ENCLOSURE 1

EXAMINATION REPORT - 50-261/OL-84-01

Facility Licensee: H. B. Robinson Steam Electric Plant
P. O. Box 790
Hartsville, SC 29550

Facility Name: H. B. Robinson

Facility Docket No. 50-261

Written and oral examinations were administered at H. B. Robinson Nuclear Plant near Hartsville, South Carolina.

Chief Examiner: Bruce A. Wilson, for 2/8/85
Edward A. Cook Date Signed

Approved by: Bruce A. Wilson 2/8/85
Bruce A. Wilson, Section Chief Date Signed

Summary:

Examinations on December 18-21, 1984

Oral examinations were administered to seven candidates; all of whom passed. Written examinations were administered to seven candidates, six of whom passed.

REPORT DETAILS

1. Persons Examined

SRO Candidates:

James A. Harding
Benjamin R. Parvin
Richard R. Stebbins

RO Candidates:

Stephen H. Atlee
Kenneth R. Darwin
Ralph C. Downey
Richard L. Randolph

Other Facility Employees Contacted:

*R. E. Morgan, General Manager
*R. S. Allen, Sr. Spec. Training
*W. M. Blaisdell, Sr. Spec. Training
*D. R. Nelson, Operations Supervisor
V. L. Smith, Training
*H. E. P. Krug, USNRC Senior Resident

*Attended Exit Meeting

2. Examiners:

*Edward A. Cook, USNRC RII
Frank Jaggar, EG&G

*Chief Examiner

3. Examination Review Meeting

At the conclusion of the written examinations, the examiners met with V. L. Smith, R. S. Allen, W. M. Blaisdell, and D. R. Nelson to review the written examination and answer key. The following comments were made by the facility reviewers:

a. SRO Exam

1. Question 6.02

Facility Comment:

(a) Answer should be PZR PORV's (Power Operated Relief Valves) vs Safety Valves.

(b) Alternate answer: "If high temperature water is sensed then a temperature directing valve (TCV-143) will automatically act to divert water around the demineralizers."

NRC Resolution: (a) PZR RORV's is proper nomenclature
(b) Alternate answer is accepted.

2. Question 6.03

Facility Comment: Due to interpretation of question and alternate answer could be: "To prevent excessive cooldown of the RCS."

NRC Resolution: Alternate answer accepted.

3. Question 6.08

Facility Comment: A reactor trip will not occur on PZR high level due to let down isolation. On a low level signal in the PZR the letdown isolation valves will shut and the heaters will trip off. As the level builds up in the PZR the letdown isolation valves will reopen. However, without operator action the PZR heaters will not reenergize. (They must be manually reset). Therefore, the resulting trip will occur from low PZR pressure.

NRC Resolution: Logic Diagram CP-300-5379-3693 reference accepted for making change.

4. Question 6.10

Facility Comment: (a) Answer: The signal for the auto start of the steam driven AFW pump comes from a loss of power to 4160 Bus 1 and 4. We recommend that this or black out signal be used as acceptable answers.

(b) Answer: Should be "Low-Low S/G Vessel (2/3) in <u>one</u> S/G."

NRC Resolution: Logic Diagram CP-300-5379-2762 reference accepted for making change.

5. Question 8.05b

Facility Comment: OMM-006 requires that the shift foreman and refueling SRO have an SRO license. However, at HBR in the past we have always used licensed RO's on the CV and SFP manipulator bridges in addition to the requirements of OMM-006.

NRC Resolution: OMM-006 and past practice accepted.

6. Question 8.06

Facility Comment: (a) Shift Foreman, Senior Reactor Operator, Control Operator, Fire Protection Tech. Aide and Auxiliary Operator.

(b) The correct answer should be "Portions of the MEL are always completed, hot or cold."

NRC Resolution: Alternate and correct answer accepted.

b. RO Exam

1. Question 2.21

Facility Comment: The answer given for the explosive range of H2 is 5% to 70% in accordance with the system description.

In the classes dealing with Mitigating Core Damage, we give slightly different numbers (4% - 75%). Also, this question could be interpreted to mean the actual explosive range where Hydrogen burns at supersonic velocities (18% - 59%). The candidate should be graded as to how the question was interpreted.

Reference - General Physics Mitigating Core Damage Page 10.6.

NRC Resolution: Question deleted. Conflicting material provided by CP&L training department during the examination review.

2. Question 3.17

Facility Comment: The CCW vent valve (RCV-609) has recently been gagged open as per MOD-835. The candidates have not yet received training on this modification; therefore, they should probably answer true. However, if they have become aware of this MOD via other paths they may answer "False" based upon this recent MOD.

NRC Resolution: Question deleted. A major plant system change occurred without the examiner being notified prior the examination.

3. Question 3.19

   Facility Comment: Answer "d" is true only to start the first condensate pump. "a" could be true if a condensate pump is already running.

   NRC Resolution: Comment accepted and the question is deleted.

4. Question 3.26

   Facility Comment: The answer key has a typographical error; "G3-AST". This should be 63-AST. In addition, we have not required candidat:s to know the numbers of various oil pressure switches. Normally this answer should be "auto stop oil pressure low."

   NRC Resolution: "Auto stop oil pressure low" is the correct answer. "63-AST low pressure" is also correct.

5. Question 4.06

   Facility Comment: Question 4.06 dealt with the hydrogen concentration in the RCS. The answer key stated 15cc/kg. The new GOP 003 states 25cc/kg. We request that either answer be accepted. The training department taught 15cc/kg but the new GOP 003 has 25cc/kg.

   NRC Resolution: Question deleted. A plant procedure changed without the examiner being notified prior to the examination.

6. Question 4.23

   Facility Concern: The answer for this question is analyzed on a case-by-case basis as there is no clear direction as to "Normal" and "Adverse" conditions in containment listed in HBR procedures.

   NRC Resolution: Comment Accepted.

4. <u>Exit Meeting</u>

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 was one generic weakness (greater than 75 percent of candidates giving incorrect answers to one examination topic) noted during the oral examination.

Candidates were not familiar with the new procedure numbers and titles. They had difficulty finding the procedure to be used for an abnormal plant condition.

The following discrepancies were noted by the Chief Examiner and are being followed up by the Senior Resident Inspector:

a. Operations staff did not complete the required shift relief procedure prior to relieving the watch (December 20, 1984, 0800-1600 shift).

b. No work permit "OWP" log book "binder" exists in the control room. No index or control feature exists if a OWP is missing.

c. No book of Tech Spec related equipment on which maintenance is in progress or tagged out exists in the control room.

d. Electrical tagouts do not include breaker numbers.

e. There is an apparent lack of control of the "control keys" for the key locker in the control room. The locker key is kept in the top draw of the Shift Foreman's desk and anyone may get access to the key without the Shift foreman knowing about it. This would gain him access to vital equipment throughout the plant without the Shift Foreman's concurrence.

f. Control room gauges on the reactor control panel were not functioning properly or had varying indications at the same parameter. Candidates could not tell if they had a "false signal" or it was meter deviation. The cold calibrated pressurizer level indication was out of commission but had not been tagged out (LI 462). All the candidates and all the licensed operators in the control room for the week were unaware of this. The plant was being controlled on a pressurizer bubble with an indicated <u>hot</u> calibrated pressurizer level of 25% (December 20, 1984). All the operators who were asked by the Chief Examiner stated that true pressurizer level is very close to the <u>hot</u> indication. Actual pressurizer level was 7.5% below the indicated level.

g. The number of people in the control room during the conduct of oral examinations appeared to be excessive. In addition to excessive numbers of people with accompanying noise, the control room is apparently also used as a lunch room. The license candidates at times found it difficult to show reference material (P&IDs, Tech Specs and procedures) to the examiners since desk space was cluttered with food, clothing, etc. The examiners felt that the conduct of the operations staff during the examinations was detrimental to the candidates performance.

5. Additional formal comments on the examinations were submitted by the H. B. Robinson staff by letter dated December 27, 1984 (RSEP/84-1260). This letter is attached to this report. Our responses to the five suggestions and comments are as follows:

   a. The point value assigned to questions has been reviewed and determined to be appropriate and consistent with the guidance in Examiner Standard ES-202. The question on hydrogen concentration cited by the letter was deleted because of the conflict of answers between what is taught to the trainees (15cc/kg) and what is contained in the new GOP-003 (25cc/kg).

   b. Concerning examination guidelines for symptomatic procedures, we expect candidates to demonstrate knowledge of the contents of normal, off-normal and emergency procedures, regardless of whether the procedure contains immediate action steps. Those steps which are not immediate actions need not be memorized, but the candidate must be able to describe conceptually the objectives and methods used to achieve these objectives for all emergency and off-normal procedures.

   c. The additional information provided by the third comment was considered during the grading of the exam.

   d. The subject of standard and emergency procedures is included under paragraph 10 CFR 55.21(j) as required content of RO written examinations. NUREG 1021, ES-202, Section B.4 contains guidance on the staff's interpretation of the subject matter of this category. We consider that "the length of time temporary procedures and special procedures are effective..." falls within the scope of "...limitations of normal operating procedures."

   e. Answers that are true will be accepted, even if not on the answer key.

Following the exam review, we found it necessary to delete four questions from the RO examination due to conflicting or inadequate reference material which was provided to us by your training department. The emphasis on NRC written examinations will continue to move toward objective, site-specific validated questions. We are endeavoring to improve our examinations such that they are a valid and reliable test of the candidates knowledge. We expect that you will make a similar commitment to the improvement of your training material.

Enclosure 3

U. S. NUCLEAR REGULATORY COMMISSION

V.L. Smith
Richard S. Allen
William M Blaisdell
Duane R Nelson

REACTOR OPERATOR LICENSE EXAMINATION

Facility: H. B. Robinson
Reactor Type: PWR-W
Date Administered: December 18, 1984
Examiner: Edward Cook
Applicant: ____________
MASTER

INSTRUCTIONS TO APPLICANT:

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

Category Value	% of Total	Applicant's Score	% of Cat. Value	Category
25.0	25	______	______	1. Principles of Nuclear Power Plant Operations, Thermodynamics, Heat Transfer and Fluid Flow
25.0	24 ~~25~~ ec	______	______	2. Plant Design Including Safety and Emergency Systems
25.0	23.5 ~~25~~ ec	______	______	3. Instruments and Controls
25.0	23 ~~25~~ ec	______	______	4. Procedures - Normal, Abnormal, Emergency & Radiological Control
100.0	95.5 ~~100~~	______	______	TOTALS

Final Grade ______ %

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

____
Applicant's Signature

1/29

PRINCIPLES OF NUCLEAR POWER PLANT OPERATIONS TOTAL POINTS = 25.0

1.01 Indicate how the following will affect unit efficiency at steady state power level. (Consider the affected parameter only and indicate Increase, Decrease or No Change)

(a) Absolute condenser pressure changes from 1.0 psi to 1.5 psi (0.5)
(b) Total S/G blowdown is changed from 35 gpm to 40 gpm (0.5)

1.02 For each of the two transients below, explain all of the reactivity effects that cause reactor power to change throughout the transient.

In your discussion STATE whether reactor power will stabilize HIGHER THAN/LOWER THAN/or the SAME AS initial power.

ASSUME:

1. Initial power = 50%
2. Rod control is in the manual mode.
3. No operator action
4. End of core cycle
5. Turbine controls are in automatic mode
6. No reactor trip.

TRANSIENTS

a. Steam Generator PORV fails open (1.5)
b. One Band D Control rod drops (No reactor trip or Turbine Runback) (2.0)

1.03 State two reasons why equilibrium Xenon has significantly more negative worth than does equilibrium samarium in an operating reactor. (2.0)

1.04 Assuming all other DNB parameters remain constant, how will the following changes affect DNBR? (Limit your answer to INCREASES, DECREASES, or DOES NOT CHANGE DNBR).

