ENCLOSURE 1

EXAMINATION REPORT - 50-400/OL-85-01

Facility Licensee: Carolina Power and Light Company
P. O. Box 1551
Raleigh, NC 27602

Facility Name: Shearon Harris

Facility Docket No. 50-400

Written and operating examinations were administered at the Shearon Harris facility near New Hill, North Carolina.

Chief Examiner: Thomas Rogers 7/26/85
Thomas Rogers — Date Signed

Approved by: Bruce A. Wilson 7/29/85
Bruce A. Wilson, Section Chief — Date Signed

Summary:

Examinations on June 11-12, 1985

Written and operating examinations were administered to six candidates; four of whom passed.

REPORT DETAILS

1. Facility Employees Contacted:

*C. S. Olexik, Jr., Project Specialist - Nuclear Operator Training
*J. H. Smith, Director, Nuclear and Simulator Training

*Attended Exit Meeting

2. Examiners:

W. C. Hemming
P. Isaksen
*T. Rogers

*Chief Examiner

3. Examination Review Meeting

At the conclusion of the written examination, the examiners met with D. D. McDade, G. M. Blinde, W. B. Geise, and C. Olexik to review the written examination and answer key. The following comments were made by the facility reviewers:

a. Instructor Certification SRO Exam

(1) Question 5.07

Facility Comment: Steam flow information given in the problem is a factor of ten low. SHNPP 100 percent rated steam flow is 12.2×10^6 lbm/hr. The provided steam flow equates to a reactor power of approximately 230 MW. The correct answer is not provided in the selections given which range from 2000 to 3000 MW.

Recommend delete question.

NRC Resolution: Question deleted.

(2) Question 5.10

Facility Comment: Shearon Harris operators are not taught that $K_{eff2} = K_{eff1} + \Delta p$ since this does not produce an exact result. To solve this question, one must assume $K_{eff2} = K_{eff1} + \Delta p$. Since this is not the method taught, the solution would involve use of the quadratic equation, which is very time-consuming; and the quadratic formula was not given on the formula sheet.

Recommend delete question.

NRC Resolution: Question deleted.

(3) Question 5.17

Facility Comment: Question infers that ΔI will become more positive regardless of other plant conditions, given that one of the options given occurs. None of the answers given will cause this under every condition. Rods below midplane at EOL might have this effect, but at BOL this is not necessarily true since ΔI is naturally negative even with an even axial fuel distribution. At BOL, question does not state to assume an initial positive MTC.

Recommend delete question.

Reference: RT-HO-1.15.

NRC Resolution: The question infers only that of the listed choices, one will cause ΔI to shift positive. Nothing is mentioned in the question text about other plant conditions and assumptions made by the candidate outside of the intent of the questions are done at the candidate's risk. Of the choices, a, b, and d will never cause ΔI to shift positive regardless of assumptions made; therefore, only c could be chosen. The question remains as is with no deletion.

(4) Question 6.12

Facility Comment: Question implies one deenergized AST solenoid causes a turbine trip, while two are actually required. This would make the answer false by itself. However, if the examinee were to assume that one solenoid did trip the turbine as inferred, the remainder of the statement is true.

Recommend delete question.

NRC Resolution: The question is not testing the candidate's knowledge of the number of solenoids used to trip the turbine, rather the operation of the solenoids when interfacing with the incoming signals. The question clearly states that a turbine trip has occurred and removes the burden of proof on this topic from the student. The question remains as is with no deletion.

(5) Question 7.04

Facility Comment: Table 1 as presented on Path 2 is only to be used while the operator is in Path 2 as directed by procedure in Block I-5 of Path 2. Operators are not required to memorize tables that are always provided with the text of a procedure. The question is really asking the examinee to recall if there is a table provided in EPP-19 and if it is identical to that provided in Path 2. As written, the question has no correct answer.

Recommend delete question.

Reference: Path 2, Block I-5.

NRC Resolution: Question deleted.

(6) Question 7.08

Facility Comment: The Harris Fuel Handling Building may contain spent fuel from H. B. Robinson and/or Brunswick Steam Electric Plant during initial fuel load movements for the Harris Plant. The question does not tell the examinee if spent fuel is present or not. If the examinee assumes it is, answers b and c are correct responses. If the examinee assumes no spent fuel on site, answer d is correct.

Recommend delete question.

Reference: AOP-13, pages 3, 4, and 6.

NRC Resolution: According to AOP-13, Section 4.0, Statement 4, the procedure is not to be implemented until radiation levels have been established unless safety of personnel is threatened. Therefore, based on the assumption that could be made by the student, answer b or d will be accepted.

(7) Question 7.10

Facility Comment: AOP-4 (Control Room Inaccessibility) states the procedure can be carried out at the remote shutdown panel if "No other accident condition exists within the primary plant; i.e., condition requiring action under Emergency Operating Procedures." The "i.e." stands for that is and not for example. Therefore, only conditions resulting in a reactor trip or safety injection would preclude use of this procedure. The question states use of this procedure is allowed if no other primary accident condition exists. In that "primary accident condition" is not fully described in the AOP, the examinee could assume an accident that may or may not result in Emergency Operating Procedure usage.

Recommend delete question.

Reference: AOP-4, page 16, Item d.

NRC Resolution: Based on the statement in the AOP, primary accident conditions are defined as those requiring use of the EOPs. Therefore, all primary accident conditions will not preclude the use of AOP-004, also as stated above. The question will remain as is with the accepted answer changed to false.

(8) Question 7.17c

Facility Comment: Question asks for maximum allowable spray ΔT per GP-2. GP-2 lists this as 320°F and refers to technical specifications. However, the revision of technical specifications supplied for the examination states the maximum ΔT as 625°F.

Recommending accepting 320°F or 625°F.

Reference: Technical Specification 3.4.9.2.

NRC Resolution: 320°F or 625°F accepted as correct answers.

(9) Question 7.19

Facility Comment: Typographical error listed Item c as correct. Item b is the correct answer.

Recommend accepting Item b as correct.

NRC Resolution: Answer key changed to reflect "b" as the correct answer.

(10) Question 7.25

Facility Comment: This question required the operator to recall from memory what the I, S, and Sub stood for in the isotope column of Appendix B to 10 CFR 20. This information is included in the footnotes to the appendix which are an integral part of 10 CFR 20. The footnotes were not provided and always are in 10 CFR 20, Appendix B. Correctly answering the question requires memorization of information that is readily available and necessitates referencing in the event of a question pertaining to radiation protection standards.

Recommend delete question.

Reference: 10 CFR 20, Appendix B.

NRC Resolution: Question deleted.

(11) Question 7.26

Facility Comment: GP-7 states cold overpressurization protection must be in effect below 275°F and refers to technical specifications. In technical specifications referenced, 350°F is used.

Recommend accepting 275°F or 350°F.

Reference: Technical Specification 3.4.9.3 and GP-7, page 6.

NRC Resolution: Accepted answers 275°F or 350°F.

(12) Question 8.03

Facility Comment: Desired response does not require the examinee to provide the time frame that action is required, but it does require the operator to memorize the six-hour action statement. Six hours allows ample time for an operator to check a technical specification item and determine the appropriate action.

Recommend delete question.

Reference: Technical Specification 3.3.4.

