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

EXAMINATION REPORT NO.	90-18 (OL)
FACILITY DOCKET NO.	50-219
FACILITY LICENSE NO.	DPR-16
LICENSEE:	GPU Nuclear Corporation P. O. Box 388 Forked River, New Jersey 08731
FACILITY:	Oyster Creek Nuclear Generating Station
EXAMINATION DATES:	October 15 - 18, 1990
EX/ 'ERS:	T. Walker, Senior Operations Engineer T. Morgan, EG&G (NRC Contractor) L. Vick, OLB, NRR (Observer) M. Leach, RIII (Observer)

CHIEF EXAMINER: Tim E Walker 12/27/90
T. Walker, Senior Operations Engineer — Date

APPROVED BY: Lee H Bettenhausen for 1/3/91
Richard J. Conte, Chief, BWR Section — Date
Operations Branch, Division of Reactor Safety

EXECUTIVE SUMMARY

Written and operating examinations were administered to two (2) senior reactor operator (SRO) upgrade applicants. Both of the applicants passed the written examination. Neither of the applicants passed the operating examination. Weaknesses in supervisory ability and use of Emergency Operating Procedures (EOPs) contributed to the failures for both applicants.

Requalification written examinations were administered to four (4) licensed reactor operators (ROs) and one (1) licensed SRO. Requalification walkthrough examinations were administered to three (3) licensed ROs and one (1) licensed SRO. All of the licensed operators passed all administered portions of the requalification examinations.

During the review of the Job Performance Measures (JPMs) selected for the requalification walkthrough examinations, the examiners identified several problems with the operating procedures associated with the JPMs. These discrepancies had been identified by training personnel during validation of the JPMs, but the problems were not corrected until after the NRC arrived on site to administer the examinations. Failure to use the requalification program to improve plant procedures was identified previously during the NRC administered requalification examinations in April, 1990.

Problems were identified with the reference material supplied by the facility for preparation of the examinations (Section 3.4). The reference materials were incomplete and out-of-date. This was the second occurrence of this problem since 1987.

Section 4.0 of the report describes findings related to the EOP flowcharts, EOP training, and equipment required for implementing the EOPs. The findings related to the flowcharts and training were corroborated by observations during administration of the examinations.

DETAILS

1.0 INTRODUCTION AND OVERVIEW

The NRC examiners administered replacement examinations to two SRO upgrade applicants. The NRC examiners also administered requalification written examinations to four licensed ROs and one licensed SRO, and requalification walkthrough examinations to three licensed ROs and one licensed SRO. The examinations were administered in accordance with NUREG 1021, Examiner Standards, Rev. 5, dated January 1, 1989.

The replacement examinations were prepared by the NRC examiners. The requalification examinations were prepared by the facility, then reviewed and approved by the NRC. Prior to administration of the examinations, a pre-examination review was conducted at the Region I office. The Manager of Plant Operations (an SRO), the Operations Training Coordinator (an SRO), the Supervisor of Operations Training (an SRO), another member of the training staff, and the Chief Examiner were present at the review. All facility individuals involved with the preparation and review of the examination materials signed security agreements to ensure that there was no compromise of the examinations.

2.0 PERSONS CONTACTED

2.1 U.S. Nuclear Regulatory Commission

E. Collins, Senior Resident Inspector
* M. Bannerjee, Resident Inspector

2.2 GPU Nuclear Corporation

* R. P. Coe, Director, Training and Education
* E. E. Fitzpatrick, Vice President and Director, Oyster Creek
* J. D. Kowalski, Manager, Plant Training, Oyster Creek
* R. Barrett, Plant Operations Director
* P. Scallon, Manager, Plant Operations
* H. Tritt, Supervisor, Operations Training
* J. Boyle, Operations Training Coordinator
M. Rossi, Operations Training Instructor
* G. Hicks, Operations Quality Assurance
* P. Thompson, Quality Assurance Auditor
* M. Heller, Oyster Creek Licensing Engineer

The examiners also held discussions with various licensed operators during the administration of the examinations.

* Denotes those present at the exit interview on October 18, 1990.

3.0 EXAMINATION RELATED FINDINGS AND CONCLUSIONS

3.1 Examination Results

Replacement:

	RO Pass/Fail	SRO Pass/Fail
Written	N / A	2 / 0
Operating	N / A	0 / 2
Overall	N / A	0 / 2

Requalification:

	RO Pass/Fail	SRO Pass/Fail
Written	4 / 0	1 / 0
Walkthrough	3 / 0	1 / 0
Overall	6 / 0	1 / 0

3.2 Operating/Walkthrough Examinations

Due to the limited number of replacement examination applicants, no specific generic strengths or weaknesses were noted on the operating tests. However, weaknesses in supervisory ability and use of procedures, specifically the Emergency Operating Procedures (EOPs) contributed to the failures for both applicants.

The control room staff was cooperative and attempted to maintain an environment conducive for conduct of the operating examinations. However, the performance of a scheduled emergency drill during administration of the replacement operating tests was distracting to the applicants and the examiners.

Due to the limited number of operators examined, no generic strengths or weaknesses were noted on the requalification walkthrough examinations.

Open (219/90-05-02). Requalification program deficiency in improving the quality of poor operating procedures. During the review of the Job Performance Measures (JPMs) to be used for the requalification walkthrough examinations, the NRC examiners identified several problems with the operating procedures associated with the JPMs. When these discrepancies were brought to the licensee's attention at the pre-examination review, the licensee representatives indicated that they had identified the same problems when validating the JPMs, but did not know if the problems had been corrected. When the NRC arrived on site to administer the examinations, the most significant of the procedural problems had not been corrected. The incorrect procedure was scheduled to be used later in the week to restore a recirculation loop to service. The procedural discrepancy was again brought to the licensee's attention at the entrance meeting and action was taken to correct the procedure the following day. The licensee indicated that the procedure had not been changed when the problem was identified during validation of JPMs to ensure that there was no compromise of the requalification examinations. Licensee representatives indicated that they understood the need to correct procedural problems identified during examination activities. The failure to correct the procedure in this instance was apparently due to a misunderstanding by operations and training personnel as to the expectations of licensee management. The lack of use of the requalification program to improve plant procedures remains unresolved pending further NRC review (219/90-05-02).

3.3 Written Examinations

Due to the limited number of replacement examination applicants, no generic strengths or weaknesses were noted on the written examination. The facility identified one error on the written examination during the post-examination review. This error was also recognized by the NRC immediately following administration of the written examinations and was corrected on the master copy of the examination prior to grading.

Due to the limited number of operators examined, no generic strengths or weaknesses were noted on the requalification written examinations.

3.4 Reference Material

The following problems were noted with the reference material provided by the facility for examination preparation:

- Materials were missing from both sets of material supplied to the NRC. System Operating Procedures (SOPs) for major safety-related systems such as the Isolation Condensers and Reactor Manual Control System were missing. Numerous pages and figures, as well as several complete chapters, were missing from the Operations Plant Manuals (OPMs) that were provided. Due to the extent of missing material, the licensee had to send a complete replacement set of OPMs to the contract examiner. Several Training Content Records (TCRs) were also missing.

- The request for reference materials sent to the facility 90 days prior to the examinations requested the licensee to supply administrative procedures that are applicable to reactor operation and safety along with procedures for radiation controls. Several procedures related to radiation control that appear to be important were not provided. For example, the procedures for the ALARA program and the Rules and Conduct for Radioactive Work were not included with the reference materials. One administrative procedure (9300-ADM-1201.02, "Awareness Reporting") that is specifically referenced by the Group Operating Supervisor (GOS) Qualification Cards was not provided.

