U. S. NUCLEAR REGULATORY COMMISSION
OPERATOR LICENSING EXAMINATION REPORT

Examination Report No.:	91-25 (OL)
Facility Docket No.:	50-354
Facility License No.:	NPF-57
Licensee:	Public Service Electric and Gas Company Post Office Box 236 Hancocks Bridge, New Jersey 08038
Facility:	Hope Creek
Examination Dates:	December 9 -13, 1991
Examiners:	M. Parrish, Examiner, INEL D. Florek, Senior Operations Engineer

Chief Examiner: Todd Fish 2/10/92
T. Fish, Senior Operations Engineer Date

Approved by: 2/13/92
Richard J. Conte, Chief Date
BWR Section, Operations Branch, DRS

EXAMINATION SUMMARY

Initial examinations were administered to seven reactor operator candidates. Four of seven candidates passed all portions of the exam, while two candidates failed the simulator portion and one candidate failed the written portion. Both simulator failures were the result of the candidates' not executing the applicable procedures in response to a failure of the reactor protection system. Performance on the written exam was marginal: five candidates including the failure, scored below 83%. Section 3 has the details.

Deficiencies were noted in several support procedures to the Emergency Operating Procedures. The facility has proposed procedure revisions which should correct the identified deficiencies. Section 4 has the details.

Job Performance Measures used during walk-through portions of the exams did not always meet the guidance established in the examiner standards. Examples include imprecise task standards, improper cues, and mis-identified critical tasks. Section 5 has the details.

Unresolved Item 354/90-18-01 was reviewed. This Item addresses the technical adequacy of the justification for a deviation from the Emergency Operating Procedure (EOP) Guidelines established for entry into Reactor Pressure Vessel (RPV) Control. Based on the facility's documentation, the justification does not appear to be adequate in that the facility's RPV Control EOP procedure does not provide equivalency to the Guidelines. Therefore, this item remains open. Section 6 has additional details.

A senior reactor operator passed a requalification retake examination which was administered during the same week as the initial exam.

DETAILS

1.0 Introduction

The NRC examiners administered initial examinations to seven Reactor Operator (RO) candidates. The examinations were administered in accordance with NUREG 1021, Examiner Standards, Revision 6. The results of the examination are summarized below:

	Pass/Fail
Written	6/1
Operating	5/2
Overall	4/3

A senior reactor operator was administered a requalification retake examination for the walk-through (Job Performance Measure) portion only. He passed the retake exam.

2.0 Preexamination Activities

The facility reviewed the examination in the Regional Office on December 4, 1991. The review team included Hope Creek's Operations Training supervisor, two Senior Reactor Operator (SRO)-licensed staff personnel, and an SRO-certified training instructor. On December 9, 1991, the simulator scenarios used for the operating section of the exam were validated on the facility's simulator. Facility staff who were involved with these reviews signed security agreements to ensure that the examination was not compromised.

3.0 Examination-Related Findings and Conclusions

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

3.1 Written Section

The following subjects were missed by at least five of the candidates, indicating a weak performance in these areas.

- Knowledge of what items on the Nuclear Control Operator's (NCO's) relief checklist must be completed prior to accepting the watch.

- Knowledge of which valve must be closed during individual rod testing

- Knowledge of what plant conditions are addressed in technical specifications in regard to reactor building pressure.

- Knowledge of the immediate operator actions associated with a given set of refuel floor alarms.

3.2 Walk-through Section

No generic strengths or weaknesses were noted with respect to the candidates' performance. However, the Job Performance Measures themselves did not always meet the guidelines established in the examiner standards. This area is discussed in more detail in section 5.

3.3 Simulator Section

Strengths

- Familiarity with control boards and systems operation.

Weaknesses

- Ability to recognize failure-to-scram (ATWS) situations.

- Use of Alarm Response Procedures (ARPs).

- Recognition of entry conditions into EOPs.

3.4 Conclusions

Regarding the written examination, it is noted that in addition to the candidate who failed, four other candidates scored between 80 and 83%. The high score was an 89.8%. These results indicate marginal performance and appear to represent real knowledge and ability weaknesses.

4.0 Deficiencies in Procedures

4.1 During the course of the examination, the NRC examiners identified deficiencies with the following facility support procedures to Emergency Operating Procedures (EOPs).

OP-EO.ZZ-301, Bypassing MSIV Isolation Interlocks, Revision 3

This procedure described items of the EOP implementation kit that were not contained in the kit as a result of a recent modification on the method to insert the jumpers. The kit was correct but the procedure was not revised to reflect the modification. In addition, step 5.1.11 directed operators to increase the turbine pressure setpoint above the reactor pressure. The procedure provided no direction to either reduce the pressure setpoint or use the bypass valve opening jack to allow the bypass valves to open to control reactor pressure. A proposed revision to the procedure addressed the required changes.

<u>OP-EO. ZZ-316, Suppression Chamber Level Reduction Using HPCI, Revision 4</u>

This procedure also required revision to reflect the recent plant modification on the method to insert the required jumpers. During the examination, the facility was conducting validation activities on a procedure revision which addressed the modification.

<u>OP-SO.BC-0001, Residual Heat Removal (RHR) System Operation, Revision 11</u>

Section 5.7 of the procedure contains the steps for placing RHR in containment/suppression pool spray. The procedure directs establishing containment spray prior to opening the valves for suppression chamber spray. This is in conflict with EOP OP-EO.ZZ-102, which directs suppression chamber spray prior to drywell spray. In addition, the procedure directs throttling of a valve (HVF027) which cannot be throttled. The facility provided information that they had identified similar concerns on August 7, 1991 but, as of December 13, 1991, had not yet issued a procedure revision. On December 19, 1991, the facility provided a modified procedure to address the specific concerns identified above.

<u>OP-SO.BF-0002, Individual Control Rod Drive (CRD) Hydraulic Control Unit (HCU) Operation</u>

An NRC-generated JPM had the following initial conditions: Rods had failed to insert, the SRO had entered step RC/Q-16 of EOP OP-EO.ZZ-101, "Reactor/Pressure Vessel Control", and had directed the RO to manually insert CRDs per SO.BF-0002. The candidates were then expected to reference SO.BF-0002 and simulate venting the over-piston area of a CRD, thereby allowing the CRD to insert. However, when they did reference the procedure, it was not clear to them which of the thirteen sections (5.1 through 5.13) of SO.BF-0002 provided the appropriate directions. This apparently was because the key words in step RC/Q-16 did not clearly correspond to any of the titles listed in the table of contents of SO.BF-0002: RC/Q-16 says to "Vent control rod over-piston volumes..." while the applicable section of SO.BF-0002, 5.12, is titled, "Directing the Effluent from the Withdraw Line Vent to a Radwaste

Drain". Once the candidates were shown this, they readily performed the JPM. Consequently, the exam team concluded that there was a deficiency in the transition between step RC/Q-16 of EOP OP-EO.ZZ-101 and SO.BF-0002 and not in the candidates' actual execution of the procedure. The facility agreed with this assessment and stated that they would revise the affected procedures as necessary to better connect step RC/Q-16 to section 5.12.

4.2 During the course of the examination, the NRC examiners identified a deficiency with the following facility operating procedure.

OP-SO-SP-001 Radiation Monitoring System, Revision 0

Section 5.4 of the procedure was designed to perform a check-source test. The sequence of the steps was questioned by the examiner. Upon further licensee investigation, the licensee representatives determined that the procedure, as written, would not perform a valid check-source test. Normally, the check-source is performed automatically by the radiation monitoring system. The facility licensee will issue a procedure revision to correct the procedure.

4.3 Conclusions

Several procedures did not reflect recent plant modifications, one contradicted steps of an EOP, and another needed enhancements so that the operators would readily know which section to refer to for instructions on how to operate the Control Rod Drive system in a seldom-used configuration. These deficiencies are being provided as feedback to the facility's procedure control program. It is noted that the facility was aware of these deficiencies and has been taking appropriate corrective actions.

5.0 Job Performance Measures (JPMs)

5.1 The requalification retake examination as well as the initial examination used JPMs from the facility JPM bank. Based on the sample of JPMs used for the examinations, a number of the JPMs conflicted with the guidance provided in the examiner standards.

1. The ES Form ES-603-1 "JPM Quality Checklist" in item 5 discusses performance standards. The standards for performance are not always sufficiently detailed to assess satisfactory performance of the step. As an example, JPM BB-003 "Jet Pump Operability," indicates in step 5.1.4.3.a to record recirculation pump A flow. The JPM standard does not indicate where the information is obtained (indicator and panel number) and what specific value would be entered to satisfy the critical

step. The JPM standard indicates flow must be within $\pm$ 1%, but does not define the base value. In JPM SF-004 "Bypass Rod RMCS," step 5.5.3, the operator is to set the binary code on the rod to be bypassed. The JPM standard does not indicate what toggle switches are to be operated and in what position each toggle switch is to be placed in. The JPMs sometimes omit actions and standards to accomplish the task. In JPM, BF-006 "Isolate a CRD HCU During Reactor Operation," tools and a ladder are required to perform the task, but they are not mentioned in the JPM steps or JPM standards. The JPM steps are not always written based on the initiating cue, but simply are the steps as written in the referenced procedure. An example is JPM BC-008 "Place the RHR System in Containment Spray", the task is to place the B loop in service, the JPM step is written as "Close HVF017A(B)..." rather than "Close HVF017B..."

2. The ES Form ES-603-1 "JPM Quality Checklist" in item 5 discusses system response cues. The cues for some of the JPMs that are simulated in the plant are not always sufficiently detailed. As an example, for JPM EG-001 "Place SACS Loop B in Service from the RSP to Supply TACS," when the operator depresses SACS pump start pushbutton, the cue provided to the operator was that the flow indicator indicates the pump is running normally. The cue should have been that the red light illuminates, the green light deenergizes, and flow indicator FI-2549 indicates a value of X gpm. The operator should determine that the pump is running normally or candidate. Similarly, when discharging a HCU accumulator in JPM BF-006 "Isolate a CRD HCU During Reactor Operation," realistic cues should be added such as gas is heard escaping from the throttled valve and that accumulator gas pressure is decreasing.

