U.S. NUCLEAR REGULATORY COMMISSION

REGION I

EXAMINATION REPORT NO:	94-04 (OL)
FACILITY DOCKET NO:	50-245
FACILITY LICENSE NO:	DPR-21
LICENSEE:	Northeast Nuclear Energy Company P. O. Box 270 Hartford, Connecticut 06141-0270
FACILITY:	Millstone Nuclear Power Station, Unit 1
EXAMINATION DATES:	January 11 and February 1-3, 1994
EXAMINERS:	T Walker, Senior Operations Engineer J. Caruso, Operations Engineer J. Hanek, Examiner (EG&G) D. Willoughby, Examiner (EG&G)

CHIEF EXAMINER: Terry E Walker 3/14/94
T. Walker, Senior Operations Engineer, BWR Section, Operations Branch, DRS — Date

APPROVED BY: Richard J Conte 3/16/94
Richard J. Conte, Chief, BWR Section, Operations Branch, DRS — Date

EXAMINATION SUMMARY

Report No. 50-245/94-04

Initial examinations were administered to two Senior Reactor Operator (SRO) instant applicants, one SRO upgrade, and three Reactor Operator (RO) applicants. All of the applicants passed all portions of the examinations. Generic weaknesses were noted on both the written examination and operating test as feedback to the training program (section 3.2). During the examinations, the examiners observed varied levels of performance on a Job Performance Measure to feed and bleed with LPCI on a loss of shutdown cooling pumps apparently due to procedural deficiencies (section 3.3).

The reference material submitted for examination preparation and license application submittal did not meet the some of the guidelines of ES-201 of NUREG-1021, "Examiner Standards," Rev. 7 (sections 3.4 and 3.5). The facility licensee took an initiative to identify the root cause of the late withdrawal of four applicants. The root cause was a programs weakness in training the operators to integrate system knowledge and varying plant conditions (section 3.5).

DETAILS

1.0 INTRODUCTION

The NRC staff administered initial examinations to two Senior Reactor Operator (SRO) instant applicants, one SRO upgrade applicant, and three Reactor Operator (RO) applicants. The examinations were administered in accordance with NUREG-1021, "Examiner Standards," Revision 7.

2.0 PREEXAMINATION ACTIVITIES

The facility reviewed the written examinations in the Region I office during the week of December 27, 1993. The simulator scenarios and Job Performance Measures (JPMs) were validated during the week of January 10, 1994, on the facility's simulator and in the plant. The facility staff who were involved with these reviews signed security agreements to ensure that the initial examinations were not compromised. No significant problems were identified with simulator fidelity once the examination materials were validated. However, there were several malfunctions and events, suggested by the NRC examiners, that the simulator was unable to model. Licensee personnel were very cooperative and suggested other methods for accomplishing most of the evaluation objectives modeled requiring modification of the examination materials. The most significant simulator fidelity problems are listed in Attachment 5.

3.0 EXAMINATION RESULTS AND RELATED FINDINGS, OBSERVATIONS AND CONCLUSIONS

3.1 Examination Results

The results of the examinations are summarized below:

	SRO Pass/Fail	RO Pass/Fail
Written	2/0	3/0
Operating	3/0	3/0
Overall	3/0	3/0

Note: The SRO upgrade applicant waived the written examination.

3.2 Facility Generic Strengths and Weaknesses

The following is a summary of the strengths and weaknesses noted during initial examination administration. This information is being provided to aid the licensee in upgrading their training program.

Written Examination

Strengths:

All of the applicants responded correctly to multiple questions; however, no specific generic strengths were noted.

Weaknesses:

Questions related to the following specific knowledge/ability topics were missed by at least half of the applicants indicating a generic weakness in the subject:

- Knowledge of the proper method for communication during a security event;
- Ability to determine the cause of improper isolation condenser operation;
- Understanding of the effect of a rod worth minimizer control rod withdrawal block due to an incorrect rod movement;
- Ability to determine the effect of an APRM failure on the rod block monitor;
- Knowledge of the reason that reactor power decreases when reactor water level is lowered.

Operating Tests: Simulator Portion

Strengths:

The following items were noted as strengths in the performance of all or most of the crew members during the dynamic simulator scenarios:

- RO turnovers when leaving the surveillance area;
- SRO crew briefs;
- Acknowledgement and repeat back of communications.

Weaknesses:

The following item was noted as a weaknesses in the performance of all or most of the crew members during the dynamic simulator scenarios:

- Ability to determine the availability and validity of reactor water level indication, specifically failure to monitor reactor pressure vessel (RPV) saturation temperature.

Operating Tests: Walkthrough Portion

No generic strengths or weaknesses were noted in applicant performance on the walkthrough portion of the examinations; however, the following examination related problems were noted:

- Varied levels of performance on a job performance measure (JPM) to feed and bleed with LPCI on a loss of shutdown cooling pumps (procedure problems are discussed in section 3.3);

- Lack of familiarity with the NRC walkthrough examination process, specifically the administrative section.

3.3 Procedures

During the examinations, the examiners observed varied levels of performance on a JPM to feed and bleed with LPCI on a loss of shutdown cooling pumps. These variations in performance appeared to be due to procedural deficiencies rather than knowledge or ability deficiencies. None of these deficiencies resulted in unsatisfactory performance of the task. However, most of the applicants had difficulty establishing a conservative cooldown rate, and several allowed reactor water level to drop significantly prior to establishing a feedpath. The following deficiencies were noted in ONP 525E, "Degraded Fire in Shutdown Cooling Pump Area," Rev. 2:

- Step 2.6.6 of ONP 525E directs the operator to establish RBCCW flow through the selected shutdown cooling (SDC) heat exchanger, but does not give any specific direction or reference another procedure for instructions. A note above step 2.6.6 states that initial flowrate should remain within the capacity of one RBCCW pump. OP 305, "Reactor Shutdown Cooling System," places two RBCCW pumps in service when placing SDC in operation. It is not clear how many RBCCW pumps should be placed in service when implementing ONP 525E.

- Step 2.6.13 of ONP 525E directs the operator to maintain reactor coolant temperature out of the SDC heat exchanger greater than 100 degrees F, but does not indicate which recorder point should be monitored. OP 305 gives specific direction to monitor recorder point 7 or 8 (heat exchanger cooling water out) when placing SDC in service. Several applicants incorrectly monitored RBCCW outlet temperature (point 7 or 8) instead of reactor coolant temperature (point 3 or 4) when establishing the feed and bleed in accordance with ONP 525E. Allowing RBCCW discharge temperature from the SDC heat exchanger to exceed 180 degrees F (cautioned not to exceed in OP 305) could result in damage to the RBCCW system or ineffective heat removal. Failure to maintain reactor coolant temperature out of the SDC heat exchangers above 100 degrees F could result in reactor vessel temperature decreasing below the 86 degree F limit with the head tensioned.

- Step 2.6.9 of ONP 525E directs the operator to station an operator at 1-LP-9A to throttle the valve manually while initially establishing flow, but does not direct any action. A note above step 2.6.9 recommends that 1-LP-9A be throttled almost closed until a conservative cooldown rate is established. Step 2.6.14 of ONP 525E directs the operator to maintain reactor vessel level by opening 1-LP-10A and manually throttling 1-LP-9A. LP-9A is normally open. If it is not throttled closed prior to opening LP-10A, cold water from the torus at full flow from one LPCI pump will be injected to the vessel when LP-10A is opened. However, the bleed path is established prior to opening LP-10A to establish the feedpath. Reactor water level will decrease until flow is established through LP-9A and LP-10A. It is not clear whether LP-9A should be throttled closed at step 2.6.9 or after LP-10A is opened in step 2.6.14.

- A note above step 2.6.14 of ONP 525E states that it is preferable to throttle with 1-LP-9A, 1-LP-9B, 1-SD-4A, and 1-SD-4B. LP-9B is closed in step 2.6.3 of ONP 525E and no other direction is given to operate the valve. Opening LP-9B would establish a drain path into the B loop of LPCI and interfere with the gravity drain from the vessel to the torus.

3.4 Reference Material

A number of reference materials necessary for examination preparation were not sent to the examiners in the initial submittal of material. These materials included the Emergency Operating Procedure (EOP) flowcharts; training material on EOPs, Emergency Plan, administrative topics, and fire protection; indices for the lesson plans, JPMs, and administrative procedures; and radiation protection procedures. There were also a number of administrative procedures, for which licensed operators were responsible, that were not sent with the reference materials. Additionally, some of the training materials that were sent were missing pages. All of the missing materials were provided promptly when requested by the examiners.

Some minor problems were encountered during administration of the examinations because changes had been made to the references after validation. Procedure changes just prior to examination administration need to be brought to the attention of the Chief Examiner to ensure the validity of the examination materials prepared.

3.5 License Applications

Preliminary license applications were submitted less than two weeks prior to administration of the written examinations. This delayed submittal did not cause any problems for this examination because the Chief Examiner was aware of the waiver requests. However, preliminary applications should be submitted at least 30 days prior to the examinations so that any waiver requests can be processed and problems with eligibility resolved in accordance with NUREG-1021.

Four license applications were not submitted as planned because the applicants' results on the facility licensee-administered final exam did not meet the facility licensee's expectations. The facility licensee conducted a root cause analysis to determine why the results of the final exam were less than expected by training department management. The root cause analysis appears to have been thorough and complete. The root cause was a program weakness in training the operators to integrate system knowledge and varying plant conditions. Several specific recommendations were identified to address the identified problems. The training department also performed an item analysis on the final exam to determine the reason between the disparity between the average class grades and the grades on the final exam.

4.0 EXIT MEETING

An exit meeting was conducted on February 4, 1994. Preliminary generic strengths and weaknesses on the operating tests were presented. The NRC examiners noted that plant conditions were good with a refueling outage in progress and there were no problems with plant access. The problems with reference material and license application submittal were also discussed.

Persons contacted and attendees at the exit meeting are listed below:

Licensee Personnel

* P. Przekop, Manager, Unit 1 Operations
M. Brown, Director, Nuclear Training
* R. Heidecker, Manager, Operator Training (Millstone Unit Nos. 1 and 2)
* C. Tabone, Supervisor, Operator Training Unit 1
* P. Fitzgerald, Licensed Operator Initial Training Coordinator
R. Schmidtknecht, Shift Supervisor, Unit 1
M. Jacobs, Licensed Operator Requalification Training Coordinator
* D. Harris, Licensing Engineer

NRC Personnel

* T. Walker, Senior Operations Engineer
* J. Caruso, Operations Engineer

* Denotes those present for the exit meeting on February 4, 1994.

Attachment 1

U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
REACTOR OPERATOR LICENSE
REGION 1

CANDIDATE'S NAME: ____________________

FACILITY: Millstone 1

REACTOR TYPE: BWR-GE3

DATE ADMINISTERED: 94/01/13

INSTRUCTIONS TO CANDIDATE:

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires a final grade of at least 80%. Examination papers will be picked up four (4) hours after the examination starts.

TEST VALUE	CANDIDATE'S SCORE	%	
99.50 ~~100.00~~ TW 1/24/94	______ FINAL GRADE	___ %	TOTALS

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

____
Candidate's Signature

ANSWER SHEET

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

MULTIPLE CHOICE

001 a b c d ____

002 MATCHING

a ____

b ____

c ____

~~d ____~~ deleted TBW 1/24/94

MULTIPLE CHOICE

003 a b c d ____

004 a b c d ____

005 a b c d ____

006 a b c d ____

007 a b c d ____

008 a b c d ____

009 a b c d ____

010 a b c d ____

011 a b c d ____

012 a b c d ____

013 a b c d ____

014 a b c d ____

015 a b c d ____

016 a b c d ____

017 a b c d ____

018 a b c d ____

019 a b c d ____

020 a b c d ____

021 a b c d ____

022 a b c d ____

023 a b c d ____

024 MATCHING

a ____

b ____

c ____

d ____

MULTIPLE CHOICE

025 a b c d ____

026 a b c d ____

027 a b c d ____

028 a b c d ____

029 a b c d ____

030 a b c d ____

031 a b c d ____

032 a b c d ____

033 a b c d ____

034 a b c d ____

035 a b c d ____

ANSWER SHEET

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

036 a b c d ____

MULTIPLE CHOICE

037 a b c d ____

038 MATCHING

a ____

b ____

c ____

d ____

MULTIPLE CHOICE

039 a b c d ____

040 a b c d ____

041 a b c d ____

042 a b c d ____

043 a b c d ____

044 a b c d ____

045 a b c d ____

046 a b c d ____

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067 a b c d ____

068 a b c d ____

069 a b c d ____

070 a b c d ____

071 a b c d ____

072 a b c d ____

073 a b c d ____

074 a b c d ____

075 a b c d ____

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

076 a b c d ____

MULTIPLE CHOICE

077 a b c d ____

078 a b c d ____

079 a b c d ____

080 a b c d ____

081 a b c d ____

082 a b c d ____

083 a b c d ____

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093 a b c d ____

094 a b c d ____

095 a b c d ____

096 a b c d ____

097 a b c d ____

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

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination, the following rules apply:

1. Cheating on the examination will result in a denial of your application and could result in more severe penalties.

2. After you complete the examination, sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination.

3. To pass the examination, you must achieve a grade of 80 percent or greater.

4. The point value for each question is indicated in parentheses after the question number.

5. There is a time limit of 4 hours for completing the examination.

6. Use only black ink or dark pencil to ensure legible copies.

7. Print your name in the blank provided on the examination cover sheet and the answer sheet.

8. Mark your answers on the answer sheet provided and do not leave any question blank.

9. If the intent of a question is unclear, ask questions of the examiner only.

10. Restroom trips are permitted, but only one applicant at a time will be allowed to leave. Avoid all contact with anyone outside the examination room to eliminate even the appearance or possibility of cheating.

11. When you complete the examination, assemble a package including the examination questions, examination aids, and answer sheets and give it to the examiner or proctor. Remember to sign the statement on the examination cover sheet.

12. After you have turned in your examination, leave the examination area as defined by the examiner.

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QUESTION: 001 (1.00)

Manipulation of controls which directly affect reactivity or power level of the reactor may be performed by:

a. a Control Operator in the license reactivation process without further supervision.

b. a Control Operator in license training under the direction of an actively licensed Senior Control Operator.

c. a Plant Equipment Operator utilizing local manual control of a recirculation MG set, in direct communication with the Reactor Operator at the controls.

d. an unlicensed on-shift Shift Technical Advisor, at a remote shutdown equipment location, under the direction of an actively licensed Control Operator.

QUESTION: 002 1.50 (~~2.00~~) TSW 1/24/94

Match the Millstone posting of radiological controlled areas in Column A with the minimum radiation dose rate that would require the posting, listed in Column B.

The items from column B may be used once, more than once, or not at all and only a single answer may occupy one answer space.

COLUMN A AREAS	COLUMN B RADIATION DOSE RATE
a. Radiation Area	1. ≥ 0.5 mrem/hr
b. Neutron Radiation Area	2. ≥ 1.0 mrem/hr
c. High Radiation Area	3. ≥ 2.5 mrem/hr
~~d. Extremely High Radiation Area~~ deleted TSW 1/24/94	4. ≥ 100 mrem/hr
	5. > 500 mrem/hr
	6. > 1000 mrem/hr

QUESTION: 003 (1.00)

A Northeast Utilities (NU) employee who is 24 years old today, can have a maximum lifetime total effective dose equivalent (TEDE) in accordance with NU TEDE administrative goals of:

a. 24 rem.

b. 30 rem.

c. 48 rem.

d. 72 rem.

QUESTION: 004 (1.00)

Given the following conditions:

-- An operator is performing a whole body frisk using a portable frisker, RM-14/HP 210.
-- Background radiation count rate in the contaminated area is at the maximum allowed for using the frisker.

What is the count rate (background + actual) at which the operator is considered to be contaminated?

a. 100 counts per minute

b. 300 counts per minute

c. 400 counts per minute

d. 500 counts per minute

QUESTION: 005 (1.00)

Which one of the following conditions require the Control Room Operators to immediately don the Scott Air Pack equipment?

a. Group 1 isolation on 120% steam line flow.

b. Group 1 isolation on main steam line low pressure.

c. Receipt of annunciator, "CONTROL ROOM AIR INTAKE HIGH RADIATION/DOWNSCALE."

d. Fire in the Main Control Board.

QUESTION: 006 (1.00)

The "A" CRD pump is properly red tagged for electrical trouble shooting. Electrical Maintenance has determined that the panel control switch requires replacement.

The red tag on the control switch:

a. is cleared before removal of the switch from the panel.

b. is removed from the switch and attached near the panel hole.

c. remains with the old switch until transferred to the new switch by the electricians.

d. is removed under a "temporary lift" until the new switch is installed.

QUESTION: 007 (1.00)

Plant personnel have been dispatched to search Unit 1 in response to a security event.

The method of communicating to the Shift Supervisor is:

a. Plant Page

b. Security Radio

c. Operations Radio

d. Plant Telephone

QUESTION: 008 (1.00)

Given the following list of safety requirements and protective equipment.

1. Stand clear of the cubicle
2. Stand to the side of the cubicle and use the left hand for the actual racking.
3. Hard hat with face shield
4. Flash Jacket with hood pulled over hard hat
5. Linesman gloves
6. Rubber mat to stand on

Select the minimum that are required when racking in a 4160V "ITE Magnablast" electrical circuit breaker, if the remote racking operator is NOT available.

a. 2 and 5

b. 1, 3 and 5

c. 2, 3, 4 and 5

d. 1, 3, 4 and 6

QUESTION: 009 (1.00)

If a Control Operator believes that a manual scram is required he/she shall announce the pertinent plant conditions, their recommendation and:

a. perform the action only when directed by the Shift Supervisor.

b. perform the action only when directed by the Supervising Control Operator.

c. wait until they receive an acknowledgement from either the Shift Supervisor or Supervising Control Operator before performing the action.

d. perform the required action without waiting for an acknowledgement from either the Shift Supervisor or Supervising Control Operator.

QUESTION: 010 (1.00)

Following the trip of a 480V circuit breaker an operator may:

a. attempt one reset only if he/she understands the cause of the trip.

b. attempt one reset after he/she explains the cause to the Supervising Control Operator.

c. attempt one reset only after receiving permission from the Shift Supervisor.

d. attempt one reset if he/she understands the cause of the trip and a second attempt after receiving permission from the Shift Supervisor.

QUESTION: 011 (1.00)

Which one of the following conditions allow the operator at the controls to leave the Surveillance Area as defined in ACP 6.01, unattended during normal plant operations?

a. To complete the Shift Surveillance Schedule logging.

b. To enter the Shift Supervisor's office to obtain a panel key.

c. To obtain a print from the outage office in order to verify a valve lineup.

d. To verify the receipt of an annunciator on a back panel.

QUESTION: 012 (1.00)

According to ACP 6.01, "Control Room Procedure", what is the MINIMUM Incident Classification made in accordance with EPIP 4701, which would meet the definition of an "Accident Situation"?

a. General Emergency

b. Site Emergency

c. Alert

d. Unusual Event

QUESTION: 013 (1.00)

Select the normal suction point for the Control Rod Drive Hydraulic Pumps during cold shutdown conditions with the condensate system secured.

a. Hotwell Reject Line

b. Condensate Storage Tank

c. Demineralized Water Header

d. Condensate Transfer Header

QUESTION: 014 (1.00)

Which one of the following rod motion sequences does NOT open the "settle valve," (control valve 120)?

a. Single Notch Insert

b. Emergency Insert

c. Continuous Insert

d. Continuous withdrawal using Rod Out Notch Override

QUESTION: 015 (1.00)

Following receipt of a "ROD CONTROL TIMER INOP" annunciator, control rod motion is inhibited.