- Numerous surveillance procedures for safety-related systems that are the responsibility of licensed operators were not provided. For example, the Core Spray Pump and Valve Operability test was not provided. No surveillance tests were provided for any of the electrical system batteries, the Area Radiation Monitors (ARMs), or the Process Radiation Monitors (PRMs).

- Several procedures related to fuel handling were deleted following submittal of the reference materials. At the time of the examinations new fuel handling equipment had recently been installed, but the applicants had not been trained on the new equipment. The licensee did not inform the NRC that the procedures had been deleted or that new fuel handling equipment had been installed. As a result, a question had to be replaced on the written examination and much of the material prepared for the fuel handling portion of the operating tests could not be used.

- Many of the OPM chapters contain out-of-date information. Most of the OPM chapters have not been revised for over two years and, therefore, do not reflect address current plant configuration. No training materials on recent modifications were provided and the NRC was not made aware of any modifications or out-of-date information in the OPMs. Several written examination questions had to be revised following the pre-examination review due to incorrect information in the OPMs.

- No current EOP training materials were provided. All the TCRs associated with the EOPs were Revision 0 and had been prepared between October, 1987 and July, 1988. The EOP bases document that was provided had not been revised since 1987. Most of the EOPs have been revised at least twice since 1988. The license applicants were taught using the requalification training materials which have been revised recently. The EOP training materials for requalification were not provided.

None of the reference material deficiencies were significant by themselves, but the combination of missing and out-of-date information severely hindered the preparation of the examinations. The time required for preparation of the written examinations was significantly increased due to the incomplete and out-of-date materials. The pre-examination review was effective in preventing problems on the written examinations and JPMs that were administered, but the review did not cover the replacement operating test. A number of the deficiencies were not identified until the week prior to the examinations while preparing for the operating tests or during the actual administration of the operating tests. The licensee was very responsive to requests for additional materials, but in many cases the deficiencies were not identified until too close to the examination week to obtain the materials.

Similar problems with incomplete and out-of-date reference materials were encountered during examinations at Oyster Creek in 1987. The deficiencies were corrected for the next few examinations, but the for ensuring that examination reference materials are complete and current appear to have been relaxed at some time prior to the most recent examinations.

4.0 EMERGENCY OPERATING PROCEDURE RELATED FINDINGS

The examiners made several observations related to the EOPs during preparation for and administration of the examinations. These observations were related to the EOP flowcharts, training on the EOPs, and equipment required for implementing the EOPs and related procedures.

- When entry into a contingency EOP (i.e., Emergency Depressurization) is required, the flowcharts do not clearly indicate which EOPs or legs of the EOPs should be exited. Training personnel indicated that the operators are trained to exit the leg of the procedure from which entry into the contingency is directed. Discussions with licensed operators and observations during administration of the examinations indicated that there is some confusion as to how to correctly transition between the EOPs and the contingency procedures.

- Passive steps that require no action (i.e., steps that indicate that Emergency Depressurization is required) are presented in the same format as action steps on the EOP flowcharts. Lack of differentiation between the format of passive and action steps can lead to incorrect transitioning between procedures. If the operator takes action directly based on a passive step, he or she could incorrectly exit the procedure leg that contains the passive step and remain in the procedure leg from which the action is directed. An error of this type was observed during the operating examinations.

- During the administration of the examinations, one of the license applicants and the on-shift Group Shift Supervisor (GSS) were unable to locate the designated table that is used when implementing the EOPs. The following day, a training instructor (a licensed SRO) was also unable to locate the table which was supposed to be located in the control room. Upon investigation, a table was located in a hallway outside of the control room. There were no indications that the table was dedicated for EOP use or that any controls were in place to ensure that the table would be available if needed. There is no desktop area in the control room suitable for use of the flowchart EOPs, therefore, it is important that a table be readily available for use. It is also important that the operating crew can locate the table quickly if an emergency occurs.

- The jumpers that are designated for use when implementing the EOPs are stored with the flowcharts in a locker in the control room. The jumpers are contained in plastic bags with the EOP step number indicated on each bag, but the bags are not organized so that they can be easily locked. An audit of the contents of the locker is performed quarterly in accordance with licensee administrative procedures, but there is no lock on the locker to protect the jumpers from unauthorized access. While there was no indication that any equipment was missing, the controls did not appear to be adequate to ensure that the jumpers would be available and could be easily accessed under emergency conditions.

5.0 EXIT INTERVIEW

An exit meeting was conducted on October 18, 1990, at the site. The licensee representatives that attended the exit meeting are listed in Section 2.0 of this report.

The results of the replacement examinations were not presented at the exit meeting. The Chief Examiner stated that the results would be contained in the Examination Report and that every effort would be made to send the results in approximately 30 working days. The licensee indicated that attempts would be made to ensure that, in the future, emergency drills are not conducted during administration of operating examinations.

Preliminary results of the requalification examinations were provided at the exit meeting. The final results would be provided following NRC management review. The concern related to correction of identified procedural deficiencies (see Section 3.2) was also discussed.

No generic strengths or weaknesses were noted due to the limited number of applicants for the replacement examinations and operators who were administered requalification examinations.

The findings related to the EOPs and the deficiencies with the reference materials (see Sections 3.4 and 4.0) were discussed. The licensee acknowledged the concerns related to the EOPs. Licensee representatives indicated that corrective actions would be taken to prevent recurrence of the problems with the reference materials.

Attachment:
1. Senior Reactor Operator Written Examination and Answer Key

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

FACILITY: Oyster Creek 1

REACTOR TYPE: BWR-GE2

DATE ADMINISTERED: 90/10/16

CANDIDATE: Master

INSTRUCTIONS TO CANDIDATE:

Points for each question are indicated in parentheses after the question. To pass this examination, you must achieve an overall grade of at least 80%. Examination papers will be picked up four and one half (4 1/2) hours after the examination starts.

NUMBER QUESTIONS	TOTAL POINTS	CANDIDATE'S POINTS	CANDIDATE'S OVERALL GRADE (%)
97	100.00		

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

Candidate's Signature

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 candidate 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 facilitate legible reproductions.

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

6. Fill in the date on the cover sheet of the examination (if necessary).

7. You may write your answers on the examination question page or on a separate sheet of paper. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.

8. If you write your answers on the examination question page and you need more space to answer a specific question, use a separate sheet of the paper provided and insert it directly after the specific question. DO NOT WRITE ON THE BACK SIDE OF THE EXAMINATION QUESTION PAGE.

9. Print your name in the upper right-hand corner of the first page of answer sheets whether you use the examination question pages or separate sheets of paper. Initial each of the following answer pages.

10. Before you turn in your examination, consecutively number each answer sheet, including any additional pages inserted when writing your answers on the examination question page.

11. If you are using separate sheets, number each answer and skip at least 3 lines between answers to allow space for grading.

12. Write "Last Page" on the last answer sheet.

13. 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.

14. The point value for each question is indicated in parentheses after the question. The amount of blank space on an examination question page is NOT an indication of the depth of answer required.

15. Show all calculations, methods, or assumptions used to obtain an answer.

16. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK. NOTE: partial credit will NOT be given on multiple choice questions.

17. Proportional grading will be applied. Any additional wrong information that is provided may count against you. For example, if a question is worth one point and asks for four responses, each of which is worth 0.25 points, and you give five responses, each of your responses will be worth 0.20 points. If one of your five responses is incorrect, 0.20 will be deducted and your total credit for that question will be 0.80 instead of 1.00 even though you got the four correct answers.

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

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

20. To pass the examination, you must achieve an overall grade of 80% or greater.

21. There is a time limit of (4 1/2) hours for completion of the examination. (or some other time if less than the full examination is taken.)