1. Reactor thermal power increases (0.5)
2. Average RCS temperature increase (0.5)
3. RCS pressure increases (0.5)
4. RCS flow increases (0.5)

CATEGORY 1 CONTINUED ON NEXT PAGE

2/24

1.05 a. True or False

One of the pump laws for centrifugal pumps states that the volume flow rate is inversely proportional to the speed of the pump. (0.5)

b. True or False

As VCT temperature decreases, mass flow rate from the positive displacement pump (PD) remains unchanged. (0.5)

c. True or False

Pump runout is the term used to describe the condition of a centrifugal pump running with no volume flow rate. (0.5)

1.06 The following statements concern subcritical multiplication. Choose the <u>one</u> [bracketed] word that makes the statements correct.

a. As Keff approaches unity, a [larger/smaller] change in neutron level results from a given change in Keff. (0.5)

b. As Keff approaches unity, a [shorter/longer] period of time is required to reach the equilibrium neutron level for a given change in Keff. (0.5)

1.07 Although the U238 resonance capture peaks broaden and flatten with increased fuel temperature, the area under the peak remains the same. Why then is there an increase in neutron capture as the fuel temperature is increased? (1.0)

1.08 a. How <u>and</u> why does the moderator temperature <u>coefficient</u> (MTC) change (more or less negative) as temperature is increased at a <u>constant boron concentration</u>. (1.0)

b. How <u>and</u> why does the moderator temperature <u>coefficient</u> (MTC) change as boron concentration is increased at a <u>constant temperature</u>. (1.0)

1.09 If steam goes through a throttling process, specifically as in a leak from a main steam header high pressure line to atmosphere, will the following parameters <u>Increase</u>, <u>Decrease</u>, or <u>Remain the Same</u>?

1. Enthalpy (0.5)
2. Pressure (0.5)
3. Entropy (0.5)
4. Specific Volume (0.5)
5. Temperature (0.5)

CATEGORY ONE CONTINUED ON NEXT PAGE

1.10 Assume one Reactor Coolant Pump trips at 30% power, without a reactor protective system actuation or a change in turbine load. Indicate whether the following parameters will INCREASE, DECREASE or REMAIN THE SAME

a. Flow in OPERATING reactor coolant loops (0.5)
b. Affected loop S/G level INITIALLY (0.5)
c. Reactor vessel Delta P (0.5)
d. Core Delta T (0.5)
e. OPERATING LOOP steam generator pressure. (0.5)

1.11 How does each of the following parameters change (INCREASE, DECREASE, or NO CHANGE) if one main steam isolation valve closes with the plant at 50% load. Assume all controls are in automatic and that no reactor trip occurs.

a. Affected loop steam generator level (INITIAL CHANGE ONLY) (0.5)
b. Affected loop steam generator pressure (0.5)
c. Affected loop cold leg temperature (0.5)
d. Unaffected loop steam generator level (INITIAL CHANGE ONLY) (0.5)
e. Unaffected loop steam generator pressure (0.5)
f. Unaffected loop cold leg temperature (0.5)

1.12 True or False

For the same constant startup rate it takes the same time to change reactor power from 20% to 40% as it does from 40% to 60%. (0.5)

1.13 True or False

The build up of Pu-239 causes a reduction in Beff from 0.007 at BOL to 0.005 at EOL. This decrease enhances the effect of a given reactivity change and causes the system to react a bit more rapidly to the changes in Keff due to the decrease in average neutron lifetime. (0.5)

1.14 True or False

The production of power by fission ceases within a few seconds of a reactor trip and only decay heat is produced. (0.5)

1.15 True or False

The use of a sliding Tavg program allows Robinson Nuclear Plant PWR to operate with a higher thermodynamic efficiency than does a constant Tavg program. (0.5)

CATEGORY ONE CONTINUED ON NEXT PAGE

1.16 A heat balance is run at full power and the power range channels are adjusted to read 100%. State whether indicated power would be equal to, greater than, or less than, actual power for the following conditions:

a. Indicated feedwater temperature was 10°F higher than actual. (0.5)

b. Heat output of the reactor coolant pumps was neglected in the calculation. (0.5)

END OF CATEGORY 1

WRITE "END OF CATEGORY 1" ON YOU ANSWER PAGE.

5/24

CATEGORY TWO - PLANT DESIGN - SAFETY AND EMERGENCY SYSTEMS

TOTAL POINTS - 25

2.01 The largest contributor to reactor vessel thermal stress which the Thermal Shield is installed to mitigate, is: (1.0)

a. neutron embrittlement from fast neutrons

b. heat generated by the absorption of gamma energy.

c. heat generated by the absorption of fast neutrons

d. neutron embrittlement from thermal neutrons

e. diversion of a portion of the core inlet flow to cool the hot reactor vessel wall.

2.02 The combined capacity of the pressurizer safety valves is equal to: (1.0)

a. The maximum surge rate resulting from a complete loss of feedwater flow without a reactor trip or any other control except a main turbine trip.

b. The maximum surge rate resulting from a complete loss of load without a reactor trip or any other automatic control.

c. The maximum surge rate resulting from an over power condition in the primary system due to an inadvertant dilution.

d. The maximum surge rate resulting from complete loss of load without reactor trip or any other control except that the secondary plant safety valves are assumed to operate when steam pressure reaches their setpoint.

e. The maximum surge rate resulting from continuous operation of the positive displacement charging pump with a water solid pressurizer.

2.03 List five ways in which the signal which initiates operation of the Safety Injection System for the injection phase is generated. (Logic required) (Setpoints not required) (2.5)

CATEGORY 2 CONTINUED ON NEXT PAGE

2.04 The containment spray system provides a means of adding sodium hydroxide to the spray system for: (1.0)

a. maintaining an alkaline pH in containment atmosphere to prevent system failure in post LOCA long term cooling systems due to excessive system corrosion from boric acid.

b. maintaining a neutral pH in the containment sample to prevent precipitation of boric acid after injection of the BIT on a LOCA.

c. maintaining the ability, of the safety injection water, for absorption of containment free iodine in post LOCA environment.

d. maintaining adequate core cooling by preventing hydrolysis of water due to high fuel clad temperature on a LOCA.

e. maintaining containment H_2 concentrations low to prevent explosive mixtures in containment atmosphere on a LOCA.

2.05 True or False (0.5)

A.S.I. signal can be reset one minute after actuation if the status of equipment is to be changed.

2.06 For post <u>LOCA</u> sampling: (1.0)

a. all "PASS" sample flow is stored in a rad liquid discharge tank prior to release off site to minimize activity release to the environment.

b. all "PASS" sample flow is returned to the containment to preclude contamination of other plant systems.

c. all "PASS" sample flow is passed through a long run of pipe to allow N^{16} to decay which limits the exposure of the chemistry technician to 3 Rem to whole body.

d. all "PASS" sample flow is purified via an ion exchanger prior to the sample being withdrawn to remove fission product gases from the sample which would be released to the atmosphere.

CATEGORY 2 CONTINUED ON NEXT PAGE

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2.07 Each of the two station batteries is sized to carry its expected shutdown loads following a plant trip and loss of all AC power for a period of: (1.0)

a. 1 hour
b. 2 hours
c. 4 hours
d. 8 hours
e. 16 hours

2.08 True or False (0.5)

Steam Generator blowdown and sample isolation valves will reopen automatically when the alarm of the radiation monitor is cleared, if the blowdown lines and the continuous blowdown sample lines were isolated due to the high radiation alarm.

2.09 H_2 is added to the Reactor Coolant System to scavenge O_2 to reduce general corrosion. The chemical reaction which takes place occurs mostly: (1.0)

a. In the gas space in the top of the VCT.

b. In the liquid space in the bottom of the VCT.

c. In the Reactor Coolant loop where the CVCS discharges into the primary.

d. In the Reactor Core.

e. In the pressurizer steam space when spray is initiated.

CATEGORY 2 CONTINUED ON NEXT PAGE

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2.10 The outlet temperature on the letdown side of the non-regenerative heat exchanger is controlled: (1.0)

a. by adjusting the component cooling water flow through the shell side via a flow control valve that receives a signal from a temperature controller that sense letdown temperature.

b. by adjusting the letdown flow via a flow control valve that receives a signal from a temperature controller that senses letdown temperature.

c. by adjusting the charging flow via a flow control valve, or positive displacement charging pump speed controller, that receives a signal from a temperature controller that senses letdown temperature.

d. by transferring heat from the letdown line to the charging line in the regenerative heat exchanger.

e. by diverting a portion of letdown flow to the hold up tanks prior to entering the mix bed ion exchangers.

2.11 All the signals which auto close steam generator blowdown valves FCV-1930A, FCV-1930B, FCV-1931A, FCV-1931B, FCV-1932A, FCV-1932B are: (pick one answer) (1.0)

a. loss of power supply, high radiation signal, and containment phase "A" isolation signal.

b. loss of power supply, auto start of auxiliary feedwater pump, and high radiation signal.

c. loss of power supply, auto start of auxiliary feedwater pump, high radiation signal, loss of both main feedwater pumps, and containment phase "A" isolation.

d. loss of power supply, high radiation signal, loss of both main feedwater pumps, and containment phase "B" isolation signal.

e. Containment phase "B" isolation signal.

CATEGORY 2 CONTINUED ON NEXT PAGE

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2.12 When venting the VCT gas space to the Waste Disposal Gas System, via CVC-258 and PCV-117, PCV-117 will: (1.0)

a. shut if VCT pressure is reduced to 15 psig which is required for RCP operation.

b. shut if the high radiation alarm is received on the vent line to the Waste Disposal Gas System indicating gross fuel clad failure.

c. shut if an Emergency Low-Low Level on VCT alarm is received to prevent bleeding the VCT dry.

d. shut if an Emergency Hi-Hi Level on VCT alarm is received to prevent carrying over VCT liquid and damaging the Waste Gas Disposal System.

e. Shuts if VCT pressure is reduced to 25 psig which is required to prevent cavitation of the coolant charging pumps.

2.13 PCV-1040 is set to maintain a 5 psi differential across RCV-014 during release of the contents of a gas decay tank to atmosphere because: (1.0)

a. an excessive head loss across RCV-014 would lead to valve seat erosion.

b. an excessive head loss across RCV-014 would cause entrained moisture flashing to steam resulting in water hammer and valve damage.

c. a low differential pressure ensures that RCV-014 will seat properly on an auto close signal from the plant stack radioactive gas monitor on high radiation levels.

d. a constant differential pressure gives a constant release rate regardless of tank pressure.

e. a higher differential pressure would force RCV-014 closed due to it being a reverse seated globe valve.

CATEGORY 2 CONTINUED ON NEXT PAGE

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2.14 The superheat, which is created by the operation of the steam dump valves, is eliminated by spray valves which: (1.0)

a. reduces the detrimental effects of the superheated steam due to direct impingement on the condenser tubes and minimizes excessive stresses.

b. Prevents a loss of the arming signal due to low condenser vacuum if all five condenser steam dump valves are open, by making use of the increased heat transfer rate for saturated steam.

c. prevents a "hot condenser" due to boiling of the circulating water in the condenser tubes adjacent to the steam dump discharge by lowering the temperature of the steam.

d. Reduces the velocity of the superheated steam after undergoing a venturi effect as it passes through the throat of the steam dump valve.

e. Prevents possible rupture of the condenser expansion joint on the condensate line, from flashing of the condensate stored in the condenser hotwell which would result from the direct admission of superheated steam into the sub cooled liquid.

2.15 List four conditions which will trip a main condensate pump. (Set points are not required) (1.0)

2.16 List three sources of water to the auxiliary feed pumps. (1.5)

2.17 List all conditions which will automatically trip a main feedwater pump. (Set points not required) (2.0)

CATEGORY 2 CONTINUED ON NEXT PAGE

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2.18 In the Steam Generator Water Level Control System: (1.0)

a. for normal operation and for large transients the predominate signal for controlling the feedwater regulating valves is level error.

b. for normal operation and for large transients the predominate signal for controlling the feedwater regulating valves is flow error.

c. for normal operation the predominate signal for controlling the feedwater regulating valves is flow error, and for large transients the level error will be most significant.

d. for normal operation the predominant signal for controlling the feedwater regulating valves is the level error, and for large transients the flow error will be most significant.

e. for normal operation the predominant signal for controlling the feedwater flow is pimp (first stage impulse pressure).