This inhibit is accomplished by which of the following methods?

a. Deenergizes the "ROD MOVEMENT CONTROL SWITCH."

b. A RWM rod block and deselecting the selected control rod

c. Deselecting the selected control rod and preventing the selecting of any other control rod

d. Deenergizing the Reactor Manual Control System directional control valves

QUESTION: 016 (1.00)

Which of the following describes the makeup function necessary to support Feedwater Coolant Injection (FWCI)?

a. The ECT pump starts on the FWCI initiation signal, the discharge valve, MW-96A modulates to maintain hotwell level.

b. The ECT pump starts on the FWCI initiation signal, the discharge valve, MW-96A opens when hotwell level decreases to less than 12 inches.

c. The ECT pump starts and the discharge valve, MW-96A opens on the FWCI initiation signal.

d. The ECT pump starts and the discharge valve, MW-96A opens when hotwell level decreases to less than 12 inches.

QUESTION: 017 (1.00)

Which of the following is the maximum number of LPRM inputs to APRM channel 1 (16 LPRM inputs) which may be bypassed without the channel being inoperable? Assume the minimum number of inputs per level requirement is met.

a. 3

b. 5

c. 8

d. 11

QUESTION: 018 (1.00)

During operation at 100% power a problem develops with reactor recirculation pump "A" upper motor bearing temperature.

Which of the following is the maximum time the plant can operate in single loop with the drywell deinerted, in order to perform the necessary repairs before it must be in COLD SHUTDOWN?

a. 8 hours

b. 12 hours

c. 24 hours

d. 48 hours

QUESTION: 019 (1.00)

A reactor startup is in progress. While attempting to withdraw the twelfth control rod to position 36 an INOP message is displayed on the rod worth minimizer (RWM) operator display with the rod at position 16.

After bypassing the RWM the operator can:

a. continue the startup with no restrictions.

b. continue the startup after an engineer verifies the control rod program.

c. continue the startup if both rod block monitor channels are operable.

d. not continue the startup until the RWM is operational.

QUESTION: 020 (1.00)

During a scram from 100% power, the hydraulic control unit accumulator piston for control rod 26-27 does NOT move because the accumulator piston is mechanically bound.

Select the response of control rod 26-27.

a. The control rod will not insert.

b. The control rod will insert until reactor and accumulator pressures equalize and then stop.

c. The control rod will only partially insert using charging water pressure.

d. The control rod will fully insert using reactor pressure.

QUESTION: 021 (1.00)

During operation at 100% power, the operator, when comparing recirculation pump seal parameters, notes the following changes from the previous round of readings:

-- seal #2 cavity pressure has decreased
-- the 1st seal leakage low flow alarm was received
-- both seal temperatures have increased

Select the cause of these changes.

a. #1 seal has failed

b. #1 seal tubular orifice is plugging

c. #2 seal has failed

d. #2 seal tubular orifice is plugging

QUESTION: 022 (1.00)

Select the reason the pump discharge valve is closed following a recirculation pump trip.

a. Prevent reverse flow through the pump from short circuiting the core, thus reducing total core flow.

b. The LPCI loop select logic will not close the discharge valve if the pump is not operating.

c. Prevent reverse pump rotation.

d. To lower recirculation pump seal temperatures.

QUESTION: 023 (1.00)

Select the reactor water cleanup (RWCU) system design feature concerning reject flow control valve, CU-33 that will prevent draining the system when isolated.

a. Automatic closure on high downstream pressure.

b. Automatic closure on low upstream pressure.

c. Interlocked closed when the system isolation valves 1-CU-2, 2A, 3, 5 and 28 are closed.

d. Interlocked closed if both the main and auxiliary RWCU pumps are not operating.

QUESTION: 024 (2.00)

Match the full core display control rod positions in column A with the indicating light colors in column B.

The items from column B may be used once, more than once, or not at all and only a single answer may occupy one answer space.

COLUMN A POSITION	COLUMN B LIGHT COLORS
a. Full In	1. Amber
b. Full In Over Travel	2. Green
c. Full Out	3. White
d. Full Out Over Travel	4. Red
	5. Green Top Bars
	6. Red Center Bars

QUESTION: 025 (1.00)

With the plant operating at 75% power, the operator has just placed the Channel "A" rod block monitor (RBM) meter function switch on CRP 937 to the "Block" position.

Placing the switch to "Block" will:

a. display the current backup high RBM trip setpoint on the panel meter.

b. send an immediate RBM control rod block signal to the reactor manual control system.

c. remove all existing RBM rod block signals being sent to the reactor manual control system.

d. display the trip reference level fed to the high trip unit on the panel meter.

QUESTION: 026 (1.00)

Select the normal and alternate power supplies to the Control Rod Drive (CRD) pump motors.

a. "A" Normal Bus 14A Alternate Bus 14E
"B" Normal Bus 14C

b. "A" Normal Bus 14E Alternate Bus 14F
"B" Normal Bus 14H

c. "A" Normal Bus 14A Alternate Bus 14B
"B" Normal Bus 14C

d. "A" Normal Bus 14E Alternate Bus 14H
"B" Normal Bus 14F

QUESTION: 027 (1.00)

Which one of the following describes the response of the scram discharge volume pilot air valves and vent and drain valves following a trip of RPS channel "B."

a. No pilot air valves reposition. Both sets of vent and drain valves remain open.

b. One pilot air valve repositions. Both sets of vent and drain valves remain open.

c. One pilot air valve repositions. One set of vent and drain valves close.

d. Two pilot air valves reposition. One set of vent and drain valves close.

QUESTION: 028 (1.00)

Which one of the following "ROD OUT NOTCH OVERRIDE SWITCH" positions alone or in combination with the "ROD MOVEMENT CONTROL SWITCH" will bypass a malfunctioning rod sequence timer?

a. "EMER ROD IN"

b. "NOTCH OVERRIDE"

c. "OFF" and "ROD IN"

d. "NOTCH OVERRIDE" and "ROD OUT NOTCH"

QUESTION: 029 (1.00)

During operation at 65% reactor power a spurious electrical fault causes the "A" reactor recirculation pump discharge valve to close to the 88% open position.

The automatic response of the reactor recirculation system is:

a. the "A" pump trips.

b. scoop tube lockup on both pumps.

c. only the "A" pump speed runs back to 32% of maximum.

d. speed runs back to 32% of maximum on both pumps.

QUESTION: 030 (1.00)

Select the statement which describes the initial changes to core d/p and reactor water level following an increase in recirculation pump speed.

a. Core d/p decreases with a corresponding decrease in water level.

b. Core d/p decreases with a corresponding increase in water level.

c. Core d/p increases with a corresponding decrease in water level.

d. Core d/p increases with a corresponding increase in water level.

QUESTION: 031 (1.00)

Twenty (20) seconds after a small Loss Of Coolant Accident coincident with a loss of normal power the following plant conditions exist:

-- All Automatic systems have functioned as designed
-- Drywell pressure is 5 psig
-- Reactor vessel level is -65 inches

Select the status of the LPCI pumps under these conditions.

a. No LPCI Pumps are running

b. All LPCI pumps are running

c. "A" and "C" LPCI pumps are running

d. "B" and "D" LPCI pumps are running

QUESTION: 032 (1.00)

Thirty (30) seconds after an automatic initiation of LPCI the operator attempts to close LP-7A, LPCI Heat Exchanger Bypass Valve.

Select the response of LP-7A.

a. The valve will not move from the open position.

b. The valve will close and remain closed.

c. The valve will close and reopen when the hand switch is released.

d. The valve will start to close until dual indication is received, then stop at that position.

QUESTION: 033 (1.00)

Which one of the following describes the logic sequence for the LPCI Loop Select if only the "A" reactor recirculation pump is running at the time of initiation?

a. Trip signal is sent to only the "A" recirculation pump, close signal sent to the recirculation cross tie and bypass valves selected on CRP 905, when reactor pressure decreases to less than 900 psig, the two second timer starts.

b. Trip signal is sent to both recirculation pumps, close signal sent to the recirculation cross tie and bypass valves selected on CRP 905, when reactor pressure decreases to less than 900 psig the two second timer starts.

c. Trip signal is sent to both recirculation pumps and the two second timer starts immediately.

d. Trip signal is sent to only the "A" recirculation pump and the two second timer starts immediately.

QUESTION: 034 (1.00)

Which of the following describes the response of the Feed Regulating Valves to an accident signal coincident with a loss of normal power (LNP)?

a. "Lock Up" as is due to a loss of control signal.

b. Close when feed pump discharge pressure decreases to less than 300 psig.

c. Fail closed from a loss of control air pressure.

d. Shift to "Flow Control" when feed flow increases above 30%.

QUESTION: 035 (1.00)

Following an automatic actuation of Feedwater Coolant Injection (FWCI) the operator has been directed to determine plant status and restore FWCI to standby readiness if allowed.

The following conditions exist:

-- "B" feedwater string pumps selected and operating
-- total feed flow 30%
-- narrow range GEMAC level +30 inches and decreasing

Why is the operator cautioned against depressing the "FEEDWATER FLOW CONTROL RESET" pushbutton with these conditions?

a. It will shift the feedwater regulating valves to the "Level Control" mode.

b. It will shift the feedwater regulating valves to the "Flow Control" mode.

c. It will trip the "B" feedwater pump.

d. It will reset the FWCI initiation signal.

QUESTION: 036 (1.00)

Select the normal and backup supplies of makeup water to the isolation condenser shell side during normal standby operation.

	NORMAL	BACKUP
a.	Demineralized water	Condensate transfer
b.	Fire water	Demineralized Water
c.	Condensate transfer	Demineralized water
d.	Fire water	Condensate transfer

QUESTION: 037 (1.00)

Following a failure of the isolation condenser to automatically initiate at -48" RPV water level, 1-IC-3, Outboard Return valve was manually opened locally from the reactor building. An initial reactor pressure decrease was observed after opening 1-IC-3. Ten minutes later the control room operator observes that reactor pressure is not decreasing.

Select the cause of this abnormal condition.

a. Decay heat is still greater than the design capacity of the isolation condenser.

b. The shell side water has heated up and is not removing as much heat.

c. The buildup of non-condensibles in the tube bundles is inhibiting condensation and natural circulation.

d. A mechanical stop on 1-IC-3 limits the amount of opening when manually operated.

QUESTION: 038 (2.00)

Match the isolation condenser valves in Column A with the power supplies listed in Column B.

The items from column B may be used once, more than once, or not at all and only a single answer may occupy one answer space.

COLUMN A VALVES	COLUMN B POWER SUPPLY
a. Inboard Steam Supply Valve, 1-IC-1	1. MCC-E3
b. OutBoard Steam Supply Valve, 1-IC-2	2. MCC-F3
c. Outboard Condensate Supply Valve, 1-IC-3	3. DC-11A-1
d. Inboard Condensate Return Valve, 1-IC-4	4. DC-11A-2
	5. DC-101-AB-1
	6. DC-101-AB-2

QUESTION: 039 (1.00)

Following a small break LOCA the Automatic Pressure Relief (APR) system failed to actuate. Reactor Pressure is stable at 300 psig with level at -60 inches and slowly decreasing. All core spray system components functioned as per design.

Which of the following describes the Core Spray system status?

a. Pumps have started but are not injecting because the "Injection" valves, CS-5A and 5B have not opened.

b. Pumps have started, "Injection" valves, CS-5A and 5B have opened and the system is injecting.

c. Pumps have started, "Injection" valves, CS-5A and 5B have opened but the system is not injecting.

d. Pumps have not started, "Injection" valves, CS-5A and 5B have not opened.

QUESTION: 040 (1.00)

Core Spray system operability testing is in progress with "Pump Disch" valve, CS-4A closed and "Injection" valve, CS-5A open.

Select the automatic response of the these valves if a valid core spray initiation signal is received in this configuration.

a. CS-5A will remain open, CS-4A will remain closed.

b. CS-5A will close, CS-4A will remain closed.

c. CS-5A will close then CS-4A will open. CS-5A will reopen when the reactor pressure permissive signal is satisfied.

d. CS-5A will remain open, CS-4A will open when the reactor pressure permissive signal is satisfied.

QUESTION: 041 (1.00)

Standby liquid control system #2 is injecting at the Technical Specification minimum flow rate. The initial volume of the liquid poison tank was at 46% which corresponds to the Technical Specification required minimum volume.

Select the minimum time required to inject the "cold shutdown boron weight" (step FSQ 25).

a. 19 minutes

b. 25 minutes

c. 31 minutes

d. 47 minutes

QUESTION: 042 (1.00)

A reactor startup is in progress with IRMs on range 2. While withdrawing SRM detectors, the operator inadvertently selects IRM detector 12 drive unit.

Select the response of the IRM system.

a. IRM 12 detector will not withdraw because companion APRM 2 is downscale.

b. IRM 12 detector will withdraw and "IRM channel A HI HI FLUX/INOP" annunciator will be received.

c. A control rod block is generated when the detector begins to withdraw.

d. No action occurs because the range selector switch for IRM 12 is on range 2 and the mode switch is not in "RUN."

QUESTION: 043 (1.00)

Given the following plant conditions:

-- fraction of rated power (FRP) 0.87
-- maximum fraction of limiting power density (MFLPD) 0.87
-- total core flow 55.2 mlbm/hr
-- total recirculation flow 25.8 mlbm/hr

What is the required setpoint of the APRM channel "A" flow biased scram?

a. 104.2

b. 106.7

c. 108.2

d. 109.2

QUESTION: 044 (1.00)

The reactor was operating at 100% power when a scram occurred and one bypass valve failed open. Reactor pressure is decreasing.

The operator should attempt to close the bypass valve by:

a. depressing the Vacuum Trip #2 pushbutton.

b. positioning the Bypass Valve Opening Jack to the "LOWER" position.

c. positioning the Bypass Valve Test Switch to the "TEST" position.

d. shifting pressure control to the manual pressure regulator (MPR).

QUESTION: 045 (1.00)

A drywell pressure increase resulting from a loss of drywell cooling has started the diesel generator (DG). The operator places the Diesel Generator LOCA Start Bypass Switch in the "BYPASS" position and stops the DG. The operator then restores drywell cooling and drywell pressure decreases to 1 psig.

Which of the followirg will start the DG under these conditions?

a. "START/STOP" switch on CRP 908 or loss of voltage on busses 14E and 14F.

b. Loss of voltage on busses 14E and 14F or reactor low-low water level.

c. "START/STOP" switch on CRP 908 or high drywell pressure.

d. Reactor low-low water level or high drywell pressure.

QUESTION: 046 (1.00)

The rod worth minimizer (RWM) has applied a control rod withdrawal block due to an incorrect rod movement.

What control rod motion is possible with the above condition?

a. A full-out control rod may be withdrawn for a coupling check.

b. The rod causing the block may only be withdrawn.

c. Any control rod in the same group as the rod causing the block may be inserted.

d. No control rod motion is possible until the RWM is placed in "Bypass."

QUESTION: 047 (1.00)

The operator has placed the reactor water cleanup (RWCU) EOP keylock switches (color brown) on CRP 902 to "BYPASS."

Select the statement that lists all RWCU isolation signals that are bypassed by this action.

a. Reactor water low-low

b. Standby liquid control system initiation

c. Reactor water low-low and non-regenerative heat exchanger outlet high temperature

d. Reactor water low-low and standby liquid control system initiation

QUESTION: 048 (1.00)

A reactor startup following a refueling outage is being performed. The first eight control rods in the established rod withdraw sequence are at a final notch position of 36.

Select the method used to verify these eight control rods are coupled to their drive assembly.

a. By exercising these control rods after the reactor is critical and verifying instrument response.

b. By electrically disconnecting the insert directional control valve, 123 and verifying "stall flow" is less then 3.5 gpm when first attempting to withdraw the rod.

c. By verifying a rod drift alarm is NOT received when first attempting to pull the rod from position 00.

d. By attempting to pull each control rod past position 48, verifying the Rod Overtravel Annunciator is NOT received, then inserting back to position 36.

QUESTION: 049 (1.00)

Given the following conditions:

-- The plant is at 55% power
-- APRM 3 fails "Downscale"
-- No operator actions have been taken

Select the expected automatic response of the rod block monitor (RBM) system.

RBM channel 7:

a. automatically shifts to APRM 2.

b. output trip functions are bypassed.

c. generates a channel "Downscale" trip.

d. generates a channel "Inoperative" trip.

QUESTION: 050 (1.00)

Which one of the following conditions will trip the control room ventilation system supply fan HVH-8 and exhaust fan HVT-11?

a. Manual initiation of the standby gas treatment system.

b. Control room Halon discharge

c. High chlorine sensed in the outside air supply

d. High reactor building ventilation radiation

QUESTION: 051 (1.00)

The plant is operating at 100% power with Off-Gas radiation monitor, 1705-03B deenergized and red tagged for maintenance.

Annunciator alarm "OFF-GAS HI HI RADIATION" is received on CRP 904.

Select the Off-Gas system response if all automatic functions operate as designed.

a. Off-Gas isolation valve will immediately close.

b. Off-Gas flow to the stack will divert to the 30 minute holdup line.

c. Off-Gas isolation valve will close in 15 minutes.

d. Off-Gas isolation valve will remain open indefinitely.

QUESTION: 052 (1.00)

The Control Room Operator records the following integrator data, six hours after the previous data was taken:

-- Drywell floor drain sump; gallons pumped since last reading = 576 gallons
-- Drywell equipment drain sump; gallons pumped since last reading = 1764 gallons

Select the Reactor Coolant System Unidentified Leakage rate.

a. 1.6 gpm.

b. 3.3 gpm.

c. 4.9 gpm.

d. 6.5 gpm.

QUESTION: 053 (1.00)

During operation at 100% power, primary containment is being vented with the "B" train of standby gas treatment (SBGT). Valve 1-AC-10, "Exhaust from Drywell and Torus to SBGT Isolation" is open.

Select the response of the SBGT system if an automatic actuation signal is received with this system configuration.

"A" train starts, . . .

a. reactor building exhaust isolation valves, 1-SG-1A and 1B remain closed, valve 1-AC-10 remains open.

b. reactor building exhaust isolation valves, 1-SG-1A and 1B open, valve 1-AC-10 closes.

c. reactor building exhaust isolation valves, 1-SG-1A and 1B open, valve 1-AC-10 remains open.

d. "B" train stops, reactor building exhaust isolation valves, 1-SG-1A and 1B open, valve 1-AC-10 closes.

QUESTION: 054 (1.00)

When determining actual core flow in single recirculation pump operation, the operator is directed to increase running recirculation pump speed slightly while observing flow in the idle loop.

The total core flow indicated on the dual-pen recorder on CRP 905 would be correct if flow through the idle loop:

a. increases.

b. decreases.

c. goes to zero.

d. does not change.

QUESTION: 055 (1.00)

The Reactor Building "HVAC ISOLATIONS-RX BLDG" keylock switches (color turquoise) have been placed in "BYPASS" on CRP 902 as directed by EOP 590.17.

What function is provided when these switches are placed in the "BYPASS" position?

a. Allows operation of reactor building HVAC with high reactor building ventilation exhaust radiation.

b. Allows operation of the reactor building HVAC transfer fans if the exhaust fans are isolated.

c. Bypasses the reactor building ventilation exhaust radiation, high drywell pressure and low RPV water level isolations of reactor building HVAC.

d. Bypasses the high drywell pressure and low RPV water level isolations of reactor building HVAC only.

QUESTION: 056 (1.00)

Select the statement that describes all of the protective actions resulting from a valid main steam line radiation signal greater than seven times full power background sensed on channels "A", "B", and "C."

a. Reactor Scram generated by high main steam line radiation, Group 1 isolation and isolation of the mechanical condenser vacuum pump.

b. Reactor scram generated by MSIV position, Group 1 isolation and isolation of the mechanical condenser vacuum pump.

c. Half scram RPS channel "A" and PCIS channel "A" Group 1 isolation signal.

d. Reactor Scram generated by high main steam line radiation, and Group 1 isolation.