3. ES-603 paragraph D.2.a(3) describes JPMs that are time critical in nature and important to the mitigation of significant plant transients. The time validation for these types of JPMs, based on the initial cues and initiating cues, is not always realistic. An example is JPM SN-002, "ADS Operation", to which the facility assigned a completion time of ten minutes. The JPM initiating cue is that the EOPs direct emergency depressurization and that the NSS orders manual initiation of ADS. It is unrealistic to allow 10 minutes to accomplish this task. In reality, and based on observations during the simulator portion of the examination when this evolution was actually performed, this task would probably be performed in less than one minute and done without reference in hand.

4. The JPMs list critical steps that have already been accomplished as part of the initiating cue. As an example, in JPM EG-001 "Place SACS Loop B in Service from the RSP to Supply TACS," the initiating cue states that all emergency transfer switches have been placed in emergency, yet a critical step of the JPM is to verify the transfer switches are in emergency.

5. The facility requalification JPM briefing sheet contains coaching information inappropriate to present at the beginning of the JPM portion of the examination and which goes beyond the scope of the briefing contained in the "Briefing Checklist - System Walk-Through" of ES-603. This includes instructions to REQUIRE the operator to locate the applicable procedures, verify procedural prerequisites, precautions and limitations for each JPM whether it is necessary or not to accomplish the assigned task. In addition, it is inappropriate to indicate in the JPM briefing that the operator is not expected to do any task from memory. Some of the JPM tasks are appropriate to do without reference to procedures such as immediate actions of emergency and other procedures appropriate to the facility.

In summary, based on the sample of JPMs used for the initial examination and requalification retake examination, which appears to be representative of the facility JPM examination bank, the examiner concluded that the facility JPMs do not fully satisfy the NRC expectations for JPMs as discussed in ES-603.

6.0 <u>Licensee Actions on Previously Identified Findings</u>

OPEN Unresolved Item (354/90-18-01). This item relates to the technical adequacy of the justification for the deviation from the BWROG Emergency Procedure Guidelines (EPGs) for the reactor water level entry condition into the Hope Creek Emergency Operating Procedures. The BWROG EPGs require entry into RPV Control if reactor water level decreases below the scram setpoint of +12.5 inches. The Hope Creek entry into RPV Control is at -38 inches. The licensee provided documentation, dated May 27, 1991, "ATS Open Item 354/90-18-01, Revised EPG Deviation Justification." Based on the licensee's documentation, the justification is not adequate in that the licensee procedures do not provide equivalency to the EPG for the condition that reactor water level cannot be maintained above +12.5 inches and actual level is above -38 inches. For the condition above, the accident mitigation strategy in the licensee EOPs depart from that specified in the BWROG EPGs. The licensee indicated that they would assess the EOPs and either revised the EOPs or provide additional justification. Based on the above, this item will remain open.

7.0 Exit Meeting

An exit meeting was conducted on December 13, 1991, following the administration of the exams. Exit attendees are listed in Attachment 5. The facility presented their comments on the written exam questions (Attachment 2). Generic findings regarding the candidates' performance on the operating portions of the exam were discussed, along with findings related to JPM's and EOP support procedures.

Attachment 1

MASTER

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Nuclear Regulatory Commission
Operator Licensing
Examination

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Official Use Only category on
date of examination.

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A N S W E R   K E Y

MULTIPLE CHOICE

001	d	023	c
002	a	024	b
003	a	025	d
004	b	026	d
005	b	027	d
006	a	028	d
007	a	029	b+a
008	c	030	d
009	b	031	c
010	c	032	a
011	c+d	033	c
012	d	034	b
013	c	035	a
014	a	036	d
015	a	037	b
016	d	038	a
017	b	039	a
018	a	040	a
019	c	041	b
020	a	042	c
021	b	043	d
022	c	044	b
		045	b

A N S W E R K E Y

046 a
047 b
048 b
049 c
050 a
051 d
052 c
053 d
054 d
055 c
056 b
057 b
058 d
059 c
060 d
061 d
062 b
063 a
064 c
065 d
066 a
067 a
068 d
069 b & C
070 c
071 b
072 c
073 b
074 c
075 d
076 b
077 b
078 c
079 d
080 b
081 a & C
082 a
083 a
084 a
085 d
086 c
087 d
~~088 b~~ Deleted from exam
089 a
090 d
091 c

A N S W E R K E Y

092 c

093 c

094 d

095 96 hours available

096 ~~161~~ 146 ± 3 psig

097 1143 +0 -5 kw additional load

098 14 LPRM inputs

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

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

CANDIDATE'S NAME: ____________________

FACILITY: Hope Creek

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 91/12/09

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	%	
98.00	________ 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 a b c d ____
003 a b c d ____
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042 a b c d ____
043 a b c d ____
044 a b c d ____
045 a b c d ____

ANSWER SHEET

Multiple Choice (Circle or X your choice)

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

046	a	b	c	d	____
047	a	b	c	d	____
048	a	b	c	d	____
049	a	b	c	d	____
050	a	b	c	d	____
051	a	b	c	d	____
052	a	b	c	d	____
053	a	b	c	d	____
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076	a	b	c	d	____
077	a	b	c	d	____
078	a	b	c	d	____
079	a	b	c	d	____
080	a	b	c	d	____
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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	____
089	a	b	c	d	____
090	a	b	c	d	____
091	a	b	c	d	____

ANSWER SHEET

Multiple Choice (Circle or X your choice)

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

092 a b c d ___

093 a b c d ___

094 a b c d ___

095 ____________________hours available

096 ____________________psig

097 ____________________Kw additional load

098 ____________________LPRM inputs

(********** 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 means an automatic denial of your application and could result in more severe penalties.

2. After the examination has been completed, you must 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. This must be done after you complete the examination.

3. Restroom trips are to be limited and only one applicant at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

4. Use black ink or dark pencil ONLY to faci`itate legible reproductions.

5. Print your name in the blank provided in the upper right-hand corner of the examination cover sheet and each answer sheet.

6. Mark your answers on the answer sheet provided. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.

7. Before you turn in your examination, consecutively number each answer sheet.

8. Use abbreviations only if they are commonly used in facility literature. Avoid using symbols such as < or > signs to avoid a simple transposition error resulting in an incorrect answer. Write it out.

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

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

11. When turning in your examination, assemble the completed examination with examination questions, examination aids and answer sheets. In addition, turn in all scrap paper.

12. To pass the examination, you must achieve a grade of 80% or greater.

13. There is a time limit of four (4) hours for completion of the examination.

4. When you are done and have turned in your examination, leave the examination area (EXAMINER WILL DEFINE THE AREA). If you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION: 001 (1.00)

The CRD hydraulic system is operating normally with the reactor at rated temperature and pressure. SELECT the effect of a failed high signal from the venturi flow element to the operating CRD flow control station.

a. The operating CRD pump will trip due to low suction pressure.

b. The CRD pressure control valve will open to control CRD system pressure.

c. CRD accumulator pressures will decrease.

d. CRDM temperatures will increase.

QUESTION: 002 (1.00)

While at 60% power during a valid withdrawal of control rod 42-23, control rod 38-15 is observed to drift into the core. SELECT the response of the reactor manual control system.

a. Rod drift alarm but NO data fault alarm and NO rod block alarm.

b. Rod drift alarm and a data fault alarm but NO rod block alarm.

c. Rod drift alarm and a rod block alarm but NO data fault alarm.

d. Rod drift alarm and an activity control disagree alarm.

QUESTION: 003 (1.00)

During a recirculation pump motor start sequence the jogging circuit opened the recirculation pump discharge valve HV-F031 to the full open position in 90 seconds. SELECT the statement that describes the expected response of the recirculation flow control system.

a. Drive motor breaker will open resulting in a trip of both the recirculation pump and MG set.

b. Field breaker will open resulting in a coastdown of the recirculation pump, but the MG set is running.

c. Pump speed will be reduced to 30% speed due to deenergization of the excitation transfer timer contact when the discharge valve was not full open at 80 seconds.

d. Recirculation pump is running since 90 seconds is the expected opening time for the jogging circuit.

QUESTION: 004 (1.00)

The plant is operating at 85% power with recirculation pump A speed control station in MAN and recirculation pump B speed control station in AUTO. Both recirculation pump speeds are initially at 90%. SELECT the response of the recirculation pumps if one primary condensate pump trips and the transient has stabilized.

a. Pump A speed is 90% - Pump B speed is 90%.

b. Pump A speed is 30% - Pump B speed is 30%.

c. Pump A speed is 90% - Pump B speed is 30%.

d. Pump A speed is 30% - Pump B speed is 90%.

QUESTION: 005 (1.00)

During an inservice test, RHR pump A is operating at 10,200 gpm through HV-F024A "A RHR pump test return valve." During the test, a valid LPCI initiation signal is generated concurrent with loss of offsite power. HV-F024A will not close and diesel generator C is not able to energize 4160V bus 10A403 . SELECT ALL the LPCI injection valves (F017) that will open when reactor pressure decreases to less than 450 psig.

a. A and B and C and D.

b. A and B and D.

c. B and D.

d. A and C and D.

QUESTION: 006 (1.00)

During a loss of all feedwater transient, the operator MANUALLY armed and depressed the HPCI initiation pushbutton. SELECT the effect of the operator actions on the HPCI system.

a. HPCI will automatically trip at +54 inches and will automatically reinitiate at -38 inches.

b. HPCI will automatically trip at +54 inches and require operator action to restart.

c. HPCI will require a manual trip at +54 inches and will automatically reinitiate at -38 inches.

d. HPCI will require a manual trip at +54 inches and require operator action to restart.