22. When you are done and have turned in your examination, leave the examination area as defined by the examiner. 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)

Which one of the following individuals can NOT authorize a temporary change to a procedure's acceptance criteria?

a. Plant Engineering Director

b. Operations Support Engineer

c. Manager Plant Material

d. PRG Chairman

QUESTION: 002 (1.00)

Which one of the following is an accurate statement regarding the use of tags at the Oyster Creek Generating Station?

a. The ORANGE tag is an electrical tag only. It is used when it is desired to test an isolated circuit with an independent source of voltage.

b. The WHITE tag is an information tag that can be used on either electrical or mechanical equipment, but is to be used only to post operational information of a temporary nature.

c. The YELLOW tag is an Electrical Equipment Caution Tag and is to be used as a warning of conditions other than normal.

d. The RED tag is to be used on mechanical equipment such as valves, mechanical controls, etc., where the operation of such equipment would create a condition unsafe to life or property.

QUESTION: 003 (1.00)

Which one of the following is NOT an example of a Temporary Variation, as described by Administrative Procedure 108, "Equipment Control?"

a. A safety ground installed to provide personnel safety while performing maintenance on I&C equipment.

b. A blank flange installed as shown on an approved system drawing.

c. A recorder bypass switch placed in bypass for an individual data point.

d. A hose connected to a pump through a normal service connection to provide a flush path after maintenance.

QUESTION: 004 (1.00)

Which one of the following systems, according to OCNGS Administrative Procedure 108, "Equipment Control," would require an independent verification of the component positions following a surveillance test?

a. Standby Gas Treatment System

b. Reactor Feedwater System

c. Turbine Bypass Valve System

d. 4160 VAC Electrical System

QUESTION: 005 (1.00)

Which one of the following Hot Work Permits would be valid for a maximum of 24 hours?

a. Work in a building containing safety-related equipment during outage conditions.

b. Work in a building containing NO safety-related equipment during outage conditions.

c. Work in a building containing safety-related equipment during plant operations.

d. Work in a building containing NO safety-related equipment during plant operations.

QUESTION: 006 (1.00)

One of the roll-up doors in the 4160 Volt Switchgear Room has been found to be inoperable and is in the open position.

Which one of the following is the correct action to be taken?

a. If the fire detectors in the room are operable, within one hour establish a continuous fire watch at the door.

b. If the fire detectors in the room are inoperable, within one hour establish a continuous fire watch at the door.

c. If the fire detectors in the room are inoperable, within one hour establish an hourly fire watch.

d. If the fire detectors in the room are operable, no action is required.

QUESTION: 007 (1.00)

Which one of the following must be completed by a licensed operator to maintain his/her license in an "active" status per the regulations of 10 CFR 55, "Operators' Licenses"?

The operator shall actively perform the functions of the appropriately licensed operator on a minimum of:

a. seven 8 hour shifts or five 12 hour shifts per calendar month.

b. seven 8 hour shifts or five 12 hour shifts per calendar quarter.

c. five 8 hour shifts or three 12 hour shifts per calendar month.

d. five 8 hour shifts or three 12 hour shifts per calendar quarter.

QUESTION: 008 (1.00)

Which one of the following Emergency Action Levels is described by the summary statement given below?

Events are in process or have occurred which involve an actual or likely major failure of plant functions needed for protection of the public. Any releases are not expected to exceed EPA Protective Action Guideline exposure levels except near Site Boundary.

a. Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

QUESTION: 009 (1.00)

Which one of the following individuals is authorized to approve unescorted access to Vital Areas, including access to the control room?

a. Operations Support Manager

b. Site Protection Lieutenant

c. Security Manager

d. Security Shift Commander

QUESTION: 010 (1.00)

The Hydrogen cooling system for the Main Generator develops a leak and results in an explosion.

Which one of the following is the correct classification of the type of fire described?

a. Class 'A'

b. Class 'B'

c. Class 'C'

d. Class 'D'

QUESTION: 011 (1.00)

Which one of the following is the minimum staffing required by the Technical Specifications, when the reactor is at 35% power and ascending to 100% during a startup?

a. One (1) licensed senior reactor operator in the control room
Two (2) licensed reactor operators, one (1) in the control room, one (1) on site
Two (2) equipment operators on site

b. Two (2) licensed senior reactor operators, one (1) in the control room, one (1) on site
Two (2) licensed reactor operators in the control room
Three (3) equipment operators on site

c. Two (2) licensed senior reactor operators in the control room
Two (2) licensed reactor operators in the control room
Three (3) equipment operators on site

d. One (1) licensed senior reactor operator on site
Two (2) licensed reactor operators, one (1) in the control room, one (1) on site
Two (2) equipment operators on site

QUESTION: 012 (2.00)

During a maintenance outage a 23 year old male worker with a lifetime exposure of 23 REM (NRC Form 4 on file), is assigned to work in a 200 mrem/hr radiation area. The worker has received 250 mrem so far this calendar quarter.

1. Which one of the following is the amount of time the worker can be in the radiation area without exceeding OCNGS administrative whole body occupational exposure limits?

   a. 1.25 hours

   b. 2.5 hours

   c. 3.75 hours

   d. 5.0 hours

2. Which one of the following is the maximum exposure this worker could receive in this calendar quarter (including his current exposure) without exceeding the 10CFR20 allowable whole body exposure limits?

   a. 3000 mrem

   b. 2000 mrem

   c. 1250 mrem

   d. 1000 mrem

QUESTION: 013 (1.00)

In accordance with Station Procedure 205.0, "Reactor Refueling":

Which one of the following individuals is responsible for notifying the bridge operator anytime that the refueling interlock rod block fails to activate with the bridge over the reactor cavity?

a. Group Shift Supervisor

b. Control Room Licensed Operator

c. Shift Technical Advisor

d. Core Engineer

QUESTION: 014 (1.00)

Which one of the following is a correct statement regarding the responsibilities of the Emergency Director (ED) and the Emergency Support Director (ESD)?

According to procedures IMP-1300.02, "Direction of Emergency Response" and IMP-1300.03, "Emergency Notification":

a. The ED is responsible for classifying the event unless overruled by the ESD, if in the ESD's judgement the event should be classified at a higher level.

b. Once an event has been classified by the ED and the NRC has been notified of its classification, NRC permission is required to change the emergency's classification.

c. Even with the ESD function activated, the ED is still responsible for approving and directing information releases to the media.

d. In the event that the NRC and the ED/ESD have differing recommendations for courses of protective action, the Plant Manager will resolve the conflict.

QUESTION: 015 (1.00)

Which one of the following items is the responsibility of the Off-Going Group Shift Supervisor doing shift turnover?

a. Conduct shift briefing with the equipment operators.

b. Preparation of the GSS turnover checklist.

c. Walk-down of control boards.

d. Sign the CRO turnover checklist.

QUESTION: 016 (1.00)

Which one of the following items, according to OCNGS Administrative Procedure 106, is required to be entered into the Group Shift Supervisor's Log?

a. Required operability testing of Technical Specification-related equipment prior to removal from service or return to service

b. Circumstances concerning the inability to perform scheduled surveillance tests

c. Circumstances concerning any violation of the Technical Specifications and operating license

d. Any telephone or radio or communications checks not specifically addressed by the Administrative Procedures

QUESTION: 017 (1.00)

Which one of the following conditions could result in exceeding the transient MCPR limit?

a. Mode Switch in Startup, operating in IRM Range 8, Recirculation Flow at 40 X 10 E6 lbm/hr.

b. Mode Switch in Run, operating in IRM Range 10, Recirculation Flow at 32 X 10 E6 lbm/hr.

c. Mode Switch in Startup, operating in IRM Range 9, Recirculation Flow at 45 X 10 E6 lbm/hr.

d. Mode Switch in Startup, operating in IRM Range 10, Recirculation Flow at 35 X 10 E6 lbm/hr.