QUESTION: 057 (1.00)

Given the following fire protection system conditions:

- -- Control switches for the jockey, diesel, unit 1 and unit 2 fire pumps are in "AUTO"
- -- A rupture in the fire header decreased header pressure to 90 psig
- -- All expected automatic actions have occurred
- -- The rupture was isolated within 10 minutes
- -- Fire header pressure has increased to 110 psig
- -- No additional operator actions have been taken

Select the expected status of the fire protection system.

The following fire protection system pumps are running:

a. Diesel Driven Fire Pump
Unit 1 Motor Driven Fire Pump

b. Jockey Pump
Diesel Driven Fire Pump

c. Jockey Pump
Unit 2 Motor Driven Fire Pump

d. Unit 1 Motor Driven Fire Pump
Unit 2 Motor Driven Fire Pump

QUESTION: 058 (1.00)

Given the following conditions:

-- A Traversing In-core Probe (TIP) trace is in progress
-- A loss of coolant accident in the drywell has just occurred

How is Primary Containment Integrity established with these conditions?

a. The shear valve must be manually fired, cutting the TIP detector cable.

b. The ball valve automatically closes, cutting the TIP detector cable.

c. The TIP detector must be manually withdrawn and the shear valve fired.

d. The TIP detector automatically withdraws and the ball valve closes.

QUESTION: 059 (1.00)

Given the following conditions:

-- Reactor Mode Switch is in "REFUEL"
-- Control rod 22-31 is withdrawn for testing
-- Rod Select Power Switch is in the "ON" position

Which one of the following will generate a rod block with these conditions?

a. Depressing the rod select pushbutton for control rod 38-31.

b. Refueling bridge over the core.

c. Turning the rod select power switch to "OFF".

d. Refueling bridge console is deenergized.

QUESTION: 060 (1.00)

Which of the following systems injects inside the reactor vessel shroud?

a. Control Rod Drive Return

b. Standby Liquid Control

c. Reactor Feedwater

d. Low Pressure Coolant Injection

QUESTION: 061 (1.00)

Given the following plant conditions:

-- reactor has scrammed
-- a reserve station services transformer (RSST) "lockout" has occurred
-- bus 14D to 14F tie breaker has failed closed

Select the automatic response of the emergency diesel generator (EDG).

a. The EDG will not start.

b. The EDG will start, the output breaker will not close.

c. The EDG will start, the output breaker will close and remain closed

d. The EDG will start, the output breaker will close, then trip open and "lockout."

QUESTION: 062 (1.00)

A loss of drywell cooling has occurred. Drywell pressure increases to 2.5 psig. Drywell temperature increases to 160 degrees F.

Which one of the following lists all sections of the Emergency Procedures that must be entered?

a. All sections of EOP 580, "Primary Containment Control."

b. Primary Containment Pressure/Drywell Temperature (PCP/DWT) section of EOP 580, "Primary Containment Control."

c. Primary Containment Pressure/Drywell Temperature (PCP/DWT) section of EOP 580, "Primary Containment Control" and all sections of EOP 570, "RPV Control."

d. All sections of EOP 570, "RPV Control" and all sections of EOP 580, "Primary Containment Control"

QUESTION: 063 (1.00)

Select the torus water level at which the EOPs consider the torus to drywell vacuum breakers submerged.

a. 13.8 ft.

b. 17.5 ft.

c. 24.0 ft.

d. 25.5 ft.

QUESTION: 064 (1.00)

A steam leak has occurred in the drywell.

- Drywell temperature is 210 degrees F.
- RPV pressure is 450 psig
- Narrow Range ATWS indicate -42 inches

Select the condition that describes level condition of the Narrow Range ATWS indicators.

a. Actual RPV level is below the lower tap for the instrument.

b. Actual water level is greater than indicated on the level instrument.

c. Actual level cannot be determined from the instrument due to potential flashing of the reference leg.

d. Indicated level is accurate for trend information but is not accurate for actual determination of level.

QUFSTION: 065 (1.00)

A loss of Torus Water Level is occurring due to a leak in the torus. The operator has determined that level cannot be maintained above 10.5 ft.

Why is emergency depressurization required if level cannot be maintained above 10.5 ft?

a. The bottom of the downcomers is at 10.5 ft.

b. The Safety Relief Valves discharge at 10.5 ft.

c. Below 10.5 ft the torus has insufficient heat capacity for depressurization.

d. Below 10.5 ft the measuring capability for torus water temperature is lost.

QUESTION: 066 (1.00)

Step LP-8 of EOP 575, "Failure to Scram", specifies systems to be used to maintain water level.

Why is Core Spray not used in this condition to maintain level?

a. Core spray water quality is inferior to the systems listed.

b. At this point in the ATWS, reactor pressure precludes use of core spray.

c. Core spray injects inside the core shroud.

d. Core spray lacks precise flow control to avoid rapid level changes.

QUESTION: 067 (1.00)

A steam leak has occurred in the drywell. EOP 580, Primary Containment Control, is being implemented. Initiation of drywell sprays is delayed due to LPCI pumps being required for Adequate Core Cooling.

The following conditions exist when drywell sprays are initiated.

- Drywell pressure = 15 psig
- Drywell bulk temperature = 300 degrees F

After drywell sprays are initiated the operator observes drywell pressure decreasing to the left side of the DW Spray Initiation Limit.

Select the correct response to this condition.

a. Continue spraying until pressure decreases to 2 psig.

b. Attempt to throttle drywell spray to maintain Drywell pressure and Drywell temperature within the DW Spray Initiation Limit

c. Secure Drywell spray when the DW Spray Initiation Limit is reached.

d. Continue spraying until drywell pressure decreases to 9 psig.

QUESTION: 068 (1.00)

A fuel element failure initiated a main steam line isolation and a scram signal. The scram signal failed to insert control rods. Torus cooling was initiated.

The following conditions exist.

- Torus water level = 11 feet
- Torus water temperature = 200 degrees F and increasing
- RPV pressure was rapidly reduced to 500 psig from 1000 psig 15 minutes ago.

Which of the following actions shall be taken if torus temperature increases to 210 F?

a. Emergency Depressurize.

b. Reduce RPV pressure to 400 psig in accordance with allowable cooldown rates.

c. Reduce RPV pressure to 400 psig immediately. Cooldown rate limits may be exceeded.

d. Wait until temperature increases to 213 F then reduce RPV pressure.

QUESTION: 069 (1.00)

Instrumentation Technicians were performing maintenance on the TIP machines when one of the TIP sources was exposed causing high radiation conditions in the Reactor Building.

Radiation monitors are reading as follows:

- TIP Cubicle (Channel 10) = 10,000 mr/hr
- TIP Drive Mechanism (Channel 11) = > 100 mr/hr
- CRD Removal Hatch Area (Channel 12) = 1000 mr/hr
- North CRD Hydraulic Control Area = > 100 mr/hr
  (Channel 13)

What actions are required per EOPs?

a. Shutdown the reactor and concurrently perform actions of ONP 509, Excessive Radioactive Levels.

b. Scram and Enter EOP 570 Sht. 1, RPV Control.

c. Continue to operate and concurrently perform actions of ONP 509, Excessive Radioactive Levels.

d. Scram and Enter EOP 570 Sh 3, Emergency Depressurization.

QUESTION: 070 (1.00)

The following conditions exist:

- EOP 585 has been entered due to high temperature in the steam tunnel
- Fire alarms have been received in the reactor building
- Main Steam Isolation valves have closed
- CRD is maintaining RPV level
- Isolation Condenser is in service
- Drywell Venting is in progress

Which of the following systems should be isolated if they are discharging into the secondary containment?

a. Control Rod Drive

b. Drywell Vent

c. Isolation Condenser

d. Fire suppression

QUESTION: 071 (1.00)

Due to a loss of power, Inst. Air or N2 to Drywell valve, 1-AC-50 has failed shut. While investigating this failure an inadvertent main steam isolation has occurred due to operator error.

Which of the following methods is to be used for controlling pressure?

a. Bypass valves.

b. Manually operate SRV's.

c. Allow SRV's to automatically control pressure.

d. Use Steam Jet Air Ejectors.

QUESTION: 072 (1.00)

Due to a leaking relief valve EOP 580 "Primary Containment Control" was entered at 85 degrees F torus water temperature.

As torus water temperature increases, when is the reactor required to be scrammed?

a. 95 degrees F

b. 105 degrees F

c. 110 degrees F

d. 120 degrees F

QUESTION: 073 (1.00)

A failure to scram has occurred. EOP 575, "Level Power Control," is presently being implemented.

Select the correct actions concerning the Main Steam Line Isolation valves.

a. If the MSIV's are open then bypass all isolation signals.

b. If the MSIV's are open then bypass only low level isolation signals.

c. If the MSIV's are closed bypass all signals then reopen the MSIV's.

d. If the MSIV's are closed bypass all signals except high radiation and high flow then reopen the MSIV's.

QUESTION: 074 (1.00)

Select the statement identifying the reason reactor power decreases when reactor water level is lowered during a failure to scram (ATWS) event.

a. Lowering level below the moisture separator removes the flowpath thereby minimizing flow through the core.

b. Lowering level reduces the pressure in the core by reducing the head of water above the core.

c. Lowering level decreases the differential pressure between outside the shroud and inside the core.

d. Lowering level reduces power by increasing the subcooling of the water entering the core.

QUESTION: 075 (1.00)

Step PCP/DWT-12 of EOP 580 is to be performed. The following plant conditions exist.

- RPV water level = - 140 inches.
- RPV Pressure = 90 psig
- Torus Pressure = 45 psig
- Four SRVs are open
- Both LPCI pumps are injecting to maintain level
- No Core Spray Pumps are running

Which statement describes the use of LPCI pumps for drywell spray?

a. LPCI pumps cannot be used in this condition for drywell spray.

b. LPCI pumps can be used for drywell spray provided RPV level remains above 2/3 core height.

c. LPCI pumps can be used for drywell spray provided RPV level remains above 1/2 core height.

d. LPCI pumps can be used for drywell spray provided four SRV's are open and RPV pressure remains above 50 psig.

QUESTION: 076 (1.00)

A failure to scram has occurred. Boron injection was commenced.

Select the condition below that will allow termination of boron injection.

a. The Reactor Shutdown Criteria is satisfied.

b. All rods are full in.

c. ONP-502 Emergency Plant Shutdown is entered following injection of 30% of the SLC tank level.

d. The entry condition for EOP 575 "ATWS - RPV Control" has cleared.

QUESTION: 077 (1.00)

A leak occurred in the RWCU system in the Reactor Building. Reactor Building Ventilation isolated when RPV level decreased below +8 inches.

Which of the following conditions would PROHIBIT restarting Reactor Building Ventilation?

a. RPV level remaining below +8 inches.

b. Drywell pressure increasing to 2.2 psig due to isolation of drywell cooling.

c. Area radiation monitors in RWCU area alarming.

d. Reactor Building Ventilation radiation increasing to 20 mrem/hr.

QUESTION: 078 (1.00)

Switch boards 101A and 101B in the 125 VDC system have been deenergized due to electrical faults. Power has remained stable at 78%.

Select the required action.

a. Maintain plant stable and request additional assistance in shutting down.

b. Locally runback recirculation pumps as soon as possible then maintain the plant stable.

c. Scram the reactor immediately.

d. Immediately commence rod insertion to reduce power. Do not attempt to adjust recirculation pump speed.

QUESTION: 079 (1.00)

Recirculation pumps are operating on the Master controller with reactor power at 83%.

Identify the response of the Recirculation pumps to a complete loss of Vital AC.

a. Scoop tubes will lockup.

b. Pumps will runback to 32% due to loss of signal from feedwater flow.

c. Recirculation pumps will trip due to loss of level indication.

d. Recirculation pump speed will decrease to minimum due to loss of signal from the master controller.

QUESTION: 080 (1.00)

An irradiated fuel bundle is being removed from the core when it is dropped resulting in the bundle lying across the top of the core. Reactor building ventilation has NOT isolated.

Identify the area or areas required to be evacuated.

a. Refuel Floor only.

b. Refuel Floor and Drywell.

c. Drywell and Reactor Building.

d. Reactor Building including the refuel floor.

QUESTION: 081 (1.00)

A loss of all Reactor Building Closed Cooling Water (RBCCW) pumps has occurred.

Which of the following systems can be cross tied with RBCCW to supply cooling to plant components?

a. Turbine Building Closed Cooling Water

b. Turbine Building Secondary Closed Cooling Water

c. Service Water

d. Emergency Service Water

QUESTION: 082 (1.00)

Following a trip of one recirculation pump from 80% power, select the condition requiring a manual reactor scram.

a. The select rod insert fails to reduce power oscillations.

b. Several LPRM downscale alarms are coming in and clearing every 2 to 4 seconds.

c. APRM power oscillations are 7% peak-to-peak on 4 channels and 8% on 2 channels.

d. Multiple LPRM upscale and downscale alarms annunciate periodically.

QUESTION: 083 (1.00)

During performance of ONP 502, Emergency Plant Shutdown, the operator attempts to reset the scram. Rod Groups 1 and 4 reset as indicated by the lights on the 905 panel but rod groups 2 and 3 will not reset.

Select the effect of continued operation in this condition.

a. The scram discharge volume will not drain.

b. A flow path exists from the reactor to the reactor building via the scram discharge volume.

c. The differential pressure across closed scram isolation valves will be excessive.

d. Potential exists for damage of CRD mechanisms if another scram signal is received with only some scram isolation valves closed.

QUESTION: 084 (1.00)

A loss of all station A.C. Power has occurred.

Select the condition that indicates when a RBCCW pump is required to be started.

a. Immediately following the diesel tying to the bus.

b. When drywell coolers are required by EOPs for maintaining drywell temperature.

c. When a CRD pump is required for maintaining RPV level.

d. When load capacity becomes available on the diesel and/or gas turbine following starting of the TBSCCW pump.

QUESTION: 085 (1.00)

A loss of both CRD pumps has occurred.

When is a reactor scram required?

a. When it is determined that neither pump can be restarted.

b. When high temperature alarms are received on more than one control rod.

c. When accumulator alarms are received on more than one control rod.

d. Ten minutes after loss of both pumps.

QUESTION: 086 (1.00)

A break has occurred in the instrument air system as indicated by annunciators and starting of the "lag" instrument air compressor. Pressure has decreased to 50 psig.

Select the automatic response that should have occurred.

a. Repositioning of reactor building ventilation dampers.

b. Reactor scram.

c. Main Steam Isolation Valve closure.

d. IA-20 from the Station Air System opening.

Surveillance area

QUESTION: 087 (1.00)

Due to a fire, the control room was evacuated prior to closing the inboard MSIV's from CRP 903.

Select the method used to close the MSIV's.

a. Remove control power fuses for the solenoids for the MSIV's.

b. Initiate a Group I isolation by deenergizing the RPS momentarily.

c. Valve out air and bleed off air to the outboard MSIV's.

d. Place Appendix R isolation switches to "ISOLATE."

QUESTION: 088 (1.00)

Shutdown cooling is in operation with one shutdown cooling pump in operation. Both reactor recirculation pumps have just been secured.

Select the flowpath that will exist in this condition that causes reduced shutdown cooling flow through the core.

a. Backflow through the B recirculation loop.

b. Backflow through the A recirculation loop.

c. Backflow through A loop jet pumps.

d. Backflow through B loop jet pumps.

QUESTION: 089 (1.00)

Following a complete Loss of Shutdown Cooling, temperature readings indicate a 1 degree F increase in bulk water temperature every 10 minutes. Assume the reactor vessel head is ON, no other parameters change and current temperature is 164 degrees F.

How much TIME is allowed before primary containment integrity MUST be established?

a. 260 minutes

b. 360 minutes

c. 480 minutes

d. 560 minutes

QUESTION: 090 (1.00)

It is desired to decrease pressure in containment with a group II isolation present. Which of the following sets of valves can be used for venting with a group II isolation signal present?

a. AC-11, 18" Torus Vent Valve, and AC-8, Discharge to Reactor Building Ventilation.

b. AC-12, 2" Torus Vent Valve, and AC-8, Discharge to Reactor Building Ventilation.

c. AC-7, 18" Drywell Vent Valve, and AC-10, Discharge to SBGT System.

d. AC-9, 2" Drywell Vent Valve, and AC-10, Discharge to SBGT System.

QUESTION: 091 (1.00)

A Main Steam Isolation has occurred causing a pressure increase. Select the conditions that comply with the PREFERRED method of using SRVs for pressure control.

a. Allow the SRV's to cycle automatically.

b. Open SRV's "A" and "E" until pressure has decreased by 100 to 150 psi. After pressure increases open SRV's "E" and "D" to achieve a similar pressure reduction.

c. Open SRV "E" and allow pressure to decrease by 50 psig. Continue to use "E" as required to maintain pressure.

d. Open SRV "A" to reduce pressure by 100 to 150 psig. After pressure increases open SRV "E" to achieve a similar pressure decrease.

QUESTION: 092 (1.00)

A loss of normal power has occurred and the Gas Turbine Generator failed to start. The diesel generator has been operating at 2800 kW for 45 minutes when a "DIESEL GENERATOR COOLANT TEMPERATURE HIGH" annunciator is received.

What is the required operator action?

a. Continue operation at 2800 kW but monitor temperature to ensure that it does not exceed 205 degrees F.

b. Reduce load to 2665 KW and investigate the alarm.

c. Continuously reduce load until the alarm clears.

d. Reduce load as necessary to prevent coolant temperature from exceeding 205 degrees F.

QUESTION: 093 (1.00)

A scram has occurred at 58% power. As power decreases the turbine is required to be tripped at what power level?

a. 100 MWE

b. 50 MWE

c. 50 MWE reverse power

d. 100 MWE reverse power

QUESTION: 094 (1.00)

A complete loss of DC has occurred. Subsequently a scram occurred. Select the required immediate action.

a. Control RPV water level with 1-FW-5C (10% valve).

b. Close the MSIV's.

c. Trip both recirculation pumps.

d. Send an operator to trip the turbine locally.

QUESTION: 095 (1.00)

Reactor was at 82% power when a loss of Vital AC occurred. The reactor has scrammed. RPV level has been restored to +30 inches as indicated on the ATWS level indication.

Select the immediate action required when RPV level is verified.

a. Take manual control of the FRVs.

b. Transfer control of feedwater flow to the Startup valve.

c. Secure one feed pump.

d. Secure all feed pumps.

QUESTION: 096 (1.00)

A loss of feedwater heating has occurred.

Which of the following conditions requires that recirculation flow be runback to minimum?

a. APRM rod block alarm occurs.

b. The instability region is entered.

c. Feedwater temperature decreases by 50 degrees F.

d. LPRM high alarms occur.

QUESTION: 097 (1.00)

Due to a fire in the cable vault the control room has been abandoned.

Which operator is responsible for controlling RPV level using CRD?

a. Reactor Building Rounds PEO

b. Supervising Control Operator

c. CRP 905 Control Operator

d. BOP Control Operator

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

ANSWER: 001 (1.00)

b.

REFERENCE:

OP276/2276/3276 page 24 LO 0672200 ONP 504 page 1

[3.3/4.3]
294001A111 ..(KA's)

ANSWER: 002 1.50 ~~(2.00)~~ TJW 1/24/94

a. 1

b. 1

c. 4

~~d. 5~~ deleted TJW 1/24/94

REFERENCE:

SHP 4906 pages 2&3 LO 12

[3.3/3.8]

294001K103 ..(KA's)

ANSWER: 003 (1.00)

a.

REFERENCE:

New 10CFR20 Training Lesson Plan page 6. LO 6.

[3.3/3.8]

294001K103 ..(KA's)

ANSWER: 004 (1.00)

c.