QUESTION: 007 (1.00)

SELECT the method that will immediately start the "A" SLCS pump.

a. Place the "TEST SWITCH PUMP A" to the test position at panel 10C011.

b. Place the key lock switch to the "ON" position on the pump A control bezel on panel 10C651.

c. Depress the "AP208 START" (red) backlit pushbutton on the pump A control bezel on panel 10C651.

d. Depress the "MANUAL INITIATION PERMISSIVE" and "MANUAL INITIATION" pushbuttons for both logic trains within Channel A of RRCS.

QUESTION: 008 (1.00)

While the reactor is critical, RPS bus A is being powered from the backup power supply MCC 10B491 through transformer 1AX432. SELECT the plant response if while restoring RPS bus A to the normal supply, the RPS Bus Transfer Switch is taken from "ALT A" through "NORMAL" to "ALT B".

a. 1/2 scram on RPS A - RPS B remains energized.

b. 1/2 scram on RPS B - RPS A remains energized.

c. Full reactor scram.

d. RPS A and RPS B remain energized.

QUESTION: 009 (1.00)

While operating at 100% power a gradual increase in recirculation flow to 92% flow occurs. The mode switch is taken to SHUTDOWN and no control rod movement is observed. If recirculation flow is maintained at 92% and the EHC can accommodate the power increase, SELECT the scram setpoint that is exceeded when the mode switch is taken to SHUTDOWN.

a. Fixed Neutron Flux-Upscale.

b. Neutron Flux-Upscale, Setdown.

c. Flow Biased Simulated Thermal Power-Upscale.

d. SRM High Count Rate.

QUESTION: 010 (1.00)

SELECT the system(s) supplied by ECCS jockey pump CP-228.

a. HPCI.

b. RCIC.

c. RHR pumps A and C and Core spray pumps A and C.

d. RHR pumps B and D and Core Spray pumps B and D.

QUESTION: 011 (1.00)

Core Spray Logic Channel C is inadvertently manually initiated with the reactor at 400 psig. SELECT the action that will **NOT** occur.

a. Emergency diesel generator C auto starts.

b. Non-1E loads on the respective vital bus are load shed.

c. HV-F005A "Core spray loop injection valve" opens.

d. HV-F031A "Core spray pump minimum flow valve" opens when the pump starts and closes when loop flow is > 775 gpm.

QUESTION: 012 (1.00)

IRM A is reading 81 on range 4 during a plant startup. The operator places the IRM A Range Switch to range 3. SELECT the system response.

a. IRM A will indicate 8.1 on range 3.

b. IRM A will indicate 9 on range 3.

c. A rod block, but no reactor 1/2 scram signal, will be generated.

d. A rod block and a reactor 1/2 scram signal will be generated.

QUESTION: 013 (1.00)

During a refueling SELECT the SRMs that must be operable to perform core alterations for fuel assembly 30-41. Utilize the enclosed Figure 2 Nuclear Instrumentation Detector Locations.

a. SRM A and SRM C.

b. SRM B and SRM D.

c. SRM B and SRM C.

d. ARM A and SRM D.

QUESTION: 014 (1.00)

Given the following initial conditions:

The reactor is operating at 100%.
The value of the A recirculation flow signal is 99%
The value of the B recirculation flow signal is 100%
Recirculation flow unit C bypassed on control room panel 10C651.
The value of the D recirculation flow signal is 101%

A component in recirculation flow unit B fails resulting in a recirculation B flow signal of 110%. SELECT the response of the APRM.

a. Upscale trip of the B flow unit and comparator trips of the A and B flow unit. No effect on the D flow unit. No scram signals generated.

b. Upscale trip of the B flow unit and comparator trips of the A, B, and D flow units. No scram signals generated

c. Inop trips of the A and C flow units and Inop scram signals on APRMs A, C, and E. No scram signals generated on APRMs B, D, and F.

d. Inop Scram signals on APRMs A, C, and E and Upscale thermal scram on APRMs B, D and F.

QUESTION: 015 (1.00)

While at rated temperature and pressure conditions in the reactor vessel, drywell temperature increases. SELECT the response of the narrow range level instrument.

a. Decreased reference leg density will cause an increase in indicated level.

b. Increased reference leg density will cause a decrease in indicated level.

c. Because both reference and variable leg densities increase, there will be no change in indicated level.

d. Because narrow range level instruments are density compensated there is no change in indicated level.

QUESTION: 016 (1.00)

Following an automatic initiation, RCIC speed is observed to be zero, the RCIC turbine trip throttle valve (4282) is open and the steam supply valve (F045) is closed. SELECT the condition that caused the RCIC response.

a. RCIC Pump suction low pressure.

b. Low reactor pressure without high drywell pressure.

c. Low reactor pressure and high drywell pressure.

d. High reactor water level.

QUESTION: 017 (1.00)

The reactor has scrammed with a loss of offsite power. The following conditions exist:

- Only EDG C and its associated ECCS pumps are running.
- Reactor water level has just reached -129 inches and is decreasing 1 inch/min.
- Reactor pressure is 550 psig and decreasing 10 psi/min.
- Reactor power is 0%.
- Drywell pressure has just reached 1.08 psig and increasing 0.30 psi/min.
- ADS has NOT been inhibited.

With the preceding conditions, SELECT the response of the ADS system.

a. The ADS valves will open in 105 seconds.

b. The ADS valves will open in 225 seconds.

c. The ADS valves will open in 300 seconds.

d. The ADS valves will not open automatically.

QUESTION: 018 (1.00)

During operation at 100% power, chilled water flow is lost to the drywell unit coolers. SELECT the description of the LONG TERM effect this will have on the primary containment parameters.

a. Drywell/torus differential pressure increases
Indicated torus water level increases

b. Drywell/torus differential pressure increases
Indicated torus water level decreases

c. Drywell/torus differential pressure decreases
Indicated torus water level decreases

d. Drywell/torus differential pressure decreases
Indicated torus water level increases

QUESTION: 019 (1.00)

SELECT the component failure mode such that if it occurred during a DBA LOCA, then drywell pressure could exceed the containment design pressure of 62 psig.

a. Reactor building to torus vacuum breakers failed open.

b. Reactor building to torus vacuum breakers failed closed.

c. Torus to drywell vacuum breakers failed open.

d. Torus to drywell vacuum breakers failed closed.

QUESTION: 020 (1.00)

SELECT the action(s) that will ONLY close all the NSSSS inboard isolation valves other than MSIVs.

a. "A" NSSSS logic manual initiation collar is armed and pushbutton is depressed.

b. "B" NSSSS logic manual initiation collar is armed and pushbutton is depressed.

c. "A" and "C" NSSSS logic channels are deenergized.

d. "A" and "D" NSSSS logic channels are deenergized.

QUESTION: 021 (1.00)

While operating RHR in shutdown cooling, reactor water level transmitter for NSSSS channel A fails downscale. SELECT the response of the RHR shutdown cooling supply valves, HV-F008 and HV-F009.

a. Both RHR shutdown cooling supply valves will automatically close.

b. Neither RHR shutdown cooling supply valve will change position automatically.

c. One of the RHR shutdown cooling supply valves automatically close and the second RHR shutdown cooling supply valve will close if Level 3 is sensed in the "B" NSSSS logic.

d. One of the RHR shutdown cooling supply valves automatically close and the second RHR shutdown cooling supply valve will close if Level 3 is sensed in the "C" NSSSS logic.

QUESTION: 022 (1.00)

While operating at 100% power at end of life a full MSIV closure transient has occurred which has armed "SRV LO-LO SET," scrammed the reactor, and isolated Primary Containment Instrument Gas. IN THE LONG TERM, SELECT the highest pressure at which the SRVs will automatically control pressure. ASSUME no operator action to control pressure and ignore the initial SRV opening pressure.

a. 1017 psig.

b. 1047 psig.

c. 1108 psig.

d. 1130 psig.

QUESTION: 023 (1.00)

GIVEN that:

- Load set = 1170 MWe
- Load limit = 100%
- Max. comb. limit = 130%
- Throttle pres. = 950 psig
- "A" pressure reg. = 920 psig

The "A" EHC pressure regulator fails upscale resulting in bypass valve opening. ASSUMING that the reactor does not scram, SELECT the action that will result in closing of the bypass valves. Attached is "EHC Logic/Pressure Control Unit" Figure 3.

a. Decrease pressure set.

b. Decrease load limit.

c. Decrease maximum combined flow.

d. Increase load set.

QUESTION: 024 (1.00)

SELECT the operator action which would result in the largest decrease in feedwater heating while operating at 100% power.

a. Opening HV-1625 around feedwater heaters one, two and the drain cooler.

b. Opening HV-1623 around feedwater heaters three, four and five.

c. Closing HV-1768A feedwater heater 6A inlet valve.

d. Opening FV-1783A RFP minimum recirculation flow control valve.

QUESTION: 025 (1.00)

While operating at 100% power RFP B and C are in AUTO on "FIC-R601 FLOW CONT". RFP A is in POSN on "TURB MODE". The "FLOW DEVN" for FIC-601A reads +10. "TURBINE GOVENOR LOWER LIMIT" light is illuminated. SELECT the RFP response if the operator depresses the AUTO push button on "FIC-R601 FLOW CONT" for RFP A. Figure 5 "RFPT A Controls" is provided as a reference.

a. RFP A speed will increase and RFP B and C speeds will decrease.

b. RFP A speed will decrease and RFP B and C speeds will remain the same.

c. RFP A speed will remain the same and RFP B and C speeds will decrease.

d. RFP A speed will remain the same and RFP B and C speeds will remain the same.

