sequoyah Nuclear Plant

Facility Docket No.: 50-327

Written examinations were administered at Sequoyah near Soddy Daisy, Tennessee.

Chief Examiner: W G Douglas 06/19/85
W. G. Douglas Date Signed

Approved by: Bruce A. Wilson 6/21/85
Bruce A. Wilson, Section Chief Date Signed

Summary:

Examinations on May 20-23, 1985

Written and oral examinations were administered to 13 candidates; 11 of whom passed. An oral examination was administered to one candidate who passed. A written examination of all categories was administered to one candidate who did not pass.

REPORT DETAILS

1. Facility Employees Contacted:

*C. S. Benton, Unit Supervisor, Simulator Section
*C. O. Brewer, Training Manager
*L. C. Bush, Operations - Assistant Group Head
*V. E. Keyser, Instructor
*B. C. Lake, Training Shift Engineer
M. J. Lorek, Instructor
*L. M. Nobles, Superintendent (O&E)
*C. H. Noe, Supervisor, Operator Training
*W. G. Payne, Instructor
*L. H. Sain, NTB

*Attended Exit Meeting

2. Examiners:

*W. G. Douglas, USNRC, Region II
F. S. Jagger, EG&G
A. J. Vinnola, EG&G

*Chief Examiner

3. Examination Review Meeting

At the conclusion of the written examinations, the examiners met with V. E. Keyser, B. C. Lake, M. J. Lorek, and W. G. Payne to review the written examination and answer key. The following comments were made by the facility reviewers:

a. SRO Exam

(1) Question 5.02

Facility Comment: Depending upon whether cycle 1 or cycle 3 is assumed, more than one curve is correct.

NRC Resolution: Agree with facility comment. Based on cycle 3 information available at review, answer b or d will be accepted for Part 1.

(2) Question 5.07.b

Facility Comment: Answer is TRUE for transient, but FALSE if new steady state is achieved.

NRC Resolution: Agree with facility comment. Part b deleted from examination.

(3) Question 5.12

NRC Resolution: Point value changed from 0.75 to 1.0 to make it consistent with other multiple choice questions.

(4) Question 5.26

Facility Comment: Answer is TRUE if cycle 1 is assumed and FALSE if cycle 3 is assumed.

NRC Resolution: Verified comment using Plant Curve Book, TI-28. Question deleted from examination.

(5) Question 6.02

Facility Comment: There are two possible correct answers for this question.

NRC Resolution: Agree with facility comment. Answers c and d verified as correct using Systems Manual, Chapter 3 and GOI-3C as reference. Answer key changed to accept c or d as correct.

(6) Question 6.16

NRC Resolution: Answer key was incorrect due to "typo". Answer key changed to accept d as correct answer.

(7) Question 6.19

NRC Resolution: Review of stated reference discovered a third possible answer. Answer key changed to accept any two for full credit.

(8) Question 7.01

Facility Comment: Answer a is also correct.

NRC Resolution: Verified answer a as correct using SOI-68.2. Answer key changed to accept a or b as correct.

(9) Question 7.11

Facility Comment: FHI-7 has been recently revised. The correct answer is 4 fuel assemblies which is not one of the four choices.

NRC Resolution: Revision was pointed out by examinee during examination. Instructions were given to disregard choices and put answer for revision to FHI-7. Answer key changed to accept 4 as correct.

(10) Question 7.17

Facility Comment: The answer to part a should be FALSE.

NRC Resolution: Agree with facility comment. Using listed reference, verified correct answer as FALSE. Answer key changed accordingly.

(11) Question 8.12

Facility Comment: Containment spray actuation is not a reactor trip signal.

NRC Resolution: Agree with facility comment. Answer key changed to required two answers at 0.5 points each. The total value of question was reduced to 1.0 point.

b. RO Exam

(1) Question 1.02 - See SRO question 5.07.b.

(2) Question 1.09 - See SRO question 5.02.

(3) Question 1.16.c

Facility Comment: Depending upon value of beta fraction assumed, answer could either be supercritical or prompt critical.

NRC Resolution: Agree with facility comment. Answer key changed to accept 3 or 4 as correct answer for part c.

(4) Question 2.04

NRC Resolution: Provided reference contained incorrect information. Using RCS system description, verified answer c as correct. Answer key changed to accept c as correct.

(5) Question 2.21

Facility Comment: Answer a is correct and answer d is incorrect.

NRC Resolution: Agree with facility comment. Verified, using stated reference and facility supplied handout. Answer key changed to accept answer a as correct.

(6) Question 2.22

Facility Comment: Calculation of AFW load using design data indicates 5% of full load may be maintained.

NRC Resolution: Stated references says 3%. However, calculation using data in stated reference yields about 5%. Answer key changed to accept b or c for full credit.

(7) Question 3.07.b

Facility Comment: The answer should be FALSE.

NRC Resolution: Using stated reference, verified the nonlinear gain is controlled by temperature error. The answer key is changed to accept FALSE as the correct answer.

(8) Question 3.12

NRC Resolution: Using stated reference, verified answers c and d as correct. Answer key changed to accept c or d for full credit.

(9) Question 3.21.d

NRC Resolution: Delta T does not directly affect the OPΔT setpoint. However, if it is logically assumed the delta T increased due to power increase which causes a Tavg increase, then the OPΔT setpoint would reduce. Since two of the three answers can be correct, this part of the question is deleted from the examination.

(10) Question 3.23

NRC Resolution: Answer key for part c was incorrect. Based on the stated reference, the correct answer is FALSE. Answer key changed accordingly.

(11) Question 4.15

Facility Comment: SOI says 200 degrees, AOI says 225 degrees. Both answers c and d should be accepted.

NRC Resolution: The RCP must be secured at the most resistive requirements, 200 degrees. The answer key is not changed.

(12) Question 4.19

Facility Comment: Parts c and d are not alarms or indications.

NRC Resolution: Agree with facility comment. Parts c and d deleted and parts a and b are worth 0.5 points each.

4. Exit Meeting

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

There were two generic weaknesses noted during the oral examinations. The first was in the area of radiation protection. The examinees were unable to adequately describe the detectors used for the various survey instruments. Also, they were unable to explain the sources of radiation (tritium, N-16, secondary activity) in the plant. The other area of generic weakness was in using Tavg and ΔT indications to figure out whether (and how) T_{cold} or T_{hot} had failed.

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.

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

ENCLOSURE 3

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: SEQUOYAH 1&2

REACTOR TYPE: PWR-WEC4

DATE ADMINISTERED: 85/05/20

EXAMINER: JAGGAR, F.

APPLICANT: MASTER COPY

INSTRUCTIONS TO APPLICANT:

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

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
30.00	25.00			1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
30.00	25.00			2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
30.00	25.00			3. INSTRUMENTS AND CONTROLS
30.00	25.00			4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
120.00	100.00			TOTALS

FINAL GRADE ________________%

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

____
APPLICANT'S SIGNATURE

QUESTION 1.01 (1.00)

Which of the following is TRUE concerning RCS operation following the loss of one Reactor Coolant Pump?

a. Core coolant velocity decreases therefore the flowrate in the remaining loops decreases.

b. Flow to the vessel from the remaining three pumps is less than 3/4 of the original flow.

c. Flow in the idle loop bypasses the core.

d. Since only thr S/G's are providing steam, the steam pressure and temperature in the remaining S/G's is increased.

QUESTION 1.02 (1.00)

TRUE or FALSE?

a. During 100% power operation, Departure from Nucleate Boiling Ratio (DNBR) is greater than the DNBR for 20% reactor power.

b. Increasing pressure of the RCS, when operating in the nucleate boiling region of the heat transfer curve will decrease the heat transfer rate (BTU/hr-square foot).

QUESTION 1.03 (1.50)

TRUE or FALSE?

a. The faster a centrifugal pump rotates, the greater the NPSH required to prevent cavitation.

b. One of the pump laws for centrifugal pumps states that the volume flow rate is inversely proportional to the speed of the pump.

c. Pump runout is the term used to describe the condition of a centrifugal pump running with no volume flow rate.

QUESTION 1.04 (2.00)

What is the most significant type of heat transfer (conduction, convection, or radiation) taking place under each of the following conditions? Consider each condition separately.

1. Nucleate boiling on the cladding surface.

2. Accident condition in which coolant is boiled and converted to steam in the reactor vessel.

3. Heat from fission thru a fuel pellet.

4. Decay heat removal by natural circulation.

QUESTION 1.05 (2.00)

Indicate how the following will affect Unit efficiency (increase, decrease, no change) at a steady state power level. (Consider each case separately.)

a. Absolute condenser pressure changes from 1 psi to 1.25 psi.

b. Total S/G blowdown is changed from 25 gpm to 40 gpm.

c. Condenser hotwell temperature changes from 125 F to 100 F. (Assume no change in condenser pressure.)

d. Steam quality changes from 99.3% to 99.7%.

QUESTION 1.06 (.50)

TRUE or FALSE?

The RHR miniflow valves prevent cavitation of the RHR pumps by opening at 550 gpm and closing at 1250 gpm.

QUESTION 1.07 (1.00)

When saturated steam is throttled, the down stream fluid:

a. loses energy as it expands.

b. DECREASES in pressure, however, the temperature may INCREASE.

c. may or may not be saturated depending on the upstream conditions.

d. has a higher mass flow rate than the upstream steam.

QUESTION 1.08 (2.00)

a. If the reactor is operating in the power range, how long will it take to raise power from 20% to 40% with a +0.5 DPM Start-up rate?

1. 12 sec.

2. 21 sec.

3. 36 sec.

4. 54 sec. (1.0)

b. How long will it take to raise power from 40% to 60% with the same +0.5 DPM Startup rate?

1. 12 sec.

2. 21 sec.

3. 36 sec.

4. 54 sec. (1.0)

QUESTION 1.09 (1.00)

Match the curves, on Fig A.4, with the following plant descriptions. Put your answers on your answer paper. i.e. "Curve 4 - g".

1. Beginning of life (BOL) - 0% power.

2. BOL - 100% power.

3. End of life (EOL) - 0% power.

4. EOL - 100% power.

QUESTION 1.10 (1.00)

Which of the following statements is TRUE?

a. It is NOT possible for the Moderator Temperature Coefficient (MTC) to ever become positive at the Sequoyah plant.

b. It is possible for the MTC to become positive, but ONLY when the reactor is in Mode 5.

c. If the MTC is positive, while the reactor is in Mode 2, Technical Specifications must be consulted because there are action statements that must be followed.

d. MTC can be positive in an under moderated core where the moderator to fuel ratio is less than the optimum value.

QUESTION 1.11 (1.00)

Which of the following best describes the effect on MTC if the RCS temperature is LOWERED?

a. It becomes less negative because boron and water molecules are swept into the core as a result of the outsurge from the pressurizer, therefore, neutrons spend more time in the resonance region.

b. It becomes less negative because the rate of change in the density of water per degree temperature change is less at lower temperature which causes a lesser change in rate in resonance escape probability.

c. It becomes more negative because thermal utilization increases and resonance escape probability decreases.

d. It becomes more negative because as temperature is lowered the moderator becomes more dense, this increases the amount of water molecules in the core therefore neutrons have a greater probability of colliding with a water molecule and this is an increased negative reactivity effect.

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

QUESTION 1.12 (4.00)

Using the attached Xenon worth curve, Fig. 1.1, answer the following.

a. Power at T0 was at 70%. What was the power level between T1 and T2?
   1. 90%

   2. 50%

   3. 20%

   4. 10% (1.0)

b. What was the length of time between T2 and T3?
   1. 1 hour

   2. 3 hours

   3. 8 hours

   4. 12 hours (1.0)

c. What happened at T2?
   1. Reactor tripped.

   2. Rods were placed in AUTO, and turbine power was raised to 100%.

   3. Reactor power was reduced to 10%.

   4. Turbine power remained constant, rods were in manual and inserted 50 steps and the steam dump valves failed open (10% of rated power). (1.0)

d. At time T4 ...
   1. All Xenon production has stopped.

   2. Iodine decay to Xenon has stopped.

   3. All Xenon production remains constant, but the burnout increases.

   4. Xenon production directly from fission has stopped, but Xenon production from decay Iodine continues. (1.0)

QUESTION 1.13 (.50)

TRUE or FALSE?

After an extended outage of 1 year, the Secondary Neutron Source is unable to release any neutrons and therefore there are insufficient neutrons available to start a chain reaction in the core.

QUESTION 1.14 (1.00)

Delayed neutrons play a major role in the operation of the core because they .

a. are born at (thermal) slow energy levels (less than 1 ev) and therefore are more apt to cause a fission as compared to being absorbed by a poison.

b. are considered as epithermal neutrons and therefore they will not travel far enough to leak out of the core.

c. are born so much later than the prompt neutrons and provide controlability during steady state operations and power transients.

d. provide 70% of the fission neutron inventory and have higher importance factors associated with them as compared to prompt neutrons.

QUESTION 1.15 (.50)

TRUE or FALSE?

Shortly after a reactor trip from the power range the reactor has a -80 sec. period because the mean life of the longest lived delayed neutron precursor group is 80.7 seconds which corresponds to a 55.9 sec. half life.

QUESTION 1.16 (3.00)

If the Source Range (SR) instruments indicate 50 cps with Keff equal to 0.9, what would the SR instrument indicate if rods were withdrawn to bring Keff equal to 0.95? Assume BOL conditions.

a. 1. 50 cps

2. 75 cps

3. 100 cps

4. 200 cps (1.0)

b. How much reactivity was added?

1. 0.0347

2. 0.0500

3. 0.0526

4. 0.0585 (1.0)

c. If the same amount of reactivity were added again, what would be the state of the reactor?

1. Sub-critical.

2. Critical.

3. Super-critical.

4. Prompt-critical. (1.0)

QUESTION 1.17 (1.50)

Refer to the attached graphs B.6.2 and B.7.1.

a. Which graph represents Total Power Coefficient? (0.5)

b. Fill in the blanks on the right side of each graph with BOL or EOL. Tear graphs from the exam and include with your answer sheets. (1.0)

QUESTION 1.18 (2.00)

Refer to TI-28, Fig. B.2.c attached to answer the following.

a. The reason the curve changes slope at the point marked "a" is because:

1. this is the point where the Power Range lower detector is no longer able to detect thermal neutrons and the rods have not moved high enough to allow the Power Range upper detector to detect thermal neutrons.

2. the rods are moving through the area that has the highest thermal neutron flux.

3. bank "B" rods have just stopped moving, while bank "C" rods continue to be withdrawn, therefore, only one bank of rods is moving instead of two banks just prior to point "a".

4. there are two effects here. One, bank "A" rods have just stopped moving, while bank "B" rods continue to be withdrawn, two, just prior to point "a", bank "A" rods were moving through an area (top of the core) that has a relatively low concentration of thermal flux. (1.0)

b. The reason for the positive slope on the curve marked "b" is because:

1. the bank "D" rods are moving into an area that has an increasing thermal flux concentration.

2. in addition to bank "C" rods moving through an area of increasing concentration of thermal flux, now bank "D" is also moving. i.e. overlap.

3. bank "C" has just stopped moving and now bank "D" rods are moving into an area of the core that has a decreasing Xenon concentration.

4. bank "D" is moving through the uppermost part of the core where the thermal flux is decreasing, therefore, the competition for neutrons is greater and rod worth is higher. (1.0)

QUESTION 1.19 (.75)

Compare the calculated Estimated Critical Position (ECP) for a startup 15 hours after a trip to the actual Critical Rod Position (ACP) if the following events/conditions occurred. Consider each independently. Limit your answer to:

a. ACP higher than ECP.

b. ACP lower than ECP.

c. ACP would not be significantly different than ECP.

1. One Reactor Coolant Pump is stopped one minute prior to criticality.

2. The steam dump pressure setpoint is increased to a value just below the code safties setpoints.

3. The startup is delayed 2 more hours.

QUESTION 1.20 (.75)

For each condition in COLUMN A find the correct heat transfer equation in COLUMN B that would be used to calculate the heat transferred.

COLUMN A	COLUMN B
a. Across the reactor (cold leg to hot leg)	1. $\dot{Q}$ = UA delta T
	2. $\dot{Q} = \dot{M}$ delta T
b. Across S/G U-tubes (primary to secondary)	3. $\dot{Q} = \dot{M}$Cp delta T
c. Across S/G secondary side (feedwater to steam)	4. $\dot{Q} = \dot{M}$ delta H
	5. $\dot{Q}$ = UA delta H

QUESTION 1.21 (1.00)

TRUE or FALSE?

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

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

QUESTION 1.22 (1.00)

Steam exiting the HP turbine is at 785 psig, 90% quality. Steam entering the LP turbine is superheated to 100 F. What is the enthalpy change of the steam?

a. 85 BTU/lbm

b. 140 BTU/lbm

c. 154 BTU/lbm

d. 705 BTU/lbm

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

QUESTION 2.01 (1.00)

The order of transfer of power sources to the 6.9 kv Shutdown Board 1B is:

a. From 6.9 kv Unit Board 1D to 6.9 kv Unit Board 1C to Emergency Diesel Generator.

b. From 6.9 kv Unit Board 1B to 6.9 kv Unit Board 1D to Emegency Diesel Generator.

c. From 6.9 kv Unit Board 1D to 6.9 kv Unit Board 1B to Emergency Diesel Generator.

d. From 6.9 kv Unit Board 1C to 6.9 kv Unit Board 1D to Emergency Diesel Generator.

QUESTION 2.02 (.50)

TRUE or FALSE?

Following the receipt of an emergency start signal, diesel generator speed and voltage may be manually adjusted from the control room.

QUESTION 2.03 (1.00)

The 125 vdc vital batteries are sized to supply the dc power required to maintain the plant in a safe shutdown condition for______ hours.

a. 0.5

b. 1.0

c. 1.5

d. 2.0

QUESTION 2.04 (1.00)

Which of the following best describes the Reactor Coolant Pump Thermal Barrier and Heat Exchanger?

a. Allows a controlled amount of relatively cool water to enter the seal section of the pump.

b. Prevents RCS water entering the lower radial bearing and the seal section of the pump.

c. Component Cooling Water flows through the cooling coils.

d. During a loss of seal injection, minimizes the amount of hot RCS water into the lower radial bearing.

QUESTION 2.05 (1.00)

Match the following parameters in COLUMN A to their values in COLUMN B.

COLUMN A	COLUMN B
a. Safety injection pump normal discharge pressure (psig).	1. 600
	2. 925
b. RHR pump normal discharge pressure (psig).	3. 1250
c. Minimum volume of cold leg accumulator (cu. ft.)	4. 1500
	5. 1800
d. Volume of UHI accumulator (cu. ft.)	

QUESTION 2.06 (1.00)

Which of the following statements is TRUE concerning the use of the Residual Heat Removal system?

a. Used during plant startup in conjunction with the CVCS to equalize boron concentration between the RCS and pressurizer.

b. Used during plant cooldown in conjunction with the CVCS to equalize boron concentration between the RCS and pressurizer.

c. Used during plant cooldown as an alternate letdown flow path.

d. Used during plant heatup as an alternate letdown flow path.

QUESTION 2.07 (1.00)

Which of the following is the preferential order of valve operation for Emergency Boration?

a. ~~F~~HCV-62-929 (Emergency borate manual valve), FCV-62-138 (Emergency borate MOV), 62-135 & 6 (RWST suction valves), BIT injection valves.

b. 62-135 & 6, BIT injection valves, ~~F~~HCV-62-929, FCV-62-138.

c. FCV-62-138, ~~F~~HCV-62-929, BIT injection valves, 62-135 & 6.

d. FCV-62-138, ~~F~~HCV-62-929, 62-135 & 6, BIT injection valves.

QUESTION 2.08 (.50)

TRUE or FALSE?

An interlock of the Manipulator Crane hoist drive circuit in the up direction permits the hoist to be operated only when the open indicating switch on the gripper is actuated.

QUESTION 2.09 (1.00)

Which of the following is a component supplied by a Component Cooling Water System Safeguards Train pump?

a. Non-regenerative heat exchanger.

b. RHR heat exchanger.

c. RCP thermal barrier heat exchanger

d. Reciprocating charging pump.

QUESTION 2.10 (1.00)

Which of the following will cause a trip of a running Main Feedwater Pump?

a. Low feedwater temperature.

b. Low Main Feed Pump turbine speed.

c. Recirculation valve open.

d. Safety Injection.

QUESTION 2.11 (2.00)

State whether the following statements are TRUE or FALSE concerning the Upper Head Injection System (UHI).

a. The membrane in the Nitrogen pipe between the water and nitrogen UHI accumulator prevents nitrogen absorbtion in ANY of the water in the UHI system.

b. The membrane prevents the injection of nitrogen into the RCS following UHI actuation.

c. The UHI system operates during a LOCA which is greater than the capacity of the charging or SI pumps.

d. The UHI system can be isolated from the RCS by closure of their motor operated valves.

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

QUESTION 2.12 (1.00)

Which of the following is a purpose of the atmospheric relief valve on the main steam line?

a. Will actuate during a trip from full load while preventing the lifting of the safety valves.

b. Used as a dummy load during plant startup.

c. Used as a steam dump when the condenser is unavailable.

d. Used to control cooldown rate during normal plant cooldown.