REFERENCE:

CHECK AT FACILITY FOR REFERENCE

[3.3/3.8]

294001K103 ..(KA's)

ANSWER: 005 (1.00)

a.

REFERENCE:

1-OPS-6.09 Rev. 0, "Control Room Habitability", 1.1.
No Facility Objective.

KA [3.5/3.8]

294001K116 ..(KA's)

ANSWER: 006 (1.00)

b.

REFERENCE:

ACP -QA-2.06A page 3

[3.9/4.5]

294001K102 ..(KA's)

ANSWER: 007 (1.00)

d.

REFERENCE:

C-OP-200 page 19 No Facility Objective

[3.1/3.2]

294001A104 ..(KA's)

ANSWER: 008 (1.00)

c.

REFERENCE:

OP 341 page 73 No Facility Objective

[3.3/3.6]

294001K107 ..(KA's)

ANSWER: 009 (1.00)

d.

REFERENCE:

OP 276/2276/3276 page 12 LO 0512200

[4.2/4.2]

294001A102 ..(KA's)

ANSWER: 010 (1.00)

c.

REFERENCE:

OP 276/2276/3276 page 19 LO 0592200

[3.3/3.6]

294001K107 ..(KA's)

ANSWER: 011 (1.00)

d.

REFERENCE:

ACP 6.01 Rev. 23, Section 6.1.1.4, page 5.
MP1 Objective # 35811101C600 0522200

KA [2.7/3.7]

294001A103 ..(KA's)

ANSWER: 012 (1.00)

c.

REFERENCE:

ACP 6.01 Rev. 23, Section 4.4, page 3.
MP1 Objective # 35811101C600 0502200

KA [2.9/4.7]

294001A116 ..(KA's)

ANSWER: 013 (1.00)

b.

REFERENCE:

Control Rod Drive Text ID 302 page 5 section 3.2.1.1. LO 11.b

[3.0/3.0]

201001K102 ..(KA's)

ANSWER: 014 (1.00)

b.

REFERENCE:

Reactor Manual Control System Text ID 32A page 12 section 5.4.3 LO 5.e

[3.5/3.5]

201002K406 ..(KA's)

ANSWER: 015 (1.00)

c.

REFERENCE:

Reactor Manual Control System Text ID 32A page 16 section 5.7 LO 15.c

[2.8/2.8]

201002A304 ..(KA's)

ANSWER: 016 (1.00)

c.

REFERENCE:

FWCI Text 334 page 7. LO 6.c

[3.7/3.7]

206000K105 ..(KA's)

ANSWER: 017 (1.00)

b.

REFERENCE:

Technical Specification Table 3.1.1 Note ** page 3/4 1-5 APRM Text page 8 LO 6.b [CHECK AT FACILITY FOR T. S. INTERPRETATION OF NOTE **]

[3.4/4.1]

215005G011 ..(KA's)

ANSWER: 018 (1.00)

ø. d

REFERENCE:

Technical Specification 3.7.A.6.c Containment Text LO 22.g

[3.3/4.1]

223001G005 ..(KA's)

ANSWER: 019 (1.00)

d.

REFERENCE:

Technical Specification 3.3.B RWM Text LO 9.a

[3.2/4.0]

201006G005 ..(KA's)

ANSWER: 020 (1.00)

d.

REFERENCE:

CRD Text page 31 LO 8.d

[3.6/3.7]

201003K404 ..(KA's)

ANSWER: 021 (1.00)

b.

REFERENCE:

Recirc Text page 66 LO 35

[2.8/2.8]

202001A410 ..(KA's)

ANSWER: 022 (1.00)

c.

REFERENCE:

OP 301 page 5 step 4.10 LO 29

[3.5/3.7]

202001G010 ..(KA's)

ANSWER: 023 (1.00)

b.

REFERENCE:

RWCU Text page 54 LO 4.1

[2.9/2.9]

204000A107 ..(KA's)

ANSWER: 024 (2.00)

a. 2

b. 5

c. 4

d. 6

REFERENCE:

Reactor Manual Control Text page 7 section 5.0 LO 2.h

[3.4/3.3]

214000A301 ..(KA's)

ANSWER: 025 (1.00)

d.

REFERENCE:

Rod Block Monitor Lesson Plan Section 5.7.6 LO 5.j

[2.9/2.9]

215002A402 ..(KA's)

ANSWER: 026 (1.00)

d.

REFERENCE:

Control Rod Drive Text ID 302 page 6 section 3.2.1.2. LO 10.a

[2.9/3.1]

201001K201 ..(KA's)

ANSWER: 027 (1.00)

b.

REFERENCE:

Control Rod Drive Text ID 302 page 16 section 4.1.6 LO 3.k

[3.5/3.4]

201001A105 ..(KA's)

ANSWER: 028 (1.00)

a.

REFERENCE:

Reactor Manual Control System Text ID 32A page 12 section 5.4.3 LO 14.b

[3.5/3.5]

201002A402 ..(KA's)

ANSWER: 029 (1.00)

a.

REFERENCE:

Reactor Recirculation Speed Control System Text 1301B Page 8 Section 5.4.1. LO 4.b

[3.0/3.0]

202002K402 ..(KA's)

ANSWER: 030 (1.00)

c.

REFERENCE:

Reactor Recirculation Flow Control System Text 31B page 17 LO 10.a & e

[3.1/3.2]

202002K109 ..(KA's)

ANSWER: 031 (1.00)

c.

REFERENCE:

LPCI Text 335 page 18 LO 18

[3.6/3.7]

203000K601 ..(KA's)

ANSWER: 032 (1.00)

c.

REFERENCE:

LPCI Text page 18 LO 22

[3.9/4.1]

203000K410 ..(KA's)

ANSWER: 033 (1.00)

b.

REFERENCE:

LPCI Text 335 page 22 LO 24

[4.2/4.6]

203000A307 ..(KA's)

ANSWER: 034 (1.00)

b.

REFERENCE:

FWCI Text 334 page 11 LO 8.a

[3.5/3.5]

206000A202 ..(KA's)

ANSWER: 035 (1.00)

a.

REFERENCE:

OP 334 page 9 step 5.4.1 LO 11

[3.1/3.1]

206000K503 ..(KA's)

ANSWER: 036 (1.00)

d.

REFERENCE:

Isolation Condenser Text 307 LO 13 e.& f.

[3.5/3.7]

207000K602 ..(KA's)

ANSWER: 037 (1.00)

c.

REFERENCE:

OP-307 page 11 step 5.3 LO 7.d

[3.8/3.8]

207000A208 ..(KA's)

ANSWER: 038 (2.00)

a. 2

b. 6

c. 5

d. 2

REFERENCE:

Isolation Condenser Text 307 pages 3&31 LO 8.a-d.

[3.6/3.8]

207000K201 ..(KA's)

ANSWER: 039 (1.00)

b.

REFERENCE:

Core Spray Text 336 pages 4, 11 and figure 7 LO 7.b., f., g.

[3.7/3.7]

209001A104 ..(KA's)

ANSWER: 040 (1.00)

d.

REFERENCE:

Core Spray Text page 11 LO 10.e OP 336

[3.6/3.6]

209001A301 ..(KA's)

ANSWER: 041 (1.00)

c.

REFERENCE:

Technical Specifications 4.4.A.1 and 3.4.C EOP 575 sheet 1 step FSQ-25
LO 6.c

[3.1/4.2]

211000G006 ..(KA's)

ANSWER: 042 (1.00)

c.

REFERENCE:

IRM Text page 14 LO 4.c

[3.7/3.7]

215003K401 ..(KA's)

ANSWER: 043 (1.00)

b.

REFERENCE:

Technical Specification 2.1.2.A.1.b LO.7.b

[3.6/3.6]

215005K505 ..(KA's)

ANSWER: 044 (1.00)

a.

REFERENCE:

OP314 page 46 Main Turbine Text page 27 No Facility Training Material on MPR/EPR or Learning Objectives Provided

[4.1/4.2]

241000A203 ..(KA's)

ANSWER: 045 (1.00)

a.

REFERENCE:

Diesel Generator Text page 37 No facility learning objectives supplied for DG system

[3.4/3.5]

264000G007 ..(KA's)

ANSWER: 046 (1.00)

a.

REFERENCE:

RWM Text pages 12 and 13 LO 6.c [CHECK AT FACILITY CONCERNING CORRECTION OF ERRORS ON PAGE 13]

[3.5/3.5]

201006K512 ..(KA's)

ANSWER: 047 (1.00)

d.

REFERENCE:

RWCU Text page 55 LO 6.g

[3.4/3.4]

204000A213 ..(KA's)

ANSWER: 048 (1.00)

a.

REFERENCE:

Technical Specification 4.3.B.1.b CRD Text LO 20.c

[3.5/3.8]

214000A203 ..(KA's)

ANSWER: 049 (1.00)

b.

REFERENCE:

Rod Block Monitor Lesson Plan Section 5.8.2 LO 7.b

[3.1/3.3]

215002A203 ..(KA's)

ANSWER: 050 (1.00)

b.

REFERENCE:

HVAC lesson text pages 54, 55 and 56 No Facility Objective

[2.9/3.0]

290003K107 ..(KA's)

ANSWER: 051 (1.00)

c.

REFERENCE:

CRAB 904A-1 Window 3-1 LO 6.e

[3.6/3.7]

272000A302 ..(KA's)

ANSWER: 052 (1.00)

a.

REFERENCE:

SP 635.1 pages 3&4 No Facility Objective

[3.4/3.6]

268000A401 ..(KA's)

ANSWER: 053 (1.00)

b.

REFERENCE:

SBGT Text page 30 No facility Learning Objective

[3.0/2.9]

261000A303 ..(KA's)

ANSWER: 054 (1.00)

b.

REFERENCE:

Reactor Recirculation Operating Procedure OP 301 page 18 section 5.10
LO 20

[3.9/3.8]

202001A412 ..(KA's)

ANSWER: 055 (1.00)

d.

REFERENCE:

EOP 590.17 page 3 HVAC Text page 37. No Facility Objective [CHECK AT FACILITY FOR DISCREPANCY IN HVAC TEXT]

[3.8/3.8]

290001G007 ..(KA's)

ANSWER: 056 (1.00)

a.

REFERENCE:

Technical Specification Table 3.1.1, 3.2.1 and 3.6.K.1 LO 7.g

[3.9/4.2]

239001A205 ..(KA's)

ANSWER: 057 (1.00)

c.

REFERENCE:

OP 332A PAGE 6
Fire Protection System Text, page 49

[3.1/3.2]

286000A302 ..(KA's)

ANSWER: 058 (1.00)

d.

REFERENCE:

TIP Text page 16 LO 4

[3.3/3.4]

215001K105 ..(KA's)

ANSWER: 059 (1.00)

~~a.~~ TCW 1/24/94 a.

REFERENCE:

Fuel Handling Text pages 34 & 35 LO 24

[3.3/4.1]

234000K402 ..(KA's)

ANSWER: 060 (1.00)

b.

REFERENCE:

Standby Liquid control Text Figure 11 LO 3.d

[3.4/3.5]

290002K112 ..(KA's)

ANSWER: 061 (1.00)

c.

REFERENCE:

Facility Exam Bank No. RO001589

[3.8/4.2]
262001K302 ..(KA's)

ANSWER: 062 (1.00)

d.

REFERENCE:

EOP 570, "RPV Control" Step RC-5.
EOP 580, "Primary Containment Control" Step PC-1.

[3.8/4.3]

295028G012 ..(KA's)

ANSWER: 063 (1.00)

c.

REFERENCE:

EOP 560.7, EOP 580 Technical Basis, page 26.

[3.1/3.3]

295029K205 ..(KA's)

ANSWER: 064 (1.00)

a. or C.

REFERENCE:

EOP 560.5, EOP 570 Technical Basis, page 18 and 19.

[4.6/4.6]

295031A201 ..(KA's)

ANSWER: 065 (1.00)

a.

REFERENCE:

EOP 560.7, EOP 580 Technical Basis, page 14.

[3.5/3.8]

295030K207 ..(KA's)

ANSWER: 066 (1.00)

c.

REFERENCE:

EOP 560.6, Technical Basis for EOP 575, page 36.

[4.0/4.2]

295037K106 ..(KA's)

ANSWER: 067 (1.00)

d.

REFERENCE:

EOP 560.7, EOP 580 Technical Basis, page 28.
EOP 590.8, Primary Containment Spray, page 3.

[3.9/3.9]

295024A117 ..(KA's)

ANSWER: 068 (1.00)

c. or a

REFERENCE:

EOP 580, Primary Containment Control, Step TWT-6.
EOP 575, Failure to Scram, Step FSP-1.

[3.8/4.5]

295026G012 ..(KA's)

ANSWER: 069 (1.00)

a.

REFERENCE:

EOP 585, Secondary Containment Control and Radioactive Release Control, Step SC-15.

[3.8/4.4]

295033G012 ..(KA's)

ANSWER: 070 (1.00)

c.

REFERENCE:

EOP 585, Secondary Containment and Radioactive Release Control, Step SC-13.

[3.7/3.9]

295032A105 ..(KA's)

ANSWER: 071 (1.00)

c.

REFERENCE:

EOP 570 Sh.1, RPV Control, Step RP-2.
Drywell Compressor System, figure 1 and page 1.

[4.4/4.4]

295025A103 ..(KA's)

ANSWER: 072 (1.00)

c.

REFERENCE:

EOP 580, Primary Containment Control, step TWT-3.

[3.4/3.8]
295026G007 ..(KA's)

ANSWER: 073 (1.00)

b.

REFERENCE:

EOP 575 Sheet 2, "Level Power Control", Step LP-2 and LP-3.

[3.8/4.1]

295037K306 ..(KA's)

ANSWER: 074 (1.00)

c.

REFERENCE:

EOP 560.6, "EOP 575 Technical Basis", page 64.

[4.1/4.5]

295037K303 ..(KA's)

ANSWER: 075 (1.00)

a.

REFERENCE:

EOP 580 "Primary Containment Control" Step PCP/DWT-12
EOP 570 Sheet 4, RPV Flooding, Table 1.

[4.6/4.8]

295031A204 ..(KA's)

ANSWER: 076 (1.00)

b.

REFERENCE:

EOP 575 Sheet 1 "Failure to Scram" Step FS-1.

[3.9/4.6]

295037G012 ..(KA's)

ANSWER: 077 (1.00)

d.

REFERENCE:

EOP 585 Secondary Containment And Radioactive Release Control Step SC-1

[4.0/3.9]

295034A103 ..(KA's)

ANSWER: 078 (1.00)

a.

REFERENCE:

ONP-506, Total Loss of Station 125 VDC Power, section 6.1.

[3.1/3.8]

295004G001 ..(KA's)

ANSWER: 079 (1.00)

a.

REFERENCE:

ONP-506B, Loss of Vital AC, section 6.C.

[3.4/3.5]

295003K204 ..(KA's)

ANSWER: 080 (1.00)

b.

REFERENCE:

ONP 519, Dropped Fuel Bundle, section 1.0.

[3.3/4.2]

295023G001 ..(KA's)

ANSWER: 081 (1.00)

a.

REFERENCE:

RBCCW Text, page 14.
Learning Objective 9.j

[3.3/3.4]

295018A101 ..(KA's)

ANSWER: 082 (1.00)

d.

REFERENCE:

ONP 526, Uncontrolled Power Oscillations, step 1.2.

[3.8/3.7]

295001G010 ..(KA's)

ANSWER: 083 (1.00)

b.

REFERENCE:

ONP 502, Emergency Plant Shutdown, page 2.

[3.5/3.6]

295006A106 ..(KA's)

ANSWER: 084 (1.00)

d.

REFERENCE:

ONP 503B, Loss of all Station A.C. Power (LNP), page 1.

[4.4/4.4]

295003A103 ..(KA's)

ANSWER: 085 (1.00)

a.

REFERENCE:

OP 302, Control Rod Drive System, page 42.

[3.1/3.2]

295022A101 ..(KA's)

ANSWER: 086 (1.00)

b.

REFERENCE:

ONP 512, Rapid and Total Loss of Instrument Air, page 1.

[3.8/3.9]

295019K201 ..(KA's)

ANSWER: 087 (1.00)

d.

REFERENCE:

ONP 525A, Degraded fire in the Control Room or Cable Vault, page 1.

[4.2/4.3]

295016A107 ..(KA's)

ANSWER: 088 (1.00)

c.

REFERENCE:

Shutdown Cooling Student Text, page 11. Obj. 9.

[3.1/3.2]

295021K207 ..(KA's)

ANSWER: 089 (1.00)

c.

REFERENCE:

Technical Specifications 3.7.A.3
ONP 531, Loss of Decay Heat Removal,

[3.5/3.6]

295021A201 ..(KA's)

ANSWER: 090 (1.00)

d.

REFERENCE:

Primary Containment Text, page 47.

[3.4/3.5]

295010A101 ..(KA's)

ANSWER: 091 (1.00)

d.

REFERENCE:

OP 337, Autopressure Relief System, section 4.2.

[3.9/4.1]

295007A104 ..(KA's)

ANSWER: 092 (1.00)

b.

REFERENCE:

OP 338, Diesel Generator, page 4.

[3.7/3.6]

295003G005 ..(KA's)

ANSWER: 093 (1.00)

b.

REFERENCE:

ONP 502, Emergency Plant Shutdown, step 1.3.

[3.8/3.6]

295005G010 ..(KA's)

ANSWER: 094 (1.00)

a.

REFERENCE:

ONP 506, Total Loss of Station 125 VDC Power, page 1.

[3.2/3.4]

295004G010 ..(KA's)

ANSWER: 095 (1.00)

d.

REFERENCE:

ONP-506B, Loss of Vital AC, section 1.3.1.

[3.9/4.1]

295003G010 ..(KA's)

ANSWER: 096 (1.00)

c.

REFERENCE:

ONP 523, Loss of Feedwater Heating, page 1.

[3.8/3.6]

295014G010 ..(KA's)

ANSWER: 097 (1.00)

d.

REFERENCE:

ONP 525A, Degraded Fire in Control Room or Cable Vault, page 3.

[4.0/4.0]

295016A109 ..(KA's)

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

ANSWER KEY

MULTIPLE CHOICE

001 b

002 MATCHING

a 1

b 1

c 4

~~d 5~~ deleted TEW 1/24/94

MULTIPLE CHOICE

003 a

004 c

005 a

006 b

007 d

008 c

009 d

010 c

011 d

012 c

013 b

014 b

015 c

016 c

017 b

018 ~~c~~ d TEW 1/24/94

019 d

020 d

021 b

022 c

023 b

024 MATCHING

a 2

b 5

c 4

d 6

MULTIPLE CHOICE

025 d

026 d

027 b

028 a

029 a

030 c

031 c

032 c

033 b

034 b

035 a

ANSWER KEY

036 d

MULTIPLE CHOICE

037 c

038 MATCHING

a 2

b 6

c 5

d 2

MULTIPLE CHOICE

039 b

040 d

041 c

042 c

043 b

044 a

045 a

046 a

047 d

048 a

049 b

050 b

051 c

052 a

053 b

054 b

055 d

056 a

057 c

058 d

059 ~~d~~ a 1/24/94

060 b

061 c

062 d

063 c

064 a or c

065 a

066 c

067 d

068 c or a

069 a

070 c

071 c

072 c

073 b

074 c

075 a

ANSWER KEY

076 b

MULTIPLE CHOICE

077 d

078 a

079 a

080 b

081 a

082 d

083 b or a

084 d

085 a

086 b

087 d

088 c

089 c

090 d

091 d

092 b

093 b

094 a

095 d

096 c

097 d

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

Attachment 2

U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
SENIOR OPERATOR LICENSE
REGION 1

MASTER COPY

CANDIDATE'S NAME:

FACILITY: Millstone 1

REACTOR TYPE: BWR-GE3

DATE ADMINISTERED: 94/01/13

INSTRUCTIONS TO CANDIDATE:

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires a final grade of at least 80%. Examination papers will be picked up four (4) hours after the examination starts.