QUESTION: 026 (1.00)

The FRVS Recirculation Fans are in AUTO. SELECT the total FRVS recirculation flow after one minute if a reactor building ventilation exhaust radiation signal of 1.8 x 10-3 μci/cc is received. ASSUME NO OPERATOR ACTIONS.

a. 0 cfm.

b. 60,000 cfm.

c. 120,000 cfm.

d. 180,000 cfm.

QUESTION: 027 (1.00)

While performing technical specification required individual control rod scram time testing, SELECT the valve that must be closed to **NOT** assist the scram time of the control rod.

a. V101 Insert riser valve.

b. V103 Drive water riser valve.

c. V104 Cooling water riser valve.

d. V113 Charging water riser valve.

QUESTION: 028 (1.00)

SELECT the action required by the rod sequence control system in the process of obtaining a 50% control rod density. Figure 6a, "RSCS Groups 1-4", is provided as a reference.

a. RSCS Group 1 must be the first group of control rods to be withdrawn to position 48. Continuous notch withdrawal to position 48 is permitted.

b. With RSCS Group 1 fully withdrawn to position 48, the second group to be moved must be RSCS Group 2. Notch withdrawal is enforced between notch positions 00 and 12.

c. With RSCS Groups 1 and 2 fully withdrawn to position 48, RSCS Group 4 can be withdrawn to position 48. Continuous notch withdrawal to position 48 is permitted.

d. With RSCS Groups 1 and 2 and 4 fully withdrawn to position 48, RSCS Group 3 control rods must be banked to notch position 04, with notch withdrawal enforced, prior to continuing with control rod movement.

QUESTION: 029 (1.00)

Reactor power is 18% during a plant startup.

SELECT the actions that will result in an INSERT BLOCK being applied to the selected control rod.

a. The operator selects (but does NOT move) a control rod which is not in the currently latched rod group.

b. A withdraw error exists and the operator selects a rod other than the error rod.

c. A control rod contained in a group lower than the currently latched group is withdrawn beyond its insert limit.

d. The operator withdraws the selected control rod beyond its withdraw limit.

QUESTION: 030 (1.00)

The recirculation pump seals are operating initially with the following conditions.

- No. 1 seal cavity pressure 1032 psig
- No. 2 seal cavity pressure 516 psig

Using the attached Figure 5 "Recirc. pump motor and seal instrumentation," SELECT the cause if the following conditions occur.

- No. 1 seal cavity pressure 1032 psig
- No. 2 seal cavity pressure 1032 psig
- FS "A" alarms low

a. Failure of No. 1 seal.

b. Failure of No. 2 seal.

c. Plugging of restricting orifice 1.

d. Plugging of restricting orifice 2.

QUESTION: 031 (1.00)

SELECT the reactor water cleanup system valves that will close in response to an automatic standby liquid control system start from the redundant reactivity control system.

a. HV-F100 "A" recirculation loop suction isolation
HV-F004 Outboard isolation

b. HV-F100 "A" recirculation loop suction isolation
HV-F106 "B" recirculation loop suction isolation
HV-F101 Bottom head drain isolation

c. HV-F001 Inboard isolation
HV-F004 Outboard isolation

d. HV-F001 Inboard isolation
HV-F106 "B" recirculation loop suction isolation

QUESTION: 032 (1.00)

SELECT the SEQUENCE of valve operation which will **NOT** result in an uncontrolled drainage of water from the reactor vessel.

a. - Close F004B, RHR PMP SUPP POOL SUCT MOV
   - Close F007B, RHR PMP MIN FL MOV
   - Open F006B, RHR PMP SUCT FROM RECIRC LOOP B ISLN MOV
   - Open F008, SHUTDOWN COOLING OUTBD ISLN MOV
   - Open F009, SHUTDOWN COOLING INBD ISLN MOV

b. - Close F007B, RHR PMP MIN FL MOV
   - Open F006B, RHR PMP SUCT FROM RECIRC LOOP B ISLN MOV
   - Open F008, SHUTDOWN COOLING OUTBD ISLN MOV
   - Open F009, SHUTDOWN COOLING INBD ISLN MOV
   - Close F004B, RHR PMP SUPP POOL SUCT MOV

c. - Open F006B, RHR PMP SUCT FROM RECIRC LOOP B ISLN MOV
   - Open F008, SHUTDOWN COOLING OUTBD ISLN MOV
   - Open F009, SHUTDOWN COOLING INBD ISLN MOV
   - Close F004B, RHR PMP SUPP POOL SUCT MOV
   - Close F007B, RHR PMP MIN FL MOV

d. - Close F004B, RHR PMP SUPP POOL SUCT MOV
   - Open F009, SHUTDOWN COOLING INBD ISLN MOV
   - Open F008, SHUTDOWN COOLING OUTBD ISLN MOV
   - Open F006B, RHR PMP SUCT FROM RECIRC LOOP B ISLN MOV
   - Close F007B, RHR PMP MIN FL MOV

QUESTION: 033 (1.00)

SELECT the condition that will **NOT** bypass the rod block monitor (RBM).

a. Control rod 03-18 is selected.

b. APRMs C and F indicate 28% power with APRM D bypassed.

c. No control rod is selected for movement.

d. APRMs D and E indicate 25% power with APRM C bypassed.

QUESTION: 034 (1.00)

The reactor is operating at 20% power when condenser vacuum begins to steadily decrease at a rate of 1 inch Hg absolute per minute. SELECT the SEQUENCE of actions that will occur if condenser vacuum continues to steadily decrease with NO OPERATOR ACTIONS.

a. - Reactor scram
   - MSIVs close
   - Main turbine trip
   - Feedwater pumps trip

b. - Main turbine trip
   - Feedwater pumps trip
   - Reactor scram
   - MSIVs close

c. - MSIVs close
   - Reactor scram
   - Feedwater pumps trip
   - Main turbine trip

d. - Main turbine trip
   - Reactor scram
   - Feedwater pumps trip
   - MSIVs close

QUESTION: 035 (1.00)

The stator water cooling pump AP119 is operating with PCV-3605 controlling stator water cooling water supply pressure at 36 psig. A malfunction of PCV-3605 causes stator water cooling water supply pressure to decrease to 22 psig. SELECT the plant response.

a. Stator water cooing pump BP119 will automatically start.

b. Main turbine runback is initiated. Reactor recirculation pump runback is not initiated. The main turbine will trip.

c. Reactor recirculation runback is initiated. Main turbine runback is not initiated.

d. Main turbine runback is initiated. Reactor recirculation pump runback is initiated

QUESTION: 036 (1.00)

All primary and secondary condensate pumps are running. SELECT the approximate system pressure downstream of the secondary condensate pumps with the secondary pumps running in a shutoff head condition.

a. 202 psig.

b. 450 psig.

c. 535 psig.

d. 720 psig.

QUESTION: 037 (1.00)

SELECT the automatic actuation that will **NOT** occur with a main generator differential overcurrent lockout signal.

a. Trip of the stator coolant water pumps.

b. Trip of the cooling fans for the main power transformers.

c. Initiation of the breaker failure protection of generator output breakers BS6-5 and BS2-6.

d. Trip of the alternator exciter field breaker.

QUESTION: 038 (1.00)

SELECT the normal 4.16KV electrical lineup for buses 10A401 (Channel A) and 10A402 (Channel B).

a. 10A401 is powered from station service transformer 1AX501
10A402 is powered from station service transformer 1BX501

b. 10A401 is powered from station service transformer 1AX502
10A402 is powered from station service transformer 1BX502

c. 10A401 is powered from station service transformer 1AX501
10A402 is powered from station service transformer 1AX501

d. 10A401 is powered from station service transformer 1BX502
10A402 is powered from station service transformer 1BX502

QUESTION: 039 (1.00)

Referring to the attached Figure 5 "UPS Power Control" and Figure 8 "Static Switch Front Panel". The operator places the manual bypass switch from "NORM" to "BYPASS TO ALTERNATE" to "ISOLATE". SELECT the effect this will have on the power supply to the load.

a. Backup power 480 VAC will supply power to the load.

b. Normal power 480 VAC will supply power to the load.

c. Alternate power 125 VDC will supply power to the load.

d. No power will be supplied to the load.

QUESTION: 040 (1.00)

SELECT the power supply to Channel A load "HPCI Relay Vertical Board 10C620."

a. 125 VDC bus 10D410.

b. 125 VDC bus 10D420.

c. 250 VDC bus 10D450.

d. 250 VDC bus 10D460.

QUESTION: 041 (1.00)

SELECT the location of the radiation monitors that are a backup to the main steam line radiation monitors to detect a fuel element failure.

a. Between first and second stage steam jet air ejectors.

b. In the gaseous radwaste holdup pipe.

c. Between the steam jet air ejector discharge and the gaseous radwaste feed gas recombiner.

d. Downstream of the gaseous radwaste HEPA filters.

QUESTION: 042 (1.00)

SELECT the condition in the Control Room Ventilation Radiation Monitoring System that will actuate the isolation mode of the control room ventilation system.

a. One of the two radiation detectors to the Channel D RM-80 reads downscale.

b. Channel C RM-23 reaches the "Alert" level.

c. Channel C RM-23 senses an "Operate Failure".

d. The RM-11 reaches the "Alert" level.

QUESTION: 043 (1.00)

SELECT the water fire suppression system that is MANUALLY initiated.

a. Wet sprinkler system in the secondary condensate pump room.

b. Preaction sprinkler system in the service water pump rooms.

c. Water spray/deluge system for the station service transformer.

d. Preaction deluge system for the FRVS recirc charcoal filter.

