U. S. NUCLEAR REGULATORY COMMISSION REGION I
OPERATOR LICENSING EXAMINATION REPORT

EXAMINATION REPORT NO. 85-03 (OL)

FACILITY DOCKET NO. 50-220

FACILITY LICENSE NO. DPR-63

LICENSEE: Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

FACILITY: Nine Mile Point, Unit 1

EXAMINATION DATES: March 11 - 14, 1985

PREPARED BY: J. A. Berry, Lead Reactor Engineer (Examiner) — Date: 4-17-85

REVIEWED BY: R. M. Keller, Chief, Projects Section 1C — Date: 4-17-85

APPROVED BY: H. B. Kister, Chief, Projects Branch No. 1 — Date: 4/18/85

SUMMARY: As part of the NRC's programmatic evaluation of Requalification Training at Nine Mile Point, Unit 1, NRC prepared written examinations were administered, in parts, to all facility personnel taking the Niagara Mohawk prepared annual requalification examinations the week of March 11, 1985. Additionally, oral requalification examinations were given to 11 licensed personnel, 7 SROs and 4 ROs.

REPORT DETAILS

TYPE OF EXAMS: Requalification

EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail
Written Exam Partial Exams	35/4	32/1
Oral Exam	4/0	6/1
Overall	35/4	32/1

1. CHIEF EXAMINER AT SITE:

   J. A. Berry, U.S. NRC - Region I

2. OTHER EXAMINERS:

   D. J. Lange, U.S. NRC - Region I
   F. J. Crescenzo, U.S. NRC - Region I
   T. L. Morgan, EG&G Idaho, Inc.
   D. E. Hill, EG&G Idaho, Inc.

3. REPORT:

   As part of the NRC's programmatic evaluation of Requalification Training at Nine Mile Point - Unit 1, NRC prepared written examinations were administered in parts, to all facility personnel taking the Niagara Mohawk prepared annual Requalification examinations the week of March 11, 1985. Additionally, oral requalification examinations were given to 11 licensed personnel, 7 SROs and 4 ROs.

The NRC written examination sections were administered as follows:

Monday, March 11 - RO Section 2 to 17 people
- SRO Sections 5 & 8 to 11 people

Tuesday, March 12 - RO Section 3 to 13 people
- SRO Section 6 to 12 people

Wednesday, March 13 - RO Sections 1 & 4 to 9 people
SRO Section 7 to 11 people

Overall, examination results were good. Five people failed NRC administered sections of the examinations, four RO's and one SRO, and one SRO failed the oral examination.

The comparison of scores on NRC sections vs. the facility sections indicated that the overall average score on the NRC exam (if sections were together) and facility exam were within 4% of each other. This is considered an acceptable range. Individual section comparisons indicated a wide disparity. Section 8 of the NRC and Facility exams were within .5% of each other in average score, but Sections 2, 3, and 6 were off by 6.71%, 9.91% and 6.56% respectively, with the NRC section scores being lower. Also, Sections 4 and 7 on the NRC exam had average scores 6.4% and 3.3% higher than the facility's sections.

The reasons for this disparity are not evident. It appears that the higher scores on the NRC Sections 4 and 7 may be due to the facility's sections being overly long, but the other section differences cannot be so explained. Probable causes may be the tension involved in taking an NRC exam, more operationally oriented (not memorization) type questions on the NRC exam, or the difference in question "style" between the two exams.

In addition to the conduct of examinations, the evaluation also consisted of a review of the NMP-1 Requalification Program Annual examinations prepared by the facility, and discussions with licensed operators and training staff members regarding the Requalification Program.

The Annual Requalification examinations prepared by the facility were considered to meet NRC requirements, but were not considered to be of high quality. Problems with the examinations included; double jeopardy questions, excessive length, many unnecessary theory calculations and questions having no relation to an operator's job, and simplistic short answer questions which failed to provide an adequate measure of depth of knowledge. The facility's Requalification examination question bank is poor, and it is felt its use contributed to the problems with the examination. To Niagara Mohawk's credit, they have identified the problems with the existing exam question bank, and have begun a task to upgrade it. Significant improvement is expected in next years exam.

Discussions with licensed operators indicate that there is dissatisfaction with the Requalification Program. Problems sited included; to much emphasis on theory that is not operationally oriented, too much self-study or reading, and unchallenging and uninteresting presentation of subject matter. These matters have been previously brought to the attention of Niagara Mohawk management by other reviews of the program. Niagara Mohawk has committed to a course of corrective action. NRC Region I will monitor the progress of the action over the next year. It is felt that the addition of a plant specific simulator training program to the Requal program will aid in improving the program.

Overall, the Nine Mile Point, Unit 1 Requalification program is satisfactory. NMPC has already begun to correct many of the programmatic problems identified. No further NRC involvement in the program is planned this year, other than monitoring of the changes being made to improve its quality.

4. Personnel Present at Exit Interview:

NRC Personnel

J. Linville, Chief, Reactor Projects Section 2C, DRP
J. A. Berry, Lead Reactor Engineer (Examiner) DRP
D. J. Lange, Reactor Engineer (Examiner), DRP
F. J. Crescerzo, Reactor Engineer (Examiner), DRP
A. J. Luptak, Resident Inspector, NMP-1

NRC Contractor Personnel

D. E. Hill, EG&G Idaho, Inc.
T. L. Morgan, EG&G Idaho, Inc.

Facility Personnel

T. W. Roman, Station Superintendent - NMP-1
K. F. Zollitsch, Training Superintendent, Niagara Mohawk
J. C. Aldrich, Operations Supervisor, NMP-1
T. Wood, Training Supervisor, NMP-1
J. T. Pavel, Asst. Training Superintendent, Niagara Mohawk
R. Seifried, Operations Training Instructor
M. Dooley, Operations Training Instructor
M. Jones, Operation Supervisor, NMP-2

5. Summary of Comments made at exit interview:

- The Chief Examiner noted that there was one person who was not a clear pass on the oral examinations.

- A discussion was held regarding Niagara Mohawk's commitment to implementation of upgrades in their Requalification Program based on previous audits.

Attachments: Written Examination(s) and Answer Key(s) (SRO/RO)

U.S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR REQUALIFICATION EXAMINATION

FACILITY: NINE MILE POINT

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/03/11

EXAMINER: BERRY, J.

NAME: _____

INSTRUCTIONS

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%

CATEGORY VALUE	% OF TOTAL	SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	100.00			4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	100.00			TOTALS

FINAL GRADE _____%

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

_____
SIGNATURE

QUESTION 4.01 (3.00)

In accordance with procedure NI-SOP-32, Failure of Reactor to Scram, what six (6) immediate actions would you take to reduce power and insert all control rods in an ATWS situation? (3.0)

QUESTION 4.02 (2.00)

Describe, in general, the four things you would do to reset a high pressure coolant injection (HPCI) initiation, assuming that the initiation signal has cleared. (2.0)

QUESTION 4.03 (2.00)

a. Why is an operator instructed to "reduce reactor power to 80% of the original power level with Reactor Recirculation flow" BEFORE removing a feedwater heater string? (1.0)

b. When two condensate booster pumps are required, the preferred linup is with #11 and #13 running; when one booster pump is required, #11 or #13 should be in service. Why is this preferred? (1.0)

QUESTION 4.04 (3.00)

a. Assuming the CSO and the NAOE were able to accomplish NOTHING in the way of securing the station prior to an evacuation, how is the reactor shutdown AND how is the shutdown verified? Your answer should include where the CSO and NAOE proceed to and their subsequent actions. (1.0)

b. After verification of a turbine trip, the SSS is to proceed to powerboard 11 & 12. What actions are to be performed at powerboard 11 & 12? (1.0)

c. How can RAW WATER be supplied to feed the reactor? (1.0)

QUESTION 4.05 (3.00)

Concerning Procedure SOP-19 (Unexplained Reactivity Change);

a. List six (6) plant parameters/indications that should be checked if an unexplained reactivity change should occur at rated power. (1.5)

b. Depending on the magnitude of the reactivity change, list three alarms that may be initiated. (PRIOR to a reactor scram) (0.75)

c. If this reactivity change is a result of decreased temperature, due to a loss of a feedwater heater string, what is your immediate action and what two (2) adverse conditions are you trying to protect against? (.75)

QUESTION 4.06 (3.00)

Concerning Procedure N1-OP-14, Containment Spray System;

a. What two (2) signals are required to automatically start the containment spray pumps? (0.5)

b. What action should be taken following a confirmed high radiation alarm in the containment spray raw water system? (0.5)

c. The containment spray Raw Water Pumps must be manually started by the control room operator? TRUE or FALSE. (0.25)

d. This procedure directs you not to manually override or shut this system down after an auto. initiation unless two conditions are met. What are these two conditions and who is authorized to make this decision ? (1.25)

QUESTION 4.07 (2.00)

During the 4:00 pm to 12:00 midnight shift, at rated power, you receive two alarms:

1. Off GAS line high pressure.
2. Off GAS line high temperature

You notice that the condenser vacuum is decreasing.

a. Based on the above indications/conditions, WHAT HAS OCCURED?, and what additional automatic actions can be expected? (1.0)

b. Based on the above situation list your immediate operator actions. (1.0)

QUESTION 4.08 (2.50)

Concerning N1-SOP-29, Pipe Break Inside Drywell;

a. Under what conditions can the automatic controls of an Emergency Core Cooling System be placed in its manual mode? (be specific) (1.0)

b. Think about the overall purpose of this procedure; ---> List at least three (3) operational functions, with respect to the Core and its Containment, you are expected to achieve to assure that the HEALTH and SAFETY of the public is protected. (1.5)

QUESTION 4.09 (2.50)

According to N1-SOP-3, Feedwater Malfunction (Decreasing FW Flow);

a. What immediate actions would you take if feedwater flow rapidly decreased due to a loss of the Shaft Feedwater Pump? (1.5)

b. Due to the above transient RX. Vessel level is decreasing at a very rapid rate. As the Shift Supervisor, at what Vessel level would you direct your operators to depressurize the vessel? (0.25)

c. Is it necessary to close the MSIVs during this transient? (0.25)

d. List three (3) conditions that could cause HPCI to automatically initiate as a result of this transient. (0.5)

QUESTION 4.10 (2.00)

Concerning Procedure SOP-15, Malfunction of the Control Rod Drive System;

a. During a power ascension,(RX. power approx. 30 %), the selected control rod starts to drift. What Automatic responses, ie; alarms/indications, would be affected? (1.0)

b. What criterion is used to define a control rod as being inoperable? (0.5)

c. How could you verify that a control rod has become uncoupled? (0.5)

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 4.01 (3.00)

1. Place Mode Switch in shutdown (This inserts an additional scram signal)
2. Trip recirculation pumps
3. Fully insert control rods using 'Emergency Rod In'
4. Reset RPS trip. Manually scram the reactor
5. Individually scram rods from 'M' panel
6. Isolate and vent scram air header locally (0.33 each)

REFERENCE
NI-SOP-32, Rev. 4, pg 8 JCK-174

ANSWER 4.02 (2.00)

1. Verify a) Feedwater flow on #11 and #12 is > 1.9 million lbm/hr.
   b) Reactor low level trip is clear (.5)
2. Switch feedwater pump #11 and #12 M/A stations to manual (.5)
3. Adjust the manual outputs until the deviation meters on the #11 #12 M/A stations are nulled. (.5)
4. Press the 'Feedwater Return to Normal After HPCI' pushbutton on the reactor control console. (.5)

REFERENCE
NI-OP-16, pg. 19 EDH-324

ANSWER 4.03 (2.00)

a. This will prevent the other feedwater heater strings from being overloaded and will preclude possible over-power of the nuclear fuel. Also power increase due to increased inlet subcooling. (1.0)

b. This preferred lineup will preclude a system feedwater disturbance due to the loss of powerboard #101. ~~Also insures HPCI availability~~. (1.0)

REFERENCE
NI-OP-16, pg. 19 EDH-317

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 4.04 (3.00)

a. The CSO proceeds to shutdown panel #12 and trips MG set 141. The NAOE proceeds to shutdown panel #11 and trips MG set 131. Verification is the 'All Rods In' white light on their respective panels. (1.0)

b. Verifies that a condensate and feedwater booster pump are operating and starts feedwater pump #11, if HPCI has failed to initiate(1.0)
Also manual transfer of PB-11&12 if auto transfer fails

c. By installing an available spool piece between the feedwater system and the fire protection water system. (1.0)
Also cross-connect to containment spray raw water through intertie valves to core spray

REFERENCE
N1-SOP-11, pg. 3-5 EDH-318

ANSWER 4.05 (3.00)

a. 1. Control Rod position.
2. Recirculation flow.
3. Reactor pressure.
4. Steam flow or temp.
5. Feedwater flow or temp.
6. Turbine Generator load.
7. Bypass,relief or safety valve flow.

(any six at 0.25 for each correct ans.)

b. 1. LPRM alarm . , 2. APRM alarm. , 3. Rod Block alarm. (0.25 each)

c. Reduce power to 80 % of the power level prior to the change using Recirc flow. (0.25) This will prevent bundle overpower (0.25) and overloading of the other feedwater heater strings. (0.25)

REFERENCE
NMP. N1-SOP-19 (pg,2&3)

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 4.06 (3.00)

a. A combination of lo-lo reactor vessel water level and high drywell pressure (3.5 psig.) (0.50)

b. The raw water pump and the containment spray pump in the affected loop should be secured. (0.25)
The loop suction and discharge valves should be closed. (0.25)

c. TRUE . (0.25)

d. 1. Sufficient evidence shows that the system is not performing its intended function. (0.50)
2. Continued operation will prolong or produce an unsafe condition. (0.50)
Shutdown of the system will be at the direction of the Station Shift Supv. (0.25)

REFERENCE
NMP. N1-OP-14, pages 1 thru 5 .

ANSWER 4.07 (2.00)

a. EXPLOSION in the Air Ejector discharge piping. (0.50)
Automatic Actions:
1.Valves BV-76-12/13 close and off gas flow goes to zero. (0.50)
2.Reactor Scram at 23 " Hg. (0.25)

b.
1.Reduce reactor load by decreasing recirc. flow.
2.Close main steam supply valve to air ejectors and mixing jet.
3.Insert control rods per rod pattern until vacuum decreases to near scram point.
4.Manually scram the reactor.
5. Initiate emergency condensers, as necessary to remove the decay heat.
6. Inform station personnel of conditions.
7. Notify Plant Superintendent.
(7 correct ans. at 0.1785 ea.)

REFERENCE
NMP. SOP- 18 ,"Explosion in the Air Ejection Disch. Piping"
Symptoms / Automatic Actions / Operator Actions

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 4.08 (2.50)

a. 1. Misoperation in automatic is confirmed by at least two independent process parameter indications. (0.50)
   2. Core cooling is assured AND this procedure,(SOP-29),directs you.(0.50

b. 1. Maintain Core Cooling.
   2. Limit the release of off-gas radiation.
   3. Place the Reactor Core and Containment in a SAFE STABLE condition.
   4. Keep the Torus bulk temp. within specified SAFETY limits.
   (any three at 0.50 each)

NOTE; Other specific Operational objectives related to SAFETY accepted.

REFERENCE
NMP. N1-SOP-32, Pipe Break Inside Drywell., Cautions, Limitations and over-all purpose and objectives. pg# 1-5 .

ANSWER 4.09 (2.50)

a. 1. Shift mode switch to refuel.
   2. Check all rods are fully inserted.
   3. Observe power level decreasing.
   4. Check for HPCI operation. Ensure that both motor driven feedwater pumps are running.
   5. Check that the emergency condensers are in operation.
   6. Check that the Core Spray pumps are running and recirculating back to the torus.
   (0.25 for each correct answer)

b. Low-Low-Low Level (- 10 inches) (0.50)

c. Yes, To conserve coolant inventory. (0.40)

d. 1. Runout flow of 1.9 x 10(6) or 3800 gpm. (0.20)
   2. Turbine Trip (0.20)
   3. Low Rx. water level (0.20)

REFERENCE
NMP. SOP-3, and Simulator scenerio Objectives # 1&2 .

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 4.10 (2.00)

a. 1. LPRM output in the vicinity of the drifting rod.
   2. Position indication for that rod.
   NOTE: The ROD DRIFT alarm will not come in due to the rod that is drifting is the one selected.
   The RWM alarm will not come in due to being > 25 % power. (1.00)

b. A control rod which cannot be moved with control rod drive pressure.(.5)

c. Control Rod overtravel alarm; (Ann. window F2-6 , Computer point B0-12) when the control rod has been fully withdrawn. (0.50)

REFERENCE → Nuclear Inst response
NMP., SOP-15, Malfunction of CRD system.

U.S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR REQUALIFICATION EXAMINATION

FACILITY: NINE MILE POINT

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/03/11

EXAMINER: BERRY, J.

NAME: _____

INSTRUCTIONS

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%
least 80%. Examination pape

CATEGORY VALUE	% OF TOTAL	SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	100.00			3. INSTRUMENTS AND CONTROLS
25.00	100.00			TOTALS

FINAL GRADE _____%

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

_____
SIGNATURE

QUESTION 3.01 (3.00)

a. What are two signals (including setpoints) that would cause the Feed System to shift to the HPCI Mode of operation and HOW would the results of a LOCA initiate each? (1.0)

B. When the Feed Pump is idle or not in the HPCI mode, the HPCI controller setpoint is blocked. This setpoint is not applied to the controller until ____?____ is produced at the ____?____. (1.0)

C. Why is the HPCI controller setpoint blocked initially? (1.0)

QUESTION 3.02 (1.50)

For each of the following systems, list what TYPE of radiation detector is used and what AUTOMATIC ACTIONS occur when the monitors trip. (NOTE: If no auto actions occur, indicate so. Assume lineups are appropriate for auto actions to occur.)

A. Air ejector offgas
B. RBCLC
C. Refueling Bridge (1.5)

QUESTION 3.03 (2.50)

Concerning the fuel zone level detectors:

a. Under what conditions does the value displayed on the fuel zone instrument come from the FUEL ZONE level transmitter? (Include system initiation signal(s) in your answer) (2.0)

b. What indication does the control room operator have that reference leg flashing is occuring in the fuel zone level instrument? (0.5)

QUESTION 3.04 (3.00)

Assume the REACTOR LEVEL CONTROL SYSTEM is being operated in 3-element control using reactor level detector channel '11'. Reactor power is at 85%, STEADY STATE.
For each of the instrument or control signal failures listed below, STATE HOW REACTOR LEVEL WILL INITIALLY RESPOND (increase, decrease, or remains constant) and BRIEFLY EXPLAIN WHY in terms of what is happening in the Level Control System immediately following the failure.