2.19 The Heater Vents and Drains System raises the hotwell condensate temperature from __(a)__ °F to __(b)__ °F at heaters No. 6A and B feedwater outlet if the plant is operating at rated load. (1.0)

2.20 The main turbine emergency back-up D-C motor drive oil pump is started automatically when the bearing oil pressure drops to __(a)__ psig. (1.0)

2.21 The Generator Hydrogen Gas System provides the operator with indication of gas pressure, temperature and purity, and enables him to prevent an explosive mixture of hydrogen and air which is __(a)%__ to __(b)%__ hydrogen by volume. (1.0) Deleted

2.22 List four (4) signals which close the containment purge supply and exhaust isolation valves. (2.0)

END OF CATEGORY 2
WRITE "END OF CATEGORY TWO" ON YOUR ANSWER PAGE

12/24

CATEGORY THREE INSTRUMENTS AND CONTROLS TOTAL POINTS 25.0

3.01 List all manual rod withdrawal stops (setpoints and logic not required) (1.0)

3.02 True or False
During cooldown, do not approach 40% steam flow indication as this will, in coincidence with low steam line pressure or low Tavg, initiate Safety Injection and Steam Line Valve Closure. (0.5)

3.03 List six (6) items which are reset when the "Start-Up Pushbutton" mounted on the control board is pushed prior to plant startup. (1.5)

3.04 True or False
A rod Control "Urgent Alarm" automatically de-energizes the lift coil and continuously energizers both gripper coils at reduced current. (0.5)

3.05 True or False
A red rod bottom light is energized for each full length control rod that is less than 35 steps from the bottom of the core. (0.5)

3.06 True or False
Receipt of an output signal from a rod bottom bistable will cause a load reduction at a rate of 150% per minute and load will continue to reduce until the turbine 70% bistables clear. (0.5)

3.07 True or False
During a reactor plant startup, with the reactor critical in the intermediate range and source range high voltage secured, the source-range-level trips are automatically reactivated, and high voltage is restored, if one intermediate range channel drops below the permissive (P6) level. (0.5)

3.08 The High Nuclear Flux (low power range) trip can: (1.0)

a. be manually blocked when 1/4 of the power range channels is above P-10

b. be manually reinstated when 2/4 of the power range channels is below P-10

c. be manually blocked when 2/4 of the power range channels is above P-10

d. be automatically reinstated when 2/4 of the power range channels is below P-10

e. be automatically blocked when 1/4 of the power range channels is above P-10

CATEGORY THREE CONTINUED ON NEXT PAGE

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3.09 True or False
The overpower ΔT (OPΔT) trip protects the core against departure from nucleate boiling (DNB) (0.5)

3.10 True or False
The Low Pressurizer Pressure trip limits the necessary range of protection afforded by the over temperature ΔT (OTΔT) trip. (0.5)

3.11 True or False
The overtemperature ΔT (OTΔT) trip is automatically blocked below the P-7 setpoint. (0.5)

3.12 True or False
The High Pressurizer Pressure Trip limits the range of required protection from the overtemperature ΔT (OTΔT) trip (0.5)

3.13 True or False
The Pressurizer High Water Level trip is automatically blocked below P-7. (0.5)

3.14 List four (4) different methods by which the reactor protection system senses low reactor coolant flow (logic not required). (1.0)

3.15 The Incore Movable Miniature Neutron Flux Detectors employ: (0.5)

(a) 90% enriched U-235
(b) 2-3% enriched U-235
(c) 2-3% enriched U-238
(d) 90% enriched Pu-240
(e) 2-3% enriched Pu-240

3.16 True or False

The Condenser Air Ejector Gas Monitor (R-15) alarm automatically shifts the air ejector exhaust from atmosphere to the plant vent. (0.5)

3.17 True or False
The Component-Cooling Liquid Radiation Monitor (R-17) alarm signal initiates automatic closure of the component cooling-surge tank vent valve. (0.5) Deleted

3.18 True or False
Steam line flow is measured as a function of the differential pressure across a <u>"flow nozzle" (venturi)</u> in the steamline downstream from each steam generator. (0.5)

3.19 To start a condensate pump the hotwell level <u>must</u> be normal <u>and</u> the discharge valve <u>must</u> be: (1.0)

Deleted

a. full open
b. no less than 95% open
c. 50% to 95% open
d. no more than 5% open
e. full shut

CATEGORY THREE CONTINUED ON NEXT PAGE

14/24

3.20 Low Pressure Heater Bypass Valve (HCV-1459) will open automatically when: (1.0)

a. both heater drain pumps trip
b. one heater drain pump is lost (one still running) and suction pressure to the main feed pumps decreases to 400 psig.
c. suction pressure to the main feed pumps decreases to less than 400 psig.
d. one heater drain pump is lost (one still running) and suction pressure to the main feed pump decreases to 300 psig
e. one main feedwater pump trips (one still running) and suction pressure to the main feed pump decreases to 350 psig

3.21 List four (4) conditions which will automatically start both motor driven auxiliary feed pumps. (1.0)

3.22 If the indicated pressurizer level is maintained constant during cooldown: (0.5)

a. actual level will stay constant
b. actual level will go down
c. actual level will go up

3.23 List ten signals which would trip the turbine (setpoints and logic are not required) (1.0)

CATEGORY THREE CONTINUED ON NEXT PAGE

15/21

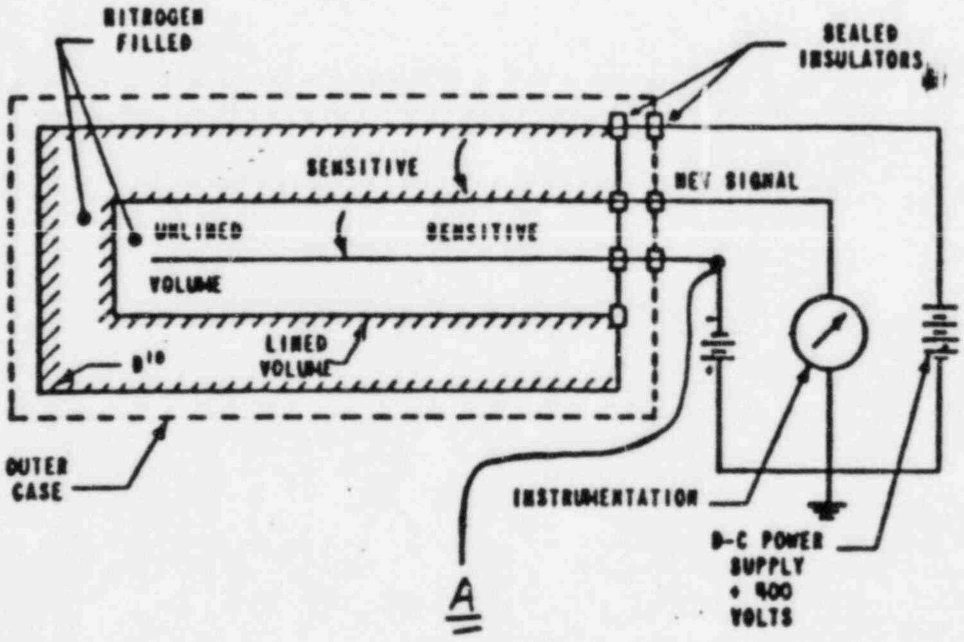
3.24 Below is a diagram for intermediate range channel one detector.

I. When point A is grounded at 100% power the control room indication for IR channel one will: (1.0)

a. fail high off scale
b. increase but remain on scale
c. will not change significantly
d. decrease but remain on scale
e. fail low off scale

II. When point A is grounded at 10^{-10} amps IR power, the control room indication for IR channel one will: (1.0)

a. fail high off scale
b. increase but remain on scale
c. will not change significantly
d. decrease but remain on scale
e. fail low off scale



CATEGORY THREE CONTINUED ON NEXT PAGE

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3.25 The valves closed by the turbine overspeed protection controller are: (1.0)

a. Governor and stop valves
b. Intercept and reheat valves
c. Governor and intercept valves
d. Stop and reheat valves
e. Governor and reheat valves

3.26 List all the signals which are sent to the reactor protection system to indicate a turbine trip (Logic and setpoints are not required). (1.0)

see answer key

3.27 At 75% power the pressure compensation on "A" loop steamline flow fails low, which causes: (1.0)

a. "A" loop steam line indicated steam flow to indicate high
b. "A" loop steam line actual steam flow to increase
c. "A" loop steam line indicated steam flow to indicate low
d. "A" loop steam line actual steam flow to decrease
e. "A" loop steam line indicated steam flow does not change

3.28 Upon loss of instrument air, how will the following valves fail? (Closed, Open)

a. HCV758 (RHR heat exchanger discharge valve) (0.5)
b. CVCS-310B (Charging to RCS loop 2) (0.5)
c. FCV 605 (RHR heat exchanger bypass valve) (0.5)
d. PCV-145 (letdown line pressure control valve) (0.5)

3.29 Where does the D.S. diesel output breaker (32b) receive control power to charge closing springs? (1.0)

3.30 With control rods in auto, and all other controls in normal, what affect will the following malfunctions have on rods. (Rod move in, rods move out, no affect)

a. Selected 1st stage pressure fails low. (0.5)
b. "A" loop control Thot fails low. (0.5)

END OF CATEGORY THREE
WRITE "END OF CATEGORY THREE" ON YOUR ANSWER SHEET.

17/24

CATEGORY FOUR - PROCEDURES

TOTAL POINTS = 25

4.01 A "temporary change" to procedures of the Plant Operating Manual may be in effect for a maximum of: (1.0)

a. 3 days

b. 7 days

c. 21 days

d. 30 days

e. 90 days

4.02 "Special Procedures" are intended to be used on a one time basis only, but on occasions when they are required to be used over a longer period of time, they may be used up to a maximum period of: (1.0)

a. one week

b. two weeks

c. three months

d. six months

e. one year -

CATEGORY FOUR CONTINUED ON NEXT PAGE

4.03 True or False (0.5)

During hot operations, a licensed Senior Reactor Operator is required to be in the Control Room at all times.

4.04 During filling and venting of the RCS (GP-001) boron concentration must be verified: (1.0)

a. continuously
b. once every one half hour
c. once every 4 hours
d. once a shift
e. once a day

4.05 For primary plant heatup (GP-002) from cold solid to hot subcritical no load TAVE, the maximum pressurizer heatup rate shall not exceed: (1.0)

a. 100 °F/Hr
b. 200 °F/Hr
c. 5 °F/Hr below 100 °F
d. 15 °F/Hr below 160 °F
e. 60 °F/Hr above 220 °F

CATEGORY FOUR CONTINUED ON NEXT PAGE

4.06 For a normal reactor startup (GP-003) the reactor will not be taken to power until hydrogen concentration is at least: (1.0)

a. 5 cc/kg

b. 15 cc/kg

c. 25 cc/kg

d. 50 cc/kg

e. 100 cc/kg

Deleted

4.07 For a reactor trip recovery, (GP-004) the maximum start up rate is: (1.0)

a. +0.3 DPM steady state and +1.0 DPM transient

b. +1.0 DPM steady state and +1.0 DPM transient

c. +1.0 DPM steady state and +3.0 DPM transient

d. +3.0 DPM steady state and +3.0 DPM transient

e. +3.0 DPM steady state and +5.0 DPM transient

4.08 For a plant shutdown from power operations (GP-006) to hot shutdown conditions the maximum rate of power reduction for normal operations is: (1.0)

a. 1%/hr

b. 5%/hr

c. 1%/min

d. 5%/min

e. manual reactor trip

CATEGORY FOUR CONTINUED ON NEXT PAGE

20/21

4.09 For a plant shutdown from power operations to hot shutdown conditions, (GP-006) the maximum allowed temperature difference between the loops is: (1.0)

a. 1° F

b. 5° F

c. 25° F

d. 50° F

e. 75° F

4.10 During draining of the Reactor coolant System, (GP-008) "The water level in the pressurizer surge line should not be allowed to decrease to the point where air will enter the Reactor Coolant loop piping, unless the proposed maintenance operations require it." The reason for this requirement is: (1.0)

a. To prevent air binding of the Reactor Coolant Pump bearings.

b. To prevent air binding of the loop flow detectors.

c. To prevent the reactor coolant pump first stage seal from air exposure.

d. To ensure that the steam generator tubes remain essentially full of coolant.

e. To ensure the RHR loop suction is below the actual RCS water level.

CATEGORY FOUR CONTINUED ON NEXT PAGE

4.11 "Prior to initiating a primary plant cooldown on natural circulation, each steam generator must be steamed for at least 30 minutes." (GP-012) the reason for this is: (1.0)

a. Prevent thermal stratification of the steam generator water due cold aux. feed water admission.

b. to ensure that pressure excursions in the reactor coolant system, from failure to remove decay heat do not result in opening the pressurizer safety valves.

c. to ensure natural circulation due to the coolant density difference between Th and Tc legs is maintained.

d. to ensure a balanced boron concentration in the Reactor Coolant System.

e. to ensure that thermal shock is minimized on the reactor vessel by allowing RCS temperatures to stabilize prior to commencing the cooldown.

