APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Examination Report: 50-298/OL-85-04 License: DPR-46

Docket: 50-298

Licensee: Nebraska Public Power District (NPPD)
P. O. Box 499
Columbus, Nebraska 68601

Facility Name: Cooper Nuclear Station (CNS)

Requalification examinations administered at Cooper Nuclear Station (CNS)

Examinations Conducted: September 23-26, 1985

Chief Examiner: David N. Graves — 11/7/85
David N. Graves Date

Approved: R. A. Cooley — 11/8/85
R. A. Cooley, Section Chief Date

Inspection Summary

An audit of the Cooper Nuclear Station Requalification Program was conducted by Region IV personnel. This audit found the requalification program to be SATISFACTORY.

DETAILS

1. Facility-Generated Written Examination

   Facility examinations to be administered to Reactor Operators (RO) and Senior Reactor Operators (SRO) were reviewed for compliance with 10 CFR 55, Appendix A. This review showed the examinations to be appropriately structured as to content. The examinations were also satisfactory when compared to the guidelines of NUREG 1021, Operator Licensing Examiner Standards, Section ES-601, Administration of NRC Requalification Program Evaluation.

2. Persons Examined

   Five SROs and three ROs were administered NRC-generated written examinations and oral operating examinations.

3. Results of NRC Administered Examination

   One RO failed the written examination and the oral operating examination. One SRO failed the written examination.

4. Examiners

   D. N. Graves, Chief Examiner
   J. E. Whittemore

5. This section pertains to the site visit conducted September 24-26, 1985, and consists of the following subsections:

   a. Examination Review Comments and Resolutions.

      Facility comments are listed by section question number and followed by the Chief Examiner's resolution.

      1) Question 1.05 (Also 5.01).

         Comments: Value of 1*.USAR uses 50 microseconds; GE Reactor Physics Review uses 100 microseconds. NRC has previously accepted values for 1* between 50 and 150 microseconds. The value of 1* used will significantly change the answers in several parts and this should be taken into account in grading.

         Resolution: Agree. Accepted 1* values from 50 to 150 microseconds.

2) Question 1.08 (Also 5.08).

Comments: Assuming question is referring to available NPSH, Rx dome pressure should also be an acceptable answer. We teach: $NPSH_{avial}=(P_{dome}+P_{height}-P_{loss})-P_{sat}$

Resolution: Accepted Rx dome pressure as subcooling.

3) Question 1.11.

Comments: Student may also answer question in terms of ensuring MCPR Safety Limit is not exceeded. This should also be an acceptable answer.

Resolution: Agree.

4) Question 2.06.

Comments: REC Temperature Control is manual or automatic. Manual by varying SW flow with RCV-451 A or B in open position. Automatic Control by TCV-451 A and B is Auto Valve position controlled by REC Temperature out of Rx. Temperature adjustable at controller.

Resolution: Agree.

5) Question 3.01c (Also 6.02c).

Comments: Question could be misinterpreted. Student may assume question implies that SRV did not open. If student assumes SRV did not open, answer would be:

Red	Green	Blue
On	Off	On

This answer should also be accepted.

Resolution: Accepted if candidate states assumption.

6) Question 3.05 (Also 6.07).

Comments: Plant Terminology/Common Usage
LI-91 A and B: Fuel Zone, Rosemount, Wide Range, Wide Range Yarways
LI-85 A and B: Narrow Range Yarways
LI-94 A, B, C: GEMAC, Narrow Range GEMAC

Resolution: Accepted.

7) Question 3.07.

Comment: Plant Terminology. Student may refer to Log Rad Monitor as Off Gas Monitor or Off Gas RAD Monitor. Either of these terms should be accepted.

Resolution: Agree.

8) Question 4.01 (Also 8.01).

Comment: Due to the fact that as a normal course of action, an announcement regarding the nature of the alarm and immediate response actions required is made over the "public address" system, we believe that this question is inappropriate for a licensed operator examination. Further, we expect that all would recognize that a "ringing gong" would not be correct for any of the emergency signals; hence, one incorrect response results in at least two incorrect answers, a "double jeopardy" situation. On this basis we believe the question should be deleted or, if not, its value be readjusted downward.

Resolution: Disagree. Question and answer stand as written.

9) Question 4.03.

Comment: May receive the answer, "Main Steam Channel A and B low pressure alarms on panel 9-5 clear." May also receive an answer of "APRM gain adjusted to 1.0" as it is in the same paragraph as the other three answers and should be considered a correct answer.

Resolution: Agree.

10) Question 4.04 (Also 7.02).

Comment: Undesirable effects. CRD supplies seal purge H_2O to the RR Pumps and RWCU Pumps. If this supply is lost, seal life could be shortened. The loss of the pumps will eliminate the possibility of moving the control rods with the RMC system as hydraulic pressure is lost. Additional information regarding possible adverse effects if the CRD pumps are lost is provided in the referenced procedure.

Resolution: Agree.

11) Question 4.09d (Also 7.08d).

Comment: We believe that there may be other ways to answer this question. Procedure 2.1.10 states, "Suppress the oscillations by inserting control rods and/or increasing core flow. It is preferred to reverse the actions that caused the flux oscillations."

Resolution: Abnormal Procedure 2.4.2.2.1, Trip of Reactor Recirculation Pumps, covers this particular situation and makes no provision for increasing flow with the remaining pump. Answer stands as written.

12) Question 5.02a.

Comment: Boiling boundary. Student may answer question in terms of "sweeping away voids" and void content. This should also be an acceptable answer.

Resolution: Agree.

13) Question 5.06.

Comment: Question could possibly be misinterpreted. (Student could answer in terms of resonance capture, and the effect of resonance escape probability on Keff.) FOR FUTURE REFERENCE: Question may be reworded as such: State two (2) effects that cause resonance capture to increase with an increase in fuel temperature.

Resolution: Noted for future reference.

14) Question 6.04.

Comment: The referenced procedure assumes Standby Mode while the question does not address EDG condition. It is felt that some answers that are addressed in attaining Standby Mode may be included as correct.

(a) DC Control Power and Maintenance Lockout Keylock is in the "On" position. Lack of this Control Power will prevent Breaker operation.
(b) DG Breaker Selector Switch in Auto. This switch being out of Auto position will prevent Breaker operation other than manually, with synch scope interlock satisfied.

In addition, to fully answer the question, i.e., "and re-energize its associated bus?", it is felt that, "IFE (GE) closed," is also a correct answer.

Resolution: Agree.

15) Question 7.01.

Comment: While responses to the second part of this question may have been provided, it is no longer an applicable question for CNS since maintenance of DW/Torus dp is no longer required.

Resolution: Agree. Points redistributed.

16) Question 8.02 b&c.

Comment: Designation of a shift communicator should also be considered a correct response.

Resolution: Agree.

17) Question 8.04.

Comment:

(a) The basis of Tech Specs (pg. 102) states "A limiting Control Rod Pattern is a pattern which results in the core being on a Thermal Hydraulic Limit (i.e., MCPR = 1.07, and LHGR - as defined in 1.0.A.4)" Ref: Tech Specs; Definition Section, pg. 1 and Basis Section, pg. 102. It seems that the procedure has shortened the Tech Spec explanation/definition for ease of use. The above answer may be given and should be considered correct.

(b) We believe that this question should be deleted. The answer is based upon information provided in the Discussion section of the referenced procedure. Administrative Procedure 0.4 "Preparation, Review and Approval of Procedure" states that the discussion section of a procedure is to "provide a short discussion of the procedure and its objectives." The discussion section is intended; therefore, as no more than an aid; consequently, we would not expect an individual to commit to memory information contained in that section. While we recognize that licensed operations personnel need to "commit to memory" immediate action steps in an emergency procedure, (Note that Procedure 10.10 does not fit that classification), subsequent actions required need not be, let alone information contained in a Discussion section.

In that respect, CNS places the responsibility for determination of limiting control rod patterns on Reactor Engineering personnel. Such a responsibility is defined in both procedures and Technical Specifications. The following references are cited:

The basis of Tech Specs (pg. 102) states "It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns."

In addition, Procedure 10.10 II.B states "G.E. Studies in this field have indicated that Engineers who specialize in Control Rod pattern development and analysis learn to tell by visual examination whether a limiting configuration exists, but there has not been any simplified method to date; general curves, tables, or overlays which would guarantee that such an assessment could be made on an absolute basis by someone else. (i.e. Operators) The combination of odd circumstances to be covered to guarantee absolute certainty would be almost infinite in extent."