(FOR EXAMPLE, your answers should include the following detail, "Causes reactor level to decrease due to a steam flow/feed flow error signal, steam flow < feed flow, resulting in a closure signal to the feedwater control valve.")

NOTE: A block diagram of the Feedwater Control System is on the following page for your use.

a. #12 FEEDWATER FLOW transmitter FAILS HIGH (1.0)

b. Channel '11' REACTOR LEVEL detector signal FAILS LOW. (1.0)

c. LOSS OF CONTROL SIGNAL to #13 FEEDWATER CONTROL VALVE. (1.0)

QUESTION 3.05 (3.00)

For each of the following, state whether a ROD BLOCK, HALF-SCRAM, FULL SCRAM, or NO PROTECTIVE ACTION is generated for that condition. NOTE: IF two or more actions are generated, i.e. rod block and a half-scram, state the most severe, i.e. half-scram. Assume NO operator actions.

a. APRM 11 Downscale, mode Switch in RUN (0.6)
b. <4 LPRM inputs to APRM 15, Mode Switch in STARTUP (0.6)
c. Both Flow Conv. Units Upscale (>107% flow), Mode Switch in RUN (0.6)
d. APRM 12 and 16 Upscale, Mode Switch in STARTUP (0.6)
e. Main Steam Line 111 ISOLATED, Mode Switch in RUN (0.6)

QUESTION 3.06 (2.00)

Describe fully how Reactor Building Closed Loop Cooling(RBCLC) temperature is regulated as the heat load on the system increases. (2.0)

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

FACILITY: NINE MILE POINT

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/03/12

EXAMINER: LANGE, D.

NAME: MASTER

INSTRUCTIONS:

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	100.00			6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.00	100.00			TOTALS

FINAL GRADE _____%

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

_____
SIGNATURE

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

FACILITY: NINE MILE POINT

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/03/12

EXAMINER: LANGE, D.

NAME: _____

INSTRUCTIONS:

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	100.00			6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.00	100.00			TOTALS

FINAL GRADE _____%

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

_____
SIGNATURE

QUESTION 6.01 (3.00)

A. What conditions must be met to satisfy the logic for ADS initiation ?
NOTE: (Include setpoints and trip logic arrangement) (2.00)

B. How would the system respond if MSERV #1 failed to open after proper logic actuation ? (all other valves respond properly) (0.50)

C. What are two types of detectors used to provide positive indication of a leaking / lifted RELIEF VALVE ? (exclude lights and annunciators) (0.50)

QUESTION 6.02 (2.50)

a. List six operational conditions that will cause an automatic closure of the Main Steam Isolation Valves. (include setpoints and bypasses) (1.50)

b. List three (3) functions of the Main Steam Line Flow Restrictors. (1.00)

QUESTION 6.03 (3.00)

Concerning the High Pressure cooling injection system (HPCI) :

a. What prevents an idle feedwater pump from starting and pumping water through a FULLY OPEN feedwater control valve following a HPCI initiation signal ? (0.75)

b. If a HPCI initiation occurs with NO LOSS of OFF-SITE POWER, state the effect on the following pumps/valves or components.
1. Condensate and feedwater pumps that are running. (0.25)
2. Idle feedwater pump. (0.25)
3. Feedwater control system. (0.25)
4. Feedwater pump controller # 11 (0.25)
5. Feedwater pump controller # 12 (0.25)

c. In addition to a HPCI initiation being blocked by protective pump lockouts, list three (3) additional INTERLOCKS that will also PREVENT an automatic start. (1.00)

QUESTION 6.04 (1.50)

Concerning the Traversing In Core Probe Sys. (TIP) :
List two (2) specific areas of information that are obtained from signals generated by the TIP system . Be sure to include how the signals are being used and what information is being obtained. (1.50)

QUESTION 6.05 (2.00)

How is the integrity of ECCS piping inside the reactor vessel verified during normal operation ? In your answer include, SENSING POINTS, SPECIFIC SYSTEM(s) WHOSE PIPING IS BEING VERIFIED, WHY IT IS VERIFIED and the response of the instrumentation to a loss of integrity . (2.00)

QUESTION 6.06 (2.25)

Concerning the Generator Stator Cooling Water system :

a. What three (3) conditions will cause a Turbine Governor Runback ? (0.75)
(SETPOINTS ARE REQUIRED)

b. Will an automatic Reactor Scram occur upon receipt of a Governor runback trip signal? If yes, from what ? If not, how could a subsequent scram be prevented ? (0.75)

c. What is the importance of regulating flow within this system to maintain pressure between 22-28 psi ? (0.75)

QUESTION 6.07 (3.00)

Concerning Reactor Vessel Level Instrumentation :

a. Using the attached figure , 2-1,(Rx.Level Inst.), indicate what control room level instruments are used to measure level parameters 1,2,3,4 & 5. Indicate what level parameter items 6,7,8,9,10,11, & 12 signify ? (1.20)

b. What plant condition(s) will automatically initiate the fuel zone level detectors ? (0.50)

c. What indications would you use to verify reference leg flashing in the fuel zone level detectors ? (be specific) (0.70)

d. What three (3) plant operation variables are used for compensation by the Fuel Zone Level Indicators ? (0.60)

QUESTION 6.08 (2.75)

Concerning the Standby Liquid Control Sys :

a. Once the SBLC sys. has initiated, what six (6) CONTROL ROOM indications could you use to verify that the system is operating properly AND injecting into the reactor vessel ? (1.50)

b. After initiation of the SBLC. sys.,is it permissible to shut the system down ? (If not, WHY ? If so, under what conditions ?) . (1.25)

QUESTION 6.09 (2.25)

Concerning the CORE SPRAY system ;

a. What protective design feature, within the core spray system, allows for running the core spray pumps at shutoff head without overheating them ? (Explain fully- be specific) (0.75)

b. Pressure in the Core Spray piping is sensed in three different places. List these three sensing points, indicating what is being sensed and any automatic actions , alarms or indications that are provided from them. (~~SETPOINTS ARE REQUIRED~~) (1.50)

QUESTION 6.10 (2.75)

According to procedure # N1-SOP-5, (Instrument Air Failure):
During your shift you experience a complete loss of instrument air header pressure. For the following air operated valves, indicate the valve action (open or closed) and the station response(if any). (2.75)

1. Main Steam Isolation Valves.
2. Feedwater Flow Control Valve.
3. Make-up valve to condenser.
4. Reactor Bldg. CLC- TCV .

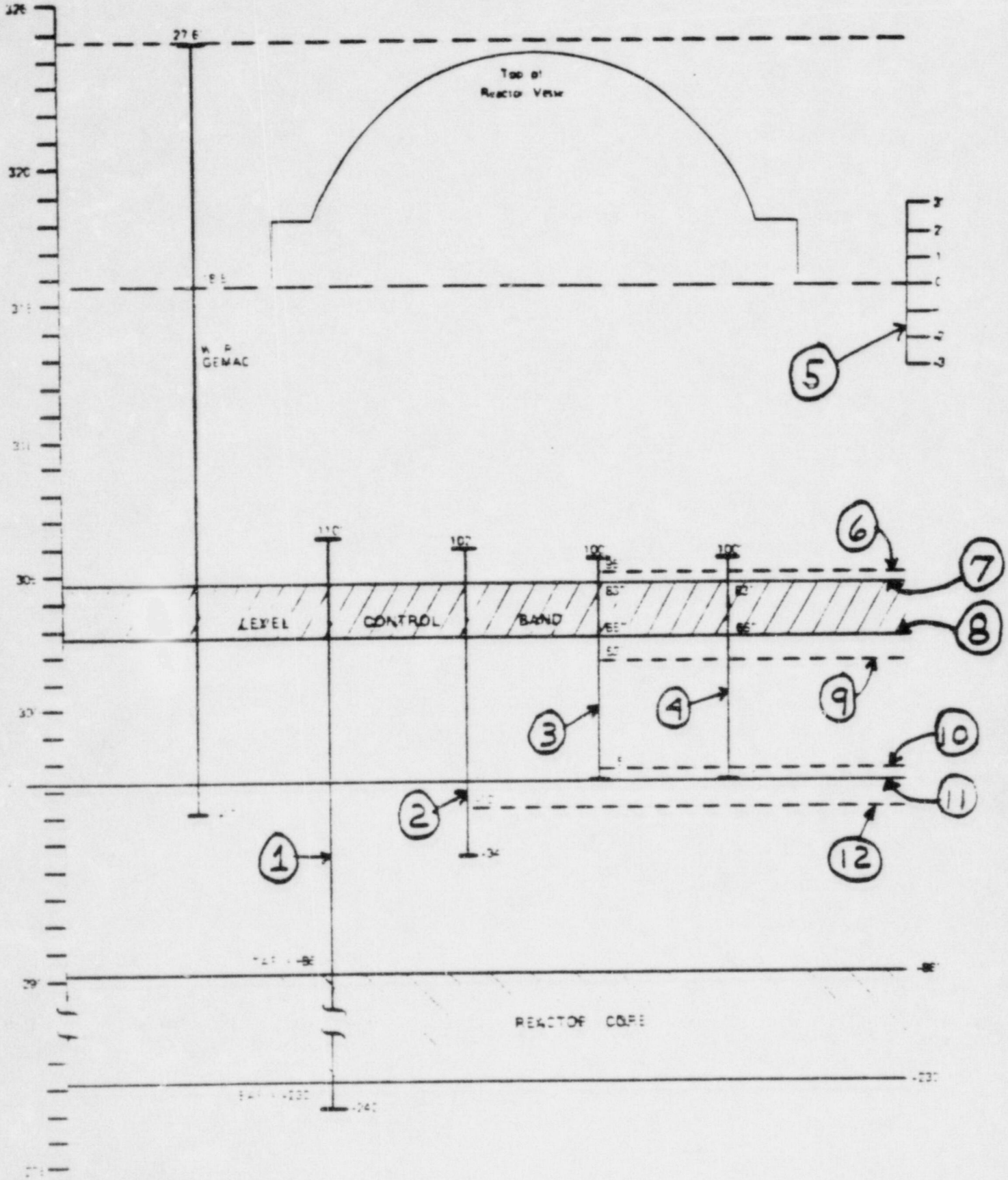

Figure II 2 1
Reactor Vessel Level Instrumentation

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

FACILITY: NINE MILE POINT

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/03/12

EXAMINER: LANGE, D.

NAME: MASTER

INSTRUCTIONS:

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	100.00			7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	100.00			TOTALS

FINAL GRADE _____%

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

_____
SIGNATURE

QUESTION 7.01 (2.50)

Concerning Procedure N1-OP-30, (4.16 Kv, 600 V, and 480 V House Service);

a. If low voltage is sensed on PB 11 or PB 12, its respective supply breaker will open. Based on this, at what point will the reserve supply breaker close ?(be specific as to value and reason why). (1.00)

b. What is the normal,(open or closed) position and why, of feeder breakers to PB-101 (R-1014/1011 ? What is the interlock function of these breakers and how can this interlock be defeated ? (1.00)

c. What interlock exists between the feeder and tie breakers on PB 16 & PB 17 . (0.50)

QUESTION 7.02 (3.00)

Concerning procedure SOP-19 (Unexplained Reactivity Change) ;

a. List six (6) plant parameters/indications that should be checked if an unexplained reactivity change should occur at rated power. (1.50)

b. Depending on the magnitude of the reactivity change list three alarms that may be initiated. (prior to a reactor scram) (0.75)

c. If this reactivity change is a result of decreased temperature, due to a loss of a feedwater heater string , what is your immediate action and what two (2) adverse conditions are you trying to protect against?(.75)

QUESTION 7.03 (2.50)

Concerning procedure N1-OP-14 , Containment Spray System :

a. What two (2) signals are required to automatically start the containment spray pumps. (0.50)

b. What action should be taken following a confirmed high radiation alarm on the containment spray raw water system ? (0.50)

c. The containment spray Raw Water Pumps must be manually started by the control room operator ? TRUE or FALSE . (0.25)

d. This procedure directs you not to manually override or shut this system down after an auto initiation unless two conditions are met. What are these two conditions and who is authorized to make this decision ? (1.25)

QUESTION 7.04 (2.50)

Concerning procedure N1-SOP-11 (Control Room Evacuation, Fire)

a. Where do Remote Shutdown Panels # 11 and # 12 receive their power from ? (0.50)

b. Being forced to evacuate the control room, an attempt should made to bring the plant to a safe shutdown condition before leaving. List, in order of preference, eight (8) immediate operator actions/verifications to be attempted prior to leaving. (2.00)

QUESTION 7.05 (3.00)

Regarding Shift Supervisor responsibilities and concerns involving the issuance and use of RADIATION WORK PERMITS,(RWP's).

a. List five (5) of the six (6) conditions , involving work, that would require the issuance and use of an RWP . (1.25)

b. List three (3) specific qualifications, duties and/or criteria that apply to all personnel assigned as " LEADMAN " on an RWP . (0.75)

c. If a maintenance activity must carry over to the next shift, the RWP for that activity, must be approved by the appropriate Rad. Protection Tech., Leadman, and be re-initialed or signed by the Station Shift Supervisor. (TRUE or FALSE) (0.25)

d. What specific qualification criteria is required for individuals authorized to use EXTENDED RWP,s ? As a Shift Supervisor , on the 12:00 mid to 8:00 am shift , how could you verify that an individual meets this criteria ? (0.75)

QUESTION 7.06 (1.50)

During your 4:00p to12:00 mid. shift there is an unexplained (slow) decrease in Primary Containment pressure. Other than a suspected loss of Primary Containment, what additional events could have caused this pressure decrease. (Explain) ? (0.25 for the event, 0.25 for the reason)

QUESTION 7.07 (2.50)

During the 4:00 pm. to 12:00 mid. shift , at rated power, you receive two alarms;

1.Off GAS line high pressure.
2.Off GAS line high temperature

You notice that the condenser vacuum is decreasing.

a. Based on the above indications/conditions, WHAT HAS OCCURED , and what additional automatic actions can be expected ? (1.25)

b. Based on the above situation list your immediate operator actions. (1.25)

QUESTION 7.08 (2.50)

Concerning N1-SOP-29 , Pipe Break Inside Drywell ;

a. Under what conditions can the automatic controls of an Emergency Core Cooling System be placed in its manual mode ? (be specific) (1.00)

b. Think about the overall purpose of this procedure :----> List at least three (3) operational functions , with respect to the Core and its Containment,that you are expected to achieve to assure that the HEALTH and SAFETY of the public is protected . (1.50)

QUESTION 7.09 (3.00)

According to Procedure N1-SOP-3,Feedwater Malfunction(Decreasing FW Flow);

a. What immediate actions would you take if feedwater flow rapidly decreased due to a loss of the Shaft Feedwater Pump. (1.50)

b. Due to the above transient RX. Vessel level is decreasing at a very rapid rate. As the Shift Supervisor, at what Vessel level would you direct your operators to depressurize the vessel ? (0.50)

c. Is it necessary to close the MSIV,s during this transient ?(Explain)(.4)

d. List three (3) conditions that could cause HPCI to automatically initiate as a result of this transient. (0.60)

QUESTION 7.10 (2.00)

Concerning Procedure SOP-15 , Malfunction of the Control Rod Drive Sys.;

a. During a power ascention, (RX. power approx.30 %), the selected control rod starts to drift. What Automatic responses, ie; alarms/indications, would be affected ? (1.00)

b. What criterion is used to define a control rod as being inoperable.(0.5)

c. How could you verify that a control rod has become uncoupled ? (0.50)

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

FACILITY: NINE MILE POINT

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/03/12

EXAMINER: LANGE, D.

NAME: MASTER

INSTRUCTIONS:

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	100.00			8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
25.00	100.00			TOTALS

FINAL GRADE _____%

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

_____
SIGNATURE

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

FACILITY: NINE MILE POINT

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/03/12

EXAMINER: LANGE, D.

NAME: _____

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25.00	100.00			TOTALS

FINAL GRADE _____%

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

_____
SIGNATURE

QUESTION 8.01 (1.00)

What action is required if the Core Maximum Peaking Factor exceeds the Design Total Peaking Factor ? (Explain the reason for your answer) (1.00)

QUESTION 8.02 (3.00)

Concerning Refueling Operations ;

a. List six (6) methods available to verify proper fuel bundle orientation. (1.50)

b. Consider the alarm- REFUEL INTERLOCK- located on the ROD BLOCK MONITOR PANEL. List two (2) conditions,(including interlocks),that this alarm could be indicating ? (1.00)

c. Under normal operations,prior to fuel handling,procedure N1-OP-34 has a prerequisite which states, " The Fuel Pool key lock switch on the " G " panel shall be placed to the Refuel position when handling fuel or irradiated fuel casks. What is the purpose of doing this ? (0.50)

QUESTION 8.03 (2.00)

List six (6) physical interlocks and/or administrative conditions which must be satisfied prior to starting a Reactor Recirculation Pump. (2.00)

QUESTION 8.04 (2.50)

At Nine Mile Point, Unit-1 the placement of electrical jumpers, changing or removal of leads and the blocking of relays are performed ONLY under specific administrativley controlled circumstances !

a. List five (5) of these circumstanses. (1.50)

b. The placement and restoration of jumpers/blocks or lifting of leads shall be accomplished by----------. Complete the sentence with the appropriate personnel and the administrative requirements they have to adhere to. (1.00)

QUESTION 8.05 (2.50)

a. The lowest point at which the Reactor Water Level can normally be monitored is approx.---------- below minimum normal water level, or -------- above the top of the active fuel. (fill in the appropriate levels) (1.00)

b. What is the significance of the above Vessel Location Tap. (0.50)

c. The actual Low-Low-Low, water level trip point,(- 10"),is 6 ft.-3 in. below the minimum normal water level,(elev.302-9"). A General Electric service information letter resulted in raising this trip set-point 20 " to conservatively account for what possible adverse condition(s) and resulting descripencies ? (1.00)

QUESTION 8.06 (1.00)

At Nine Mile Point, procedural controls will assure that the IRM scram is maintained up to 20 % Flow. How is this accomplished ? (1.00)

QUESTION 8.07 (1.50)

Concerning the Limiting Condition For Operation (Operability Requirements) as described in Technical Specifications. When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation only if the following two (2) conditions are satisfied. LIST THESE TWO CONDITIONS. (1.50)

QUESTION 8.08 (2.50)

Assume the Reactor has failed to scram (SOP-31) by manual or automatic means. What CRITERIA would you use to determine when to initiate the STANDBY LIQUID CONTROL SYSTEM ? (Be specific). (2.50)

QUESTION 8.09 (2.00)

In accordance with the Technical Specifications, the reactor was scrammed due to Suppression Chamber water temperature being greater than 110 degrees F. The reactor is now in HOT SHUTDOWN, Suppression Pool Cooling is ON, and Suppression Chamber water temperature is 92 degrees F. Using the attached section of Technical Specfications can you commence a startup ?? (Fully Explain) (2.00)

QUESTION 8.10 (2.00)

* NOTE: USE THE ATTACHED SECTION OF THE TECHNICAL SPECIFICATIONS TO *
* ANSWER THE FOLLOWING QUESTION. FULLY REFERENCE ALL SECTIONS YOU USE. *

During a shift turnover, with the plant operating at 75% power, you are informed that the BI-Weekly Closure Surveillance Test has exceeded the maximum allowable extension interval and will be performed on your shift. Halfway through the test, " ONE" Outboard MSIV FAILS to meet the specified closing time. In accordance with the Tech Specs:

a. What situation exists due to the surveillance test being outside of the test frequency schedule. (0.75)

b. What actions must be taken due to the fact that the MSIV has failed it's closing time test ? (1.25)

QUESTION 8.11 (2.50)

Concerning NMP- Technicisl Specification DEFINITIONS, answer the following either TRUE or FALSE.

a. The TEST INTERVALS that are specified in Tech. Spec.s are only valid during periods of power operation and do not apply in the event of extended Station Shutdown. (0.50)

b. An OPERATING CYCLE is that portion of station operation between the end of one operating cycle and the end of the next operating cycle. (0.50)

c. CORE ALTERATION is the addition, removal, relocation, or other manual movement,including control rod movement with the control rod drive hydraulic system, of fuel or controls in the reactor core . (0.50)

d. A FIRE WATCH PATROL is a patrol that requires an area with inoperable fire protection equipment to be inspected at least every four (4) hrs. (0.50)

e. A TRIP SYSTEM is an arrangement of sensors and auxillary equipment required to generate and transmit a plant parameter to an instrument channel for the purpose of satisfying a component response. (0.50)

QUESTION 8.12 (2.50)

During your shift, 12:00mid - 8:00AM , with the plant at 100 % steady state power, one of your operators informs you that the closed position indication light for MSERV # 6 is not indicating. The problem is determined not to be a burned out light bulb but a maintenance problem. You are informed that this probl[illegible] not be worked on until 8:00 am, when maintenance personnel ar[illegible]le.