NRC Resolution: Knowledge of entry conditions that will place the plant in an LCO are required for both an RO and SRO. Basic requirements to satisfy the LCO are required at the SRO level. The question remains as is with no deletion.

(13) Question 8.13

Facility Comment: Question asks if a precaution in a General Procedure can be changed by AP-7. The correct answer listed is d, which says, in effect, that AP-7 does not allow a temporary change to precautions. However, d has a typographical error and contains AOP-7 versus AP-7. Answer a says technical specifications provide a valid temporary change mechanism. The examinee would answer d if he assumed that the typo meant AP-7. However, if read as is, a is the best answer.

Recommend accept answers a and d.

Reference: AP-7 and Technical Specification Section 6.8.3.

NRC Resolution: Using AP-7, Precaution and Limitation are not allowed to be changed; therefore, the typographical error in d creates a no answer situation. The question is deleted.

(14) Question 8.22

Facility Comment: Questions requires examinee to recall from memory individuals required to be notified 24 hours after a safety limit violation. Ample time is provided to refer to technical specifications, and the operator will not be the one to contact the Vice President - Harris and the Corporate Nuclear Safety Section.

Recommend delete question.

Reference: Technical Specification Section 6.7.

NRC Resolution: Because of the nature of the violation, any SRO licensed individual should have a sound knowledge of the personnel reporting requirements. Even with time limits ignored, response c is the only one with the correct personnel. Question remains as is with no deletion.

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 no generic weaknesses noted during the oral examination.

The cooperation given to the examiners 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
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

MASTER COPY

FACILITY: SHEARON HARRIS 1

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 85/06/10

EXAMINER: HEMMING, W

APPLICANT: ______________________

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 (1.00)

Which of the following statements is CORRECT concerning the inverse multiplication plot?

(a) The vertical axis is the initial count rate and the horizontal axis is the final count rate.

(b) The vertical axis is the initial count rate divided by the final count rate and the horizontal axis is control rod reactivity.

(c) The vertical axis is control rod reactivity and the horizontal axis is the final count rate divided by the initial count rate.

(d) The vertical axis is the final count rate divided by the initial count rate and the horizontal axis is control rod reactivity.

QUESTION 5.02 (1.00)

As boron concentration increases:

(a) MTC becomes less negative due to the increased neutron leakage.

(b) MTC becomes more negative due to the increased neutron leakage.

(c) MTC becomes less negative due to the increased neutron absorption in the reactor coolant.

(d) MTC becomes more negative due to the increased neutron absorption in the reactor coolant.

QUESTION 5.03 (1.00)

Which of the following actions will cause the actual critical position to be LOWER than the estimated critical position?

(a) Overfeeding the steam generators.

(b) Increasing the steam dump pressure setpoint by 30 psi.

(c) Underestimating the actual boron concentration by 5 ppm.

(d) Allowing Tave to increase 2 F.

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

QUESTION 5.04 (1.00)

The Quadrant Power Tilt Ratio limitation is applicable:

(a) Anytime the reactor is in Mode 1.

(b) Only when one power range channel is inoperable.

(c) Only when reactor power is greater than 50%.

(d) Only during dropped rod recoveries.

QUESTION 5.05 (1.00)

To increase the VOLUMETRIC flow rate in a constant volume positive displacement pump:

(a) Reduce system resistance to flow.

(b) Increase pump speed.

(c) Increase net positive suction head.

(d) Increase fluid density.

QUESTION 5.06 (2.00)

(a) Define "Conversion Ratio".

~~(b) State a typical value for conversion ratio in a large PWR.~~ Part "b" deleted

(c) List two effects conversion ratio has on reactor operations.

QUESTION 5.07 (1.00)

Steam flow from the S/Gs is 1×10^{6} lbm/hr at 960 psia. Condenser pressure is 1 psia. Plant efficiency is 37 percent with a turbine efficiency of 70 percent and a pump efficiency of 60 percent. What is the reactor thermal power?

(a) 2285 MWt.

deleted

(b) 2325 MWt.

(c) 2587 MWt.

(d) 2785 MWt.

QUESTION 5.08 (1.00)

Which of the below is the approximate value for 100% power equilibrium xenon reactivity.

(a) 1650 pcm

(b) 2280 pcm

(c) 2600 pcm

(d) 2780 pcm

QUESTION 5.09 (2.00)

TRUE OR FALSE?

The following concern SAMARIUM.

(a) The change in Samarium concentration following a reactor trip will diminish the shutdown margin.

(b) Samarium is produced as a result of the beta decay of Promethium.

(c) The equilibrium Samarium concentration is directly proportional to reactor power.

(d) Samarium beta decays to Europium.

QUESTION 5.10 (1.00)

~~Assume the reactor is subcritical with an initial count rate of 25 counts per second. Rods are withdrawn to add 300 pcm of reactivity, resulting in a stable count rate of 40 counts per second. Which of the following is the value of Keff after the rod withdrawal?~~

~~(a) .950~~

~~(b) .990~~

~~(c) .995~~

~~(d) .999~~

deleted

QUESTION 5.11 (1.00)

At BOL, the major contributor to fast fission is:

(a) Uranium 235

(b) Uranium 238

(c) Plutonium 239

(d) Plutonium 241

QUESTION 5.12 (1.00)

Importance Factor is ______ than one because delayed neutrons ______________________.

(a) less; are less likely to leak from the core.

(b) less; do not cause fast fission of Uranium 238.

(c) greater; are less likely to leak from the core.

(d) greater; do not cause fast fission.

QUESTION 5.13 (1.00)

With a startup rate of .5 decades per minute, reactor power will ~~increase~~ by a factor of 5 approximately every:
increase

(a) 60 seconds.

(b) 72 seconds.

(c) 84 seconds.

(d) 96 seconds.

QUESTION 5.14 (1.00)

As the core ages, control rod worth ___________.
As the relative thermal neutron flux which the control rod experiences increases, control rod worth ___________.

(a) increases; increases

(b) increases; decreases

(c) decreases; increases

(d) decreases; decreases

QUESTION 5.15 (1.00)

Differential boron worth for a given Tavg is more negative at lower boron concentrations because:

(a) of the thermal flux redistribution at lower boron concentrations.

(b) of less competition between boron atoms.

(c) fewer fuel atoms are present, reducing the thermal utilization coefficient.

(d) of the harder neutron flux spectrum at lower boron concentrations.

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

QUESTION 5.16 (1.00)

The Tech. Specs. for the control of Axial Power Distribution are designed to:

(a) minimize the effects of xenon redistribution during load-follow maneuvers.

(b) serve as backup protection against a dropped or misaligned control rod.

(c) ensure adequate control rod reactivity.

(d) limit potential reactivity insertions due to a control rod ejection accident.

QUESTION 5.17 (1.00)

In which of the following situations will the further insertion of control rods cause Delta I to become more positive?

(a) Buildup of Xenon in the top of the core with rods fully withdrawn.

(b) Positive MTC during a reactor startup.

(c) Bank D control rods inserted to the core midplane.

(d) Excessively negative MTC at EOL.

QUESTION 5.18 (2.00)

Technical Specification 3.10 lists two DNB-related parameter limits which shall be maintained during power operation. List the two limits (include their values).