QUESTION: 044 (1.00)

In accordance with technical specifications, SELECT the statement that applies to reactor building pressure.

a. Need not be negative with thermal power less than 1% and the reactor coolant temperatures less than 200°F.

b. Must be negative any time reactor coolant temperature is greater than 200°F.

c. Need not be negative when irradiated fuel is being handled in the fuel pool with primary containment established.

d. Must be negative within 12 hours of placing the Mode Switch to STARTUP.

QUESTION: 045 (1.00)

The control room ventilation system A is operating normally with the CREF system selected to the O.A. MODE. Refer to Figure 1 "Control Room Supply System A Overview." SELECT the system flow response to a control room ventilation high radiation isolation at the points marked on the Figure 1.

a. A - 0
   B - 0
   C - 4000 scfm
   D - 4000 scfm
   E - 14500 scfm

b. A - 0
   B - 1000 scfm
   C - 4000 scfm
   D - 3000 scfm
   E - 14500 scfm

c. A - 3000 scfm
   B - 1000 scfm
   C - 4000 scfm
   D - 0
   E - 14500 scfm

d. A - 0
   B - 4000 scfm
   C - 4000 scfm
   D - 0
   E - 14500 scfm

QUESTION: 046 (1.00)

SELECT the response of the Traversing Incore Probe (TIP) to a reactor scram in which RPV level decreases to -40 inches when one detector is in the core. ASSUME the detector is being operated in the manual mode at the TIP control panel.

a. The TIP detector is automatically withdrawn from the core and when the detector has reached the "in-shield" position, the ball valve will automatically close.

b. The TIP detector is automatically withdrawn from the core and when the detector has reached the "in-shield" position the ball valve "Valve Open" light will dim allowing the operator to manually close the ball valve.

c. The TIP detector will remain in the core and the shear valve will automatically operate.

d. The TIP detector remains in the manual mode and must be withdrawn from the core manually.

QUESTION: 047 (1.00)

SELECT the system capable of providing emergency fill of the spent fuel pool.

a. Reactor Auxiliary Cooling System.

b. Service Water System.

c. Safety Auxiliary Cooling System.

d. Core spray system.

QUESTION: 048 (1.00)

SELECT the Refueling Platform Interlock Status Display Module light that will illuminate for the following condition.

- The refueling platform is over the reactor
- A load is on the grapple
- A control rod is withdrawn

a. Backup Hoist Limit.

b. Fuel Hoist Interlock.

c. Faulty Lockout.

d. Bridge Reverse Stop.

QUESTION: 049 (1.00)

Using the attached Figure 1 "MSIV Sealing System," SELECT the valve(s) that satisfy BOTH of the following conditions.

Condition 1 - Will not open unless the main steam line pressure is less than 25 psig.

Condition 2 - Considered containment isolation valve(s).

a. Valve 5829A.

b. Valves 5829A and 5829B.

c. Valves 5834A and 5835A and 5836A and 5837A.

d. Valves 5834A(B) and 5835A(B) and 5836A(B) and 5837A(B).

QUESTION: 050 (1.00)

SELECT the limit that the end of cycle RPT prevents exceeding.

a. MCPR.

b. MAPLHGR.

c. CMFLPD.

d. FRTP.

QUESTION: 051 (1.00)

While operating at power RPV pressure increases. SELECT the impact this will have on RPV water level and reactor power.

a. RPV water level increases - reactor power increases.

b. RPV water level increases - reactor power decreases.

c. RPV water level decreases - reactor power decreases.

d. RPV water level decreases - reactor power increases.

QUESTION: 052 (1.00)

RPS has failed to initiate a scram signal from 100% power. SELECT the method of reactor power reduction that will result in reactor SHUTDOWN in the SHORTEST time. ASSUME each method is successful when performed.

a. Trip of both recirculation pumps and injection of boron into the RPV.

b. Inject boron into the RPV and control RPV water level between 2/3 core height and top of active fuel.

c. Vent the scram air header.

d. Locally vent the over-piston volume of the control rod drive units.

QUESTION: 053 (1.00)

Hope Creek is operating at 95% power. A reactivity transient occurs which causes:

- APRM indication to increase
- No change in total core flow
- Generator MW to decrease
- Feedwater flow to decrease

SELECT the cause of the reactivity transient.

a. Continuous withdrawal of a shallow control rod.

b. Reactor recirculation pump speed increase.

c. Loss of feedwater heating due to closure of extraction steam bleeder trip valve.

d. HPCI initiation.

QUESTION: 054 (1.00)

DRYWELL PRESSURE HI/LO annunciator has alarmed with a current drywell pressure value of 1.55 psig. SELECT the possible cause of the alarm condition.

a. Reactor power decrease from 100% to 75%.

b. Decrease in Chilled water temperature.

c. Excessive venting of the torus.

d. Decrease in drywell cooling fan flow.

QUESTION: 055 (1.00)

In accordance with OP-EO.ZZ-102 B "Primary Containment Control," drywell sprays have been initiated at a suppression chamber pressure of 11 psig and a drywell temperature of 350°F. SELECT the condition and reason when drywell sprays are required to be terminated.

a. When drywell pressure decreases to less than the value contained in the Drywell Spray Initiation Curve (DWP-P-1) terminate drywell sprays to restore the RHR pumps for adequate core cooling.

b. When suppression chamber pressure decreases to less than 9.5 psig (suppression chamber spray initiation pressure) to preclude cyclic condensation of steam at the downcomer openings of the drywell vents.

c. When drywell pressure decreases to less than the high drywell pressure entry condition terminate drywell sprays to prevent exceeding the negative design pressure of the primary containment.

d. When all entry conditions into primary containment control have cleared and hydrogen concentration in the drywell and suppression pool is below 0.5% to assure that the hydrogen will not ignite.

QUESTION: 056 (1.00)

The reactor scrammed from 100% power due to loss of two reactor feedpumps. The reactor level has stabilized on one reactor feedpump without any operator actions. In carrying out the subsequent actions of OP-AB.ZZ-200 "Reactor Low Water Level," the operator resets the SETPOINT SETDOWN logic prior to matching the "Rx level Controller" setpoint to "Rx level." SELECT the plant response that will occur.

a. Feedwater control system will lock up the reactor feedwater pump turbine.

b. Feedwater control system will increase reactor feedwater pump speed.

c. Feedwater control system will decrease reactor feedwater pump speed.

d. No automatic actions will occur and the operator has to manually adjust reactor feedwater pump speed.

QUESTION: 057 (1.00)

SELECT the reactor pressure, drywell temperature and level instrument that would allow valid determination of a reactor water level of -100 inches. RPV saturation curve from EOP Caution 1 is provided as a reference.

a. RPV pressure - 300 psig
Drywell temperature - 400°F
Narrow range A

b. RPV pressure - 200 psig
Drywell temperature - 350°F
Wide range B

c. RPV pressure - 100 psig
Drywell temperature - 250°F
Fuel zone A

d. RPV pressure - 100 psig
Drywell temperature - 350°F
Upset range

QUESTION: 058 (1.00)

SELECT the method that has the most capacity (lbm/hr) for controlling reactor pressure following a reactor scram with failure of the turbine bypass valves and SRV's to operate.

a. One reactor feedpump operating with the pump discharge valve closed and the pump running on minimum flow.

b. Operating the RWCU in the recirculation mode at rated reactor pressure.

c. RCIC operating at rated flow CST to CST at a reactor pressure of 1020 psig.

d. HPCI operating at rated flow CST to CST at a reactor pressure of 1020 psig.

QUESTION: 059 (1.00)

During a failure to scram condition the standby liquid control (SLC) system has been manually initiated due to a high suppression pool temperature. After operator actions, twelve control rods are not able to be inserted to position 02. SELECT the condition that permits terminating SLC injection.

a. Indicated SLC tank level decreases to 3100 gallons.

b. Indicated SLC tank level decreases to 1200 gallons.

c. Indicated SLC tank level decreases to 325 gallons.

d. Suppression pool temperature decreases to less than 110°F.

QUESTION: 060 (1.00)

SELECT the condition that will require emergency RPV depressurization.

a. Suppression chamber pressure and suppression pool level are below the pressure suppression curve.

b. Suppression pool level and RPV pressure are below the SRV tailpipe level limit curve.

c. Unisolated leak in the RWCU room with radiation levels stabilizing at 100 times normal.

d. During an ATWS RPV water level cannot be maintained above -190 inches.

QUESTION: 061 (1.00)

SELECT the condition that will require a reactor scram as part of the immediate operator actions of the abnormal procedures.

a. While at 100% power the indicated total core flow differs by more than 10% from the established total core flow value derived from recirculation loop flow measurements.

b. Recirculation pump A trips while below the 80% rod line.

c. During an approach to criticality following refueling, a control rod drive travels past position "48" when fully withdrawn.

d. During an approach to criticality following refueling, CRD charging water header pressure is lost and two accumulator alarms are present.

QUESTION: 062 (1.00)

SELECT the condition that would represent thermal hydraulic instability following a trip of a recirculation pump.

a. LPRM peak to peak oscillations of greater than 10%.

b. APRM peak to peak oscillations of greater than 10%.

c. APRM downscale alarm.

d. LPRM downscale alarm.

QUESTION: 063 (1.00)

SELECT the condition that will result in the SLOWEST main condenser vacuum loss decay rate at 100% power.

a. Isolation of the steam jet air ejectors.

b. Loss of three circulating water pumps.

c. Opening of vacuum breaker valves.

d. Loss of sealing steam to turbine shaft gland seals.

QUESTION: 064 (1.00)

SELECT the component load that will be automatically sequenced onto the bus for a loss of offsite power (LOP) but **NOT** for a LOCA.

a. SACS pump.

b. R/B FRVS Vent fan.

c. Drywell cooler fan.

d. SSWS pump.