Using the attached sections of Tech. Specs , can the plant continue to operate under this condition ? Fully reference all sections of T.S related to this situation, giving a brief description and bases for any actions you would take. (2.50)

Ans. Key.

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

ANSWER 5.01 (2.75)

A. NO ,
Thermodynamic efficiency is a comparison of energy in versus energy out. [0.5] The increase in generator output resulted from decreasing the amount of steam diverted to the HP FW heater. [0.5] This condition requires additional energy output from the reactor to raise FW temp to the same saturation temp as before [0.5] Thus, thermodynamic efficiency of the plant has gone down. [0.5] More delta T across the heater would have caused more extraction steam to have been removed from the turbine.
B. Reactor Power ,(CMWT), increases (0.25), due to the core inlet temp-decreasing thus causing more heat to be added to reach the same core exit enthalpy . (0.50)

REFERENCE
NMP. Operations Tech. Module-9,Chapter #6.

ANSWER 5.02 (3.00)

a. Adds negative reactivity [0.25] due to the increase in neutron leakage - Moderator temperature coefficient. [0.50]
b. Adds negative reactivity [0.25] due to the increase in neutron capture in the fuel - Doppler coefficient. [0.50]
c. Adds positive reactivity [0.25] due to the decrease in neutron leakage - Moderator temperature coefficient. [0.50]
d. Adds negative reactivity [0.25] due to the increase in neutron leakage - Void coefficient. [0.50

REFERENCE
NMP.-- Reactor Theory

ANSWER 5.03 (2.25)

a. VOID COEFFICIENT (0.50), adds negative reactivity (0.25)
b. FUEL TEMP. COEFFICIENT (0.50), adds negative reactivity (0.25)
c. MODERATOR TEMPERATURE COEFFICIENT (0.50), adds positive reactivity (0.25

REFERENCE
NMP. Reactor Theory .

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

ANSWER 5.04 (2.00)

At 100 % power. (0.75) At 4 % power, you are at operating pressure but at low feedwater flow rate. NPSH is low due to T inlet being high. As power increases, pump inlet temperature is reduced due to mixing in the downcomer. T-inlet is lower so P-sat. at inlet is lower, therefor NPSH is higher. (1.25)

REFERENCE
NMP. Thermodynamics and fluid flow.

ANSWER 5.05 (2.00)

First, convert psig. to psia. by adding 14.7 psi. Then,refering to the steam tables; 900 psia. = 532 deg.F
610 psia. = 488 deg.F
532 deg.F - 488 deg.F =44 deg.F / half hour, or 88 deg./hr (1.50)

NO. The cooldown limit of 100 deg.F/hr has not been exceeded. (0.50)

REFERENCE
NMP. Module # 9 Part 2 Properties of Matter, pg. 17 thru 23 .

ANSWER 5.06 (1.50)

The cold water injection SUB-COOLS the moisture separator drains. This sub-cooling prevents TWO-PHASE FLOW in the moisture separator drain tank piping to the feedwater heaters. (1.50)

REFERENCE
NMP. N1-OP-31, Tandem Compound Reheat Turbine, Rev. # 5, pg# 75.

ANSWER 5.07 (2.25)

Flowrate is proportional to speed;
(speed)2-is proportional to head;
(speed)3-is proportional to power;
From the above relationships, since the discharge head decreased by a factor of four (4), the following new figures apply;

CAPACITY = 15,000 GPM (0.75)
SPEED = 1500 RPM (0.75)
POWER = decreases to (1/8) the original value.

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

REFERENCE
NMP- Fluid Llow. and BWR Technology Basic Pump Law Relationships.

ANSWER 5.08 (2.50)

After about 10 hours the SDM will increase to approx. 3 % as a result of Xenon peaking. After 20 to 30 hours the SDM will be back at 1 % as Xenon decays to its equilibrium value .From this point on the SDM decreases with the Xenon concentration and the Reactor will go critical if no other action is taken. (2.50)

REFERENCE
NMP. Oper. Tech. Module 1-Ch. 16 , pg.128 and Module 1,Ch,7,pg.55.

ANSWER 5.09 (2.25)

a. 13.4 Kw/ft- To limit clad plastic strain to 1 % during transient operation. (1.00)

b. 1. Radial position in the core. (0.25)
2. Axial position in the core. (0.25) (any 3 at 0.75 each)
3. Position of the rod in the bundle. (0.25)
4. Manfacturing tolerances. (0.25) Look at other ans

c. To protect the fuel cladding during a DRY-OUT,LOCA - 2200 deg.F Limit. Due to heat radiation problems in the fuel nodes. (0.50)

REFERENCE
NMP-OPER.TECH. Module 10, pg. 40 to 45.

ANSWER 5.10 (1.50)

TRUE. (0.50) If a fuel bundle dries out during a LOCA , the edge and corner rods could dissipate more heat easily than central rods. The edge and corner rods can radiate heat away from the fuel bundle while the central rods radiate much of their heat to other central rods. (1.00) (The primary heat transfer mechanism is thermal radiation.)

REFERENCE
NMP- Oper. Tech. Module 10, ch.#7 pg #-41

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

ANSWER 5.11 (1.50)

YES. (0.50) To maintain the removal of non-condensable gasses produced from the decomposition of water , activation products and noble gasses produced in the fuel and leaking into the coolant via. cladding cracks. (1.00)

REFERENCE
NMP. Oper. Tech. Module 5 pg. 41 to 45.

ANSWER 5.12 (1.50)

This accident was analyzed for three reactor operating modes. The HOT STAND BY condition results in the most severe condition. (0.50) This is because of the higher reactivity worths than at full power,and because of the larger concentration of fission products than at cold conditions. (1.00)

REFERENCE
NMP - Oper. Tech. Module #12, pg. 59.

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

ANSWER 6.01 (3.00)

A. * High DW pressure(.25), 3.5 psig(.25), one out of two taken twice(.2)
   * Low-Low-Low water level(.25), -10 inches(.25), one out of two taken ~~twice~~ once(.2)
   * Time delay timed out(.2), 120 seconds(.2), logic is 1 of 2(.2) (12 sec) TAKEN ONCE.

B. 5 seconds after the 120 second timer started(.25), if the primary valve #1 was not open, its backup valve #2 would open(.25)

C. Acoustic monitor (0.25)
   Temperature elements (0.25)

REFERENCE
Operation Technology, Module 4, Part 8, ADS

ANSWER 6.02 (2.50)

1. Low-Low-Low condenser vacuum (7" Hg).
   Bypassed when < 600 psig. and mode switch is in STARTUP or REFUEL.
2. Main steam line high radiation. (5 x NFPB).
3. Main steam line high flow . (105 psid or 120 % steam flow).
4. Reactor low pressure. (850 psig. with mode switch in run).
5. High area temperature in the steam tunnel. (200 deg.F).
6. Low-Low vessel level. (+ 5").

(0.25 for each correct ans.)

b. 1. Restricts discharge and protects vessel internals from large D/P.(.33
   2. Provides signal for MSIV closure. (0.33)
   3. Provides steam flow signal to FWCS. (0.33)

REFERENCE
NMP. Oper. Tech. Module # 2, Chapter 9 .

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

ANSWER 6.03 (3.00)

a. The flow control valve is prevented from opening following a HPCI initiation signal until feedwater pump discharge pressure is sensed. (0.75)

b.1. Remain in operation. (0.25)
2. The idle FW Pump will start and be up to speed in approx. (10 sec.) (.25)
3. The FW control sys. will switch to single element control if it had been in three element. (0.25)
4. Attempt to maintain RX. level at (65) (0.25)
5. " " " " " (71) (0.25)

c. 1. The feed pump automatic start is blocked by aux.oil pressure less than (8 psig.) (0.33)
2. The feed pumps will trip if suction pressure drops below 200 psig. or if aux. oil pressure drops below 3 psig. (0.33)
3. The feedwater booster pumps automatic start is blocked by suction pressure less than 35 psig. (0.33)

REFERENCE
NMP. N1-OP-46 , HPCI. and OPER. TECH. Module #4, part #9.

ANSWER 6.04 (1.50)

a 1. Used to calibrate individual LPRM detectors. (0.50) to map the core
2. Used by the prosess computer (0.50) to determine MCPR and local heat flux conditions. (0.25)

REFERENCE
NMP.- Oper. Tech , Module 3 part 5.

ANSWER 6.05 (2.00)

A differential pressure sensor is used to confirm the integrity of the CORE SPRAY piping within the reactor vessel (between the inside of the vessel and the core shroud).
To continuously monitor the integrity of the core spray piping, a Delta P switch measures the pressure difference between the two loops, which is effectively the inside of each Core Spray sparager pipe, just outside of the Rx vessel. If the core spray sparager is intact, this pressure difference will be zero. If integrity is lost, this pressure differential will include the pressure drop across the steam seperator. Alarms at 5 psid in the control room (2.00)

REFERENCE
NMP. Oper. Tech. Module # 4, Part 10. Core Spray. pg.4-61.

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

ANSWER 6.06 (2.25)

a. High temperature- > 83 deg.C (0.25) Low pressure - < 17 psig. (
Low system flow < 442 gpm. (0.25)

b. NO, (0.25) , Immediately reduce Rx. Recirc. flow to minimum, in an attempt to prevent a scram. (0.50)

c. System flow is regulated to maintain inlet pressure low enough to prevent water from entering the stator windings in the event of a leak. (if a leak develops hydrogen will leak into the cooling water). (0.75)

REFERENCE
NMP. N1-OP-44, Gen. Stator Cooling Water Sys., pg. 1-5 .

ANSWER 6.07 (3.00)

a. 1. Fuel Zone Ind.
3. Hi/Lo, Lo-Lo Rosemount.
4. Narrow Range GE-MAC.
5. Flange Level GE-MAC.
6. Turbine Trip Signal.
7. High Level Alarm.
9. Low Level Scram.
10. Low-Low isolation.
11. Instrument Zero.
12. Low-Low-Low ADS.
2. Lo-Lo-Lo- Ro correct ans.) (1.20 for full credit)

b. This sys. is initiated by manual or automatic tripping of all five (5) Recirc. MG set drive motor breakers. (0.50)

c. 1. Core Level / Torus Mon. Sys. Trouble annunciator will alarm. (0.35)
2. The digital level indicators on panel " F " will flash on and off. (0.35)

d. 1. Drywell pressure (0.20)
2. Reactor Pressure (0.20)
3. Reactor building temperature (0.20)

REFERENCE
NMP. Oper. Tech. Module 2, Ch. 2, pg.16,17,& fig.2-1.

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

ANSWER 6.08 (2.75)

a. 1. Continuity lights go out. 2. Alarm indication.
3. Milliamp meters on back of 09-03 indicate current flow to firing ckt.
4. Decrease in power.
5. Selected pump has red light indicating pump is running.
6. SBLC pressure > reactor pressure.
7. SBLC tank level decreasing. (any six at 0.25 each)

b. YES. (0.25) If the SBLC. tank level approaches zero (0.50) or the SBLC pump begins to loose discharge pressure. (0.50) This is indicated by fluctuation of pump amperage and press. Need SS approval.

REFERENCE
NMP . Oper. Tech. Module 4-81 and N1-OP-12

ANSWER 6.09 (2.25)

a. A relief valve in each core spray loop provides a Minimum Flow Recirculation path to the TORUS when the pumps are running at shutoff head. (0.75)

b. 1. Pressure on the suction side of the Core Spray pumps. (0.25)
ALARM on panel K at (2.5 psig.) LOW SUCTION PRESSURE. (0.25)
2. Pressure is sensed downstream of the flow orifice. (0.25)
ALARM on panel K at (225 psig.) CORE SPRAY LOOP LOW PRESSURE. (0.25)
3. Pressure is sensed on the disch.of the core spray topping pumps.(0.25
Provides ALARM at (445 psig) CORE SPRAY LOOP HIGH PRESSURE and remote indication of core spray system pressure. (0.25)

REFERENCE
NMP.Oper. Tech. Module # 4 pg. 59-61 and N1-OP-2.

ANSWER 6.10 (2.75)

1. Main Steam Isolation Valves CLOSE (0.25) causing a REACTOR SCRAM on valve position. (0.50)

2. Feedwater FCV - locks-up as is (0.25) causing a LOSS OF HPCI FLOW CONTROL. (0.50)

3. MAKE -UP valve to Condenser - CLOSES (0.25) causing HOTWELL LEVEL to DECREASE. (0.50)

4. RX.Bldg.CLC-TCV, OPENS (0.25) Temp. decrease on closed loop water.(0.25)

REFERENCE
NMP. N1-SOP-5 . Instrument Air Failure. pg.# 2,3 .

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

ANSWER 8.01 (1.00)

Action required- The scram trip setting must be adjusted . (0.25)
Reason - To ensure that the LHGR limit is not violated for any combination of MTPF and REACTOR CORE THERMAL POWER. (0.75)

REFERENCE
NMP-1, Tech.Spec. bases for 2.1.2:
1

ANSWER 8.02 (3.00)

a. 1. Fuel assembly serial # are readable from the associated control rod.
2. Lugs on the fuel assembly bail handle point at the associated control rod.
3. Channel spacer buttons are above the associated control rod.
4. Channel fastener spring clips are above the associated control rod.
5. Gadolinium rods have longer end plugs which protrude through the upper tie plate.
6. Overall core symmetry.
(0.25 for each correct answer)

b. 1. The mode switch is in refuel with one control rod withdrawn. (0.25)
An attempt to move the refuel platform with a fuel element over the core will result in de-energizing of the hoist motor. (0.25)
2. The mode switch is in refuel with the refuel platform loaded and over the core ;(0.25) This condition inserts a rod block to prevent control rod withdrawal. (0.25)

c. This places the fuel pool high radiation monitor, on the refueling bridge,on the emergency ventiation circuit (alarms at 1000 mr/hr).(0.50)

REFERENCE
Operations Technology, module 2, Chapter 3., and N1-OP-34, Refueling Procd.

ANSWER 8.03 (2.00)

1. The 86 relays are reset.
2. The Suction valve is open.
3. The discharge valve is shut.
4. The discharge bypass valve is open.
5. The scoop tube is reset.
6. Local MA station in manual and set at 20 % .
7. Reactor Recirc pump must not be shutdown > 2 hrs. during power operation
8. Do not exceed 775 Kw power consumption.
(any six at 0.33 each)

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

REFERENCE
NMP - N1-OP-1, Rev. 23, pg# 12.

ANSWER 8.04 (2.50)

a. 1. In accordance with approved maintenance procedures.
   2. In " " " test or surviellance procedures.
   3. In " " an approved modification.
   4. To facilitate the conduct of tests and checks.
   5. To preserve the safety, function, and/or integrity of the station or systems.
   6. For non-routine activities during refueling outage.
   (any 5 at 0.30 each)

b. Appropriate personnel----> A licensed operator OR a qualified instrument technician or electrician. (0.25)

   Administrative requirements ;
   1. The action taken must be verified by a second such QUALIFIED person.
   2. Only with prior approval of the Station Shift Supervisor.
   3. Only with the knowledge of the Chief Shift Operator.
   (three (3) required for full credit @ 0.25 ea)

REFERENCE
NMP. ; APN.-7A, Placement of jumpers/blocks or lifting of leads.

ANSWER 8.05 (2.50)

a. Seven (7) ft. eleven (11) in. below minimum normal water level. (0.50)
   Four (4) ft. eight (8) in. above the top of the active fuel. (0.50)

b. This is the location of the Reactor Vessel Tap for the Low-Low-Low, water level instrumentation. (0.50) or, INST. "0" accepted if used in (c) "ADS SET POINT."

c. This trip point was raised 20 inches to conservatively account for possible differences in ACTUAL to INDICATED water level (0.50) due to potentially HIGH DRYWELL TEMPERATURES (0.50).

REFERENCE
NMP- Tech. Spec. 2.1.1, Bases for Fuel Cladding- Safety Limit, pg. 13

ANSWER 8.06 (1.00)

This is accomplished by keeping the Reactor Mode Switch in the Startup position until 20 % flow is exceeded (0.50) and the APRM's are on scale(0.50).