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

QUESTION 5.19 (1.00)

Which of the following is NOT necessary to cause brittle fracture?

(a) Pre-existing defects

(b) Load stress greater than yield stress

(c) Temperataure below the nil ductility transition temperature

(d) Residual stresses

QUESTION 5.20 (1.00)

In order to maintain a 200 F subcooling margin in the RCS when reducing RCS pressure to 1600 psig, steam generator pressure must be reduced to approximately:

(a) 245 psig

(b) 445 psig

(c) 645 psig

(d) 845 psig

QUESTION 5.21 (1.00)

When the flow rate through a centrifugal pump is increased by opening the discharge valve, the required NPSH _______, and the available NPSH ______.

(a) increases; increases

(b) increases; decreases

(c) decreases; increases

(d) decreases; decreases

QUESTION 5.22 (1.00)

Which of the following conditions is NOT indicative of pump runout?

(a) abnormally high discharge pressure

(b) excessive current in the pump motor

(c) failure of the coupling between the pump shaft and the motor shaft

(d) available NPSH less than required NPSH

QUESTION 5.23 (1.00)

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 5.24 (2.00)

The hot channel factor limits will be met for normal operation provided four conditions are observed. List these four conditions.

QUESTION 5.25 (1.00)

If reactor power increases, DNBR will ______________
If RCS pressure increases, DNBR will ______________

a. increase, increase.

b. increase, decrease.

c. decrease, increase.

d. decrease, decrease.

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

QUESTION 5.26 (1.00)

With a 1 decade per minute startup-rate, reactor power will double approximately every ________ seconds.

a. 9

b. 18

c. 27

d. 54

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

QUESTION 6.01 (2.50)

TRUE OR FALSE?

The following statements concern the construction and operation of the POWER RANGE NUCLEAR INSTRUMENTATION detector.

a. Is lined with Boron-10.

b. Has Boron-triflouride (BF3) gas in the detecter.

c. Is a fission chamber.

d. Operates in the proportional region of the gas amplification curve. (Detector voltage vs. current curve).

e. Uses no compensation circuitry to remove gamma current.

QUESTION 6.02 (2.00)

TRUE OR FALSE?

The following statements concern the response of the ROD CONTROL system.

a. An urgent failure in a power cabinet sends a signal to the logic cabinet, inhibiting all automatic rod motion.

b. At the C-3 and/or C-4 setpoint, all automatic and manual rod motion in inhibited.

c. If turbine power falls below 15%, automatic rod withdrawl is blocked.

d. At 103% reactor power, automatic rod insertion is inhibited.

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

QUESTION 6.03 (1.50)

What would happen (INCREASE, DECREASE, NO EFFECT) to each of the below if the parameter change following each occurred.

a. Indicated 100% Steam Flow -- if steam pressure output failed to 50% of it's full value.

b. Indicated Power--if cold leg temperature decreases by 5-F while maintaining 100% actual reactor power.

c. Control bank rod height--if Tcold input to a Tave channel fails low.

QUESTION 6.04 (3.00)

With plant load at 50% and the Chemical and Volume Control System (CVCS) in a normal lineup, the charging system is put in manual at 30 GPM discharge flow. Assuming no operator action state the sequence of events that will lead to a reactor trip. Include setpoints where applicable.

QUESTION 6.05 (1.50)

The following concern the reactor coolant low flow trip CIRCUITRY. For the following conditions, state whether an automatic reactor trip WILL or WILL NOT occur.

a. ONE RCP voltage is below its undervoltage setpoint while at 35% reactor power.

b. TWO RCP breakers are opened while the reactor is at 5% power.

c. ONE RCP trips on underfrequency while at 75% reactor power.

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

QUESTION 6.06 (1.00)

What is the required flowrate that auxiliary feedwater must supply to the S/G's on a loss of normal feedwater with site power available.

a. 380 gpm.

b. 500 gpm.

c. 800 gpm.

d. 830 gpm.

QUESTION 6.07 (2.00)

Any combination of the following equipment will assure adequate heat removal to keep containment pressure below the design pressure during injection phase: (fill in the blanks placing answers on your answer page.)

______ out of two CSS pumps.

______ out of four Containment Cooling Units.

______ CSS pump(s) and ______ Containment Cooling Unit(s).

QUESTION 6.08 (1.00)

Regarding the Chemical and Volume Control System, if an unsaturated bed of H-OH resin is placed in service, what will be the results?

a. Oxygen in the primary will increase.

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

c. Positive reactivity will be added to the reactor.

d. Boron will be released.

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

QUESTION 6.09 (2.00)

Complete the following statements concerning the Boron Thermal Regeneration System (BTRS) by filling in the blanks. Place your answers on your answer page.

The BTRS capacity is limited to approximately a ________ ppm change at BOL and a ________ ppm change at EOL.

At BOL, a boron concentration dilution of 100 ppm using BTRS takes approximately _______ hour(s) and at EOL, the same dilution takes _______ hour(s).

QUESTION 6.10 (.50)

TRUE OR FALSE?

A flow restrictor is inserted into the RCS hot leg bypass manifold so that hot leg bypass flow will be equal to cold leg bypass flow.

QUESTION 6.11 (.50)

TRUE OR FALSE?

The piping elbow installed in the RCS to create a delta-P for flow measurement has 3 low pressure taps and 1 high pressure tap.

QUESTION 6.12 (.50)

TRUE OR FALSE?

After a turbine trip has been initiated from the turbine's Emergency Trip System the solenoid causing the trip remains DE-ENERGIZED after the parameter that caused it to de-energize returns to normal.

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

QUESTION 6.13 (1.00)

How is the presence of water in the casing of the High Pressure Turbine detected?

a. Installed thermocouples.

b. Tell-tale drains.

c. Impulse traps.

d. There is no direct detection system installed, only increased noise and vibration are available.

QUESTION 6.14 (1.50)

For each reactor core bypass flowpath, state the amount of the bypass flow. (In percent of total core flow.)

a. Nozzle Bypass Flow.

b. Head Cooling Bypass Flow.

c. Control Rod and Instrument Thimble Bypass Flow.

QUESTION 6.15 (1.50)

a. List the two sequencers that are used in the Engineered Safeguards System to sequence electrical loads onto the Emergency Diesel Generator. (1.0)

b. In a saftey injection situation with loss of off-site power, the sequencer used is delayed until the diesel generator output breaker shuts. How many seconds is added to this seqeuncer's normal timing by this delay? Why is the sequencer delayed? (0.5)

QUESTION 6.16 (1.00)

When is a 2 out of 4 protection logic required to be used?

a. When four detectors are used.

b. When both protection and control are from the same detector.

c. When no alternate or backup protection exists for that parameter.

d. When the detectors are not envronmentally qualified.

QUESTION 6.17 (3.00)

Using the attached drawing, RPS-TP-1.0, Reactor Core Safety Limits vs Protection Boundry, indicate on your answer sheet what each line or area labeled 1-6 represent.

QUESTION 6.18 (1.00)

TRUE OR FALSE?

The following concern PRESSURIZER LEVEL indication.

a. If a leak develops in the reference leg, pressurizer level will indicate low.

b. Operating at a temperature above the calibration temperature (650 F) will cause pressurizer level to indicate low.