QUESTION: 018 (1.00)

Which one of the following signals will result in an ISOLATION of an Isolation Condenser?

a. Steam header flow exceeds 400% for 24 seconds.

b. Condensate return header flow exceeds 250% for 35 seconds.

c. Steam header flow exceeds 350% for 35 seconds.

d. Condensate return header flow exceeds 300% for 15 seconds.

QUESTION: 019 (1.00)

Which one of the following is the reactor water level at which the Isolation Condenser must be isolated?

a. 180" or 15' above TAF

b. 160" or 13'4" above TAF

c. 137" or 11'5" above TAF

d. 89" or 7'2" above TAF

QUESTION: 020 (1.00)

Which one of the following accurately describes the combination of valves that can be operated from the Remote Shutdown Panel?

a. - D.C. Condensate Return from "B" I.C.
   - Condensate Transfer Makeup to "B" I.C. Shell

b. - D.C. Condensate Return from "B" I.C.
   - A.C. Steam Inlet Valve to "B" I.C.

c. - D.C. Steam Inlet Valve to "B" I.C.
   - A.C. Condensate return from "B" I.C.

d. - Condensate Transfer Makeup to "B" I.C. Shell
   - A.C. Condensate return from "B" I.C.

QUESTION: 021 (2.00)

Answer the two following questions regarding the Standby Liquid Control (SLC) System.

1. Which one of the following is the minimum allowed injection time for the SLC System?

   a. 120 min

   b. 60 min

   c. 35 min

   d. 26 min

2. Which one of the following accurately describes the basis for the minimum allowed SLC injection time?

   a. To prevent restart when reactivity is added by void collapse.

   b. To prevent power chugging caused by improper mixing.

   c. To prevent restart when reactivity is added during cooldown following Xenon peak.

   d. To prevent injection line clogging caused by boron solidification.

QUESTION: 022 (1.00)

Which one of the following gives the Tech Spec Bases for the Main Steam Line Low Pressure MSIV Closure?

a. Anticipates the pressure, neutron flux and heat flux increase caused by the rapid closure of the valves and failure of the turbine bypass system to ensure reactor pressure safety limit is not exceeded.

b. With the Isolation at this pressure, the loss of inventory is minimized, therefore the fuel cladding integrity limit is insured.

c. Provided to terminate a steam flow increase transient so that the maximum steam flow obtained corresponds to the Kf curve maximum flow limit.

d. Protection against fast reactor depressurization and the resulting rapid cooldown of the vessel to ensure the fuel cladding integrity safety limit is not exceeded.

QUESTION: 023 (1.00)

Which one of the following accurately describes the EFFECT on the RPS system if the Mode Switch is in STARTUP, reactor pressure is at 700 psig and an IRM switch is placed in range ten (10)?

a. Main Steam Line Isolation will occur, but the reactor will NOT scram because reactor pressure is below 825 psig.

b. Main Steam Line Isolation will NOT occur, but the reactor will scram because reactor pressure is below 825 psig.

c. Main Steam Line Isolation will occur and the reactor will scram because reactor pressure is below 825 psig.

d. Main Steam Line Isolation will NOT occur and the reactor will NOT scram because reactor pressure is below 825 psig.

QUESTION: 024 (2.00)

For each of the Reactor Level instruments listed in Column 'A', MATCH the type of compensation that instrument has from Column 'B'.
The items in Column 'B' may be used once, more than once, or not at all.

COLUMN 'A' INSTRUMENTS	COLUMN 'B' COMPENSATION
a. Wide Range GE-MAC.	1. No Compensation
b. Narrow Range GE-MAC	2. Density compensated using reactor pressure.
c. Lo-Level Yarway	3. Compensated using heated reference leg.
d. Lo-Lo- Level Yarway	4. Density compensated using MSL pressure.

QUESTION: 025 (1.00)

Which one of the following level instruments provides the signal to the Automatic Depressurization System logic?

a. Narrow Range GE-MAC

b. Lo Level Yarway

c. Lo Lo Level Yarway

d. Lo Lo Lo Barton Indicating Switch

QUESTION: 026 (1.00)

ADS has initiated (EMRVs OPEN). Which one of the following actions alone will close the EMRVs?

a. Resetting the Drywell Pressure signal

b. Resetting the 120 Second ADS Timer

c. Tripping the Core Spray Booster Pumps

d. Taking the individual EMRV switches to OFF

QUESTION: 027 (1.00)

Which one of the following is the correct sequence of EMRV operating when manually controlling reactor pressure?

a. D, C, B, A, E

b. C, B, D, E, A

c. A, D, B, C, E

d. B, A, E, D, C

QUESTION: 028 (1.00)

The Containment Spray System has received an automatic start signal. Which one of the following accurately describes the proper start sequence?

40 seconds after the initiation signal:

a. containment spray pumps 'A' and 'C' start, 20 sec. after sensing 1500 pgm flow to the drywell the 5% valve opens, ESW pumps 'A' and 'C' start 45 seconds after associated containment spray pumps start.

b. containment spray pumps 'B' and 'D' start, 20 sec. after sensing 1500 pgm flow to the drywell the 5% valve opens, ESW pumps 'B' and 'D' start 45 seconds after associated containment spray pumps start.

c. containment spray pumps 'A' and 'C' start, 20 sec. after sensing 1500 pgm flow to the drywell the ESW pumps 'A' and 'C' start, 5% valve opens 45 seconds after associated containment spray pumps start.

d. containment spray pumps 'B' and 'D' start, 20 sec. after sensing 1500 pgm flow to the drywell the ESW pumps 'B' and 'D' start, 5% valve opens 45 seconds after associated containment spray pumps start.

QUESTION: 029 (1.00)

While at 100% power, an operator holds the control for Main Stop valve #2 Internal Bypass in the Fast Lower position.

Which of the following best describes how the Turbine Stop Valves will respond?

a. #2 internal bypass and the four main stop valves close.

b. #2 internal bypass and #2 main stop valve close. #1, #3 and #4 main stop valves remain open.

c. #2 internal bypass opens. The four main stop valves remain open.

d. #2 internal bypass opens. The #2 main stop valve closes. #1, #3 and #4 main stop valves remain open.

QUESTION: 030 (1.00)

Which one of the following turbine control devices is normally used to control the position of the turbine control valves during paralleling and picking up initial turbine load?

a. EPR

b. Load Limiter

c. Speed load changer

d. MPR

QUESTION: 031 (1.00)

A plant startup is proceeding and the MPR is set at 150 psig. Assume no operator actions take place as reactor pressure increases from 140 psig to 160 psig.

Which one of the following describes the expected response?

a. Turbine bypass valves start to open.

b. Turbine bypass valves and control intercept valves start to open.

c. Turbine control valves and control intercept valves start to open.

d. Turbine control valves and bypass valves start to open.

QUESTION: 032 (1.00)

The plant is at 90% power and the 'A' transmitter is selected for input to the Feedwater Control System. Which one of the following component failures would result in an actual Reactor Vessel water level increase?

a. Feed flow signal fails upscale

b. Steam flow signal fails upscale

c. 'A' level transmitter fails upscale

d. Feedwater temperature signal fails downscale

QUESTION: 033 (1.00)

Which one of the following correctly indicates when the SRMs should be fully withdrawn from the core during Reactor Startup?

a. IRM switches on range 5

b. IRM switches on range 6

c. IRM switches on range 7

d. IRM switches on range 8

QUESTION: 034 (1.00)

General Operating Procedure 201.2 "Plant Heatup to Hot Standby" requires removal of the RWCU Auxiliary Cleanup Pump from service when reactor pressure reaches approximately 100 psig.