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

REFERENCE
NMP. Tech. Spec. 2.1.2 Bases for Fuel Cladding. pg-16.

ANSWER 8.07 (1.50)

1. Its Corresponding NORMAL or EMERGENCY power source is operable. (0.75)
2. All of its redundent system(s), subsystem(s), train(s), component(s) and device(s) are operable. (0.75)

REFERENCE
NMP. Tech. Spec. Sec. 3.0, Operability Requirements. pg. 25.

ANSWER 8.08 (2.50)

a. Increasing Reactor power (0.50); or if two or more adjacent control rods (0.50) or thirty or more control rods cannot be inserted below position 06 (0.50) and reactor water level cannot be maintained (0.50) or if suppression pool temperature cannot be maintained (0.50).

REFERENCE
NMP- N1-SOP-32, rev#4, pg. #9.

ANSWER 8.09 (2.00)

NO.(.75) Operation shall not be resumed until the pool temperature is reduced to below the power operation limit specified within the shaded area of figure 3.3.2a when downcommer submergence is > 4 ft. or figure 3.3.2b when downcommer submergence is > 3 ft. (1.25)

REFERENCE
Technical Specification 3.3.2 specfication (E)
Technical Specification 3.3.2 Figures a&b.

ANSWER 8.10 (2.00)

a. The LCO for the MSIVs are stated in Section 3.2.7. Two main steam line isolation valves per main steam line shall be operable with closing times greater than or equal to 3 secs and less than or equal to 10 sec. Operating outside of this specification is a violation of T.S. and therefore is a reportable occurance. (0.75)
b. Action 'b' of 3.2.7 states that with the one MSIV INOP due to exceeding the allowable closing time, the affected steam line shall be isolated . If the problem is not corrected , initiate an orderly shutdown within one hour and have the reactor in cold shutdown within 10 hours. (1.25

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

REFERENCE
NMP- TS- sec.3.2.7 Reactor coolant system isol. valves.

ANSWER 8.11 (2.50)

a. TRUE
b. FALSE
c. FALSE
d. FALSE
e. FALSE (0.50 for each correct ans.)

REFERENCE
NMP- Tech. Spec. definitions.

ANSWER 8.12 (2.50)

With no indication of valve position and possible logic or maintance problem existing and unable to be corrected, the valve has to be considered inoperable according to the definition of operability. (0.50)
The LCO for pressure relief systems-Solenoid Actuated Relief Valves , as stated in section 3.1.5 , requires all six valves to be operable when you are at operating temperature and pressure.(0.25) The primary bases is for depressurization to allow for full flow core spray operation in the event of a small line break. (0.25)
The LCO for pressure relief systems- as stated in section 3.2.9 requires that only 5 of the 6 valves need to be operable. (0.25)
The primary bases for this is to limit reactor overpressure below the lowest safety valve set point in the event of rapid reactor isolation. (0.25)
The two conflicting LCO's are determined for different bases requirements. The conservative LCO for section 3.1.5, where all 6 valves shall be operable, is the most limiting and should be adhered to. (0.50) The action to be followed is the same for both LCO's-----> Be 110 psig. or less and saturation temperature, respectively, within ten (10) hours. (0.50)

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

ANSWER 7.01 (2.50)

a. The respective supply breaker will not close until voltage has decayed to 20 % of the normal value. (0.25). This delay precludes re-energizing motors on PB-11 or PB-12 when their voltage may be considerably out of phase with the incoming voltage . (0.75)

b. R-1014 is normally closed and R-1011 is normally open. (0.25). This provides a better balance of loading between T-1015 and T-101N . (0.25) R-1014 and R-1011 are interlocked such that only one can be closed at a time. (0.25) This interlock can be defeated by a control switch in the control room. (0.25)

c. Only two (2) of the three can be closed at any one time. (0.50)

REFERENCE
NMP. N1-OP-30 Rev. #6,pages 1, 5, 6 .

ANSWER 7.02 (3.00)

a. 1. Control Rod position.
2. Recirculation flow.
3. Reactor pressure.
4. Steam flow or temp.
5. Feedwater flow or temp.
6. Turbine Generator load.
7. Bypass,relief or safety valve flow.

(any six at 0.25 for each correct ans.)

b. 1. LPRM alarm . , 2. APRM alarm. , 3. Rod Block alarm. (0.25 each)

c. Reduce power to 80 % of the power level prior to the change using Recirc flow. (0.25) This will prevent bundle overpower (0.25) and overloading of the other feedwater heater strings. (0.25)

REFERENCE
NMP. N1-SOP-19 (pg,2&3)

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

ANSWER 7.03 (2.50)

a. A combination of lo-lo reactor vessel water level and high drywell pressure (3.5 psig.) (0.50)

b. The raw water pump and the containment spray pump in the affected loop should be secured. (0.25)
The loop suction and discharge valves should be closed. (0.25)

c. TRUE . (0.25)

d. 1. Sufficient evidence shows that the system is not performing its intended function. (0.50)
2. Continued operation will prolong or produce an unsafe condition. (0.50)
Shutdown of the system will be at the direction of the Station Shift Supv. (0.25)

REFERENCE
NMP. N1-OP-14, pages 1 thru 5 .

ANSWER 7.04 (2.50)

a. Panels # 11 & 12 are powered from RPS continuous power motor generators 162 and 172 . (0.50)

b. 1. Scram the Reactor
2. Trip the 345 KV breakers and trip machine
3. Verify the Rx. Scram.
4. Verify the Turbine Tripped
5. Initiate emergency cooling
6. Operate Manual Isolation Switches - Vessel Isolation Channel # 11 and # 12 on the console & verify MSIV & RX. Water Cleanup Isolation.
7. Sound Fire Alarm and identify area if known.
8. Verify HPCI initiation.
(eight correct answers at 0.25 each)

REFERENCE
NMP. , N1-SOP-11 Control Room Evacuation. and OP Tech. Module # 4-87,89

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

ANSWER 7.05 (3.00)

a. 1. Contamination levels greater than 10,000 dpm/cm2 .
2. Airborn radioactivity requiring the use of respiratory equipment.
3. Neutron radiation exposure.
4. High radiation area entries.
5. Unknown conditions in an area to be entered.
6. Maintenance of equipment, controls or instrumentation in Radiation areas or High radiation areas.
(any five (5) at 0.25 for each correct ans.)

b. The LEADMAN must be qualified in Radiation Protection. (0.25)
2. The LEADMAN or a qualified alternate must be ON THE JOB AT ALL TIMES .
3. The LEADMAN may be responsible for ONLY ONE RWP at a time.
(0.25 for each correct answer)

c. FALSE - The SSS signature is only required upon issue of the RWP. (0.25)

d. Only those people who meet the following criteria ;
1. Qualified in Radiation Protection. (0.20)
2. Demonstrated their knowledge of Self Monitoring by taking and passing a comprehensive Self Monitoring Qual.Course (0.30)

The names of those who are approved to use the EXTENDED RWP are contained in a LOG in the Control Room. (0.25)

REFERENCE
NMP. Procedure # RP-2 , Radiation Work Permit Procedure, pg.,2,3,6,7.

ANSWER 7.06 (1.50)

Three events;
1. Decrease in Supp. pool level; As level decreases the nitrogen must occupy a larger volume, resulting in decreased Drywell pressure.
2. Increase in barometric pressure; The Drywell pressure instruments are all refrenced to atmosphere therefore an increase in atmospheric press. causes a decrease in D/W press. PBCLC
3. A decrease in D/W or chilled water temp; An increase in cooling capacity in the D/W from the D/W coolers and chillers will decrease D/W pressure. (Other reasonable ans.accepted if substantiated)
(0.25 for each event; 0.25 for each reason)

REFERENCE
NMP-

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

ANSWER 7.07 (2.50)

a. EXPLOSION in the Air Ejector discharge piping. (0.50)
Automatic Actions;
1.Valves BV-76-12/13 close and(off gas flow goes to zero.) (0.50)
2.Reactor Scram at 23 " Hg. (0.25)

b.
1.Reduce reactor load by decreasing recirc. flow.
2.Close main steam supply valve to air ejectors and mixing jet.
3.Insert control rods per rod pattern until vacuum decreases to near scram point.
4.Manually scram the reactor.
5. Initiate emergency condensers, as necessary to remove the decay heat.
6. Inform station personnel of conditions.
7. Notify Plant Superintendent.
(7 correct ans. at 0.1785 ea.)

REFERENCE
NMP. SOP- 18 ,"Explosion in the Air Ejection Disch. Piping"
(Symptoms / Automatic Actions / Operator Actions)

ANSWER 7.08 (2.50)

a. 1. Misoperation in automatic is confirmed by at least two independent process parameter indications. (0.50)
2. Core cooling is assured AND this procedure,(SOP-29),directs you.(0.50

b. 1. Maintain Core Cooling.
2. Limit the release of off-gas radiation.
3. Place the Reactor Core and Containment in a SAFE STABLE condition.
4. Keep the Torus bulk temp. within specified SAFETY limits.
(any three at 0.50 each)

NOTE: Other specific Operational objectives related to SAFETY accepted.

REFERENCE
NMP. N1-SOP-33, Pipe Break Inside Drywell., Cautions, Limitations and over-all purpose and objectives. pg# 1-5 .

ANSWERS -- NINE MILE POINT -85/03/12-LANGE, D.

ANSWER 7.09 (3.00)

a. 1. Shift mode switch to refuel.
   2. Check all rods are fully inserted.
   3. Observe power level decreasing.
   4. Check for HPCI operation. Ensure that both moter driven feedwater pumps are running.
   5. Chcek that the emergency condensers are in operation.
   6. Check that the Core Spray pumps are running and recirculating back to the torus.

(0.25 for each correct answer)

b. Low-Low-Low Level (- 10 inches) (0.50)

c. Yes, To conserve coolant inventory. (0.40)

d. 1. Runout flow of 1.9 x 10(6) or 3800 gpm. (0.20)
   2. Turbine Trip (0.20)
   3. Low Rx. water level (0.20)

REFERENCE
NMP. SOP-3, and Simulator scenerio Objectives # 1&2 .

ANSWER 7.10 (2.00)

a. 1. LPRM output in the vicinity of the drifting rod.
   2. Position indication for that rod.

NOTE: The ROD DRIFT alarm will not come in due to the rod that is drifting is the one selected.
The RWM alarm will not come in due to being > 25 % power. (1.00)

b. A control rod which cannot be moved with control rod drive pressure. .5)

c. Control Rod overtravel alarm; (Ann. window F2-6 , Computer point B0-12) when the control rod has been fully withdrawn. (0.50)

REFERENCE
NMP., SOP-15, Malfunction of CRD system.

U.S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR REQUALIFICATION EXAMINATION

FACILITY: NINE MILE POINT

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/03/11

EXAMINER: BERRY, J.

NAME: _____

INSTRUCTIONS

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%

CATEGORY VALUE	% OF TOTAL	SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	100.00			4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	100.00			TOTALS

FINAL GRADE _____%

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

_____
SIGNATURE

QUESTION 4.01 (3.00)

In accordance with procedure NI-SOP-32, Failure of Reactor to Scram, what six (6) immediate actions would you take to reduce power and insert all control rods in an ATWS situation? (3.0)

QUESTION 4.02 (2.00)

Describe, in general, the four things you would do to reset a high pressure coolant injection (HPCI) initiation, assuming that the initiation signal has cleared. (2.0)

QUESTION 4.03 (2.00)

a. Why is an operator instructed to "reduce reactor power to 80% of the original power level with Reactor Recirculation flow" BEFORE removing a feedwater heater string? (1.0)

b. When two condensate booster pumps are required, the preferred linup is with #11 and #13 running; when one booster pump is required, #11 or #13 should be in service. Why is this preferred? (1.0)

QUESTION 4.04 (3.00)

a. Assuming the CSO and the NAOE were able to accomplish NOTHING in the way of securing the station prior to an evacuation, how is the reactor shutdown AND how is the shutdown verified? Your answer should include where the CSO and NAOE proceed to and their subsequent actions. (1.0)

b. After verification of a turbine trip, the SSS is to proceed to powerboard 11 & 12. What actions are to be performed at powerboard 11 & 12? (1.0)

c. How can RAW WATER be supplied to feed the reactor? (1.0)

QUESTION 4.05 (3.00)

Concerning Procedure SOP-19 (Unexplained Reactivity Change);

a. List six (6) plant parameters/indications that should be checked if an unexplained reactivity change should occur at rated power. (1.5)

b. Depending on the magnitude of the reactivity change, list three alarms that may be initiated. (PRIOR to a reactor scram) (0.75)

c. If this reactivity change is a result of decreased temperature, due to a loss of a feedwater heater string, what is your immediate action and what two (2) adverse conditions are you trying to protect against? (.75)

QUESTION 4.06 (3.00)

Concerning Procedure N1-OP-14, Containment Spray System:

a. What two (2) signals are required to automatically start the containment spray pumps? (0.5)

b. What action should be taken following a confirmed high radiation alarm in the containment spray raw water system? (0.5)

c. The containment spray Raw Water Pumps must be manually started by the control room operator? TRUE or FALSE. (0.25)

d. This procedure directs you not to manually override or shut this system down after an auto. initiation unless two conditions are met. What are these two conditions and who is authorized to make this decision ? (1.25)

QUESTION 4.07 (2.00)

During the 4:00 pm to 12:00 midnight shift, at rated power, you receive two alarms:

1. Off GAS line high pressure.
2. Off GAS line high temperature

You notice that the condenser vacuum is decreasing.

a. Based on the above indications/conditions, WHAT HAS OCCURED?, and what additional automatic actions can be expected? (1.0)

b. Based on the above situation list your immediate operator actions. (1.0)

QUESTION 4.08 (2.50)

Concerning N1-SOP-29, Pipe Break Inside Drywell:

a. Under what conditions can the automatic controls of an Emergency Core Cooling System be placed in its manual mode? (be specific) (1.0)

b. Think about the overall purpose of this procedure; ---- List at least three (3) operational functions, with respect to the Core and its Containment, you are expected to achieve to assure that the HEALTH and SAFETY of the public is protected. (1.5)

QUESTION 4.09 (2.50)

According to N1-SOP-3, Feedwater Malfunction (Decreasing FW Flow):

a. What immediate actions would you take if feedwater flow rapidly decreased due to a loss of the Shaft Feedwater Pump? (1.5)

b. Due to the above transient RX. Vessel level is decreasing at a very rapid rate. As the Shift Supervisor, at what Vessel level would you direct your operators to depressurize the vessel? (0.25)

c. Is it necessary to close the MSIVs during this transient? (0.25)

d. List three (3) conditions that could cause HPCI to automatically initiate as a result of this transient. (0.5)

QUESTION 4.10 (2.00)

Concerning Procedure SOP-15, Malfunction of the Control Rod Drive System

a. During a power ascension,(RX. power approx. 30 %), the selected control rod starts to drift. What Automatic responses, ie; alarms/indications, would be affected? (1.0)

b. What criterion is used to define a control rod as being inoperable? (0.5)

c. How could you verify that a control rod has become uncoupled? (0.5)

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 4.01 (3.00)

1. Place Mode Switch in shutdown (This inserts an additional scram signal)
2. Trip recirculation pumps
3. Fully insert control rods using 'Emergency Rod In'
4. Reset RPS trip. Manually scram the reactor
5. Individually scram rods from 'M' panel
6. Isolate and vent scram air header locally (0.33 each)

REFERENCE
NI-SOP-32, Rev. 4, pg 8 JCK-174

ANSWER 4.02 (2.00)

1. Verify a) Feedwater flow on #11 and #12 is < 1.9 million lbm/hr.
   b) Reactor low level trip is clear (.5)
2. Switch feedwater pump #11 and #12 M/A stations to manual (.5)
3. Adjust the manual outputs until the deviation meters on the #11 #12 M/A stations are nulled. (.5)
4. Press the 'Feedwater Return to Normal After HPCI' pushbutton on the reactor control console. (.5)

REFERENCE
NI-OP-16, pg. 19 EDH-324

ANSWER 4.03 (2.00)

a. This will prevent the other feedwater heater strings from being overloaded and will preclude possible over-power of the nuclear fuel. Also power increase due to increased inlet subcooling. (1.0)

b. This preferred lineup will preclude a system feedwater disturbance due to the loss of powerboard #101. ~~Also insures HPCI availability~~. (1.0)

REFERENCE
NI-OP-16, pg. 19 EDH-117

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 4.04 (3.00)

a. The CSO proceeds to shutdown panel #12 and trips MG set 141. The NAOE proceeds to shutdown panel #11 and trips MG set 131. Verification is the 'All Rods In' white light on their respective panels. (1.0)

b. Verifies that a condensate and feedwater booster pump are operating and starts feedwater pump #11, if HPCI has failed to initiate(1.0)
Also manual transfer of PB-11&12 if auto transfer fails

c. By installing an available spool piece between the feedwater system and the fire protection water system. (1.0)
Also cross-connect to containment spray raw water through intertie valves to core spray

REFERENCE
N1-SOP-11, pg. 3-5 EDH-318

ANSWER 4.05 (3.00)

a. 1. Control Rod position.
2. Recirculation flow.
3. Reactor pressure.
4. Steam flow or temp.
5. Feedwater flow or temp.
6. Turbine Generator load.
7. Bypass,relief or safety valve flow.

(any six at 0.25 for each correct ans.)

b. 1. LPRM alarm . , 2. APRM alarm. , 3. Rod Block alarm. (0.25 each)

c. Reduce power to 80 % of the power level prior to the change using Recirc flow. (0.25) This will prevent bundle overpower (0.25) and overloading of the other feedwater heater strings. (0.25)

REFERENCE
NMP. N1-SOP-19 (pg,2&3)

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 4.06 (3.00)

a. A combination of lo-lo reactor vessel water level and high drywell pressure (3.5 psig.) (0.50)

b. The raw water pump and the containment spray pump in the affected loop should be secured. (0.25)
The loop suction and discharge valves should be closed. (0.25)

c. TRUE . (0.25)

d. 1. Sufficient evidence shows that the system is not performing its intended function. (0.50)
2. Continued operation will prolong or produce an unsafe condition. (0.50)
Shutdown of the system will be at the direction of the Station Shift Supv. (0.25)

REFERENCE
NMP. N1-OP-14, pages 1 thru 5 .