TEST VALUE	CANDIDATE'S SCORE	%	
98.50 ~~100.00~~		___ %	TOTALS
	FINAL GRADE		

780 1/24/94

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

Candidate's Signature

MASTER COPY

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

MULTIPLE CHOICE

001 a b c d ____

002 MATCHING

a ____

b ____

c ____

~~d ____~~ deleted TAW 1/24/94

MULTIPLE CHOICE

003 a b c d ____

004 a b c d ____

005 a b c d ____

006 a b c d ____

007 a b c d ____

008 a b c d ____

009 a b c d ____

010 a b c d ____

011 a b c d ____

012 a b c d ____

013 a b c d ____

014 a b c d ____

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016 a b c d ____

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021 a b c d ____

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036 a b c d ____

037 a b c d ____

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

038 MATCHING

a ____

b ____

c ____

d ____

MULTIPLE CHOICE

039 a b c d ____

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042 a b c d ____

043 a b c d ____

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074 a b c d ____

075 a b c d ____

076 a b c d ____

077 a b c d ____

078 a b c d ____

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

079 a b c d ____

MULTIPLE CHOICE

080 a b c d ____

081 a b c d ____

082 a b c d ____

083 a b c d ____

084 a b c d ____

085 a b c d ____

086 a b c d ____

087 a b c d ____

088 a b c d ____

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093 a b c d ____

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095 a b c d ____

096 a b c d ____

097 a b c d ____

098 a b c d ____

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

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination, the following rules apply:

1. Cheating on the examination will result in a denial of your application and could result in more severe penalties.

2. After you complete the examination, sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination.

3. To pass the examination, you must achieve a grade of 80 percent or greater.

4. The point value for each question is indicated in parentheses after the question number.

5. There is a time limit of 4 hours for completing the examination.

6. Use only black ink or dark pencil to ensure legible copies.

7. Print your name in the blank provided on the examination cover sheet and the answer sheet.

8. Mark your answers on the answer sheet provided and do not leave any question blank.

9. If the intent of a question is unclear, ask questions of the examiner only.

10. Restroom trips are permitted, but only one applicant at a time will be allowed to leave. Avoid all contact with anyone outside the examination room to eliminate even the appearance or possibility of cheating.

11. When you complete the examination, assemble a package including the examination questions, examination aids, and answer sheets and give it to the examiner or proctor. Remember to sign the statement on the examination cover sheet.

12. After you have turned in your examination, leave the examination area as defined by the examiner.

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QUESTION: 001 (1.00)

Manipulation of controls which directly affect reactivity or power level of the reactor may be performed by:

a. a Control Operator in the license reactivation process without further supervision.

b. a Control Operator in license training under the direction of an actively licensed Senior Control Operator.

c. a Plant Equipment Operator utilizing local manual control of a recirculation MG set, in direct communication with the Reactor Operator at the controls.

d. an unlicensed on-shift Shift Technical Advisor, at a remote shutdown equipment location, under the direction of an actively licensed Control Operator.

QUESTION: 002 1.50 ~~(2.00)~~ TDW 1/24/94

Match the Millstone posting of radiological controlled areas in Column A with the minimum radiation dose rate that would require the posting, listed in Column B.

The items from column B may be used once, more than once, or not at all and only a single answer may occupy one answer space.

COLUMN A AREAS	COLUMN B RADIATION DOSE RATE
a. Radiation Area	1. ≥ 0.5 mrem/hr
b. Neutron Radiation Area	2. ≥ 1.0 mrem/hr
c. High Radiation Area	3. ≥ 2.5 mrem/hr
~~d. Extremely High Radiation Area~~ deleted TDW 1/24/94	4. ≥ 100 mrem/hr
	5. > 500 mrem/hr
	6. > 1000 mrem/hr

QUESTION: 003 (1.00)

A Northeast Utilities (NU) employee who is 24 years old today, can have a maximum lifetime total effective dose equivalent (TEDE) in accordance with NU TEDE administrative goals of:

a. 24 rem.

b. 30 rem.

c. 48 rem.

d. 72 rem.

QUESTION: 004 (1.00)

Given the following conditions:

-- An operator is performing a whole body frisk using a portable frisker, RM-14/HP 210.
-- Background radiation count rate in the contaminated area is at the maximum allowed for using the frisker.

What is the count rate (background + actual) at which the operator is considered to be contaminated?

a. 100 counts per minute

b. 300 counts per minute

c. 400 counts per minute

d. 500 counts per minute

QUESTION: 005 (1.00)

Which one of the following conditions require the Control Room Operators to immediately don the Scott Air Pack equipment?

a. Group 1 isolation on 120% steam line flow.

b. Group 1 isolation on main steam line low pressure.

c. Receipt of annunciator, "CONTROL ROOM AIR INTAKE HIGH RADIATION/DOWNSCALE."

d. Fire in the Main Control Board.

QUESTION: 006 (1.00)

The "A" CRD pump is properly red tagged for electrical trouble shooting. Electrical Maintenance has determined that the panel control switch requires replacement.

The red tag on the control switch:

a. is cleared before removal of the switch from the panel.

b. is removed from the sw .ch and attached near the panel hole.

c. remains with the old switch until transferred to the new switch by the electricians.

d. is removed under a "temporary lift" until the new switch is installed.

QUESTION: 007 (1.00)

Plant personnel have been dispatched to search Unit 1 in response to a security event.

The method of communicating to the Shift Supervisor is:

a. Plant Page

b. Security Radio

c. Operations Radio

d. Plant Telephone

QUESTION: 008 (1.00)

Given the following list of safety requirements and protective equipment.

1. Stand clear of the cubicle
2. Stand to the side of the cubicle and use the left hand for the actual racking.
3. Hard hat with face shield
4. Flash Jacket with hood pulled over hard hat
5. Linesman gloves
6. Rubber mat to stand on

Select the minimum that are required when racking in a 4160V "ITE Magnablast" electrical circuit breaker, if the remote racking operator is NOT available.

a. 2 and 5

b. 1, 3 and 5

c. 2, 3, 4 and 5

d. 1, 3, 4 and 6

QUESTION: 009 (1.00)

The Double Line Trip Detector, Breaker Failure, Unit Rejection Scheme (DBURS) has failed and is out of service.

Which one of the following is responsible for station coordination of power maneuvers with this condition?

a. CONVEX

b. Unit 1 Shift Supervisor

c. Unit 2 Shift Supervisor

d. Unit 3 Shift Supervisor

QUESTION: 010 (1.00)

A Plant Equipment Operator directed to perform a tagout on a 4160V ITE "Magnablast" electrical circuit breaker, removes the control power fuses for the wrong breaker.

Select the correct statement concerning operation of that breaker if it was closed at the time the DC control power fuses were removed.

a. It will trip open as soon as the control power fuses were removed, any further breaker operations must be local.

b. It will not operate remotely, it can be tripped locally once with no further operation possible.

c. It will not operate remotely, it can be locally tripped open, then closed and tripped open again.

d. It can only be tripped when the racking mechanism is engaged.

QUESTION: 011 (1.00)

How are electrical fuses that are removed for a station tagout controlled?

a. The fuses are tagged and delivered to the clearance requester.

b. The fuses are bagged, tagged and placed in a drawer of the SCO's desk.

c. The fuses are bagged, tagged and placed in the panel near the fuse holder.

d. The fuses are tagged and stored in a cabinet in the Shift Supervisor's office.

QUESTION: 012 (1.00)

Who by title has the responsibility for maintaining control of the TIP Room (H4) key?

a. On-duty Health Physics Supervisor

b. On-duty Security Shift Supervisor

c. On-shift Health Physics Technician

d. On-shift Shift Supervisor

QUESTION: 013 (1.00)

A recently licensed Senior Reactor Operator (SRO) has performed the functions of an SRO during one 8 hour shift since January 1, 1994.

Which one of the following will maintain his/her SRO license in an "active status" in accordance with 10 CFR 55.53?

a. Two 8 hour shifts performing SRO functions during February and four 8 hour shifts performing SRO functions during March.

b. Five 8 hour shifts performing SRO functions during March and three 12 hour shifts performing SRO functions during April.

c. Three 8 hour shifts performing SRO functions during January and six 12 hour shifts performing Reactor Operator functions during March.

d. Seven 8 hour shifts performing Reactor Operator functions during January and four 8 hour shifts performing SRO functions during February.

QUESTION: 014 (1.00)

Which of the following duties can the Director of Station Emergency Operations (DSEO) delegate?

a. KI use

b. Nuclear Incident Report Form Approval

c. Approval of contaminated personnel leaving the site

d. Performance of Plant Actions outside Technical Specifications

QUESTION: 015 (1.00)

A "Lift Truck" being used at the screenhouse to load a service water pump motor onto a truck, ruptures a hydraulic hose. The resultant oil spill discharges approximately eight 8 gallons of oil on the ground and less than one quart into Niantic Bay.

This oil spill:

a. is not classified as a reportable event to either state or federal authorities.

b. is classified as a State Posture Code ECHO and must be reported to the NRC within one hour, identified as a four hour report.

c. is classified as a State Posture Code ECHO and must be reported to the NRC within four hours.

d. is classified as a State Posture Code DELTA-ONE and must be reported to the NRC within one hour.

QUESTION: 016 (1.00)

A plant startup is in progress. Five minutes after placing the Reactor Mode Switch in "RUN" Chemistry reports that reactor coolant chloride concentration has increased from 0.08 ppm to 0.14 ppm in the last 4 hours.

Based on this condition the Shift Supervisor:

a. must hold the startup until reactor water cleanup can reduce chloride to less than 0.1 ppm.

b. must commence an orderly reactor shutdown and place the reactor in cold shutdown within 24 hours.

c. may continue the startup because the chloride concentration is still within specifications.

d. may continue the startup if dissolved oxygen content of the reactor coolant system is less than 0.3 ppm.

QUESTION: 017 (1.00)

During operation at 100% power the reactor building operator on rounds reports the heat tracing on the suction side of the standby liquid control (SLC) pumps is damaged and inoperable.

The following conditions exist:

-- liquid poison tank concentration 13%
-- liquid poison tank volume 1850 gallons
-- reactor building ambient temperature at SLC pumps 80 degrees F

Select the action required based on these conditions:

a. Commence an orderly reactor shutdown because the SCL system does not meet the definition of operable.

b. Commence an orderly reactor shutdown because the SCL solution temperature does not meet the requirements of Technical Specifications.

c. Operation may be continued for up to 7 days with this condition existing.

d. Operation may be continued indefinitely at this boron concentration as long as reactor building ambient temperature remains above 70 degrees F.

QUESTION: 018 (1.00)

With the plant operating at 100% power APRM channel 3 has the following LPRMS bypassed:

20-37A 44-13B 04-37C
36-21A

With these conditions LPRM 04-21A fails downscale.

What action is required based on this condition?

a. No action is required, continue operation with no restrictions.

b. Bypass LPRM 04-21A only.

c. Bypass LPRM 04-21A and APRM channel 3.

d. Bypass LPRM 04-21A, APRM channel 3 and insert a trip in RPS channel A.

QUESTION: 019 (1.00)

During operation at 100% power an engineering review reveals that the previous turbine bypass valves functional test was not satisfactory.

Select the required action.

a. Reduce power so that turbine first stage pressure is less than the value corresponding to 50% of rated reactor thermal power.

b. Reduce power to within the capacity of 5 bypass valves.

c. Commence an orderly shutdown and have the turbine off the line within 24 hours.

d. Take the reactor out of the "RUN" mode within 12 hours.

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QUESTION: 020 (1.00)

The reactor is operating at 100% power with the following conditions:

-- control rod 18-23 is electrically disarmed at position 24
-- control rod 14-23 is electrically disarmed at position 00
-- hydraulic accumulator for control rod 22-27 indicates HI-LEVEL
-- control rod 22-27 is at position 36

The PEO reports that when attempting to drain hydraulic accumulator 22-27 a steady stream of water under pressure is coming from the charging block drain connection and does not appear to be slowing.

Select the action allowed based on these conditions.

a. Insert control rod 22-27 to position 00, valve out hydraulically and continue operation with no restrictions.

b. Leave control rod 22-27 at its present position, electrically disarm and continue operation with no restrictions.

c. Leave control rod 22-27 at its present position, valve out hydraulically and commence an orderly shutdown to place the reactor in cold shutdown within 24 hours.

d. Insert control rod 22-27 to position 00, and continue operation with no restrictions.

QUESTION: 021 (1.00)

The reactor is operating at 25% power during startup after a refueling outage. An engineering review indicates that shutdown cooling inside containment isolation supply valve, 1-SD-1 had a stroke time of 58 seconds during its most recent surveillance test.

Select the action required.

a. No action is required as long as the valve remains closed.

b. Verify and record daily that shutdown cooling pump suction valves 1-SD-2A and 2B are closed.

c. Open the power supply breaker to the valve motor and continue operation with no restrictions.

d. Initiate an orderly shutdown and have the reactor in cold shutdown within 24 hours.

QUESTION: 022 (1.00)

Chemistry has reported that a tube leak in the main condenser has increased condensate demineralizer influent conductivity to 0.31 umho/cm and estimates that the unused ion exchange capacity of the demineralizer resin will reach a minimum value of 5 pounds as chloride ions in 7 days.

What action is required based on these conditions?

a. Commence an orderly shutdown and place the reactor in cold shutdown within 24 hours.

b. Enter a seven day LCO to place the reactor in cold shutdown.

c. Calculate and log the remaining ion exchange capacity of the resin daily and replace before it reaches a minimum value of 5 pounds as chloride ions

d. Ultrasonically clean each inservice condensate demineralizer, calculate and log the remaining ion exchange capacity of the resin after each cleaning.

QUESTION: 023 (1.00)

During operation at 100% power, main steam line differential pressure indicating switch, dPIS 261-2B (channel B) fails downscale.

Select the required action based on this condition.

a. No action is required if all of the remaining dPISs on that steam line are operable.

b. Reduce power to 75% and isolate the affected steam line.

c. Initiate an orderly load reduction and have the reactor in "HOT STANDBY" within eight hours.

d. Manually trip containment isolation logic channel "B."

QUESTION: 024 (1.00)

Following a Loss of Coolant Accident, Feed Water Coolant Injection (FWCI) initiated and the feed regulating valves (FRVs) shifted to the "FLOW CONTROL" mode of operation.

Select the condition which will shift FWCI out of "FLOW CONTROL."

a. Reactor pressure decreases to less than 300 psig.

b. Feed flow decreases to less than 115%.

c. Vessel level increases to +48 inches indicated on the narrow range Yarway.

d. Vessel level increases to +37 inches indicated on the narrow range GEMACs.

QUESTION: 025 (1.00)

Which one of the following plant areas is protected by a Carbon Dioxide Fire Protection system?

a. Fire Pump House

b. Gas Turbine Enclosure

c. Diesel Generator Room

d. Cable Vault

QUESTION: 026 (1.00)

Select the normal and alternate power supplies to the Control Rod Drive (CRD) pump motors.

a. "A" Normal Bus 14A Alternate Bus 14E
"B" Normal Bus 14C

b. "A" Normal Bus 14E Alternate Bus 14F
"B" Normal Bus 14H

c. "A" Normal Bus 14A Alternate Bus 14B
"B" Normal Bus 14C

d. "A" Normal Bus 14E Alternate Bus 14H
"B" Normal Bus 14F

QUESTION: 027 (1.00)

Which one of the following describes the response of the scram discharge volume pilot air valves and vent and drain valves following a trip of RPS channel "B."

a. No pilot air valves reposition. Both sets of vent and drain valves remain open.

b. One pilot air valve repositions. Both sets of vent and drain valves remain open.

c. One pilot air valve repositions. One set of vent and drain valves close.

d. Two pilot air valves reposition. One set of vent and drain valves close.

QUESTION: 028 (1.00)

Which one of the following "ROD OUT NOTCH OVERRIDE SWITCH" positions alone or in combination with the "ROD MOVEMENT CONTROL SWITCH" will bypass a malfunctioning rod sequence timer?

a. "EMER ROD IN"

b. "NOTCH OVERRIDE"

c. "OFF" and "ROD IN"

d. "NOTCH OVERRIDE" and "ROD OUT NOTCH"

QUESTION: 029 (1.00)

During operation at 65% reactor power a spurious electrical fault causes the "A" reactor recirculation pump discharge valve to close to the 88% open position.

The automatic response of the reactor recirculation system is:

a. the "A" pump trips.

b. scoop tube lockup on both pumps.

c. only the "A" pump speed runs back to 32% of maximum.

d. speed runs back to 32% of maximum on both pumps.

QUESTION: 030 (1.00)

Select the statement which describes the initial changes to core d/p and reactor water level following an increase in recirculation pump speed.

a. Core d/p decreases with a corresponding decrease in water level.

b. Core d/p decreases with a corresponding increase in water level.

c. Core d/p increases with a corresponding decrease in water level.

d. Core d/p increases with a corresponding increase in water level.

QUESTION: 031 (1.00)

Twenty (20) seconds after a small Loss Of Coolant Accident coincident with a loss of normal power the following plant conditions exist:

-- All Automatic systems have functioned as designed
-- Drywell pressure is 5 psig
-- Reactor vessel level is -65 inches

Select the status of the LPCI pumps under these conditions.

a. No LPCI Pumps are running

b. All LPCI pumps are running

c. "A" and "[illegible]" LPCI pumps are running

d. "B" and "D" LPCI pumps are running

QUESTION: 032 (1.00)

Thirty (30) seconds after an automatic initiation of LPCI the operator attempts to close LP-7A, LPCI Heat Exchanger Bypass Valve.

Select the response of LP-7A.

a. The valve will not move from the open position.

b. The valve will close and remain closed.

c. The valve will close and reopen when the hand switch is released.

d. The valve will start to close until dual indication is received, then stop at that position.

QUESTION: 033 (1.00)

Which one of the following describes the logic sequence for the LPCI Loop Select if only the "A" reactor recirculation pump is running at the time of initiation?

a. Trip signal is sent to only the "A" recirculation pump, close signal sent to the recirculation cross tie and bypass valves selected on CRP 905, when reactor pressure decreases to less than 900 psig, the two second timer starts.

b. Trip signal is sent to both recirculation pumps, close signal sent to the recirculation cross tie and bypass valves selected on CRP 905, when reactor pressure decreases to less than 900 psig the two second timer starts.

c. Trip signal is sent to both recirculation pumps and the two second timer starts immediately.

d. Trip signal is sent to only the "A" recirculation pump and the two second timer starts immediately.

QUESTION: 034 (1.00)

Which of the following describes the response of the Feed Regulating Valves to an accident signal coincident with a loss of normal power (LNP)?

a. "Lock Up" as is due to a loss of control signal.

b. Close when feed pump discharge pressure decreases to less than 300 psig.

c. Fail closed from a loss of control air pressure.

d. Shift to "Flow Control" when feed flow increases above 30%.

QUESTION: 035 (1.00)

Following an automatic actuation of Feedwater Coolant Injection (FWCI) the operator has been directed to determine plant status and restore FWCI to standby readiness if allowed.

The following conditions exist:

-- "B" feedwater string pumps selected and operating
-- total feed flow 30%
-- narrow range GEMAC level +30 inches and decreasing

Why is the operator cautioned against depressing the "FEEDWATER FLOW CONTROL RESET" pushbutton with these conditions?

a. It will shift the feedwater regulating valves to the "Level Control" mode.

b. It will shift the feedwater regulating valves to the "Flow Control" mode.

c. It will trip the "B" feedwater pump.

d. It will reset the FWCI initiation signal.