QUESTION 6.19 (1.00)

How would an open or a short of the RTD effect a RTD bridge circuit output?

a. An open would fail the indication high and a short would fail it low.

b. An open would fail the indication low and a short would fail it high.

c. Both failures would cause the indication to fail high.

d. Both failures would cause the indication to fail low.

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

QUESTION 6.20 (1.00)

What is the accuracy of the Digital Rod Position Indication System (DRPI)?

a. +/- 1 step.

b. +/- 4 steps.

c. +/- 12 steps.

d. + 10, -4 steps.

QUESTION 6.21 (1.00)

What will cause the Nuclear Instrumentation Channel Current Comparator to alarm?

a. If any one channel exceeds 1.02 times the average of all the channels.

b. If the difference between any two channels exceeds 2%.

c. If any channel exceeds the auctioneered high channel by 2%.

d. If any channel exceeds the average of all channels by +2%, -1%.

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

QUESTION 7.01 (1.00)

An inadvertant reactor trip has occurred. During performance of procedure EOP-EPP-004, Reactor Trip Response, a safety injection occurs. Where are you to proceed?

a. Path 2, entry point J.

b. Path 1, entry point C.

c. EOP-EPP-004, step 1.

d. Path 1, entry point A.

QUESTION 7.02 (1.00)

During performance of EOP-EPP-001, Loss of AC Power to 1A-SA and 1B-SB Busses, you are informed by the STA that the Critical Safety Function Status Tree for Heat Sink is in a yellow path.
He recommends that you proceed to procedure FRP-H.2, Response to S/G Overpressure. What action do you take.

a. Acknowledge the information and continue to proceed in EOP-EPP-001.

b. Proceed in EOP-EPP-001 and use FRP-H.2 in conjunction with it.

c. Go to FRP-H.2, use it until the situation is under control and then return immediately to EOP-EPP-001.

d. Proceed to Path 1, entry point C.

QUESTION 7.03 (1.00)

Procedure EOP-EPP-001, Loss of AC Power to 1A-SA and 1B-SB Busses, cautions the operator not to depressurize the S/G's below 165 psig. Why is the limit imposed?

a. To prevent nitrogen from being injected into the primary.

b. To prevent voiding in the vessel head region.

c. To ensure the nuclear instruments continue to read properly.

d. To ensure the S/G's continue as reliable heat sinks.

QUESTION 7.04 (1.00)

~~When is Table 1 on EOP Path 2 used?~~

deleted

~~a. Post SGTR Using Backfill, EOP-EPP-17.~~

~~b. Post SGTR Cooldown Using Steam Dumps, EOP-EPP-19.~~

~~c. SGTR With Loss of Reactor Coolant, Subcooled Recovery, EOP-EPP-20.~~

~~d. SGTR With Loss of Reactor Coolant, Saturated Recovery, EOP-EPP-21.~~

QUESTION 7.05 (1.00)

The Control Room is declared inaccessable and an evacuation declared. The reactor is tripped from the main control board and an operator stationed at the Auxiliary Control Panel. What procedure is to be used?

a. EOP Path 1, entry at "reactor trip or SI block".

b. EOP Path 1, entry point A.

c. AOP-004, Control Room Inaccessability, and the EOP Network procedures as they apply.

d. AOP-004, Control Room Inaccessability.

QUESTION 7.06 (1.00)

If high radiation exsists in the Control Room, who orders an evacuation.

a. Shift Foreman.

b. Shift Foreman with concurrence from the Control Room Senior Control Operator.

c. Shift Foreman with permission from the Manager-Operations.

d. Manager-Operations.

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

QUESTION 7.07 (1.00)

An excessive turbine eccentricity /vibration alarm is received while operating the main turbine on the grid. The turbine is tripped and all steam is cutoff. Upon engagement of the turning gear, eccentricity reads .003 in. What action is to be taken per AOP-006, Turbine Vibration.

a. Continue operation on the turning gear and continue with procedure.

b. Stop the turning gear operation and proceed with Follow-up Actions.

c. Inform the Supervisor-Operations and continue investigation.

d. Continue operations on the turning gear and advise the Manager-Operations of the reading.

QUESTION 7.08 (1.00)

During initial fuel loading, it is reported to the Control Room that a new fuel assembly has been dropped in the Spent Fuel Pit. What action(s) is/are to be taken from the Control Room?

a. No action until directed by the Operations Manager.

b. Evacuate all personnel until Radiation Control personnel have established radiation levels.

c. Implement AOP-13, Fuel Handling Accident.

d. No action required as no spent fuel is onsite.

QUESTION 7.09 (1.00)

When control is shifted to the Auxilary Control Panel, what automatic functions are removed?

a. All functions associated with P-7.

b. All functions of P-10 and P-6.

c. All automatic SI actuation signals.

d. All automatic SI actuation signals and functions of P-7.

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

QUESTION 7.10 (.50)

TRUE OR FALSE?

In order to use AOP-004, Control Room Inaccessability, no other accident conditions can exist in the primary plant.

QUESTION 7.11 (.50)

TRUE OR FALSE?

According to AOP-15, Secondary Load Rejection, activation of the load drop anticipator will arm the steam dumps thus precluding excessive steam release out of the Moisture Seperator Reheater relief valves.

QUESTION 7.12 (1.00)

When using AOP-16, Excessive Primary Plant Leakage, when should safety injection be initiated?

a. If leakage exceeds the capacity of one charging pump.

b. If VCT level cannot be maintained with letdown isolated.

c. If pressurizer level cannot be maintained greater than 15%.

d. If the leakage exceeds RCS makeup capability.

QUESTION 7.13 (1.00)

In a lifesaving situation, what is the emergency exposure limit?

a. 25 rems wholebody, 100 rems to hands and forearms.

b. 100 rems wholebody, 200 rems to hands and forearms.

c. 75 rems wholebody, 200 rems hands and forearms.

d. 75 rems wholebody, 100 rems hands and forearms.

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

QUESTION 7.14 (.50)

TRUE OR FALSE?

Emergency doses gained for any reason are NOT to be included in an individuals exposure history record.

QUESTION 7.15 (.50)

TRUE OR FALSE?

If a qualified female radiation worker becomes pregnant, she may continue to work in a job that requires occupational radiation exposure if she chooses to do so.

QUESTION 7.16 (1.00)

When frisking out of a contaminated area, what constitutes skin or clothing contamination using an Eberline RM-14 and what is the maximum background that may be present at the final exit point?

a. 100 cpm > background with a maximum 100 cpm final exit background.

b. 75 cpm > background with a maximum 200 cpm final exit background.

c. 100 cpm > background with a maximum 200 cpm final exit background.

d. 200 cpm > background with a maximum 100 cpm final exit background.

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

QUESTION 7.17 (4.00)

Answer each of the following according to the precautions in procedure GP-02, Normal Plant Heatup From Cold Solid to Hot Subcritical.

a. Reactivity can be added to a subcritical reactor by more than one method at a time providing that one of the methods is from ________________. (two words)

b. Heatup and cooldown rates for the RCS shall not exceed ______F/hr.

c. The maximum spray water to pressurizer temperature differential shall be _______F.

d. The shutdown margin shall be greater than or equal to ________pcm for 3 loop operation.

e. Moderator Temperature Coefficient (MTC) shall be maintained between _________ and ________ pcm/F.

f. To allow the shutdown banks to be left inserted while the reactor is subcritical with positive reactivity being inserted requires the approval of the ____________________. (two words)

g. With the RCS temperature less than 70 F, S/G pressure must be determined to be less than 200 psig at least once every ______.

h. When the RHR loops are in service, the reactor coolant pressure must not exceed _______ psig as determined by PI 403.