QUESTION 2.13 (1.00)

What signal inputs are used to control the position of the governor on the Turbine Driven Auxiliary Feed Pump?

a. Turbine speed and pump discharge flow.

b. Turbine speed and pump discharge pressure.

c. Steam pressure and pump discharge pressure.

d. Steam flow and pump discharge flow.

QUESTION 2.14 (1.00)

What is the normal operating speed of the Emergency Diesel Generator?

a. 800 RPM

b. 850 RPM

c. 900 RPM

d. 950 RPM

QUESTION 2.15 (1.00)

Match the following Emergency Diesel Generator speeds to the event that occurs as the generator is started.

a. 40 RPM	1. Engine running alarm.
b. 200 RPM	2. Field flash.
c. 550 RPM	3. Opens ERCW valve to Jacket Water heat exch.
d. 850 RPM	4. Diesel muffler room exhaust fan starts.
	5. Diesel engine lube oil pump starts.

QUESTION 2.16 (1.00)

When operating the RHR Heat Exchanger outlet flow control valves, FCV-74-16 & FCV-74-28 from the Unit-1 and Unit-2 control rooms, state if the following are TRUE or False.

a. Reset FCV-74-16 counter-clockwise (to the left) on Unit-1.
b. Reset FCV-74-28 counter-clockwise (to the left) on Unit-2.

QUESTION 2.17 (1.00)

The purpose of the Reactor Coolant Pump (RCP) Seal standpipe is to provide a:

a. final collection point for the #3 seal leakoff.

b. lubricating water supply for #3 seal.

c. final collection point for the #2 seal leakoff.

d. pressure head for #3 seal.

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

QUESTION 2.18 (1.00)

The purpose of the interlock that prevents the letdown isolation valves from opening or shutting unless all three orifice isolation valves are shut is to prevent:

a. exceeding design flow rates of the demineralizers.

b. excessive heatup rates across the regen. heat exchanger.

c. flashing of water on the shell of the regen. heat exchanger.

d. unnecessary lifting of relief valves downstream of orifices.

QUESTION 2.19 (1.00)

What are TWO reasons for maintaining a minimum spray bypass flow to the pressurizer? Choose only one of the following combinations.

a. 1. Prevent excessive cooling to the surge line.
   2. Reduce the delta pressure across the spray valves.

b. 1. Reduce thermal shock to the spray nozzle.
   2. Ensure that the backup heaters cycle on.

c. 1. Prevent excessive cooling to the spray line.
   2. Equalize boron between pressurizer and the RCS.

d. 1. Minimize stress to the surge line thermal sleeve.
   2. Remove gases from the RCS.

QUESTION 2.20 (1.00)

The Condensate Storage Tank minimum water volume of 190,000 gallons is sufficient to maintain the plant in hot standby for how many hours?

a. 2 hours

b. 4 hours

c. 6 hours

d. 8 hours

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

QUESTION 2.21 (1.00)

Which action below is taken to protect the Motor Driven Auxiliary Feedwater Pumps from a runout condition?

a. Flow orifices in the discharge line of each pump.

b. Trip of motor breaker.

c. Automatic closure of the flow control valves.

d. Recirculation of the excess flow to CST.

QUESTION 2.22 (1.00)

When ALL the Auxiliary Feedwater Pumps are used for normal plant startup, what approx. percentage of full load can be maintained?

a. 1

b. 3

c. 5

d. 7

QUESTION 2.23 (1.00)

Which of the following provides a direct input to cause the feedwater isolation valves to close automatically?

a. Reactor trip from 100% power.

b. Steam Generator Hi Hi level.

c. Steam Generator Hi level.

d. Both Main Feed pumps trip.

QUESTION 2.24 (1.00)

The Component Cooling Water system in conjunction with the RHR system is designed to reduce the RCS temperature to _____ F within _____ hours after shutdown.

a. 350, 16

b. 120, 24

c. 250, 18

d. 140, 16

QUESTION 2.25 (2.00)

For the following components, indicate whether they will receive an OPEN, CLOSE, or NO signal as a result of a safety injection initiation signal (with Phase "A").

a. Control room supply ducts
b. Main feed bypass valves
c. SI accumulator discharge isolation valves
d. Normal charging header isolation valves
e. Main steam isolation valves
f. RWST to SI pump suction valves
g. Seal water return isolation valve
h. Component cooling isolation valve from RHR system
i. Component cooling isolation from letdown heat exchanger
j. Steam supply valves to turbine-driven feed pump

QUESTION 2.26 (1.00)

Reactor Coolant Pump #3 is started with the plant in Mode 5, after its seal has been replaced. After operating for approximately 20 minutes the following is observed.

1. #1 seal delta P >390#
2. Standpipe low level
3. #1 seal leakoff has increased

ASSUME:

1. Plant pressure is at 400#
2. Seal injection at 6 gpm.

Which of the following is a probable cause for the abnormal indications?

a. #3 seal failure.

b. VCT pressure is low.

c. #1 seal bypass is closed.

d. RCDT pressure has increased.

QUESTION 2.27 (.50)

TRUE or FALSE

The automatic switchover of the charging pump's suction from the VCT to the RWST is designed to maintain proper seal injection flow to the RCP's.

QUESTION 2.28 (1.00)

Which of the following components served by the Component Cooling Water System (CCW) would NOT cause an increase in both CCW surge tank level and CCW radiation monitor indication if a failure of the component were to occur?

a. Reactor Coolant Pump bearing coolers.

b. RHR pump seal coolers.

c. Seal Water Heat Exchanger.

d. Spent Fuel Pit Heat Exchanger.

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

QUESTION 2.29 (.50)

TRUE or FALSE?

The Reheat Stop Valves close on a turbine trip to control turbine overspeed due to expansion of entrained steam in the MSR's through the low pressure turbine.

QUESTION 2.30 (1.00)

If an unsaturated bed of H-OH resin is placed in service, what will be the result?

a. RCS Oxygen concentration will increase.

b. No ion exchange will occur for the first 12 hours of operation.

c. RCS Boron concentration will decrease.

d. RCS Boron concentration will increase.

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

QUESTION 3.01 (1.00)

Which of the following is true concerning the operation of the Reactor Trip (RT) and Reactor Trip Bypass (BY) Breakers?

a. The Train B trip signal trips RTB and BYB.

b. To allow testing of the RT's, BOTH BY's may be closed while the reactor is at power.

c. Tripping is accomplished by an undervoltage relay, normally held open by 48 volt dc power from the logic panels.

d. The Train A trip signal trips RTA and BYB.

QUESTION 3.02 (1.00)

Which of the following Reactor Protection signals is actuated by a 2/4 coincidence?

a. Pressurizer Low Pressure.

b. Pressurizer High Level.

c. Low RCS Flow.

d. RCP Low Voltage.

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

QUESTION 3.03 (1.00)

Match the following permissives in COLUMN A with their function in COLUMN B. (i.e. e. 6)

COLUMN A	COLUMN B
a. P-6	1. Below setpoint allows operation with one RCP off.
b. P-9	2. Below setpoint blocks reactor trip due to turbine trip.
c. P-10	3. Allows manual block of high steam flow SI and blocks steam dumps below setpoint.
d. P-12	4. Allows manual block of SRM trip above setpoint.
	5. At setpoint allows manual block of IRM and PRM low setpoint trips and rod stops and removes P-7.

QUESTION 3.04 (1.00)

Which of the following is a correct statement concerning ECCS equipment operation?

a. A Safety Injection signal starts the diesel generators and aligns them to the shutdown boards.

b. A Safety Injection signal starts the charging pumps, SI pumps, RHR pumps and Containment Spray pumps.

c. A Safety Injection signal opens the valves in the boron injection tank recirculation lines.

d. A Safety Injection signal will initiate a reactor trip, isolate the feedwater system, and start the auxiliary feedwater pumps.

QUESTION 3.05 (1.50)

On the attached pressurizer level control diagram, Figure 3.1, fill in the setpoints or actions indicated a. through f. (Put your answers on your answer paper.)

QUESTION 3.06 (1.00)

Match the following total temperature error signals (Tavg -Tref) in COLUMN A to their corresponding automatic rod speed signals in COLUMN B.

COLUMN A	COLUMN B
a. -0.5 F	1. 0 steps/ min.
b. -2.0 F	2. 8 steps/min.
c. -4.0 F	3. 40 steps/min.
d. -6.0 F	4. 48 steps/min.
	5. 72 steps/min.

QUESTION 3.07 (1.50)

For the following statements concerning the rod control system, indicate whether each is TRUE or FALSE.

a. The programmed Tavg is increased linearly with power.

b. The nonlinear gain unit adjusts circuit gain depending on turbine power.

c. The rod drive mechanisms receive their power from two parallel motor generator sets through two series Reactor Trip Breakers.

QUESTION 3.08 (1.50)

TRUE or FALSE?

a. The ~~stop~~ throttle and reheat stop valves are closed by the Overspeed Protection Controller.

b. The governor and intercept valves are the ONLY valves closed by the Overspeed Trip Mechanism.

c. The ~~stop~~ throttle, reheat stop, and intercept valves are opened when the turbine is latched.

c. Valves opened when the turbine is latched.

c

QUESTION 3.09 (1.00)

What signals are sent to the reactor protection system to indicate a turbine trip?

a. Governor valves closed and Auto-stop-oil pressure low.

Throttle
b. ~~Stop~~ valves closed and Auto-stop-oil pressure low.

c. Governor valves closed and EHC fluid pressure low.

throttle
d. ~~Stop~~ valves closed and EHC fluid pressure low.

QUESTION 3.10 (.75)

Unit 1 is operating at 45% power with all systems in automatic control. For each of the following conditions, give the direction of inital rod motion. (In, Out, or None)

a. A steam generator Atmospheric Relief Valve fails open.

b. A feedwater heater string becomes isolated.

c. The lower detector of the power range channel N-44 fails high.

QUESTION 3.11 (1.00)

Select the answer which most closely describes the normal status of the Steam Dump system. Assume unit is at 100% power.

a. The system is disarmed in the steam pressure mode with load rejection controller selected to effect steam dump valve operation upon receipt of an error signal between Tavg and Tno-load of >/= 5 F.

b. The system is disarmed in the Tavg mode with the ~~turbine~~ reactor trip controller selected to effect steam dump valve operation upon a ~~turbine~~ reactor trip signal when Tavg increases to >/= high Tavg setpoint.

c. The system is disarmed in the Tavg mode with the load rejection controller selected to effect steam dump valve operation upon receipt of an error signal between Tavg and Tref >/= 5 F if a load rejection arming signal is supplied.

d. The system is disarmed in the Tavg mode with the load rejection controller selected and having any error signal between Tavg and Tno-load present and sufficient to effect steam dump valve operation immediately upon receipt of a load rejection arming signal.

QUESTION 3.12 (1.00)

Which of the following statements concerning the Steam Dump Control System is correct?

a. The steam dump valves fail open on loss of air.

b. In order to cooldown below 540 F, the steam dump mode selector switch must be momentarily taken to "Reset" and returned to "Steam Pressure".

c. When in the Tavg mode, the steam dumps are armed by the reactor trip breakers opening.

d. In the load rejection mode, the steam dump valves may receive a signal to modulate open or a signal to trip open depending upon the magnitude of the error signal.

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

QUESTION 3.13 (1.00)

What signal "ARMS" the Steam Dump system when in the Tavg mode?

a. 5% / 2 min.

b. 10% / 2 min.

c. 10 F Tavg/Tref error.

d. 13 F Tavg/ Tref error.

QUESTION 3.14 (1.00)

If the Intermediate Range Compensating Voltage failed low, would the indicated flux level be HIGHER, LOWER, or the SAME AS the actual flux at:

a. 100% power indicated on the power range.

b. 10^{-10} amps indicated on the intermediate range.

QUESTION 3.15 (1.00)

Which of the following statements describes the signal path from the Source Range detector to the Source Range level meter on the MCB?

a. Detector, Pre Amp, Discriminator, Log Integrator, Meter

b. Detector, Log Integrator, Pulse Shaper, Pulse Counter, Meter

c. Detector, Pre Amp, Log Integrator, Discriminator, Meter

d. Detector, Log Amp, Meter

QUESTION 3.16 (1.00)

What is the signal used to develop the Control Rod Insertion Limits?

a. Auctioneered high Tavg and Auctioneered high delta T.

b. Delta T and Tavg.

c. Auctioneered high delta T.

d. Auctioneered high Tavg.

QUESTION 3.17 (1.00)

Which of the following is a function of the Power Range Instruments?

a. Provides 20% power rod withdrawal stop.

b. Provides input to the Rod Insertion Limit computer.

c. Provides indication of Startup Rate on M-4.

d. Provides input to OTdT protection circuit.

QUESTION 3.18 (1.00)

The THREE input signals to the Steam Generator Water Level Control are:

a. Tavg, compensated feed flow, uncompensated steam flow.

b. Feed flow, compensated steam flow, Feed pressure.

c. Compensated feed flow, water level, compensated steam flow.

d. Uncompensated feed flow, compensated steam flow, water level.

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

QUESTION 3.19 (1.00)

Considering only the Steam Generator Level Control System, what would be the response of the INITIAL feedwater flow to the S/G if the controlling S/G pressure transmitter failed LOW during 50% power operations?

a. The flow would decrease due to the loss of the steam pressure input to the steam flow signal.

b. The flow would remain the same due to the steam pressure not affecting the steam flow.

c. The flow would increase due to the steam pressure input to the feed control valve position controller.

d. The flow would increase due to the loss of steam pressure input to the steam flow signal.

QUESTION 3.20 (.50)

TRUE or FALSE?

The indicated steam generator level would INCREASE if the differential pressure transmitter reference leg temperature decreased by 10 degrees. (ASSUME actual level does NOT change.)

QUESTION 3.21 (2.00)

How would a significant increase in the following affect the OTdT setpoint? (Increase, Decrease, or No effect).

a. Tavg.

b. RCS pressure.

c. Delta flux.

~~d. Actual delta T.~~ delete

QUESTION 3.22 (1.00)

Which of the following Radiation Monitors has an automatic isolation function?

a. Fuel Pool Radiation Monitor (RE-90-102).

b. Condenser Vaccuum Pump High Range Air Exhaust Monitor (RE-90-99).

c. ERCW Liquid Effluent Monitor (RE-90-133).

d. Shield Building Vent Monitor (RE-90-100).

QUESTION 3.23 (2.00)

TRUE or FALSE

a. The Source Range instrument uses a fission chamber for detecting neutrons.

b. Compensating current is used in the Intermediate Range detector to eliminate the gamma signal contribution.

c. The Power Range High Flux Setpoint Deviation alarm is automatically defeated below 50% power.

d. If control rods are in automatic and reactor power is at 50% when Power Range channel NI-42 fails low, the rods will automatically drive out.

QUESTION 3.24 (1.00)

Which of the following statements about temperature detectors is correct?

a. The thermocouple is connected to one leg of a bridge circuit and as the temperature changes the output voltage across the bridge changes.

b. When a thermocouple fails open it will respond in the same manner as an RTD and will indicate a full scale reading on the meter.

c. When a faster responding temperature signal is needed a direct immersion (wet bulb type) RTD is used instead of the thermowell mounted RTD.

d. A RTD is comprised of two wires of dissimilar metals in contact with each other and generates a voltage that is proportional to the temperature difference between the open ends of the wires.

QUESTION 3.25 (1.25)

Select the Emergency Diesel Generator (EDG) PARAMETER that will be adjusted by the METHOD and conditions listed. (i.e. f. 5.)

METHOD	EDG PARAMETER
a. Voltage adjust switch WITH EDG output breaker open	1. KW
b. Voltage adjust switch WITH EDG output breaker closed and normal 6.9 kv shutdown board supply breaker open.	2. VARS
c. Voltage adjust switch WITH EDG output breaker closed and normal 6.9 kv shutdown board supply breaker closed.	3. Voltage
d. Speed switch WITH EDG output breaker open.	4. Frequency
e. Speed switch WITH EDG output breaker and normal 6.9 kv supply breaker closed.	

QUESTION 3.26 (1.00)

Which of the following, by itself, will cause an automatic start of the Turbine Driven Auxiliary Feed Pump?

a. Actual Low-Low level in Loop #3 Steam Generator.

b. Loss of either Main Feed Pump @ 50% reactor power.

c. 25 seconds after a loss of offsite power.

d. Immediately after a loss of offsite power.

QUESTION 3.27 (1.00)

Which of the following is NOT a purpose of the time delay in tripping the main ~~turbine~~ generator breaker after a reactor trip?

a. Reduce arcing of the generator output breakers.

b. Prevent overspeeding of the main turbine.

c. Prevent lifting Steam Generator safeties following the trip.

d. Remove excess steam from the moisture/seperator reheaters.

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

QUESTION 4.01 (1.00)

It is necessary to dilute 200 ppm to get the critical boron concentration prior to pulling the control banks. Prior to the dilution the source range instruments read 30 and 37 cps. After diluting 100 ppm the same instruments read 62 and 75 cps. Which of the following is the proper operator action in accordance with GOI-2?

a. Stop the dilution and borate back to the original count rate.

b. Stop the dilution and evaluate the situation.

c. Continue the dilution and continuously monitor the count rate.

d. Continue the dilution and recalculate the ECC.

QUESTION 4.02 (1.00)

According to a precaution in GOI-~~2~~ 1 during solid plant operation, what is the minimum Reactor Coolant System (RCS) temperature ~~for~~ while ~~starting~~ operating a Reactor Coolant Pump (RCP)?

a. 100 F

b. 160 F

c. 190 F

d. 250 F

QUESTION 4.03 (1.00)

Fill in the blanks in accordance with the Precautions Section of the "GOI's".

a. A load change rate of +/-_____%/min. or a step change of _____% should not be exceeded.

b. The boron concentration difference between the pressurizer and the RCS must not exceed _____ppm.

c. The maximum allowable heatup rate for the RCS is _____deg./hr.

QUESTION 4.04 (1.00)

GOI-2 "Plant Startup from Hot Standby to Minimum Load" states that the shutdown banks must be at the fully withdrawn position whenever positive reactivity is being inserted, except when ...

a. the Shutdown Margin has been calculated to be 900 pcm.

b. the RCS is borated to the cold shutdown concentration with plant cooldown in progress.

c. the reactor is in the source range with the High Flux at Shutdown alarm operable.

d. the actual boron concentration is greater than the predicted critical boron concentration.

QUESTION 4.05 (1.00)

TRUE or FALSE?

During solid plant operations with pressure being maintained by the low pressure letdown valve, FCV-62-81 in automatic ...

a. if the RHR system pressure exceeds 700 psig, the RHR suction from Loop 4 hot leg valves will close.

b. with RCS pressure at 300 psig and no steam bubble in the pressurizer, it is permissible to isolate the RHR suction line from the RCS.

QUESTION 4.06 (1.00)

During normal CVCS operation, which of the following is an abnormal condition and would require operator action to correct?

a. VCT pressure is 15 psig.

b. The temperature of the fluid leaving the letdown heat exchangers is 127 F.

c. The RCP seal injection water tempeature is 120 F and flow to the seals is 8 gpm/pump.

d. RCP seal differential pressure is 300 psid.

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

QUESTION 4.07 (1.00)

At what point during plant cooldown from hot standby using GOI-3B is the remaining Turbine-driven Main Feed Water Pump removed from service?

a. Tavg is between 540 F and 525 F.

b. Main steam pressure drops to 1000 psig.

c. Tavg is between 440 F and 425 F.

d. RCS pressure is between 450 psig and 425 psig.

QUESTION 4.08 (1.00)

When an RCP is stopped, either Component Cooling Water through the thermal barrier, or seal water to the RCP must be continued until the RCS temperature is reduced below: (Choose the most correct answer)

a. 150 F.

b. 160 F.

c. 170 F.

d. 175 F.

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

QUESTION 4.09 (1.00)

Using the following 5 actions, which of the below sequences is correct for STARTING the first Control Rod Drive M.G. set?

1. Flash the field.
2. Close the Aux. 150 VAC supply breaker to rod drives.
3. Adjust generator voltage.
4. Close the motor circuit breaker.
5. Close the generator circuit breaker.

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

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

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

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

QUESTION 4.10 (1.00)

Which of the following statements concerning the procedure for a dropped RCCA is correct?

a. Upon starting recovery of the dropped RCCA, an URGENT FAILURE alarm will occur because the lift coils for the other rods in the group have been disconnected.

b. The delta flux target band is not applicable during a dropped RCCA malfunction and recovery.

c. If two or more RCCA's have dropped, manually trip the reactor and proceed in accordance with EP-1.00.

d. Recovery from a dropped RCCA will be facilitated if Tavg is higher than Tref prior to commencing withdrawal of the dropped RCCA.

QUESTION 4.11 (1.00)

During an inadvertent dilution accident while at 100% power, which of the following will be the most probable cause of a reactor trip?

a. Pressurizer low pressure.

b. Over-temperature dT.

c. Over-power dT.

d. Power range monitor positive rate.