QUESTION: 036 (1.00)

Select the normal and backup supplies of makeup water to the isolation condenser shell side during normal standby operation.

	NORMAL	BACKUP
a.	Demineralized water	Condensate transfer
b.	Fire water	Demineralized Water
c.	Condensate transfer	Demineralized water
d.	Fire water	Condensate transfer

QUESTION: 037 (1.00)

Following a failure of the isolation condenser to automatically initiate at -48" RPV water level, 1-IC-3, Outboard Return valve was manually opened locally from the reactor building. An initial reactor pressure decrease was observed after opening 1-IC-3. Ten minutes later the control room operator observes that reactor pressure is not decreasing.

Select the cause of this abnormal condition.

a. Decay heat is still greater than the design capacity of the isolation condenser.

b. The shell side water has heated up and is not removing as much heat.

c. The buildup of non-condensibles in the tube bundles is inhibiting condensation and natural circulation.

d. A mechanical stop on 1-IC-3 limits the amount of opening when manually operated.

QUESTION: 038 (2.00)

Match the isolation condenser valves in Column A with the power supplies listed in Column B.

The items from column B may be used once, more than once, or not at all and only a single answer may occupy one answer space.

COLUMN A VALVES -------	COLUMN B POWER SUPPLY ------------
a. Inboard Steam Supply Valve, 1-IC-1	1. MCC-E3
b. OutBoard Steam Supply Valve, 1-IC-2	2. MCC-F3
c. Outboard Condensate Supply Valve, 1-IC-3	3. DC-11A-1
d. Inboard Condensate Return Valve, 1-IC-4	4. DC-11A-2
	5. DC-101-AB-1
	6. DC-101-AB-2

QUESTION: 039 (1.00)

Following a small break LOCA the Automatic Pressure Relief (APR) system failed to actuate. Reactor Pressure is stable at 300 psig with level at -60 inches and slowly decreasing. All core spray system components functioned as per design.

Which of the following describes the Core Spray system status?

a. Pumps have started but are not injecting because the "Injection" valves, CS-5A and 5B have not opened.

b. Pumps have started, "Injection" valves, CS-5A and 5B have opened and the system is injecting.

c. Pumps have started, "Injection" valves, CS-5A and 5B have opened but the system is not injecting.

d. Pumps have not started, "Injection" valves, CS-5A and 5B have not opened.

QUESTION: 040 (1.00)

Core Spray system operability testing is in progress with "Pump Disch" valve, CS-4A closed and "Injection" valve, CS-5A open.

Select the automatic response of the these valves if a valid core spray initiation signal is received in this configuration.

a. CS-5A will remain open, CS-4A will remain closed.

b. CS-5A will close, CS-4A will remain closed.

c. CS-5A will close then CS-4A will open. CS-5A will reopen when the reactor pressure permissive signal is satisfied.

d. CS-5A will remain open, CS-4A will open when the reactor pressure permissive signal is satisfied.

QUESTION: 041 (1.00)

Standby liquid control system #2 is injecting at the Technical Specification minimum flow rate. The initial volume of the liquid poison tank was at 46% which corresponds to the Technical Specification required minimum volume.

Select the minimum time required to inject the "cold shutdown boron weight" (step FSQ 25).

a. 19 minutes

b. 25 minutes

c. 31 minutes

d. 47 minutes

QUESTION: 042 (1.00)

A reactor startup is in progress with IRMs on range 2. While withdrawing SRM detectors, the operator inadvertently selects IRM detector 12 drive unit.

Select the response of the IRM system.

a. IRM 12 detector will not withdraw because companion APRM 2 is downscale.

b. IRM 12 detector will withdraw and "IRM channel A HI HI FLUX/INOP" annunciator will be received.

c. A control rod block is generated when the detector begins to withdraw.

d. No action occurs because the range selector switch for IRM 12 is on range 2 and the mode switch is not in "RUN."

QUESTION: 043 (1.00)

Given the following plant conditions:

-- fraction of rated power (FRP) 0.87
-- maximum fraction of limiting power density (MFLPD) 0.87
-- total core flow 55.2 mlbm/hr
-- total recirculation flow 25.8 mlbm/hr

What is the required setpoint of the APRM channel "A" flow biased scram?

a. 104.2

b. 106.7

c. 108.2

d. 109.2

QUESTION: 044 (1.00)

The reactor was operating at 100% power when a scram occurred and one bypass valve failed open. Reactor pressure is decreasing.

The operator should attempt to close the bypass valve by:

a. depressing the Vacuum Trip #2 pushbutton.

b. positioning the Bypass Valve Opening Jack to the "LOWER" position.

c. positioning the Bypass Valve Test Switch to the "TEST" position.

d. shifting pressure control to the manual pressure regulator (MPR).

QUESTION: 045 (1.00)

A drywell pressure increase resulting from a loss of drywell cooling has started the diesel generator (DG). The operator places the Diesel Generator LOCA Start Bypass Switch in the "BYPASS" position and stops the DG. The operator then restores drywell cooling and drywell pressure decreases to 1 psig.

Which of the following will start the DG under these conditions?

a. "START/STOP" switch on CRP 908 or loss of voltage on busses 14E and 14F.

b. Loss of voltage on busses 14E and 14F or reactor low-low water level.

c. "START/STOP" switch on CRP 908 or high drywell pressure.

d. Reactor low-low water level or high drywell pressure.

QUESTION: 046 (1.00)

The rod worth minimizer (RWM) has applied a control rod withdrawal block due to an incorrect rod movement.

What control rod motion is possible with the above condition?

a. A full-out control rod may be withdrawn for a coupling check.

b. The rod causing the block may only be withdrawn.

c. Any control rod in the same group as the rod causing the block may be inserted.

d. No control rod motion is possible until the RWM is placed in "Bypass."

QUESTION: 047 (1.00)

The operator has placed the reactor water cleanup (RWCU) EOP keylock switches (color brown) on CRP 902 to "BYPASS."

Select the statement that lists all RWCU isolation signals that are bypassed by this action.

a. Reactor water low-low

b. Standby liquid control system initiation

c. Reactor water low-low and non-regenerative heat exchanger outlet high temperature

d. Reactor water low-low and standby liquid control system initiation

QUESTION: 048 (1.00)

A reactor startup following a refueling outage is being performed. The first eight control rods in the established rod withdraw sequence are at a final notch position of 36.

Select the method used to verify these eight control rods are coupled to their drive assembly.

a. By exercising these control rods after the reactor is critical and verifying instrument response.

b. By electrically disconnecting the insert directional control valve, 123 and verifying "stall flow" is less then 3.5 gpm when first attempting to withdraw the rod.

c. By verifying a rod drift alarm is NOT received when first attempting to pull the rod from position 00.

d. By attempting to pull each control rod past position 48, verifying the Rod Overtravel Annunciator is NOT received, then inserting back to position 36.

QUESTION: 049 (1.00)

Given the following conditions:

-- The plant is at 55% power
-- APRM 3 fails "Downscale"
-- No operator actions have been taken

Select the expected automatic response of the rod block monitor (RBM) system.

RBM channel 7:

a. automatically shifts to APRM 2.

b. output trip functions are bypassed.

c. generates a channel "Downscale" trip.

d. generates a channel "Inoperative" trip.

QUESTION: 050 (1.00)

Which one of the following conditions will trip the control room ventilation system supply fan HVH-8 and exhaust fan HVT-11?

a. Manual initiation of the standby gas treatment system.

b. Control room Halon discharge

c. High chlorine sensed in the outside air supply

d. High reactor building ventilation radiation

QUESTION: 051 (1.00)

The plant is operating at 100% power with Off-Gas radiation monitor, 1705-03B deenergized and red tagged for maintenance.

Annunciator alarm "OFF-GAS HI HI RADIATION" is received on CRP 904.

Select the Off-Gas system response if all automatic functions operate as designed.

a. Off-Gas isolation valve will immediately close.

b. Off-Gas flow to the stack will divert to the 30 minute holdup line.

c. Off-Gas isolation valve will close in 15 minutes.

d. Off-Gas isolation valve will remain open indefinitely.

QUESTION: 052 (1.00)

The Control Room Operator records the following integrator data, six hours after the previous data was taken:

-- Drywell floor drain sump; gallons pumped since last reading = 576 gallons
-- Drywell equipment drain sump; gallons pumped since last reading = 1764 gallons

Select the Reactor Coolant System Unidentified Leakage rate.

a. 1.6 gpm.

b. 3.3 gpm.

c. 4.9 gpm.

d. 6.5 gpm.

QUESTION: 053 (1.00)

During operation at 100% power, primary containment is being vented with the "B" train of standby gas treatment (SBGT). Valve 1-AC-10, "Exhaust from Drywell and Torus to SBGT Isolation" is open.

Select the response of the SBGT system if an automatic actuation signal is received with this system configuration.

"A" train starts, . . .

a. reactor building exhaust isolation valves, 1-SG-1A and 1B remain closed, valve 1-AC-10 remains open.

b. reactor building exhaust isolation valves, 1-SG-1A and 1B open, valve 1-AC-10 closes.

c. reactor building exhaust isolation valves, 1-SG-1A and 1B open, valve 1-AC-10 remains open.

d. "B" train stops, reactor building exhaust isolation valves, 1-SG-1A and 1B open, valve 1-AC-10 closes.

QUESTION: 054 (1.00)

When determining actual core flow in single recirculation pump operation, the operator is directed to increase running recirculation pump speed slightly while observing flow in the idle loop.

The total core flow indicated on the dual-pen recorder on CRP 905 would be correct if flow through the idle loop:

a. increases.

b. decreases.

c. goes to zero.

d. does not change.

QUESTION: 055 (1.00)

The Reactor Building "HVAC ISOLATIONS-RX BLDG" keylock switches (color turquoise) have been placed in "BYPASS" on CRP 902 as directed by EOP 590.17.

What function is provided when these switches are placed in the "BYPASS" position?

a. Allows operation of reactor building HVAC with high reactor building ventilation exhaust radiation.

b. Allows operation of the reactor building HVAC transfer fans if the exhaust fans are isolated.

c. Bypasses the reactor building ventilation exhaust radiation, high drywell pressure and low RPV water level isolations of reactor building HVAC.

d. Bypasses the high drywell pressure and low RPV water level isolations of reactor building HVAC only.

QUESTION: 056 (1.00)

The plant is at 175 degrees F operating on shutdown cooling. The Reactor Vessel Head has been removed.

Following an isolation of shutdown cooling which of the following methods can be used to remove heat from the core?

a. Perform a feed an bleed using RWCU and Condensate.

b. Initiate the isolation condenser.

c. Start a second CRD pump.

d. Raise vessel level to 40 inches to initiate natural circulation.

QUESTION: 057 (1.00)

A small break has occurred in the drywell causing drywell pressure to increase. The operator initiated drywell venting to attempt to control drywell pressure increase.

EOP 580 requires that venting the drywell be secured at what condition AND what is the reason for securing venting.

a. 212 degrees F drywell temperature because this is an indication that the cause of the pressure increase is a leak in the drywell.

b. 2 psig drywell pressure because venting via normal paths above this pressure may damage ventilation components.

c. 212 degrees F drywell temperature because damage could occur to SBGT components above this temperature.

d. 2 psig drywell pressure because further venting may result in unacceptable reduction of non-condensibles in the drywell.

QUESTION: 058 (1.00)

Step PCP/DWT-10 of EOP 580 "Primary Containment Control" verifies that torus level is less than 24 ft.

What is the basis for this step?

a. Ensuring that the torus to drywell vacuum breakers are not submerged.

b. Ensuring that the torus spray ring is not submerged.

c. Ensuring that the torus vent penetration is not submerged.

d. Ensuring that connections to the reactor building to torus vacuum breakers are not submerged.

QUESTION: 059 (1.00)

EOP 570, RPV Flooding, is being performed.

The following conditions exist.

- Time since shutdown 50 minutes.
- Torus pressure - 15 psig.
- Torus level - 20.5 ft.

Which of the following conditions would allow injection to be terminated in order to determine if water level indication was restored?

	Number of Open SRVs	RPV Pressure	Flooding Interval
a.	3	80 psig	80 minutes
b.	4	100 psig	50 minutes
c.	1	220 psig	65 minutes
d.	2	120 psig	75 minutes

QUESTION: 060 (1.00)

A loss of feed has occurred causing EOP 570 to be implemented. The following conditions exist.

- RPV water level = -60 inches and slowly increasing
- CRD has been maximized.
- RPV pressure = 800 psig and being controlled by Isolation Condenser.
- Feedwater is being restored and estimated to be available within 5 minutes.

The operator observes the SRV's being opened by the Autopressure Reduction System.

What action should be taken?

a. Continue in EOP 570, "RPV Control." Place the "APR Bypass" switch to "Bypass" to close the SRV's.

b. Continue in EOP 570, "RPV Control." Allow pressure to reduce.

c. Enter EOP 570 sheet 3, "Emergency Depressurization/Steam Cooling." Place SRV's to open.

d. Continue in EOP 570, "RPV Control." Place SRV's to open.

QUESTION: 061 (1.00)

A fuel element failure caused a main steam isolation and reactor scram. The following conditions exist:

- Indicated RPV level = -170 inches and decreasing
- RPV pressure = 575 psig and slowly decreasing
- Isolation Condenser Vent Radiation Monitor Alarming
- Core Spray and LPCI pumps operating on minimum flow
- Health Physics technicians report contamination on east side of reactor building.

What action(s) should be taken?

a. Maintain the Isolation Condenser in service unless not required for cooldown.

b. Maintain Isolation Condenser in service and initiate an Emergency Depressurization.

c. Isolate Isolation Condenser and initiate an Emergency Depressurization.

d. Isolate Isolation Condenser when injection from Core Spray or LPCI is verified.

QUESTION: 062 (1.00)

While performing EOP 575 "Failure to Scram" and EOP 580 "Primary Containment Control" the operator observes the following conditions:

- A failure to scram has occurred and reactor power is 15%
- Bypass valves tripped due to loss of vacuum
- Drywell temperature = 360 F
- RPV pressure = 125 psig

Which one of the following sections of EOP's should be implemented or continue to be implemented?

a. Section FLD "ATWS RPV Flooding" of EOP 575.

b. Section FSL "RPV Level" of EOP 575.

c. Section FSP "RPV Pressure" of EOP 575.

d. Section DEP "ATWS Emergency Depressurization" of EOP 575.

QUESTION: 063 (1.00)

Torus water level is increasing due to a break on the reactor recirculation line. The operators are implementing the torus water level portion of Primary Containment Control. The operator has determined that torus level and RPV pressure cannot be maintained below the SRV tail pipe level limit.

If adequate core cooling can be assured which of the following systems should be secured?

a. Core Spray

b. Feedwater Coolant Injection

c. Low Pressure Coolant Injection

d. Control Rod Drive

QUESTION: 064 (1.00)

A steam leak has occurred in the drywell.

- Drywell temperature is 210 degrees F.
- RPV pressure is 450 psig
- Narrow Range ATWS indicate -42 inches

Select the condition that describes level condition of the Narrow Range ATWS indicators.

a. Actual RPV level is below the lower tap for the instrument.

b. Actual water level is greater than indicated on the level instrument.

c. Actual level cannot be determined from the instrument due to potential flashing of the reference leg.

d. Indicated level is accurate for trend information but is not accurate for actual determination of level.

QUESTION: 065 (1.00)

A loss of Torus Water Level is occurring due to a leak in the torus. The operator has determined that level cannot be maintained above 10.5 ft.

Why is emergency depressurization required if level cannot be maintained above 10.5 ft?

a. The bottom of the downcomers is at 10.5 ft.

b. The Safety Relief Valves discharge at 10.5 ft.

c. Below 10.5 ft the torus has insufficient heat capacity for depressurization.

d. Below 10.5 ft the measuring capability for torus water temperature is lost.

QUESTION: 066 (1.00)

Step LP-8 of EOP 575, "Failure to Scram", specifies systems to be used to maintain water level.

Why is Core Spray not used in this condition to maintain level?

a. Core spray water quality is inferior to the systems listed.

b. At this point in the ATWS, reactor pressure precludes use of core spray.

c. Core spray injects inside the core shroud.

d. Core spray lacks precise flow control to avoid rapid level changes.

QUESTION: 067 (1.00)

A steam leak has occurred in the drywell. EOP 580, Primary Containment Control, is being implemented. Initiation of drywell sprays is delayed due to LPCI pumps being required for Adequate Core Cooling.

The following conditions exist when drywell sprays are initiated.

- Drywell pressure = 15 psig
- Drywell bulk temperature = 300 degrees F

After drywell sprays are initiated the operator observes drywell pressure decreasing to the left side of the DW Spray Initiation Limit.

Select the correct response to this condition.

a. Continue spraying until pressure decreases to 2 psig.

b. Attempt to throttle drywell spray to maintain Drywell pressure and Drywell temperature within the DW Spray Initiation Limit.

c. Secure Drywell spray when the DW Spray Initiation Limit is reached.

d. Continue spraying until drywell pressure decreases to 9 psig.

QUESTION: 063 (1.00)

A fuel element failure initiated a main steam line isolation and a scram signal. The scram signal failed to insert control rods. Torus cooling was initiated.

The following conditions exist.

- Torus water level = 11 feet
- Torus water temperature = 200 degrees F and increasing
- RPV pressure was rapidly reduced to 500 psig from 1000 psig 15 minutes ago.

Which of the following actions shall be taken if torus temperature increases to 210 F?

a. Emergency Depressurize.

b. Reduce RPV pressure to 400 psig in accordance with allowable cooldown rates.

c. Reduce RPV pressure to 400 psig immediately. Cooldown rate limits may be exceeded.

d. Wait until temperature increases to 213 F then reduce RPV pressure.

QUESTION: 069 (1.00)

Instrumentation Technicians were performing maintenance on the TIP machines when one of the TIP sources was exposed causing high radiation conditions in the Reactor Building.

Radiation monitors are reading as follows:

- TIP Cubicle (Channel 10) = 10,000 mr/hr
- TIP Drive Mechanism (Channel 11) = > 100 mr/hr
- CRD Removal Hatch Area (Channel 12) = 1000 mr/hr
- North CRD Hydraulic Control Area (Channel 13) = > 100 mr/hr

What actions are required per EOPs?

a. Shutdown the reactor and concurrently perform actions of ONP 509, Excessive Radioactive Levels.

b. Scram and Enter EOP 570 Sht. 1, RPV Control.

c. Continue to operate and concurrently perform actions of ONP 509, Excessive Radioactive Levels.

d. Scram and Enter EOP 570 Sh 3, Emergency Depressurization.

QUESTION: 070 (1.00)

The following conditions exist:

- EOP 585 has been entered due to high temperature in the steam tunnel
- Fire alarms have been received in the reactor building
- Main Steam Isolation valves have closed
- CRD is maintaining RPV level
- Isolation Conde^ er is in service
- Drywell Venti in progress

Which of the following systems should be isolated if they are discharging into the secondary containment?

a. Control Rod Drive

b. Drywell Vent

c. Isolation Condenser

d. Fire suppression

QUESTION: 071 (1.00)

Due to a loss of power, Inst. Ai. or N2 to Drywell valve, 1-AC-50 has failed shut. While investigating this failure an inadvertent main steam isolation has occurred due to operator error.

Which of the following methods is to be used for controlling pressure?

a. Bypass valves.

b. Manually operate SRV's.

c. Allow SRV's to automatically control pressure.

d. Use Steam Jet Air Ejectors.

QUESTION: 072 (1.00)

Due to a leaking relief valve EOP 580 "Primary Containment Control" was entered at 85 degrees F torus water temperature.