Also, Procedure 10.10.C pg 1 states "As such, each of the above abnormal withdrawal sequences shall be considered a potential limiting Control Rod pattern, unless, specific analyses are performed which demonstrates a limiting pattern does not exist. Also, Procedure 2.4.1.4; Step I.A states "The Reactor Engineer will notify the Shift Supervisor and have it entered in the Operations Log when a limiting Control Rod Pattern exists."

Further, the answer requires that three (3) responses be provided. This comprises all of the potential responses identified in the Discussion section. Hence, to obtain full credit, all of the elements within the Discussion section need to be addressed. We believe that, in itself, is inappropriate. Finally, the point value assigned amounts to 1/6 of the total point value in that section. This is an excessive value assigned to a question which we feel to be inappropriate due to the fact that Reactor Engineering Personnel are specified in the Technical Specifications as being responsible for identifying control rod patterns.

Resolution:

(a) Agree

(b) Disagree. The question does not ask to identify a specific limiting control rod pattern. The question states the general way to obtain a limiting pattern is by an abnormal withdrawal sequence. It is reasonable to expect an SRO to be aware of conditions under which abnormal patterns (not specific patterns) could be encountered. Question and answer stand.

b. Exit Meeting Summary

At the conclusion of the site-visit, the examiners met with utility representatives to discuss the results of the examinations. The

following personnel were present for the exit meeting:

NRC	Facility
D. L. Dubois, Senior Resident Inspector	R. D. Black
D. N. Graves	D. L. Reeves, Jr.
	T. Sandner
	J. Sayer
	P. V. Thomason

The meeting was started by Mr. Graves announcing preliminary results of the oral operating examination with one RO not being a "clear pass" as of the exit meeting. The facility was informed that this did not mean the candidate failed, but that a more detailed review of his performance must be made.

Observations made during the examination were discussed as follows:

1) Several SROs were indecisive and hesitant concerning actions required by the Emergency Plan. Given a set of conditions, several were hesitant to make decisions or commit to a specific course of action.

2) In at least one instance, the Emergency Action Level classification chart identifies one event under two classifications. Specifically, a main steam line break with loss of primary containment integrity is addressed specifically as a Site Emergency yet also satisfies the criteria for a loss of two of three fission product barriers, which is a General Emergency.

The meeting concluded with the examiners thanking the facility staff for their cooperation and informing them that final results would be forthcoming as soon as possible.

c. Requalification Program Evaluation

The Requalification Program Evaluation Report is presented on the next page.

d. Master Copies of the RO and SRO Written Requalification Examination

Copies of RO and SRO examinations follow the Requalification Program Evaluation Report.

REQUALIFICATION PROGRAM EVALUATION REPORT

Facility: COOPER NUCLEAR STATION
Examiner: D. N. Graves
Date(s) of Evaluation: September 24-26, 1985
Areas Evaluated: XX Written XX Oral ___ Simulator

Written Examination

1. Overall evaluation of examination: (NRC written) 6 of 8 candidates passed

2. Evaluation of facility examination grading: N/A

Oral Examination

1. Overall evaluation 7 of 8 candidates passed NRC administered oral exams

2. Number conducted 8

Simulator Examination

1. Overall evaluation N/A

2. Number conducted N/A

Overall Program Evaluation

Satisfactory XX Marginal ___ Unsatisfactory ___ (List major deficiency areas with brief descriptive comments)

Submitted: David N. Graves
Examiner

Forwarded: R. G. Cooley
Section Chief

Approved: Eric H. Johnson
for Branch Chief

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: COOPER

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 85/09/26

EXAMINER: GRAVES, D.

APPLICANT: ____________________

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answe 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
15.00	25.00	________	________	1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
15.00	25.00	________	________	2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
15.00	25.00	________	________	3. INSTRUMENTS AND CONTROLS
15.00	25.00	________	________	4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
60.00	100.00	________	________	TOTALS

FINAL GRADE ______________%

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

____
APPLICANT'S SIGNATURE

QUESTION 1.01 (2.00)

a. What combination of recirculation flow and control rod position will yield the MAXIMUM PERCENT VOIDS in the core? (Specific values not required) (1.0)

b. Why does the void percentage DECREASE as power is raised from the level in part "a" to 100% power? (1.0)

QUESTION 1.02 (2.00)

What are five (5) mechanisms or methods by which reactivity additions are made in the power range? (2.0)

QUESTION 1.03 (2.00)

Consider a turbine trip from approximately 25% power. Briefly explain what would be expected to occur to the plant in the next 15 minutes with no operator action. Include parameters such as power, pressure, temperature, etc. Provide reasons for all changes. (2.0)

QUESTION 1.04 (2.00)

Concerning control rod worths during a reactor startup from 100% PEAK XENON versus a startup under XENON-FREE conditions, which statement is correct? JUSTIFY YOUR CHOICE. (2.0)

a. PERIPHERAL control rod worth will be LOWER during the PEAK XENON startup than during the XENON-FREE startup.

b. CENTRAL control rod worth will be HIGHER during the PEAK XENON startup than during the XENON-FREE startup.

c. BOTH control rod worths will be the SAME regardless of core Xenon conditions.

d. PERIPHERAL control rod worth will be HIGHER during the PEAK XENON startup than during the XENON-FREE startup.

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

QUESTION 1.05 (1.00)

How much (by what factor) would power increase in one second in a prompt critical reactor at EOL? (1.0)

QUESTION 1.06 (1.00)

Using the Steam Tables, calculate a reactor cooldown rate (F/hr) for a reactor pressure decrease from 1000 psig to 250 psig in one hour and forty-five minutes (105 minutes total). Show all work for full credit. (1.0)

QUESTION 1.07 (1.00)

Select the answer below which best describes Net Positive Suction Head: (1.0)

a. The difference between the pump suction pressure and the pump discharge pressure.

b. The pressure above which the pump can no longer provide flow

c. The difference between Psat at the pump discharge and the actual pump discharge pressure.

d. The difference between Psat at the pump suction and the actual pump suction pressure.

QUESTION 1.08 (1.00)

What are the three (3) major parameters affecting the NPSH of the recirculation pumps? (1.0)

QUESTION 1.09 (1.00)

During a LOCA, large amounts of hydrogen may be produced. Why will adequate core cooling prevent a great deal of this hydrogen from being produced? (1.0)

QUESTION 1.10 (1.00)

Which type of fuel, new fuel or exposed fuel, would have the longer thermal time constant? EXPLAIN. (1.0)

QUESTION 1.11 (1.00)

How is it assured that 99.9% of the fuel pins in the core do not experience transition boiling during a transient? (1.0)

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

QUESTION 2.01 (2.00)

a. What are six (6) areas served by the Low Pressure CO2 System? Indicate whether the area is covered AUTOMATICALLY or requires MANUAL actions. (1.5)

b. After the initial 50 second discharge of CO2, how are subsequent discharges initiated (2 methods or locations required)? (0.5)

QUESTION 2.02 (2.00)

For each of the following diesel generator controls, describe its effect(s) on diesel generator performance/operation. Include how the listed controls affect automatic and manual starts and the relationship(s) between the controls as applicable.

a. Control Mode Selector Switch (when in REMOTE) (0.5)

b. Diesel Control Isolation Switch (when in LOCAL) (0.5)

c. Diesel Generator Stop Pushbutton (on D/G engine panel) (0.5)

d. Emergency Stop Pushbutton (on D/G engine panel) (0.5)

QUESTION 2.03 (2.00)

For each of the following items (a - d), indicate whether it describes proper system response or not. If it does not, describe how the discussed event should occur.

a. With the RCIC system operating, a low level occurs in an ECST. The pump suction from the ECST (RCIC-MO-18) closes and the pump suction from the suppression chamber (RCIC-MO-41) then opens. (0.5)

b. The RCIC system is operating with reactor level increasing. When the high level setpoint is reached, the steam supply inboard and outboard isolation valves (RCIC-MO-15 and 16) close. (0.5)

c. The RCIC system is operating in the TEST mode, discharging to the ECST. A valid low reactor level initiation signal is received. The test circuitry is automatically bypassed, the test bypass to ECST closes, and the flow controller controls RCIC flow automatically. (0.5)

d. With the RCIC system operating, a high steam line space temperature isolation signal is generated. The following valves close as a result of the isolation: Steam supply inboard and outboard isolation valves (RCIC-MO-15 and 16), the minimum flow valve (RCIC-MO-27), the RCIC pump discharge valve (RCIC-MO-20), and the injection valve (RCIC-MO-21). NOTE: The above listed valves are not the only valves that are affected by the isolation. EVALUATE ONLY THE VALVES LISTED. DO NOT ATTEMPT TO COMPLETE THE LIST. (0.5)