QUESTION 7.18 (2.00)

Answer each of the following concerning ECP's and GP-03, Reactor Startup From Hot Standby to Critical.

a. The reactor must not be taken critical below the minimum ________.

b. The maximum allowable difference between estimated critical position (ECP) and actual critical position is _________ pcm.

c. If the limit in part b. above is exceeded, the reactor must be ___________ and the ECP ____________

d. 1/M data points should be taken whenever there is a substantial increase in _____________ with a minimum of _____ points plotted.

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

QUESTION 7.19 (1.00)

When is procedure GP-04, Recovery From a Reactor Trip, applicable?

a. Anytime after the cause for the trip is found and corrected.

b. Anytime that less than 6 hours has elasped from trip to starting control bank withdrawl.

c. Anytime that less than 6 hours has elasped from trip to starting control bank withdrawl and boron concentration has not been significantly altered.

d. Anytime that control bank withdrawl can be started prior to peak Xenon.

QUESTION 7.20 (1.00)

According to GP-004, Recovery From A Reactor Trip, what must be done if a confident determination of Xenon cannot be made?

a. Startup the reactor using an inverse count rate ratio.

b. Insure a minimum of 24 hours passes before control bank withdrawl.

c. Calculate an ECP using the most conservative Xenon value obtainable.

d. Startup the reactor limiting SUR to .5 DPM.

QUESTION 7.21 (1.00)

As Shift Foreman while in a refueling mode, it is brought to your attention that the boron concentration is 1850 ppm. Calculating Keff indicates it is .94. What action should be taken.

a. No action is required as Keff is < .95.

b. Immediately borate the RCS to a concentration of 2000 ppm.

c. Stop all core alterations and notify the operations manager for permission to continue.

d. Continue refueling using an inverse count ratio plot and verify Keff < .95 every hour as long as boron concentration is < 2000 ppm.

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

QUESTION 7.22 (1.00)

While loading fuel the audible output from the source range channel 31 is lost. Switching the audible to channel 32 restores the audible. What action is required?

a. Core alterations must stop until the audible on channel 31 is fixed.

b. Core alterations may continue as long as boron samples are done every 72 hours.

c. Core alterations must stop until the Operations Manager approves continued operations.

d. Core alterations may continue providing that a portable source range instrument is placed in containment within one hour.

QUESTION 7.23 (1.00)

Technical Specification 3.4.1.1 requires all reactor coolant loops to be operational in modes 1 and 2. This T.S. is exempted whenever:

a. Power is below the P-6 setpoint.

b. Power is below the P-7 setpoint and the Intermediate/ Power range reactor trip low power setpoints are set at < 25%.

c. Special tests for control rod worth or shutdown margin are being performed.

d. Startup tests are being performed.

QUESTION 7.24 (1.00)

According to 10 CFR 20.102b, before permitting any individual in a restricted area to receive exposure in excess of the limits of 10 CFR 20.101a (1.25 rem/qtr.), what must be done?

a. File only form NRC-4.

b. File form NRC-4 and calculate the additional dose allowed.

c. File form NRC-4 and undergo an approved General Employee Training course on radiation exposure.

d. File form NRC-4 and form NRC-396.

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

QUESTION 7.25 (1.50)

10 CFR 20, Appendix B, contains concentration limits in air and water for all elements. Refering to the attached page from the appendix, fill in the blanks below on your answer page.

Deleted

a. If the letter "I" is observed in the column next to the isotope designation, it stands for ______________.

b. If the letter "S" is observed in the column next to the isotope designation, it stands for ______________.

c. If "SUB" is observed in the column next to the isotope designation, it stands for __________________________________.

QUESTION 7.26 (1.50)

Answer the following according to GP-07, Normal Plant Cooldown.

a. How many Steam Generators must be in operation with Tave above 200 F?

b. If the RCS is less than or equal to _______ F, and not vented to the containment, two ________ must be operable.

c. Pressurizer boron concentration should not differ from RCS boron concentration by more than ______ ppm during normal operations, and _____ ppm during transient conditions.

QUESTION 7.27 (1.00)

According to procedure EOP-EPP-5, Natural Circulation Cooldown, how long must the reactor vessel upper head region be cooled to prevent upper head voiding when the RCS is depressurized with the CRDM fans are inoperable?

a. 9 hrs.

b. 19 hrs.

c. 29 hrs.

d. 39 hrs.

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

QUESTION 8.01 (2.00)

List all the Critical Safety Functions status Trees (CSFST) in their proper order of priority

QUESTION 8.02 (1.00)

The concentration and temperature of the boric acid solution in the Boron Acid Tanks shall be verified every 7 days. The chemist sampled the ~~BIT~~ BAT under the following schedule. (All samples were taken at about 1200).

January 1--January 8--January 16--January 24--January 31

On what day(s) was/were the Tech. Spec. surveillance for the Boric Acid Tanks violated?

a. January 24

b. January 16 and 24

c. January 16

d. January 16, 24, and 31

QUESTION 8.03 (1.00)

The unit is operating at 50% load when the main generator governor valve #3 fails open and the remaining three valves reposition to maintain load at 50%.

List the TWO possible actions, as stated in Tech. Specs, that may be taken to keep the turbine in an operating status? (Time limits not required.)

QUESTION 8.04 (3.00)

For each of the following leak locations give the maximum allowable leak rate AND the basis for each as listed in Tech. Specs.

a. Unknown location.

b. Through a pressurizer code safety valve to the Pressurizer Relief Tank.

c. Through the wall of the line between the pressurizer relief valves and the pressurizer.

d. Reactor Coolant Pump seals.

e. Steam Generator tube leakage.

QUESTION 8.05 (1.50)

List the THREE overall bases that the specifications in Tech. Spec. section 3.1.3, Moveable Control Assemblies, ensures.

QUESTION 8.06 (1.50)

List THREE of the five bases behind Tech. Spec. 3.1.1.4, Minimum Temperature for Criticality.

QUESTION 8.07 (1.00)

During performance of procedure PEP-101, Initial Emergency Actions, who may relieve the Shift Foreman and conduct this procedure?

a. No one.

b. Only the Site Emergency Director- Technical Support Center.

c. Any designated alternate trained to do so.

d. Only the Plant Operations Director.

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

QUESTION 8.08 (1.00)

According to Shearon Harris procedure PEP-101, Initial Emergency Actions, what action is the Shift Foreman NOT allowed to delegate?

QUESTION 8.09 (1.00)

You are the Shift Foreman when a casualty occurs which creates an Alert situation. Which of the following is the proper order of succession for the Site Emergency Coordinator- Technical Support Center?

a. Control Room Senior Control Operator, Reactor Control Room Operator, any trained designate.

b. Manager Operations, Manager Start-up, Manager Maintenance, any trained designate.

c. Plant General Manger, Manager Start-up, Manager Operations, Manager Maintenance.

d. Plant General Manager, Manager Start-up, Manager Operations, any trained designate.