Which one of the following describes the expected response of the Reactor Water Cleanup System if reactor pressure increases to 130 psig without removing the Auxiliary Cleanup Pump from service?

a. The Auxiliary Cleanup Pump would experience thrust bearing and mechanical seal damage.

b. The Auxiliary Cleanup Pump would continue to operate normally.

c. The Reactor Water Cleanup System would auto transfer to the cleanup recirc pumps.

d. The Auxiliary Cleanup Pump would automatically trip.

QUESTION: 035 (1.00)

During Reactor operations at 100% power, a Containment High-Range Radiation Monitor is found to be inoperable.

Which one of the following is the required actions that must be taken?

a. No action required, minimum number of instruments and trip systems are not exceeded.

b. Restore the monitor to an operable condition within 7 days. Submit a special report to the NRC within 14 days following the event.

c. Isolate vent and purge pathways or place the reactor in a cold shutdown condition.

d. Commence a reactor shutdown and place the reactor in a cold shutdown condition within 30 hours.

QUESTION: 036 (1.00)

With the plant at full power , the alarm "HOTWELL CONDUCT HI" is received and the operators confirm that the Condenser 1B N Hotwell cell indicates 10 micro mhos/cm.

Which one of the following actions should the operators take?

a. Raise the affected condenser level and isolate the affected condenser.

b. Bypass the alarm on the affected cell and monitor the reading locally.

c. Scram the reactor and close the MSIVs.

d. Commence a normal reactor shutdown within 4 hours.

QUESTION: 037 (1.00)

Which one of the following statements correctly describes the operation of the Reactor Feedwater Pump Runout Protection Circuitry?

a. The flow control valve locks up at 2.67 x 10 E6 lbs/hour feedwater flow and can only be manually reset when the high flow signal clears.

b. The flow control valve locks up at 90% of rated reactor feed pump current and will automatically reset on a Reactor Water Level 175 inches above TAF increasing.

c. The flow control valve locks up at 2.67 x 10 E6 lbs/hour feedwater flow and will reset automatically when the high flow signal clears.

d. The flow control valve locks up at 90% of rated reactor feed pump current and can only be manually reset when Reactor Water Level is 175 inches above TAF and increasing.

QUESTION: 038 (1.00)

Which one of the following signals, when received, would result in an automatic start of the Standby Gas Treatment System?

a. Radiation in the Reactor Building Ventilation Exhaust of 15 mr/hr.

b. Radiation in the Reactor Building Operating Floor Elev. 119' of 55 mr/hr for 3 minutes.

c. Drywell pressure of 2.25 psig.

d. Reactor Water Level of 115" above TAF.

QUESTION: 039 (1.00)

A plant transient caused automatic actuation of the Alternate Rod Injection (ARI) System. Which one of the following describes the procedure for resetting the ARI actuation?

a. ARI is automatically reset when the Reactor Protection System (RPS) is reset.

b. No operator action is required, ARI automatically resets when initiating conditions clear.

c. ARI can be manually reset following a 45 second time delay by depressing the reset push buttons.

d. No operator action is required, ARI automatically resets following a 45 second time delay.

QUESTION: 040 (1.00)

On panel LSP-DG2 the Alternate Mode Selector switch is placed in DEADLINE, then the Control Transfer switches are sequentially transferred to ALTERNATE. The DG2 Alternate Emergency Start Switch is momentarily placed in START.

Which one of the following accurately describes the expected response of the Emergency Diesel Generator #2?

a. DG2 will start, idle for 90 seconds at 450 rpm, accelerate to rated speed, synchronize to the bus and pick up rated load.

b. DG2 will start, accelerate to rated speed and the output breaker will close in approximately 15 seconds.

c. DG2 will start, idle at 450 rpm until manually put into "RUN", accelerate to rated speed in 90 seconds and synchronize to the bus.

d. DG2 will start, accelerate to rated speed, but will NOT synchronize to the bus.

QUESTION: 041 (1.00)

Which one of the following Emergency Diesel Generators protective circuitry trips is bypassed when a fast start signal is received?

a. Generator Breaker trip

b. Positive Crankcase Pressure

c. Sequence Fault

d. Engine Overspeed

QUESTION: 042 (1.00)

The APRM flow converters have a 16,000 gpm mismatch trip unit circuit associated with them. Which one of the following describes when this mismatch will occur?

a. When total recirc flow measured by one flow converter differs from that of the other flow converter by more than 16,000 gpm.

b. When one recirc loop flow differs from another recirc loop flow by more than 16,000 gpm.

c. When total recirc flow is 16,000 gpm above the flow biased APRM trip setpoint.

d. When one recirc loop flow differs from the average recirc loop flow by more than 16,000 gpm.

QUESTION: 043 (1.00)

The reactor is operating at 75% power and a Core Spray Pump Operability Test is in progress in accordance with surveillance procedure 610.4.002. The Core Spray Pump has been running for 15 minutes, when a spurious High Drywell Pressure initiation signal is received.

Which one of the following correctly describes the expected response of the Core Spray System?

a. The Parallel Isolation Valves will open
   The Discharge Valves will close
   The Test to Torus Valves will remain open

b. The Parallel Isolation Valves will remain closed
   The Discharge Valves will open
   The Test to Torus Valves will close

c. The Parallel Isolation Valves will open
   The Discharge Valves will remain closed
   The Test to Torus Valves will remain open

d. The Parallel Isolation Valves will remain closed
   The Discharge Valves will remain closed
   The Test to Torus Valves will close

QUESTION: 044 (1.00)

Which one of the following is an accurate precaution or limitation for the recirculation pumps?

a. To insure minimum vibration, individual pump speeds will not exceed 30 cps, until at least four (4) pumps are in service and the fifth discharge valve is shut.

b. Do not operate recirculating pumps continuously above 36 cps until system temperature is in the normal operating range.

c. Recirculating pumps should not be operated continuously at full speed (57.5 cps) with the plant at hot standby with one (1) or more recirculating loops shut down.

d. With system temperature in the normal operating range, recirculating pumps may not be operated above 56 cps with drywell temperature above 130 F.

QUESTION: 045 (1.00)

Which one of the following supplies the signals to the Rod Worth Minimizer to determine when the Low Power Setpoint and Low Power Alarm Point are reached?

a. First Stage Main Turbine Pressure signal

b. Average APRM signal from RPS

c. Main Steam System total flow signals

d. Average LPRM signal from 'A' or 'B' APRM

QUESTION: 046 (1.00)

Which one of the following will NOT initiate a Rod Worth Minimizer rod withdrawal block?

a. The operator has selected a rod differing from any of the three insert error rods.

b. A withdrawal error has been produced by the operator while inserting rods.

c. A withdrawal error exists upon system initialization.

d. A fault is detected in the rod block relay hardware.

QUESTION: 047 (1.00)

During a core fuel offload, jumpers are used to defeat the one rod free movement permissive interlock for control rods in empty fuel cell locations.

Which one of the following combinations describes the rod position indication backlighting that will be available to warn the operator, in the control room, that a jumper is installed?

a. amber - red

b. red - green

c. green - white

d. white - amber

QUESTION: 048 (1.00)

The reactor mode switch is in REFUEL and all rods are fully inserted. The CRO selects rod 34-51 in preparation for taking it to the full out position (position 48). When the rod is at position 24, the Rod Select Power Switch is inadvertently placed to OFF, then switched back to ON.