As torus water temperature increases, when is the reactor required to be scrammed?

a. 95 degrees F

b. 105 degrees F

c. 110 degrees F

d. 120 degrees F

QUESTION: 073 (1.00)

A failure to scram has occurred. EOP 575, "Level Power Control," is presently being implemented.

Select the correct actions concerning the Main Steam Line Isolation valves.

a. If the MSIV's are open then bypass all isolation signals.

b. If the MSIV's are open then bypass only low level isolation signals.

c. If the MSIV's are closed bypass all signals then reopen the MSIV's.

d. If the MSIV's are closed bypass all signals except high radiation and high flow then reopen the MSIV's.

QUESTION: 074 (1.00)

Select the statement identifying the reason reactor power decreases when reactor water level is lowered during a failure to scram (ATWS) event.

a. Lowering level below the moisture separator removes the flowpath thereby minimizing flow through the core.

b. Lowering level reduces the pressure in the core by reducing the head of water above the core.

c. Lowering level decreases the differential pressure between outside the shroud and inside the core.

d. Lowering level reduces power by increasing the subcooling of the water entering the core.

QUESTION: 075 (1.00)

Step PCP/DWT-12 of EOP 580 is to be performed. The following plant conditions exist.

- RPV water level = - 140 inches.
- RPV Pressure = 90 psig
- Torus Pressure = 45 psig
- Four SRVs are open
- Both LPCI pumps are injecting to maintain level
- No Core Spray Pumps are running

Which statement describes the use of LPCI pumps for drywell spray?

a. LPCI pumps cannot be used in this condition for drywell spray.

b. LPCI pumps can be used for drywell spray provided RPV level remains above 2/3 core height.

c. LPCI pumps can be used for drywell spray provided RPV level remains above 1/2 core height.

d. LPCI pumps can be used for drywell spray provided four SRV's are open and RPV pressure remains above 50 psig.

QUESTION: 076 (1.00)

A failure to scram has occurred. Boron injection was commenced.

Select the condition below that will allow termination of boron injection.

a. The Reactor Shutdown Criteria is satisfied.

b. All rods are full in.

c. ONP-502 Emergency Plant Shutdown is entered following injection of 30% of the SLC tank level.

d. The entry condition for EOP 575 "ATWS - RPV Control" has cleared.

QUESTION: 077 (1.00)

A leak occurred in the RWCU system in the Reactor Building. Reactor Building Ventilation isolated when RPV level decreased below +8 inches.

Which of the following conditions would PROHIBIT restarting Reactor Building Ventilation?

a. RPV level remaining below +8 inches.

b. Drywell pressure increasing to 2.2 psig due to isolation of drywell cooling.

c. Area radiation monitors in RWCU area alarming.

d. Reactor Building Ventilation radiation increasing to 20 mrem/hr.

QUESTION: 078 (1.00)

Switch boards 101A and 101B in the 125 VDC system have been deenergized due to electrical faults. Power has remained stable at 78%.

Select the required action.

a. Maintain plant stable and request additional assistance in shutting down.

b. Locally runback recirculation pumps as soon as possible then maintain the plant stable.

c. Scram the reactor immediately.

d. Immediately commence rod insertion to reduce power. Do not attempt to adjust recirculation pump speed.

QUESTION: 079 (1.00)

Recirculation pumps are operating on the Master controller with reactor power at 83%.

Identify the response of the Recirculation pumps to a complete loss of Vital AC.

a. Scoop tubes will lockup.

b. Pumps will runback to 32% due to loss of signal from feedwater flow.

c. Recirculation pumps will trip due to loss of level indication.

d. Recirculation pump speed will decrease to minimum due to loss of signal from the master controller.

QUESTION: 080 (1.00)

An irradiated fuel bundle is being removed from the core when it is dropped resulting in the bundle lying across the top of the core. Reactor building ventilation has NOT isolated.

Identify the area or areas required to be evacuated.

a. Refuel Floor only.

b. Refuel Floor and Drywell.

c. Drywell and Reactor Building.

d. Reactor Building including the refuel floor.

QUESTION: 081 (1.00)

A loss of all Reactor Building Closed Cooling Water (RBCCW) pumps has occurred.

Which of the following systems can be cross tied with RBCCW to supply cooling to plant components?

a. Turbine Building Closed Cooling Water

b. Turbine Building Secondary Closed Cooling Water

c. Service Water

d. Emergency Service Water

QUESTION: 082 (1.00)

Following a trip of one recirculation pump from 80% power, select the condition requiring a manual reactor scram.

a. The select rod insert fails to reduce power oscillations.

b. Several LPRM downscale alarms are coming in and clearing every 2 to 4 seconds.

c. APRM power oscillations are 7% peak-to-peak on 4 channels and 8% on 2 channels.

d. Multiple LPRM upscale and downscale alarms annunciate periodically.

QUESTION: 083 (1.00)

During performance of ONP 502, Emergency Plant Shutdown, the operator attempts to reset the scram. Rod Groups 1 and 4 reset as indicated by the lights on the 905 panel but rod groups 2 and 3 will not reset.

Select the effect of continued operation in this condition.

a. The scram discharge volume will not drain.

b. A flow path exists from the reactor to the reactor building via the scram discharge volume.

c. The differential pressure across closed scram isolation valves will be excessive.

d. Potential exists for damage of CRD mechanisms if another scram signal is received with only some scram isolation valves closed.

QUESTION: 084 (1.00)

A loss of all station A.C. Power has occurred.

Select the condition that indicates when a RBCCW pump is required to be started.

a. Immediately following the diesel tying to the bus.

b. When drywell coolers are required by EOPs for maintaining drywell temperature.

c. When a CRD pump is required for maintaining RPV level.

d. When load capacity becomes available on the diesel and/or gas turbine following starting of the TBSCCW pump.

QUESTION: 085 (1.00)

A loss of both CRD pumps has occurred.

When is a reactor scram required?

a. When it is determined that neither pump can be restarted.

b. When high temperature alarms are received on more than one control rod.

c. When accumulator alarms are received on more than one control rod.

d. Ten minutes after loss of both pumps.

QUESTION: 086 (1.00)

A break has occurred in the instrument air system as indicated by annunciators and starting of the "lag" instrument air compressor. Pressure has decreased to 50 psig.

Select the automatic response that should have occurred.

a. Repositioning of reactor building ventilation dampers.

b. Reactor scram.

c. Main Steam Isolation Valve closure.

d. IA-20 from the Station Air System opening.

QUESTION: 087 (1.00)

Due to a fire, the control room surveillance alm was evacuated prior to closing the inboard MSIV's from CRP 903.

Select the method used to close the MSIV's.

a. Remove control power fuses for the solenoids for the MSIV's.

b. Initiate a Group I isolation by deenergizing the RPS momentarily.

c. Valve out air and bleed off air to the outboard MSIV's.

d. Place Appendix R isolation switches to "ISOLATE."

QUESTION: 088 (1.00)

Shutdown cooling is in operation with one shutdown cooling pump in operation. Both reactor recirculation pumps have just been secured.

Select the flowpath that will exist in this condition that causes reduced shutdown cooling flow through the core.

a. Backflow through the B recirculation loop.

b. Backflow through the A recirculation loop.

c. Backflow through A loop jet pumps.

d. Backflow through B loop jet pumps.

QUESTION: 089 (1.00)

Following a complete Loss of Shutdown Cooling, temperature readings indicate a 1 degree F increase in bulk water temperature every 10 minutes. Assume the reactor vessel head is ON, no other parameters change and current temperature is 164 degrees F.

How much TIME is allowed before primary containment integrity MUST be established?

a. 260 minutes

b. 360 minutes

c. 480 minutes

d. 560 minutes

QUESTION: 090 (1.00)

It is desired to decrease pressure in containment with a group II isolation present. Which of the following sets of valves can be used for venting with a group II isolation signal present?

a. AC-11, Torus Vent Valve, and AC-8, Discharge to Reactor Building Ventilation.

b. AC-12, Torus Vent Valve, and AC-8, Discharge to Reactor Building Ventilation.

c. AC-7, Drywell Vent Valve, and AC-10, Discharge to SBGT System.

d. AC-9, Drywell Vent Valve, and AC-10, Discharge to SBGT System.

QUESTION: 091 (1.00)

A Main Steam Isolation has occurred causing a pressure increase. Select the conditions that comply with the PREFERRED method of using SRVs for pressure control.

a. Allow the SRV's to cycle automatically.

b. Open SRV's "A" and "E" until pressure has decreased by 100 to 150 psi. After pressure increases open SRV's "E" and "D" to achieve a similar pressure reduction.

c. Open SRV "E" and allow pressure to decrease by 50 psig. Continue to use "E" as required to maintain pressure.

d. Open SRV "A" to reduce pressure by 100 to 150 psig. After pressure increases open SRV "E" to achieve a similar pressure decrease.

QUESTION: 092 (1.00)

A loss of normal power has occurred and the Gas Turbine Generator failed to start. The diesel generator has been operating at 2800 kW for 45 minutes when a "DIESEL GENERATOR COOLANT TEMPERATURE HIGH" annunciator is received.

What is the required operator action?

a. Continue operation at 2800 kW but monitor temperature to ensure that it does not exceed 205 degrees F.

b. Reduce load to 2665 KW and investigate the alarm.

c. Continuously reduce load until the alarm clears.

d. Reduce load as necessary to prevent coolant temperature from exceeding 205 degrees F.

QUESTION: 093 (1.00)

A scram has occurred at 58% power. As power decreases the turbine is required to be tripped at what power level?

a. 100 MWE

b. 50 MWE

c. 50 MWE reverse power

d. 100 MWE reverse power

QUESTION: 094 (1.00)

A complete loss of DC has occurred. Subsequently a scram occurred. Select the required immediate action.

a. Control RPV water level with 1-FW-5C (10% valve).

b. Close the MSIV's.

c. Trip both recirculation pumps.

d. Send an operator to trip the turbine locally.

QUESTION: 095 (1.00)

Reactor was at 82% power when a loss of Vital AC occurred. The reactor has scrammed. RPV level has been restored to +30 inches as indicated on the ATWS level indication.

Select the immediate action required when RPV level is verified.

a. Take manual control of the FRVs.

b. Transfer control of feedwater flow to the Startup valve.

c. Secure one feed pump.

d. Secure all feed pumps.

QUESTION: 096 (1.00)

A loss of feedwater heating has occurred.

Which of the following conditions requires that recirculation flow be runback to minimum?

a. APRM rod block alarm occurs.

b. The instability region is entered.

c. Feedwater temperature decreases by 50 degrees F.

d. LPRM high alarms occur.

QUESTION: 097 (1.00)

The plant was operating at 100% power when a turbine blade broke and pierced the turbine casing causing a turbine trip and reactor scram. A small fire was reported in the vicinity of the turbine. One half of the control rods failed to insert. Reactor power decreased to Range 5 of the intermediate range.

Five minutes after the transient, the following conditions exist;

- Reactor power is on Range 3 of the IRMs.
- RPV level is stable at +30 using CRD.
- The isolation condenser is in service controlling reactor pressure.
- The fire has been extinguished.

Select the required Emergency Action Level classification.

a. None

b. Unusual Event

c. Alert

d. Site Area Emergency

QUESTION: 098 (1.00)

A Main Steam Line Isolation has occurred due to High Radiation. The dose rate one foot from a 0.5 ml RCS sample is 50 mr/hr. RPV level is -101" and the SCO has directed the operator to vent the drywell in response to drywell pressure exceeding the Pressure Suppression Curve.

What would the emergency action level for this condition?

a. Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

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

ANSWER: 001 (1.00)

b.

REFERENCE:

OP276/2276/3276 page 24 LO 0672200 ONP 504 page 1

[3.3/4.3]
294001A111 ..(KA's)

ANSWER: 002 (2.00)

a. 1

b. 1

c. 4

~~d. 5~~ deleted TBW 1/24/94

REFERENCE:

SHP 4906 pages 2&3 LO 12

[3.3/3.8]

294001K103 ..(KA's)

ANSWER: 003 (1.00)

a.

REFERENCE:

New 10CFR20 Training Lesson Plan page 6. LO 6.

[3.3/3.8]

294001K103 ..(KA's)

MASTER COPY

ANSWER: 004 (1.00)

c.

REFERENCE:

CHECK AT FACILITY FOR REFERENCE

[3.3/3.8]

294001K103 ..(KA's)

ANSWER: 005 (1.00)

a.

REFERENCE:

1-OPS-6.09 Rev. 0, "Control Room Habitability", 1.1.
No Facility Objective.

KA [3.5/3.8]

294001K116 ..(KA's)

ANSWER: 006 (1.00)

b.

REFERENCE:

ACP -QA-2.06A page 3

[3.9/4.5]

294001K102 ..(KA's)

ANSWER: 007 (1.00)

d.

REFERENCE:

C-OP-200 page 19 No Facility Objective

[3.1/3.2]

294001A104 ..(KA's)

ANSWER: 008 (1.00)

c.

REFERENCE:

OP 341 page 73 No Facility Objective

[3.3/3.6]

294001K107 ..(KA's)

ANSWER: 009 (1.00)

b.

REFERENCE:

OP 275/227/3275 page 3. LO 05122200

[3.6/4.2]

294001A110 ..(KA's)

ANSWER: 010 (1.00)

c.

REFERENCE:

4160 VAC Text page 11 & OP 341 page 52 LO 11.b

[3.3/3.6]

294001K107 ..(KA's)

ANSWER: 011 (1.00)

c.

REFERENCE:

1-OPS-6.25 Rev. 1, "Fuse Control", Discussion 2., page 1.
No Facility Objective.

KA [3.9/4.5]

294001K102 ..(KA's)

ANSWER: 012 (1.00)

d.

REFERENCE:

ACP-12.09 Rev. 1 "High Radiation Area Key Control", Section 5.3, page 4.
No Facility Objective.

KA [3.2/3.7]

294001K105 ..(KA's)

ANSWER: 013 (1.00)

a.

REFERENCE:

10 CFR 55.53 No Facility Objective

[2.7/3.7]

294001A103 ..(KA's)

ANSWER: 014 (1.00)

d.

REFERENCE:

EPOP 4411, Director of Station Emergency Operations, page 2.

[2.9/4.7]

294001A116 ..(KA's)

ANSWER: 015 (1.00)

b.

REFERENCE:

EPIP 4400 Attachment 1, sheet 4 of 23 NOTE at top of page and Sheet 9
No Facility Objective

[2.9/4.7]

294001A116 ..(KA's)

ANSWER: 016 (1.00)

c.

REFERENCE:

Technical Specification 3.6.C.3.a and Bases 3.6.C No Facility
Objective

[2.9/3.4]

294001A114 ..(KA's)

ANSWER: 017 (1.00)

d.

REFERENCE:

Technical Specification 3.4.C LO 13 [CAF FOR T.S. INTERPRETATION]
LO 13

[3.6/4.4]

211000G005 ..(KA's)

ANSWER: 018 (1.00)

~~a.~~ d.

REFERENCE:

Technical Specification Table 3.1.1 notes 10 and ** ODI 1-OPS-6.26
APRM Text page 26 LO 6.a&b.

[3.5/3.6]

215005K306 ..(KA's)

ANSWER: 019 (1.00)

d.

REFERENCE:

Technical Specification 3.14 RPS Text LO 8.f

[2.9/3.8]

241000G005 ..(KA's)

ANSWER: 020 (1.00)

deleted
TDW 1/21/94

d.

REFERENCE:

Technical Specification 3.3.D LO 20.e.

[3.3/3.9]

201003G005 ..(KA's)

ANSWER: 021 (1.00)

b.

REFERENCE:

Technical Specification 3.7.D.2 and 4.7.D.2 SDC Text LO 9.a

[2.8/2.9]

205000K502 ..(KA's)

ANSWER: 022 (1.00)

c.

REFERENCE:

Technical Specification 3.6.J.1 and 4.6.J.1.b LO 9.a

[2.9/3.6]

256000G005 ..(KA's)

ANSWER: 023 (1.00)

d.

REFERENCE:

Technical Specification Table 3.2.1 action 3. Main Steam Text LO 12.a

[3.2/4.1]

239001G005 ..(KA's)

ANSWER: 024 (1.00)

d.

REFERENCE:

FWCI Lesson Text 334 page 9 LO 6.a

[4.3/4.4]

206000A101 ..(KA's)

ANSWER: 025 (1.00)

b.

REFERENCE:

Technical Specification 4.12.a [NO FACILITY REFERENCE MATERIAL PROVIDED]

[3.2/3.3]

286000K109 ..(KA's)

ANSWER: 026 (1.00)

d.

REFERENCE:

Control Rod Drive Text ID 302 page 6 section 3.2.1.2. LO 10.a

[2.9/3.1]

201001K201 ..(KA's)

ANSWER: 027 (1.00)

b.

REFERENCE:

Control Rod Drive Text ID 302 page 16 section 4.1.6 LO 3.k

[3.5/3.4]

201001A105 ..(KA's)

ANSWER: 028 (1.00)

a.

REFERENCE:

Reactor Manual Control System Text ID 32A page 12 section 5.4.3 LO 14.b

[3.5/3.5]

201002A402 ..(KA's)

ANSWER: 029 (1.00)

a.

REFERENCE:

Reactor Recirculation Speed Control System Text 1301B Page 8 Section 5.4.1. LO 4.b

[3.0/3.0]

202002K402 ..(KA's)

ANSWER: 030 (1.00)

c.

REFERENCE:

Reactor Recirculation Flow Control System Text 31B page 17 LO 10.a & e

[3.1/3.2]

202002K109 ..(KA's)

ANSWER: 031 (1.00)

c.

REFERENCE:

LPCI Text 335 page 18 LO 18

[3.6/3.7]

203000K601 ..(KA's)

ANSWER: 032 (1.00)

c.

REFERENCE:

LPCI Text page 18 LO 22

[3.9/4.1]

203000K410 ..(KA's)

ANSWER: 033 (1.00)

b.

REFERENCE:

LPCI Text 335 page 22 LO 24

[4.2/4.6]

203000A307 ..(KA's)

ANSWER: 034 (1.00)

b.

REFERENCE:

FWCI Text 334 page 11 LO 8.a

[3.5/3.5]

206000A202 ..(KA's)

ANSWER: 035 (1.00)

a.

REFERENCE:

OP 334 page 9 step 5.4.1 LO 11

[3.1/3.1]

206000K503 ..(KA's)

ANSWER: 036 (1.00)

d.

REFERENCE:

Isolation Condenser Text 307 LO 13 e.& f.

[3.5/3.7]

207000K602 ..(KA's)

ANSWER: 037 (1.00)

c.

REFERENCE:

OP-307 page 11 step 5.3 LO 7.d

[3.8/3.8]

207000A208 ..(KA's)

ANSWER: 038 (2.00)

a. 2

b. 6

c. 5

d. 2

REFERENCE:

Isolation Condenser Text 307 pages 3&31 LO 8.a-d.

[3.6/3.8]

207000K201 ..(KA's)

ANSWER: 039 (1.00)

b.

REFERENCE:

Core Spray Text 336 pages 4, 11 and figure 7 LO 7.b., f., g.

[3.7/3.7]

209001A104 ..(KA's)

ANSWER: 040 (1.00)

d.

REFERENCE:

Core Spray Text page 11 LO 10.e OP 336

[3.6/3.6]

209001A301 ..(KA's)

ANSWER: 041 (1.00)

c.

REFERENCE:

Technical Specifications 4.4.A.1 and 3.4.C EOP 575 sheet 1 step FSQ-25 LO 6.c

[3.1/4.2]

211000G006 ..(KA's)

ANSWER: 042 (1.00)

c.

REFERENCE:

IRM Text page 14 LO 4.c

[3.7/3.7]

215003K401 ..(KA's)

ANSWER: 043 (1.00)

b.

REFERENCE:

Technical Specification 2.1.2.A.1.b LO.7.b

[3.6/3.6]

215005K505 ..(KA's)

ANSWER: 044 (1.00)

a.