QUESTION 8.10 (1.00)

While in the Emergency Plan, who are the only individuals authorized to request off-site assistance? (Other than law enforcement.)

a. Site Emergency Coordinator or Emergency Response Manager.

b. Site Emergency Coordinator or Plant Operations Director.

c. Plant Operations Director or Representative to the State Emergency Response Team.

d. Plant General Manager or Plant Operations Director.

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

QUESTION 8.11 (1.00)

In regards to the Emergency Plan, who has responsibility to determine the need for, and to direct an evacuation of hazardous areas, along with directing personnel to a safe area?

a. Work Group Supervisors.

b. Site Emergency Coordinator.

c. Each individual.

d. Plant Operations Director.

QUESTION 8.12 (1.00)

Who is responsible to determine when an OWP (Operations Work Permit) is required?

a. Manager Operations.

b. Operations Supervisor or designate.

c. Shift Foreman.

d. Manager Maintenance.

QUESTION 8.13 (1.00)

While performing GP-02 on the midshift, a precaution and limitation which is not applicable at this time prevents you from proceeding. Using AP-007, Temporary and Advanced Changes to Plant Procedures, what can be done?

a. A temporary change form must be filled out with approval by 2 interim approvers.

deleted

b. An advanced change form must be filled out with approval by 2 qualified Safety Reviewers.

c. A temporary change form must be filled out with approval from 2 qualified Safety Reviewers.

d. Nothing can be done to allow continuance per AOP-007.

QUESTION 8.14 (1.00)

When is an Equipment Inoperable Record (EIR) attached to the front of a Shift turnover Package (STP)?

a. For any piece of equipment declared inoperable by the off-going shift.

b. For any piece of equipment declared inoperable by the off-going shift which is not likely to be restored in the next on-coming shift.

c. For Tech. Spec. related equipment declared inoperable by the off-going shift.

d. For Tech. Spec. related equipment declared inoperable by the off-going shift which is not likely to be restored in the next on-coming shift.

QUESTION 8.15 (1.00)

Which statement concerning Guidance For Voluntary LCO's, AP-019, is correct?

a. The TOTAL length of time required to complete the work cannot exceed 80% of the LCO time limit for both Group I and Group II LCO's.

b. Group II LCO's are more limiting than Group I.

c. Group I LCO's must be worked on a continuous 24 hour basis.

d. Group II LCO's must have the Plant General Managers approval prior to voluntary entry.

QUESTION 8.16 (.50)

TRUE OR FALSE?

If a worker remains in the work area to maintain control of the lifted leads, wire removal tags are not needed providing the piece of equipment is not Tech. Spec. related.

QUESTION 8.17 (.50)

TRUE OR FALSE?

Lifted leads already identified in other approved procedures are excluded from procedure AP-24, Temporary Bypass, Jumper, and Wire Removal Control.

QUESTION 8.18 (1.00)

Following a reactor trip with the cause found and corrected, the Shift Foreman has the authority to:

a. withdraw the shutdown banks.

b. alter Keff to a maximum of .95.

c. alter any combination of reactivity, one method at a time, to 500 pcm below the ECP.

d. withdraw the shutdown banks to within 1000 pcm of the ECP.

QUESTION 8.19 (1.00)

According to Tech. Spec. Section 6, which of the below is the correct maximum for working hours when substantial amounts of overtime are needed on a temporary basis?

a. Not more than 16 hours straight including shift turnover time.

b. Not more than 16 hours straight excluding shift turnover time.

c. Not more than 16 hours straight.

d. Not more than two 16 hour shifts in a 48 hour period.

QUESTION 8.20 (3.00)

Answer the following concerning Facility Staffing as established in Tech. Spec., section 6.

a. The Fire Brigade shall be composed of ________ members.

b. Core alterations shall be supervised directly by either a licensed SRO or a licensed ____________________________________ (5 words)

c. State the number of Radiation Control Techicians that must be onsite when fuel is in the core.

d. The minimum shift crew manning outlined in table 6.2-1 may be reduced by one, except for the ____________, for ________ hour(s).

e. When in modes 1-4, in the absense of the Shift Foreman, any indivdual with a valid SRO license may be designated to assume control except the ________.

f. When in modes 5 or 6, any individual with a valid _______________ or _______________ may assume control. (Two words each)

QUESTION 8.21 (2.00)

Refering to the list of events below, choose the events that must be reported to the NRC within ONE HOUR per 10 CFR 50.72. (More than one answer is possible.)

a. Declaration of an "Unusual Event" per the emergency plan.

b. Automatic actuation of the Auxiliary Feedwater System.

c. An airborne release 2 times the limits of 10 CFR 20, Appendix B, Table II, in an unrestricted area, averaged over one hour.

d. An actual low pressure safety injection actuation.

e. A plant shutdown due to exceeding the time limits of an LCO in Tech. Specs.

f. A fire in the auxiliary building.

QUESTION 8.22 (1.00)

Who must be notified in the event that a safety limit is violated?

a. NRC within 1 hour, Operations Manager within 1 hour, and the Plant General Manager within 1 hour.

b. NRC within 1 hour, Plant General Manager within 1 hour, and Vice President of Harris Nuclear Project within 24 hours.

c. NRC within 1 hour, Vice President of Harris Nuclear Project within 24 hours, and Manager of Corporate Nuclear Safety within 24 hours.

d. NRC within 1 hour, Vice President of Harris Nuclear Project within 24 hours, and Plant Nuclear Safety Committee within 24 hours.

QUESTION 8.23 (1.00)

Read the following statement and choose the correct definition title.

"An area of such a size that an individual located at any point on it's boundry for two hours immediately following the onset of the postulated fission product release would not receive a total radiation dose of 25 rems whole body or 300 rems to the thyroid from iodine."

a. The Low Population Zone Boundry.

b. Restricted Area Boundry.

c. Population Center Boundry.

d. Exclusion Area Boundry.

QUESTION 8.24 (1.00)

Which of the following constitutes an IMPROPER valve arrangement for containment isolation as defined in 10 CFR 50 ,appendix A?

a. A check valve inside containment and an automatic isolation valve outside containment.

b. A locked closed valve inside containment and a locked closed valve outside containment.

c. An automatic isolation valve inside containment and a check valve outside containment.

d. An automatic isolation valve inside containment and a locked closed valve outside containment.