QUESTION: 065 (1.00)

SELECT the system AND procedure type that can be used to increase suppression pool water level with drywell pressure of 5 psig and suppression pool water level of 55 inches.

a. HPCI using a SO procedure.

b. Condensate Transfer using an EO procedure.

c. Torus water cleanup using a SO procedure.

d. Core spray using an EO procedure.

QUESTION: 066 (1.00)

Procedure OP-EO.ZZ-102 "Primary Containment Control" requires that suppression pool level be maintained below 125 inches. SELECT the basis for 125 inches.

a. The level that ensures the suppression chamber to drywell vacuum breakers are not submerged.

b. The level that ensures the HPCI exhaust line is not submerged.

c. The level that ensures emergency depressurization will not result in exceeding the heat capacity temperature limit.

d. The level that is the top of the indicating range for suppression pool level.

QUESTION: 067 (1.00)

SELECT the condition that would result in entry into OP-EO.ZZ-103 "Reactor Building Control."

a. "Reactor Cleanup Demineralizer System Equipment" digital alarm point of annunciator C6-A2 "RADIATION MONITORING ALARM/TRBL" high radiation alarm.

b. Reactor building HVAC exhaust radiation level of 1 x 10-4 μci/ml.

c. RWCU pump room temperature of 130°F.

d. RCIC pump room water level of 0.75 inches.

QUESTION: 068 (1.00)

Procedure OP-EO.ZZ-104 "Radiation Release Control," directs that if the turbine building ventilation system is shutdown the operator is to restart the turbine building ventilation system. SELECT the basis for this action.

a. Results in a positive pressure inside the turbine building to limit the intrusion of radioactivity into the turbine building.

b. Results in recirculation of the turbine building ventilation and reduction in the amount of radioactivity released.

c. Results in the radioactivity being discharged as a ground release to limit the dispersion of radioactivity.

d. Results in the radioactivity being discharged in a monitored release.

QUESTION: 069 (1.00)

The feeder breaker for an entire 250 volt distribution panel has tripped. SELECT the operator actions in the required sequence.

a. 1. Attempt of RECLOSE feeder breaker once and if not successful
   2. Determine cause of fault
   3. Correct cause of fault
   4. RECLOSE feeder breaker

b. 1. OPEN all individual supply breakers
   2. Determine cause of fault
   3. Correct cause of fault
   4. RECLOSE the feeder breaker
   5. Individually CLOSE supply breakers

c. 1. Determine cause of fault
   2. Correct cause of fault
   3. OPEN all individual supply breakers
   4. RECLOSE the feeder breaker
   5. Individually CLOSE supply breakers

d. 1. Determine cause of fault
   2. Correct cause of fault
   3. RECLOSE the feeder breaker

QUESTION: 070 (1.00)

During preparation for refueling with the fuel pool at the normal level, the gate between the reactor well and fuel pool catastrophically fails with the refueling bellows not yet in place. SELECT the fuel pool level response.

a. Fuel pool level will be maintained above the technical specification minimum fuel pool level.

b. Fuel pool level will decrease approximately 12 feet.

c. Fuel pool level will decrease to approximately 8 feet above the top of the fuel.

d. Fuel pool level will decrease to below the top of fuel with two thirds of the fuel covered.

QUESTION: 071 (1.00)

During establishing shutdown cooling, a RPV reactor water level transient below Level 2 has occurred coincident with an inability to use the RHR pumps. Assuming RPV water level cannot be raised above Level 2, SELECT the alternate method of decay heat removal that will be effective with these conditions.

a. RWCU maximizing RACS to the non-regenerative heat exchangers.

b. Condensate transfer system via the ECCS injection lines maximizing flow to the RPV.

c. Vessel head spray maximizing mixing of RPV water.

d. Maximizing fuel pool cooling.

QUESTION: 072 (1.00)

During fuel movement the following alarms are generated:

REFUELING FL AIRBORNE ACTIVITY HI
R B AIRBORNE ACTIVITY HI
RADIATION MONITORING ALARM/TRBL
NEW FUEL CRITICALITY RAD HI

SELECT the action that is an immediate operator action.

a. Suspend refueling operations after any fuel assembly attached to the fuel handling grapple is placed in the fuel pool.

b. Evacuate all unnecessary personnel from the reactor building.

c. Verify reactor building ventilation system isolation and verify FRVS start.

d. Notify radiation protection to respond and control access to the reactor building.

QUESTION: 073 (1.00)

SELECT the automatic action, and the basis for the automatic action, at RPV water level 8.

a. Initiates feedwater level control setpoint setdown to reduce the feedwater flow entering the vessel.

b. RCIC F045 valve closes to prevent overfilling the reactor vessel.

c. Trip the HPCI turbine to prevent HPCI turbine blade damage.

d. Close the MSIVs to limit the amount of water entering the steam lines.

QUESTION: 074 (1.00)

Following a control room evacuation SELECT the desired mode of RPV water level and reactor pressure control.

a. Feedwater pumps controlling RPV water level and turbine bypass valves controlling reactor pressure.

b. RCIC controlling RPV water level and HPCI controlling reactor pressure.

c. RCIC controlling RPV water level and SRVs controlling reactor pressure.

d. HPCI controlling RPV water level and SRVs controlling reactor pressure.

QUESTION: 075 (1.00)

SELECT the response of the condensate and feedwater system valves to a loss of instrument air.

a. - Primary condensate pump minimum flow recirculation valve (HV-1710) - fails open.
   - Secondary condensate pump minimum flow recirculation valves (HV-1650A-C) - fail open.
   - RFP recirc control valves (FV-1783A-C) - fail open.

b. - Primary condensate pump minimum flow recirculation valve (HV-1710) - fails open.
   - Secondary condensate pump minimum flow recirculation valves (HV-1650A-C) - fail open.
   - RFP recirc control valves (FV-1783A-C) - fail closed.

c. - Primary condensate pump minimum flow recirculation valve (HV-1710) - fails open.
   - Secondary condensate pump minimum flow recirculation valves (HV-1650A-C) are motor operated and are not affected.
   - RFP recirc control valves (FV-1783A-C) - fail closed.

d. - Primary condensate pump minimum flow recirculation valve (HV-1710) is motor operated and is not affected.
   - Secondary condensate pump minimum flow recirculation valves (HV-1650A-C) - fail open.
   - RFP recirc control valves (FV-1783A-C) - fail open.

QUESTION: 076 (1.00)

SELECT the system that supplies the cooling water flow to the containment instrument gas compressor cooler.

a. RACS.

b. SACS.

c. TACS.

d. SSWS.

QUESTION: 077 (1.00)

SELECT the containment vent path that if used to control containment hydrogen concentration would result in an unscrubbed, unmonitored and untreated radioactive release to the environment.

a. Vent the containment via the drywell 26 inch exhaust.

b. Vent the containment via the drywell supply and ILRT piping.

c. Vent the containment via the suppression chamber 24 inch exhaust.

d. Vent the containment via the suppression chamber supply and ILRT piping.

QUESTION: 078 (1.00)

During alignment of RACS to supply backup to the Chilled Water System for drywell cooling, the operator performed actions in the following sequence.

CLOSED - HV9532-1 CHW ISLN RTN VLV
CLOSED - HV9532-2 CHW ISLN SUP VLV
DEPRESSED - Containment Clg Sply Select Loop A Sply/Rtn Chw/RACS Sel Open RACS Pb
DEPRESSED - Containment Clg Sply Select Loop B Sply/Rtn Chw/RACS Sel Open RACS Pb
OBSERVED - SEE BELOW
OPENED - HV9532-1 CHW ISLN RTN VLV
OPENED - HV9532-2 CHW ISLN SUP VLV

SELECT what the operator should have observed.

a. RACS pumps trip on low-low system flow.

b. RACS pumps trip on low-low head tank level.

c. RACS/Chilled Water Isolation valves (HV-9531A1,2,3,4; B1,2,3,4) reposition as designed.

d. RACS/Chilled Water Isolation valves (HV-9531A1,2,3,4; B1,2,3,4) do not reposition.

QUESTION: 079 (1.00)

Following a reactor scram the following procedures have been entered.

OP-AB.ZZ-126 Abnormal Release of Gaseous Radioactivity
OP-EO.ZZ-103 Reactor Building Control
OP-EO.ZZ-100 Scram
OP-EO.ZZ-99 Post Scram Recovery

Reactor Building Ventilation System has automatically isolated and FRVS has automatically initiated. The refuel floor HVAC radiation level has peaked at 5 x 10-4 μci/ml.

SELECT the status of the Hydrogen/Oxygen Analyzer System.

a. In service.

b. Isolated on low reactor water level.

c. Isolated on refuel floor HVAC high-high radiation.

d. Isolated on reactor building vent exhaust high-high radiation.

QUESTION: 080 (1.00)

The plant is operating in the RUN mode with main steamline radiation monitor C failed upscale. SELECT the plant response if the normal power supply to RPS MG B from MCC 00B484 fails.

a. Loss of indication on MSL rad monitors B and D and a 1/2 scram condition exists.

b. Loss of indication on MSL rad monitors B and D and a full scram.

c. No loss of indication on MSL rad monitors B and D and a 1/2 scram condition exists.

d. No loss of indication on MSL rad monitors B and D and a full scram.

QUESTION: 081 (1.00)

SELECT the component that will be the last to be loaded onto the bus by the LOCA load sequencer.

a. Service water pump.

b. FRVS Recirc fan.

c. DG room recirc fan.

d. SACS pump.

QUESTION: 082 (1.00)

SELECT how secondary containment over pressure protection is provided for the HPCI room.

a. Blowout panel relieves at 0.25 psid to the torus compartment.

b. Blowout panel relieves at 1.5 psid to the torus compartment.

c. Blowout panel relieves at 0.25 psid to the steam vent.

d. Blowout panel relieves at 1.5 psid to the steam vent.