4.12 List all Reactor Operator immediate actions for failure of the control rod bank to move on a normal turbine load increase. (1.0)

4.13 List all Reactor Operator immediate actions for Loss of a feedwater pump at full turbine load. (1.0)

CATEGORY FOUR CONTINUED ON NEXT PAGE

23/24

4.14 List four (4) Reactor Operator immediate actions for a slowly decreasing condenser vacuum. (1.0)

4.15 List the Reactor Operator immediate actions for a 35% electrical load rejection when operating at rated elecrtical load. (1.0)

4.16 In accordance with AOP-016, Excessive Primary Plant Leakage, list four (4) symptoms of a primary to secondary leak. (1.5)

4.17 List the Reactor Operator immediate actions for loss of component cooling water to RCP motor oil coolers at 100% power. (1.0)

4.18 The reactor is at 100% power when the main turbine trips due to excessive vibration but the control rods do not trip. List all reactor operator actions which you will take to add negeative reactivity to the core which is observed to be critical. (2.0)

4.19 List all reactor operator actions which you will take to reduce steam flow should a reactor trip occur with no turbine trip. (1.0)

4.20 State SI termination cirteria for normal containment conditions. (2.0)

CATEGORY FOUR CONTINUED ON NEXT PAGE

4.21 Switch to alternate AFW water supply if CST level decreases to less than: (1.0)

a. 5%

b. 10%

c. 15%

d. 20%

e. 25%

4.22 State RCP trip criteria for a small break LOCA. (state all assumptions) (1.0)

4.23 The end path procedures (EPP-Foldout) gives various setpoints and important parameters which are used to determine important operator actions. The numbers provided are different for the same parameter depending on containment conditions. What are "normal", and what are "adverse" containment conditions. (1.0)

Deleted

END OF CATEGORY FOUR

WRITE "END OF CATEGORY FOUR" ON YOUR ANSWER SHEET

24/24

Answers
MASTER

1.01 a. Decrease (0.5)
b. Decrease (0.5)

General Physics Co., HTFF February 1981, Section 1.6 Typical PWR Cycle

1.02 a. 1. + reactivity: from the lowered RCS temp. (0.5)
2. - Reactivity: from increased fuel temp. (power increase) (0.5)
3. Power stabilizes at a higher level (0.5)

b. 1. - reactivity: from dropped rod (0.5)
2. + reactivity: from the decreased fuel temp. (power decrease) (0.5)
3. - reactivity: from the increased fuel temp. (as power is turned and increases) (0.5)
4. Power stabilizes at the same level (0.5)

Reference: Westinghouse PWR core physics - Reactivity Coefficients
HBR SES 24

1.03 The production of Xe from fission and decay is greater. (1.0)
The absorption cross section for Xe is much higher. (1.0)

Reference: Westinghouse PWR Core Physics pg. I-5.63 and I-5.77
HBR SES 37, 38, 39

1.04 1. Decreases (0.5)
2. Decreases (0.5)
3. Increases (0.5)
4. Increases (0.5)

Reference: General Physics Co., Rx PWR Limits pg. 243
General Physics Co., Boiling Heat Transfer pg. 122

1.05 a. False (0.5)

Reference: General Physics HTFF Pump laws pg 322

b. False (0.5)

Reference: General Physics Co., Pump Laws pg. 327
General Physics Co., Units of Fluid Flow pg. 288

c. False (0.5)

Reference: General Physics Co., Pump Terminology pg. 320

1.06 a. <u>Larger</u> (0.5)

Reference: Westinghouse Nuclear Sources and Subcritical Multiplication - Relationship Between Subcritical Multiplication and Keff pg. I-4.13
HBR SES 41, 42

b. <u>Longer</u> (0.5)

Reference: Westinghouse Nuclear Sources and Subcritical Multiplication - Subcritical Multiplication Kinetics, pg. I-4.26
HBR SES 41, 42

1.07 The neutron sees a significant absorption cross section over a wider range of energies and the self shielding is reduced. (1.0)

Reference: Westinghouse PWR Core Physics
Fuel Temperature Coefficient pg. I-5.16
HBR SES 27, 28

1.08 a. MTC becomes more <u>negative</u> because the density change per degree -F is greater at higher temperatures (0.5) (0.5)

Reference: Westinghouse PWR Core Physics
Moderator Coefficient pg. I-5.8
HBR SES 26

b. MTC becomes <u>less negative</u> (decreases). The number of boron atoms (poison) in the core decreases more per °F change at higher boron concentration. (0.5) (0.5)

Reference: Westinghouse PWR Core Physics
Moderator Coefficient I-5.10
HBR SES 26

1.09 1. Remain the same (0.5)
2. Decrease (0.5)
3. Increase (0.5)
4. Increase (0.5)
5. Decrease - (0.5)

Reference: Fundamentals of Classical Thermodynamics, pg. 134
Gordon J. Van Wylen
Richard E. Sonntag 1976

1.10 a. Increase (0.5)
b. Decrease (0.5)
c. Decrease (0.5)
d. Increase (0.5)
e. Decrease (0.5)

Reference: General Physics Co., HTFF
Fluid Flow Applications for System and Components.

1.11 a. Decrease (0.5)
b. Increase (0.5)
c. Increase (0.5)
d. Increase (0.5)
e. Decrease (0.5)
f. Decrease (0.5)

Reference: GPC HTFF
Fluid Flow Applications

1.12 False (0.5)

Reference: HBR SES 43, 44, 45, 46

1.13 True (0.5)

Reference: HBR SES 23

1.14 False (0.5)

Reference: HBR SES 44

1.15 True (0.5)

Reference: General Physics HT FF 1.5, 1.6, 1.7

1.16 a. Less than (0.5)

b. Greater than (0.5)

Reference: HBR QB 1-17

2.0 Answers

2.01 (b) absorption of gamma energy. (1.0)

REF: RCS 9/49

2.02 (d) (1.0)

REF: RCS pgs. 25/49

2.03 1. Two out of three low pressurizer pressure. (0.5)
2. Two of three high differential steam line pressure. (0.5)
3. High steam line flow in 2/3 SG outlet lines coincident with low T_{ave} in 2/3 loops or low pressure in 2/3 SG outlet lines. (0.5)
4. Two of three high containment pressure (0.5)
5. Manual actuation (0.5)

REF: Safety Injection System pg. 3/28

2.04 (c) a (1.0)

REF: Safety Injection System pg. 4/28 p 8/28

2.05 False (2 min) (0.5)

REF: Rx Safeguards System pg. 11/17

2.06 (b) (1.0)

REF: PASS pg. 5/34

2.07 (d) (1.0)

REF: Electrical pg. 10/40

2.08 True (0.5)

REF: SG B/D System p 3/11

2.09 (d) (1.0)

REF: not provide by utility CVCS pg. 2/52
Westinghouse - Water Chemistry Hand Book

2.10 (a) (1.0)

REF: CVCS System pg. 5/52

2.11 (c) (1.0)

REF: SGB/D System pg. 5/11

2.12 (a) (1.0)

REF: CVCS System pg. 14 of 52

2.13 (d) (1.0)

REF" WGS pg. 5/19

2.14 (a) (1.0)

REF: Cond. System pg. 8/21

2.15 (a) Hotwell level low (0.25)

(b) Electrical Breaker Over load. (0.25)

(c) Under voltage on their respective 4160V bus (0.25)

(d) Stop switch on RTGB (0.25)

REF: Cond. System pg. 13/21

2.16 (a) CST (0.5)

(b) Service Water (0.5)

(c) Deep well (0.5)

REF: Feedwater pg 2/25

2.17 (1) Electrical overload (0.25)
(2) Undervoltage on associated bus (0.25)
(3) Loss of condensate pump (0.25)
(4) Low lube oil pressure (0.25)
(5) Low suction pressure (0.25)
(6) Safeguards actuation (SI signal) (0.25)
(7) Hi steam generator level (0.25)
(8) Minimum flow - blocked for 30 secs. after starting (0.25)

REF: Feedwater System pg. 12/25

2.18 (d) (1.0)

REF: Feedwater System pg. 16/25

2.19 (a) 102°F (+20 -10) (0.5)

(b) 441 °F (+20 -20) (0.5)

REF: Heater Vents & Drains System pg. 14/20

2.20 (a) 6-8 psig (1.0)

REF: Turb + Cont. System pg. 18/80

2.21 (a) 5% ~~(0.5)~~

(b) 70% Deleted ~~(0.5)~~

REF: Generator pg. 6/53

2.22 (a) Safety inje ion (0.5)

(b) Containment high radiation (0.5)

(c) manual containment isolation (0.5)

(d) automatic spray signal (0.5)

REF: Containment pg. 24/51

ANSWERS

3.01 1. Power Range (High Range) Nuclear Overpower (0.25)
2. Intermediate Range Nuclear Overpower (0.25)
3. OPΔT (0.25)
4. OTΔT (0.25)

Reference: Rod Cont Sys pg 11/27

3.02 True (0.5)
Reference: Reactor Control and Prot sys
P + L OP-001 pg 5/19

3.03 1. All step counters on the control board (0.25)
2. The master cycler reversible counter (0.25)
3. All slave cycler counters (0.25)
4. The bank overlap counter (0.25)
5. All internal memory and alarm circuits (0.25)
6. All pulse to analog converters in the Rod Position Indication System (0.25)

Reference: Rod Control Sys pg 10/27

3.04 True (0.5)
Reference: Rod Control Sys p 16/27

3.05 False (0.5)
Reference: IRPI p 5/8

3.06 False (200% per minute) (0.5)
Reference: IRPI p 7/8

3.07 False (both channels) (0.5)
Reference: NIS pg 3/37

3.08 (C) (1.0)
Reference Rx Prot Sys pg 4/20

3.09 False (0.5)
Reference Rx Prot Sys p 5/20

3.10 True (0.5)
Reference: Rx Prot Sys p 8/20

3.11 False (0.5)
Reference: Rx Prot Sys p 5/20 + 8/20

3.12 True (0.5)
Reference: Rx Prot Sys p 8/20

3.13 True (0.5)
Reference: Rx Prot Sys p 9/20

3.14 1. Measured low flow in the reactor coolant piping (0.25)
2. Reactor Coolant Pump Circuit Breaker Open (0.25)
3. Undervoltage on 4160 V bus 1, 2, 4 (0.25)
4. Under frequency on 4160 V bus 1, 2, 4 (0.25)

Reference: Rx Prot Sys 10/20

3.15 (a) (0.5)

Reference Incore Inst Sys Pg 3/8

3.16 True (0.5)
Reference: Rad Mon Sys p 14/38

3.17 True Deleted ~~(0.5)~~
Reference: Rad Mon Sys p 16/38

3.18 True (0.5)
Reference: MS Sys p 7/26

3.19 (d) Deleted ~~(1.0)~~
Reference: Cond Sys p 12/21

3.20 (d) (1.0)
Reference: Condensate Sys p 13/21

3.21 1. One steam generator low-low level (0.25)
2. Breakers of both feedwater pumps being open (0.25)
3. Blackout (0.25)
4. Safeguard conditions signal (0.25)

Reference: Feedwater Sys p. 22/25

3.22 (b) (0.5)
Reference: HBRQB 3-9

3.23 a. generator trip
b. low condenser vacuum
c. turbine bearing low oil pressure
d. thrust bearing failure
e. turbine overspeed mechanical trip
f. manual trip at front standard
g. high S/G level in any S/G
h. loss of three circulating waterpumps
i. loss of EH governor DC power (±15V and ±48V)
(Answer continued on next page)

j. loss of both main feed pumps
k. Rx trip
l. manual at RTGB

(0.1 each) X <u>10</u> = total = (1.0)

Reference: HBRQB 3-10

3.24 I (c) (1.0)
II (b) (1.0)

Reference: HBRAB 3-18

3.25 (c) Governor and intercept valves. (1.0)

Reference: HBRAB 3-25

3.26 (a) stop valves closed (0.5)
(b) ~~83~~ AST low pressure (auto stop oil) (0.5)
63

Reference: HBRQB 3-29

3.27 (c) (1.0)

Reference: HBRQB 3-13

3.28 (a) Closed (0.5)
(b) Open (0.5)
(c) Closed (0.5)
(d) Open (0.5)

Reference: HBRQB 2-6-18

3.29 From D. S. Diesel/output (1.0)

Reference: HBRQB 2-54

3.30 (a) rods move in (0.5)
(b) no affect (0.5)

Reference: HBRQB 2-57

4.0 Answers

4.01 (c) 21 days (1.0)

REF: AP-004 pg. 8/46

4.02 (d) six months (1.0)

REF: AP-009 pg. 7/12

4.03 True (0.5)

REF: Cond. of Ops pg. 2/30

4.04 (b) once every one half hour (1.0)

REF: GP-001 pg. 7/34

4.05 (a) 100°F/Hr (1.0)

REF: GP-002 pg. 15/39

4.06 (b) 15 cc/kg ~~(1.0)~~ Deleted.

REF: GP-003 pg. 14/24

4.07 (b) +1.0 DPM steady state and +1.0 DPM transient (1.0)

REF: GP-004 pg. 8/15

4.08 (d) 5%/min (1.0)

REF: GP-006 pg. 2/10

4.09 (c) 25°F (1.0)

REF: GP-006 pg. 3/10

4.10 (d) (1.0)

REF: GP-008 pg. 5/11

4.11 (d) (1.0)

REF: GP-0012

4.12 (a) Transfer Rod Control to Manual

(b) Manually position the Control Bank to restore equilibrium conditions.