REFERENCE:

OP314 page 46 Main Turbine Text page 27 No Facility Training Materia. on MPR/EPR or Learning Objectives Provided

[4.1/4.2]

241000A203 ..(KA's)

ANSWER: 045 (1.00)

a.

REFERENCE:

Diesel Generator Text page 37 No facility learning objectives supplied for DG system

[3.4/3.5]

264000G007 ..(KA's)

ANSWER: 046 (1.00)

a.

REFERENCE:

RWM Text pages 12 and 13 LO 6.c [CHECK AT FACILITY CONCERNING CORRECTION OF ERRORS ON PAGE 13]

[3.5/3.5]

201006K512 ..(KA's)

ANSWER: 047 (1.00)

d.

REFERENCE:

RWCU Text page 55 LO 6.g

[3.4/3.4]

204000A213 ..(KA's)

ANSWER: 048 (1.00)

a.

REFERENCE:

Technical Specification 4.3.B.1.b CRD Text LO 20.c

[3.5/3.8]

214000A203 ..(KA's)

ANSWER: 049 (1.00)

b.

REFERENCE:

Rod Block Monitor Lesson Plan Section 5.8.2 LO 7.b

[3.1/3.3]

215002A203 ..(KA's)

ANSWER: 050 (1.00)

b.

REFERENCE:

HVAC lesson text pages 54, 55 and 56 No Facility Objective

[2.9/3.0]

290003K107 ..(KA's)

ANSWER: 051 (1.00)

c.

REFERENCE:

CRAB 904A-1 Window 3-1 LO 6.e

[3.6/3.7]

272000A302 ..(KA's)

ANSWER: 052 (1.00)

a.

REFERENCE:

SP 635.1 pages 3&4 No Facility Objective

[3.4/3.6

268000A401 ..(KA's)

ANSWER: 053 (1.00)

b.

REFERENCE:

SBGT Text page 30 No facility Learning Objective

[3.0/2.9]

261000A303 ..(KA's)

ANSWER: 054 (1.00)

b.

REFERENCE:

Reactor Recirculation Operating Procedure OP 301 page 18 section 5.10
LO 20

[3.9/3.8]

202001A412 ..(KA's)

ANSWER: 055 (1.00)

d.

REFERENCE:

EOP 590.17 page 3 HVAC Text page 37. No Facility Objective [CHECK AT FACILITY FOR DISCREPANCY IN HVAC TEXT]

[3.8/3.8]

290001G007 ..(KA's)

ANSWER: 056 (1.00)

a.

REFERENCE:

ONP 531, Loss of Decay Heat Removal, page 3.

[3.7/3.7]

295021A104 ..(KA's)

ANSWER: 057 (1.00)

d.

REFERENCE:

EOP 580, Primary Containment Control, Step PCP/DWT 3 and 4.
EOP 560.7, Technical Basis for EOP 580, page 23.

[3.5/4.0]

295024K307 ..(KA's)

ANSWER: 058 (1.00)

a.

REFERENCE:

EOP 560.7, Technical Basis for EOP 580, page 26.

[3.1/3.3]

295029K205 ..(KA's)

ANSWER: 059 (1.00)

d. or a.

REFERENCE:

EOP 570, RPV Flooding, Steps FLD-11 to FLD-14.

[3.8/4.3]

295028G012 ..(KA's)

ANSWER: 060 (1.00)

c.

REFERENCE:

EOP 570, "RPV Control", step RP-1.

[4.4/4.4]

295031A106 ..(KA's)

ANSWER: 061 (1.00)

c.

REFERENCE:

EOP 560.3, Figure 4, "Correction for WR Level"
EOP 585, "Secondary Containment and Radioactive Release Control", Step RR-4.
EOP 570, Alternative Level Control, Step ALC-9.

[3.9/4.5]

295038G012 ..(KA's)

ANSWER: 062 (1.00)

a.

REFERENCE:

EOP 575, Steps FSL-1, FSP-1, DEP-1.
EOP 560.5, EOP 570 Technical Basis, page 18.

[4.3/4.7]

295012G011 ..(KA's)

ANSWER: 063 (1.00)

b.

REFERENCE:

EOP 580, Primary Containment Control, step TWL-11

[3.7/4.4]
295029G012 ..(KA's)

ANSWER: 064 (1.00)

a. or C

REFERENCE:

EOP 560.5, EOP 570 Technical Basis, page 18 and 19.

[4.6/4.6]

295031A201 ..(KA's)

ANSWER: 065 (1.00)

a.

REFERENCE:

EOP 560.7, EOP 580 Technical Basis, page 14.

[3.5/3.8]

295030K207 ..(KA's)

ANSWER: 066 (1.00)

c.

REFERENCE:

EOP 560.6, Technical Basis for EOP 575, page 36.

[4.0/4.2]

295037K106 ..(KA's)

ANSWER: 067 (1.00)

d.

REFERENCE:

EOP 560.7, EOP 580 Technical Basis, page 28.
EOP 590.8, Primary Containment Spray, page 3.

[3.9/3.9]

295024A117 ..(KA's)

ANSWER: 068 (1.00)

c. or a

REFERENCE:

EOP 580, Primary Containment Control, Step TWT-6.
EOP 575, Failure to Scram, Step FSP-1.

[3.8/4.5]

295026G012 ..(KA's)

ANSWER: 069 (1.00)

a.

REFERENCE:

EOP 585, Secondary Containment Control and Radioactive Release Control, Step SC-15.

[3.8/4.4]

295033G012 ..(KA's)

ANSWER: 070 (1.00)

c.

REFERENCE:

EOP 585, Secondary Containment and Radioactive Release Control, Step SC-13.

[3.7/3.9]

295032A105 ..(KA's)

ANSWER: 071 (1.00)

c.

REFERENCE:

EOP 570 Sh.1, RPV Control, Step RP-2.
Drywell Compressor System, figure 1 and page 1.

[4.4/4.4]

295025A103 ..(KA's)

ANSWER: 072 (1.00)

c.

REFERENCE:

EOP 580, Primary Containment Control, step TWT-3.

[3.4/3.8]
295026G007 ..(KA's)

ANSWER: 073 (1.00)

b.

REFERENCE:

EOP 575 Sheet 2, "Level Power Control", Step LP-2 and LP-3.

[3.8/4.1]

295037K306 ..(KA's)

ANSWER: 074 (1.00)

c.

REFERENCE:

EOP 560.6, "EOP 575 Technical Basis", page 64.

[4.1/4.5]

295037K303 ..(KA's)

ANSWER: 075 (1.00)

a.

REFERENCE:

EOP 580 "Primary Containment Control" Step PCP/DWT-12
EOP 570 Sheet 4, RPV Flooding, Table 1.

[4.6/4.8]

295031A204 ..(KA's)

ANSWER: 076 (1.00)

b.

REFERENCE:

EOP 575 Sheet 1 "Failure to Scram" Step FS-1.

[3.9/4.6]

295037G012 ..(KA's)

ANSWER: 077 (1.00)

d.

REFERENCE:

EOP 585 Secondary Containment And Radioactive Release Control Step SC-1

[4.0/3.9]

295034A103 ..(KA's)

ANSWER: 078 (1.00)

a.

REFERENCE:

ONP-506, Total Loss of Station 125 VDC Power, section 6.1.

[3.1/3.8]

295004G001 ..(KA's)

ANSWER: 079 (1.00)

a.

REFERENCE:

ONP-506B, Loss of Vital AC, section 6.C.

[3.4/3.5]

295003K204 ..(KA's)

ANSWER: 080 (1.00)

b.

REFERENCE:

ONP 519, Dropped Fuel Bundle, section 1.0.

[3.3/4.2]

295023G001 ..(KA's)

ANSWER: 081 (1.00)

a.

REFERENCE:

RBCCW Text, page 14.
Learning Objective 9.j

[3.3/3.4]

295018A101 ..(KA's)

ANSWER: 082 (1.00)

d.

REFERENCE:

ONP 526, Uncontrolled Power Oscillations, step 1.2.

[3.8/3.7]

295001G010 ..(KA's)

ANSWER: 083 (1.00)

b. or a.

REFERENCE:

ONP 502, Emergency Plant Shutdown, page 2.

[3.5/3.6]

295006A106 ..(KA's)

ANSWER: 084 (1.00)

d.

REFERENCE:

ONP 503B, Loss of all Station A.C. Power (LNP), page 1.

[4.4/4.4]

295003A103 ..(KA's)

ANSWER: 085 (1.00)

a.

REFERENCE:

OP 302, Control Rod Drive System, page 42.

[3.1/3.2]

295022A101 ..(KA's)

ANSWER: 086 (1.00)

b.

REFERENCE:

ONP 512, Rapid and Total Loss of Instrument Air, page 1.

[3.8/3.9]

295019K201 ..(KA's)

ANSWER: 087 (1.00)

d.

REFERENCE:

ONP 525A, Degraded fire in the Control Room or Cable Vault, page 1.

[4.2/4.3]

295016A107 ..(KA's)

ANSWER: 088 (1.00)

c.

REFERENCE:

Shutdown Cooling Student Text, page 11. Obj. 9.

[3.1/3.2]

295021K207 ..(KA's)

ANSWER: 089 (1.00)

c.

REFERENCE:

Technical Specifications 3.7.A.3
ONP 531, Loss of Decay Heat Removal,

[3.5/3.6]

295021A201 ..(KA's)

ANSWER: 090 (1.00)

d.

REFERENCE:

Primary Containment Text, page 47.

[3.4/3.5]

295010A101 ..(KA's)

ANSWER: 091 (1.00)

d.

REFERENCE:

OP 337, Autopressure Relief System, section 4.2.

[3.9/4.1]

295007A104 ..(KA's)

ANSWER: 092 (1.00)

b.

REFERENCE:

OP 338, Diesel Generator, page 4.

[3.7/3.6]

295003G005 ..(KA's)

ANSWER: 093 (1.00)

b.

REFERENCE:

ONP 502, Emergency Plant Shutdown, step 1.3.

[3.8/3.6]

295005G010 ..(KA's)

ANSWER: 094 (1.00)

a.

REFERENCE:

ONP 506, Total Loss of Station 125 VDC Power, page 1.

[3.2/3.4]

295004G010 ..(KA's)

ANSWER: 095 (1.00)

d.

REFERENCE:

ONP-506B, Loss of Vital AC, section 1.3.1.

[3.9/4.1]

295003G010 ..(KA's)

ANSWER: 096 (1.00)

c.

REFERENCE:

ONP 523, Loss of Feedwater Heating, page 1.

[3.8/3.6]

295014G010 ..(KA's)

ANSWER: 097 (1.00)

c.

REFERENCE:

EPIP form 4400-1.

[2.9/4.2]

295005G002 ..(KA's)

ANSWER: 098 (1.00)

c.

REFERENCE:

EPIP Form 4400-1.

[3.1/4.7]

295038G002 ..(KA's)

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

ANSWER KEY

MASTER COPY

MULTIPLE CHOICE

001 b

002 MATCHING

a 1

b 1

c 4

~~d 5~~ deleted TAW 1/24/94

MULTIPLE CHOICE

003 a

004 c

005 a

006 b

007 d

008 c

009 b

010 c

011 c

012 d

013 a

014 d

015 b

016 c

017 d

018 d

019 d

~~020 d~~ deleted TAW 1/24/94

021 b

022 c

023 d

024 d

025 b

026 d

027 b

028 a

029 a

030 c

031 c

032 c

033 b

034 b

035 a

036 d

037 c

MASTER COPY

ANSWER KEY

038	MATCHING	056	a
	a 2	057	d
	b 6	058	a
	c 5	059	d or a
	d 2	060	c
	MULTIPLE CHOICE	061	c
039	b	062	a
040	d	063	b
041	c	064	a or c
042	c	065	a
043	b	066	c
044	a	067	d
045	a	068	c or a
046	a	069	a
047	d	070	c
048	a	071	c
049	b	072	c
050	b	073	b
051	c	074	c
052	a	075	a
053	b	076	b
054	b	077	d
055	d	078	a

ANSWER KEY

079 a

MULTIPLE CHOICE

080 b

081 a

082 d

083 b or a

084 d

085 a

086 b

087 d

088 c

089 c

090 d

091 d

092 b

093 b

094 a

095 d

096 c

097 c

098 c

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

Attachment 3

NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 665-5000

January 17, 1994
MP-94-30

Mr. Thomas Martin
Branch Chief
U.S. Nuclear Regulatory Commission, Region 1
475 Allendale Road
King of Prussia, PA 19406

Dear Mr. Martin:

Attached is a list of comments regarding the written examination administered to Millstone Unit No. 1 Initial License candidates on January 11, 1994.

These comments were the results of a review of the examination conducted by members of the Millstone Unit No. 1 training staff. References are provided to substantiate these comments.

Please contact Mr. Chris Tabone, Supervisor, Operator Training, Millstone Unit No. 1, with any questions concerning our comments.

Sincerely,

Donald B. Miller, Jr.
Senior Vice President
Millstone Station

Attachment 1

SRO Exam

Question 2 - The question asks for the minimum dose rates for the area to be posted as listed. It appears that either a typographical error was made and answer No. 5 should have been 500 REM/hr or the answer key should have listed answer No. 6 as the correct response because extremely high rad area is defined as an area > 500 R/hr.

Reference: Radiation Worker Training Manual and SHP 4906, 3.14

Recommendation: Credit given for answer No. 6 only and the exam bank corrected.

Question 20 - The question asks for the Technical Specification actions based on two inoperable accumulators in a nine-rod array. Millstone Unit No. 1 Technical Specifications require the rod with an inoperable accumulator be inserted to the "full-in" position and have its directional control valves electrically disarmed. This requirement makes none of the answers correct.

Reference: Millstone Unit No. 1 Technical Specifications Section 3.3.D

Recommendation: Delete question from exam and correct answer "d" to include electrical disarment.

Question 59 - The question asks for the requirements to allow injection to be terminated to restore level indication during RPV flooding. Answers "a" and "d" are correct responses based on the minimum core flooding interval (Table 2).

Reference: EOP 570 FLD-13 and 14

Recommendation: Credit given for answer "a" and "d". Correct exam bank so that only one correct answer exists.

Question 64 - The question asks for the condition of narrow range ATWS indications due to high drywell temperature. Millstone Unit No. 1 does not use drywell bulk temperature for determining validity of these level instruments. Determining minimum temperatures would require readings from TR 1602-6 pts. 1 and 9. Also, the attached memo was presented to all operators to ensure accurate interpretation of caution #1. This memo corrects misleading information from the EOP bases text. Based on this information, answer "a" is incorrect. Interpretation that RPV pressure was rapidly reduced to 450 psig leads to answer "c" being correct based on caution 2 which states that this instrument should not be used during rapid depressurization < 500 psig.

Reference: Memo (EOP issues covered by training). EOP bases text, pages 19, 20 and 22

Recommendation: Delete question from exam or give credit for answer "c" only. Correct exam bank to use the correct temperature points and include specific information on RPV pressure response.

Question 83 - The question asks for the effect of operation with only one half of RPS reset. With the correct answer being "b", the question assumes the "Scram Discharge Volume Isolation Test Switch" is returned to "Normal" (ONP 502, step 2.13.5). One SRO asked the proctor if resetting the scram (ONP 502, step 2.13.3) was the last action performed and he was told "Yes". If this was true the correct answer would be "a" because at that point the operator would NOT have opened the scram discharge volume vent and drain valves.

Reference: ONP 502, Emergency Plant Shutdown

Recommendation: Credit given for answer "a" and "b". Correct exam bank to address final position of control switches.

<u>RO Exam</u>

Question 2 - See SRO exam, question 2.

Question 59 - The question asks which refueling interlock has generated the rod block. The refueling interlock on refueling bridge position at one time used power from the refueling bridge console to supply the logic. This logic was modified to use an actual limit switch on the refueling bridge trolley to prevent bridge console power interruptions from giving the control room unnecessary rod blocks (see attached wiring diagrams). The system text is under revision at the present time and the operators have been trained on this modification. Therefore, answer "d" is incorrect and answer "a" is correct if it is assumed that the rod select power switch was turned "off" then returned to "on" (see lesson text).

Reference: Refueling Text, Control Wiring Diagrams, 520 and 525

Recommendation: Credit given for answer "a" only. Correct exam bank to include present modification

Question 64 - See SRO exam, question 64.

Question 68 - The question asks for the action taken on high torus water temperature. In accordance with Millstone Unit No. 1 EOPs and EOP User's Guidelines, as TWT approaches the HCTL, RPV pressure should be maintained below the

HCTL. However, if you find yourself in the unsafe region of the HCTL curve, emergency depressurization is required. This question forces the operator to interpolate (allowed in User's Guidelines) between to TWL lines. The point picked for this question is very close, if not in the unsafe region of the HCTL curve. Depending on exactly where you place yourself on the HCTL curve could make answer "a" or "c" correct. In both cases, depressurization would eventually occur either by gradually reducing pressure or performing a emergency depressurization.

References: EOP 580 and HCTL curve.

Recommendation: Credit given for answer "a" and "c". Correct exam bank to clearly place plant in safe or unsafe region.

Question 83 - See SRO exam, question 83.

ATTACHMENT 4

NRC RESOLUTION OF FACILITY COMMENTS

Question #	
RO/SRO 2	Comment partially accepted. Because neither distractor 5 nor 6 is the correct posting for an Extremely High Radiation Area, Part d. of this question was deleted reducing the total exam point value by 0.5.
SRO 20	Comment accepted. Because there was no totally correct answer, this question was deleted from the SRO exam reducing the total point value by 1.0.
RO 59	Comment accepted. Based on the information provided by the facility, it was determined that item a. is the correct answer and distractor d. is incorrect. Credit was given for item a. only.
SRO 59	Comment accepted. Credit was given for both items a. and d. The confusion caused by this question was a direct result of the poor reproduction copy of the EOP flowcharts submitted by the facility for exam preparation.
RO/SRO 64	Comment partially accepted. Based on the information provided by the facility, it was determined that answer a. was not clearly incorrect. Credit was given for both items a. and c..
RO/SRO 68	Comment accepted. Section 6.1.5.1 of EOP-550.2, "Unit One Emergency Operating Procedures User's Guide," discusses interpolation of values on the X and Y axes of figures in the EOPs. Interpolation between curves within a figure is not addressed. This question was developed based on the assumption that interpolation between curves was not allowed; therefore, the more conservative torus water level curve would have been used. Because interpolation between curves is not precluded by EOP-550.2, item a. is a correct answer. Credit was given for both items a. and c..
RO/SRO 83	Comment accepted. The proctor's record of the applicant's question differed from the question described in the facility comment. The proctor's record indicated that the applicant asked "has the operator done everything right and the scram just will not reset"? He was told "yes, the operator has done nothing additional that would cause it not to reset." Based on this, item a. would not be a correct answer.

However, because the question did not specifically address the final position of the control switches, item a. was accepted as a correct answer. Credit was given for both items a. and b.

Note: RO question #18 was modified prior to administration of the examination. The correct answer was changed to item d.

ATTACHMENT 5

SIMULATION FACILITY REPORT

Facility License: DPR-21

Facility Docket No: 50-245

Operating Test Preparation and Administration: February 1-3, 1994

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information that may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating tests, the following items were observed:

ITEM	DESCRIPTION
Power Oscillations	There is no malfunction or other method available to model reactor power instabilities for high power/low flow conditions. (The training department indicated that a malfunction is being developed to model power oscillations.)
Control Rod Drift	There is no malfunction available to simulate a control rod drifting out of the core. Instructor overrides were used to stick the rod movement switches causing the selected rod to move out continuously. These overrides did not cause the same results during examination administration that they did during validation. They also did not provide a plausible failure mode.