QUESTION 4.12 (1.00)

Which of the following would be a symptom of a Power Range instrument failing HIGH?

a. Rods stepping out.

b. Tavg increase.

c. OPdT reactor trip.

d. Rods stepping in.

QUESTION 4.13 (1.00)

Prior to a reactor startup with normal operating temperature and pressure the following RCS leakages exist. For each leak rate below, indicate if you ~~would~~ could STARTUP or REMAIN SHUTDOWN.

1. 0.5 GPH from a cracked weld on a narrow range temperature instrument manifold.

2. 1.0 GPM from a manual valve packing gland.

3. 0.4 GPM tube leakage on one Steam generator.

4. Leak from an unknown source of 1.2 GPM.

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

QUESTION 4.14 (1.00)

If the minimum temperature for criticality Technical Specification is violated, what option is available?

a. Restore to within its limits within 15 minutes or be in Hot Standby within the next 15 minutes.

b. Restore to within its limits within 15 minutes or be in Hot Standby within the next 6 hours.

c. Restore to within its limits within 1 hour or be in Hot Standby within the next 15 minutes.

d. Restore to within its limits within 15 minutes or be in Hot Standby within the next 1 hour.

QUESTION 4.15 (1.00)

If CCW flow to the RCP motor is lost, at what Upper or Lower bearing tempeature must the effected RCP be stopped?

a. 180 F.

b. 185 F.

c. 200 F.

d. 225 F.

QUESTION 4.16 (1.00)

Which one of the following is a symptom of "Rods fail to insert following a decrease in turbine load"?

a. Low pressurizer pressure.

b. "Pressurizer level high backup heaters on" alarm.

c. "Reactor coolant loops Tref-Tauct. high-low" alarm and Tref-Tavg = +5 F.

d. Rod insertion low limit alarm.

QUESTION 4.17 (1.00)

An operator receives the following whole body exposures of radiation:

30 RAD of Gamma
4 RAD of Fast Neutrons
3 REM of Thermal Neutrons

What is the total dose the operator received? Show all work.

QUESTION 4.18 (1.00)

Which of the following would be a result of a loss of a RCP with reactor power at 30%? Assume NO operator action.

a. Tavg increase.

b. Effected S/G level increase.

c. Reactor trip.

d. Uneffected loop flow decrease.

QUESTION 4.19 (1.00)

Match the following RCP seal alarms/indications in COLUMN A to the causes in COLUMN B. (i.e. f. 6.)

COLUMN A	COLUMN B
a. #1 Seal high flow alarm	1. Increased flow to RCDT.
b. #1 Seal low flow alarm	2. Loss of injection water.
c. #2 Seal high flow	3. #1 Seal dP too low.
d. #3 Seal high flow	4. Increased filling of seal standpipe.

Delete c & d

Do not raise points for a & b

QUESTION 4.20 (1.00)

Which of the following would NOT require Emergency Boration?

a. Failure of a control rod to fully insert following a reactor trip.

b. Excessive control rod withdrawal when at power.

c. Failure of Boric Acid Flow Controller FC-62-139 to function properly.

d. Uncontrolled reactor heatup following a reactor trip.

QUESTION 4.21 (1.00)

Arrange the following Critical Safety Function Status Tree colors by numbering in order of priority. Place answers on answer paper.

a. Green

b. Yellow

c. Red

d. Orange

QUESTION 4.22 (1.00)

FR-S.1 "Response to Nuclear Power Generation/ATWS" is entered from what procedure?

a. ES-0.1 Reactor Trip Response.

b. E-1 Loss of Reactor or Secondary Coolant.

c. FR-C.1 Response to Inadequate Core Cooling.

d. E-0 Reactor Trip or Safety Injection.

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

QUESTION 4.23 (1.00)

Following a reactor trip, how much Boron (gallons) must be added for each control rod not fully inserted?

a. 350

b. 400

c. 450

d. 500

QUESTION 4.24 (1.00)

Which of the following is NOT an immediate operator action for a Safety Injection as stated in E-0?

a. Verify Containment Isolation.

b. Check Tavg.

c. Verify AFW status

d. Verify Steam Dumps actuated.

QUESTION 4.25 (1.00)

TRUE or FALSE?

a. The transfer of ECCS suction to the containment sump is accomplished when RWST level is < 29%.

b. When RWST level reaches 0%, the Containment Spray Pumps are shifted to the Containment Sump.

QUESTION 4.26 (.50)

TRUE or FALSE?

Areas where dose rates are 500 mr/hr are required to be locked and access controlled.

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

QUESTION 4.27 (1.00)

Which of the following conditions would require re-initiation of safety injection according to ES-0.2 "Safety Injection Termination"?

	RCS PRESSURE	SUBCOOLING	PZR LEVEL%
a.	stable	25	15
b.	increasing	45	15
c.	stable	45	35
d.	stable	45	30

QUESTION 4.28 (1.00)

From the choices below pick one that best completes the following statement

A trip after a long period of reactor shutdown leaves little decay heat to be removed thus causing the possibility of excessice cooling of the reactor coolant if too much feedwater is being added. The operator should NEVER restore the steam generator water level, after a plant trip, at the cost of a reduction of the ______________________________.

a. plant pressure

b. shutdown margin

c. CST level

d. steam generator pressure.

QUESTION 4.29 (1.50)

For the power levels in Column A, find the one associated conditions or actions in Column B, as stated in GOI-5A.

COLUMN A	COLUMN B
a. 20 %	1. P-8 light goes out.
b. 30 %	2. P-9 light goes out.
c. 35 %	3. Observe turbine startup drains closed.
d. 50 %	4. Open HP drains to the No. 1 heater shells.
	5. Start two condensate demin pumps.
	6. Prior to exceeding _________ % power, steam generator chemistry must be below the limits for exceeding this specific power level.

QUESTION 4.30 (1.00)

In accordance with the "Loss of Reactor or Secondary Coolant" procedure, E-1, if one charging pump is operating and RCS pressure is uncontrollably decreasing, at what RCS pressure must the Reactor Coolant Pumps be stopped?

a. 1450 psig.

b. 1350 psig.

c. 1250 psig.

d. 1150 psig.

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

EQUATION SHEET

$f = ma$

$v = s/t$

Cycle efficiency = (Net work out)/(Energy in)

$w = mg$

$s = V_o t + 1/2\, at^2$

$E = mc^2$

$KE = 1/2\, mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

$A = A_o e^{-\lambda t}$

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

$W = v \Delta P$

$A = \frac{\pi D^2}{4}$

$t_{1/2}eff = \frac{[(t_{1/2})(t_b)]}{[(t_{1/2}) + (t_b)]}$

$\Delta E = 931\, \Delta m$

$\dot{m} = V_{av} A \rho$

$I = I_o e^{-\Sigma x}$

$\dot{Q} = \dot{m} Cp \Delta t$

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

$Pwr = W_f \Delta h$

$I = I_o e^{-\mu x}$

$I = I_o\, 10^{-x/TVL}$

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

$P = P_o 10^{sur(t)}$

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

$SUR = 26.06/T$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$SUR = 26\rho/\ell^* + (\beta - \rho)T$

$T = (\ell^*/\rho) + [(\beta - \rho)/\bar{\lambda}\rho]$

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

$T = (\beta - \rho)/(\bar{\lambda}\rho)$

$\rho = (K_{eff}-1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$M = 1/(1 - K_{eff}) = CR_1/CR_o$

$M = (1 - K_{effo})/(1 - K_{eff1})$

$SDM = (1 - K_{eff})/K_{eff}$

$\ell^* = 10^{-4}$ seconds

$\bar{\lambda} = 0.1$ seconds^{-1}

$\rho = [(\ell^*/(T\, K_{eff})] + [\bar{\beta}_{eff}/(1 + \bar{\lambda}T)]$

$I_1 d_1 = I_2 d_2$

$I_1 d_1^{\,2} = I_2 d_2^{\,2}$

$P = (\Sigma\phi V)/(3 \times 10^{10})$

$\Sigma = \sigma N$

$R/hr = (0.5\ CE)/d^2$(meters)

$R/hr = 6\ CE/d^2$ (feet)

Water Parameters

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.48 gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.
1 ft. H_2O = 0.4335 lbf/in.

Miscellaneous Conversions

1 curie = 3.7×10^{10} dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1 in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)
1 BTU = 778 ft-lbf

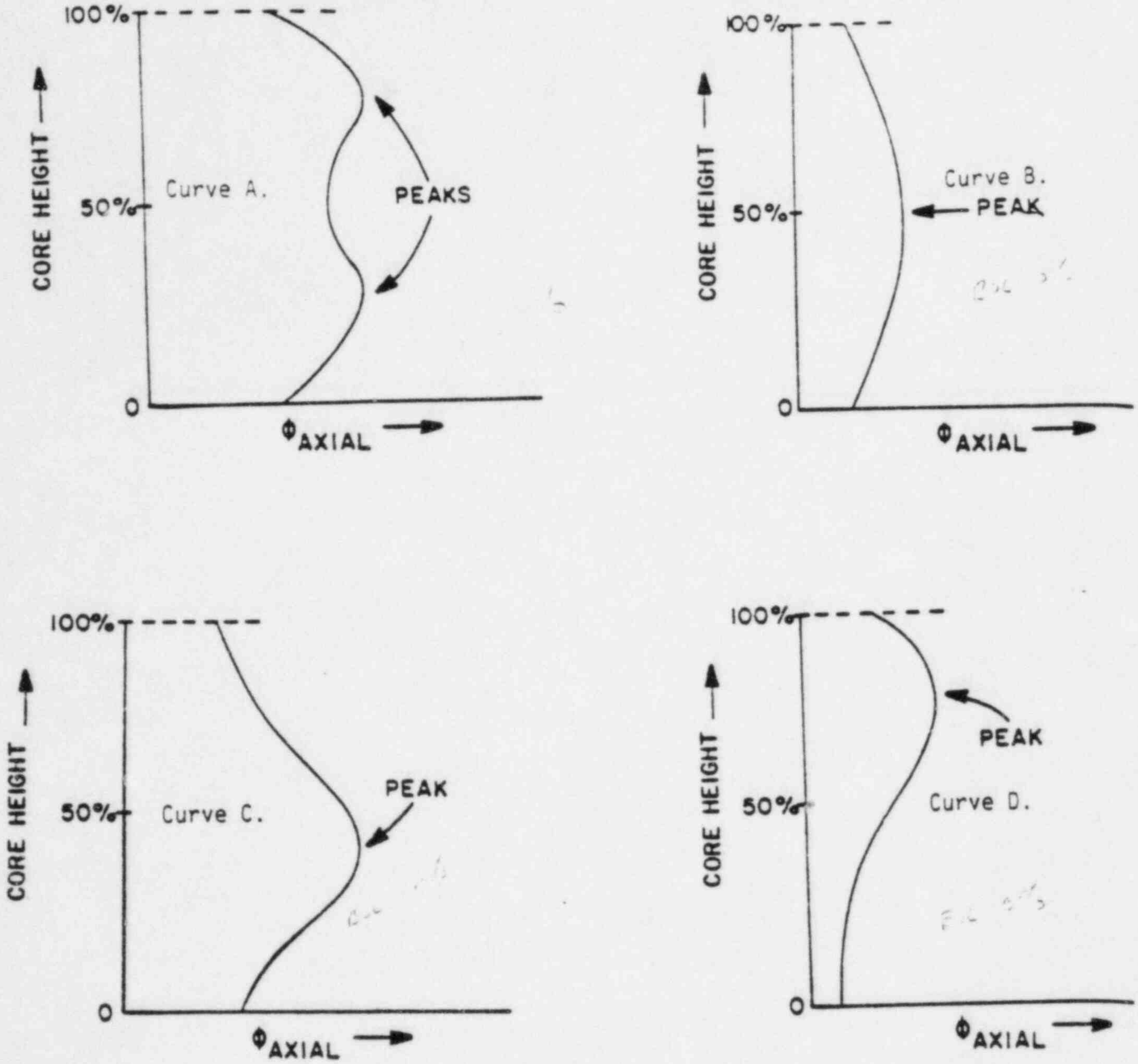
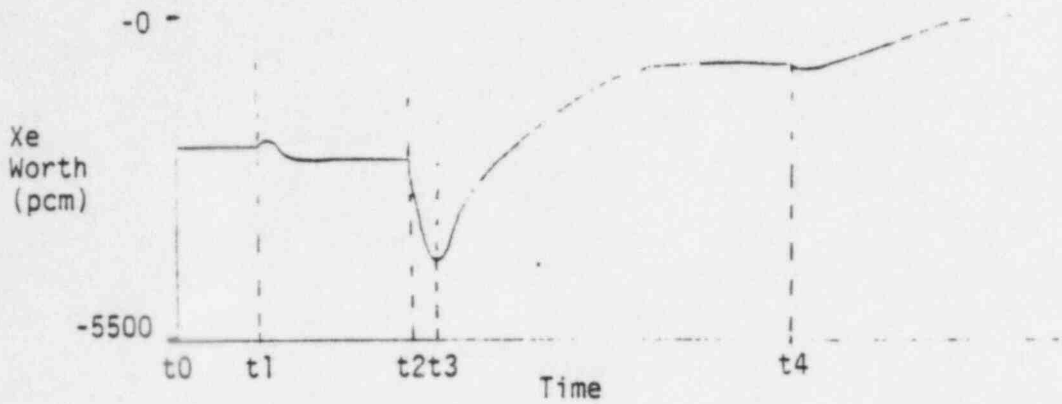

100%
50%
0
CORE HEIGHT
Curve A.
PEAKS
Φ_{AXIAL}
100%
50%
0
CORE HEIGHT
Curve B.
PEAK
Φ_{AXIAL}
100%
50%
0
CORE HEIGHT
Curve C.
PEAK
Φ_{AXIAL}
100%
50%
0
CORE HEIGHT
Curve D.
PEAK
Φ_{AXIAL}

Figure A - 4.



XENON vs. TIME CURVE

FIGURE 1.1

(10-81)