QUESTION 2.04 (1.50)

a. The DRYWELL COOLING SYSTEM is operating with three (3) units running and one in STANDBY. How does the system respond to a LOCA signal? (0.5)

b. What system supplies cooling water to the DRYWELL COOLING SYSTEM? (0.5)

c. What will cause a DRYWELL COOLING SYSTEM unit to automatically start? (0.5)

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

QUESTION 2.05 (1.50)

a. What two (2) mechanisms or methods are used in the Offgas System to maintain or reduce hydrogen concentration of the offgas? (1.0)

b. Why is there a maximum power limit imposed on the use of the mechanical vacuum pumps? (0.5)

QUESTION 2.06 (1.50)

a. How is the temperature of the REC system controlled? Include whether control is automatic or manual, and how the temperature is adjusted. NO VALUES are required. (1.0)

b. What will cause the REC critical loops to automatically be placed into service? (0.5)

QUESTION 2.07 (1.50)

a. Where does the Standby Gas Treatment System line up to take a suction on an automatic initiation due to a refueling accident? (0.5)

b. Other than the normal automatic initiation supply and dilution air, what are two (2) additional areas or components that can provide a supply to the Standby Gas Treatment System? (1.0)

QUESTION 2.08 (2.00)

a. What provides the NORMAL and BACKUP cooling supplies to the three (3) plant air compressors? BE SPECIFIC. (1.5)

b. Is the normal to backup cooling water relationship a manual or automatic feature? If automatic, include what parameter(s) cause(es) the transfer to occur. Setpoints not required. (0.5)

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

QUESTION 2.09 (1.00)

Control Rod Drive Hydraulic System flow is automatically maintained at approximately 45 gpm. Describe the flow path for this water, with no CRD motion, from the CRD Flow Control Valve to where it mixes with the majority of the reactor coolant. (1.0)

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

QUESTION 3.01 (3.00)

Describe the indication (ON or OFF) of the three (3) indicating lights for a Safety/Relief Valve under each of the following conditions: (3.0)

a. No actuation signals present, valve shut

b. The valve is leaking significantly

c. The valve handswitch is in OPEN

d. The valve opens due to high reactor pressure

e. ADS logic actuated

QUESTION 3.02 (1.50)

Following a reactor scram, the four rod display position goes blank, but the green full-in light on the full core display for that control rod is lighted. Is this normal? If so, explain why it occurs. If not, describe the probable cause. (1.5)

QUESTION 3.03 (2.00)

a. Which two (2) SRM rod blocks are bypassed when the IRM's are on range 5? Setpoints not required. (1.0)

b. Which two (2) SRM rod blocks are bypassed (in addition to the 2 above) when the IRM's are on range 8 or above? Setpoints not required. (1.0)

QUESTION 3.04 (1.50)

Describe three (3) rod blocks associated with refueling equipment. Include required Mode Switch position(s), as applicable. (1.5)

QUESTION 3.05 (2.50)

a. List eight (8) indications of reactor vessel water level in the Control Room. Include on which panel the indication is located. (2.0)

b. How far is INSTRUMENT ZERO (0 on most level ranges) above the top of the active fuel (TAF)? (0.5)

QUESTION 3.06 (2.00)

Indicate whether each of the following statements about the operation of the DEH system in Mode 4 (Turbine Follow - Reactor Manual) are TRUE or FALSE. For those that are FALSE, BRIEFLY EXPLAIN WHY.

a. Raising the pressure control signal above the load reference will cause the bypass valves to open. (0.5)

b. Reducing the valve position limiter setting to below the pressure control signal will result in an increase in reactor pressure. (0.5)

c. Reactor power is controlled by the operator adjusting the load reference signal. (0.5)

d. Governor valve control will remain in AUTOMATIC if a loss of the speed loop (speed signal) occurs. (0.5)

QUESTION 3.07 (2.00)

a. What combination(s) of Off Gas Radiation Monitoring System trips will start the Off Gas Valve Timer? (1.0)

b. What four (4) valves are isolated (shut) at the completion of the Off Gas Valve Timer cycle? Valves may be identified by number or description. (1.0)

QUESTION 3.08 (.50)

Which of the following provides the signal for a Turbine Control Valve (TCV) Fast Closure scram? (0.5)

1. TCV position limit switches

2. Rate of TCV position change

3. Power to the TCV fast acting solenoids

4. Turbine control fluid pressure

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

QUESTION 4.01 (1.50)

Match the emergency signal (a - c) with its tone description. (1.5)

_____a. Emergency alarm	1. distinct steady tone
_____b. Fire alarm	2. one steady up and down tone
_____c. All clear	3. ringing gong
	4. distinct pulse tone

QUESTION 4.02 (2.00)

A control rod coupling test is to be performed. Briefly describe how the check is performed AND provide four (4) indications that the operator would see if the rod was uncoupled. (2.0)

QUESTION 4.03 (1.50)

List three (3) verifications that should be made prior to placing the Mode Switch to RUN during a plant startup. (1.5)

QUESTION 4.04 (2.00)

With the reactor operating at power, a loss of both CRD pumps occurs. What are two (2) undesirable effects of this loss and what makes these effects undesirable? (2.0)

QUESTION 4.05 (2.00)

a. The control room is filling with a noxious vapor from an undetermined source. The Shift Supervisor decides to implement the "Toxic Gas in Control Room" procedure. What IMMEDIATE ACTIONS are required of operating personnel by this procedure? (1.0)

b. The above actions are determined to be inadequate and the control room is to be evacuated. What actions should be taken, if possible, prior to leaving the control room, by the Control Room Operators? (1.0)

QUESTION 4.06 (1.50)

a. A gaitronics announcement is made concerning a Class Bravo fire. What is meant by a Class Bravo fire? (1.0)

b. If the fire involves radioactive material(s), an individual should not remain in the area for longer than _________ without respiratory protection. (0.5)

QUESTION 4.07 (1.00)

Boron injection into the reactor is required per EOP-1, RPV Control, and the Standby Liquid Control System is incapable of injecting into the RPV. Describe basically how boron injection to the RPV is accomplished under the above circumstances. Specific procedural steps ARE NOT required. (1.0)

QUESTION 4.08 (1.50)

What are the Emergency Dose Exposure Limits per EPIP 5.7.12, Emergency Radiation Exposure Control, for:

a. Sampling under accident conditions (0.5)

b. Corrective or protective actions (0.5)

c. Life-saving actions (0.5)

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

QUESTION 4.09 (2.00)

The plant is operating at power with both recirculation pumps operating at minimum speed. ONE pump trips and backflow is NOT established in the idle jet pumps.

a. In this condition, how will indicated core flow compare with actual core flow? (0.5)

b. In the condition above, in what region of the power/flow map will the plant appear to be operating? (0.5)

c. What action must the operator take to ensure that a correct core flow signal is used by the process computer for core parameter calculations? (0.5)

d. What action should be taken to avoid or control abnormal neutron flux oscillations (not actions to observe flux oscillations, but to control them)? (0.5)

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

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

NRC LICENSE EXAMINATION HANDOUT

EQUATIONS, CONSTANTS, AND CONVERSIONS

$\dot{Q} = \dot{m}*C_p*\text{deltaT}$ $\qquad$ $\dot{Q} = U*A*\text{deltaT}$

$P = P_o*10^{sur*(t)}$ $\qquad$ $P = P_o*e^{t/T}$ $\qquad$ $SUR = 26/T$

$T = 1^*/\rho + (\beta-\rho)/\bar{\lambda}\rho$ $\qquad$ $T = 1/(\rho-\beta)$ $\qquad$ $T = (\beta-\rho)/\bar{\lambda}\rho$

$\rho = (K_{eff}-1)/K_{eff} = \text{delta}K_{eff}/K_{eff}$ $\qquad$ $\rho = 1^*/TK_{eff} + \bar{\beta}_{eff}/(1+\bar{\lambda}T)$

$\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$ $\qquad$ $\bar{\lambda} = 0.1 \text{ seconds}^{-1}$

$I = I_o*e^{-ux}$

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

$R/hr = 6*CE/d^2$ feet

Water Parameters

1 gallon = 8.345 lb_m = 3.87 liters
1 ft^3 = 7.48 gallons
Density @ STP = 62.4 lb_m/ft^3 = 1 gm/cm^3
Heat of vaporization = 970 Btu/lb_m
Heat of fusion = 144 Btu/lb_m
1 atmosphere = 14.7 psia = 29.9 inches Hg.