ANSWER 4.07 (2.00)

a. EXPLOSION in the Air Ejector discharge piping. (0.50)
Automatic Actions:
1.Valves BV-76-12/13 close and off gas flow goes to zero. (0.50)
2.Reactor Scram at 23 " Hg. (0.25)

b.
1.Reduce reactor load by decreasing recirc. flow.
2.Close main steam supply valve to air ejectors and mixing jet.
3.Insert control rods per rod pattern until vacuum decreases to near scram point.
4.Manually scram the reactor.
5. Initiate emergency condensers, as necessary to remove the decay heat.
6. Inform station personnel of conditions.
7. Notify Plant Superintendent.
(7 correct ans. at 0.1785 ea.)

REFERENCE
NMP. SOP- 18 ,"Explosion in the Air Ejection Disch. Piping"
Symptoms / Automatic Actions / Operator Actions

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 4.08 (2.50)

a. 1. Misoperation in automatic is confirmed by at least two independent process parameter indications. (0.50)
   2. Core cooling is assured AND this procedure,(SOP-29),directs you.(0.50

b. 1. Maintain Core Cooling.
   2. Limit the release of off-gas radiation.
   3. Place the Reactor Core and Containment in a SAFE STABLE condition.
   4. Keep the Torus bulk temp. within specified SAFETY limits.
   (any three at 0.50 each)

NOTE: Other specific Operational objectives related to SAFETY accepted.

REFERENCE
NMP. N1-SOP-33, Pipe Break Inside Drywell., Cautions, Limitations and overall purpose and objectives. pg# 1-5 .

ANSWER 4.09 (2.50)

a. 1. Shift mode switch to refuel.
   2. Check all rods are fully inserted.
   3. Observe power level decreasing.
   4. Check for HPCI operation. Ensure that both moter driven feedwater pumps are running.
   5. Chcek that the emergency condensers are in operation.
   6. Check that the Core Spray pumps are running and recirculating back to the torus.
   (0.25 for each correct answer)

b. Low-Low-Low Level (- 10 inches) (0.50)

c. Yes, To conserve coolant inventory. (0.40)

d. 1. Runout flow of 1.9 x 10(6) or 3800 gpm. (0.20)
   2. Turbine Trip (0.20)
   3. Low Rx. water level (0.20)

REFERENCE
NMP. SOP-3, and Simulator scenerio Objectives # 182 .

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 4.10 (2.00)

a. 1. LPRM output in the vicinity of the drifting rod.
   2. Position indication for that rod.
   NOTE: The ROD DRIFT alarm will not come in due to the rod that is drifting is the one selected.
   The RWM alarm will not come in due to being > 25 % power. (1.00)

b. A control rod which cannot be moved with control rod drive pressure.(.5)

c. Control Rod overtravel alarm; (Ann. window F2-6 , Computer point BO-12) when the control rod has been fully withdrawn. (0.50)

REFERENCE → Nuclear Inst response
NMP., SOP-15, Malfunction of CRD system.

U.S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR REQUALIFICATION EXAMINATION

FACILITY: NINE MILE POINT

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/03/11

EXAMINER: BERRY, J.

NAME: _____

INSTRUCTIONS

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%
least 80%. Examination pape

CATEGORY VALUE	% OF TOTAL	SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	100.00			3. INSTRUMENTS AND CONTROLS
25.00	100.00			TOTALS

FINAL GRADE _____%

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

_____
SIGNATURE

QUESTION 3.01 (3.00)

a. What are two signals (including setpoints) that would cause the Feed System to shift to the HPCI Mode of operation and HOW would the results of a LOCA initiate each? (1.0)

B. When the Feed Pump is idle or not in the HPCI mode, the HPCI controller setpoint is blocked. This setpoint is not applied to the controller until ____?____ is produced at the ____?____. (1.0)

C. Why is the HPCI controller setpoint blocked initially? (1.0)

QUESTION 3.02 (1.50)

For each of the following systems, list what TYPE of radiation detector is used and what AUTOMATIC ACTIONS occur when the monitors trip. (NOTE: If no auto actions occur, indicate so. Assume lineups are appropriate for auto actions to occur.)

A. Air ejector offgas
B. RBCLC
C. Refueling Bridge (1.5)

QUESTION 3.03 (2.50)

Concerning the fuel zone level detectors:

a. Under what conditions does the value displayed on the fuel zone instrument come from the FUEL ZONE level transmitter? (Include system initiation signal(s) in your answer) (2.0)

b. What indication does the control room operator have that reference leg flashing is occuring in the fuel zone level instrument? (0.5)

QUESTION 3.04 (3.00)

Assume the REACTOR LEVEL CONTROL SYSTEM is being operated in 3-element control using reactor level detector channel '11'. Reactor power is at 85%, STEADY STATE.
For each of the instrument or control signal failures listed below, STATE HOW REACTOR LEVEL WILL INITIALLY RESPOND (increase, decrease, or remains constant) and BRIEFLY EXPLAIN WHY in terms of what is happening in the Level Control System immediately following the failure.

(FOR EXAMPLE, your answers should include the following detail, "Causes reactor level to decrease due to a steam flow/feed flow error signal, steam flow < feed flow, resulting in a closure signal to the feedwater control valve.")

NOTE: A block diagram of the Feedwater Control System is on the following page for your use.

a. #12 FEEDWATER FLOW transmitter FAILS HIGH (1.0)

b. Channel '11' REACTOR LEVEL detector signal FAILS LOW. (1.0)

c. LOSS OF CONTROL SIGNAL to #13 FEEDWATER CONTROL VALVE. (1.0)

QUESTION 3.05 (3.00)

For each of the following, state whether a ROD BLOCK, HALF-SCRAM, FULL SCRAM, or NO PROTECTIVE ACTION is generated for that condition. NOTE: IF two or more actions are generated, i.e. rod block and a half-scram, state the most severe, i.e. half-scram. Assume NO operator actions.

a. APRM 11 Downscale, Mode Switch in RUN (0.6)
b. <4 LPRM inputs to APRM 15, Mode Switch in STARTUP (0.6)
c. Both Flow Conv. Units Upscale (>107% flow), Mode Switch in RUN (0.6)
d. APRM 12 and 16 Upscale, Mode Switch in STARTUP (0.6)
e. Main Steam Line 111 ISOLATED, Mode Switch in RUN (0.6)

QUESTION 3.06 (2.00)

Describe fully how Reactor Building Closed Loop Cooling(RBCLC) temperature is regulated as the heat load on the system increases. (2.0)

QUESTION 3.07 (2.00)

A. Will a normal transfer of an RPS Bus from its normal to its emergency power supply cause any protective action? WHY or WHY NOT? (1.0)

B. Will a shift of a REACTOR TRIP Bus from its normal to its emergency power supply cause any protective action? WHY or WHY NOT? (1.0)

QUESTION 3.08 (2.50)

A. The Main Steam Line flow restrictors are used to develop main steam flow signals. What are two (2) CONTROL FUNCTIONS provided by this signal? (1.0)

B. Pressure switches on the main condenser will actuate the PCIS to shut the MSIV's if vacuum decreases to ____?____. (.5)

C. WHEN and HOW is the low vacuum trip of the MSIV's bypassed? (1.0)

QUESTION 3.09 (3.00)

Concerning Reactor Vessel Level Instrumentation:

a. Using the attached figure 2-1, (Rx Level Inst.), list what level instruments are used to measure the indicated parameters 1, 2, 3, 4, & 5. (1.0)

b. Indicate what level parameter items 6 through 12 signify. (1.25)

c. What three (3) plant variables are used for compensation by the Fuel Zone Level Indicators? (0.75)

QUESTION 3.10 (2.50)

Concerning the Standby Liquid Control System:

a. Once the SBLCS has initiated, what six (6) CONTROL ROOM indications could you use to verify that the system is operating properly AND injecting into the reactor vessel? (1.5)

b. After initiation of the SBLCS is it permissible to shut the system down? If not, WHY? If so, under what conditions? (1.0)

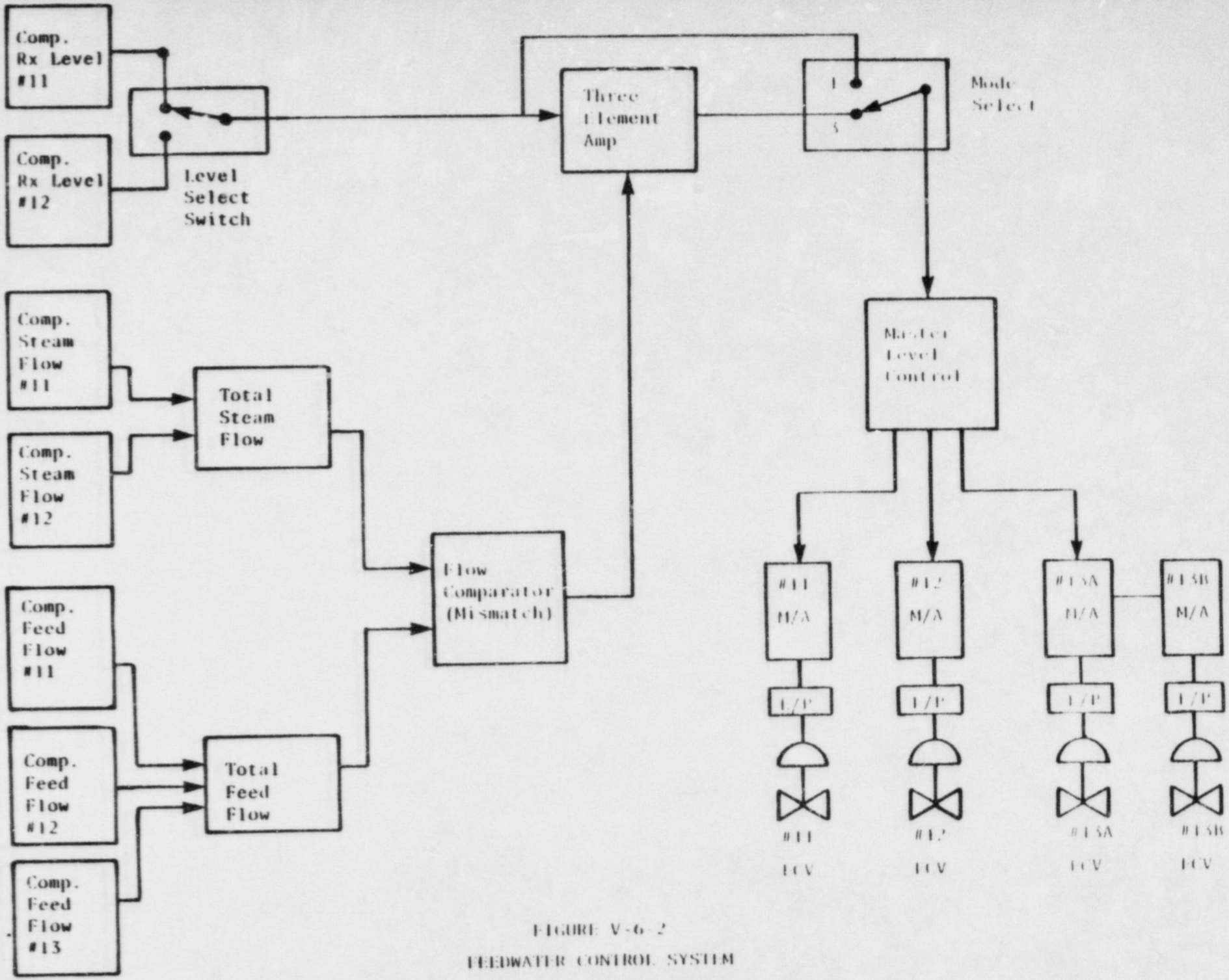


FIGURE V-6-2

FEEDWATER CONTROL SYSTEM

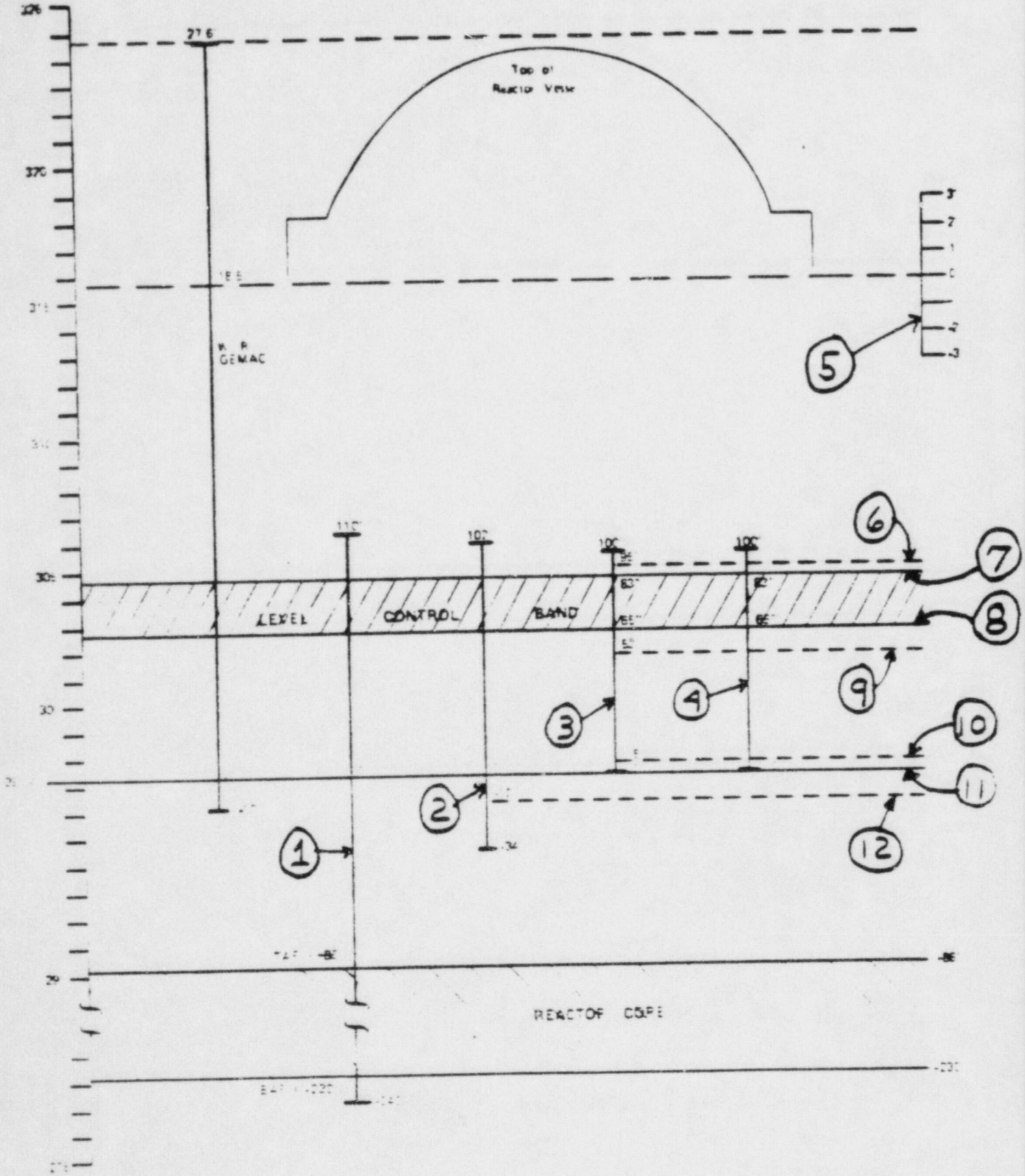


Figure II 2.1
Reactor Vessel Level Instrumentation

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

FACILITY: NINE MILE POINT

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/03/12

EXAMINER: LANGE, D.

NAME: MASTER

INSTRUCTIONS:

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	100.00			5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.00	100.00			TOTALS

FINAL GRADE _____%

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

_____
SIGNATURE

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

FACILITY: NINE MILE POINT

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/03/12

EXAMINER: LANGE, D.

NAME: _____

INSTRUCTIONS:

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.

ATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	100.00			5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.00	100.00			TOTALS

FINAL GRADE _____%

ll work done on this examination is my own. I have neither iven nor received aid.