U.S. NUCLEAR REGULATORY COMMISSION

OPERATOR/SENIOR OPERATOR LICENSING EXAMINATION
ANSWERS

NAME

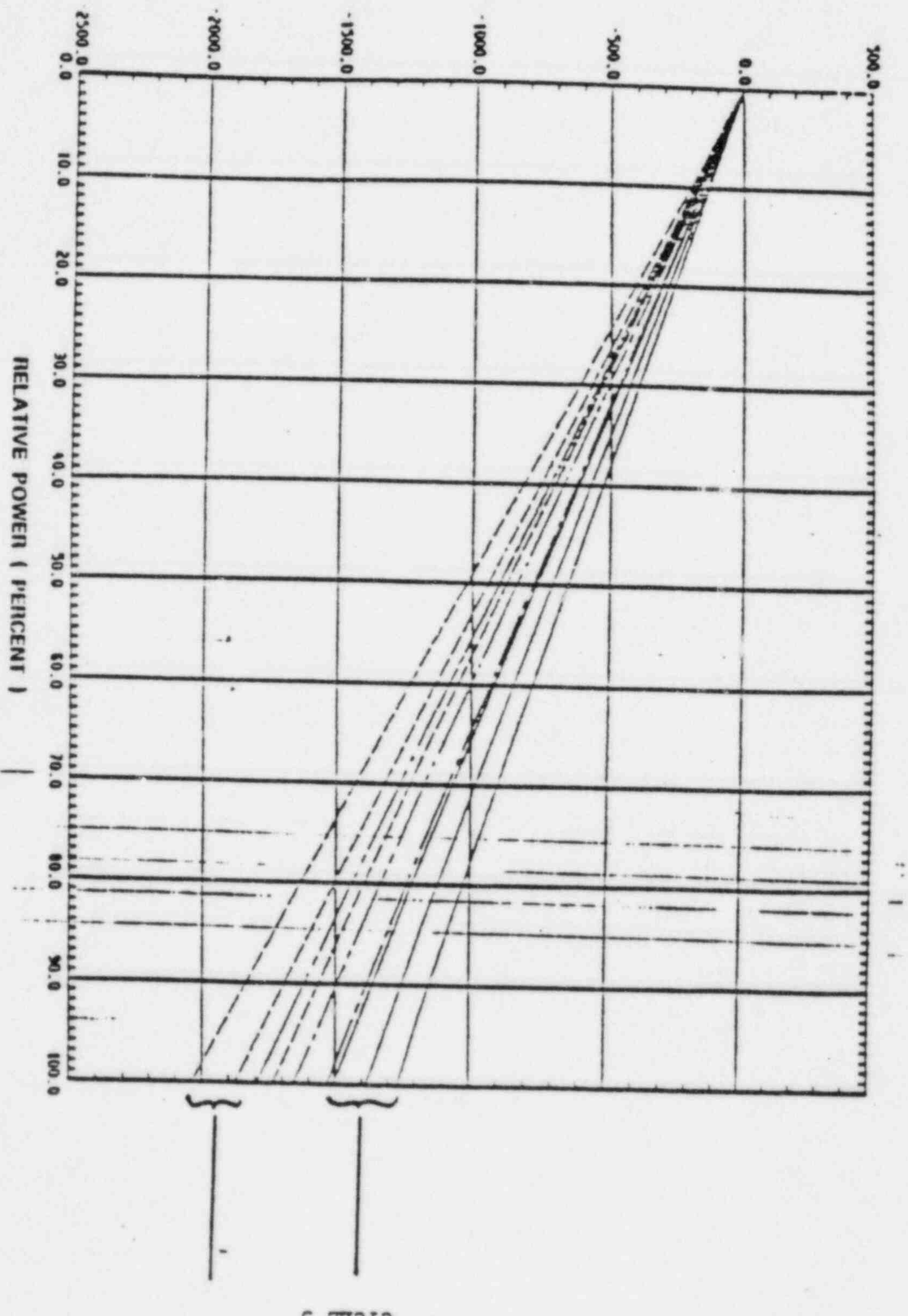


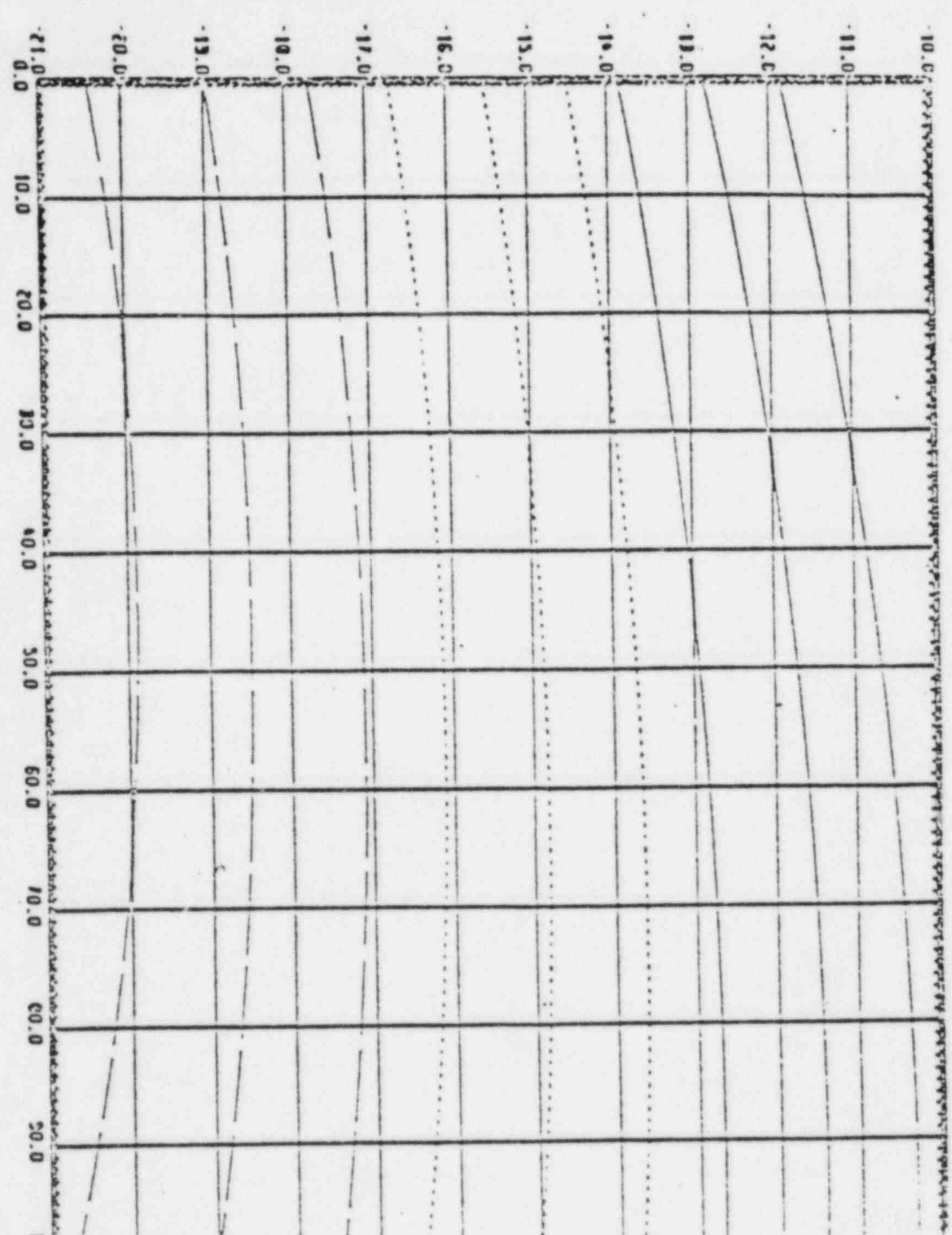
FIGURE ~~6.0-13~~ B.6.2

CYCLE 3

UNIT 2

SQNP
TI-28
Figure B.6.2
Page 1 of 1
Rev. 61

(10-81) U.S. NUCLEAR REGULATORY COMMISSION OPERATOR/SENIOR OPERATOR LICENSING EXAMINATION ANSWERS	NAME



B.7.1

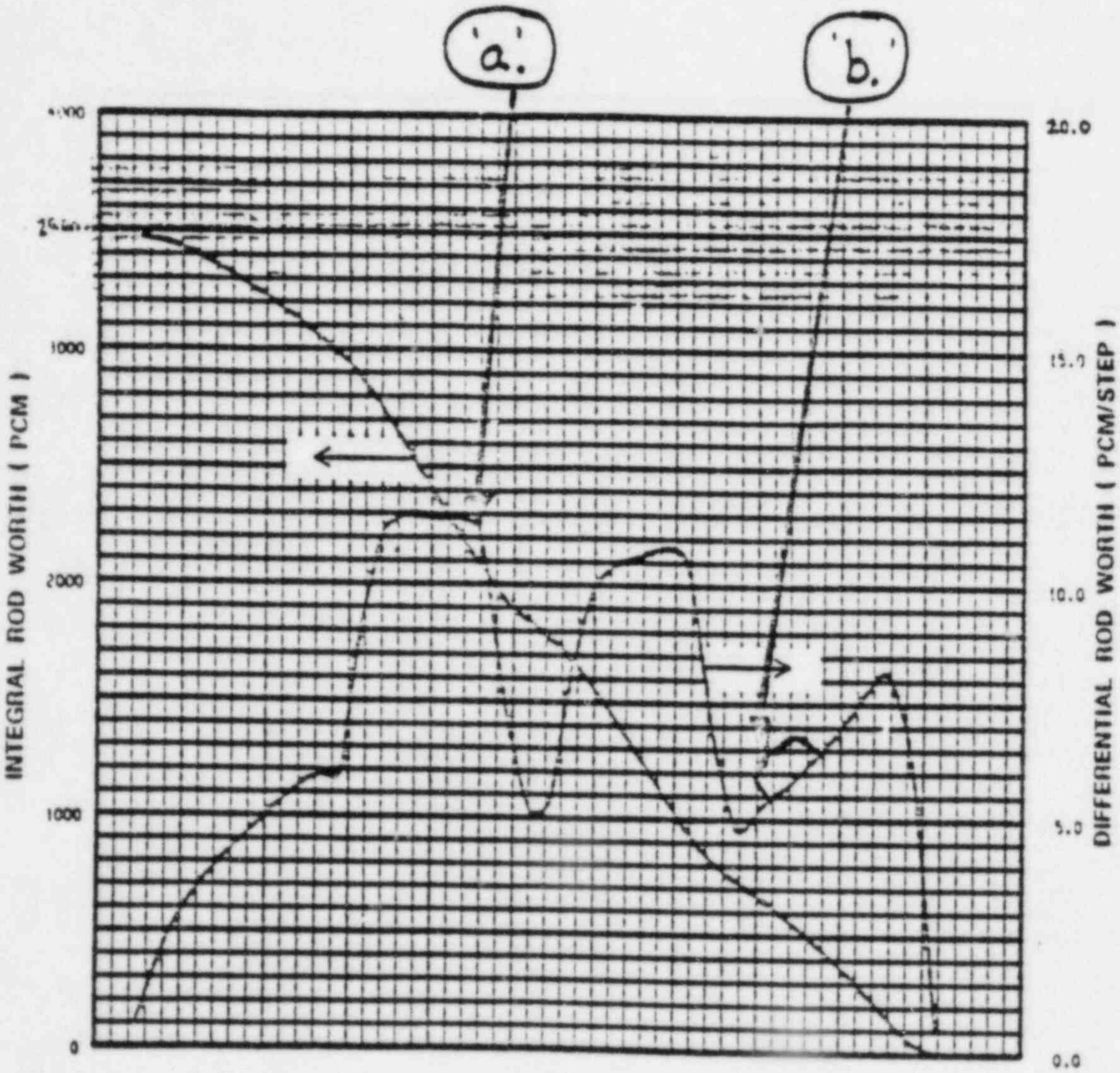
SONP
TI-28
Figure B.7.1
Page 1 of 1
Rev. 51

CYCLE 3

SQNP
TI-28
Figure B.2.C
Page 3 of 3
Rev. 61

CYCLE 3

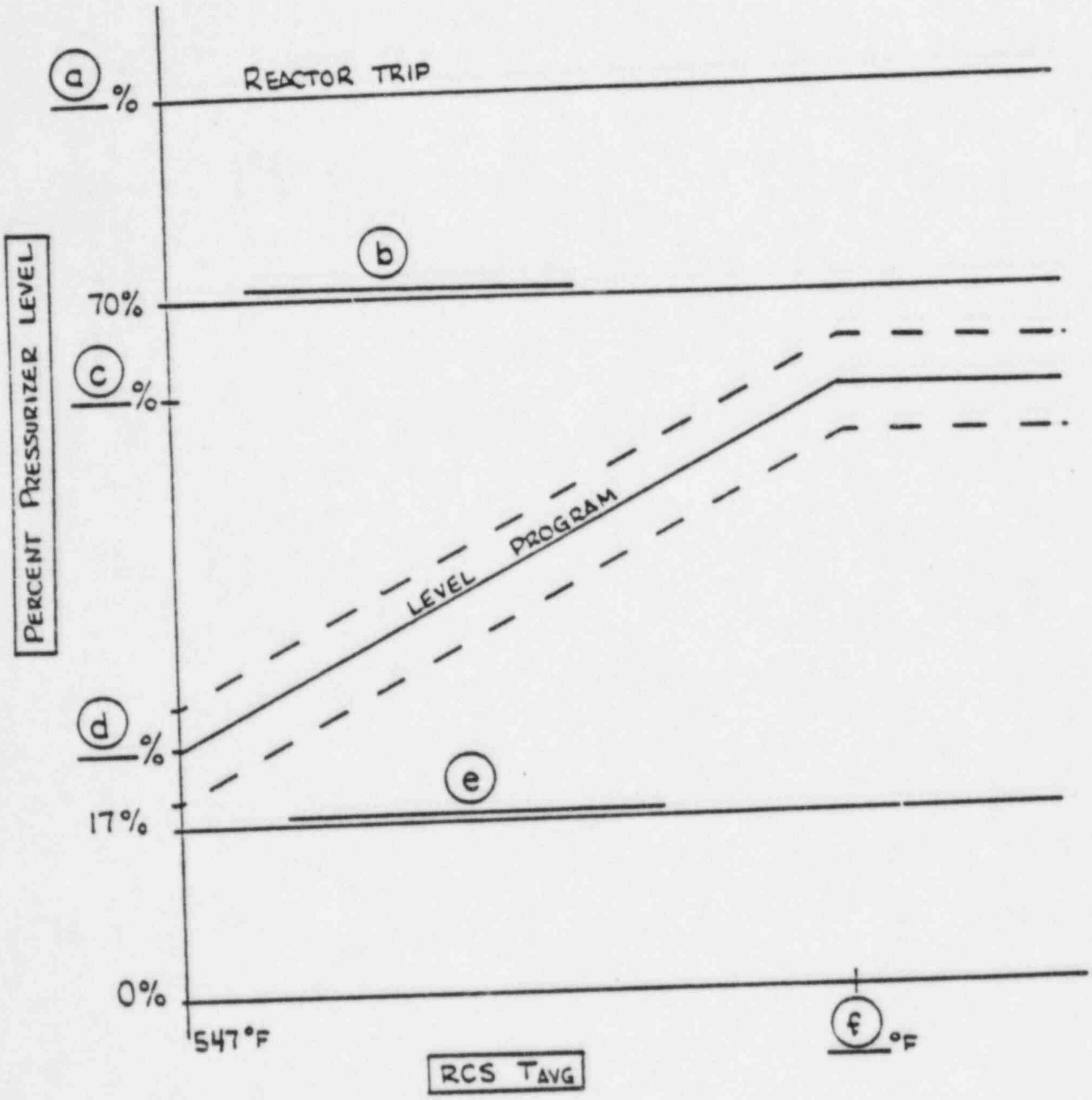
UNIT 2



ROD BANK POSITION STEPS WITHDRAWN

~~FIGURE~~ Figure B.2.C.

DIFFERENTIAL AND INTEGRAL ROD WORTH VERSUS STEPS WITHDRAWN AT EOL



a
%
REACTOR TRIP
b
70%
c
%
LEVEL PROGRAM
d
%
e
17%
0%
547°F
f
°F
PERCENT PRESSURIZER LEVEL
RCS TAVG

Fig. 3.1

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. The lower containment atmosphere particulate radioactivity monitoring system,

b. The containment pocket sump level monitoring system, and

c. The lower containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous or particulate radioactive monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. R16

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. The lower containment atmosphere gaseous and particulate monitoring systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Containment pocket sump level monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months.

MAR 25 1982

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 40 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. 1 GPM leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1. R16

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by: R16

MAR 25 1982

SURVEILLANCE REQUIREMENTS (Continued)

a. Monitoring the lower containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment pocket sump inventory and discharge at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into Mode 3 or 4. R16

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing requirements required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months. R16

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4. R16

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
63-560	Accumulator Discharge
63-561	Accumulator Discharge
63-562	Accumulator Discharge
63-563	Accumulator Discharge
63-622	Accumulator Discharge
63-623	Accumulator Discharge
63-624	Accumulator Discharge
63-625	Accumulator Discharge
63-551	Safety Injection (Cold Leg)
63-553	Safety Injection (Cold Leg)
63-557	Safety Injection (Cold Leg)
63-555	Safety Injection (Cold Leg)
63-632	Residual Heat Removal (Cold Leg)
63-633	Residual Heat Removal (Cold Leg)
63-634	Residual Heat Removal (Cold Leg)
63-635	Residual Heat Removal (Cold Leg)
63-641	Residual Heat Removal/Safety Injection (Hot Leg)
63-644	Residual Heat Removal/Safety Injection (Hot Leg)
63-558	Safety Injection (Hot Leg)
63-559	Safety Injection (Hot Leg)
63-543	Safety Injection (Hot Leg)
63-545	Safety Injection (Hot Leg)
63-547	Safety Injection (Hot Leg)
63-549	Safety Injection (Hot Leg)
63-640	Residual Heat Removal (Hot Leg)
63-643	Residual Heat Removal (Hot Leg)
87-558	Upper Head Injection
87-599	Upper Head Injection
87-560	Upper Head Injection
87-561	Upper Head Injection
87-562	Upper Head Injection
87-563	Upper Head Injection
FCV-74-1*	Residual Heat Removal
FCV-74-2*	Residual Heat Removal

R16

R16

R16

*These valves do not have to be leak tested following manual or automatic actuation or flow through the valve.

MAR 25 1982

MASTER COPY

ANSWER 1.01 (1.00)

c.

REFERENCE
WNTC, HTFF, Chap. 12, p. 15

ANSWER 1.02 (1.00)

a. False

b. True (Delete - true for trans. False for steady state) [0.5 ea.]

REFERENCE
SQNP, HTFF text p. 202

ANSWER 1.03 (1.50)

a. True

b. False

c. False [0.5 each]

REFERENCE
WNTC Thermal Hydraulic Principles and Applications to the Pressurized Water Reactor, Chapter 10, pp 32,38,49, Chapter 11, p 27

ANSWER 1.04 (2.00)

1. Convection

2. Radiation/convection (large Delta T)

3. Conduction

4. Convection (natural) [0.5 ea.]

REFERENCE
SQNP, HTFF text, pp. 191-206

ANSWER 1.05 (2.00)

a. Decrease

b. Decrease

c. Increase

d. Decrease [0.5 ea.]

REFERENCE
SQNP, HTFF, page 15

ANSWER 1.06 (.50)

TRUE

REFERENCE
System Descriptions, Chapter 3, pp 6 & 7

ANSWER 1.07 (1.00)

c.

REFERENCE
SQNP, HTFF, p. 11 & 12

ANSWER 1.08 (2.00)

a. 3

b. 2 [1.0 ea.]

REFERENCE
SQNP, Q & A Bank , sec 1-11

ANSWERS -- SEQUOYAH 1&2 -85/05/20-JAGGAR, F.

ANSWER 1.09 (1.00)

1. B or D per facility request depending on new or recycle core

2. C

3. D

4. A

REFERENCE
WNT, Chap. 3, pp. 3-44 to 3-53

ANSWER 1.10 (1.00)

c.

REFERENCE
SQNP, T.S. 3.1.1.3; Review of Reactivity Coefficients p. 4

ANSWER 1.11 (1.00)

b.

REFERENCE
SQNP, 6 Factor Formula lesson, pp. 3 - 5

ANSWER 1.12 (4.00)

a. 1

b. 3

c. 3

d. 4

REFERENCE
SQNP, Review of core poisons lesson, p. 6

ANSWER 1.13 (.50)

FALSE.

REFERENCE
SQNP, Review of Neutron Physics lesson, p. 3

ANSWER 1.14 (1.00)

c.

REFERENCE
SQNP, Review of Neutron Kinetics, p. 5

ANSWER 1.15 (.50)

TRUE.

REFERENCE
SQNP, Review of Neutron Kinetics, p. 5

ANSWER 1.16 (3.00)

a. 3

b. 4

c. 3 OR 4 ~~depending on~~ based on present (Cycle 3) value for β

REFERENCE
SQNP, Subcritical Multiplication lesson, p. 5; Review of Kinetics, p. 3

ANSWER 1.17 (1.50)

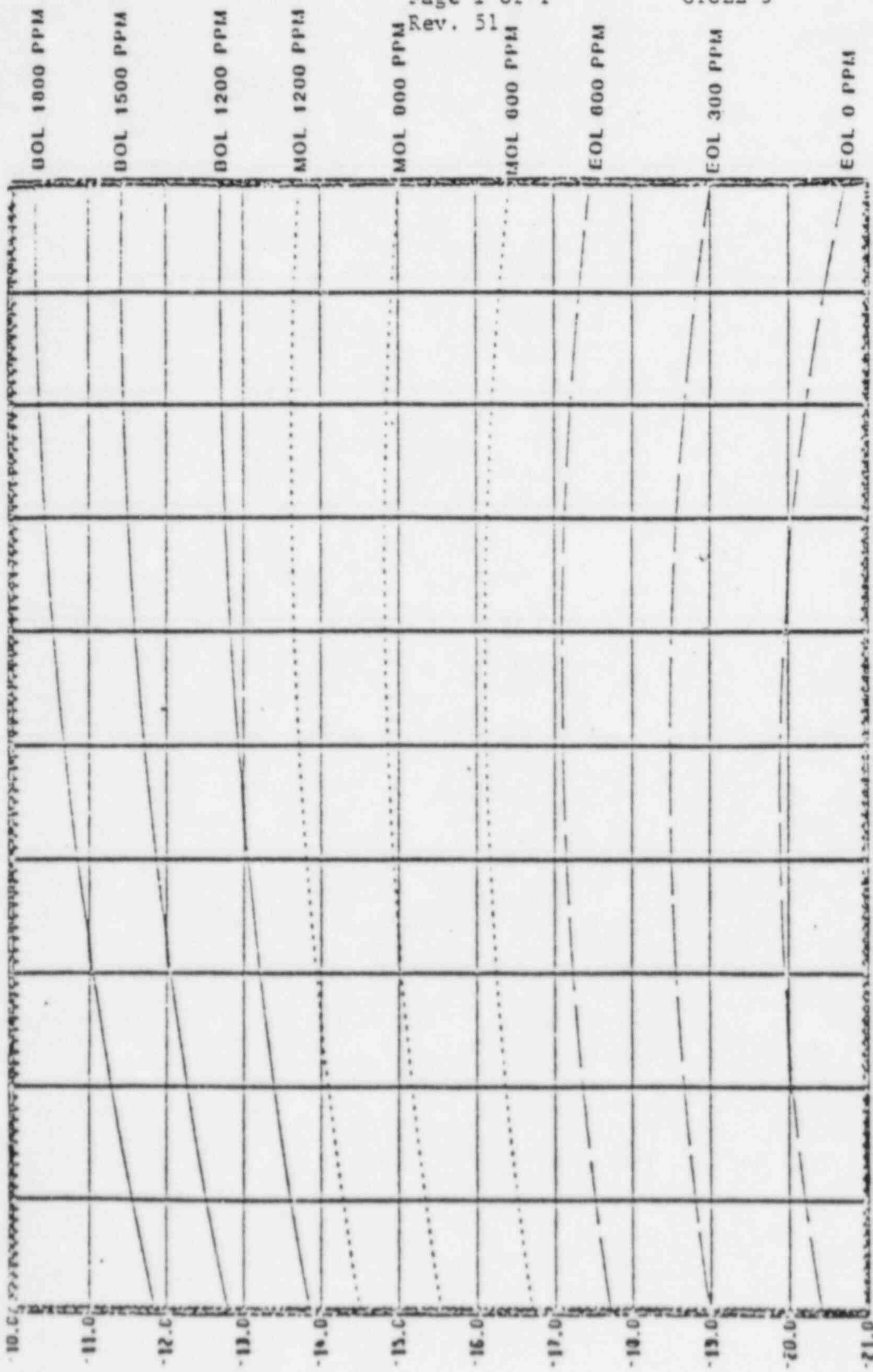
a. 5.7.1 (0.5)

b. See graphs attached. [0.5 ea.] (1.0)

SONP
TI-28
Figure B.7.1
Page 1 of 1
Rev. 51

CYCLE 3

UNIT 1



TOTAL POWER COEFFICIENT VERSUS POWER LEVEL AT BOL, MOL, AND EOL

48

UNIT 2

CYCLE 3

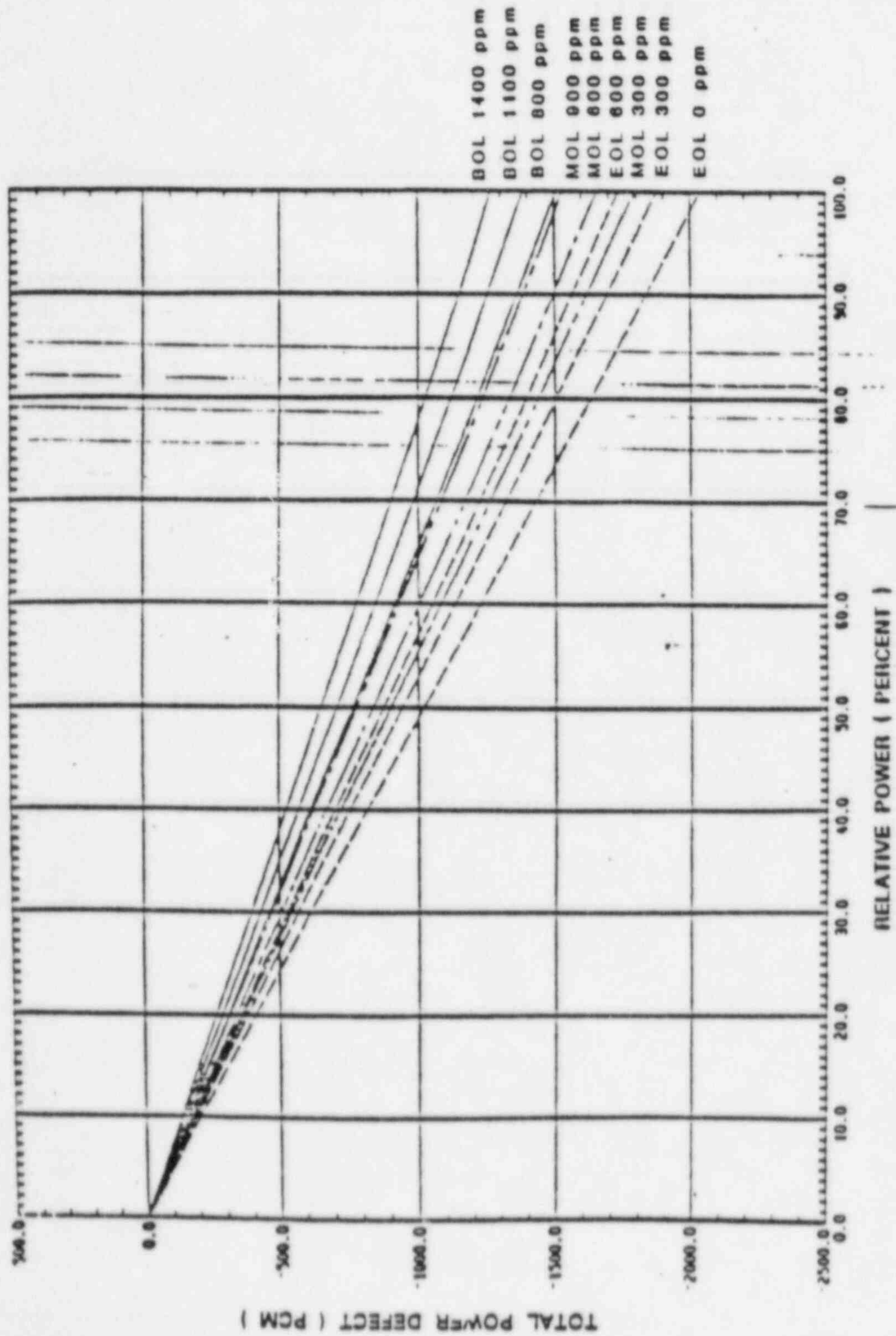


FIGURE ~~6.0-13~~ B.6.2

TOTAL POWER DEFECTS VERSUS POWER LEVEL AT BOL, MOL, AND EOL

*Entire Page

REFERENCE
SQNP, Curve book, Fig's B.6.2, B.7.1

ANSWER 1.18 (2.00)

a. 3

b. 1 [1.0 ea.]