Miscellaneous Conversions

1 curie = 3.7×10^{10} disintegrations per second
1 kilogram = 2.21 lb_m
1 horsepower = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1 inch = 2.54 centimeters
degrees F = 9/5 degrees C + 32
degrees C = 5/9 (degrees F - 32)
1 Btu = 778 $ft\text{-}lb_f$

ANSWERS -- COOPER   -85/09/26-GRAVES, D.

ANSWER 1.01 (2.00)

a. At the 100% rod line (0.5) with the recirc pumps at minimum speed (0.5).
b. Void percentage decreases to counter the increased negative reactivity from the fuel temperature coefficient (1.0) or similar explanation.

REFERENCE
GE Reactor Physics Review, pg 52

ANSWER 1.02 (2.00)

- control rod movement
- recirculation flow changes
- fuel temperature changes
- fission product poison changes
- void content change (pressure changes)

(5 at 0.4 each)

REFERENCE
GE Reactor Theory Review, pg 51

ANSWER 1.03 (2.00)

BPV's open to pass the steam previously going to the turbine (0.5). Feedwater temperature will decrease due to a loss of extraction steam (0.5). Reactor power will increase due to the decrease in feedwater temperature (0.5). Reactor pressure will increase due to the increase in reactor power (0.5).

REFERENCE
BWR-4 Transients

ANSWER 1.04 (2.00)

"d" is the correct answer(0.5). The highest xenon concentration will be in the center of the core(0.5), the high flux region from the previous operating period(0.5). This will increase the flux in the area of the peripheral rods(0.5) thus increasing their worth.

REFERENCE
GE Reactor Physics Review, pg 36-37

ANSWER 1.05 (1.00)

T=l-star/p + (B-p)/tp (0.1)
so for prompt critical neglect delayed term & T=l-star/p (0.1)
l* = 10EE-04 seconds (accept 50 to 150 microseconds) (0.1)
p ~ .005 (0.1)
T = 10EE-4/.005 = 0.02 seconds (0.25)
P/Po=eEEtime/T (0.1)
P/Po=eEE50 = 5.18 X 10EE21 (0.25)
Numerical values do not have to be exact for full credit. Reasonable assumptions accepted.

REFERENCE
GE Reactor Theory Review, pg 20

ANSWER 1.06 (1.00)

Obtain corresponding temperatures from steam tables by interpolation:
1000 psig = 546.3 deg F (0.25)
250 psig = 406.0 deg F (0.25)
Determine the temperature change: 546.3 - 406.0 = 140.3 deg F (0.25)
Determine the rate of cooldown: 140.3/1.75 hr = 80.2 deg/hr (0.25)

REFERENCE
Steam Tables

ANSWER 1.07 (1.00)

d (1.0)

REFERENCE
GE Thermodynamics, Heat Transfer and Fluid Flow, pg 7-91

ANSWER 1.08 (1.00)

- height of the column of water above the eye of the pump
- amount of subcooling of the above water or reactor dome pressure
- irreversible flow losses in the suction line

(3 at 0.33 each)

REFERENCE
GE Thermodynamics, Heat Transfer and Fluid Flow, pg 7-93

ANSWER 1.09 (1.00)

The hydrogen generation becomes measurable when the cladding temperature becomes elevated (> 2200 deg F). Adequate core cooling prevents the required temperature from being reached. (or similar explanation) (1.0)

REFERENCE
CNS Mitigating Core Damage, Gas Generation Section

ANSWER 1.10 (1.00)

New fuel would have the longer thermal time constant (0.5) due to the increase in fuel to clad contact in exposed fuel (0.5).

REFERENCE
General Electric Heat Thermodynamics, Heat Transfer, and Fluid Flow pg 9-131, question 34
Problem Solutions, pg 9-5

ANSWER 1.11 (1.00)

Transition boiling is avoided by maintaining MCPR above the operating or safety limit (1.0).

REFERENCE
GE Thermodynamics, Heat Transfer and Fluid Flow, pg 9-93

ANSWER 2.01 (2.00)

a. Reactor building MG set room - Manual
Control building cable spreading room - manual
Control room entrance - manual
Turbine building switchgear room - manual
Main generator - manual
Turbine bearing #1 area - automatic
Turbine bearing #2 area - automatic
Turbine bearing #3 area - automatic
(6 areas at C.15 each, 6 initiations at 0.1 each)

b. - Reset button on the sprinkler control and fire alarm panel in the Control Room (0.25)
- Manual pushbuttons in the NW corner and on the North wall of the turbine generator operating floor shield wall (0.25)

REFERENCE
OP 2.2.2, Carbon Dioxide System, Rev 14, pg 2,5

ANSWER 2.02 (2.00)

a. Allows the D/G to be manually started and stopped from the Control Room (0.5).

b. When in LOCAL, disconnects all remote control and automatic start signals to the diesel (0.25) as well as Control Room indication for the D/G (0.25).

c. Allows local shutdown of the D/G (0.25) if the Control Mode Selector Switch is in LOCAL (0.25).

d. Allows shutting down the D/G under any conditions (0.5)

REFERENCE
SOP 2.2.20, Standby AC Power System (D/G), Rev 19, pg 14

ANSWER 2.03 (2.00)

a. No, the torus suction valve opens first, then the ECST suction valve shuts (0.5).

b. No, the steam supply block valve (RCIC-MO-131) shuts (0.5).

c. Yes (0.5)

d. No, the discharge and injection valves (RCIC-MO-20 and 21) do not close on an isolation (0.5).

REFERENCE
SOP 2.2.67, RCIC, Rev 24, pg 2-4
RCIC System Description, pg 5

ANSWER 2.04 (1.50)

a. All units that are running will stop automatically (0.5).

b. REC (0.5)

c. High area temperature near CRD hydraulic piping (0.5)

REFERENCE
SOP 2.2.40, HVAC Drywell Cooling, Rev 7, pg 2, 4

ANSWER 2.05 (1.50)

a. - dilution (0.5)
 - recombination (0.5)

b. Due to the possibility of combustion within the vacuum pump (0.5)

REFERENCE
Offgas and Augmented Offgas System Description, pg 8, 10

ANSWER 2.06 (1.50)

a. Temperature control is manual or automatic (0.5). REC temperature is adjusted by varying the service water flow through the REC heat exchangers (0.5).

b. When any of the Core Standby Cooling Systems (Core Spray, RHR, HPCI, or RCIC) are placed into service (0.5). Also accept Group VI isolation.

REFERENCE
REC System Description, pg 16

ANSWER 2.07 (1.50)

a. Reactor building exhaust plenum (0.5)

b. - HPCI gland steam condenser exhauster (0.5)
   - Drywell and torus purge exhaust (0.5)

REFERENCE
SBGTS System Description, pg 6, Figure 1

ANSWER 2.08 (2.00)

a.

	NORMAL	BACKUP
Compressor A:	REC	TEC
Compressor B:	TEC	REC
Compressor C:	TEC	REC

(6 at 0.25 each)

b. Manual (0.5)

REFERENCE
Plant Air System Description, pg 6

ANSWER 2.09 (1.00)

The coolant flows from the flow control valve into the cooling water header (0.33) and into the insert line of each individual CRD (0.33). From there it flows through the CRD into the reactor vessel (0.33).

REFERENCE
CRD System Description, pg 21-22, Figure 1B

ANSWER 3.01 (3.00)

	RED	GREEN	BLUE
a.	off	on	on
b.	off	on	off
c.	on	off	off
d.	off	on	off
e.	on	off	off

(15 at 0.2 each)

REFERENCE
Nuclear System Pressure Relief System Description, pg 4,6,8,9

ANSWER 3.02 (1.50)

Yes, it is normal (0.5). The drive piston moves the RPIS magnet past the "00" reed switch and actuates only the green full-in "overtravel" reed switch (1.0).