_____
SIGNATURE

QUESTION 5.01 (2.75)

NOTE: Answer the following questions from a theoretical standpoint, not from a Nine Mile Point system design standpoint.

a. With the plant operating at 90 % power , extraction steam to the fifth stage feedwater heaters is removed. An engineer, observing that turbine load increased by 15 MWe after the extraction steam removal, concludes that this action has improved the plant's thermodynamic efficiency (NOT heat rate). Do you agree with this conclusion? Explain your answer fully. (2.00)

b. For the above condition, will reactor power (CMWT) increase, decrease, or remain the same . (Briefly Explain) (0.75)

QUESTION 5.02 (3.00)

Indicate whether the following will INCREASE or DECREASE reactivity during operation AND briefly EXPLAIN why.

a. Moderator temperature increases while below saturation temperature. (.75)
b. Fuel temperature increases. (.75)
c. Loss of a feedwater heater. (.75)
d. A sudden reduction in reactor primary system pressure. (.75)

QUESTION 5.03 (2.25)

For each of the events listed below, state which reactivity coefficient will respond first and if it adds positive or negative reactivity.

a. Relief Valve opening at 100 % power . (0.75)
b. Rod drop at 100 % power. (0.75)
c. Isolation of a feedwater heater string at 75 % power. (0.75)

QUESTION 5.04 (2.00)

Will the Recirculation Pumps have more NPSH at 4 % or 100 % power? Explain your answer fully. (2.00)

[

QUESTION 5.05 (2.00)

During a cooldown of the reactor vessel from outside the control room, reactor pressure decreased from 885 psig to 595 psig in one half hour. Has your reactor cooldown limit been exceeded ? (show all work) (2.00)

QUESTION 5.06 (1.50)

The discharge of the Feedwater Booster Pumps provide cold water injection into the moisture separator drain system. From a theoretical standpoint, why is this done ? (1.50)

QUESTION 5.07 (2.25)

Relationships exist between Centrifugal Pump parameters that are refered to as BASIC PUMP LAWS . Use these pump laws to answer the following.
A centrifugal pump is operating at 30,000 GPM , 3,000 RPM, with a discharge head of 200 psig. If the discharge head decreases to 50 psig, what are the new CAPACITY, SPEED, and POWER REQUIREMENTS of the pump. (2.25)

QUESTION 5.08 (2.50)

The Reactor has been at 100 % power for 90 days when a Reactor Scram occurs The temperature is maintained at 540 deg.F, and the Shutdown margin 10 min. after the scram is 1% delta K/K . Describe what happens to the Shutdown margin during the next three (3) days if the temperature is maintained at 540 degF. ? (2.50)

QUESTION 5.09 (2.25)

At NMP-Unit 1, Fuel thermal safety limits are established for the purpose of protecting fuel clad integrity during accident or steady state power operation.

a. For the 8x8 fuel, what is the LHGR limit, and why was it specifically established ? (1.00)

b. The actual LHGR of a fuel rod is dependent upon what (3) variables ? (0.75)

c. Why was the (M)APLHGR thermal limit established ? (0.50)

QUESTION 5.10 (1.50)

" The Central fuel rods are more likely to exceed the 2200 deg.F limit during a LOCA even though the edge and corner fuel rods have higher LOCAL PEAKING FACTORS ".---> TRUE or FALSE ? (Justify your answer) (1.50)

QUESTION 5.11 (1.50)

If the Main Condenser and associated systems were absolutely AIR TIGHT would there be any need for the Steam Jet Air Ejectors during full power operation ? (Explain your answer). (1.50)

QUESTION 5.12 (1.50)

A Rod Drop Accident is expected to be more severe at:

1. Power Operation
2. Hot Standby
3. Cold Conditions

(Choose the correct answer and justify your decision) (1.50)

TABLE II-3-1

PROPERTIES OF SATURATED STEAM AND SATURATED WATER (TEMPERATURE)

Temp F	Press. psia	Volume, ft³/lb			Enthalpy, Btu/lb			Entropy, Btu/lb x F			Temp F
		Water	Evap	Steam	Water	Evap	Steam	Water	Evap	Steam	
		v_f	v_{fg}	v_g	h_f	h_{fg}	h_g	s_f	s_{fg}	s_g	
32	0.08859	0.01602	3305	3305	−0.02	1075.5	1075.5	0.0000	2.1873	2.1873	32
35	0.09991	0.01602	2948	2948	3.00	1073.8	1076.8	0.0061	2.1706	2.1767	35
40	0.12163	0.01602	2446	2446	8.03	1071.0	1079.0	0.0162	2.1432	2.1594	40
45	0.14744	0.01602	2037.7	2037.8	13.04	1068.1	1081.2	0.0262	2.1164	2.1426	45
50	0.17796	0.01602	1704.8	1704.8	18.05	1065.3	1083.4	0.0361	2.0901	2.1262	50
60	0.2561	0.01603	1207.6	1207.6	28.06	1059.7	1087.7	0.0555	2.0391	2.0946	60
70	0.3629	0.01605	868.3	868.4	38.05	1054.0	1092.1	0.0745	1.9900	2.0645	70
80	0.5069	0.01607	633.3	633.3	48.04	1048.4	1096.4	0.0932	1.9426	2.0358	80
90	0.6981	0.01610	468.1	468.1	58.02	1042.7	1100.8	0.1115	1.8970	2.0085	90
100	0.9492	0.01613	350.4	350.4	68.00	1037.1	1105.1	0.1295	1.8530	1.9825	100
110	1.2750	0.01617	265.4	265.4	77.96	1031.4	1109.3	0.1472	1.8105	1.9577	110
120	1.6927	0.01620	203.25	203.26	87.97	1025.6	1113.6	0.1646	1.7693	1.9339	120
130	2.2230	0.01625	157.32	157.33	97.96	1019.8	1117.8	0.1817	1.7295	1.9112	130
140	2.8892	0.01629	122.98	123.00	107.95	1014.0	1122.0	0.1985	1.6910	1.8895	140
150	3.718	0.01634	97.05	97.07	117.95	1008.2	1126.1	0.2150	1.6536	1.8686	150
160	4.741	0.01640	77.27	77.29	127.96	1002.2	1130.2	0.2313	1.6174	1.8487	160
170	5.993	0.01645	62.04	62.06	137.97	996.2	1134.2	0.2473	1.5822	1.8295	170
180	7.511	0.01651	50.21	50.22	148.00	990.2	1138.2	0.2631	1.5480	1.8111	180
190	9.340	0.01657	40.94	40.96	158.04	984.1	1142.1	0.2787	1.5148	1.7934	190
200	11.526	0.01664	33.62	33.64	168.09	977.9	1146.0	0.2940	1.4824	1.7764	200
210	14.123	0.01671	27.80	27.82	178.15	971.6	1149.7	0.3091	1.4509	1.7600	210
212	14.696	0.01672	26.78	26.80	180.17	970.3	1150.5	0.3121	1.4447	1.7568	212
220	17.186	0.01678	23.13	23.15	188.23	965.2	1153.4	0.3241	1.4201	1.7442	220
230	20.779	0.01685	19.364	19.381	198.33	958.7	1157.1	0.3388	1.3902	1.7290	230
240	24.968	0.01693	16.304	16.321	208.45	952.1	1160.6	0.3533	1.3609	1.7142	240
250	29.825	0.01701	13.802	13.819	218.59	945.4	1164.0	0.3677	1.3323	1.7000	250
260	35.427	0.01709	11.745	11.762	228.78	938.6	1167.4	0.3819	1.3043	1.6862	260
270	41.855	0.01718	10.042	10.060	238.95	931.7	1170.6	0.3960	1.2769	1.6729	270
280	49.200	0.01726	8.627	8.644	249.17	924.6	1173.8	0.4098	1.2501	1.6599	280
290	57.550	0.01736	7.443	7.460	259.4	917.4	1176.8	0.4236	1.2238	1.6473	290
300	67.005	0.01745	6.445	6.466	269.7	910.0	1179.7	0.4372	1.1979	1.6351	300
310	77.67	0.01755	5.609	5.626	280.0	902.5	1182.5	0.4506	1.1726	1.6232	310
320	89.64	0.01766	4.896	4.914	290.4	894.8	1185.2	0.4640	1.1477	1.6116	320
340	117.99	0.01787	3.770	3.788	311.3	878.8	1190.1	0.4902	1.0990	1.5892	340
360	153.01	0.01811	2.939	2.957	332.3	862.1	1194.4	0.5161	1.0517	1.5678	360
380	195.73	0.01836	2.317	2.335	353.6	844.5	1198.0	0.5416	1.0057	1.5473	380
400	247.26	0.01864	1.8444	1.8630	375.1	825.9	1201.0	0.5667	0.9607	1.5274	400
420	308.78	0.01894	1.4808	1.4997	396.9	806.2	1203.1	0.5915	0.9165	1.5080	420
440	381.54	0.01926	1.1976	1.2169	419.0	785.4	1204.4	0.6161	0.8729	1.4890	440
460	466.9	0.0196	0.9746	0.9942	441.5	763.2	1204.8	0.6405	0.8299	1.4704	460
480	566.2	0.0200	0.7972	0.8172	464.5	739.6	1204.1	0.6648	0.7871	1.4518	480
500	680.9	0.0204	0.6545	0.6749	487.9	714.3	1202.2	0.6890	0.7443	1.4333	500
520	812.5	0.0209	0.5386	0.5596	512.0	687.0	1199.0	0.7133	0.7013	1.4146	520
540	962.8	0.0215	0.4437	0.4651	536.8	657.6	1194.3	0.7378	0.6577	1.3954	540
560	1133.4	0.0221	0.3651	0.3871	562.4	625.3	1187.7	0.7625	0.6132	1.3757	560
580	1326.2	0.0228	0.2994	0.3222	589.1	589.9	1179.0	0.7876	0.5673	1.3550	580
600	1543.2	0.0236	0.2435	0.2675	617.1	550.6	1167.7	0.8134	0.5196	1.3330	600
620	1786.9	0.0247	0.1962	0.2208	646.9	506.3	1153.2	0.8403	0.4689	1.3092	620
640	2059.9	0.0260	0.1543	0.1802	679.1	454.6	1133.7	0.8686	0.4134	1.2821	640
660	2365.7	0.0277	0.1166	0.1443	714.9	392.1	1107.0	0.8995	0.3502	1.2498	660
680	2708.6	0.0304	0.0808	0.1112	758.5	310.1	1068.5	0.9365	0.2720	1.2086	680
700	3094.3	0.0366	0.0386	0.0752	822.4	172.7	995.2	0.9901	0.1490	1.1390	700
705.5	3208.2	0.0508	0	0.0508	906.0	0	906.0	1.0612	0	1.0612	705.5

EQUATION SHEET

$f = ma$

$v = s/t$

Cycle efficiency = (Network out)/(Energy in)

$w = mg$

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

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

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

$NPSH = P_{in} - P_{sat}$

$\dot{m} \propto \rho AV$

$\Delta E = 931\ \Delta m$

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

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

$Q = UA\Delta h$

$Pwr = W_f \Delta h$

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

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

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

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

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

$SUR = 26.06/T$

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

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

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

$CR_1(1 - K_{eff_1}) = CR_2(1 - K_{eff_2})$

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

$M = 1/(1 - K_{eff}) = CR_1/CR_2$

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

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

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

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

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

$\ell^* = 10^{-5}$ seconds

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

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

$I_1 d_1 = I_2 d_2$

$I_1 d_1^2 = I_2 d_2^2$

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

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

$\Sigma = \sigma N$

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

$NPSH = \text{Static head} - h_f - P_{sat}$

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.

Miscellaneous Conversions

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

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 3.01 (3.00)

A. 1. Low level of 53"(.25) due to loss of inventory through the break(.25)
   2. Turbine trip (.25) due to the scram on DW pressure, level or valve position(0.25)
   3. 1900000 lbm/hr on either motor driven feed pump(0.25) due to feed system trying to regain level lost from the break(0.25)

B. discharge pressure(.5), feed pump(.5) (1.0)

C. To ensure the feed pump starts against a closed discharge valve. (1.0)

xxxpart A. Reactor can scram on High DW pressure, low level,MSL valve position

REFERENCE
OP-46 HPCI JCK-185

ANSWER 3.02 (1.50)

A. ion chamber(.1) Offgas isolation valve 'stack blocking valve' and drain valve shuts(.3)

B. scintillation(.1) None(.3)

C. G-M(.1) Normal Rx Bldg ventilation trips(.3) and Emergency ventilation starts(.3)

REFERENCE
Operation Technology, Module 5, Part 15, Process Rad Monitor JCK-188

ANSWER 3.03 (2.50)

a. System is initiated by all five reactor recirculation motor generator set drive motor breakers tripping. At this point, the displayed level comes from the triple-low (Rosemount) transmitter. Whenever one of the core spray isolation valves opens OR level falls below triple-low (-10"), the level displayed on the fuel zone meter comes from the fuel zone level transmitters. (2.0)

b. 'Core Level/Torus Temperature Monitor System Trouble' energizes. (0.5) (Verbatim not required) Also the digital readout on FZ indicator will flash.

REFERENCE
Operations Technology, mod.II, Chap.2, 4.49,50,51 EDH-309

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 3.04 (3.00)

a. Causes reactor level to DECREASE [0.25] due to the Level Control System having a STEAM FLOW/FEED FLOW ERROR, STEAM FLOW < FEED FLOW [0.5] resulting in a CLOSURE SIGNAL TO THE FEEDWATER CONTROL VALVES [0.25]. (1.0)

b. Causes reactor level to INCREASE [0.25] due to the Level Control System having a LEVEL ERROR, LEVEL SET > INDICATED LEVEL [0.5] resulting in an OPEN SIGNAL TO THE FEEDWATER CONTROL VALVES [0.25]. (1.0)

c. Reactor level should REMAIN CONSTANT [0.25] because the #13 FEEDWATER CONTROL VALVE WILL LOCK-UP [.75]. (1.0)
Note: Loss of electrical will not lock up valve.

REFERENCE
Operations Technology Mod V pt. 6, N1-OP-16 pg. 16 EDH-312

ANSWER 3.05 (3.00)

a. rod block
b. half-scram
c. rod block
d. full scram
e. half-scram
(0.6 each) (3.0)

REFERENCE
Operations Technology, Mod.III, pg. 20-22,34 EDH-314

ANSWER 3.06 (2.00)

Three valves are automatically positioned to regulate RBCLC temperature: RBCLC heat exchanger inlet valve, RBCLC heat exchanger bypass valve, and RBCLC heat exchanger service water outlet valve(1.0). Initially, the controller modulates the RBCLC inlet and bypass valves(.5). As the heat load increases further, the controller will cause the service water outlet valve to open more fully(.5). (2.0)

REFERENCE
Operations Technology, Module 5, Chapter 7 JCK-182

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 3.07 (2.00)

A. No(.25) This power system has the capability to be synchronized to its to its emergency supply(.75) prior to transferring therefore there is no interuption of power. (1.0)

B. ~~Yes~~ NO.(.25) Synch scopes installed. A half scram will result because the RPS trip bus supplied by that MG set loses power(.75) (1.0)

REFERENCE
NI-OP-48 Motor Generator Sets pg 2-5 JCK-184

ANSWER 3.08 (2.50)

A. Feed water control system input(.5)
MSIV closure(.5) (1.0)

B. 7" Hg (.5)

C. Reactor Mode switch in Startup or Refuel(.5) and reactor pressure is less than 600 psig(.5) (1.0)

REFERENCE
Operation Technology, Module 2, Part 9, Main Steam
Operation Technology, Module 4, Part 2, Coolant Isolation JCK-187

ANSWER 3.09 (3.00)

a. 1. Fuel Zone Ind.
& 3. Hi Lo, Lo-Lo Rosemount.
b. 4. Narrow Range GE-MAC.
5. Flange Level GE-MAC.
6. Turbine Trip Signal.

c. Drywell ~~Pressure~~ (Temperature)
Reactor Pressure
Reactor Building Temperature

7. High Level Alarm.
8. Low Level Scram.
10. Low-Low isolation.
11. Instrument Zero.
12. Low-Low-Low ADS.

8. Low level alarm

2. Lo-Lo-Lo-Rosemount

REFERENCE
NMP. Oper. Tech. Module 2, Ch. 2, pg.16,17,& fig.2-1.

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 3.10 (2.50)

a. 1. Continuity lights go out. 2. Alarm indication.
3. Milliamp meters on back of 09-03 indicate current flow to firing ckt.
4. Decrease in power.
5. Selected pump has red light indicating pump is running.
6. SBLC pressure > reactor pressure.
7. SBLC tank level decreasing. (any six at 0.25 each)

b. YES. (0.25) If the SBLC. tank level approaches zero (0.50) or the SBLC pump begins to loose discharge pressure. (0.50) This is indicated by fluctuation of pump amperage and press. Need SS approval.

REFERENCE
NMP . Oper. Tech. Module 4-81 and N1-OP-12

U.S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR REQUALIFICATION EXAMINATION

FACILITY: NINE MILE POINT

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/03/11

EXAMINER: BERRY, J.

NAME: _____

INSTRUCTIONS

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%

CATEGORY VALUE	% OF TOTAL	SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	100.00			2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
25.00	100.00			TOTALS

FINAL GRADE _____%

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

_____
SIGNATURE

QUESTION 2.01 (3.00)

Answer the following with regard to the Emergency Cooling System:

a. What are two sources of makeup to the Emergency Condenser Makeup Tank? (0.5)

b. What are the initiation signals for the system? Include setpoints. (1.0)

c. How may the automatic initiation feature be overridden? (.5)

d. After system initiation and pressure is < 1000 psig, How is the cooldown rate controlled? (1.0)

QUESTION 2.02 (2.00)

WHAT particular hazard is involved with extended light load operation of a Diesel Generator and HOW is this hazard minimized? (2.0)

QUESTION 2.03 (3.00)

Fill in the indicated blanks A-I:

AIR OPERATED VALVES	NORMAL POSITION DURING OPERATION	VALVE ACTION ON LOSS OF AIR	STATION EFFECT ON AIR LOSS
MSIV's	OPEN	CLOSED	REACTOR SCRAM ON VALVE POSITION
EMERGENCY VENTILATION	CLOSED	A.	B.
SPENT FUEL POOL LCV	THROTTLING	C.	RISE IN LEVEL IF COND. XFER PUMPS IN OPERATION
TURBINE BLDG CLC - TCV	THROTTLING	D.	E.
EMERG. CONDEN. COND. RETURN	CLOSED	F.	G.
MU TK TIE VALVE	OPEN	H.	I.

QUESTION 2.04 (2.00)

If the Main Condenser and associated systems were absolutely AIR TIGHT would there be any need for the Steam Jet Air Ejectors during full power operation? (Explain your answer) (2.0)

QUESTION 2.05 (3.00)

A. What conditions must be met to satisfy the logic for ADS initiation? Include setpoints and trip logic arrangement (2.0)

B. How would the system respond if MSERV #1 failed to open after proper logic actuation? (All other valves respond properly) (0.5)

C. What are two types of detectors used to provide positive indication of a leaking/lifted RELIEF VALVE? (exclude lights and annunciators) (0.5)

QUESTION 2.06 (3.00)

Concerning the High Pressure Cooling Injection system (HPCI);

a. What prevents an idle feedwater pump from starting and pumping water through a FULLY OPEN feedwater control valve following HPCI initiation? (0.75)

b. If a HPCI initiation occurs with NO LOSS of OFF-SITE POWER, state the affect on the following valves, pumps or components.
   1. Condensate and feedwater pumps that are running. (0.25)
   2. Idle feedwater pump. (0.25)
   3. Feedwater Control System (0.25)
   4. Feedwater pump controller # 11 (0.25)
   5. Feedwater pump controller # 12 (0.25)

c. In addition to a HPCI initiation being blocked by protective pump lockouts, list three (3) additional INTERLOCKS that will also PREVENT an automatic start. (1.0)

QUESTION 2.07 (2.00)

How is the integrity of ECCS piping inside the reactor vessel verified during normal operation. In your answer include: SENSING POINTS, SPECIFIC SYSTEMS WHOSE PIPING IS VERIFIED, WHY IT IS VERIFIED and the response of the instrumentation to a loss of integrity. (2.0)

QUESTION 2.08 (2.00)

Concerning the Generator Stator Cooling Water System;

a. What three (3) conditions will cause a Turbine Governor Runback? (Setpoints are required) (0.75)

b. Will an automatic Reactor Scram occur on a Governor runback trip signal? If yes, from what? If not, how could a subsequent scram be prevented? (0.5)

c. What is the importance of regulating flow within this system to maintain pressure between 22-28 psi? (0.75)

QUESTION 2.09 (2.00)

Concerning the CORE SPRAY system;

a. What protective design feature, within the core spray system, allows for running the core spray pumps at shutoff head without overheating them. (Explain fully- be specific) (0.5)

b. Pressure in the Core Spray piping is sensed in three different places. List these three sensing points, indicating what is being sensed and any automatic actions, alarms, or indications that are provided from them. (~~SETPOINTS ARE REQUIRED~~) delete (1.5)

QUESTION 2.10 (3.00)

Concerning Refueling Operations;

a. List six (6) methods available to verify proper fuel bundle orientation. (1.5)

b. Consider the REFUEL INTERLOCK alarm located on the ROD BLOCK MONITOR PANEL. List two (2) conditions, including interlocks, that this alarm could be indicating? (1.0)

c. Under normal operations, prior to fuel handling , Procedure N1-OP-34 has a prerequisite which states, 'The Fuel Pool key lock switch on the 'G' panel shall be placed to the Refuel position when handling fuel or irradiated fuel casks.' What is the purpose of doing this? (0.5

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 2.01 (3.00)

A. CST via Condensate transfer system, Fire Water System (0.5)

B. 1080 psig(.33) or
low low reactor level of 5 inches(.33) for a period of
10 seconds(.33) (1.0)

C. Shut the steam supply valve(s) (.5)

D. Alternate opening and closing one condensate return valve (1.0)

REFERENCE
Operations Technology, Module 4, Part 7, Emergency Cooling System
JCK-162

ANSWER 2.02 (2.00)

The accumulation of oil in the engine exhaust system called 'SOUPING'(.5) could result in a fire(.5). The engine is operated at some minimum load for a period of time ~~or~~ AND until inspection shows that the exhaust stack is clean(1.0). (2.0)

REFERENCE
N1-OP-45 Rev 6, pg 11 JCK-169

ANSWER 2.03 (3.00)

A. Open B. None C. Open

D. Open E. Decrease of temperature on closed loop water

F. Open G. Emergency Condenser in operation

H. Closed I. Isolates Make-up Tanks (from each other)

(0.33 each)

REFERENCE
N1-SOP-5, Rev. 3, pg. 2. 3.48 EDH-307

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 2.04 (2.00)

YES. (0.75) To maintain the removal of non-condensable gasses produced from the decomposition of water , activation products and noble gasses produced in the fuel and leaking into the coolant via. cladding cracks. (1.25)

REFERENCE
NMP. Oper. Tech. Module 5 pg. 41 to 45.