REFERENCE
SQNP, Review of Core Poison lesson, pp. 4 - 5

ANSWER 1.19 (.75)

1. c

2. a

3. b [0.25 ea.]

REFERENCE
SQNP, Review of Core Poisons, pp. 4 - 7

ANSWER 1.20 (.75)

a. 3 or 4

b. 1

c. 4

REFERENCE
SQNP, HTFF text, pp. 174 -185

ANSWER 1.21 (1.00)

a. FALSE

b. TRUE [0.5 ea.]

REFERENCE
General Physics, HT&FF, pp. 155 and 320 and Subcooled Liquid Density Tables

ANSWER 1.22 (1.00)

c

REFERENCE
SQNP, HTFF text, pp. 23 - 24

ANSWER 2.01 (1.00)

d.

REFERENCE
SQNP System Description, Electrical Distribution, p. 5.7-13

ANSWER 2.02 (.50)

FALSE.

REFERENCE
SQNP System Description, Electrical Distribution, p. 5.2-13

ANSWER 2.03 (1.00)

d.

REFERENCE
SQNP System Description, Electrical Distribution, p. 5.3-3

ANSWER 2.04 (1.00)

~~a~~ c. Reference is incorrect

REFERENCE
SQNP System Description, RCS, pp. 12 & 13 of 38

ANSWER 2.05 (1.00)

a. 4

b. 1

c. 2

d. 5

REFERENCE
SQNP System Descriptions, ECCS, p. 4.2-27 - 29; RHR, p. 4.1-17

ANSWER 2.06 (1.00)

d.

REFERENCE
SQNP System Descriptions, RHR, pp. 4 & 9 of 11

ANSWER 2.07 (1.00)

d.

REFERENCE
SQNP System Description, CVCS, p. 14 of 40
AOI-34.

ANSWER 2.08 (.50)

FALSE.

REFERENCE
SQNP System Description, Fuel Handling, p. 9-6

ANSWER 2.09 (1.00)

b.

REFERENCE
SQNP System Description, Component Cooling System, p. 7.1-17

ANSWER 2.10 (1.00)

d.

REFERENCE
SQNP System Descriptions, Condensate and Feedwater, pp. 9 & 10 of 11

ANSWER 2.11 (2.00)

a. FALSE

b. FALSE

c. TRUE

d. FALSE

REFERENCE
SQNP System Description, ECCS, pp. 4.2-7 & 8

ANSWER 2.12 (1.00)

c.

REFERENCE
SQNP System Description, MainSteam, p. 7 of 10

ANSWER 2.13 (1.00)

a.

REFERENCE
SQNP System Description, AFW, p. 6 of 8

ANSWER 2.14 (1.00)

c.

REFERENCE
SQNP Diesel Generator Handout

ANSWER 2.15 (1.00)

a. 3

b. 4

c. 2

d. 1

REFERENCE
SQNP Diesel Generator Handout

ANSWER 2.16 (1.00)

a. TRUE

B. FALSE

REFERENCE
SOI 74-1A; pp 5 & 7.

ANSWER 2.17 (1.00)

d.

REFERENCE
SQNP System Description, RCS, p. 13 of 38

ANSWER 2.18 (1.00)

c.

REFERENCE
SQNP System Description, CVCS, p. 9 of 40

ANSWER 2.19 (1.00)

c.

REFERENCE
SQNP System Description, RCS, p. 28 of 38

ANSWER 2.20 (1.00)

a.

REFERENCE
SQNP System Description, p. 4 of 8

ANSWER 2.21 (1.00)

~~d.~~ a

REFERENCE
SQNP System Description, AFW, p. 5 of 8

ANSWER 2.22 (1.00)

b. May except c if use calculation (5.3 — 5.8)

REFERENCE
SQNP System Description, AFW, p. 3 of 8

ANSWER 2.23 (1.00)

b.

REFERENCE
SQNP Condensate and Feedwater, p. 10 of 11

ANSWER 2.24 (1.00)

d.

REFERENCE
SQNP System Description, RHR, p. 4.1-3

ANSWER 2.25 (2.00)

a. CLOSE
b. CLOSE
c. OPEN
d. CLOSE
e. NO
f. OPEN
g. CLOSE
h. NO
i. NO
j. NO [0.2 ea.]

REFERENCE
SQNP System Description, ECCS, CVCS, MNSTM, CCW

ANSWER 2.26 (1.00)

b.

REFERENCE
SQNP SOI 68.2 p. 2 of 11

ANSWER 2.27 (.50)

FALSE

REFERENCE
SQNP System Descriptions, CVCS, p. 3-28

ANSWER 2.28 (1.00)

a.

REFERENCE
SQNP System Description, CCWS, p. 7.1-4

ANSWER 2.29 (.50)

TRUE.

REFERENCE
SQNP System Description, Steam Systems, p. 10.1-11

ANSWER 2.30 (1.00)

c.

REFERENCE
SQNP SOI 62.1B, p. 2 of 8

ANSWER 3.01 (1.00)

d.

REFERENCE
SQNP System Description, Reactor Protection, pp. 6 of 13

ANSWER 3.02 (1.00)

a.

REFERENCE
SQNP System Description, Reactor Protection, p. 9 of 13

ANSWER 3.03 (1.00)

a. 4

b. 2

c. 5

d. 3 [0.25 ea.]

REFERENCE
SQNP System Description, Reactor Protection, pp. 10 & 11 of 13

ANSWER 3.04 (1.00)

d.

REFERENCE
SQNP System Description, Protection System, p. 11.10-32; ECCS, p. 4.2-10,11

ANSWER 3.05 (1.50)

a. 92%

b. High level alarm

c. 60%

d. 24.7%

e. Letdown line isolation or low-low level heater cutout

f. 578.2 F [0.25 ea.]

REFERENCE
SQNP PLS, pp. 16, 28, 34

ANSWER 3.06 (1.00)

a. 1

b. 2

c. 3

d. 5

REFERENCE
SQNP System Description, Rod Control, pp. 6 & 7 of 11

ANSWER 3.07 (1.50)

a. TRUE

b. ~~TRUE~~ false

c. TRUE

REFERENCE
SQNP System Description, Rod Control, pp. 5 - 9 of 11

ANSWER 3.08 (1.50)

a. FALSE

b. FALSE

c. TRUE [0.5 ea.]

REFERENCE
SQNP System Description, Turbine Control, p. 5; E-H Control, p. 10.3-12, 10.3-20

ANSWER 3.09 (1.00)

b.

REFERENCE
SQNP System Description, Protection System, p. 11.10-131

ANSWER 3.10 (.75)

a. Rods out.

b. Rods out.

c. Rods in.

REFERENCE
SQNP System Description, Rod Control, p. 5 & 6 of 1

ANSWER 3.11 (1.00)

c.

REFERENCE
SQNP System Description, Steam Dump Control, p. 7 of 8

ANSWER 3.12 (1.00)

d. or C

REFERENCE
SQNP, System Description, Steam Dump Control System, pp. 2 - 8

ANSWER 3.13 (1.00)

b.

REFERENCE
SQNP System Descriptions, Steam Dump Control, p. 7 of 8

ANSWER 3.14 (1.00)

a. Same

b. higher [0.5 ea.]

REFERENCE
SQNP System Descriptions, Excore Instrumentation, pp. 11 & 12 of 18

ANSWER 3.15 (1.00)

a.

REFERENCE
SQNP System Description, Excore Nuclear Instrumentation, p. 7 of 18

ANSWER 3.16 (1.00)

c.

REFERENCE
SQNP System Description, Rod Control, p. 11 of 11

ANSWER 3.17 (1.00)

d.

REFERENCE
SQNP System Description, Excore Instrumentation, p. 13 of 18

ANSWER 3.18 (1.00)

d.

REFERENCE
SQNP System Description, SGLCS, p. 11.7-1

ANSWER 3.19 (1.00)

a.

REFERENCE
SQNP System Description, SGLCS, p. 11.7-8 thru 11.7.10

ANSWER 3.20 (.50)

FALSE.

REFERENCE
SQNP System Description, SGLCS, 11.7

ANSWER 3.21 (2.00)

a. Decrease.

b. Increase.

c. Decrease.

~~d. No effect~~. or Decrease therefore question must be deleted [0.5 ea.]

REFERENCE
SQNP System Description, Reactor Protection, p. 11.10-24, 25

ANSWER 3.22 (1.00)

a.

REFERENCE
SQNP System Description, RMS, pp. 41 - 49 of 53

ANSWER 3.23 (2.00)

a. FALSE

b. TRUE

c. ~~TRUE~~ False Same SD.

d. FALSE [0.25 ea.]

REFERENCE
SQNP System Description, Excore Instrumentation, p. 6, 17, 10 of 18

ANSWER 3.24 (1.00)

c.

REFERENCE
Nuclear Power Plant Instrumentation Systems Manual, Ch. 4

ANSWER 3.25 (1.25)

a. 3

b. 3

c. 2

d. 4

e. 1 [0.25 ea.]

REFERENCE
SQNP SOI-82

ANSWER 3.26 (1.00)

d.

REFERENCE
SQNP System Description, Auxiliary Feedwater System, p.3 of 8

ANSWER 3.27 (1.00)

a.

REFERENCE
SQNP System Description, Electrical Distribution, p. 5.2-6

ANSWER 4.01 (1.00)

b.

REFERENCE
SQNP GOI-2, pg. 1

ANSWER 4.02 (1.00)

b.

REFERENCE
SQNP GOI-1, pg. 3

ANSWER 4.03 (1.00)

a. 5, 10

b. 50

c. 100

REFERENCE
SQNP GOI-1, p. 2; GOI-5, p. 2

ANSWER 4.04 (1.00)

b.

REFERENCE
SQNP GOI-2, p. 2

ANSWER 4.05 (1.00)

a. TRUE

b. FALSE

REFERENCE
SQNP SOI-74.1, pp. 3, 4

ANSWER 4.06 (1.00)

a.

REFERENCE
SQNP SOI-62.1B, pp. 8, 9

ANSWER 4.07 (1.00)

c.

REFERENCE
SQNP ~~S~~OI-3B, p. 15
G

ANSWER 4.08 (1.00)

a.

REFERENCE
SQNP SOI 68.2, p.3

ANSWER 4.09 (1.00)

c.

REFERENCE
SQNP SOI-85.1A, p. 3 of 4

ANSWER 4.10 (1.00)

a.

REFERENCE
SQNP AOI-2D, pp. 10 - 12

ANSWER 4.11 (1.00)

b.

REFERENCE
SQNP AOI-3D, p. 1 of 2

ANSWER 4.12 (1.00)

d.

REFERENCE
SQNP AOI-4D, p. 1 of 2

ANSWER 4.13 (1.00)

1. Shutdown
2. Startup
3. Shutdown
4. Shutdown [0.25 each]

REFERENCE
SQNP Technical Specifications 3/4.4.6.2

ANSWER 4.14 (1.00)

a.

REFERENCE
SQNP Tech. Spec. 3/4.3.1.1.4

ANSWER 4.15 (1.00)

c. or d SOI says 200, AOI says 225

REFERENCE
SQNP AOI-15A, p. 2 of 3

ANSWER 4.16 (1.00)

b.

REFERENCE
SQNP AOI-2, p. 2

ANSWER 4.17 (1.00)

30 RAD Gamma x 1 Rem/Rad	=	30 Rem
4 RAD Fast Neutrons x 10 Rem/Rad	=	40 Rem
3 REM Thermal Neutrons	=	3 Rem
		73 Rem

REFERENCE
10 CFR 20.1 and 10 CFR 55.21

ANSWER 4.18 (1.00)

a.

REFERENCE
SQNP AOI-5, pp. 2, 3

ANSWER 4.19 (1.00)

a. 2

b. 3 if seal leakoff or 4 if seal supply

c. ~~2~~ 4

d. 4

Delete c & d the Column A indications are really the causes and Column B was the indications

REFERENCE
SQNP AOI-23A, pp. 1 - 3 of 4

ANSWER 4.20 (1.00)

d.

REFERENCE
SQNP AOI-34A, p. 1 Of 3

ANSWER 4.21 (1.00)

a. 4

b. 3

c. 1

d. 2

REFERENCE
SQNP FRG's

ANSWER 4.22 (1.00)

d.

REFERENCE
SQNP FR-S.1, p. 1 of 5

ANSWER 4.23 (1.00)

a.

REFERENCE
SQNP ES-0.1, p. 2 of 13

ANSWER 4.24 (1.00)

d.

REFERENCE
SQNP E-0 pp. 2 - 5

ANSWER 4.25 (1.00)

a. TRUE

b. FALSE.

REFERENCE
SQNP ES-1.3 p. 1 of 4; ES-1.2 p. 1 of 3, App. A

ANSWER 4.26 (.50)

FALSE

REFERENCE
SQNP RCI-1, p. 4

ANSWER 4.27 (1.00)

a

REFERENCE
SQNP ES-0.2, pp 3, 4

ANSWER 4.28 (1.00)

b

REFERENCE
SQNP PL3, p. 6

ANSWER 4.29 (1.50)

a. 3
b. [illegible]
c. 1
d. 2

REFERENCE
SQNP GOI-5A, pp 5-8

ANSWER 4.30 (1.00)

c.

REFERENCE
SQNP E-1, p. 2 of 11

ENCLOSURE 3

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

Reviewed By:		
1. Ed Keyser	FACILITY:	SEQUOYAH 1&2
2. C.O. Brewer	REACTOR TYPE:	PWR-WEC4
3. W.G Payne	DATE ADMINISTERED:	85/05/20
4. B.C. Lake	EXAMINER:	VIDNOLA, A.
5. M.J. Lorek	APPLICANT:	MASTER

INSTRUCTIONS TO APPLICANT:

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

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
30.00	25.00	________	________	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
30.00	25.00	________	________	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
30.00	25.00	________	________	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
30.00	25.00	________	________	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
120.00	100.00	________	________	TOTALS

FINAL GRADE ________________%

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

____
APPLICANT'S SIGNATURE

QUESTION 5.01 (2.00)

a. If the reactor is operating in the power range, how long will it take to raise power from 20% to 40% with a +0.5 DPM Startup rate?

1. 12 seconds

2. 21 seconds

3. 36 seconds

4. 54 seconds

b. How long will it take to raise power from 40% to 60% with the same +0.5 DPM Startup rate?

1. 12 seconds

2. 21 seconds

3. 36 seconds

4. 54 seconds

QUESTION 5.02 (1.00)

Match the curves, on Figure A-4 on the next page, with the following plant descriptions. Put your answers on your answer paper, e.g. "Curve A -6.".

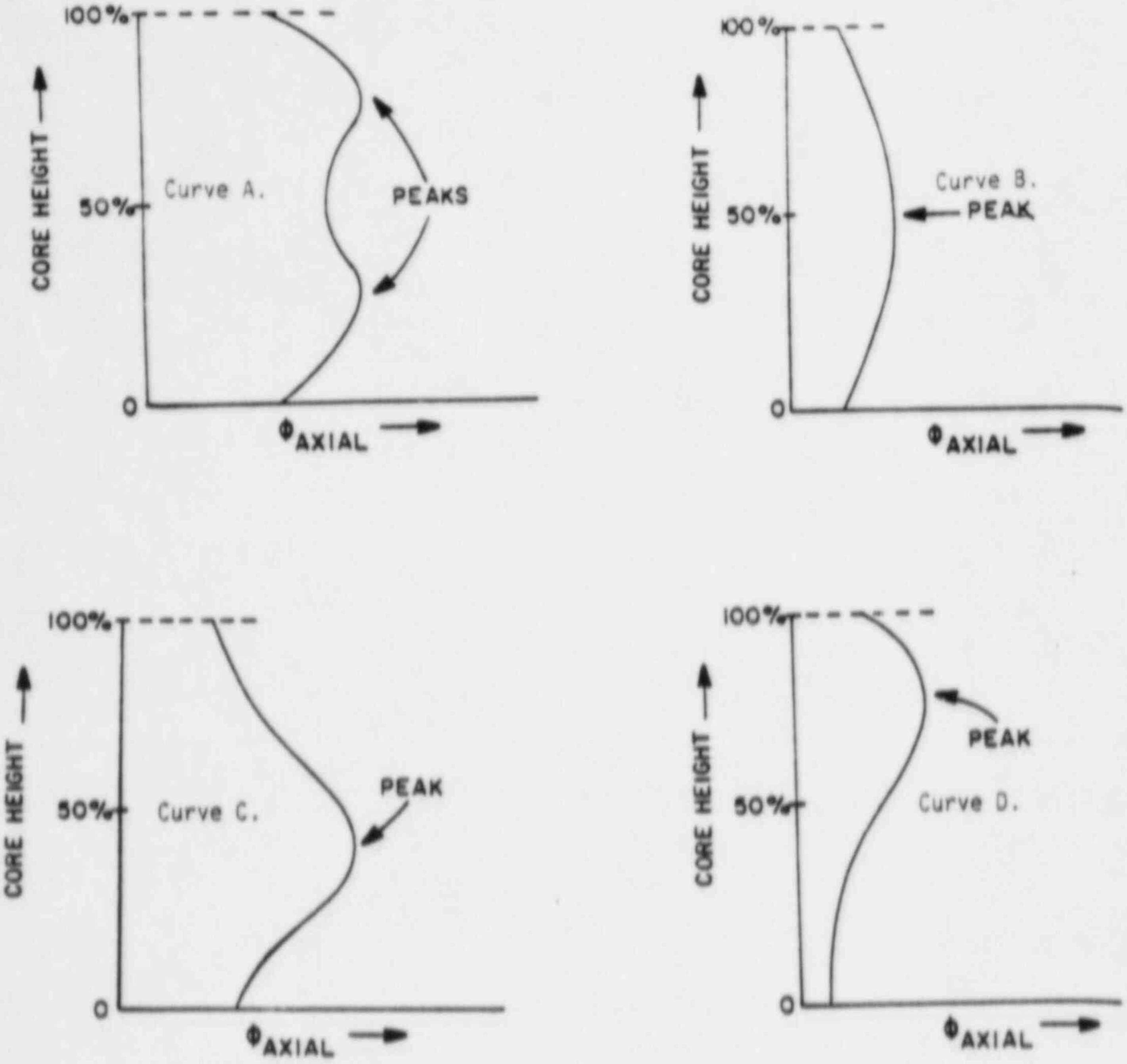
1. Beginning of life (BOL) - 0% power.

2. BOL - 100% power.

3. End of life (EOL) - 0% power.

4. EOL - 100% power.

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



100%
50%
0
CORE HEIGHT
Curve A.
PEAKS
ΦAXIAL
100%
50%
0
CORE HEIGHT
Curve B.
PEAK
ΦAXIAL
100%
50%
0
CORE HEIGHT
Curve C.
PEAK
ΦAXIAL
100%
50%
0
CORE HEIGHT
PEAK
Curve D.
ΦAXIAL

Figure A - 4.

QUESTION 5.03 (1.00)

Which of the following statements is TRUE?

a. It is NOT possible for the Moderator Temperature Coefficient (MTC) to ever become positive at the Sequoyah plant.

b. It is possible for the MTC to become positive, but ONLY when the reactor is in Mode 5.

c. If the MTC is positive, while the reactor is in Mode 2, Technical Specifications must be consulted because there are action statements that must be followed.

d. MTC can be positive in an under-moderated core where the moderator to fuel ratio is less than the optimum value.

QUESTION 5.04 (1.00)

Which of the following best describes the effect on MTC if the RCS temperature is LOWERED?

a. It becomes less negative because boron and water molecules are swept into the core as a result of the outsurge from the Pressurizer, therefore, neutrons spend more time in the resonance region.

b. It becomes less negative because the rate of change in the density of water per degree temperature change is less at lower temperature which causes a lesser change in rate in resonance escape probability.

c. It becomes more negative because thermal utilization increases and resonance escape probability decreases.

d. It becomes more negative because as temperature is lowered the moderator becomes more dense, this increases the amount of water molecules in the core therefore neutrons have a greater probability of colliding with a water molecule and this is an increased negative reactivity effect.