REFERENCE
CRD System Description, pg 18
RMC System Text

ANSWER 3.03 (2.00)

a. - SRM downscale (0.5)
   - Retract Not Permitted (0.5)

b. - SRM Inop (0.5)
   - SRM Upscale (0.5)

REFERENCE
Instrumentation Operating Procedure 4.1.1, SRM's, Rev 6, pg 2

ANSWER 3.04 (1.50)

ROD BLOCK	MODE SWITCH POSITION
- Service platform jib crane loaded	Startup/Hot Standby or Refuel
- Refueling platform over core	Startup/Hot Standby
- Refueling platfrom over core with any of its 3 hoists loaded	Refuel
- Refueling platform over core with fuel grapple not fully up	Refuel

(3 blocks at 0.25 each, 3 positions at 0.25 each)

REFERENCE
Instrumentation Operating Procedure 4.3, Reactor Manual Control System, Rev 10, pg 6, 7

ANSWER 3.05 (2.50)

a. - Post-accident monitor A or LI-91A. Located on panel 9-3
- Post-accident monitor B or LI-91B. Located on panel 9-3
- Wide range for vessel flooding or LI-86. Located on panel 9-4
- ECCS Yarway A or LI-85A. Located on panel 9-5
- ECCS Yarway B or LI-85B. Located on panel 9-5
- GEMAC feedwater control A or LI-94A. Located on panel 9-5
- GEMAC feedwater control B or LI-94B. Located on panel 9-5
- GEMAC feedwater control C or LI-94C. Located on panel 9-5
- Narrow Range Recorder or LR-6-97. Located on panel 9-5
- Wide range during shutdown for core coverage or LR-6-98. Located on panel 9-5

(8 indications required at .125 each, locations at .125 each)

NOTE: LI-91A & B: Fuel Zone, Rosemount, Wide Range, Wide Range Yarways
LI-85A & B: Narrow Range Yarways
LI-94A, B,& C: GEMAC, Narrow Range GEMAC
These are alternate names for the above instruments.

b. 164.19 inches. Accept 160 - 170 inches (0.5)

REFERENCE
Instrumentation Operating Procedure 4.6.1, Reactor Vessel Water Level Indication, Rev 9, pg 2,
Attachment "C" to the above procedure

ANSWER 3.06 (2.00)

a. True (0.5)
b. False (0.25). Bypass valves will open to maintain pressure at its present value (0.25).
c. False (0.25). Reactor power is controlled with control rods and recirculation flow (0.25).
d. True (0.5).

REFERENCE
DEH System Description

ANSWER 3.07 (2.00)

a. Log Rad Monitor High - High (0.25) Log Rad Monitor = Off Gas Rad Mon.
   Log Rad Monitor Downscale (0.25)
   Log Rad Monitor Inop (0.25)
   Any combination of one of the above in each channel will activate the Off Gas Valve Timer , i.e. a High - High in channel A and a Downscale in channel B (0.25).

b. OG - 254AV or ERP inlet valve (0.25)
   OG - 902AV or AOG outlet valve to the ERP (0.25)
   OG - AO-12 or holdup pipe drain valve (0.25)
   OG - AO-13 or Offgas filter drain valve (0.25)

REFERENCE
Instrumentation Operating Procedure 4.7.2, Air Ejector Off Gas Radiation Monitoring System, Rev 9, pg 1, 2
Off Gas System Description, figure 2

ANSWER 3.08 (.50)

4 or turbine control fluid pressure (0.5)

REFERENCE
RPS System Description, pg 9

ANSWER 4.01 (1.50)

a. 1
b. 4
c. 2
(0.5 each)

REFERENCE
SOP 2.2.4, Communications, Rev 13, pg 6

ANSWER 4.02 (2.00)

With the control rod at position 48 (0.5), initiate a rod out signal (0.5). If the control rod is uncoupled: 1) The indicating number 48 will go out, 2) the background will go blank, 3) and the red full indicating light will go out. 4) ROD OVERTRAVEL and 5) ROD DRIFT annunciators will alarm when the timer cycle is complete.
(4 of 5 indications required at 0.25 each)

REFERENCE
Instrumentation Operating Procedure 4.3, Reactor Manual Control System, Rev 10, pg 12

ANSWER 4.03 (1.50)

- APRMs are not downscale
- Vacuum is established
- Reactor pressure is greater than 825 psig or Main Steam Channel A and B Low Pressure alarms on 9-5 clear
- APRM gain adjusted to 1.0
(3 required at 0.5 each)

REFERENCE
General Operating Procedure 2.1.1, Cold Startup Procedure, Rev 39, pg 10

ANSWER 4.04 (2.00)

The loss of the CRD pumps causes a loss of cooling water flow (0.5), which shortens CRD seal life (0.5), and also allows the CRD hydraulic accumulators to slowly depressurize (0.5) which reduces the scram capability of the reactor (0.5), especially at low pressures. Also accept loss of seal purge to RWCU and RR pumps which can shorten seal life and loss of ability to move control rods with RMCS as hydraulic pressure is lost.

REFERENCE
Abnormal Procedure 2.4.1.1.4, Loss of CRD Pump, Rev 5, pg 2
System Procedure 2.2.8, Control Rod Drive System, pg 2

ANSWER 4.05 (2.00)

a. - Start the Control Room Ventilation Booster Fan BF-C-1A (0.33)
   - Secure the Control Room Ventilation Supply Fans SF-C-1A & B (0.33)
   - Essential Control Room personnel obtain self contained breathing apparatus and use as necessary (0.33)

b. - Scram the reactor (0.25)
   - Verify all control rods inserted (0.25)
   - Trip the main turbine (0.25)
   - Ensure the reactor feed pump turning gear control switches are in Auto (0.25)

REFERENCE
Abnormal Procedure 2.4.8.5, Toxic Gas in Control Room, Rev 2, pg 1
Emergency Procedure 5.2.1, Shutdown from Outside the Control Room, Rev 13, pg 1

ANSWER 4.06 (1.50)

a. Flammable or combustible liquids, flammable gases, greases, and similar materials (1.0).

b. 2 minutes (0.5)

REFERENCE
Emergency Procedure 5.4.1, General Fire Procedure, Rev 17, pg 1, 5

ANSWER 4.07 (1.00)

Filling the RWCU demineralizers with borated water and injecting this via the RWCU system (1.0).

REFERENCE
Emergency Procedure 5.2.14, Alternate Means to Inject Boron to RPV, Rev 0

ANSWER 4.08 (1.50)

a. 5 REM (0.5)
b. 25 REM (0.5)
c. 75 REM (0.5)

REFERENCE
Emergency Plan Implementing Procedure 5.7.12, Emergency Radiation Exposure Control, Rev 5, Attachment "A"

ANSWER 4.09 (2.00)

a. Indicated core flow will be lower than actual (0.5)

b. To the left of the natural circulation line (0.5)

c. The operator must enter a substitute value for core flow equal to the natural circulation flow for the existing power level (0.5).

d. Insert control rods until power is < or = 75% load line (0.5).

REFERENCE
General Operating Procedure 2.1.10, Station Power Changes, Rev 9, Attachment "A"
General Operating Procedure 2.1.15, Reactor Recirculation Pump Startup and Shutdown, Rev 13, pg 3, 4
Abnormal Procedure 2.4.2.2.1, Trip of Reactor Recirculation Pumps, Rev 12, pg 1-3

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

FACILITY: COOPER

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 85/09/26

EXAMINER: GRAVES, D.

APPLICANT: ______________________

INSTRUCTIONS TO APPLICANT:

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

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
15.00	25.00	________	________	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
15.00	25.00	________	________	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
15.00	25.00	________	________	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
15.00	25.00	________	________	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
60.00	100.00	________	________	TOTALS

FINAL GRADE ________________%

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

____
APPLICANT'S SIGNATURE

QUESTION 5.01 (1.00)

How much (by what factor) would power increase in one second in a prompt critical reactor at EOL? (1.0)

QUESTION 5.02 (2.00)

The reactor is operating at 75% power. Recirculation flow is subsequently increased to provide 100% power and 100% flow. Describe the effect that the above increase will have on each of the below items. Continue your description until steady state conditions are reached.

a. Core Void Content (1.0)

b. Core Net Reactivity (1.0)

QUESTION 5.03 (2.00)

Concerning control rod worths during a reactor startup from 100% PEAK XENON versus a startup under XENON-FREE conditions, which statement is correct? JUSTIFY YOUR CHOICE. (2.0)

a. PERIPHERAL control rod worth will be LOWER during the PEAK XENON startup than during the XENON-FREE startup.

b. CENTRAL control rod worth will be HIGHER during the PEAK XENON startup than during the XENON-FREE startup.

c. BOTH control rod worths will be the SAME regardless of core Xenon conditions.

d. PERIPHERAL control rod worth will be HIGHER during the PEAK XENON startup than during the XENON-FREE startup.