ANSWER 2.05 (3.00)

A. * High DW pressure(.25), 3.5 psig(.25), one out of two taken twice(.2)
   * Low-Low-Low water level(.25), -10 inches(.25), one out of two taken twice(.2)
   * Time delay timed out(.2), 120 seconds(.2), logic is 1 of 2(.2)

B. 5 seconds after the 120 second timer started(.25), if the primary valve #1 was not open, its backup valve #2 would open(.25)

C. Acoustic monitor (0.25)
   Temperature elements (0.25)

REFERENCE
Operation Technology, Module 4, Part 8, ADS

ANSWER 2.06 (3.00)

a. The flow control valve is prevented from opening following a HPCI initiation signal until feedwater pump discharge pressure is sensed. (0.75)

b.1. Remain in operation. (0.25)
   2. The idle FW Pump will start and be up to speed in approx. 10 sec.(.25)
   3. The FW control sys. will switch to single element control if it had been in three element. (0.25)
   4. Attempt to maintain RX. level at 65 " (0.25)
   5. " " " " 71 " (0.25)

c. 1. The feed pump automatic start is blocked by aux.oil pressure less than 8 psig. (0.33)
   2. The feed pumps will trip if suction pressure drops below 200 psig. or if aux. oil pressure drops below 3 psig. (0.33)
   3. The feedwater booster pumps automatic start is blocked by suction pressure less than 35 psig. (0.33)

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

REFERENCE
NMP. N1-OP-46 , HPCI. and OPER. TECH. Module #4, part #9.

ANSWER 2.07 (2.00)

A differential pressure sensor is used to confirm the integrity of the CORE SPRAY piping within the reactor vessel (between the inside of the vessel and the core shroud).
To continuously monitor the integrity of the core spray piping, a Delta P switch measures the pressure difference between the two loops, which is effectively the inside of each Core Spray sparager pipe, just outside of the Rx vessel. If the core spray sparager is intact, this pressure difference will be zero. If integrity is lost, this pressure differential will include the pressure drop across the steam seperator. Alarms at 5 psid in the control room (2.00)

REFERENCE
NMP. Oper. Tech. Module # 4, Part 10. Core Spray. pg.4-61.

ANSWER 2.08 (2.00)

a. High temperature- > 83 deg.C (0.25) Low pressure - < 17 psig. (
Low system flow < 442 gpm. (0.25)

b. NO, (0.25) , Immediately reduce Rx. Recirc. flow to minimum, in an attempt to prevent a scram. (0.50)

c. System flow is regulated to maintain inlet pressure low enough to prevent water from entering the stator windings in the event of a leak. (if a leak develops hydrogen will leak into the cooling water). (0.75)

REFERENCE
NMP. N1-OP-44, Gen. Stator Cooling Water Sys., pg. 1-5 .

ANSWER 2.09 (2.00)

a. A relief valve in each core spray loop provides a Minimum Flow Recirculation path to the TORUS when the pumps are running at shutoff head. (0.75)

b. 1. Pressure on the suction side of the Core Spray pumps. (0.25)
ALARM on panel K ~~at 2.5 psig~~. LOW SUCTION PRESSURE. (0.25)
2. Pressure is sensed downstream of the flow orifice. (0.25)
ALARM on panel K ~~at 225 psig~~. CORE SPRAY LOOP LOW PRESSURE. (0.25)
3. Pressure is sensed on the disch.of the core spray topping pumps.(0.25)
Provides ALARM ~~at 440 psig~~.CORE SPRAY LOOP HIGH PRESSURE and remote indication of core spray system pressure. (0.25)

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

REFERENCE
NMP.Oper. Tech. Module # 4 pg. 59-61 and N1-OP-2.

ANSWER 2.10 (3.00)

a. 1. Fuel assembly serial # are readable from the associated control rod.
   2. Lugs on the fuel assembly bail handle point at the associated control rod.
   3. Channel spacer buttons are above the associated control rod.
   4. Channel fastener spring clips are above the associated control rod.
   5. Gadolinium rods have longer end plugs which protrude through the upper tie plate.
   6. Overall core symmetry.

(0.25 for each correct answer)

b. 1. The mode switch is in refuel with one control rod withdrawn. (0.25) An attempt to move the refuel platform with a fuel element over the core will result in de-energizing of the hoist motor. (0.25)
   2. The mode switch is in refuel with the refuel platform loaded and over the core ;(0.25) This condition inserts a rod block to prevent control rod withdrawal. (0.25)

c. This places the fuel pool high radiation monitor, on the refueling bridge,on the emergency ventiation circuit (alarms at 1000 mr/hr).(0.50)

REFERENCE
Operations Technology, Module 2, Chapter 3., and N1-OP-34, Refueling Proc.

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR REQUALIFICATION EXAMINATION

FACILITY: NINE MILE POINT

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 85/03/11

EXAMINER: BERRY, J.

NAME :

INSTRUCTIONS

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%

CATEGORY VALUE	% OF TOTAL	SCORE	% OF CATEGORY VALUE	CATEGORY
~~25.00~~	100.00			1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
25.00	100.00			TOTALS

FINAL GRADE _____%

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

_____
SIGNATURE

QUESTION 1.01 (2.00)

MATCH the Failure Mechanism from column (1) AND the Limiting Condition from column (2) WITH the associated Power Distribution Limits (a-c) below. (A 'letter-number-number' sequence is sufficient.)

a. Linear Heat Generation Rate (LHGR)
b. Average Planer Linear Heat Generation Rate (APLHGR)
c. Minimum Critical Power Ratio (MCPR)

1 - FAILURE MECHANISM	2 - LIMITING CONDITION
1. FUEL CLAD CRACKING DUE TO LACK OF COOLING CAUSED BY DNB	1. 1% PLASTIC STRAIN
2. FUEL CLAD CRACKING DUE TO HIGH STRESS FROM PELLET EXPANSION	2. PREVENT TRANSITION BOILING
3. GROSS CLAD FAILURE DUE TO DECAY HEAT & STORED HEAT FOLLOWING A LOCA	3. LIMIT CLAD TEMP TO 2200 F

(2.0)

QUESTION 1.02 (1.50)

Concerning control rod worth during a reactor startup with 100% peak xenon versus a startup with xenon free conditions, WHICH STATEMENT IS CORRECT? JUSTIFY YOUR CHOICE. (1.5)

1. PERIPHERAL control rod worth will be LOWER during the 100% peak xenon startup than during the xenon free startup.
2. CENTRAL control rod worth will be HIGHER during the 100% peak xenon startup than during the xenon free startup.
3. PERIPHERAL control rod worth will be HIGHER during the 100% peak xenon startup than during the xenon free startup.
4. BOTH CENTRAL and PERIPHERAL control rod worth WILL BE THE SAME regardless of core xenon concentration.

QUESTION 1.03 (2.00)

Starting at 35% on IRM Range 2, reactor power increases for 7 minutes on a 70 second period. What will be the IRM indication after 7 minutes? What fraction of reactor power is this? (2.0)

QUESTION 1.04 (.50)

If the turbine exhaust pressure increases due to a change in circulating water temperature, which of the following is true? (0.5)

a. Exhaust quality increases and exhaust enthalpy decreases

b. Exhaust quality decreases and exhaust enthalpy decreases

c. Exhaust quality decreases and exhaust enthalpy increases

d. Exhaust quality increases and exhaust enthalpy increases

QUESTION 1.05 (~~3.00~~) 2

a. Determine the condenser hotwell subcooling (condensate depression) if the condenser vacuum is 27.9" Hg. and the condensate temperature is 90 degrees F. (1.0)

b. What is one disadvantage of condensate depression? (0.5)

c. How does increased condensate depression affect condensate pump net positive suction head? (0.5)

d. Give two examples of how you, as an operator, can increase condensate depression. (1.0)

QUESTION 1.06 (2.50)

The reactor has been operating at 100% power for one month when a scram occurs in which several control rods FAIL TO FULLY INSERT. Enough rods DO insert to bring the reactor subcritical at the time of scram. If reactor moderator temperature is maintained CONSTANT, and control rods are NOT moved, about HOW LONG will the operator have to wait before he can be reasonably sure that the reactor will remain subcritical? EXPLAIN. (2.5)

QUESTION 1.07 (2.00)

Although steam is known to be a poorer heat conductor than water, NUCLEATE BOILING is a BETTER heat transfer mechanism than SINGLE PHASE CONVECTION. Explain this apparent contradiction. (2.0)

QUESTION 1.08 (1.50)

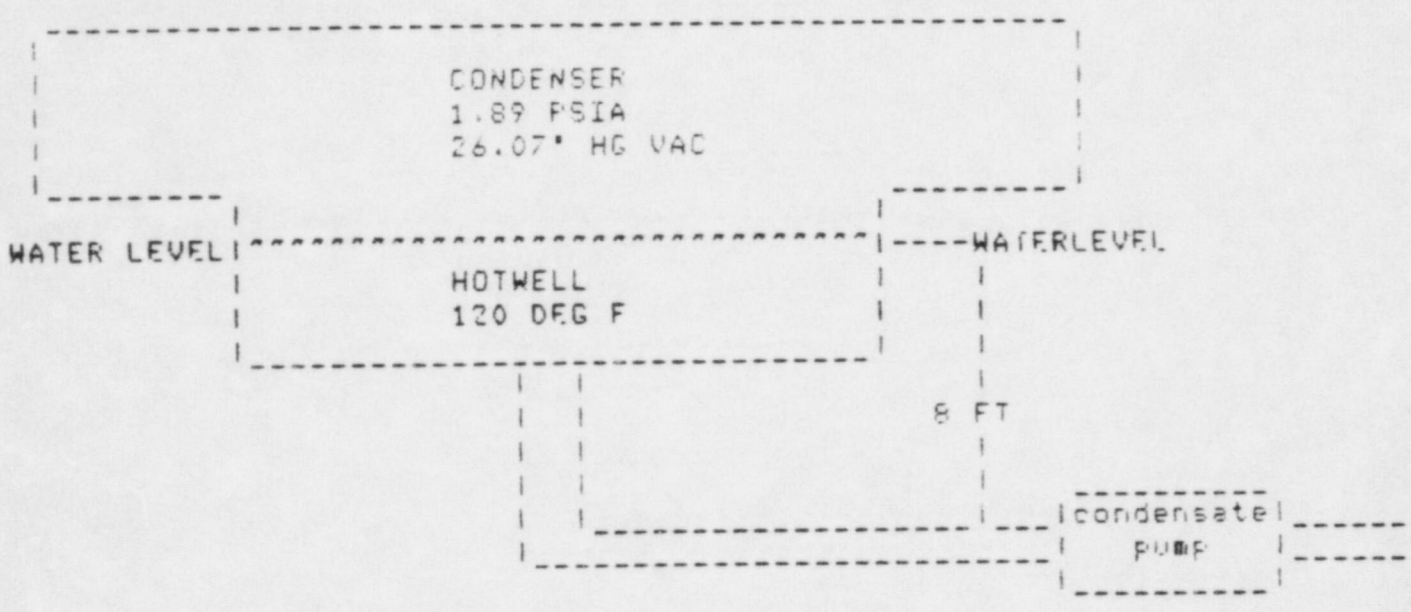
ANSWER THE FOLLOWING THREE MULTIPLE CHOICE QUESTIONS REGARDING CONTROL ROD EFFECTS ON CORE POWER.

a. Withdrawal of a 'deep control rod' generally has an appreciable effect on total core power output because: (0.5)
   1. The power increase is spread throughout the core by the relatively high void content in the area of withdrawal.
   2. The power increase is large due to the low void content present in the area of withdrawal.
   3. The power increase is spread throughout the core by the relatively high moderator temperature in the area of withdrawal.
   4. The power increase is large due to the minimal 'rod shadowing' present in the area of withdrawal.

b. Withdrawal of a 'shallow control rod' generally changes the power shape, while affecting total core power very little, because: (0.5)
   1. The power increase is small due to the shadowing effect of nearby control rods.
   2. The power increase is small due to the high void content present in the top of the core.
   3. The relatively large local power increase is off-set by an increase in void content.
   4. The relatively large local power increase is off-set by an increase in fuel temperature.

c. The 'reverse power effect' or 'reverse reactivity effect' occasionally observed when a shallow control rod is withdrawn one or two notches is due to: (0.5)
   1. A relatively large local power increase being off-set by a void related power decrease.
   2. A relatively large local power increase being off-set by a moderator temperature realted power decrease.
   3. A relatively small local power decrease due to the 'shadowing' effect of nearby control rods.
   4. A relatively small local power decrease due to increased local doppler effects.

QUESTION 1.09 (2.00)



Using the above figure, calculate:

a. Pressure at the condensate pump inlet (1.0)

b. Degrees F subcooling of the condensate in the hotwell (1.0)

SHOW ALL ASSUMPTIONS AND WORK - STEAM TABLES ARE ATTACHED

QUESTION 1.10 (1.50)

The moderator temperature coefficient is not considered to be a significant reactivity coefficient because its affect is limited primarily to the reactor startup range. Why is it not considered significant in the POWER RANGE during NORMAL operation? (1.5)

QUESTION 1.11 (3.00)

A fuel pin, over a period of time, has a uniform coating of corrosion products about 0.001 inches thick buildup on its surface. Assuming that power generation within the fuel pin REMAINS CONSTANT during the time of the buildup, would you expect the following temperatures to increase, decrease, or remain the same during the buildup? EXPLAIN EACH ANSWER.

a. Fuel temperature. (1.0)
b. Cladding temperature. (1.0)
c. Coolant temperature surrounding the lower portion of the fuel pin (prior to the onset of boiling). (1.0)

QUESTION 1.12 (2.00)

a. If the thermocouple in the discharge of a Electromatic Safety Relief Valve measures only the temperature of the discharged steam, what temperature would you expect to see on the recorder if an SRV was leaking while the reactor was at 100% power? EXPLAIN. (Steam Tables and Mollier Diagram are attached.) (1.0)
b. Would the steam flow through an SRV be seen on the Control Room steam flow instruments if the SRV FAILED OPEN during operation? EXPLAIN. (1.0)

QUESTION 1.13 (1.50)

Give ONE undesirable result of each of the following.(Be more specific than 'pump failure'):

a. Operating a centrifugal pump for extended periods of time with the discharge valve shut. (0.5)
b. Starting a centrifugal pump with the discharge valve full open. (0.5)
c. Operating a motor driven pump under 'PUMP RUNOUT' conditions. (0.5)

TABLE II-3-1

PROPERTIES OF SATURATED STEAM AND SATURATED WATER (TEMPERATURE)