QUESTION 5.05 (4.00)

Use the attached Xenon worth curve, (on the next page) to answer the following four questions.

a. Power at T0 was at 70%. What was the power level between T1 and T2?

1. 90%

2. 50%

3. 20%

4. 10%

b. What was the length of time between T2 and T3?

1. 1 hour

2. 3 hours

3. 8 hours

4. 12 hours

c. What happened at T2?

1. Reactor tripped.

2. Rods were placed in AUTO, and turbine power was raised to 100%.

3. Reactor power was reduced to 10%.

4. Turbine power remained constant, rods were in manual and inserted 50 steps and the steam dump valves failed open(10% of rated power).

d. At time T4, what happened?

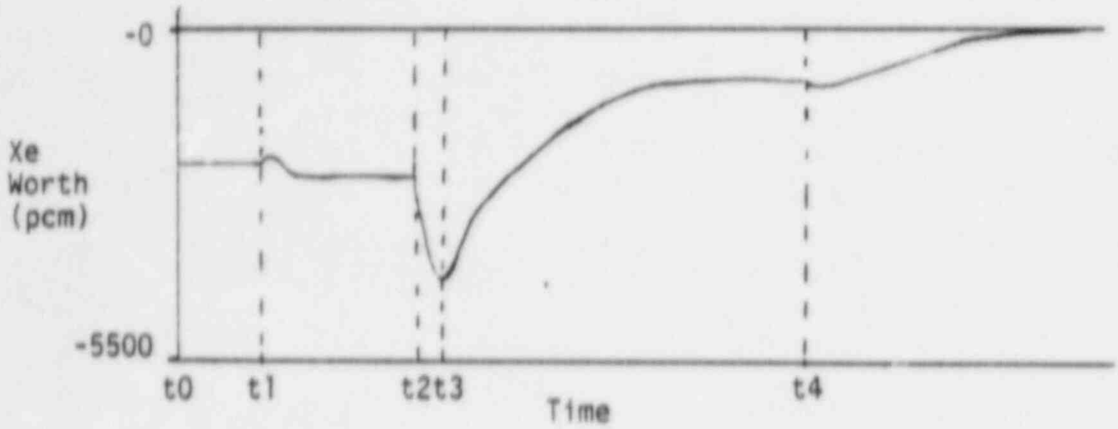
1. All Xenon production has stopped.

2. Iodine decay to Xenon has stopped.

3. All Xenon production remains constant, but the burnout increases.

4. Xenon production directly from fission has stopped, but Xenon production from decay of Iodine continues.

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



XENON vs. TIME CURVE

FIGURE 1.1

QUESTION 5.06 (1.00)

Which of the following is TRUE concerning RCS operation following the loss of one Reactor Coolant Pump?

a. Core coolant velocity decreases therefore the flowrate in the remaining loops decreases.

b. Flow to the vessel from the remaining three pumps is less than 3/4 of the original flow.

c. Flow in the idle loop bypasses the core.

d. Since only three S/G's are providing steam, the steam pressure and temperature in the remaining S/G's is increased.

QUESTION 5.07 (1.00)

TRUE or FALSE?

a. During 100% power operation, Departure from Nucleate Boiling Ratio (DNBR) is greater than the DNBR for 20% reactor power.

b. Increasing pressure of the RCS, when operating in the nucleate boiling region of the heat transfer curve will decrease the heat transfer rate in (BTU/hr-square foot).

QUESTION 5.08 (1.00)

TRUE or FALSE?

a. During a RCS heatup, as temperature gets higher, it will take a smaller letdown flow rate to maintain a constant Pressurizer level.

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

QUESTION 5.09 (1.00)

The reactor is operating at 100 % power, all-rods-out, near end of cycle prior to a scheduled power reduction to 50 % power for a surveillance. The Unit Operator observes AFD (delta I) to be in the doghouse and decides to lower power and temperature by borating, while leaving rods fully withdrawn. Actual Tavg follows the programmed Tavg.

Which of the following best describes the AFD initial change?

a. AFD will change in the POSITIVE direction; because relatively more POSITIVE reactivity is added to the top half of the core.

b. AFD will change in the NEGATIVE direction; because relatively more NEGATIVE reactivity is added to the top half of the core.

c. AFD will not change because rods have not moved.

d. AFD will change in the NEGATIVE direction; because relatively more NEGATIVE reactivity is added to the bottom half of the core.

QUESTION 5.10 (1.00)

Which of the below choices does a sufficient Shutdown Margin NOT ensure?

a. The reactor can be made subcritical from all operating conditions.

b. The reactor will not be made critical with the Reactor Coolant System average temperature less than 541 degrees F.

c. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits.

d. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

QUESTION 5.11 (.75)

For each condition in COLUMN A, find the correct heat transfer equation in COLUMN B that would be used to calculate the heat transferred.

COLUMN A	COLUMN B
a. Across the reactor (cold leg to hot leg)	1. $\dot{Q}$ = UA delta T
b. Across S/G U-tubes (primary to secondary)	2. $\dot{Q} = \dot{M}$ delta T
c. Across S/G secondary side (feedwater to steam)	3. $\dot{Q} = \dot{M}$ Cp delta T
	4. $\dot{Q} = \dot{M}$ delta H
	5. $\dot{Q}$ = UA delta H

QUESTION 5.12 (.75)

Which of the following is a "tell-tale" sign that the Point of Adding Heat has been reached?

a. Decrease in the average temperature.

b. Increase in the Pressurizer level.

c. Increase in the Start-up rate.

d. Decrease in the Reactor Coolant System pressure.

QUESTION 5.13 (.50)

TRUE or FALSE?

Convective heat transfer capability is increased by decreasing coolant flow velocity.

QUESTION 5.14 (1.00)

Heat Flux Hot Channel Factor, RCS Flowrate and Nuclear Enthalpy Rise Hot Channel Factor are power distribution limits and are determined periodically. This periodic surveillance is sufficient to ensure the limits are maintained provided four conditions are met. One condition to be met is the control rod insertion limits are maintained.

Which of the following choices is NOT one of the other three conditions?

a. The reactor shall not be made critical with a positive MTC.

b. Control rods in a single group are to move together with no single rod differing by more than 13 steps from the group demand position.

-> c. Control rods are to be sequenced with proper overlapping groups.

d. Axial power distribution (AFD) is to be maintained within limits.

QUESTION 5.15 (.50)

TRUE or FALSE?

Increasing the RCS boron concentration causes the thermal utilization factor to increase, therefore the MTC becomes less negative.

QUESTION 5.16 (.50)

TRUE or FALSE?

Thermal neutron flux is higher in the fuel rod than in the moderator.

QUESTION 5.17 (.50)

TRUE or FALSE?

The use of a sliding Tavg program provides plant operation with a higher thermodynamic efficiency than does operation with a constant Tavg program.

QUESTION 5.18 (1.50)

TRUE or FALSE?

a. One of the pump laws for centrifugal pumps states that the volume flow rate is proportional to the speed of the pump.

b. As VCT temperature decreases, volume flow rate from the positive displacement (PD) pump increases.

c. Pump runout is the term used to describe the condition of a centrifugal pump running with no volume flow rate.

QUESTION 5.19 (1.00)

Which of the below describes the inverse multiplication plot (graph)?

	The Vertical Axis	The Horizontal Axis
a.	initial count rate	final count rate
b.	initial count rate divided by final count rate	control rod reactivity (No. of assemblies)
c.	control rod reactivity (No. of assemblies)	final count rate divided by the initial count rate
d.	final count rate divided by initial count rate	control rod reactivity (No. of assemblies)

QUESTION 5.20 (1.00)

Which of the below items will cause the control rod differential rod worth to increase?

a. An increase in the boron concentration.

b. An increase in the Moderator temperature.

c. A decrease in the neutron fast flux.

d. An increase in the fission product concentration.

QUESTION 5.21 (1.00)

Which of the below choices correctly completes the following statement?

Assuming all other factors are identical, the MASS flow rate of fluid through a 10 inch diameter pipe will be approximately ____________ times as great as the MASS flow rate through a 2 inch diameter pipe.

a. 2.5

b. 5.0

c. 12.5

d. 25.0

QUESTION 5.22 (1.00)

Which of the following nuclides is NOT a fissile nuclide, i.e. fissionable by a thermal neutron, during power operations?

a. Uranium 235

b. Uranium 238

c. Plutonium 239

d. Plutonium 241

QUESTION 5.23 (1.00)

To what approximate pressure must the steam generator pressure be reduced to maintain a 200 degree F subcooling margin in the RCS when reducing RCS pressure to 1600 psig?

a. 845 psig.

b. 645 psig.

c. 445 psig.

d. 245 psig.

QUESTION 5.24 (1.00)

From which of the below choices does the majority of Tritium originate?

a. Directly from fission as a fission fragment.

b. Activation of Deuterium by a neutron.

c. Activation of Lithium by a neutron.

d. Activation of Boron by a neutron.

QUESTION 5.25 (1.00)

If RCP's are tripped following a LOCA, and the break has been isolated, which of the following situations would be MOST desirable?

	PZR Press.	HOT LEG Temp.	COLD LEG Temp.
a.	600	500	480
b.	800	530	520
c.	1000	540	530
d.	1200	575	565

QUESTION 5.26 (.50)

TRUE or FALSE?

The Doppler only power coefficient (PCM/%power) at hot full power becomes more negative during the life of the core (BOL to EOL).

QUESTION 5.27 (1.00)

TRUE or FALSE?

a. As Keff approaches unity, a smaller change in neutron level will result for identical changes in Keff.

b. As Keff approaches unity, a longer period of time is required to reach the equilibrium neutron level for identical changes in Keff.

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

QUESTION 5.28 (1.00)

TRUE or FALSE?

a. The differential temperature necessary to transfer heat is inversely proportional to heat flux.

b. The latent heat of vaporization is another term for the latent heat of condensation.

QUESTION 5.29 (.50)

TRUE or FALSE?

High energy neutron exposure increases the possibility of brittle fracture of the reactor vessel by increasing the compressive stress on the reactor vessel.

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

QUESTION 6.01 (1.00)

The purpose of the CVCS demineralizers is to:

a. Remove all chemicals from the RCS fluids.

b. Remove soluable and insoluable material from the RCS.

c. Replace insoluable material with soluable ions.

d. Provide a method for boron control during reactor operations.

QUESTION 6.02 (1.00)

Near the end of a plant cooldown how is the Pressurizer (PZR) volume cooled after the RCPs are secured?

a. Ambient losses determine the cooling of the PZR.

b. RCPs are to be run for 5 minutes, every 15 min. to provide spray flow.

c. PZR level is raised and vapor is vented to the PRT.

d. Aux spray path is used from the regen heat exchanger outlet.

QUESTION 6.03 (1.00)

Which of the below features enhances the operation of the ice condenser and containment spray for heat removal?

a. Containment design, such that the delta P between upper and lower containment drives the air circulation.

b. Ventilation coolers and recirculation fans are used to mix the air and provide additional cooling.

c. Air return fans provide flow to return the air from the upper containment to the lower containment.

d. Pressure operated doors open to allow upper containment air to flow through to the lower containment.

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

QUESTION 6.04 (.50)

TRUE or FALSE?

Increasing RCS pressure while solid on RHR is most effectively and and quickly accomplished by increasing the temperature control setting of the RHR Heat Exchanger outlet temperature controller.

QUESTION 6.05 (1.00)

Why are the protection bistables associated with a failed instrument placed in a tripped condition?

a. Ensures that the required protection coincidence will be met.

b. Prevents inadvertant tripping of the plant during repair work.

c. Ensures that a failed channel will not cause a trip when another valid signal is present.

d. Ensures adequate plant reliablity is maintained by ensuring that two additional signals are needed to trip.

QUESTION 6.06 (.50)

TRUE or FALSE?

After tripping a bistable in a 2/4 logic system, one of three remaining signals reaching the bistable setpoint will cause a trip, even though the logic SYSTEM remains as a 2/4 system.

QUESTION 6.07 (1.00)

Which of the following protects the Upper Head Injection Accumulator from overpressurization, if inleakage or temperature changes occurred during normal operations?

a. Relief valves

b. Surge tank

c. Rupture disk

d. Alarms for operator action

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

QUESTION 6.08 (1.00)

What interlock must be satisfied prior to opening the containment sump suction valves to the RHR system?

a. RHR pump must be off.

b. Low alarm on RWST.

c. RHR discharge valves shut.

d. RWST valves shut.

QUESTION 6.09 (1.50)

Refer to the Logic Diagram on Figure 6-1, on the next page, to answer the following.

a. What are the setpoint values for the items labeled "A"?

b. What is the coincidence of the bistables labeled "B"?

c. What is the P-11 permissive setpoint labeled "C"?

QUESTION 6.10 (1.00)

The plant is at 100% power and stable. A maintenance person inadvertently trips the turbine but the reactor does not trip. After 30 seconds the AUO de-energizes the rod drive MG sets. Assume no further operator action.

Which of the below is the response of the plant?

a. Steam dump arms, all valves trip open and Tavg is reduced to Tref no-load setpoint.

b. Steam dumps open, turbine trip controller reduces Tavg to Tref plus the 2 degrees F dead band (549 degrees).

c. Steam dump arms and valves will open as a result of a signal from the load rejection controller and reduces Tavg to Tref no-load plus the 2 degrees dead band (549 degrees).

d. Steam dump arms and valves will open as a result of a signal from the load rejection controller until the rods drop, at which time the reactor trip controller reduces Tavg to Tref.

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

LOW PRESSURIZER PRESSURE ESF ACTUATION LOGIC

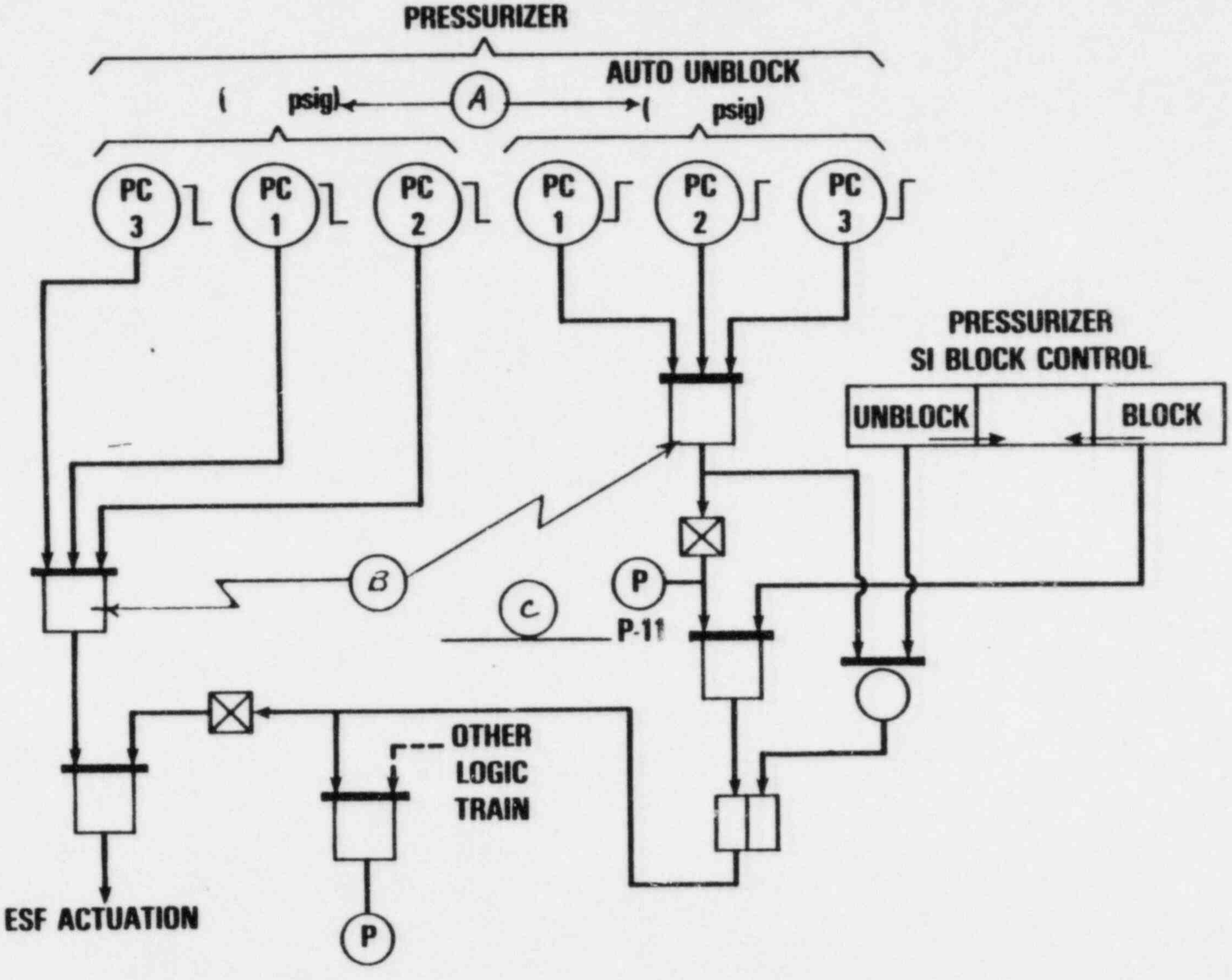


FIGURE 6 - 1

QUESTION 6.11 (1.00)

Which statement is correct concerning the level control valves on the Motor Driven Auxiliary Feedwater Pump discharge?

a. They are hydraulic operated by the pump and reservoir mounted locally.

b. They fail closed.

c. When pressure downstream of the valves drops to 500 psig, they will close.

d. Can be operated from the local control panel, just outside the Terry Turbine Room.

QUESTION 6.12 (1.00)

Which of the below choices correctly completes the following statement?

The Turbine Driven Auxiliary Feedwater Pump ______________________________.

a. will trip on a thermal overload of its trip/throttle valve.

b. can operate properly on 50 psig steam.

c. will automaticly start if one main feedpump trips at 60% power.

d. suction valves will transfer to ERCW when suction pressure drops to 23 psig. for 2 seconds.

QUESTION 6.13 (1.00)

The controlling Pressurizer (PZR) level channel (459) fails high during 100% power operation. Assuming NO operator action is taken, which of the following best describes the response of the plant?

a. Charging flow goes to minimum, PZR level decreases, letdown isolates and the plant continues to operate at the same power.

b. Charging flow goes to minimum, PZR level decreases, letdown isolates and the plant eventually trips on high PZR level.

c. Charging flow goes the maximum, PZR level increases and the plant trips on high PZR level.

d. Charging flow remains the same, PZR level increases due to letdown isolating and the plant trips on high PZR level.

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

QUESTION 6.14 (1.00)

Besides the overspeed shutdown, which of the following diesel engine/ generator shutdowns is enabled during an emergency start of the diesel?

a. Voltage restraint overcurrent relay, (51V).

b. Generator differential relay, (87).

c. Phase balance relay, (46).

d. Low lube oil pressure.

QUESTION 6.15 (1.00)

Which statement describes the signal path from the Source Range detector to the Source Range level meter on the main control board?

a. Detector, Pre Amp, Discriminator, Log Integrator, Meter

b. Detector, Log Integrator, Pulse Shaper, Pulse Counter, Meter

c. Detector, Pre Amp, Log Integrator, Discriminator, Meter

d. Detector, Log Amp, Meter

QUESTION 6.16 (1.00)

Which statement concerning the Power Range Nuclear Instrument detectors is CORRECT?

a. They are uncompensated ion chambers that operate in the proportional region of the gas amplification (detector characteristics) curve.

b. Uses compensation inner chamber current to cancel out gamma current measured in the outer chamber.

c. Uses Boron Trifluoride gas to make it neutron sensitive.

d. Its detector current is calibrated using the data from a secondary heat balance.

QUESTION 6.17 (1.00)

Using the following 7 actions, which of the below sequences is the correct sequence for a rod withdrawl?

Seven Rod Control Actions

1. Movable gripper coil energized.
2. Stationary coil energized.
3. Lift coil energized.
4. Stationary gripper energized.
5. Movable coil DEENERGIZED.
6. Stationary coil DEENERGIZED.
7. Lift coil DEENERGIZED.

Sequences to Choose From

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

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

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

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

QUESTION 6.18 (1.00)

Which statement concerning the Rod Control System is CORRECT?

a. The power cabinet provides AC power pulses to drive the control rod drive mechanism.

b. The reactor control unit generates a rod speed and direction signal in response to three ERROR signals.

c. Turbine impulse pressure provides signals to the rate comparator, summing unit and the variable gain unit in the rod control circuits.

d. Rod power is supplied by two motor generator sets with a 260VDC output through an isolation transformer.

QUESTION 6.19 (1.00)

What are TWO interlocks that would prevent the automatic transfer of a 6.9 KV Shutdown Board from its normal supply to alternate supply?