Which one of the following describes the actions that must be performed in order to continue withdrawing rod 34-51 to position 48?

a. Continue with the operation, using single or continuous notch withdrawal for rod 34-51.

b. Drive rod 34-51 in two (2) notches, then continue withdrawing rod 34-51.

c. Push the select pushbutton for rod 34-51, then continue withdrawing rod 34-51.

d. Push the select pushbutton for rod 34-51, fully insert rod 34-51, then withdraw rod 34-51.

QUESTION: 049 (1.00)

Which one of the following is the correct power supply for each of the Reactor Feed Water Pumps?

a. Reactor Feed Pump 1A feed from 4160 Bus 1A
Reactor Feed Pump 1B feed from 4160 Bus 1B
Reactor Feed Pump 1C feed from 4160 Bus 1C

b. Reactor Feed Pump 1A feed from 4160 Bus 1A
Reactor Feed Pump 1B feed from 4160 Bus 1B
Reactor Feed Pump 1C feed from 4160 Bus 1B

c. Reactor Feed Pump 1A feed from 4160 Bus 1B
Reactor Feed Pump 1B feed from 4160 Bus 1A
Reactor Feed Pump 1C feed from 4160 Bus 1A

d. Reactor Feed Pump 1A feed from 4160 Bus 1A
Reactor Feed Pump 1B feed from 4160 Bus 1B
Reactor Feed Pump 1C feed from 4160 Bus 1A

QUESTION: 050 (1.00)

Which one of the following is a accurate statement regarding the 125 VDC system?

a. Bus 'A' supplies train 'B' safety-related loads
Bus 'B' supplies train 'A' safety-related loads
Bus 'C' supplies Non-safety-related loads

b. Bus 'A' supplies Non-safety-related loads
Bus 'B' supplies train 'B' safety-related loads
Bus 'C' supplies train 'A' safety-related loads

c. Bus 'A' supplies train 'A' safety-related loads
Bus 'B' supplies Non-safety-related loads
Bus 'C' supplies train 'B' safety-related loads

d. Bus 'A' supplies train 'A' safety-related loads
Bus 'B' supplies train 'B' safety-related loads
Bus 'C' supplies Non-safety-related loads

QUESTION: 051 (1.00)

Which one of the following conditions will result in an automatic isolation of the Main Steam Isolation Valves?

a. Reactor water level at 76" above T.A.F.

b. Steam line radiation at ten (10) times background at 50% rated power.

c. Steam line flow rate at 110% of rated single-line flow.

d. Main steam tunnel temperature at 45 degrees F above ambient temperature at power.

QUESTION: 052 (1.00)

Which one of the following valves/components will fail CLOSED or to MINIMUM on a complete loss of Instrument/Service Air?

a. CRD Flow Control Valves

b. Shutdown Cooling Recirc Valves

c. Feedwater Regulating Valves

d. Recirc M/G set

QUESTION: 053 (1.00)

The Normal system lineup exists and the primary, secondary, and alternate power supplies are available to the Continuous Instrument Panel 3 (CIP-3).

Which one of the following describes the expected system configuration if the secondary source [125 VDC Distr Center 'B'] of power is lost to the rotary invertor supplying CIP-3?

a. Power to CIP-3 continues to be supplied from its normal, primary source VMCC - 1B2 via the rotary invertor.

b. Power to CIP-3 automatically transfers to VMCC 1A2 via transformer IT-3.

c. Power to CIP-3 continues to be supplied from its normal, primary source VMCC - 1A2 via the rotary invertor.

d. Power to CIP-3 automatically transfers to VMCC 1B2 via transformer IT-3.

QUESTION: 054 (1.00)

Which one of the following correctly describes the response of the Reactor Building Heating and Ventilation (RBHV) system when the loss of vital AC to the RBHV system occurs?

RBHV system fans shutdown and the:

a. RBHV intake and exhaust valves isolate. SGTS will NOT automatically start.

b. RBHV intake and exhaust valves remain open. SGTS automatically starts.

c. RBHV intake and exhaust valves remain open. SGTS will NOT automatically start.

d. RBHV intake and exhaust valves isolate. SGTS automatically starts.

QUESTION: 055 (1.00)

While operating at 60% reactor power, 5 recirc loop operation, one of the Recirculation Pumps trip resulting in total core flow of 40%. [Power to flow map is attached.]

Which one of the following describes the immediate actions required?

a. Initiate an manual reactor scram.

b. Insert the CRAM array to reduce reactor power below the 80% rod line.

c. Increase recirculation flow with remaining pumps, to return to desired area of map.

d. Take no immediate actions and allow power to stabilize and check for power oscillation.

QUESTION: 056 (1.00)

While operating at 87% power a RBCCW pump trips followed by CCW FLOW LO alarms on 'B' & 'E' recirculation pumps.

Which one of the following describes the required action to be taken by the Operator?

Scram the reactor in accordance with ABN-3200.01, "Reactor Scram", then:

a. Trip ALL operating recirculation pumps, and confirm that ALL recirculation pump suction and discharge valves are OPEN.

b. Trip the AFFECTED recirculation pumps, and IDLE the effected recirculation loops.

c. Trip the AFFECTED recirculation pumps, and ISOLATE the effected recirculation loops.

d. Trip ALL operating recirculation pumps, and SHUT all but ONE set of recirculation pump suction and discharge valves.

QUESTION: 057 (1.00)

The plant is operating at 100% power with Turbine Building Closed Cooling Water (TBCCW) pump 1 out of service for maintenance, when TBCCW pump 3 trips. Which one of the following actions is required in accordance with ABN-3200.20 "TBCCW Failure Response?" ("DISCH PRESS LO" alarm lit)

a. Scram the Reactor, trip all operating recirculation pumps, and be in cold shutdown within 24 hours.

b. Close the TBCCW heat exchanger bypass valve and lineup the Fire Protection System to the station air compressors.

c. Reduce reactor power by reducing recirculation flow when Recirculation MG Set temperatures exceed the specified limits.

d. Monitor all equipment cooled by TBCCW and increase Service Water Flow to the TBCCW heat exchangers.

QUESTION: 058 (1.00)

While operating at 95% power the following alarms are received.

- COND VAC LO/TURB TRIP I [Ann Window J-1-c]
- COND VAC LO/TURB TRIP II [Ann Window J-2-c]
- COND VAC LOW 25 INCHES [Ann Window Q-3-c]
- COND VAC TRIP 1 22 INCHES [Ann Window Q-2-c]

Which one of the following correctly describes the automatic functions that should have occurred?

a. No automatic actions should have occurred at this time.

b. Reactor Scram only

c. Reactor Scram and Main Turbine trip only

d. Reactor Scram, Main Turbine trip, and Bypass valves interlocked closed.

QUESTION: 059 (1.00)

Following a complete loss of 125 VDC, a reactor limiting safety system setpoint has been exceeded and fuel integrity is threatened.

Which one of the following describes the required actions in accordance with 2000-ABN-3200.13?

a. Start both Emergency Diesel Generators, parallel and load the diesels to their respective buses.

b. Place the Isolation Condensers in standby readiness with the DC valves manually opened.

c. Maintain the plant power output constant irrespective of Technical Specification requirements.

d. Open the Isolation condenser AC inlet and outlet valves, then scram the reactor.

QUESTION: 060 (1.00)

While operating at 75% reactor power, the Torus water level is accidentally allowed to increase to 156".