(c) If the control bank will not move then adjust turbine load or RCS boron concentration to restore equilibrium conditons.

Need (2/3) (0.5) each = (1.0)

REF: AOP-001 pg. 7/31

4.13 (a) Attempt to Start affected pump. (0.25)

(b) Reduce load to match existing feedwater flow capability. (0.25)

(c) Verify Tavg and reactor power are being maintained automatically, or manually insert control rods to maintain reactor power and Tavg. (0.25)

(d) Verify Steam Generator levels are being maintained at normal operating level. (0.25)

REF: AOP-010 pg. 5/7

4.14 (a) Start standby circulating pump. (0.25)

(b) Verify standby vacuum pump is running. (0.25)

(c) Verify condenser vacuum breaker valves are closed. (0.25)

(d) If condenser vacuum is approaching the low vacuum trip point (20 ±2 in Hg), then reduce turbine generator load. (0.25)

REF: AOP-012 pg. 4/6

4.15 (a) Verify Tavg and reactor power are being maintained automatically, or manually insert control rods to maintain reactor power and Tavg. (0.33)

(b) Verify steam generator levels are being maintained at normal levels. (0.33)

(c) Verify proper operation of stam dumps and steam generator power operated relief valves. (0.33)

REF: AOP-015 pg. 5/6

4.16 (a) Steam generator blowdown radiation monitor R-19

(b) Condenser vacuum exhaust radiation monitor R-15

(c) Lab analysis of steam generator gross activity

(d) Excessive RCS makeup system operation

3/4 (0.5) each = (1.5)

REF: AOP-016

4.17 (a) Verify component cooling valves to and from the reactor coolant pumps are open and at least one CCW pump is running. (0.5)

(b) If any of the following limits are reached, then stop the affected reactor coolant pumps: (0.5)

1. Upper bearing temperature exceeds 200°F.
2. lower bearing temperature exceeds 225°F.
3. Any RCP has operated for two minutes with no component cooling water flow to either motor oil cooler.

REF: AOP-018 pg. 6/22

4.18 (a) Try manual trip (0.4)

(b) Manually insert control rods (0.4)

(c) locally trip rod drive motor generator sets (0.4)

(d) Initiate emergency boration of RCS (0.4)

(e) Check all dilution paths isolated (0.4)

REF: FRP-S.1

4.19 (1) Manually trip turbine (0.5)

(2) Run back turbine (0.5)

REF: EPP path 1

4.20 (1) RCS Subcooling - greater than 25°F (0.5)

(2) Total feed flow to intact S/G (0.5)

a. greater than 300 GPM or $0.2X10^6$ ~~PPH~~

or

b. level in at least one intact S/G greater than 10%

(3) RCS pressure a. greater than 1520 Psig (0.5)

b. stable or increasing

(4) RZR level greater than 10% (0.5)

REF: EPP Foldout

4.21 (b) (1.0)

REF: EPP Foldouts

4.22 (1) SI pumps - at least one running (0.5)

(2) RCS subcooling - less than 25°F [35°F] (0.5)

REF: EPP - Foldouts

4.23 CAF ~~(1.0)~~

Deleted.

Reference —

Enclosure 3
(2 of 2)

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

FACILITY: ROBINSON

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 84/12/18

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
25.00	25.00	________	________	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.00	25.00	________	________	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.00	25.00	________	________	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.00	________	________	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
100.00	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 (1.90)

Multiple Choice.

a. With the plant operating at 85% power and all systems in a normal/automatic configuration, the operator borates 100 PCM. Shutdown Margin will:

1. Increase
2. Increases until rods move
3. Decrease
4. Decreases until rods move
5. Remain unchanged, whether or not rods move (0.4)

b. State the value of the Shutdown Margin required by Technical Specifications for:

1. Cold Shutdown.
2. Refueling. (0.5)

c. HOW and WHY does the Shutdown Margin requirement change as the core ages? (1.0)

QUESTION 5.02 (3.00)

The calculated Shutdown Margin is 10% delta k/k assuming the most reactive control rod worth is 1000 PCM. The Source Range count rate is 50 cps. Show all work and state any assumptions made for the following;

a. Determine the final count rate after the shutdown banks are fully withdrawn. (Assume the shutdown bank rod worth is 5600 PCM). (1.5)

b. Determine the final count rate after 100 ppm dilution of the RCS FOLLOWING a, above. (1.5)

QUESTION 5.03 (2.25)

a. Compare Xenon-135 and Samarium-149 fission product poisons by EXPLAINING the differences for the following: (actual values are not necessary).

1. Time to reach equilibrium conditions after startup. (0.5)
2. Magnitude of negative reactivity at full power equilibrium conditions. (0.75)

b. Explain why the following statement is correct. "Equilibrium Xenon concentration at 50% power is NOT approximately half its concentration at 100% power". (1.0)

QUESTION 5.04 (3.00)

a. How does the target delta-flux change (more positive, more negative or no change) with increasing core burnup? (0.5)

b. The straight target delta-flux line is not the natural behavior of the unrodded core as a function of power with equilibrium xenon. How then, is the delta-flux maintained within the target band at 50 % power and xenon equilibrium? (0.5)

c. What is the long term affect on delta-flux of a load reduction made by significant insertion of control rods from full power with no operator action. At BOL? At EOL? Assume auto rod control. (1.0)

d. Delta-flux is about to go outside the target band in the negative direction with the plant at full power. Provide TWO operator control actions that are available to the operator to maintain delta-flux within the target band? (1.0)

QUESTION 5.05 (2.00)

For each of the following conditions, which of the two choices would the INDIVIDUAL (differential or integral as indicated) rod worth be greater?

	Rod Worth	Condition	Choice 1	Choice 2	
a.	Integral	Tavg	150-F	500-F	(0.5)
b.	Integral	Core life	BOL	EOL.	(0.5)
c.	Differential	Rod position	40% inserted	70% inserted	(0.5)
d.	Differential	Rod in Bank C which is next to an assembly with:	an inserted rod	the rod withdrawn	(0.5)

QUESTION 5.06 (2.25)

Differential boron worth is defined as the change in reactivity due to a change in boron concentration. Explain both HOW AND WHY the following factors affect differential boron worth (more negative, less negative or no change). Consider each factor seperately.

a. Boron concentration increase (0.75)

b. Moderator temperature decrease (0.75)

c. Core burnup from MOL to EOL (0.75)

QUESTION 5.07 (2.60)

Would fuel center line temperature INCREASE, DECREASE, or REMAIN THE SAME in each of the following situations? BRIEFLY EXPLAIN your answer.

a. Power decreases with constant Tave.

b. Tave increases with constant power.

c. Core age increases (MOL to EOL) with constant power.

d. Pressurizer pressure increases with constant power. (2.6)

QUESTION 5.08 (2.00)

a. At what axial location in a PWR core is the critical heat flux at the MINIMUM? (0.5)

b. How does the minimum critical heat flux change (increase, decrease, or no change) as the following parameters are INCREASED? Consider each separately.

1. Tave
2. RCS pressure
3. RCS flow
4. Reactor power (1.0)

c. At what core height (Bottom, 1/4, 1/2, 3/4, Top) does the "Fuel Surface Temperature" reach its maximum value? (0.5)

QUESTION 5.09 (2.50)

a. Brittle fracture of any carbon steel pressure vessel can occur at stresses well below yield stress if TWO other conditions are present. What are these TWO conditions? (1.0)

b. How do heatup/cooldown rate limits on the reactor coolant system reduce the probability of brittle fracture? (0.5)

c. Why does the concern about brittle fracture of the reactor pressure vessel increase as the plant ages? Include in your answer the specific material PROPERTY that is affected. (1.0)

QUESTION 5.10 (1.50)

True or False

a. The differential temperature necessary to transfer heat is inversely proportional to heat flux. (0.25)

b. Pump runout is the term used to describe a centrifugal pump when it is operating with its discharge valve shut. (0.25)

c. The latent heat of vaporization is another term for the latent heat of condensation. (0.25)

d. One of the pump laws for centrifugal pumps states that power required by the pump motor is directly proportional to the square of the pump speed. (0.25)

e. The faster a centrifugal pump rotates, the greater the NPSH required to prevent cavitation. (0.25)

f. When comparing a parallel-flow heat exchanger to a counter-flow heat exchanger, the temperature difference between the two fluids along the LENGTH of the heat exchanger tubes is MORE uniform for the parallel-flow heat exchanger. (0.25)

QUESTION 5.11 (2.00)

a. If during a cooldown on natural circulation, the RCS pressure was 1200 psig, what would be the maximum steam generator pressure to assure adequate subcooling? (1.0)

b. During natural circulation cooldown, a steam bubble may form in the reactor vessel head area. What is the primary indication of this bubble formation? (0.5)

c. If Natural Circulation flow is lost, how will the following parameters respond (INCREASE, DECREASE, or NO CHANGE)?

1. Steam generator pressure.

2. Reactor coolant loop Tc. (0.5)

QUESTION 6.01 (2.00)

a. How are the RTD bypass manifold penetrations on the RCS cold legs different from the RTD bypass manifold penetrations on the RCS hot legs? (0.5)

b. Where in the bypass manifold system is the flow orifice located for the "Loop RTD Bypass Flow Low" alarm circuit? (0.5)

c. Why is a low flow condition in the RTD bypass manifold system of concern? (0.5)

d. TRUE or FALSE

Reactor Protection System circuits are provided inputs from the wide range RCS loop temperature instrumentation? (0.5)

QUESTION 6.02 (2.50)

a. If the RCS pressure is 375 psia, RCS temperature is 340 F, and warming of the Residual Heat Removal System (RHR) has NOT started, HOW is the RCS protected against an overpressurization? (1.0)

b. Would resin damage occur in the CVCS ion exchangers if the RHR system were lined up to the CVCS and the RCS temperature was above ~~140~~ 130 F? Assume all automatic protective features function correctly. EXPLAIN your answer. (1.5)

QUESTION 6.03 (3.00)

The operation of any trip device and the resultant loss of Auto-Stop Oil pressure will open the Interface Emergency Trip Valve.

a. What valves in the steam system will reposition when the Emergency Trip Valve opens? (1.2)

b. Explain the sequence of events that occurs to cause the above valves to close. (1.2)

c. Why must the valves in "a" above function correctly on a Main Generator fault with Reactor Trip? (0.6)

QUESTION 6.04 (2.00)

TRUE or FALSE

a. An urgent failure in a power cabinet sends a signal to the logic cabinet. (All automatic rod motion is inhibited.) (0.5)

b. At the OTdT setpoint or the OPdT setpoint ALL automatic rod motion is inhibited. (0.5)

c. If turbine power drops below 15%, automatic rod withdrawal is blocked. (0.5)

d. At 103% reactor power, automatic rod insertion is inhibited. (0.5)

QUESTION 6.05 (2.40)

During operation at 80% power, a "B" S/G safety valve fails full open. (Control rods are at 220 steps)

a. Will the S/G level control system detect the added steam flow? Explain. (0.6)

b. Will any of this added load be shared by the other S/G's? Explain. (1.2)

c. How can nuclear instrumentation and RCS temperature indication be used by the operator to determine that this problem has occurred? (0.6)

QUESTION 6.06 (2.70)

a. What are the TWO conditions that will cause automatic startup of the Emergency Diesel Generators? (0.6)

b. What are FIVE of the six conditions that will cause automatic trip of the Emergency Diesel ENGINES when not running in the Emergency Mode? (1.5)

c. What are the THREE conditions that will cause automatic trip of the Emergency Diesel Generator OUTPUT BREAKERS when running in the Emergency Mode? (0.6)

QUESTION 6.07 (2.40)

The plant is operating at 80% power when a Thot RTD fails high. EXPLAIN how this failure will affect the following. Consider each item independently. Assume no operator action and all control systems are in automatic.

a. Rod insertion limit setpoint (0.6)

b. Charging flow (initially) (0.6)

c. Control rod bank position (0.6)

d. Steam dump control system (0.6)

QUESTION 6.08 (3.00)

List the sequence of events (control and protection) that leads to a reactor trip when the controlling Pressurizer LEVEL channel fails HIGH.

ASSUME- No operator action and initial plant conditions are in a normal/automatic configuration at 50% load. (Setpoints of control and protective events are not required.) (3.0)

QUESTION 6.09 (3.00)

Indicate whether the following statements are TRUE for OT Delta-T, OP Delta-T, OR BOTH, (OT Delta-T and OP Delta-T) protection instruments.