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

QUESTION 6.20 (1.00)

On a sustained (greater than 1.5 seconds) loss of voltage to a 6.9 kV Shutdown Board, what is the sequence of events?

a. The diesel is started, and in 2.0 seconds the bus is stripped, then the diesel breaker shuts upon reaching 800 RPM and 6.9 kV.

b. The bus is stripped, except the 480v shutdown transformers, and in an additional 3.5 seconds, the diesel is started and its breaker shuts upon reaching 6.9 kV and 900 RPM.

c. 3.5 seconds later the diesel is started with the bus being completely stripped and then the diesel breaker shuts within the following 10 seconds.

d. The diesel is started, then after an additional 3.5 seconds, the bus is stripped except the 480 v shutdown board transformers, and the diesel breaker shuts upon reaching 900 RPM and 6.9 kV.

QUESTION 6.21 (1.00)

With normal power unavailable and ONE vital battery out of service, how long will the remaining THREE batteries be capable of supplying all loads required for safe shutdown of BOTH units?

a. 30 minutes.

b. 1 hour

c. 1.5 hours

d. 2 hours

QUESTION 6.22 (1.00)

TRUE or FALSE?

a. A Diesel Generator should never be isolated on a Shutdown Board during surveillances.

b. If a Diesel Generator was loaded at 40 % for four hours, it is then permissable to shutdown the Diesel without further loading until the next surveillance.

QUESTION 6.23 (2.00)

List the automatic actions that occur if one Main Feed Pump trips when the plant is operating above 80 % power.

QUESTION 6.24 (1.00)

What action will occur if the #3 heater drain bypass to the condenser valve leaves its fully closed position with plant at 95% power?

a. Standby drain pump starts.

b. Main Turbine runback to 85%.

c. Main Turbine runback to 60%.

d. Main Feed Regulating (Control) BYPASS valves open fully.

QUESTION 6.25 (2.00)

For the following components, indicate whether they will receive an OPEN, CLOSE, or NO signal as a result of a safety injection (with Phase 'A') initiation signal.

a. Control room supply ducts
b. Main feed bypass valves
c. SI accumulator discharge isolation valves
d. Normal charging header isolation valves
e. Main steam isolation valves
f. RWST to SI pump suction valves
g. Seal water return isolation valve
h. Component cooling isolation valve from RHR system
i. Component cooling isolation from letdown heat exchanger
j. Steam supply valves to turbine-driven feed pump

QUESTION 6.26 (1.00)

Considering only the Steam Generator Level Control System, what would be the response of the INITIAL feedwater flow to the S/G if the controlling S/G pressure transmitter failed LOW during 50% power operations?

a. The flow would decrease due to the loss of the steam pressure input to the steam flow signal.

b. The flow would remain the same due to the steam pressure not affecting the steam flow.

c. The flow would increase due to the steam pressure input to the feed control valve position controller.

d. The flow would increase due to the loss of steam pressure input to the steam flow signal.

QUESTION 6.27 (.50)

TRUE or FALSE?

The Pressurizer PORV arming temperature setpoint for Unit 2 was recently changed to that of Unit 1, i.e. 380 degrees F.

QUESTION 6.28 (1.00)

Which of the following will cause a trip of a running Main Feedwater Pump?

a. Low feedwater temperature.

b. Low Main Feed Pump turbine speed.

c. Recirculation valve open.

d. Safety Injection.

QUESTION 6.29 (1.00)

If an unsaturated bed of H-OH resin is placed in service, what will be the result?

a. RCS Oxygen concentration will increase.

b. No ion exchange will occur for the first 12 hours of operation.

c. RCS Boron concentration will decrease.

d. RCS Boron concentration will increase.

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

QUESTION 7.01 (1.00)

GOI-1 gives a precaution limiting the opening of the RCP seal bypass return valve, 62-53. It must be open if any #1 seal return flow is less than 1gpm and the RCS temperature is high enough to cause the lower radial bearing temperature to exceed the alarm point.

What is the other limitation required prior to opening 62-53?

a. CVCS injection to the seals are at least 6 gpm.

b. RCS pressure is at least 100 psig.

c. The RCP thermal barrier flow is normal.

d. The RCP #1 seal delta Pressure is less than 400 psid.

QUESTION 7.02 (1.00)

If the reactor trip breakers are closed and the steam generators are under nitrogen pressure, the nitrogen pressure must be vented off the steam generators prior to opening the MSIV's.

Why must this be done?

a. To prevent SIS actuation on steam generator high delta Pressure.

b. To prevent damage to the MSIV seats.

c. To prevent ESFAS actuation on high steamline flow.

d. To prevent EST actuation on steam generator lo-lo level.

QUESTION 7.03 (1.00)

When the RCS pressure is below 500 psig, why is a centrifugal charging pump required to be used instead of the positive displacement pump?

a. The smoother flow from the centrifugal pump is desired instead of the pulsation flow of the positive displacement pump.

b. The flow from the centrifugal pump can be regulated to a lower value than the flow from the positive displacement pump.

c. The seal injection flow is easier to control with the centrifugal pump than the positive displacement pump.

d. One centrifugal pump must be tagged out for overpressurization reasons in case of an SI and the second is desired to be running.

QUESTION 7.04 (1.00)

The control rods were withdrawn 5 steps to prevent "thermal lock-up" during RCS heatup. If the control rods were NOT fully inserted using bank-select prior to withdrawing rods using manual, what would be the result according to GOI-2, "Plant Startup from Hot Standby to Minimum Load"?

a. Error in rod height for the ECP critical data.

b. Rod bottom lights malfunction.

c. Rod bank overlap malfunction.

d. Rod upper limit stop malfunction.

QUESTION 7.05 (1.00)

According to a note in GOI-2, what condition must be met prior to exceeding 600 RPM on the main turbine?

a. Main Feedwater Regulating valves are to be in automatic.

b. Tavg is to be at the no-load value.

c. The low pressure turbine inlet metal temperature must be greater than 400 degrees F.

d. Steam dumps must be in Tavg mode.

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

QUESTION 7.06 (2.50)

TRUE or FALSE?

a. When the axial flux difference monitor is inoperable, the AFD must be logged once a shift by performing SI-44.

b. If the loop boron concentration is changed by 10 ppm or greater, pressurizer sprays will be actuated by manual operation of sprays.

c. Any off-frequency turbine operation is to be reported to the results section for record keeping.

d. If the "Rod Control Banks Limit Low" alarm comes in when critical, commence boration to clear the alarm.

e. When the quadrant power tilt ratio alarm is inoperable, the QPTR must be calculated every 12 hours by performing SI-133.

QUESTION 7.07 (2.00)

For the power levels in Column A, find the one associated conditions or actions in Column B, as stated in GOI-5A.

COLUMN A	COLUMN B
a. 20 %	1. P-8 light goes out.
b. 30 %	2. P-9 light goes out.
c. 35 %	3. Observe turbine startup drains closed.
d. 50 %	4. Open HP drains to the No. 1 heater shells.
	5. Start two condensate demin pumps.
	6. Prior to exceeding _________ % power, steam generator chemistry must be below the limits for exceeding this specific power level.

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

QUESTION 7.08 (1.00)

In accordance with ES-1.2, "Transfer to RHR Containment Sump", what is the guidance for performing all actions?

a. Slowly and deliberately with time taken to analyze the proceeding steps.

b. Complete each step as rapidly as possible unless action does not take place, then complete corrective measures.

c. Stop all running pumps, then transfer the valve lineup per the procedure.

d. Quickly in a precise, orderly sequence, without interruption of changeover operation until all actions are completed.

QUESTION 7.09 (1.00)

After going into the recirculation mode, at what RWST level should the Containment Spray Pumps suction be realigned to the containment sump?

a. 0%

b. 8%

c. 15%

d. 25%

QUESTION 7.10 (2.00)

LIST the THIRTEEN immediate actions to be taken for a Saftey Injection, in accordance with Emergency Procedure, E-0.

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

QUESTION 7.11 (1.00)

What is the maximum quantity of fuel (assemblies) that shall be allowed out of approved storage locations at any one time during fuel-handling operations?

a. 1
b. 3
c. 6
d. 8

Disregard & put your answer for the newly revised procedure

QUESTION 7.12 (2.00)

Match the whole body radiation exposure terms in Column A to their limit in Column B.

CAUTION: Some answers could be used more than once.

COLUMN A	COLUMN B
a. 10CFR20 limit/qtr without NRC Form 4	1. 0.3 REM
b. TVA limits/qtr for TVA (occupational) workers	2. 1.25 REM
	3. 0.75 REM
c. TVA limit/qtr for Non-TVA personnel without their history	4. 5.0 REM
d. 10CFR20 limit/qtr with an NRC Form 4	5. 3.0 REM

[0.5 each]

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

QUESTION 7.13 (1.00)

What group of THREE indications below are used to monitor RCS cooldown during natural circulation according to ES-0.3, "Natural Circulation Cooldown"?

a. Tcold (NR), Pzr level, Core exit TC's.

b. Thot (NR), RCS pressure, RCS subcooling.

c. Thot (WR), Core exit TC's, RCS subcooling.

d. Tcold (WR), Pzr pressure, Thot (NR).

QUESTION 7.14 (1.00)

GOI-2, "Plant Startup from Hot Standby to Minimum Load" states that the shutdown banks must be at the fully withdrawn position whenever positive reactivity is being inserted.

When can exceptions to this rule be applied?

a. When the Shutdown Margin has been calculated to be 900 pcm.

b. When the RCS has been borated to the cold shutdown concentration and the plant is being cooled down.

c. When the reactor is in the source range with the High Flux at Shutdown alarm operable.

d. When the actual boron concentration is greater than the predicted critical boron concentration.

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

QUESTION 7.15 (1.50)

Are the three statements below TRUE or FALSE?

ASSUME the plant (RCS) is solid with pressure being maintained by the low pressure letdown valve, FCV-62-81, in automatic.

a. The stopping of an RHR pump will cause a decrease in RCS pressure.

b. If the RHR system pressure exceeds 700 psig, the RHR suction valves from Loop 4 hot leg will close.

c. With RCS pressure at 300 psig (no steam bubble in the pressurizer), it is permissible to isolate the RHR suction line from the RCS.

QUESTION 7.16 (1.00)

During normal CVCS operation, which of the following is an abnormal condition and would require operator action to correct?

a. VCT pressure is 15 psig.

b. The temperature of the fluid leaving the letdown heat exchangers is 127 F.

c. The RCP seal injection water tempeature is 120 F and flow to the seals is 8 gpm/pump.

d. RCP seal differential pressure is 250 psid.

QUESTION 7.17 (1.50)

TRUE or FALSE?

a. The transfer of ECCS suction to the containment sump is accomplished when RWST level is greater than 29 %.

b. When the RWST level reaches 0 %, ~~the~~ all equipment taking suction on the RWST ~~are~~ stopped. is automatically stopped.

c. Transfer to hot leg recirculation is accomplished 15 hours after initiation of Safety Injection.

QUESTION 7.18 (.50)

TRUE or FALSE?

Areas where dose rates are 500 mr/hr, are required to be locked and access controlled.

QUESTION 7.19 (1.00)

Which of the following statements concerning the procedure for a dropped RCCA is correct?

a. Upon starting recovery of the dropped RCCA, an URGENT FAILURE alarm will occur because the lift coils for the other rods in the group have been disconnected.

b. The delta flux target band is not applicable during a dropped RCCA malfunction and recovery.

c. If two or more RCCA's have dropped, manually trip the reactor and proceed in accordance with EP-1.00.

d. Recovery from a dropped RCCA will be facilitated if Tavg is higher than Tref prior to commencing withdrawal of the dropped RCCA.

QUESTION 7.20 (1.00)

Following a reactor trip, how many gallons must be emergency borated for each control rod not fully inserted?

a. 350

b. 400

c. 450

d. 500

QUESTION 7.21 (.50)

TRUE or FALSE

Fuel Handling equipment interlocks shall only be bypassed by approval of, and under the direct supervision of, the fuel handling SRO.

QUESTION 7.22 (1.00)

At least ___(how many)___ steam generator(s) must be maintained available for RCS cooldown in accordance with procedure E-2, "Faulted Steam Generator Isolation".

QUESTION 7.23 (1.00)

To terminate SI with adverse containment conditions, the pressurizer level must be greater than ______% in accordance with ES-0.2.

a. 20

b. 30

c. 40

d. 50

QUESTION 7.24 (.50)

TRUE or FALSE?

During a steam generator tube rupture and subsequent controlled RCS depressurization, the RCP Trip Criteria can be disregarded.

depressurization

QUESTION 7.25 (1.00)

In accordance with the "Loss of Reactor or Secondary Coolant" procedure, E-1, if one charging pump is operating and RCS pressure is uncontrollably decreasing, at what RCS pressure must the Reactor Coolant Pumps be stopped?

a. 1450 psig.

b. 1350 psig.

c. 1250 psig.

d. 1150 psig.

QUESTION 7.26 (1.00)

Which of the following conditions would REQUIRE fuel shuffle operations be immediately stopped? Assume the initial nucleus of ten assemblies are loaded, AND exclude ANTICIPATED change in count rates due to detector and/or source movement.

a. An increase in count rate by a factor of 2 on ANY nuclear channel or by a factor of 1.5 on ALL nuclear channels during any single loading step.

b. An increase in count rate by a factor of 3 on ANY nuclear channel or by a factor of 1.5 on ALL nuclear channels during any single loading step.

c. An increase in count rate by a factor of 4 on ANY nuclear channel or by a factor of 2 on ALL nuclear channels during any single loading step.

d. An increase in count rate by a factor of 5 on ANY nuclear channel or by a factor of 2 on ALL nuclear channels during any single loading step.

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

QUESTION 8.01 (1.00)

Which of the below classifications of drawings is permitted to be used in the Main Control Room?

a. Information Only.

b. As-Designed.

c. Workplan Drawing Copy.

d. Safeguard Information.

QUESTION 8.02 (1.00)

Concerning AI-30 Procedure, "Nuclear Plant Method of Operation", which one of the below statements is correct?

a. Under emergency conditions, a licensed reactor operator may APPROVE reasonable action that departs from a license condition or Technical Specifications.

b. The operator is required to have Category "B" instructions present when performing work.

c. There is no specific written guidance concerning the repeat back of verbal communications, however it is a good operating practice.

d. Permission shall be received from the Lead Unit Operator prior to the performance of any maintenance, test, or modification activity on, or that may affect, plant equipment.

QUESTION 8.03 (1.00)

Which statement is correct about independent verifications?

a. Independent verification shall be provided for return to service of equipment that is placed in an "off normal" configuration for the operating mode to allow surveillance testing to be done.

b. When clearances are hung independent verification is required for the placing of tags and not for the removing of tags.

c. Independent checks on remote, hard to reach equipment can be verified by the second person by verbal report from the first person.

d. Bypassing independent verifications can be done by the Shift Engineer if an "after-the-fact" review is submitted by a licensed SRO.

QUESTION 8.04 (2.00)

List FIVE circumstances which require direct notification within one hour to the NRC Operations Center via the ENS.

QUESTION 8.05 (3.00)

For the items in Column A, identify the proper color code from Column B for plant instruments and common equipment.

COLUMN A	COLUMN B
a. Unit 0	1. Red
b. Unit 1	2. Blue
c. Unit 2	3. Green
d. Alarm point (not trip)	4. Black
e. Trip point	5. Yellow
f. Operating Band	6. Orange
	7. White
	8. Brown

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

QUESTION 8.06 (1.00)

Which of the following is a direct responsibility of the Assistant Shift Engineer (Shift Supervisor) as stated in the AI-2 Procedure?

a. Serves as site emergency director until relieved as specified in the Radiological Emergency Plan.

b. Can exercise control over any action which could affect reactivity of the reactor for which he is responsible.

c. It is his responsibility to analyze the cause and determine if operation can continue safely before returning to power.

d. Responsible for the safe and correct operation of all electrical boards and maintains a constant source of power to the equipment and controls of the unit that he/she is assigned to.

QUESTION 8.07 (1.00)

Before a person may be placed on the official clearance list he/she must complete which of the below items?

a. An oral checkout with the shift engineer.

b. A written exam with the grade of at least 70%.

c. A 3.0 hour training course on clearances.

d. Two complicated clearances under supervision of an approved person.

QUESTION 8.08 (1.00)

When utilizing the Emergency Clearance procedure, who is responsible for all work performed and the safety of all workers involved in the clearance?

a. The Shift Engineer.

b. The person clearing the equipment.

c. The Plant Manager.

d. The Emergency Director.

QUESTION 8.09 (1.00)

Who should review a Temporary Alteration (TA), to determine the need for specific training or for formal information for the licensed operators to assure their awareness of the TA and its implications?

a. NRC, Region II.

b. The Unit ASE.

c. The Shift Technical Advisor.

d. The PORC.

QUESTION 8.10 (1.00)

From the choices below pick the term that best completes the following statement.

Only _________________ changes should be installed using plant instructions or Maintenance Requests (MR's).

a. long-term

b. PORC-approved

c. SQA-approved

d. short-term

QUESTION 8.11 (1.00)

From the choices below pick one that best completes the following statement

A trip after a long period of reactor shutdown leaves little decay heat to be removed thus causing the possibility of excessice cooling of the reactor coolant if too much feedwater is being added. The operator should NEVER restore the steam generator water level, after a plant trip, at the cost of a reduction of the _________________________________.

a. plant pressure

b. shutdown margin

c. CST level

d. steam generator pressure.

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

QUESTION 8.12 (1.50)

List the THREE reactor trip CIRCUITS that may be administratively BYPASSED for maintenance on a single channel.

QUESTION 8.13 (1.00)

What is the minimum Technical Specification quadrant power tilt ratio (QPTR) which requires actions to be taken when operating at 75% power?

a. 1.02.

b. 1.03.

c. 1.05.

d. 1.09.

QUESTION 8.14 (1.00)

During cold shutdown, Technical Specifications require the two RHR loops to be OPERABLE, with a note modifying this requirement to allow substitution of certain equipment for one of the RHR loops.

Which of the below groups of equipment can be substituted for the one RHR loop?

a. Four filled RCS loops with one Safety Injection Pump and RWST level > 50%.

b. Four filled RCS loops with one OPERABLE RCPs and one Safety Injection pump.

c. Four filled RCS loops with two OPERABLE RCPs and one OPERABLE auxiliary feed pump with CST level > 50%.

d. Four filled RCS loops with at least two steam generators having levels >= 10% WR.

QUESTION 8.15 (3.00)

Match the condition in Column A to the maximum system leakage as stated in Technical Specifications from Column B.

COLUMN A	COLUMN B (gpm)
a. controlled leakage	1. Zero
b. RCS pressure isolation valves	2. 0.5
c. Primary-to-secondary	3. 1.0
d. Pressure boundary leakage	4. 5.0
e. Identified leakage	5. 10.0
f. Unidentified leakage	6. 20.0
	7. 40.0

QUESTION 8.16 (1.00)

During refueling operations it is discovered that the Keff is greater than 0.95 but the boron concentration is at 2050 ppm.

What action must be taken, in addition to suspending all core alterations?

a. Borate at greater than or equal to 10 gpm with a solution greater than or equal to 20,000 ppm boron, until Keff is less than 0.95.

b. No action required since the boron concentration is greater than 2000 ppm.

c. Borate at 10 gpm with 20,000 ppm boron until the boron concentration is 100 ppm higher than previous concentration.

d. Notify the reactor engineers and immediately recalculate the Keff value.

QUESTION 8.17 (1.00)

How are loads in excess of 2000 pounds prohibited from travel over fuel assemblies in the storage pool when suspended from the Spent Fuel Area Crane?

a. All loads are less than limit except fuel and exceptions are made for the fuel movement.

b. The area is restricted by adminstrative control.

c. The crane is prevented from moving in the area by interlocks and stops.

d. The crane is not rated for loads in excess of 2000 pounds.

QUESTION 8.18 (1.00)

What actions shall take place in the event a Safety Limit is violated?

a. Place unit in HOT SHUTDOWN in 1 hour and notify NRC in 24 hours.

b. Place unit in COLD SHUTDOWN within 30 hours and notify NRC in 6 hours.

c. Place unit in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within 30 hours and notify the NRC within 24 hours.

d. Place unit in HOT STANDBY within 1 hour and notify the NRC within 1 hour.

QUESTION 8.19 (1.00)

What is the MINIMUM crew composition (SS, SRO, RO, AC, STA) as defined in the Technical Specification when both units are operating in mode 1?

a. 7

b. 9

c. 10

d. 12

QUESTION 8.20 (1.00)

Each time the main control room drawing files are updated to provide new "as-constructed" drawings, with which of the below labels are the new drawings identified?

a. CONTROL COPY

b. SAFEGUARDS INFORMATION

c. FIELD CONTROL COPY

d. THIS DRAWING EXPIRES AFTER

QUESTION 8.21 (1.00)

Whose approval is required prior to the implementation of Handwritten Instructions?

a. SE-SRO and Operations Supervisor

b. ASE-SRO and UO

c. ASE-SRO and SE-SRO

d. SE-SRO and the preparer

QUESTION 8.22 (1.00)

Whenever a condition requires a temporary (one day) deviation from normal system alignement, these deviations are to be tracked by entries in which of the below logs?

a. Status Log.

b. Deviation Log.

c. Configuration Log.

d. Surveillance Log.