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

ANSWER 5.01 (1.00)

B

REFERENCE
Nuclear Reactor Theory for the Power Plant Operator, Pages 16-18

ANSWER 5.02 (1.00)

C

REFERENCE
RT-HO-1.10, Pages 14-15

ANSWER 5.03 (1.00)

A

REFERENCE
RT-HO-1.14, Pages 10-14

ANSWER 5.04 (1.00)

C

REFERENCE
S.H.T.S. Section 3.2.4., Pages 3/4-2-11

ANSWER 5.05 (1.00)

B

REFERENCE
FF-LP-1.1, Pages 26-27

ANSWER 5.06 (2.00)

(a) The ratio of the amount of Pu-239 produced to the amount of U-235 depleted

$$\frac{\text{Pu 239 produced}}{\text{U 235 depleted}}$$ (1.0)

(b) Extended core life
Faster core response time (1.0)

REFERENCE
RT-HO-1.9, Pages 15-17, 35 and Fig. RT-TP-164
RT-HO-1.6, Pages 26-27

~~ANSWER 5.07 (1.00)~~

deleted

~~REFERENCE~~
~~Thermo LP 1.4, Page 60~~

ANSWER 5.08 (1.00)

D

REFERENCE
S.H. Curve Book, Curve C-1

ANSWER 5.09 (2.00)

(a) FALSE

(b) TRUE

(c) FALSE

(d) FALSE

REFERENCE
S.H. RT-HO-1.11, Pages 22-25

~~ANSWER 5.10 (1.00)~~

deleted

~~REFERENCE~~
~~RT-HO-1.6~~

ANSWER 5.11 (1.00)

A

REFERENCE
RT-HO-1.5, Page 12

ANSWER 5.12 (1.00)

B

REFERENCE
RT-HO-1.6, Pages 27-28

ANSWER 5.13 (1.00)

C

REFERENCE
RT-HO-1.6, Pages 10-12

ANSWER 5.14 (1.00)

A

REFERENCE
RT-HO-1.13, Pages 16-18

ANSWER 5.15 (1.00)

B

REFERENCE
RT-HO-1.12, Page 14

ANSWER 5.16 (1.00)

A

REFERENCE
S.H.T.S. Section 3/4 2.1, Pages B3/4 2-1

ANSWER 5.17 (1.00)

C

REFERENCE
RT-HO-1.15

ANSWER 5.18 (2.00)

RCS Temp $\leq$ 593 F

PZR Pressure $\geq$ 2205

REFERENCE
S.H.T.S. 3.2.5, Pages 3/4 2-14

ANSWER 5.19 (1.00)

B

REFERENCE
HT-LP-1.2, Page 20

ANSWER 5.20 (1.00)

A

REFERENCE
Thermo-LP-1.1 and steam tables

ANSWER 5.21 (1.00)

B

REFERENCE
FF-LP-1.1, Pages 27-30

ANSWER 5.22 (1.00)

A

REFERENCE
FF-TP-52.0

ANSWER 5.23 (1.00)

D

REFERENCE
FF-LP-1.1 Section 2.2

ANSWER 5.24 (2.00)

Control rods in a single bank move together [.25] with no individual rod insertion differing by more than +/- 13 steps from bank demand.

Control banks are sequenced with overlapping banks. [.5]

Control bank insertion limits are not violated. [.5]

Axial power distribution control limits are observed. [.5] (2.0)

REFERENCE
S.H. T.S. 3/4.2.2/2.3 Pp. B 3/4 2-2, 2-3.

ANSWER 5.25 (1.00)

c.

REFERENCE
S.H. Heat Transfer HT-LP-1.2.

ANSWER 5.26 (1.00)

b.

REFERENCE
S. H. Reactor Theory, RT-HO-1.6, Pp 6-12.

ANSWER 6.01 (2.50)

a. TRUE

b. FALSE

c. FALSE

d. FALSE

e. TRUE [0.5 each]

REFERENCE
NIS-HO-1.0, p. 13

ANSWER 6.02 (2.00)

a. TRUE

b. FALSE

c. TRUE

d. FALSE [0.5 each]

REFERENCE
RODCS-HO-1.0, Pp 14-22.

ANSWER 6.03 (1.50)

a. DECREASE

b. DECREASE

c. NO EFFECT

REFERENCE
NIS-HO-1.0, Pp-6-10
RODCS-HO-1.0
SGWLC-HO-1.0, p-8

ANSWER 6.04 (3.00)

Answer must include the following as a minimum for full credit.

Pzr. level will decrease due to charging < letdown.
17% Pzr. level isolates letdown.
Pzr. level increases due to charging > letdown.
At 92% reactor trip occurs.

REFERENCE
PZRLC-HO-1.0, Pp 11-15.

ANSWER 6.05 (1.50)

a. NO TRIP

b. NO TRIP

c. TRIP

REFERENCE
RPS-HO-1.0, p17

ANSWER 6.06 (1.00)

b.

REFERENCE
AFS-HO-1.0, p-7.

ANSWER 6.07 (2.00)

Two (2)
Four (4).
One (1), two (2)

REFERENCE
CSS-HO-1.0, p-5.

ANSWER 6.08 (1.00)

c.

REFERENCE
CVCS-HO-1.0, p-13.

ANSWER 6.09 (2.00)

200, 100.
3-4, 14-18.

REFERENCE
BTRS-HO-1.0, p-8.

ANSWER 6.10 (.50)

FALSE

REFERENCE
RCS-HO-1.0, p-17.

ANSWER 6.11 (.50)

TRUE

REFERENCE
RCS-HO-1.0, p-18.

ANSWER 6.12 (.50)

TRUE

REFERENCE
EHC-HO-1.0, p-24.

ANSWER 6.13 (1.00)

a.

REFERENCE
MT-HO-1.0, p-6.

ANSWER 6.14 (1.50)

1%, .5%, 2%.

REFERENCE
RVI-HO-1.0, p-15

ANSWER 6.15 (1.50)

a. Loss of Offsite Power.
LOCA. [.5 each]

b. 10 seconds.
To allow the diesel time to start. [.25 each]

REFERENCE
SEQ-HO-1.0, p-9

ANSWER 6.16 (1.00)

b.

REFERENCE
PRS-HO-1.0, p-7.

ANSWER 6.17 (3.00)

1. S/G Safeties
2. OT delta T
3. OP delta T
4. Nuclear Overpower
5. Acceptable Operation
6. Unacceptable Operation

REFERENCE
Drawing RPS-TP-1.0, file 10.2.

ANSWER 6.18 (1.00)

a. FALSE

b. TRUE.

REFERENCE
PZRLC-HO-1.0, p-10.

ANSWER 6.19 (1.00)

a.

REFERENCE
RCTEMP-HO-1.0, p-21.

ANSWER 6.20 (1.00)

b.

REFERENCE
RODCS-HO-1.0, p-10.

ANSWER 6.21 (1.00)

b.

REFERENCE
NIS-HO-1.0, p-28.

ANSWERS -- SHEARON HARRIS 1 -85/06/10-HEMMING, W.

ANSWER 7.01 (1.00)

d.

REFERENCE
S.H. EOP-EPP-004, p 3.

ANSWER 7.02 (1.00)

a.

REFERENCE
S.H. EOP-EPP-001, p 3.

ANSWER 7.03 (1.00)

a.

REFERENCE
S.H. EOP-EPP-001, p. 12.

~~ANSWER 7.04 (1.00)~~

deleted

~~REFERENCE~~
~~S.H. EOP-EPP-19, p 7~~

ANSWER 7.05 (1.00)

d.

REFERENCE
S.H. AOP-004, p 3.

ANSWER 7.06 (1.00)

a.

REFERENCE
S.H. AOP-005, p 10.

ANSWER 7.07 (1.00)

b.

REFERENCE
S.H. AOP-006, p 4.

ANSWER 7.08 (1.00)

b or d.

REFERENCE
S.H. AOP-13, p 6.

ANSWER 7.09 (1.00)

c.

REFERENCE
S.H. AOP-004, p 4.