QUESTION 5.04 (3.00)

The reactor is operating at 100% power when HPCI inadvertantly initiates. Describe the response of the following parameters during the transient, including why the parameter changes as it does. Assume NO SCRAM occurs. Continue your description until steady state conditions are reached.

a. Reactor Pressure (1.0)

b. Reactor Water Level (1.0)

c. Feedwater Flow (1.0)

QUESTION 5.05 (1.50)

List the three (3) primary sources of hydrogen during an accident. (1.5)

QUESTION 5.06 (1.00)

Give two (2) reasons or factors that cause the fuel temperature coefficient to be negative. (1.0)

QUESTION 5.07 (2.00)

Using the steam tables, indicate whether water at each of the following is SUBCOOLED, SATURATED, or SUPERHEATED. (2.0)

a. 200 psig, 387.7 F

b. 1000 psig, 544.6 F

c. 1200 psig, 603.9 F

d. 900 psig, 531.1 F

QUESTION 5.08 (1.00)

What are the three (3) major parameters affecting the NPSH of the recirculation pumps? (1.0)

QUESTION 5.09 (1.00)

Pellet/cladding interaction, a stress corrosion type cracking that can occur during rapid power increases in irradiated fuel, occurs due to two (2) factors. List both factors. (1.0)

QUESTION 5.10 (.50)

a. TRUE or FALSE? APLHGR is a function of fuel burnup. (0.25)
b. FILL IN THE BLANKS. The APLHGR limit is based on keeping __(i)__ below __(ii)__ degrees F. (0.25)

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

QUESTION 6.01 (2.00)

a. What are six (6) areas served by the Low Pressure CO2 System? Indicate whether the area is covered AUTOMATICALLY or requires MANUAL actions. (1.5)

b. After the initial 50 second discharge of CO2, how are subsequent discharges initiated (2 methods or locations required)? (0.5)

QUESTION 6.02 (3.00)

Describe the indication (ON or OFF) of the three (3) indicating lights for a Safety/Relief Valve under each of the following conditions: (3.0)

a. No actuation signals present, valve shut

b. The valve is leaking significantly

c. The valve handswitch is in OPEN

d. The valve opens due to high reactor pressure

e. ADS logic actuated

QUESTION 6.03 (1.50)

Following a reactor scram, the four rod display position goes blank, but the green full-in light on the full core display for that control rod is lighted. Is this normal? If so, explain why it occurs. If not, describe the probable cause. (1.5)

QUESTION 6.04 (1.50)

What are six (6) of the eight (8) permissives that must be met in order for the emergency diesel generator breaker to close and re-energize its associated bus? (1.5)

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

QUESTION 6.05 (2.00)

The HPCI system was in a normal standby lineup when an initiation signal was received. For each of the following situations, state whether it is indicative of a malfunction or not, and explain your choice. All parameters were within their respective normal ranges when the initiation signal was received.

a. During pump startup, discharge pressure is 75 psig, flow is 0 gpm and the minimum flow valve (HPCI-MO-25) is SHUT. (0.5)

b. The GLAND SEAL CONDENSER CONDENSATE PUMP indicates NOT running. (0.5)

c. The GLAND SEAL CONDENSER BLOWER indicates NOT running. (0.5)

d. The HPCI pump suction torus valve (HPCI-MO-58) strokes open. (0.5)

QUESTION 6.06 (2.00)

Describe four (4) rod blocks associated with refueling equipment. Include required Mode Switch position(s), as applicable. (2.0)

QUESTION 6.07 (2.00)

List eight (8) indications of reactor vessel water level in the Control Room. Include on which panel each indication is located. (2.0)

QUESTION 6.08 (1.00)

a. LPRM accuracy may be affected significantly by depletion of the fission chamber. How does this depletion affect the LPRM readings (higher or lower than actual power)? (0.5)

b. A Core Thermal Power and APRM Calibration program (OD-3) is performed and shows APRM A with a Gain Adjustment Factor of 1.03. What does this tell the operator about the relationship between actual and indicated power on APRM channel A? (0.5)

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

QUESTION 7.01 (2.50)

During a reactor and plant startup, primary containment oxygen content must be ___(a)___ within ___(b)___ hours after ___(c)___. (2.5)

QUESTION 7.02 (2.00)

With the reactor operating at power, a loss of both CRD pumps occurs. What are two (2) undesirable effects of this loss and what makes these effects undesirable? (2.0)

QUESTION 7.03 (1.00)

With the reactor operating at power, an accumulator trouble alarm sounds. Investigation reveals a nitrogen leak on the accumulator. The control rod is FULLY INSERTED. Is that control rod OPERABLE? If not, how can it be made operable? (1.0)

QUESTION 7.04 (2.50)

a. The control room is filling with a noxious vapor from an undetermined source. The Shift Supervisor decides to implement the "Toxic Gas in Control Room" procedure. What IMMEDIATE ACTIONS are required of operating personnel by this procedure? (1.0)

b. The above actions are determined to be inadequate and the control room is to be evacuated. What actions should be taken, if possible, prior to leaving the control room, by the Shift Supervisor and the Control Room Operators? (1.5)

QUESTION 7.05 (1.50)

a. A gaitronics announcement is made concerning a Class Bravo fire. What is meant by a Class Bravo fire? (1.0)

b. If the fire involves radioactive material(s), an individual should not remain in the area for longer than _________ without respiratory protection. (0.5)

QUESTION 7.06 (1.00)

Boron injection into the reactor is required per EOP-1, RPV Control, and the Standby Liquid Control System is incapable of injecting into the RPV. Describe basically how boron injection to the RPV is accomplished under the above circumstances. Specific procedural steps ARE NOT required. (1.0)

QUESTION 7.07 (2.50)

Emergency Dose Exposure Limits are defined for three (3) categories in EPIP 5.7.12, Emergency Radiation Exposure Control. List the three categories AND their associated exposure limits. (2.5)

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

QUESTION 7.08 (2.00)

The plant is operating at power with both recirculation pumps operating at minimum speed. ONE pump trips and backflow is NOT established in the idle jet pumps.

a. In this condition, how will indicated core flow compare with actual core flow? (0.5)

b. In the condition above, in what region of the power/flow map will the plant appear to be operating? (0.5)

c. What action must the operator take to ensure that a correct core flow signal is used by the process computer for core parameter calculations? (0.5)

d. What action should be taken to avoid or control abnormal neutron flux oscillations (not actions to observe flux oscillations, but to control them)? (0.5)

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

QUESTION 8.01 (1.50)

Match the emergency signal (a - c) with its tone description. (1.5)

_____a. Emergency alarm	1. distinct steady tone
_____b. Fire alarm	2. one steady up and down tone
_____c. All clear	3. ringing gong
	4. distinct pulse tone

QUESTION 8.02 (1.50)

For each of the following three Government Communications Systems, state the color of the telephones, the person (by title or organization) who should establish the communication, and whether or not the system is in the Control Room.

a. Health Physics Network (HPN) (0.5)

b. Emergency Notification System (ENS) (0.5)

c. Nebraska State Patrol (NSP) Hotline (0.5)

QUESTION 8.03 (1.50)

Briefly describe how a TEMPORARY setpoint change to a Technical Specification setpoint is performed administratively. Include the appropriate documentation and approvals/concurrences required. (1.5)

QUESTION 8.04 (2.50)

a. Define a LIMITING CONTROL ROD PATTERN. (1.0)

b. It has been concluded that such an occurrence could result only from an abnormal withdrawal sequence at a high power level. List or describe three (3) operations or conditions that could result in an abnormal withdrawal sequence. (1.5)

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

QUESTION 8.05 (2.00)

Pertaining to CNS Procedure 0.9, Equipment Clearance and Release Orders:

a. List the three (3) methods given that may be used for verifying the position of manual valves in the main flow path of safety related equipment when they are returned to service. (1.5)

b. Who may sign the CLEARANCE RELEASED BY blank when the person who signed the CLEARANCE ISSUED TO blank is not on site and the tags need to be picked up? (0.5)

QUESTION 8.06 (2.00)

The Shift Supersivor shall immediately (or as soon as possible) notify the Operations Supervisor and Division Manager of Nuclear Operations if any of FOUR (4) GENERAL SITUATIONS exist. What are these four general situations? (2.0)

QUESTION 8.07 (1.50)

Identify three (3) situations or conditions that warrant using a small magnetic base red arrow on a white background as a highlighting device. (1.5)

QUESTION 8.08 (1.50)

The reactor is operating at 100% power when a turbine trip occurs. A review of the alarm printer shows the reactor scrammed on APRM high flux. Does this violate Technical Specifications? If so, JUSTIFY YOUR ANSWER. (1.5)