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

QUESTION 8.23 (1.00)

Who has the responsibility for initiation of a hold order clearance on the incore flux drive motor control power, prior to personnel entering into the lower containment or annulus?

a. Health Physics Representative

b. Shift Engineer

c. Public Safety Officer

d. ASE-SRO

QUESTION 8.24 (.50)

TRUE or FALSE?

The inoperability of vital inverter 2-II is cause for the Unit 1 Technical Specification LCO action statement to be entered, when Unit 1 is in mode 1.

QUESTION 8.25 (1.00)

If a reactor trip was caused by personnel error, who (by job title) is allowed to grant approval for the subsquent startup of the reactor? LIST ALL APPLICABLE JOB TITLES.

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

ANSWER 5.01 (2.00)

a. 3.

b. 2.

REFERENCE
SQNP, Q & A Bank , sec 1-11.
SQNP, Review of Neutron Kinetics Lesson Plan, p d.

ANSWER 5.02 (1.00)

Curve 1. - B or D

Curve 2. - C

Curve 3. - D

Curve 4. - A

REFERENCE
1. WNT, Chapter 3, pp 3-44 to 3-53.
2. Cycle 3 information available at review.

ANSWER 5.03 (1.00)

c.

REFERENCE
SQNP, T.S. 3.1.1.3; AND Review of Reactivity Coefficients Lesson Plan, p 4.

ANSWER 5.04 (1.00)

b.

REFERENCE
SQNP, 6 Factor Formula Lesson Plan, pp 3 - 5.

ANSWER 5.05 (4.00)

a. 1.

b. 3.

c. 3.

d. 4.

REFERENCE
SQNP, Review of core poisons lesson, p. 6

ANSWER 5.06 (1.00)

c.

REFERENCE
WNTC, HTFF, Chapter 12, p 15.

ANSWER 5.07 ~~(1.00)~~ 0.5

a. False.

b. True. False for Steady State therefore not graded

REFERENCE
SQNP, HTFF text p. 202

ANSWER 5.08 (1.00)

a. FALSE

b. TRUE [0.5 ea.]

REFERENCE
General Physics, HT&FF, pp. 155 and 320 and Subcooled Liquid Density Tables

ANSWER 5.09 (1.00)

a.

REFERENCE
WNTC, Reactor Control, Chapter 8. pp 19-22.

ANSWER 5.10 (1.00)

D.

REFERENCE
Tecnnical Specifications, B 3/4 1-1.

ANSWER 5.11 (.75)

a. 3. or 4.

b. 1.

c. 4.

REFERENCE
SQNP, HTFF text, pp. 174 -185

ANSWER 5.12 (.75)

b.

REFERENCE
WNTC, Reactor Control, Chapter 9, p 17.

ANSWER 5.13 (.50)

False.

REFERENCE
SQNP, HTFF text, pp 169-170.

ANSWER 5.14 (1.00)

a.

REFERENCE
SQNP, Technical Specifications, p B3/4 2-2.

ANSWER 5.15 (.50)

False.

REFERENCE
SQN, Review of Reactivity Coefficients Lesson Plan, pp 4 - 10.

ANSWER 5.16 (.50)

False.

REFERENCE
SQN, Review of MNeutron Kinetics Lesson Plan, pp 6 - 8.

ANSWER 5.17 (.50)

TRUE.

REFERENCE
SQN, HT,FF,THERMO, Lesson Plan, pp 12 - 15.

ANSWER 5.18 (1.50)

a. True.
b. False.
c. False.

REFERENCE
SQNP; HTFF, pp 16, 17 and 19.

ANSWER 5.19 (1.00)

b.

REFERENCE

SQN, Neutron Sources and Subcritical Multiplication Lesson Plan, pp 4 - 6.

ANSWER 5.20 (1.00)

b.

REFERENCE
SQN, Review of Core Poisons Lesson Plan, pp 3 - 5.

ANSWER 5.21 (1.00)

d.

REFERENCE

SQN, HT,FF,THERMO, pp 15 - 17.

ANSWER 5.22 (1.00)

b.

REFERENCE
SQN, Review of Basic Nuclear Concepts Lesson Plan, p 4.

ANSWER 5.23 (1.00)

d.

REFERENCE

Steam Tables.

ANSWER 5.24 (1.00)

d.

REFERENCE
WNTO, Radiation, Chemistry and Corrosion... for Nuclear Power Plants, pp 7 - 13 & 14.

ANSWER 5.25 (1.00)

c.

REFERENCE
Steam Tables

ANSWER 5.26 (.50)

False. (True for cycle 3)

REFERENCE
SQN, Review of Reactivity Coefficients Lesson Plan.

ANSWER 5.27 (1.00)

a. False.
b. True.

REFERENCE
WNTO Station Nuc Eng Text; 1-3.13 through 19

ANSWER 5.28 (1.00)

a. False.
b. True.

REFERENCE
Westinghouse Thermal Science, Chapters 3,5 & 10.

ANSWER 5.29 (.50)

False.

REFERENCE

WNTC, Thermal-Hydraulic Principles and Applications, pp 13 - 18.

ANSWER 6.01 (1.00)

b.

REFERENCE
System Manual, Chapter 3, pp 4, 10
SQNP System Descriptions, CVCS

ANSWER 6.02 (1.00)

d. or C

REFERENCE
System Manual, Chapter 3, pp 3-5
GOI-3C, pp 29-32.

ANSWER 6.03 (1.00)

c.

REFERENCE
System Manual, Chapter 4, p. 4.0-2

ANSWER 6.04 (.50)

False.

REFERENCE
System Manual, Chapter 4, p. 4.1-8

ANSWER 6.05 (1.00)

a.

REFERENCE
Reactor Protection Lesson, p. 8 of 13.

ANSWER 6.06 (.50)

True.

REFERENCE
Reactor Protection Lesson, p. 8 of 13, item d.

ANSWER 6.07 (1.00)

b

REFERENCE
System Manual, Chapter 4.2, p. 4.2-20

ANSWER 6.08 (1.00)

d

REFERENCE
System Manual, Chapter 4.2, pp 4.2-13 and 4.2-45

ANSWER 6.09 (1.50)

a. 1870, 1970
b. 2/3
c. 1970 [0.5 each]

REFERENCE
Reactor Protection Lesson, pp 11-13.

ANSWER 6.10 (1.00)

c.

REFERENCE
System Description, Chapter 7, pp 7 & 8 of 8

ANSWER 6.11 (1.00)

c.

REFERENCE
Aux Feed Lesson Plan, p. 5 of 8.

ANSWER 6.12 (1.00)

a

REFERENCE
Aux. Feed Lesson Plan, pp 4-6.

ANSWER 6.13 (1.00)

b

REFERENCE
System Manual, Chapter 11.9, pp 11.9-1 thru 5.
SQNP RCS Lesson Plan, pp 24-31.

ANSWER 6.14 (1.00)

b.

REFERENCE
SQNP Diesels handout, p. 6.

ANSWER 6.15 (1.00)

a

REFERENCE
System Manual, Chapter 11.5, Fig. 11.5-1, p. 11.5-39

ANSWER 6.16 (1.00)

a or d

REFERENCE
SQNP Excore NI Lesson Plan, p 14.
Examiner speaking with Plant Engineers and Inst. mech. during site visit.

ANSWER 6.17 (1.00)

b.

REFERENCE
Rod Control Lesson, p. 4 of 11.

ANSWER 6.18 (1.00)

c

REFERENCE
SQNP Rod Control Lesson, pp 5 & 6 of 11.
Systems Manual, Chapter 11.1, p. 11.1-63.

ANSWER 6.19 (1.00)

1. Lack of normal voltage on alternate feeder (bus).
2. Overcurrent on shutdown board. [0.5 each]
3. Transfer switch not in automatic

REFERENCE
Review of Elec. Distribution, p. 4.

ANSWER 6.20 (1.00)

d.

REFERENCE
Review of Elec. Distribution, p 4.

ANSWER 6.21 (1.00)

a

REFERENCE
Review of Elec. Distribution Lesson, p 10.

ANSWER 6.22 (1.00)

a. True.

b. False.

REFERENCE
SQNP Diesel handout, p. 7.

ANSWER 6.23 (2.00)

1. Auto start of all aux feed pumps, [0.5].
2. Operating MFPT goes to max. speed, [0.3].
3. Isolation of the affected MFPT condenser, [0.5].
4. Main Turbine runback to 75% power, [0.5]
5. Steam generator blowdown isolates, [0.2].

REFERENCE
System Manual, Chapter 10.2, p. 10.2-7

ANSWER 6.24 (1.00)

b.

REFERENCE
System Manual, Chapter 10.2, p. 10.2-7

ANSWER 6.25 (2.00)

a. CLOSE
b. CLOSE
c. OPEN
d. CLOSE
e. NO
f. OPEN
g. CLOSE
h. NO
i. NO
j. NO [0.2 ea.]

REFERENCE
SQNP System Description, ECCS, CVCS, MNSTM, CCW

ANSWER 6.26 (1.00)

a.

REFERENCE
SQNP System Description, SGLCS, p. 11.7-8 thru 11.7.10

ANSWER 6.27 (.50)

False.

REFERENCE
SQN, GOI-1, p 4.

ANSWER 6.28 (1.00)

d.

REFERENCE
SQNP System Descriptions, Condensate and Feedwater, pp. 9 & 10 of 11

ANSWER 6.29 (1.00)

c.

REFERENCE
SQNP SOI 62.1B, p. 2 of 8

ANSWERS -- SEQUOYAH 1&2 -85/05/20-VINNOLA, A.

ANSWER 7.01 (1.00)

b or a

REFERENCE
SQNP GOI-1, p. 4 ; SOI 68.2 , p 5.

ANSWER 7.02 (1.00)

d

REFERENCE
SQNP GOI-1, p. 4; precaution T.

ANSWER 7.03 (1.00)

b

REFERENCE
SQNP GOI-1, p. 4

ANSWER 7.04 (1.00)

c

REFERENCE
SQNP GOI-2, p. 9

ANSWER 7.05 (1.00)

b

REFERENCE
SQNP GOI-2, p. 16

ANSWERS -- SEQUOYAH 1&2 -85/05/20-VINNOLA, A.

ANSWER 7.06 (2.50)

a. False.
b. True.
c. True.
d. True.
e. True. [0.5 each]

REFERENCE
SQNP GOI-5A, pp 2 & 3

ANSWER 7.07 (2.00)

a. 3
b. 6
c. 1
d. 2

REFERENCE
SQNP GOI-5A, pp 5-8

ANSWER 7.08 (1.00)

d

REFERENCE
SQNP ES-1.2, p. 1

ANSWER 7.09 (1.00)

b.

REFERENCE
SQNP ES-1.2, p. 7.

ANSWER 7.10 (2.00)

1. Verify Reactor Trip.
2. Verify Turbine Trip.
3. Verify Shutdown Boards Energized.
4. Check if SI Actuated.
5. Verify ECCS status.
6. Verify Cntmt. Isolation.
7. Verify MFW Isolation.
8. Verify AFW status.
9. Verify CCS Pumps Running.
10. Verify ERCW Pumps Running.
11. Verify EGTS and ABGTS Running.
12. Check Cntmt. press less than 2.81 psig.
13. Check Tavg.

REFERENCE
SQNP E-0, pp 2-5

ANSWER 7.11 (1.00)

~~e~~ 4

REFERENCE
SQNP FHI-7, p. 1, new revision

ANSWER 7.12 (2.00)

a. 2
b. 5
c. 1
d. 5 [0.5 each]

REFERENCE
SQNP RCI-1, p.7

ANSWER 7.13 (1.00)

c

REFERENCE
SQNP ES-0.3, p. 6

ANSWER 7.14 (1.00)

b.

REFERENCE
SQNP GOI-2, p. 2

ANSWER 7.15 (1.50)

a. False

b. True

c. False

REFERENCE
SQNP SOI-74.1, pp. 3, 4

ANSWER 7.16 (1.00)

a.

REFERENCE
SQNP SOI-62.18, pp. 8, 9

ANSWER 7.17 (1.50)

a. ~~True.~~ False

b. True.

c. False.

REFERENCE
SQNP ES-1.3 p. 1 of 4; ES-1.2 p. 1 of 3, App. A

ANSWER 7.18 (.50)

False

REFERENCE
SQNP RCI-1, p. 4

ANSWER 7.19 (1.00)

a.

REFERENCE
SQNP AOI-2D, pp. 10 - 12

ANSWER 7.20 (1.00)

a.

REFERENCE
SQNP ES-0.1, p. 2 of 13

ANSWER 7.21 (.50)

True

REFERENCE
SQNP FHI-7, p. 2

ANSWER 7.22 (1.00)

ONE (1)

REFERENCE
SQNP E-2, p. 3

ANSWER 7.23 (1.00)

d

REFERENCE
SQNP ES-0.2, p. 4

ANSWER 7.24 (.50)

True.

REFERENCE
SQNP, E-3, p 4.

ANSWER 7.25 (1.00)

c.

REFERENCE
SQNP E-1, p. 2 of 11

ANSWER 7.26 (1.00)

d

REFERENCE
SQNP, FHI-7, p 4.

ANSWER 8.01 (1.00)

c

REFERENCE
SQNP AI-25, p. 5

ANSWER 8.02 (1.00)

d

REFERENCE
SQNP AI-30, pp 1-4

ANSWER 8.03 (1.00)

a

REFERENCE
SQNP AI-37, pp 1-4

ANSWER 8.04 (2.00)

SEE NEXT THREE PAGES, WHICH ARE EXCERPTS FROM AI-18.
ANY ITEM LISTED IN 10CFR50.72 WILL BE ACCEPTED INCLUDING ANY OF THE BELOW!

1. Initiation of REP.
2. Exceeding a Tech. Spec safety limit.
3. Any event that places the unit in an unexpected or uncontrolled condition.
4. Loss of physical security effectiveness, sabotage or acts of sabotage.
5. Shutdown due to a Tech. Spec. LCO.
6. Personnel error or procedural inadequacy which prevents or could prevent the fulfillment of safety functions important to safety.
7. SIS actuation.
8. Accidental, unplanned, or uncontrolled radioactive release.
9. Any fatality or serious injury requiring transport to offsite medical facility.
10. Personnel contamination requiring extensive onsite decontamination or outside assistance.
11. Any event meeting the 10CFR20.403 criteria.
12. Strikes by operating employees or security guards, or honoring of picket lines by these employees.

REFERENCE
10CFR50.72
AI - 18, pp 64 - 67, 73 - 74.

ANSWER 8.05 (3.00)

a. 1-Red
b. 5-Yellow
c. 6-Orange
d. 5-Yellow
e. 1-Red
f. 3-Green [0.5 each]

REFERENCE
SQNP AI-2, p. 9
SQNP Question and Answer Bank, sec. 8, p. 6

ANSWER 8.06 (1.00)

b

FILE PACKAGE NO. 18 FILE COVER PAGE
NOTIFICATION AND LICENSEE EVENT REPORT (LER)

Responsible Section: Operations/Compliance
Report Initiation Conditions: Conditional
References - Program procedures 1200R03 and 1200R06
10 CFR 50.72
10 CFR 50.73
10 CFR 20.403
10 CFR 20.405
10 CFR 50.54 (x)
Sequoyah Technical Specifications
NUREG 1022
Standard Practice SQA-84

NOTE: Section 6.9.1.12 and 6.9.1.13 of the technical specifications are no longer applicable after January 1, 1984, and should not be used for determining reportability.

Type of Notification/Report

I. Immediate Notification - NRC

A. The Immediate Notification Criteria of 10 CFR 50.72 is divided into 1 hour and 4 hour phone calls. Notify the NRC Operations Center (red phone) within the applicable one or four hour time limit for any item which is identified in the Immediate Notification Criteria.

NOTE: Use checklist (pages 79 and 80) as a guideline for the type of information which may be requested by the NRC Operations Center.

Complete a PRO form (SQA-84).

B. Immediate Notification Criteria

1. The following criteria require 1 hour notification

(50.72.a.1.i) a. The declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan.

(50.72.b.1.i.A) b. The initiation of any nuclear plant shutdown required by the plant's technical specifications.

(50.72.b.1.i.B) c. Any deviation from the plant's technical speicifications authorized pursuant to 10 CFR 50.54(x).

(50.72.b.1.ii) d. Any event or condition during operation that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or results in the nuclear power plant being:

8.04

File Package No. 18 (Continued)

d. (Continued) (i) In an unanalyzed condition that significantly compromises plant safety;

(ii) In a condition that is outside the design basis of the plant; or

(iii) In a condition not covered by the plant's operating and emergency procedures.

(50.72.b.1.iii) e. Any natural phenomenon or other external condition that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant.

(50.72.b.1.iv) f. Any event that results or should have resulted in emergency core cooling system (ECCS) discharge into the reactor coolant system as a result of a valid signal.

(50.72.b.1.v) g. Any event that results in a major loss of emergency assessment capability, offsite response capability, or communications capability (e.g., significant portion of control room indication, emergency notification system, or offsite notification system).

(50.72.b.1.vi) h. Any event that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.

(20.403.a) i. Any event meeting the criteria of 10 CFR 20.403 for notification, (see following list for 10 CFR 20.403 reporting requirements, also see additional reporting requirements in file package 19).

(i) Any incident involving byproduct, source, or special nuclear material possessed by the licensee that may have caused or threatens to cause:

(a) Exposure of the whole body of an individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual of 150 rems or more of radiation; or exposure of the feet, ankles, hands, or forearms of any individual to 375 rems or more of radiation; or

File Package No. 18 (Continued)

(b) The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 5,000 times the limits specified for such materials in 10 CFR 20, Appendix B, Table II, or

(c) A loss of one working week or more of the operation of any facilities affected, or

(d) Damage to property in excess of $200,000.

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b. The following criteria require 4 hour notification:

(50.72.b.2.i) 1. Any event, found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety.

(50.72.b.2.ii) 2. Any event or condition that results in manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS). However, actuation of an ESF, including the RPS, that results from and is part of the preplanned sequence during testing or reactor operation need not be reported.

(50.72.b.2.iii) 3. Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(i) Shut down the reactor and maintain it in a safe shutdown condition;

(ii) Remove residual heat;

(iii) Control the release of radioactive material; or

(iv) Mitigate the consequences of an accident.

(50.72.b.2.iv.A) 4. Any airborne radioactive release that exceeds 2 times the applicable concentrations of the limits specified in Appendix B, Table II of 10 CFR 20 in unrestricted areas, when averaged over a time period of one hour.

REFERENCE
SQNP AI-2, pp 2-5

ANSWER 8.07 (1.00)

c

REFERENCE
SQNP AI-3, p. 8

ANSWER 8.08 (1.00)

b

REFERENCE
SQNP AI-3, p. 20

ANSWER 8.09 (1.00)

c. (STA)

REFERENCE
SQNP AI-9, p. 4

ANSWER 8.10 (1.00)

d

REFERENCE
SQNP AI-9, p. 5

ANSWER 8.11 (1.00)

b

REFERENCE
SQNP PLS, p. 6

ANSWER 8.12 (1.50)

1. Source range high neutron flux trip
2. Intermediate range high neutron flux trip
~~3. Containment high-high pressure spray actuation [0.5 each]~~
(#3 is not a Reactor Trip)

REFERENCE
SQNP PLS, p. 7

ANSWER 8.13 (1.00)

a

REFERENCE
SQNP Technical Specifications 3.2.4, p. 3/4 2-15

ANSWER 8.14 (1.00)

d

REFERENCE
SQNP Technical Specification 3.4.1.4, p. 3/4 4-2b

ANSWER 8.15 (3.00)

a. 7. - 40 gpm
b. 3. - 1 gpm
c. 3. - 1 gpm
d. 1. - Zero gpm
e. 5. - 10 gpm
f. 3. - 1 gpm

REFERENCE
SQNP Technical Specification 3.4.6.2, p. 3/4 4-14

ANSWER 8.16 (1.00)

a

REFERENCE
SQNP Technical Specification 3.9.1, p. 3/4 9-1

ANSWER 8.17 (1.00)

c

REFERENCE
SQNP Tecnnical Specification 3.9.7, p. 3/4 9-7

ANSWER 8.18 (1.00)

d

REFERENCE
SQNP Techinecal Specification 6.7, p. 6-14

ANSWER 8.19 (1.00)

b

REFERENCE
SQNP Tecnnical Specifications 6.2.2 and Table 6.2-1

ANSWER 8.20 (1.00)

a

REFERENCE
SQNP AI-25, p. 6

ANSWER 8.21 (1.00)

c

REFERENCE
SQNP OSLA58, p. 11 and Appendix F

ANSWER 8.22 (1.00)

c

REFERENCE
SQNP OSLA58, p. 1

ANSWER 8.23 (1.00)

b

REFERENCE
SQNP AI-8, p. 2

ANSWER 8.24 (.50)

True.

REFERENCE
Tech. Spec. 3/4.8 & 12-20-83 occurrence.

ANSWER 8.25 (1.00)

Plant Manager. (Plant Supt.)

Plant Superintendent (O&E). (Asst Supt)

Operations Superintendent.

Ref. AI-18, p 52I; AI-2, p 8.