1. Protects the core from DNB.

2. Protects the core from overpower (Kw/ft).

3. Backup for the high neutron flux trip.

4. Circuitry dynamically compensates for piping delays to the loop temperature detectors.

5. Requires RCS pressure within the high and low reactor trip setpoints in order to be valid. (3.0)

QUESTION 6.10 (2.00)

a. What signals will AUTOMATICALLY start the Motor-Driven Auxiliary Feedwater Pumps? (No setpoints required) (1.0)

b. What signals will AUTOMATICALLY start the Turbine-driven Auxiliary Feedwater Pump? (No setpoints required) (0.5)

c. What is the basis for the 35,000 gallon minimum (Technical Specification requirement) in the Condensate Storage Tank? (0.5)

QUESTION 7.01 (2.80)

A plant startup is in progress in accordance with GP-003 (NORMAL PLANT STARTUP FROM HOT SHUTDOWN TO CRITICAL).

a. It is necessary to dilute 200 PPM boron 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 of boron the same instruments read 62 and 75 CPS. Should the operator continue with the planned dilution of another 100 PPM? Explain. (0.6)

b. If criticality is achieved greater than the 500 PCM below the ECP, what TWO operator actions must be taken? (1.2)

c. Source Range Channels 31 & 32 are reading 10 4 CPS. Intermediate Range Channel 35 is reading 10 -10 AMPS and Channel 36 is reading 10 -11 AMPS. Which I/R channel is reading correctly? Explain. (1.0)

QUESTION 7.02 (2.00)

Robinson Procedure, GP-006 states the following precaution:

"All shutdown banks must be at the fully withdrawn position whenever positive reactivity is being inserted by boron removal and Xenon decay or reactor coolant temperature changes".

State the TWO exceptions to this precaution. (2.0)

QUESTION 7.03 - (3.00)

a. List FOUR conditions that require Emergency Boration as stated in AOP-002. (2.0)

b. State the FOUR flow paths for delivering boric acid to the charging pump suction. (1.0)

QUESTION 7.04 (2.00)

During power operation, a loss of an instrument bus occurs;

a. In accordance with AOP-024 (Loss of Instrument Bus) what means should be used to determine which instrument bus was lost? (0.5)

b. As a result of a Loss of Instrument Bus, under what conditions will the following occur?

1. Turbine Runback.

2. Reactor Trip. (1.5)

QUESTION 7.05 (2.00)

a. While performing "Natural Circulation Cooldown" (EPP-5), you are procedurally directed to depressurize the RCS.

1. What method will be used to depressurize the RCS if Letdown is in service? (0.5)

2. What method will be used if Letdown is NOT in service? (0.5)

b. Why is the depressurization more restrictive if CRDM fans are not running? (1.0)

QUESTION 7.06 (2.40)

TRUE OR FALSE

a. If while monitoring CSFST's a yellow condition is noticed, the operator shall combat the condition via appropriate FRP while continuing to progress through the flow path.

b. If performance of an EPP step requires undoing an action that mitigated a RED FRP, then that step is left up to the discretion of the operator.

c. Upon discovery of a S/G tube rupture, PATH-2 may be entered directly without entry to PATH-1.

d. While combatting a RED FRP, a loss of E-1 and E-2 occurs, the operator should discontinue the FRP and move to EPP-1, Loss of all AC. (2.4)

QUESTION 7.07 (1.90)

During your Licensing Examination you will be required to escort your examiner through the plant. What is the administrative whole body exposure limit of the examiner (a visitor) assigned to you in accordance with DP-003? ANSWER THE QUESTION IF THE EXAMINER PROVIDES HIS DOSE RECORDS AND ALSO IF THE EXAMINER DOES NOT PROVIDE HIS DOSE RECORDS! (1.9)

QUESTION 7.08 (1.40)

If a spent fuel assembly was damaged during refueling operations in the Refueling Canal:

a. What automatic actions occur if Radiation Monitor R-11 or R-12 reaches the high alarm setpoint? (0.6)

b. In addition to verifying that the automatic actions occur, what are the required immediate actions for this event? (Include actions performed by the control room operator AND personnel in the containment.) (0.8)

QUESTION 7.09 (2.00)

a. According to EPP-9 when is the "Transfer to Cold Leg Recirculation" required? (Include any parameter values) (0.5)

b. During the performance of EPP-9, what is the maximum time limit between termination of RHR flow during the injection phase and reinitiation of RHR flow in the recirculation phase? (0.5)

c. At what level in the RWST are the SI pumps, RHR pumps, and CV Spray pumps stopped? (0.5)

d. When is the "Transfer to Hot Leg Recirculation" (EPP-10) started? (0.5)

QUESTION 7.10 (2.50)

a. During a "Loss of Component Cooling Water" (AOP-014), state the THREE conditions that would require stopping ALL THREE Reactor Coolant Pumps (RCP). (Include applicable parameter values) (1.5)

b. What is the maximum #1 seal leafoff temperature allowed before a RCP must be stopped? (0.5)

c. At what minimum flow rate of Component Cooling Water to the Thermal Barrier Cooler must the RCP be stopped? (0.5)

QUESTION 7.11 (3.00)

a. Assume the plant is operating at full power and the Axial Flux Difference (AFD) has been outside the target band for the last 5 minutes. What are the TWO actions specified which you may choose between to meet the Technical Specification requirements? (1.0)

b. Assume that it is 0310 on 5/19/84 and the plant is presently at 45% power. Considering the AFD penalty history below, at what date and time may power be increased above 50%? EXPLAIN. (Show all work.) Assume no deviation outside the band after 0310 on 5/19/84.

DATE	TIME WENT OUT OF BAND	TIME BACK IN BAND	POWER
5/18/84	0310	0318	85%
5/18/84	1557	1637	65%
5/19/84	0148	0310	45%

(2.0)

QUESTION 8.01 (2.60)

a. During an "Immediatly Reportable" event, what are the TWO time period constraints within which such an event must be reported? (0.8)

b. What is the time constraint within which a "Prompt Reportable" occurrence must be reported to the NRC via telephone? (0.3)

c. What are TWO documents the Shift Foreman could use to determine if an event is immediatly reportable? (1.0)

d. Who has the responsibility for contacting outside emergency response agencies during the use of the Emergency Plan? (0.5)

QUESTION 8.02 (2.00)

According to Technical Specifications what action should be taken for the following abnormal situations?

a. Alarms that would provide indication of out of specification axial flux difference are out of service. (1.0)

b. Quadrant Power Tilt Ratio is 1.092 with no indication of rod misalignment. (1.0)

QUESTION 8.03 (2.50)

a. Who has the primary responsibility for the following Radiation Work Permit (RWP) tasks? Answer for both Routine And Non-Routine RWP's.

1. Writing.

2. Reviewing.

3. Approving. (1.5)

b. Under what condition(s) may the requirement for submitting a Pre-Submittal RWP form be delayed? (1.0)

QUESTION 8.04 (1.80)

The following pertain to Temporary Procedure changes.

a. Who must approve a Temporary Change prior to its becoming effective? (0.7)

b. What is the period of time that a Temporary Change is applicable before it must be deleted or reviewed? (0.5)

c. How may a Temporary Change be recognized by person performing a procedure? (0.6)

QUESTION 8.05 (3.50)

a. Name the individual (by title) who is responsible for the following refueling operations:

1. Advising the Shift Foreman that conditions are satisfactory for disengaging the gripper from a fuel assembly. (0.5)

2. Providing a temporary relief for the containment upender operator. (0.5)

3. Ensuring all prerequisites are met prior to commencing a specific fuel handling evolution. (0.5)

4. Maintaining 1/M plot. (0.5)

b. Name, by title, the various members of the refueling operations organization that must be SRO or RO licensed. Indicate which license (RO or SRO) the position requires. (1.5)

QUESTION 8.06 (2.80)

a. List the required shift complement when the RCS is >200 F (0.8)

b. Where is the shift relief conducted for the Unit 2 Shift Foreman? (0.5)

c. When performing a normal shift relief, who must sign the Minimum Equipment List? (1.0)

d. When may the requirement for reviewing and signing the MEL be waived ? (0.5)

QUESTION 8.07 (2.50)

Answer the following in accordance with OMM-015.

a. Who are the reviewing authorities for all Operations Surveillance Testing? (1.0)

b. TRUE or False

The reviewer may accept a test if one or more of the Acceptance Criteria is not met provided certain conditions exist. (0.5)

c. If the test results of equipment tested under a Technical Specification Test with NO LCO time requirements does not meet the Acceptance Criteria, can the reviewer accept the test? Briefly Explain Your Answer. (1.0)

QUESTION 8.08 (2.80)

a. Briefly explain the check made on RTGB redundant indicators each shift. (0.8)

b. How is the check in "a" above documented? (0.5)

c. What is the criteria for a satisfactory check? (1.0)

d. How can an indicator that has NOT passed a satisfactory check be visually identified in the control room? (0.5)

QUESTION 8.09 (2.50)

a. Who is responsible for performing the required weekly setpoint check on all Process and Area Radiation Monitors? (0.5)

b. Under what condition may it become necessary or allowable to change a monitor setpoint? (1.0)

c. Explain how a change in setpoint would be documented. (1.0)

QUESTION 8.10 (2.00)

Provide the two reasons why there is a minimum reactor cavity level requirement during refueling operations. (2.0)

EQUATION SHEET

$f = ma$

$v = s/t$

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

$w = mg$

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

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

$W = v\ \Delta P$

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

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

$\Delta E = 931\ \Delta m$

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

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

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

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

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

$Pwr = W_f \Delta h$

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

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

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

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

$SUR = 26.06/T$

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

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

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

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

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

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

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

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

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

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

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

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

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

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

$I_1 d_1 = I_2 d_2$

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

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

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

$\Sigma = \sigma N$

$R/hr = 6\ CE/d^2$ (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
1in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)
1 BTU = 778 ft-lbf

ANSWER 5.01 (1.90)

a. Increase. (0.4)

b. 1. 1% dk/k
2. 10% dk/k (0.5)

c. SDM increases (more limiting) [0.5] as core ages because the hypothetical steam break accident is more severe at EOL. [0.5] (1.0)

REFERENCE
HBR RXTH Handout, Session 50, p.2; Tech. Specs. p. 3.10-10

ANSWER 5.02 (3.00)

a. CR(f) = CR(i) [(1 - Ki)/ (1 - Kf)]

Reactivity in core (i)= -10,000 pcm + (-1000 pcm) = -11,000 pcm
Reactivity in core (f)= -11,000 pcm + 5600 pcm = -5400 pcm

SDM = (1 - Keff)/Keff OR p = (Keff - 1)/Keff ; Keff =1/(1 - p)

Ki = 1/(1 + 0.11) Ki = 0.9009

Kf = 1/(1 + 0.054) Kf = 0.9488

$$CR(f) = 50 \text{ cps } \frac{(1-0.9009)}{(1-0.9488)} = 96.7 \text{ cps}$$ (1.5)

b. Assume Boron worth = 10 pcm/ppm

K(i) = 0.9488 CR(i) = 96.7 cps

Reactivity in core (i) = -5400 pcm + (100 ppm x 10 pcm/ppm)

= -5400 pcm + 1000 pcm = -4400 pcm

K(f) = 1/(1 + 0.44) = 0.9579

$$CR(f) = 96.7 \times \frac{(1-0.9488)}{(1-0.9579)} = 96.7 \times 1.216 = 117.6 \text{ cps}$$ (1.5)

REFERENCE
HBR RXTH Handout, Session 42, p. 3-4

ANSWER 5.03 (2.25)

a. 1. Samarium takes much longer to reach equilibrium conditions [0.25] because it is formed by decay of a precursor with a longer half life than for Xenon [0.25]. (0.5)

2. Xenon has the most worth at equilibrium [0.25] because it is more abundant [0.25] and because it has a higher neutron absorption cross-section [0.25]. (0.75)

b. Equilibrium Xenon is flux dependent and non-linear ~~[0.5]~~ since burnout is relatively less significant at 50% power than at 100% power (production is flux dependent but removal terms are not both flux dependent) ~~[0.5]~~. 1.0 (1.0)

REFERENCE
HBR RXTH Handout, Session 37, p. 9; Session 38, p. 4,8

ANSWER 5.04 (3.00)

a. More ~~positive~~ negative (0.5)

b. Control Bank D rods must be inserted (as required, to drive the delta flux negative to within the band) (0.5)

c. BOL - Self dampening delta-flux oscillations (xenon oscillations)
EOL - Diverging delta-flux oscillations [0.5 each] (1.0)

d. Withdraw control rods (if not fully withdrawn) [0.5] or, reduce reactor power [0.5] (1.0)