QUESTION: 083 (1.00)

SELECT the system that requires an Independent Verification if a local manual valve in the system is manipulated.

a. SACS.

b. Circ water.

c. RACS.

d. EHC.

QUESTION: 084 (1.00)

With the reactor at 95% power, SELECT the individuals that are authorized to reduce recirculation pump speed by performing local operation of the recirculation pump scoop tube.

I. NSS under the direction of the NCO at the controls.

II. NEO under the direction of the NCO at the controls.

III. NEO under the direction of an NSS at the scoop tube.

IV. An individual in the licensed operator training program under the direction of the NCO at the controls and in the presence of an NSS at the scoop tube.

a. I and IV.

b. II and III.

c. I and III.

d. II and IV.

QUESTION: 085 (1.00)

An room is posted as an airborne contaminated area at a level of 1 MPC SELECT the maximum amount of time an individual can spend in the room during a calendar quarter.

a. 1 hour.

b. 40 hours.

c. 260 hours.

d. 520 hours.

QUESTION: 086 (1.00)

SELECT the responsibility of the Nuclear Control Operator during the implementation of the Safety Tagging program.

a. Verify that work on equipment under the jurisdiction of the Electric System Operator (ESO)has been cleared with the ESO.

b. Complete the Tagging Request and Work Request/Order Log.

c. Instruct the assigned Nuclear Equipment Operator in the specific tagging operation and the purpose of the tagging request.

d. Position the electrical and mechanical components in the field in accordance with the tagging request.

QUESTION: 087 (1.00)

In accordance with NC.NA-AP.ZZ-0005(Q) "Station Operating Practices," SELECT the normal complement of Fire Brigade personnel for power operations.

a. 3.

b. 4.

c. 5.

d. 6.

QUESTION: 088 (1.00)

A licensed operator has the 10CFR55 required medical examination on January 15, 1992. SELECT the latest date that the next medical examination must be performed to not violate the medical requirements of the 10CFR55 operators license.

a. January 15, 1993.

b. January 15, 1994.

c. January 15, 1995.

d. January 15, 1998.

Deleted from exam

QUESTION: 089 (1.00)

In accordance with HC.OP-AP-ZZ-0107 "Shift Relief and Turnover," SELECT the NCO Relief Checklist items that must be completed prior to accepting the shift.

I. Most Recent P-1.

II. Surveillance Log.

III. Inop Instrument/Alarm Log.

IV. Action Statement Log.

a. I and II.

b. II and IV.

c. II and III.

d. I and III and IV.

QUESTION: 090 (1.00)

SELECT the emergency classification that would be declared during an accident that resulted in loss of the fission barriers of reactor coolant and loss of containment integrity and required offsite protective action recommendations.

a. Unusual event.

b. Alert.

c. Site Area.

d. General Emergency.

QUESTION: 091 (1.00)

During a hydrogen gas charging evolution of the Main Generator, SELECT the precaution that must be observed.

a. Do not exceed hydrogen purity of 75% with air present.

b. Use only non-coated carbcn steel tools in the vicinity of hydrogen piping.

c. Do not refill the generator if it has been allowed to go below atmospheric pressure.

d. Do no admit hydrogen to the generator if carbon dioxide is present.

QUESTION: 092 (1.00)

SELECT the procedure change that would be acceptable to use an On-the-Spot Change.

a. Change to the purpose of the procedure.

b. Addition of new acceptance criteria to the procedure.

c. Addition of an equivalent operation of a system to that currently contained in the procedure.

d. Removal of a QA hold point at the request of the QA supervisor.

QUESTION: 093 (1.00)

SELECT the tag that is represented by the following:

> Identifies the mechanical or low voltage electrical blocking (600 volts or less) points between any circuit or equipment that is energized and the de-energized equipment upon which work is to be performed. Operation of the equipment is permitted by the worker named on the tag or by persons directed by the worker.

a. Red Blocking Tag.

b. Yellow Permissive Tag.

c. Worker's Blocking Tag.

d. White Caution tag.

QUESTION: 094 (1.00)

Referring to the attached Figure of the area of the main control room, SELECT the areas defined for the licensed operator "at the controls" during normal operations.

a. Areas I and II and III.

b. Areas II and III and IV.

c. Areas I and II and III and IV and V.

d. Areas II and III.

QUESTION: 095 (1.00)

Using the attached procedure HC.OP-AP.ZZ-0104 "Containment Prepurge Cleanup, Inerting, Deinerting or Pressure Control" and the data contained in Attachment 1 of the procedure, DETERMINE the number of hours available in Attachment 2 to open the drywell and suppression chamber purge supply and exhaust valves.

______________ hours available (MARK ANSWER ON ANSWER SHEET)

QUESTION: 096 (1.00)

Reactor pressure has been reduced to 470 psig and has been held at that pressure for 1 hour. DETERMINE the LOWEST reactor pressure that the reactor can be depressurized over the next 1 hour without exceeding the maximum TECHNICAL SPECIFICATION allowable reactor coolant system cooldown rate.

______________ psig (MARK ANSWER ON ANSWER SHEET)

QUESTION: 097 (1.00)

FILL IN THE BLANK

Diesel generator A is currently loaded at 3730 KW. DETERMINE the additional load that the diesel generator can accommodate for the next 2 hours and **NOT EXCEED** a 10% overload condition.

_______________ Kw additional load (MARK ANSWER ON ANSWER SHEET)

QUESTION: 098 (1.00)

DETERMINE the minimum number of LPRM inputs required to avoid an automatic APRM inoperative trip.

_______________ LPRM inputs. (MARK ANSWER ON ANSWER SHEET)

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

PSE&G

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038

Hope Creek Generating Station

December 13, 1991
NTD-91-3274

Mr. Lee H. Bettenhausen, Chief
Operations Branch
Division of Reactor Safety
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Dear Mr. Bettenhausen:

EXAMINATION REVIEW COMMENTS - HOPE CREEK LICENSE EXAMINATION

Attached are comments on six questions used on the written examination administered at Hope Creek Generating Station on December 9, 1991. These comments have been developed following the examination and are presented in the same order as originally numbered on the examination. They are in addition to comments provided by Messrs. Bauer, Kirwin, Wynn, and Gott during the pre-examination review conducted on December 4, 1991.

The following format has been used to document specific comments:

A. NRC question, answer and reference;

B. facility comment including a recommendation for resolution; and

C. support documentation,.

If you have any questions, comments, or need additional information, please call G. Mecchi, (609) 339-3857, or W. Gott, (609) 339-3769. They will provide the requested information or will see that you are contacted by the appropriate person.

Sincerely yours,

J. Hagan

Attachments (6)

BC General Manager - Nuclear Services
Manager - Licensing and Regulation
Manager - Nuclear Training
Operations Manager - Hope Creek
Principal Trainer - Operations Training
Prin. Trng. Supr. - Hope Creek Operations

A. NRC Question, Answer and Reference

QUESTION: 011 (1.00)

Core Spray Logic Channel C is inadvertently manually initiated with the reactor at 400 psig. SELECT the action that will **NOT** occur.

a. Emergency diesel generator C auto starts.

b. Non-1E loads on the respective vital bus are load shed.

c. HV-F005A "Core spray loop injection valve" opens.

d. HV-F031A "Core spray pump minimum flow valve" opens when the pump starts and closes when loop flow is > 775 gpm.

ANSWER: 011 (1.00)

c

REFERENCE:

Lesson plan 302H-000.00H-000027-09 Core Spray pg 42
Learning objective 4

[3.3/3.1]

209001A208 ..(KA's)

B. Facility Comments

This question specifically asks for actions that will not occur, and provides two actions that meet the stated requirement. The question implies that the core spray system is in its normal standby line-up (HV-F031A OPEN, HV-F005A CLOSED). This assumption is required at a minimum for any choice to be correct. Since the core spray minimum flow valve HV-F031A is in its normal (OPEN) position, it can not open on pump start. The core spray loop injection valve (HV-F005A) will not open on Logic C. If HV-F005A does not open, loop flow can not increase to 775 gpm and the minimum flow valve (HV-F031A) will not close. So there are two responses that are correct for this question.

It is recommended that both responses, c. and d., be accepted as correct answers.

C. Support Documentation

HC.OP-AP.ZZ-0107(Q), Attachment 6, page 3 of 7
HC.OP-ST.BE-0001(Q), Attachment 2, page 10 of 12

QUESTION: 029 (1.00)

[illegible]tor power is 18% during a plant startup.

SELECT the actions that will result in an INSERT BLOCK being applied to the selected control rod.

a. The operator selects (but does NOT move) a control rod which is not in the currently latched rod group.

b. A withdraw error exists and the operator selects a rod other than the error rod.

c. A control rod contained in a group lower than the currently latched group is withdrawn beyond its insert limit.

d. The operator withdraws the selected control rod beyond its withdraw limit.

REFERENCE:

Lesson plan 302H-000.00H-000010-08 RSCS Pg 21-22
Learning objective 7

[3.2/3.3]

201004K101 ..(KA's)

ANSWER: 029 (1.00)

b.

B. Facility Comments

The question does not require limiting response to only the Rod Worth Minimizer (RWM). At a power level of 18%, Rod Sequence Control System (RSCS) is active and will also generate an INSERT BLOCK. If a rod is selected and is not in the currently latched group (answer a.), RSCS will cause an INSERT ROD BLOCK to occur.

It is recommended that both answers, a. and b., be counted as correct.

C. Support Documentation

HC.OP-IO.ZZ-0004(Q), 5.1.10.c
Lesson Plan 302H-000.00H-000010-08, page 23

QUESTION: 059 (1.00)

During a failure to scram condition the standby liquid control (SLC) system has been manually initiated due to a high suppression pool temperature. After operator actions, twelve control rods are not able to be inserted to position 02. SELECT the condition that permits terminating SLC injection.

a. Indicated SLC tank level decreases to 3100 gallons.

b. Indicated SLC tank level decreases to 1200 gallons.

c. Indicated SLC tank level decreases to 325 gallons.

d. Suppression pool temperature decreases to less than 110°F.