Temp F	Press. psia	Volume, ft³/lb			Enthalpy, Btu/lb			Entropy, Btu/lb x F			Temp F
		Water	Evap	Steam	Water	Evap	Steam	Water	Evap	Steam	
		v_f	v_{fg}	v_g	h_f	h_{fg}	h_g	s_f	s_{fg}	s_g	
32	0.08859	0.01602	3305	3305	-0.02	1075.5	1075.5	0.0000	2.1873	2.1873	32
35	0.09991	0.01602	2948	2948	3.00	1073.8	1076.8	0.0061	2.1706	2.1767	35
40	0.12163	0.01602	2446	2446	8.03	1071.0	1079.0	0.0162	2.1432	2.1594	40
45	0.14744	0.01602	2037.7	2037.8	13.04	1068.1	1081.2	0.0262	2.1164	2.1426	45
50	0.17796	0.01602	1704.8	1704.8	18.05	1065.3	1083.4	0.0361	2.0901	2.1262	50
60	0.2561	0.01603	1207.6	1207.6	28.06	1059.7	1087.7	0.0555	2.0391	2.0946	60
70	0.3629	0.01605	868.3	868.4	38.05	1054.0	1092.1	0.0745	1.9900	2.0645	70
80	0.5068	0.01607	633.3	633.3	48.04	1048.4	1096.4	0.0932	1.9426	2.0359	80
90	0.6981	0.01610	468.1	468.1	58.02	1042.7	1100.8	0.1115	1.8970	2.0086	90
100	0.9492	0.01613	350.4	350.4	68.00	1037.1	1105.1	0.1295	1.8530	1.9825	100
110	1.2750	0.01617	265.4	265.4	77.98	1031.4	1109.3	0.1472	1.8105	1.9577	110
120	1.6927	0.01620	203.25	203.26	87.97	1025.6	1113.6	0.1646	1.7693	1.9339	120
130	2.2230	0.01625	157.32	157.33	97.96	1019.8	1117.8	0.1817	1.7295	1.9112	130
140	2.8892	0.01629	122.98	123.00	107.95	1014.0	1122.0	0.1985	1.6910	1.8895	140
150	3.718	0.01634	97.05	97.07	117.95	1008.2	1126.1	0.2150	1.6536	1.8686	150
160	4.741	0.01640	77.27	77.29	127.96	1002.2	1130.2	0.2313	1.6174	1.8487	160
170	5.993	0.01645	62.04	62.06	137.97	996.2	1134.2	0.2473	1.5822	1.8295	170
180	7.511	0.01651	50.21	50.22	148.00	990.2	1138.2	0.2631	1.5480	1.8111	180
190	9.340	0.01657	40.94	40.96	158.04	984.1	1142.1	0.2787	1.5148	1.7934	190
200	11.526	0.01664	33.62	33.64	168.09	977.9	1146.0	0.2940	1.4824	1.7764	200
210	14.123	0.01671	27.80	27.82	178.15	971.6	1149.7	0.3091	1.4509	1.7600	210
212	14.696	0.01672	26.78	26.80	180.17	970.3	1150.5	0.3121	1.4447	1.7568	212
220	17.186	0.01678	23.13	23.15	188.23	965.2	1153.4	0.3241	1.4201	1.7442	220
230	20.779	0.01685	19.364	19.381	198.33	958.7	1157.1	0.3388	1.3902	1.7290	230
240	24.968	0.01693	16.304	16.321	208.45	952.1	1160.6	0.3533	1.3609	1.7142	240
250	29.825	0.01701	13.802	13.819	218.59	945.4	1164.0	0.3677	1.3323	1.7000	250
260	35.427	0.01709	11.745	11.762	228.76	938.6	1167.4	0.3819	1.3043	1.6862	260
270	41.856	0.01718	10.042	10.060	238.95	931.7	1170.6	0.3960	1.2769	1.6729	270
280	49.200	0.01726	8.627	8.644	249.17	924.6	1173.8	0.4098	1.2501	1.6599	280
290	57.550	0.01736	7.443	7.460	259.4	917.4	1176.8	0.4236	1.2238	1.6473	290
300	67.005	0.01745	6.448	6.466	269.7	910.0	1179.7	0.4372	1.1979	1.6351	300
310	77.67	0.01755	5.609	5.626	280.0	902.5	1182.5	0.4506	1.1726	1.6232	310
320	89.64	0.01766	4.896	4.914	290.4	894.8	1185.2	0.4640	1.1477	1.6116	320
340	117.99	0.01787	3.770	3.788	311.3	878.8	1190.1	0.4902	1.0990	1.5892	340
360	153.01	0.01811	2.939	2.957	332.3	862.1	1194.4	0.5161	1.0517	1.5678	360
380	195.73	0.01836	2.317	2.335	353.6	844.5	1198.0	0.5416	1.0057	1.5473	380
400	247.26	0.01864	1.8444	1.8630	375.1	825.9	1201.0	0.5667	0.9607	1.5274	400
420	308.78	0.01894	1.4808	1.4997	396.9	806.2	1203.1	0.5915	0.9165	1.5080	420
440	381.54	0.01926	1.1976	1.2169	419.0	785.4	1204.4	0.6161	0.8729	1.4890	440
460	466.9	0.0196	0.9746	0.9942	441.5	763.2	1204.8	0.6405	0.8299	1.4704	460
480	566.2	0.0200	0.7972	0.8172	464.5	739.6	1204.1	0.6648	0.7871	1.4518	480
500	680.9	0.0204	0.6545	0.6749	487.9	714.3	1202.2	0.6890	0.7443	1.4333	500
520	812.5	0.0209	0.5386	0.5596	512.0	687.0	1199.0	0.7133	0.7013	1.4146	520
540	962.8	0.0215	0.4437	0.4651	536.8	657.5	1194.3	0.7378	0.6577	1.3954	540
560	1133.4	0.0221	0.3661	0.3883	562.4	625.3	1187.7	0.7625	0.6132	1.3757	560
580	1326.2	0.0228	0.2994	0.3222	589.1	589.9	1179.0	0.7876	0.5673	1.3550	580
600	1543.2	0.0236	0.2438	0.2675	617.1	550.6	1167.7	0.8134	0.5196	1.3330	600
620	1786.9	0.0247	0.1960	0.2208	646.9	506.3	1153.2	0.8403	0.4689	1.3092	620
640	2059.9	0.0260	0.1543	0.1802	679.1	454.6	1133.7	0.8686	0.4134	1.2821	640
660	2365.7	0.0277	0.1166	0.1443	714.9	392.1	1107.0	0.8995	0.3502	1.2498	660
680	2708.6	0.0304	0.0808	0.1112	758.5	310.1	1068.5	0.9365	0.2720	1.2086	680
700	3094.3	0.0366	0.0386	0.0752	822.4	172.7	995.2	0.9901	0.1490	1.1390	700
705.5	3208.2	0.0508	0	0.0508	906.0	0	906.0	1.0612	0	1.0612	705.5

EQUATION SHEET

$f = ma$

$v = s/t$

Cycle efficiency = (Network out)/(Energy in)

$w = mg$

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

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

$NPSH = P_{in} - P_{sat}$

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

$\dot{m} \propto \rho AV$

$\Delta E = 931\ \Delta m$

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

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

$Q = UA\Delta h$

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

$Pwr = W_f \Delta h$

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

$TVL = 1.3/\mu$

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

$HVL = -0.693/\mu$

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

$SUR = 26.06/T$

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

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

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

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

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

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

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

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

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

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

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

$\ell^* = 10^{-5}$ seconds

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

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

$I_1 d_1 = I_2 d_2$

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

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

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

$\Sigma = \sigma N$

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

$NPSH = \text{Static head} - h_1 - P_{sat}$

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.

Miscellaneous Conversions

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

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 1.01 (2.00)

	Failure Mechanism	Limited Condition
A. LHGR	2	1
B. APLHGR	3	3
C. MCPR	1	2

(6 answers req. @ 0.33each)

REFERENCE
Operations Technology Manual, Module 10 JCK-150

ANSWER 1.02 (1.50)

C is the correct choice (0.5)
The highest Xenon concentration will be in the center of the core [0.33] due to it's being the high flux region from the previous operating period [0.33]. This will increase the flux levels in the area of the peripheral control rods, thus increasing their worthg.[0.33]

REFERENCE
Operations Technology Manual, Module 1, Chap. 14

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 1.03 (2.00)

P=(Po)(e raised to t/T)

t=7 minutes= 420 seconds
T=70 seconds

P=(35)(e raised to 420/70)
P=(35)(403.4)
P=14119% on range 2
P=1411.9 on range 4
P=141.19 on range 6
P=14.119% on range 8
or
P=1.4119% on range 10 (1.5)

100 on range 10=10% of rated
So, 1.4119% on range 10 = 1.4119% of 10%
Fraction power=.14119% of rated or .0014119 of rated (0.5)

REFERENCE
Operations Technology Manual, Module 1, Chap 10 & Module 3, Chap. 3
NMP Question Bank
JCK-156

ANSWER 1.04 (.50)

d - Exhaust quality increases and exhaust enthalpy increases (0.5)

REFERENCE
Operations Technology, Mod. IX, Chap.2 EDH-297

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

ANSWER 1.05 ~~(3.00)~~ 2.0

A. 29.9" - 27.9" = 2" Hg absolute (0.25)
2" Hg absolute = .98 psia (0.25)
Tsat for .98 psia = 100 F (0.25)
100 F - 90 F = 10 F condensate depression (0.25) (1.0)
B. Plant efficiency is reduced (0.5)
C. NPSH increases (0.5)
D. Reduce turbine load, Increase circ water flow, raise condenser pressure, Decrease circ water temp., Adjust tempering gate position, Increase hotwell level (2 required) (1.0)

REFERENCE
Steam Tables, Fluid Mechanics and Thermodynamics Study Guide DNG102

ANSWER 1.06 (2.50)

60 - 70 hours (1.0)
It will take approx. 70 hours for the Xenon to peak and then decay after the scram. If the positive reactivity inserted by the decay of Xenon is less than the shutdown reactivity due to rods, then the reactor will remain subcritical. (1.5)

REFERENCE
SNPS Reactor Physics Module - Lesson 11, pages 7-154 to 7-161
Student Objectives #1 & 2

ANSWER 1.07 (2.00)

The bubbles caused by nucleate boiling serve to agitate the stagnent fluid film next to the surface, thus improving thermal conductivity.[1.0] Also, each bubble, as it leaves the surface, carries off more energy than is possible by natural convection.[1.0]

REFERENCE
Heat Transfer and Thermodynamics

ANSWER 1.08 (1.50)

B. - 1 [0.5]
D. - 3 [0.5]
C. - 1 [0.5]

ANSWERS -- NINE MILE POINT -85/03/11-BERRY, J.

REFERENCE
Reactor Physics Module

ANSWER 1.09 (2.00)

a. P = Patm + Pwater column (0.5)
= 1.89 psia + 8 ft / 2.3 ft / psia (0.4)
= 5.4 psia (0.1)

b. Interpolating from steam tables 1.8 - 2.0 psia (0.4) yields Psat for 1.89 psia to be 124 degrees F
Subcooling = Psat - Pactual (0.4)
Subcooling = 124 - 120 = 4 degrees subcooling (0.1)

REFERENCE
Fluid Mechanics
Heat Transfer and Thermodynamics

ANSWER 1.10 (1.50)

Once the reactor reaches the power producing range, pressure is controlled, and the moderator temperature changes very little over the power range, therefore the MTC effects are minimized. (1.5)

REFERENCE
Reactor Physics

ANSWER 1.11 (3.00)

a. Fuel temperature would increase [0.25] to get the needed delta T to transfer the heat to the coolant. The corrosion layer will require some delta T across it to transfer heat [0.25]

b. Cladding temperature would also increase [0.25] because the pin temperature increased and the cladding is now transferring heat to the corrosion film instead of the coolant.[0.25]

c. Coolant temperature remains the same [0.25] since it is a function of pressure, which is maintained constant by the ~~EHC~~ MHC system. [0.25]

REFERENCE
Heat Transfer

ANSWER 1.12 (2.00)

a. Since the steam would undergo adiabatic expansion, from the Mollier diagram the maximum temperature would be approx. 325 degrees F. (1.0)

b. ~~NO~~ YES (0.5) SRVs come off ~~before~~ After the flow restrictors which measure steam flow. (0.5)

REFERENCE

Reactor Theory
Main Steam System
Thermodynamics/Steam Tables

ANSWER 1.13 (1.50)

A. The pump will eventually add a sufficient amount of heat to the fluid to cause cavitation. Also will accept overheating of the pump. (1.0)

B. Could cause excessively long starting currents or water hammer if the downstream piping was not filled. (1.0)

C. Causes excessive motor amps to be drawn and the high current could cause damage to the motor windings. (1.0)

REFERENCE
Fluid Flow Basic Principles

Docket No. 50-220

Niagara Mohawk Power Corporation
ATTN: Mr. B. G. Hooten
Executive Director
Nuclear Operations
c/o Miss Catherine R. Seibert
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

SUBJECT: EXAMINATION REPORT NO. 50-220/85-03 (OL)

This transmits the Examination Report of Requalification Examinations conducted by USNRC Region I at Nine Mile Point, Unit 1 the week of March 11, 1985. At the exit interview held on March 14, 1985, the preliminary results of these examinations were discussed.

Although your overall Requalification Program rating is satisfactory, we note that some of the problems identified in this report have been brought to your attention by other reviews, and that you have committed to corrective action. NRC Region I will continue to monitor the progress of these actions as they relate to the improvement of your Requalification Program.

No reply to this letter is required. Your cooperation in this matter is appreciated.

Sincerely,

Samuel J. Collins, Chief
Projects Branch No. 2
Division of Reactor Projects

Enclosure: Examination Report No. 50-220/85-03 (OL) w/attachments

cc: w/enclosure and attachments:
T. Perkins, General Superintendent, Nuclear Generation
Plant Training Manager
Senior Resident Inspector
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
State of New York

cc w/o attachment to enclosures:
T. E. Lempges, Vice President, Nuclear Generation
D. Palmer, Manager of Quality Assurance
T. Roman, Station Superintendent
J. Aldrich, Supervisor Operations
W. Drews, Technical Superintendent
Troy B. Conner, Jr., Esquire
John W. Keib, Esquire
Director, Power Division

bcc w/o attachments to enclosure:
DRP Section Chief
Examiner
Chief, OLB/DHFS, NRR
OL File 12.0
Region I Docket Room (w/concurrences)
Master Exam File
D. Weiss, LFMB
A. J. Vinnola, EG&G Idaho

U. S. NUCLEAR REGULATORY COMMISSION REGION I
OPERATOR LICENSING EXAMINATION REPORT

EXAMINATION REPORT NO. 85-03 (OL)

FACILITY DOCKET NO. 50-220

FACILITY LICENSE NO. DPR-63

LICENSEE: Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

FACILITY: Nine Mile Point, Unit 1

EXAMINATION DATES: March 11 - 14, 1985

PREPARED BY: _____ _____
J. A. Berry, Lead Reactor Engineer (Examiner) Date

REVIEWED BY: _____ _____
R. M. Keller, Chief, Projects Section 1C Date

APPROVED BY: _____ _____
H. B. Kister, Chief, Projects Branch No. 1 Date

SUMMARY: As part of the NRC programmatic evaluation of Requalification Training at Nine Mile Point, Unit 1, NRC prepared written examinations were administered, in parts, to all facility personnel taking the Niagara Mohawk prepared annual requalification examinations the week of March 11, 1985. Additionally, oral requalification examinations were given to 11 licensed personnel, 7 SROs and 4 ROs.

REPORT DETAILS

TYPE OF EXAMS: Requalification

EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail
Written Exam Partial Exams	35/4	32/1
Oral Exam	4/0	6/1
Overall	35/4	32/1

1. CHIEF EXAMINER AT SITE:

   J. A. Berry, U.S. NRC - Region I

2. OTHER EXAMINERS:

   D. J. Lange, U.S. NRC - Region I
   F. J. Crescenzo, U.S. NRC - Region I
   T. L. Morgan, EG&G Idaho, Inc.
   D. E. Hill, EG&G Idaho, Inc.

3. REPORT:

   As part of the NRC's programmatic evaluation of Requalification Training at Nine Mile Point - Unit 1, NRC prepared written examinations were administered in parts, to all facility personnel taking the Niagara Mohawk prepared annual Requalification examinations the week of March 11, 1985. Additionally, oral requalification examinations were given to 11 licensed personnel, 7 SROs and 4 ROs.

The NRC written examination sections were administered as follows:

Monday, March 11 - RO Section 2 to 17 people
- SRO Sections 5 & 8 to 11 people

Tuesday, March 12 - RO Section 3 to 13 people
- SRO Section 6 to 12 people

Wednesday, March 13 - RO Sections 1 & 4 to 9 people
SRO Section 7 to 11 people

Overall, examination results were good. Five people failed NRC administered sections of the examinations, four RO's and one SRO, and one SRO failed the oral examination.

The comparison of scores on NRC sections vs. the facility sections indicated that the overall average score on the NRC exam (if sections were together) and facility exam were within 4% of each other. This is considered an acceptable range. Individual section comparisons indicated a wide disparity. Section 8 of the NRC and Facility exams were within .5% of each other in average score, but Sections 2, 3, and 6 were off by 6.71%, 9.91% and 6.56% respectively, with the NRC section scores being lower. Also, Sections 4 and 7 on the NRC exam had average scores 6.4% and 3.3% higher than the facility's sections.

The reasons for this disparity are not evident. It appears that the higher scores on the NRC Sections 4 and 7 may be due to the facility's sections being overly long, but the other section differences cannot be so explained. Probable causes may be the tension involved in taking an NRC exam, more operationally oriented (not memorization) type questions on the NRC exam, or the difference in question "style" between the two exams.

In addition to the conduct of examinations, the evaluation also consisted of a review of the NMP-1 Requalification Program Annual examinations prepared by the facility, and discussions with licensed operators and training staff members regarding the Requalification Program.

The Annual Requalification examinations prepared by the facility were considered to meet NRC requirements, but were not considered to be of high quality. Problems with the examinations included; double jeopardy questions, excessive length, many unnecessary theory calculations and questions having no relation to an operator's job, and simplistic short answer questions which failed to provide an adequate measure of depth of knowledge. The facility's Requalification examination question bank is poor, and it is felt its use contributed to the problems with the examination. To Niagara Mohawk's credit, they have identified the problems with the existing exam question bank, and have began a task to upgrade it. Significant improvement is expected in next years exam.

Discussions with licensed operators indicate that there is dissatisfaction with the Requalification Program. Problems sited included; to much emphasis on theory that is not operationally oriented, too much self-study or reading, and unchallenging and uninteresting presentation of subject matter. These matters have been previously brought to the attention of Niagara Mohawk management by other reviews of the program. Niagara Mohawk has committed to a course of corrective action. NRC Region I will monitor the progress of the action over the next year. It is felt that the addition of a plant specific simulator training program to the Requal program will aid in improving the program.

Overall, the Nine Mile Point, Unit 1 Requalification program is satisfactory. NMPC has already begun to correct many of the programmatic problems identified. No further NRC involvement in the program is planned this year, other than monitoring of the changes being made to improve its quality.

4. Personnel Present at Exit Interview:

NRC Personnel

J. Linville, Chief, Reactor Projects Section 2C, DRP
J. A. Berry, Lead Reactor Engineer (Examiner) DRP
D. J. Lange, Reactor Engineer (Examiner), DRP
F. J. Crescenzo, Reactor Engineer (Examiner), DRP
A. J. Luptak, Resident Inspector, NMP-1

NRC Contractor Personnel

D. E. Hill, EG&G Idaho, Inc.
T. L. Morgan, EG&G Idaho, Inc.

Facility Personnel

T. W. Roman, Station Superintendent - NMP-1
K. F. Zollitsch, Training Superintendent, Niagara Mohawk
J. C. Aldrich, Operations Supervisor, NMP-1
T. Wood, Training Supervisor, NMP-1
J. T. Pavel, Asst. Training Superintendent, Niagara Mohawk
R. Seifried, Operations Training Instructor
M. Dooley, Operations Training Instructor
M. Jones, Operation Supervisor, NMP-2

5. Summary of Comments made at exit interview:

- The Chief Examiner noted that there was one person who was not a clear pass on the oral examinations.

- A discussion was held regarding Niagara Mohawk's commitment to implementation of upgrades in their Requalification Program based on previous audits.

Attachments: Written Examination(s) and Answer Key(s) (SRO/RO)