QUESTION 8.09 (1.00)

Once emergency conditions have been detected and classified, initial state and local notifications must be completed within ___(a)___ and initial NRC notification must be made within ___(b)___. (1.0)

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

NRC LICENSE EXAMINATION HANDOUT

EQUATIONS, CONSTANTS, AND CONVERSIONS

$\dot{Q} = \dot{m}*C_p*\text{deltaT}$ $\qquad$ $\dot{Q} = U*A*\text{deltaT}$

$P = P_o*10^{sur*(t)}$ $\qquad$ $P = P_o*e^{t/T}$ $\qquad$ $SUR = 26/T$

$T = l^*/\rho + (\beta-\rho)/\bar{\lambda}\rho$ $\qquad$ $T = l/(\rho-\beta)$ $\qquad$ $T = (\beta-\rho)/\bar{\lambda}\rho$

$\rho = (K_{eff}-1)/K_{eff} = \text{delta}K_{eff}/K_{eff}$ $\qquad$ $\rho = l^*/TK_{eff} + \bar{\beta}_{eff}/(1+\bar{\lambda}T)$

$\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$ $\qquad$ $\bar{\lambda} = 0.1 \text{ seconds}^{-1}$

$I = I_o*e^{-ux}$

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

$R/hr = 6*CE/d^2$ feet

Water Parameters

1 gallon = 8.345 lb_m = 3.87 liters
1 ft^3 = 7.48 gallons
Density @ STP = 62.4 lb_m/ft^3 = 1 gm/cm^3
Heat of vaporization = 970 Btu/lb_m
Heat of fusion = 144 Btu/lb_m
1 atmosphere = 14.7 psia = 29.9 inches Hg.

Miscellaneous Conversions

1 curie = 3.7 x 10^{10} disintegrations per second
1 kilogram = 2.21 lb_m
1 horsepower = 2.54 x 10^3 Btu/hr
1 mw = 3.41 x 10^6 Btu/hr
1 inch = 2.54 centimeters
degrees F = 9/5 degrees C + 32
degrees C = 5/9 (degrees F - 32)
1 Btu = 778 ft-lb_f

ANSWER 5.01 (1.00)

T=l-star/p + (B-p)/tp	(0.1)
so for prompt critical neglect delayed term & T=l-star/p	(0.1)
1* = 10EE-04 seconds (accept 50 to 150 microseconds)	(0.1)
p ~ .005	(0.1)
T = 10EE-4/.005 = 0.02 seconds	(0.25)
P/Po=eEEtime/T	(0.1)
P/Po=eEE50 = 5.18 X 10EE21	(0.25)

Numerical values do not have to be exact for full credit. Reasonable assumptions accepted.

REFERENCE
GE Reactor Theory Review, pg 20

ANSWER 5.02 (2.00)

a. Voids initially decrease (0.25) as the increased flow moves the boiling boundary farther into the core (0.25). As power increases, the rate of boiling increases (0.25), and the boiling boundary returns to near its original level (0.25).

b. The decrease in void content initially causes a positive reactivity addition (0.25). As power increases, negative reactivity is added due to increased void formation (0.25) and the increased fuel temperature (0.25). Net reactivity returns to 0 at steady state conditions (0.25).

REFERENCE
GE Reactor Physics Review, Figure 66

ANSWER 5.03 (2.00)

"d" is the correct answer(0.5). The highest xenon concentration will be in the center of the core(0.5), the high flux region from the previous operating period(0.5). This will increase the flux in the area of the peripheral rods(0.5) thus increasing their worth.

REFERENCE
GE Reactor Physics Review, pg 36-37

ANSWER 5.04 (3.00)

a. Reactor pressure would increase (0.5) due to the increase in reactor power caused by the increased subcooling (0.5).

b. Reactor level will increase (0.5). A level error must be generated to reestablish steady state conditions in the FWLCS (0.5).

c. Feedwater flow will decrease (0.5). The HPCI injection is providing a portion of the required feed for the reactor and this is not sensed by the Feed Flow detectors (0.5).

REFERENCE
BWR-4 Transients

ANSWER 5.05 (1.50)

- Zirconium-water reaction
- Steel-water reaction
- Radiolysis of water
  (3 at 0.5 each)

REFERENCE
Mitigating Core Damage, Gas Generation Section

ANSWER 5.06 (1.00)

- Doppler broadening (0.5)
- Self-shielding (0.5)

REFERENCE
GE Reactor Physics Review, pg 30

ANSWER 5.07 (2.00)

a. Saturated
b. Subcooled
c. Superheated
d. Subcooled
(0.5 each)

REFERENCE
Steam Tables

ANSWER 5.08 (1.00)

- height of the column of water above the eye of the pump
- amount of subcooling of the above water or reactor dome pressure
- irreversible flow losses in the suction line
(3 at 0.33 each)

REFERENCE
GE Thermodynamics, Heat Transfer and Fluid Flow, pg 7-93

ANSWER 5.09 (1.00)

High localized stresses caused by differential pellet/clad expansion (0.5) and the presence of embrittling fission product species (0.5) such as Iodine and Cadmium.

REFERENCE
GE Thermodynamics, Heat Transfer and Fluid Flow, pg 9-124

ANSWER 5.10 (.50)

a. TRUE. (0.25)
b. i. clad temperature (0.125)
ii. 2200 (0.125)

REFERENCE
GE Thermodynamics, Heat Transfer and Fluid Flow, pgs 9-71 and 9-72

ANSWER 6.01 (2.00)

a. Reactor building MG set room - Manual
Control building cable spreading room - manual
Control room entrance - manual
Turbine building switchgear room - manual
Main generator - manual
Turbine bearing #1 area - automatic
Turbine bearing #2 area - automatic
Turbine bearing #3 area - automatic
(6 areas at 0.15 each, 6 initiations at 0.1 each)

b. - Reset button on the sprinkler control and fire alarm panel in the Control Room (0.25)
- Manual pushbuttons in the NW corner and on the North wall of the turbine generator operating floor shield wall (0.25)

REFERENCE
OP 2.2.2, Carbon Dioxide System, Rev 14, pg 2,5

ANSWER 6.02 (3.00)

	RED	GREEN	BLUE
a.	off	on	on
b.	off	on	off
c.	on	off	off
d.	off	on	off
e.	on	off	off

(15 at 0.2 each)

REFERENCE
Nuclear System Pressure Relief System Description, pg 4,6,8,9

ANSWER 6.03 (1.50)

Yes, it is normal (0.5). The drive piston moves the RPIS magnet past the "00" reed switch and actuates only the green full-in "overtravel" reed switch (1.0).

REFERENCE
CRD System Description, pg 18
RMC System Text

ANSWER 6.04 (1.50)

- Emergency transformer 4.16 KV ACB open (1FS/1GS)
- Diesel generator lockout relay reset
- Diesel generator tie breaker lockout relay reset
- Bus 1F (1G) tie breaker 1AF (1BG) OR Bus 1A (1B) tie breaker 1FA (1GB) open
- Diesel generator at rated speed
- Diesel generator at rated voltage (94 %)
- Sustained loss of critical bus voltage > 5 seconds
- Diesel Control Isolation Switches in REMOTE
- DC Control Power and Maintenance Lockout Keylock in "ON"
- D/G bkr Selector Switch in AUTO

(6 required at 0.25 each)

REFERENCE
SOP 2.2.20, Standby AC Power System (D/G), Rev 19, pg 25

ANSWER 6.05 (2.00)

a. Malfunction. The minimum flow valve should open when the initiation signal is received. Any explanation that demonstrates this fact is acceptable (0.5).

b. This is not indicative of a problem. The condensate pump cycles on level in the hotwell and is independent of the HPCI initiation signal (0.5).

c. Malfunction. The blower should start on the initiation signal and run continuously (0.5).

d. Malfunction. The torus suction valve should not receive an open signal due to the HPCI initiation (0.5).

REFERENCE
SOP 2.2.33, HPCI, Rev 26, pg 4-7

ANSWERS -- COOPER -85/09/26-GRAVES, D.