REFERENCE:

Lesson plan 302H-000.00H-00124C pg 30 Objective 8

[3.8/3.8]

295025K303 .'KA's)

ANSWER: 059 (1.00)

c

B. Facility Comments

The question asks the applicant to select the condition that permits terminating SLC injection. Step RC/Q-2 of EOP-101 addresses the action "TERMINATE BORON INJECTION". Neither of the conditions specified in this step, i.e., "ALL CONTROL RODS ARE INSERTED TO BEYOND POSITION 02", nor "REACTOR ENGINEERING HAS DETERMINED THAT THE REACTOR WILL STAY SHUTDOWN UNDER ALL CONDITIONS WITHOUT BORON", are included in the responses to the question. TERMINATE is defined as to - "take the appropriate action to stop the stated action, process, or evolution. Generally, the most direct action which will stop the stated action is preferred; however, a wide variety of actions may be employed".

Step RC/Q-21 specifies that "If SLC TANK WATER LEVEL DROPS TO 325 GALLONS THEN VERIFY TRIP OF SLC PUMPS". VERIFY means to - "use available indications (status lights, plant and system parameters) and/or physical observation to determine that the specific action has occurred; if the specific action has not occurred, then take the necessary actions to cause it to occur". The value 325 gallons corresponds to SLC pump Auto Trip setpoint on SLC tank low level. This trip setpoint is established to prevent damage to the SLC pumps from loss of suction head.

Step RC/Q-23 specifies that "WHEN THE COLD SHUTDOWN BORON WEIGHT OF BORON HAS BEEN INJECTED INTO THE RPV (LESS THAN 1200 GALLONS IN THE SLC TANK) THEN CONTINUE IN THIS PROCEDURE AT RC/Q-16". This step does not tell the operator to terminate boron injection. However, requiring the RO applicant to recall specific setpoints from steps deep within the EOPs, goes far beyond the stated requirement to commit EOP entry conditions to memory. When the language used to compose the responses to the question deviate from the very specific language used in the EOPs, the correct response identified on the answer key can NOT be shown to follow what is asked in the question.

The reference cited for this question is in error. It should be Lesson Plan 302H-000.00H-00124 B, page 30, Objective 8, "Given any step of the procedure, explain the reason for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step."

Based on the above discussion, both answers b. and c. should be considered as correct.

C. Support Documentation

EOP - 101 RC/Q LEG
OP-EO.ZZ-101(Q), page 5, Section 3.0 Definitions - 3.13 & 3.14

QUESTION: 069 (1.00)

The feeder breaker for an entire 250 volt distribution panel has tripped. SELECT the operator actions in the required sequence.

a. 1. Attempt of RECLOSE feeder breaker once and if not successful
   2. Determine cause of fault
   3. Correct cause of fault
   4. RECLOSE feeder breaker

b. 1. OPEN all individual supply breakers
   2. Determine cause of fault
   3. Correct cause of fault
   4. RECLOSE the feeder breaker
   5. Individually CLOSE supply breakers

c. 1. Determine cause of fault
   2. Correct cause of fault
   3. OPEN all individual supply breakers
   4. RECLOSE the feeder breaker
   5. Individually CLOSE supply breakers

d. 1. Determine cause of fault
   2. Correct cause of fault
   3. RECLOSE the feeder breaker

ANSWER: 069 (1.00)

b

REFERENCE:

OP-AB.ZZ-0149 pg 2

[3.2/3.3]

295004A204 ..(KA's)

B. <u>Facility Comments</u>

The sequence of actions contained in both answers b. and c. of this question are equally valid in addressing the problem posed. Abnormal operating procedures (AOP) "250VDC System Malfunction" (HC.OP-AB.ZZ-0149) and "125VDC System Malfunction" (HC.OP-AB.ZZ-0150) both provide guidance regarding tripped feeder breakers in the subsequent action sections, 4.6 and 4.8, respectively. Both caution the operator to "Never reclose a tripped breaker until the cause of the fault is determined <u>and</u> corrected," Caution 4.6 and 4.8, respectively. Additionally, Section 5.1.1.3.b, "Use of Operations Department Procedure" (OP-AP.ZZ-102), in referring to use of AOPs, states "The Subsequent Actions shall be performed with the procedure immediately accessible or at the direction of a person with the procedure immediately accessible." The question places the candidate in the position of recalling the required sequence when more than one valid sequence is provided without having access to the specific procedure.

It is recommended that either answers b. and c. be accepted as correct or the question be removed from the examination.

C. <u>Support Documentation</u>

HC.OP-AB.ZZ-0149(Q), Caution 4.6
HC.OP-AB.ZZ-0150(Q), Caution 4.8
HC.OP-AP.ZZ-102(Q), 5.1.1.3

QUESTION: 081 (1.00)

SELECT the component that will be the last to be loaded onto the bus by the LOCA load sequencer.

a. Service water pump.

b. FRVS Recirc fan.

c. DG room recirc fan.

d. SACS pump.

ANSWER: 081 (1.00)

a

REFERENCE:

Lesson plan 302H-000.00H-000066-12 Table 1C
Objective 18

[4.1/4.1]

295003K203 ..(KA's)

B. Facility Comments

There are four separate LOCA load sequencers (Channels A, B, C and D); each with similar, but slightly different loads. In addition, there are four separate LOP sequencers (Channels A, B, C and D); each with similar, but slightly different loads. Likewise the loads on the LOCA and LOP are similar to one another, but different. To correctly answer this question, the applicant would be required to commit eight lists to memory. The objective referenced for this task specifically states that available reference material is to be given. Even with reference materials available, identification of the component that is the last to be loaded is not always an easy matter.

The implication of this question must be that none of the loads listed are already loaded onto the bus at the time they receive the LOCA sequencer start signal; otherwise they would simply continue to run. The respective DG room recirc fan (E, F, G or H) is the last component to receive a LOCA start signal from its appropriate channel (A, B, C or D) and if <u>it is not running</u> it should be the last to start, albeit from a redundant start signal. The argument could be made that if the DG room recirc fan did not start on the first start signal, why should it be expected to start later on the redundant signal? If that were true (the DG room recirc fan would not run for some reason), the original answer, Service Water pump, would be correct.

It is recommended that both answers, a. and c., be considered correct.

C. Support Documentation

HC.OP-AB.ZZ-0135(Q), Tables 5.1 & 5.2

QUESTION: 088 (1.00)

A licensed operator has the 10CFR55 required medical examination on January 15, 1992. SELECT the latest date that the next medical examination must be performed to not violate the medical requirements of the 10CFR55 operators license.

a. January 15, 1993.

b. January 15, 1994.

c. January 15, 1995.

d. January 15, 1998.

ANSWER: 088 (1.00)

b

REFERENCE:

10CFR55.21

[4.2/4.2]

294001A102 ..(KA's)

B. Facility Comments

While 10CFR55.21 requires a "medical examination by a physician every two years," making answer b. correct, answer a. should also be accepted since the licensed operator medical examinations at Hope Creek are governed by Nuclear Medical Department Procedure NMD-025. According to NMD-025:

1. One of the responsibilities of the Medical Director - Nuclear is to "initiate and track annual physical." (Section 7.1.1)

2. The physical examination "will be consistent with the requirements of ANSI/ANS 3.4 1983, ANSI Z88.6 1984, and NUREG 0041." (Section 9.1)

3. "License renewal medical examinations shall be conducted on an annual (10 month) basis as requalification exams..." (Section 11.2)

The 10CFR55 required medical examinations are not directly scheduled by the individual licensed operator and are performed annually. The discriminatory value of the question for Hope Creek licensed operators is minimal and either both a. or b. answers should be accepted.

C. Support Documentation

Nuclear Medical Department Procedure NMD-025

Attachment 3

NRC Resolution of Facility Comments

1. Question 11. Accept facility comment that (c) and (d) are correct answers. The answer key will be revised.

2. Question 29. Accept facility comment that (a) and (b) are correct answers. The answer key will be revised.

3. Question 59. For the conditions specified in the stem of the question, the only permissible condition to terminate the SLC manual initiation is when the SLC pump trips at a SLC tank level indication of 325 gallons. The facility comment is not accepted.

4. Question 69. Accept facility comment that (b) and (c) be considered as correct answers. The answer key will be revised.

5. Question 81. Accept facility comment that (a) and (c) be considered as correct answers. The answer key will be revised.

6. Question 88. The question does not relate to the facility administrative requirements for medical examinations but to the requirements as specified in 10 CFR 55. The facility comment is not accepted. However upon further NRC review, we have determined that the written examination is not the forum to ask this question and it has been deleted from the examination.

Attachment 4

Simulator Facility Report

Licensee: Public Service Electric and Gas

Facility: Hope Creek

Docket No.: 50-354

Operating tests administered: December 10-11, 1991

This form is to be used only to report observations. These observations do not constitute audit or inspection findings. They also do not affect NRC certification or approval of the simulator other than to provide information which 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
None	

Attachment 5

EXIT ATTENDEES

December 13, 1991

Public Service Electric and Gas

J. Hagen, General Manager, Hope Creek
R. Hovey, Operations Manager, Hope Creek
A. Orticelle, Manager, Nuclear Training
G. Mecchi, Principal Training Supervisor, Operations Training
W. Gott, Principal Training Supervisor, Hope Creek
R. Griffith, Sr., Manager, Station QA - Hope Creek
M. Cirelly, Hope Creek Station Licensing Engineer

U. S. Nuclear Regulatory Commission

T. Fish, Senior Operations Engineer
D. Florek, Senior Operations Engineer
K. Lathrop, Resident Inspector