REFERENCE
~~FMP-009~~; FMP-006, 2.3, 2.4; Tech. Spec. 3.10.2.7
Cycle 9 Data Fig. 6.21 and Table 6.22

ANSWER 5.05 (2.00)

a. 2

b. 2 will accept 1 for Cycle 10

c 1

d. 2 [0.5 ea.] (2.0)

REFERENCE
HBR RXTH Handout, Session 36, p. 2-3

ANSWER 5.06 (2.25)

a. Delta boron worth becomes less negative [0.25] due to increased competition for neutrons by more boron atoms OR due to fast to thermal flux ratio increase [0.5]. (0.75)

b. Delta boron worth becomes more negative [0.25] because density of moderator increases, increasing the possibility of a neutron being absorbed by boron OR due to thermal to fast flux ratio increase [0.5]. (0.75)

c. Delta boron worth becomes more negative [0.25] due to the buildup of fission product absorbers [0.5]. Will also accept burnout of Gd ∴ less competition with boron. (0.75)

REFERENCE
HBR RXTH Handout, Session 33, pp. 3

ANSWER 5.07 (2.60)

a. Decrease[0.35], smaller delta T required to transfer more energy from RCS. [0.35]

b. Increase[0.35], center line temperature responds to RCS temperature in order to maintain constant delta T across cladding. [0.35]

c. Decrease[0.35], fuel swelling and clad creep reduce clad gap which reduces delta T across gap and lowers center line temp. [0.35]

d. No change[0.25], pressure has little effect on heat transfer in subcooled fluids. [0.25] Will also accept increase due to reduced nucleate boiling and resultant ↓ in heat transfer rate. (2.6)

REFERENCE
GP HTFF Chap. 4, p. 235, 217-219

ANSWER 5.08 (2.00)

a. Top of the core (0.5)

b. 1. Decrease
   2. Increase
   3. Increase
   4. Decrease (1.0)

c. 3/4 (0.5)

REFERENCE
GP HTFF, Chap. 4, p. 226, 224, 229

ANSWER 5.09 (2.50)

a. 1) Presence of a flaw (or crack of sufficient size). [0.5]
   2) Low temperature [0.5]. (1.0)

b. Reduces the thermal stress. (Reduced DT across the RV wall reduces total/thermal/tensile stress.) (0.5)

c. Neutron exposure (integrated) [0.5] makes the material more brittle (raises NDT) (Reduces ductility.) [0.5] (1.0)

REFERENCE
WNTC Thermodynamics, Volume II, Chapter 13, p. 55
Tech. Spec. p. 3.1-6, -7

ANSWER 5.10 - (1.50)

a. False
b. False
c. True
d. False
e. True
f. False [0.25 each] (1.5)

REFERENCE
GP HTFF, pp. 31, 114, 164, 320, 322

ANSWER 5.11 (2.00)

a. Tsat for 1200 psig is 567 F (from steam tables).
567 - 50 = 517 F (subcooling of 50 F).
Psat for 517 F is about 800 psig. (1.0)

b. Erratic pressurizer level indication. (0.5)

c. 1. Decrease.

2. No change. (0.5)

REFERENCE
Steam Tables; EPP-6 p. 8; GP HTFF p. 357

ANSWER 6.01 (2.00)

a. Three scoop type penetrations (spaced 120 degrees apart) in each hot leg,[0.25] and a single penetration in each cold leg.[0.25] (0.5)

b. In the combined return line from both manifolds. (0.5)

c. Alerts the operator to a possible unreliable narrow range temperature indication. (0.5)

d. False. (0.5)

REFERENCE
HBR SD-001 p. 28; DWG. 5379-1971 sh. 1

ANSWER 6.02 (2.50)

a. RCS is protected by the Pressurizer PORV's ~~primary safety valves~~ [0.5] in the low pressure mode. [0.5] (1.0)

b. No. [0.75] RHR connection to the CVCS is upstream of the letdown heat exchanger, (which cools the RCS water to below 140 F) [0.75]. Additional correct responses - TCV-143 will divert water around the demineralizers. (1.5)

REFERENCE
GP-007 p. 12; CVCS DWG. 5379-685 sh.1; SD-021 p. 6, 26

ANSWER 6.03 (3.00)

a. stop valves
governor valves
reheat valves
intercept valves
(extraction steam non-return checks) [0.3 ea.] (1.2)

b. The Emergency Trip fluid is depressurized when the trip valve opens [0.3] allowing the EH fluid dump valves on each turbine valve to open [0.3] the EH pressure holding the turbine valves open is relieved [0.3] spring action closes the turbine valves [0.3] (1.2)

c. To prevent turbine damage from overspeed. Will also accept - prevent excessive RCS cooldown. (0.6)

REFERENCE
SD-033 p. 1, 2, 16

ANSWER 6.04 (2.00)

a. TRUE

b. FALSE

c. TRUE

d. FALSE [0.5 each] (2.0)

REFERENCE
SD-007 pp. 11,16,19

ANSWER 6.05 (2.40)

a. Yes [0.3] The steam flow nozzles are upstream of the safety valve tap offs [0.3] (0.6)

(and flow from)

b. Yes [0.3] The safety lifting lowers pressure in the "B" S/G [0.3] turbine load is set [0.3] thus, the other S/G's steaming rate increases [0.3] (1.2)

c. Nuclear power increases above turbine power [0.3] Tave (lowers) deviates from Tref [0.3] (0.6)

REFERENCE
SD-025 p. 1,2

ANSWER 6.06 (2.70)

a. 1. SI Signal
   2. Low voltage on the respective emergency bus (blackout). (0.6)

b. 1. Overspeed
   2. High jacket water temperature
   3. Low lube oil pressure
   4. Low jacket water pressure
   5. 10 sec. Overcrank
   6. High crank case pressure [5 required, 0.3 each] (1.5)

c. 1. Reverse power
   2. Over current
   3. Over voltage [0.2 ea.] (0.6)

REFERENCE
SD-005 p. 12,13,15; SD-006 p. 7,13

ANSWER 6.07 (2.40)

a. Raises the limit, because high dT indicates a higher power. (0.6)

b. Increases to raise pressurizer level to 100% program, because of the higher Tave (0.6)

c. Rods move in, because of the Auct. Tave/Tref mismatch (0.6)

d. No effect, the demand signal is present (Tave/Tref) but there is no arming signal. (0.6)

REFERENCE
SD-025 p. 15; SD-007 p. 20; SD-021 p.23; SD-001 p. 29,34

ANSWER 6.08 (3.00)

1. ~~Backup heaters on [0.4]~~
2. Charging flow to minimum [0.6]
3. Letdown isolation (LCV-460 and orifice valves) [0.7]
Letdown will [illegible]
→ 4. Backup and control heaters cutout [0.4]
5. Reactor trip on ~~high~~ Low PZR ~~level~~ pressure [0.9] (3.0)

REFERENCE
SD-021 p. 13,23; SD-001 p. 34; Logic Diagrams sh. 17.

ANSWER 6.09 (3.00)

A. 1. OT Delta-T
2. OP Delta-T
3. OP Delta-T
4. Both
5. OT Delta-T [0.6 each] (3.0)

REFERENCE
SD-011 p. 5-7

ANSWER 6.10 (2.00)

a. - Low-Low S/G level (2/3) in ~~two~~ one S/G.
   - SI signal.
   - Trip of both MFP's.
   - Blackout. [0.25 ea.] (1.0)

b. - Blackout (also loss of power to Bus 1 + 4)
   - Low-Low S/G level (2/3) in two S/Gs [0.25 ea.] (0.5)

c. To maintain the plant in Hot Standby for 2 hrs. (0.5)

REFERENCE
HBR Tech Spec. p. 3.4-3, -5

ANSWERS -- ROBINSON -84/12/18-JAGGAR, F.

ANSWER 7.01 (2.80)

a. No, he must immediately suspend the dilution. Will also accept; If the counts are doubled the Shutdown Margin is halfed, (shutdown reactivity was decreased by approximately 50%) thus by adding the same amount of reactivity again the reactor would be critical. (0.6)

b. Reinsert control banks (shutdown)
Recalculate the ECC [0.6 ea.] (1.2)

c. Channel 35 [0.5], compare S/R to I/R, 10 4 CPS = 10 -10 AMPS [0.5] (1.0)

REFERENCE
GP-002 p. 16; GP-003 pp. 15, 18; SD-010 FIG.1

ANSWER 7.02 (2.00)

a. With Plant Managers approval,[0.25] the RCS temperature and boron concentration are being maintained at the hot shutdown, Xenon free condition.[0.75] (1.0)

b. With Plant Managers approval,[0.25] the RCS has been borated to the cold shutdown concentraton AND the plant is being cooled down or heated up.[0.75] (1.0)

REFERENCE
HBR GP-006 p. 2; QB 4-7-38

ANSWER 7.03 (3.00)

a. 1. Failure of 2 or more RPI's to indicate rods inserted after a normal shutdown or trip.
2. Control rod height below the insertion limit.
3. Uncontrolled RCS cooldown following a trip.
4. Unexplained or uncontrolled reactivity increase [0.5 ea.] (2.0)

b. 1. Normal flow path through blender via FCV-113A & B.
2. Through MOV-350 (Emergency Boration).
3. Through LCV-115B or CVC-358.
4. Through FCV-113A & 114B. [0.25 ea] (1.0)

ANSWERS -- ROBINSON -84/12/18-JAGGAR, F.

REFERENCE
HBR AOP-002 p. 3-5

ANSWER 7.04 (2.00)

a. Observe the bistable status lights AND Power Range indication. (0.5)

b. 1. A turbine runback will occur if a Rod Drop signal is initiated [0.5]. (NIS -5%/5sec)

2. A reactor trip will occur if Instrument Bus 1 or 2 is lost [0.5] AND the Source or Intermediate trips are NOT bypassed. [0.5] (1.5)

REFERENCE
AOP-024 p. 3, 4

ANSWER 7.05 (2.00)

a. 1. Auxiliary pzr spray (0.5)
2. Pzr PORV (0.5)

b. The potential for void formation in the reactor vessel head increases if the vessel head cooling provided by the CRDM fans is not available. (1.0)

REFERENCE
EPP-5 pp. 7, 10

ANSWER 7.06 (2.40)

a. False

b. True

c. False

d. True [0.6 ea.] (2.4)

REFERENCE
HBR Question Bank 4-7-75

ANSWER 7.07 (1.90)

WITHOUT DOSE RECORDS;
1. 375 mrem/cal. qtr. [0.9]

WITH DOSE RECORDS;
1. 1,250 mrem/cal. qtr.[1.0] OR

2. 3,000 mrem/cal. qtr.[0.8] if the requirements of 10 CFR 20 [0.1] are met and authorization of the examiner's employer is obtained. [0.1] (1.9)

REFERENCE
OP-003 p. 7

ANSWER 7.08 (1.40)

a. 1. Purge valves close.
   2. Vacuum and pressure relief valves close.
   3. HVE-1A & B stop. [0.2 ea.] (0.6)

b. Control Room Operator
   1. Verify alarm.
   2. Notify ECRC to take samples.
   3. Announce Alarm.
   4. Evacuate containment (sound alarm).
   5. Implement actions per Emergency Plan.

   Personnel in Containment
   1. Go to personnel monitoring station.
   2. Notify control room
   3. Ensure all doors are closed (or temporary covers erected over openings).
   [0.1 ea.] (0.8)

REFERENCE
FHP-035 p. 5; AOP-005 p. 7

ANSWER 7.09 (2.00)

a. When the RWST level decreases to 27%. (0.5)

b. 10 minutes. (0.5)

c. 9% (0.5)

d. Approx. 18 hours after a LOCA has occurred. (0.5)

REFERENCE
HBR EPP-9 pp. 3-5; EPP-10 p. 3

ANSWER 7.10 (2.50)

a. 1. Two minutes of RCP operation following the loss of cooling water flow.