ANSWER 6.06 (2.00)

ROD BLOCK	MODE SWITCH POSITION
- Service platform jib crane loaded	Startup/Hot Standby or Refuel
- Refueling platform over core	Startup/Hot Standby
- Refueling platfrom over core with any of its 3 hoists loaded	Refuel
- Refueling platform over core with fuel grapple not fully up	Refuel

(0.25 for each block, 0.25 for each position)

REFERENCE
Instrumentation Operating Procedure 4.3, Reactor Manual Control System, Rev 10, pg 6, 7

ANSWER 6.07 (2.00)

- Post-accident monitor A or LI-91A. Located on panel 9-3
- Post-accident monitor B or LI-91B. Located on panel 9-3
- Wide range for vessel flooding or LI-86. Located on panel 9-4
- ECCS Yarway A or LI-85A. Located on panel 9-5
- ECCS Yarway B or LI-85B. Located on panel 9-5
- GEMAC feedwater control A or LI-94A. Located on panel 9-5
- GEMAC feedwater control B or LI-94B. Located on panel 9-5
- GEMAC feedwater control C or LI-94C. Located on panel 9-5
- Narrow Range Recorder or LR-6-97. Located on panel 9-5
- Wide range during shutdown for core coverage or LR-6-98. Located on panel 9-5

(8 indications required at .125 each, locations at .125 each)

NOTE: LI-91A & B: Fuel Zone, Rosemount, Wide Range, Wide Range Yarways
LI-85A & B: Narrow Range Yarways
LI-94A, B, & C: GEMAC, Narrow Range GEMAC
These are alternate names for the above level indicators.

REFERENCE
Instrumentation Operating Procedure 4.6.1, Reactor Vessel Water Level Indication, Rev 9, pg 2

ANSWER 6.08 (1.00)

a. Readings will decrease as the detector ages (0.5).

b. The APRM reading is lower than actual power (0.5).

REFERENCE
Nuclear Performance Procedure 10.5, LPRM Calibration, Rev 17, pg 1
Nuclear Performance Procedure 10.1, APRM Calibration, Rev 14, pg 4

ANSWER 7.01 (2.50)

a. < 4%
b. 24
c. going to the RUN Mode
(0.833 each)

REFERENCE
General Operating Procedure 2.1.1, Cold Startup Procedure, Rev 39, pg 11

ANSWER 7.02 (2.00)

The loss of the CRD pumps causes a loss of cooling water flow (0.5), which shortens CRD seal life (0.5), and also allows the CRD hydraulic accumulators to slowly depressurize (0.5) which reduces the scram capability of the reactor (0.5), especially at low pressures. Also accept loss of seal purge to the RWCU and RR pumps which can shorten seal life, and loss of ability to move control rods with RMCS as hydraulic pressure is lost.

REFERENCE
Abnormal Procedure 2.4.1.1.4, Loss of CRD Pump, Rev 5, pg 2
System Procedure 2.2.8, Control Rod Drive System, pg 2

ANSWER 7.03 (1.00)

The control rod is INOPERABLE (0.5). To make it operable, the leak must be repaired and the accumulator trouble cleared (0.5)

REFERENCE
CNS Technical Specification 3.3.A.2.e

ANSWER 7.04 (2.50)

a. - Start the Control Room Ventilation Booster Fan BF-C-1A (0.33)
   - Secure the Control Room Ventilation Supply Fans SF-C-1A & B (0.33)
   - Essential Control Room personnel obtain self contained breathing apparatus and use as necessary (0.33)

b. - Announce the event over the station paging system (0.25)
   - Direct operations personnel to assemble in some designated area area (0.25)
   - Scram the reactor (0.25)
   - Verify all control rods inserted (0.25)
   - Trip the main turbine (0.25)
   - Ensure the reactor feed pump turning gear control switches are in Auto (0.25)

REFERENCE
Abnormal Procedure 2.4.8.5, Toxic Gas in Control Room, Rev 2, pg 1
Emergency Procedure 5.2.1, Shutdown from Outside the Control Room, Rev 13, pg 1

ANSWER 7.05 (1.50)

a. Flammable or combustible liquids, flammable gases, greases, and similar materials (1.0).

b. 2 minutes (0.5)

REFERENCE
Emergency Procedure 5.4.1, General Fire Procedure, Rev 17, pg 1, 5

ANSWER 7.06 (1.00)

Filling the RWCU demineralizers with borated water and injecting this via the RWCU system (1.0).

REFERENCE
Emergency Procedure 5.2.14, Alternate Means to Inject Boron to RPV, Rev 0

ANSWER 7.07 (2.50)

- Sampling under accident conditions (0.33): 5 REM (0.5)
- Corrective or protective actions (0.33): 25 REM (0.5)
- Life saving actions (0.33): 75 REM (0.5)

REFERENCE
Emergency Plan Implementing Procedure 5.7.12, Emergency Radiation Exposure Control, Rev 5, pg 1, Attachment "A"

ANSWER 7.08 (2.00)

a. Indicated core flow will be lower than actual (0.5)

b. To the left of the natural circulation line (0.5)

c. The operator must enter a substitute value for core flow equal to the natural circulation flow for the existing power level (0.5).

d. Insert control rods until power is < or = 75% load line (0.5).

REFERENCE
General Operating Procedure 2.1.10, Station Power Changes, Rev 9, Attachment "A"
General Operating Procedure 2.1.15, Reactor Recirculation Pump Startup and Shutdown, Rev 13, pg 3, 4
Abnormal Procedure 2.4.2.2.1, Trip of Reactor Recirculation Pumps, Rev 12, pg 1-3

ANSWER 8.01 (1.50)

a. 1
b. 4
c. 2
(0.5 each)

REFERENCE
SOP 2.2.4, Communications, Rev 13, pg 6

ANSWER 8.02 (1.50)

a. - beige (0.1)
   - NRC (0.2)
   - NOT in Control Room (0.2)

b. - red (0.1)
   - DMNO, STA, SS, communicator (any of the four acceptable) (0.2)
   - IN Control Room (0.2)

c. - green (0.1)
   - Emergency Director (or communicator) (0.2)
   - IN Control Room (0.2)

REFERENCE
SOP 2.2.4, Communications, Rev 14, pg 3, 11-13
Procedure 2.0.5, Steps II.B.1.d, pg 1 and Step III.E, pg 5

ANSWER 8.03 (1.50)

A Setpoint Change Request form (0.25) marked TEMPORARY (0.25) will be filled out and signed by 2 SRO's (1.0).

REFERENCE
Instrumentation Operating Procedure 4.0.1, Instrument Setpoint Control, Rev 0, pg 3

ANSWER 8.04 (2.50)

a. One which contains a rod which, if completely withdrawn (0.25), could result in a MCPR of less than 1.07 (0.75).

b. - Interchange of normal control rod patterns (0.5)
   - Establishment of special control rod patterns as an aid for identifying core regions having failed fuel assemblies (0.5)
   - Establishment of special control rod patterns resulting from control rod drive system malfunctions (0.5)

REFERENCE
Nuclear Performance Procedure 10.10, Limiting Control Rod Pattern Determination, Rev 7, pg 1

ANSWER 8.05 (2.00)

a. 1. Position light indication in the control room if the valve is so equipped (0.5)
   2. Locally by a second operator (0.5)
   3. Satisfactory performance of a flow operability surveillance (0.5)

b. One of the individual's supervisors (0.5)

REFERENCE
CNS Procedure 0.9, Equipment Clearance and Release Orders, Rev 2, pg 2, 3

ANSWER 8.06 (2.00)

- Entry into a limiting condition for operation as required by Technical Specifications (0.5)
- Any plant condition which requires any of the Emergency, Abnormal, or Emergency Operating Procedure to be implemented (0.5)
- Any continuing off-normal condition which limits plant power capability or could limit power production if corrective action was not taken (0.5)
- A condition for entry into the Emergency Plan Implementing Procedures (0.5)

REFERENCE
Conduct of Operations Procedure 2.0.1, Operations Department Policy, Rev 2, pg 2

ANSWER 8.07 (1.50)

- Any time a control is placed in an out of normal position
- Annunciators and other indications that may be out of service
- Annunciators or other indications that may require particular attention
- Nuisance alarms

(3 required at 0.5 each)

REFERENCE
Conduct of Operations Procedure 2.0.3, Control Room Conduct and Manning, Rev 0, pg 2

ANSWER 8.08 (1.50)

Yes (0.5). A safety limit shall be assumed to be exceeded when a scram is accomplished by a menas other than the expected scram signal (1.0). The scram should have occurred due to the Turbine Stop Valve Closure.

REFERENCE
CNS Technical Specifications 1.1.C, pg 6, 12, 13

ANSWER 8.09 (1.00)

a. 15 minutes (0.5)
b. 1 hour (0.5)

REFERENCE
EPIP 5.7.6, Notification, Rev 5, pg 1