Which one of the following choices describes the correct actions to be taken?

a. Enter EMG-3200.02 "Primary Containment Control", and commence lowering Torus water level, and commence a normal reactor shutdown to be in Cold Shutdown within 24 hours.

b. Enter normal operating procedures for lowering Torus level, and commence lowering the water level, and maintain steady state reactor power conditions.

c. Enter normal operating procedures for lowering Torus level, and commence lowering the water level, and commence a normal reactor shutdown to cold shutdown within 30 hours.

d. Enter EMG-3200.02 "Primary Containment Control", and commence lowering Torus water level, and maintain steady state reactor power conditions.

QUESTION: 061 (1.00)

Which one of the following RPS Scram signals is provided to anticipate the pressure and neutron flux increases caused by the rapid closure of the turbine stop valve(s) and failure of the turbine bypass system?

a. High Pressure Scram

b. MSIV Closure Scram

c. APRM High Flux Scram

d. Generator Load Rejection Scram

QUESTION: 062 (1.00)

EMG-3200.08 "RPV Flooding" step FLD-2.1, in the ATWS Path, asks if at least 3 EMRVs can be opened. If no, then the procedure skips the step of closing the MSIV's and IC isolation valves.

Which one of the following correctly describes the bases for leaving the MSIVs and the Isolation Condenser valves open if 3 EMRVs can NOT be opened?

a. Allows a flow path for steam to the main condenser for condensing and injecting back into the RPV.

b. Prevents pressure build up in the RPV during flooding to help minimize the possibility of the leak size increasing.

c. Provides an additional depressurization path to allow for the injection of low pressure injection systems.

d. Allows a flow path for the coolant to prevent the torus from over filling and negating the requirement for containment flooding.

QUESTION: 063 (1.00)

Which one of the following conditions would require entry into EMG-3200.11 "Secondary Containment Control"?

a. Equipment Hatch Area radiation level of 50 mr/hr.

b. Floor drain sump 1-7 water level at -22 ft. 3 in.

c. A reactor Building Ventilation Exhaust radiation level at 11 mr/hr.

d. Secondary Containment differential Pressure at 1 in of water.

QUESTION: 064 (1.00)

Which one of the following sets of parameters would indicate a Fire and NOT a primary leak into one of the secondary containment areas?

a. 51' 3" S/D HX Areas (C-4) radiation level is 240 mr/hr.
51' Shutdown cooling (IB06E-J) temperature is 220 degrees F.

b. 23' 6" CRD Modules Area (C-7) radiation level is 20 mr/hr.
Trunnion Room (IB13A-D) temperature is 220 degrees F.

c. 51' 3" Cleanup Pumps (C-1) radiation level is 90 mr/hr.
51' Cleanup HX Room (IB06-13/14) temperature is 220 degrees F.

d. 95' 3" Iso Cond Area (C-3) radiation level is 400 mr/hr.
95' Iso Condensers (IB06A-D) temperature is 220 degrees F

QUESTION: 065 (1.00)

EMG-3200.11 'Secondary Containment Control Step SC/T-2 states "WHEN Any area temperature exceeds its maximum NORMAL operating temperature listed in Table 11 THEN Isolate all systems that are discharging into the area except..."

Which one of the following systems would be isolated when the above step is performed?

a. Systems required to shut down the reactor

b. Systems required to assure adequate core cooling

c. Systems required to reduce area temperatures are maintained

d. Systems required to suppress a fire

QUESTION: 066 (1.00)

An accident has occurred on the refueling floor. An electrical short on the bridge crane has resulted in a dropped fuel bundle. An operator has been injured and is unconscious. Radiation levels in the area of the injured man are 3400 mr/hr.

Which one of the following is the stay time for the rescue team?

a. 22 minutes

b. 7.3 minutes

c. 22 hours

d. 7.3 hours

QUESTION: 067 (1.00)

The reactor is shutdown at 180 degrees F with all recirculation pumps tripped, when shutdown cooling is lost and cannot be reestablished.

Which one of the following actions should the operator take to establish or enhance natural circulation?

a. Maintain reactor water level at its normal level and open at least three (3) EMRVs.

b. Raise reactor water level to at least 180" above TAF and open Isolation Condenser Vents.

c. Lower reactor water level to 137" above TAF and open one (1) EMRV.

d. Raise reactor water level to at least 185" above TAF and ensure at least one recirculation loop is unisolated.

QUESTION: 068 (1.00)

While at power an inadvertent primary containment isolation occurs because of a false High Drywell Pressure signal.

Which one of the following systems would still be in operation following the isolation?

a. Reactor Building Heating and Ventilation System

b. Reactor Water Cleanup System

c. Reactor Building Closed Cooling Water System

d. Containment Atmosphere Control System.

QUESTION: 069 (1.00)

The following plant conditions exist:

-- A Loss of Coolant Accident (LOCA) has occurred.
-- All ECCS systems are functioning normally.
-- 4160 V bus 1~~A~~C has lost power.
-- Emergency Diesel Generator (EDG) #1 has started, the output breaker has shut and the bus is reenergized.
-- All normal loads have auto sequenced onto the bus.
-- Turbine Building and Reactor Building loads have been manually restarted.
-- 4160 V bus 1~~A~~C steady state load is 2975 KW.

In accordance with OCNGS 341, "Emergency Diesel Generator Operation", which one of the following actions is required to be taken?

a. Closely monitor diesel generator operation for the duration of the emergency.

b. Immediately reduce load to less than 2900 KW by manual load shedding of unnecessary or non-vital loads.

c. Load must be reduced to less than 2850 KW within 2 hours or transered to Manual Parallel operation.

d. Load reduction to less than 2800 KW is required immediately by manual trip of Turbine Building, or other non-vital, loads.

QUESTION: 070 (1.00)

A loss of power to the EPR system occurs while the reactor is operating at 75% power. Which one of the following accurately describes the initial effect on reactor pressure and the cause?

a. Reactor pressure increases, due to the control valves going SHUT in response to the MPR becoming the controlling signal.

b. Reactor pressure decreases, due to the control valves going OPEN in response to the MPR becoming the controlling signal.

c. Reactor pressure increases, due to the control valves going OPEN in response to the EPR maintaining the controlling signal.

d. Reactor pressure decreases, due to the control valves going SHUT in response to the EPR maintaining the controlling signal.

QUESTION: 071 (1.00)

A automatic reactor scram has occurred, 2 control rods did not insert and are at position 48.

Which one of the following methods of control rod insertion would be the most likely to be affective for inserting these rods?

a. Deenergize the scram solenoids by manually opening the RPS breakers.

b. Vent the scram air header by manually opening the air header vent valves.

c. Place the Mode switch to REFUEL and drive the control rods into the core.

d. Reset the scram and manually insert a scram by depressing the scram buttons.

QUESTION: 072 (1.00)

An ATWS condition exists. It has NOT been determined that the reactor will remain shutdown under all conditions without boron. EMG-3200.09 has been entered.

Step LP-1 of EMG-3200.09, "Level/Power Control", requires placing the ADS Timer Switch in "Bypass". Which one of the following is the justification for this action?

a. Uncontrolled depressurization could cause large RPV level perturbations at a point when level is required to be held constant.

b. Depressurization could move the RPV Saturation Temperature Curve into the Unsafe Region affecting the usability of RPV water level instrumentation.

c. Once depressurized below the shutoff head of the Core Spray system, the relatively cold water may cause large positive reactivity additions.

d. Uncontrolled depressurization could cause a large loss of RPV inventory potentially lowering water level below the top of active fuel (TAF).

QUESTION: 073 (1.00)

The plant is operating at 100% power when a fire occurs that is directly affecting operation from the Control Room. The Shift Supervisor has entered OCNGS ABN-3200.30, "Control Room Evacuation", and has determined that time and plant conditions do not permit completion of all operator actions prior to leaving the Control Room.