ANSWER 7.10 (.50)

~~TRUE~~. FALSE

REFERENCE
S.H. AOP-004, p 16.

ANSWERS -- SHEARON HARRIS 1 -85/06/10-HEMMING, W.

ANSWER 7.11 (.50)

FALSE.

REFERENCE
S.H. AOP-15, p 5.

ANSWER 7.12 (1.00)

d.

REFERENCE
S.H. AOP-16, p 6.

ANSWER 7.13 (1.00)

c.

REFERENCE
S.H. RC and PM, p 4-3 and preceeding change insert.

ANSWER 7.14 (.50)

FALSE.

REFERENCE
S.H. RC and PM, change insert between pages 4-2 and 4-3

ANSWER 7.15 (.50)

TRUE.

REFERENCE
S.H. RC and PM, p 4-4.

ANSWER 7.16 (1.00)

c.

REFERENCE
S.H. RC and PM, p 4-11.

ANSWER 7.17 (4.00)

a. Xenon decay.

b. 100. (+/- 5)

c. 320. (+/- 5) or 625°F

d. 1770.

e. -42, 0. (+/- 2)

f. Operations Manager.

g. hour.

h. 400. (+/- 10) [.5 total each letter]

REFERENCE
S.H. GP-02, Pp 14-17.

ANSWER 7.18 (2.00)

a. Rod Insertion Limit (RIL)

b. 500

c. Shutdown, recalculated

d. Countrate, 4 [.5 total for each letter]

REFERENCE
S.H. GP-03, Pp 17, 19.

ANSWER 7.19 (1.00)

~~a~~. b.

REFERENCE
S.H. GP-004, p 3.

ANSWER 7.20 (1.00)

a.

REFERENCE
S.H. GP-004, p 13.

ANSWER 7.21 (1.00)

b.

REFERENCE
S.H. Technical Specifications 3.9.1, P 3/4 9-1.

ANSWER 7.22 (1.00)

b.

REFERENCE
S.H. Technical Specifications 3.9.2, p 3/4 9-2.

ANSWER 7.23 (1.00)

d.

REFERENCE
S.H. Technical Specifications 3.10.4. p 3/4 10-4

ANSWERS -- SHEARON HARRIS 1 -85/06/10-HEMMING, W.

ANSWER 7.24 (1.00)

b.

REFERENCE
10 CFR 20, 20.102

~~ANSWER 7.25 (1.50)~~

~~a. Insoluble~~ Deleted

~~b. Soluble~~

~~c. Immersed in a Semipherical Cloud.~~

REFERENCE
10 CFR 20, Appendix B.

ANSWER 7.26 (1.50)

a. 3.

b. 275, PORV's. or 350

c. 10, 50.

REFERENCE
S.H. GP-07, Pp 5-7.

ANSWER 7.27 (1.00)

c.

REFERENCE
S.H. EOP-EPP-5, p 14.

ANSWER 8.01 (2.00)

Subcriticality
Core cooling
Heat sink
Integrity
Containment
Inventory [.5 ea. name] [.5 for order, no partial credit] (2.0)

REFERENCE
Critical Safety Function Status Trees

ANSWER 8.02 (1.00)

a

REFERENCE
Technical Specifications 4.02 p. 3/4 0-1, 3/4 0-2

ANSWER 8.03 (1.00)

1. Return the governor valve to operable status
2. Close at least one valve in the affected steam lead [0.5 ea.]

REFERENCE
Technical Specifications p. 3/4 3-83, B3/4 3-5

ANSWER 8.04 (3.00)

a. 1 gpm [0.2]--it is sufficiently low to allow for early detection of additional leakage [0.4].

b. 10 gpm [0.2]--allowance for leakage from known sources which would not interfere with detection of unidentified leakage [0.4]

c. 0 gpm [0.2]--may be indicative of an impending gross failure [0.4]

d. 31 gpm (at 2235 psig) [0.2]-- that SI flow will not be less than assumed in accident analysis in event of a LOCA [0.4].

e. 1 gpm for all S/G's or 500 gpd for any one S/G. [0.2]
-- ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR 100 limits in the event of a SGTR or Steam Line Break. [0.4]

~~e. 1 gpm TOTAL for all S/G's or 500 gpd for any one S/G. [0.2] -- ensures the dosage contribution from tube leakage will be a small fraction of Part 100 limits in event of a SGTR or a Steam Line Break [0.4]~~

REFERENCE
Technical Specifications 3/4.4.6

ANSWER 8.05 (1.50)

1. Maintain acceptable power distribution limits.
2. Maintain minimum shutdown margin.
3. Limit effects of rod misalignment.

REFERENCE
Technical Specifications pp. B 3/4 1-2, 1-4

ANSWER 8.06 (1.50)

Any THREE of the below:

1. MTC is within it's analyzed range.
2. Protection instrumentation is within it's operating range.
3. P-12 interlock is above it's setpoint.
4. Pressurizer is operable.
5. Reactor vessel is above minimun RT/NDT temperature.

REFERENCE
Technical Specifications, section 3.1.1.4, p B 3/4 1-2

ANSWER 8.07 (1.00)

c.

REFERENCE
S.H. PEP-101, p-4.

ANSWER 8.08 (1.00)

The final classification decision.

REFERENCE
S.H. PEP-101, p 4.

ANSWER 8.09 (1.00)

c.

REFERENCE
S.H. PEP-103, p-5.

ANSWER 8.10 (1.00)

a.

REFERENCE
S.H. PEP-301, p-7.

ANSWER 8.11 (1.00)

b

REFERENCE
S.H. PEP-381, p-4

ANSWER 8.12 (1.00)

c.

REFERENCE
S.H. OMM-005, p-6

~~ANSWER 8.13 (1.00)~~

deleted

~~REFERENCE~~
~~S.H. AP-007, p-6~~

ANSWER 8.14 (1.00)

c.

REFERENCE
S.H. OMM-002, p-4

ANSWER 8.15 (1.00)

c.

REFERENCE
S.H. AP-019, p-5 & 6.

ANSWER 8.16 (.50)

FALSE.

REFERENCE
S.H. AP-24, p-6.

ANSWER 8.17 (.50)

TRUE.

REFERENCE
S.H. AP-24, p-5

ANSWER 8.18 (1.00)

d.

REFERENCE
S.H. OMM-01, p-10.

ANSWER 8.19 (1.00)

b

REFERENCE
S.H. T.S. section 6, p 6-4.

ANSWER 8.20 (3.00)

a. Five (5).

b. SRO Limited to Fuel Handling.

c. One (1).

d. Shift Foreman, two (2).

e. STA.

f. RO license, SRO license. (point value)

REFERENCE
S.H. T.S. section 6, p 1 & 5.

ANSWER 8.21 (2.00)

a, d, e, f.

5. Any event that results in a major loss of emergency assessment, offsite response, or communications capability.

6. Any event that threatens the safety of the plant or hampers site personnel in the safe operation of the plant.

REFERENCE
10 CFR 20.403.

ANSWER 8.22 (1.00)

c.

REFERENCE
S.H. T.S. Section 6.7.

ANSWER 8.23 (1.00)

d.

REFERENCE
10 CFR 100.11.

ANSWER 8.24 (1.00)

c.

REFERENCE
10 CFR 50, appendix A.