2. Upper bearing temperature reaches 200 F.

3. Lower bearing temperature reaches 225 F. [0.5 ea.] (1.5)

b. 170 F. (0.5)

c. 25 gpm. (0.5)

REFERENCE
AOP-014 p. 5; AOP-18 p. 14

ANSWER 7.11 (3.00)

a. 1. Restore the indicated AFD to within the target band immediately, [0.5] or
2. Reduce the thermal power to <90% of rated thermal power. [0.5] (1.0)

b. Accumulated penalty over the past 24 hours is 89 minutes. [1.0]
The penalty will be reduced to 60 minutes at 1618 minutes on 5/19/84 and then power may be increased. [1.0] (2.0)

REFERENCE
HBR Tech Spec 3.10 p. 3.10-5, -6

ANSWER 8.01 (2.60)

a. 1,4 hours (0.8)

b. 24 hours (0.3)

c. 10.CFR 50 (72), HBR Tech. Specs., AP-30 [any two 0.5 ea.] (1.0)

d. Emergency Coordinator. (Will accept Shift Foreman) (0.5)

REFERENCE
WJE 200
10.CFR.50 (72), HBR.T.S. 6.6, PEP 2.6.21

ANSWER 8.02 (2.00)

a. Log AFD hourly (to ensure compliance with specification). (1.0)

b. Proceed to Hot Shutdown immediatly. (1.0)

REFERENCE
WJE 201
T.S. 3.10.2.10, 3.10.3.3, 3.10.2.9.b

ANSWER 8.03 (2.50)

a. Routine: Writing - Technician
Review - Rad. Con. Supervisor
Approval - Rad. Con. Supervisor

Non-Routine: Writing - Technician
Review - Rad. Con. Foreman
Approval- Rad. Con. Foreman [0.25 each] (1.5)

b. 1. For RWP's in support of a priority 1 work request. (0.5)
2. RWP's in support of a declared emergency. (0.5)

REFERENCE
WJE 202
HPP-006, Pp. 5,9

ANSWER 8.04 (1.80)

a. Two members of the plant staff [0.35], one of which has an SRO license. [0.35] (0.7)

b. 21 days. (0.5)

c. Identified by a vertical bar in the right hand margin with a "T" change number opposite the revised text. (0.6)

REFERENCE
AP-004, p. 27 of 46

ANSWER 8.05 (3.50)

a. 1. Reactivity monitor (1/m plotter) (0.5)

2. Refueling SRO (0.5)

3. Shift Foreman (0.5)

4. Reactivity monitor (0.5)

b. Facility to provide correct answer and documentation of same. (1.5)

Shift Foreman — SRO
Refueling SRO — SRO
CV Bridge — RO
SFP Bridge — RO

REFERENCE
WJE 204
OMM-006, Pp. 5-7

Ref. Review Comments 12-18-84

ANSWER 8.06 (2.80)

a. 1 Shift Foreman.
1 Senior Control Operator.
2 Control Operators.
2 Auxiliary Operators.
1 Shift Engineer.
1 Fire Protection Technical Aide. [0.1 ea. person] (0.8)

b. Control Room. (0.5)

c. ~~On coming and Off going~~ Shift Foreman, ~~Additional required answer.~~ (1.0)
SRO, CO, FPTA, AO.

d. When RCS temperature is less than 200 degrees. (0.5)
Will also accept that portions of MEL are always completed.

REFERENCE
AP-027, pp. 17 of 30, 18 of 30, 2 of 30

ANSWER 8.07 (2.50)

a. The Unit 2 Shift Forman [0.75] and The Fire Protection Technical Aide. [0.25] (1.0)

b. True (0.5)

c. Yes, if a Work Request is initiated and documented on the Certification and Review Form. (1.0)

REFERENCE
WJE 206
OMM-015, Pp. 5,6,7,8

ANSWER 8.08 (2.80)

a. Determine the deviation between the highest and lowest reading redundant indicators. (0.8)

b. Control operators ~~log sheet~~ stet / will also accept Minimum Equipment List. (0.5)

c. Deviations must be less than 4% of the indicators full range. (1.0)

d. A yellow sticker will be placed adjacent to the faulty indicator. (0.5)

REFERENCE
WJE 207
OMM-1006, Pp. 2-4

ANSWER 8.09 (2.50)

a. The Control Operator. (0.5)

b. 1. Planned release (liquid or gas).
2. Relaxed containment integrity. (1.0)

[accept 1 answer above, 1.0]

c. A setpoint change record will be filled out. (1.0)

REFERENCE
WJE 208
OMM-014, Pp. 5,11,12,13

ANSWER 8.10 (2.00)

a. Prevents an assembly from being partially uncovered during transfer over vessel flange to provide personnel radiation protection. (1.0)

b. Provides large enough heat sink so that sufficient time is available to initiate emergency cooling should RHR fail. (1.0)

REFERENCE
WJE 209
Tech. Spec., Pp. 3.8.4

SRO EXAM REVIEW - DECEMBER 18, 1984

COMMENTS

Section 1

Question 5b

Rod worth could be higher at BOL or EOL based on which core cycle the candidate is examining. Cycle 9 has rod worth higher at EOL where as cycle 10 has rod worth higher at BOL. Since the answer required no explanation, either answer, BOL or EOL could be given.

Question 6c answer

Differential boron worth increases over core life due to the burn out of Gadelinia and therefore less competition affects of Gd and the boron.

Reference - cycle 9 handout.

Question 7

The question is not clear as to what constitutes core age, one cycle BOC to EOC or BOC cycle 1 to EOC cycle 3.

Alt. answer

During the first cycle centerline temperature increases due to the contamination of the fuel rod fill gas with fission gasses and during the next two cycles it decreases due to clad creep and fuel pellet swell.

Question 4c

The question is unclear on what scenario is happening and what type of answer is required. This procedure has not been used since the very early cycles of operation of HBR and has recently been deleted. Therefore, we request this question be deleted.

Section 6

Question 2

a. Answer should be PZR PORV's (Power Operated Relief Valves) vs Safety Valves.

b. Alternate answer: "If high temperature water is sensed then a temperature directing valve (TCV-143) will automatically act to divert water around the demineralizers."

Question 3c

Due to interpretation of the question an alternate answer could be: "To prevent excessive cooldown of the RCS."

Question 8 answer

A reactor trip will not occur on PZR high level due to letdown isolation. On a low level signal in the PZR the letdown isolation valves will shut and the heaters will trip off. As the level builds up in the PZR the letdown isolation valves will reopen. However, without operator action the PZR heaters will not reenergize. (They must be manually reset) Therefore, the resulting trip will occur from low PZR pressure.

Reference: Logic Diagram CP-300-5379-3693

Question 10

b. Answer: The signal for the auto start of the steam driven AFW pump comes from a loss of power to 4160 Bus 1 and 4. We recommend that this or black out signal be used as acceptable answers.

a. Answer: Should be "Low-Low S/G Vessel (2/3) in <u>one</u> S/G."

Reference: Logic Diagram CP-300-5379-2762

<u>Section 7</u>

No comments

<u>Section 8</u>

Question 5b answer

OMM-006 requires that the shift foreman and the refueling SRO have an SRO license. However, at HBR in the past we have always used licensed RO's on the CV and SFP manipulator bridges in addition to the requirements of OMM-006.

Question 6c and d answers

c. Shift Foreman, Senior Reactor Operator, Control Operator, Fire Protection Tech. Aide and Auxiliary Operator.

d. The correct answer should be "Portions of the MEL are always completed, hot or cold."

RESOLUTION OF FACILITY COMMENTS FOR
SRO EXAMINATION AT H. B. ROBINSON
ON DECEMBER 19, 1984

Facility Comment Question 5.05b

Rod worth could be higher at BOL or EOL based on which core cycle the candidate is examining. Cycle 9 has rod worth higher at EOL whereas cycle 10 has rod worth higher at BOL. Since the answer required no explanation, either answer, BOL or EOL could be given.

Examiner Comment

Agree with facility. Either answer may be correct.

Examiner Resolution

Answer key changed to accept either answer if cycle 10 is referenced.

Facility Comment Question 5.06c

Differential boron worth increases over core life due to the burn out of Gadelinia and therefore less competition affects of Gd and the boron.

Examiner Comment

Agree. Based on cycle 9 data provided.

Examiner Resolution

Answer key changed to reflect additional answer.

Facility Comment 5.07

The question is not clear as to what constitutes core age, one cycle BOC to EOC or BOC cycle 1 to EOC cycle 3.

Examiner Comment

Agree. Question could have been more specific, however, the reference provided is not that specific.

Examiner Resolution

If explanation includes reference to multiple cycles, credit will be given for increasing FCT during cycle 1 and decreasing through remaining cycles.

Facility Comment 5.04c

The question is unclear on what scenario is happening and what type of answer is required. This procedure has not been used since the very early cycles of operation of HBR and has recently been deleted. Therefore, we request this question be deleted.

Examiner Response

The reference stated was supplied with the material from the facility. The concept of Xenon oscillations is still valid.

Examiner Resolution

Question will not be deleted.

Facility Comment 6.02a&b

a. Answer should be PZR PORV's (Power Operated Relief Valves) vs Safety Valves.

b. Alternate answer: "If high temperature water is sensed then a temperature directing valve (TCV-143) will automatically act to divert water around the demineralizers."

Examiner Comment

a. Agree

b. Agree

Examiner Resolution

a. Answer key changed.

b. The additional correct response added to answer key.

NOTE: The question will be changed to reflect the fact that H. B. Robinson teaches that resin damage will not occur below 145°F.

Facility Comment 6.03c

Due to interpretation of the question, an alternate answer could be: "To prevent excessive cooldown of the RCS."

Examiner Comment

Agree

Examiner Resolution

Additional correct response added to answer key.

Facility Comment 6.08 Answer

A reactor trip will not occur on PZR high level due to letdown isolation. On a low level signal in the PZR the letdown isolation valves will shut and the heaters will trip off. As the level builds up in the PZR the letdown isolation valves will reopen. However, without operator action the PZR heaters will not reenergize. (They must be manually reset). Therefore, the resulting trip will occur from low PZR pressure.

Reference: Logic Diagram CP-300-5379-3693

Examiner Response

Agree

Examiner Resolution

Answer key will be changed to reflect the supplied reference data.

Facility Comment 6.10a&b Answer

a. Answer: Should be "Low-Low S/G Vessel (2/3) in one S/G."

b. Answer: The signal for the auto start of the steam driven AFW pump comes from a loss of power to 4160 Bus 1 and 4. We recommend that this or black out signal to be used as acceptable answers.

Reference: Logic Diagram CP-300-5379-2762

Examiner Response

a. Agree

b. Agree

Examiner Resolution

a. Additional answer added to key.

b. Answer key changed.

Facility Comment 8.05b Answer

OMM-006 requires that the shift foreman and the refueling SRO have an SRO license. However, at HBR in the past we have always used licensed RO's on the CV and SFP manipulator bridges in addition to the requirements of OMM-006.

Examiner Response

Thank you.

Examiner Resolution

Provided answer was requested by the examiner. Answer will be added to the answer key.

Facility Comment 8.06c&d Answer

c. Shift Foreman, Senior Reactor Operator, Control Operator, Fire Protection Tech. Aide and Auxiliary Operator.

d. The correct answer should be "Portions of the MEL are always completed, hot or cold."

Examiner Comment

c. Agree

d. Agree

Examiner Resolution

c. Answer key changed to reflect proper answer.

d. Answer key will be added to reflect an additional correct answer.

ENCLOSURE 4

CP&L

Carolina Power & Light Company

ROBINSON NUCLEAR PROJECT DEPARTMENT
POST OFFICE BOX 790
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DEC 27 1984

Robinson File No: 13510 Serial: RSEP/84-1260

Mr. A. Gibson
Chief, Operations Branch
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 2900
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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
RO/SRO EXAM - DECEMBER 17 - 21, 1984

Dear Mr. Gibson:

A Reactor Operator, Senior Reactor Operator (RO/SRO) examination was administered by Mr. Ed Cook during the week of December 17. The following suggestions and comments on this examination are provided per NUREG-1021.

1. It is suggested that the point value of the exam be reviewed for consistency.

   Example: 1 point for AOP immediate actions
   1 point for knowing a number (RCS H_2 concentration)

2. The examination included questions on actions to take per the EOP network. The new EOP network was designed so the Operator would not have to memorize these actions. It was intended to remove the burden of memorization off the Operator. Therefore, questions requiring the memorization of actions in the EOPs should not be included in the exam.

3. SRO Exam, Section 6, Question 2(b). Resin damage does not occur until the temperature increases beyond 145°F which is the setpoint for TCV-143. Please consider this point when grading this question.

4. The length of time temporary procedures and special procedures are effective is not included in our RO training program and does not appear to be in 10CFR55. This has not been considered as required RO knowledge.

~~8502010304~~ 2pp.

5. When grading the exams, we suggest that you accept answers that are true even if not listed on the answer key.

   Example: There may be many symptoms for primary to secondary leak other than those listed in the answer key.

Should you have any questions, please contact Mr. Steve Allen at 803-383-4524 Extension 341.

Very truly yours,

R. E. Morgan

R. E. Morgan
General Manager
H. B. Robinson S.E. Plant

JCS/wp

Attachment

cc: G. P. Beatty, Jr.
C. A. Bethea
F. L. Lowery
D. R. Nelson
D. C. Stadler
A. C. Tollison
S. R. Zimmerman