Based on the above conditions, which one of the following operator actions can be deferred until after the Control Room is evacuated?

a. Trip all five (5) Reactor Recirculation Pumps

b. Initiate the "B" Isolation Condenser

c. Trip all three (3) Reactor Feedwater Pumps

d. Close the Main Steam Isolation Valves

QUESTION: 074 (1.00)

Surveillance Procedure 602.4.003, "Electromatic Relief Valve Operability Test", is in progress. Suppression pool temperature is 100 degrees F and increasing. Which one of the following is the set of parameters that correctly answers the following?

-- The EMRV surveillance must be stopped prior to exceeding a Suppression Pool temperature of ____________ degrees F.

-- Suppression Pool temperature must be reduced below the Normal Power Operation limit within ____________ hours.

-- The Normal Power Operation limit is ____________ degrees F.

a. 110 F, 12 hrs, 95 F

b. 110 F, 24 hrs, 90 F

c. 105 F, 24 hrs, 95 F

d. 105 F, 12 hrs, 90 F

QUESTION: 075 (1.00)

Using the attached Figure 'F' from EMG-3200.02 "Primary Containment Control" determine which one of the following set of parameters would require an Emergency RPV Depressurization.

a. - Torus Temperature 165 degrees F
   - Torus Water Level 135 inches
   - RPV Pressure 950 psig

b. - Torus Temperature 175 degrees F
   - Torus Water Level 145 inches
   - RPV Pressure 350 psig

c. - Torus Temperature 155 degrees F
   - Torus Water Level 115 inches
   - RPV Pressure 850 psig

d. - Torus Temperature 170 degrees F
   - Torus Water Level 125 inches
   - RPV Pressure 250 psig

QUESTION: 076 (1.00)

EMG-3200.04, "Emergency RPV Depressurization", directs the operator to close all EMRVs if Torus water level drops below 90 inches. Which one of the following is the reason for this requirement?

a. Torus temperature detectors are uncovered resulting in inaccurate temperature indication.

b. There is insufficient inventory in the Torus to prevent exceeding the heat capacity temperature limit on a blowdown.

c. The downcomers are NOT sufficiently covered causing excessive drag on the components.

d. The Y-quenchers are not sufficiently covered, which could result in pressurization of the torus on a blowdown.

QUESTION: 077 (1.00)

Oyster Creek Technical Specification 3.5.A states the minimum water volume in the Suppression Pool shall not be less than 82,000 cu. ft. Which one of the following is the bases for this specification?

a. To prevent reaching ECCS Net Positive Suction Head (NPSH) limits at maximum design operating temperature.

b. To prevent Core Spray and Containment Spray corner room ambient temperatures from affecting motor temperature limits.

c. To prevent exceeding Drywell or Torus design pressures during a reactor blowdown from 1060 psig.

d. To prevent reaching Core Spray Vortex limits during ECCS maximum design flow rates.

QUESTION: 078 (1.00)

Which one of the following sets of conditions describes a situation in which adequate core cooling is assured?

a. Core Spray System is in operation and maintaining reactor water level at -20 inches while implementing Containment Flooding.

b. Feedwater System is in operation and maintaining reactor water level at -5 inches while implementing Level Restoration.

c. Level is being maintained at -25 inches while implementing Level/Power control.

d. Level is being maintained at -45 inches with one EMRV open while implementing Steam Cooling.

QUESTION: 079 (1.00)

While operating at 75% reactor power, a loss of feedwater heating occurs. The Shift Supervisor enters ABN-3200.16, "Loss of FW Heaters", and directs a power reduction. Which one of the following conditions REQUIRES an immediate reactor scram?

a. APRM channel 2 oscillating from 61% to 65% with a 4 second period
APRM channel 3 oscillating from 62% to 66% with a 4 second period
APRM channel 4 oscillating from 60% to 64% with a 3 second period

b. APRM channel 6 oscillating from 60% to 68% with a 2 second period
APRM channel 7 oscillating from 61% to 64% with a 3 second period
APRM channel 8 oscillating from 58% to 66% with a 4 second period

c. APRM channel 3 oscillating from 59% to 65% with a 4 second period
APRM channel 4 oscillating from 60% to 64% with a 3 second period
APRM channel 5 oscillating from 61% to 68% with a 2 second period

d. APRM channel 6 oscillating from 62% to 68% with a 2 second period
APRM channel 7 oscillating from 60% to 65% with a 3 second period
APRM channel 8 oscillating from 63% to 65% with a 2 second period

QUESTION: 080 (1.00)

Which one of the following thermal limits could be violated if reactor power oscillations occur?

a. Minimum Critical Power Ratio (MCPR)

b. Linear Heat Generation Rate (LHGR)

c. Average Planar Linear Heat Generation Rate (APLHGR)

d. Maximum Fraction of Limiting Power Density (MFLPD)

QUESTION: 081 (1.00)

The plant is in the Startup Mode and rods are being pulled to achieve criticality. The reactor goes critical prior to the 1% ECP margin. Which one of the following actions must be performed?

a. Stop pulling control rods and maintain reactor power constant until the ECP is recalculated.

b. Insert control rods in reverse order and bring the reactor subcritical until the reason for the error in the ECP is determined.

c. Scram the reactor and commence a cooldown, then determine the reason for the error in the ECP.

d. Continue with the reactor startup with a second CRO at the controls to verify rod pattern while the ECP is recalculated.

QUESTION: 082 (1.00)

Refueling Operations are in progress. The reactor mode switch is locked in the "REFUEL" position. The core is subcritical by 0.5% delta k. An evaluation has been performed to ensure that actual core criticality for the planned evolutions is bounded by previous analysis. All refueling interlocks are operable unless otherwise stated below.

All control rods are fully inserted unless otherwise stated below.

Which one of the following planned refueling operations can be performed without violating Technical Specifications?

a. Loading of fuel into reactor core cell 10-11 using the fuel grapple hoist with the load limit switch set at 450 pounds. SRMs 23 and 24 are inoperable. Control rod 22-51 is withdrawn to notch 02 for testing.

b. Removal of control rod 42-47 using the service platform hoist with the load limit switch set at 400 pounds. SRMs 22 and 23 are inoperable. Control rod 42-11 is withdrawn to notch 12 for testing. Refuel interlocks for rod 42-47 are bypassed and the fuel assemblies are removed from core cell 42-47.

c. Removal of the control rod drive mechanism for control rod 38-19 using the frame mounted auxiliary hoist with the load limit switch set at 375 pounds. SRMs 21, 22, and 23 are inoperable. Control rod 50-19 is withdrawn to notch 02 for testing.

d. Removal of control rod 34-47 using the trolley mounted auxiliary hoist with the load limit switch set at 475 pounds. SRM 24 is inoperable. Control rod 10-15 is removed. Refuel interlocks for rods 34-47 and 10-15 are bypassed. The fuel assemblies are removed from core cells 34-47 and 10-15.

QUESTION: 083 (1.00)

Which one of the following statements is CORRECT concerning the Minimum RPV Flooding Pressure of 83 psig above Torus pressure referred to in step FLD-4 of EMG-3200.08, "RPV Flooding"?

a. Once RPV pressure decreases below the Minimum RPV Flooding Pressure, adequate core cooling is assured.

b. Once RPV pressure decreases below the Minimum RPV Flooding Pressure, steam flow through the core does not provide adequate core cooling.

c. If there are no EMRVs open and pressure remains above the Minimum RPV Flooding Pressure, sufficient natural circulation flow through the core exists to provide adequate core cooling.

d. If 3 EMRVs are open and pressure remains above the Minimum RPV Flooding Pressure, insufficient natural circulation to provide adequate core cooling exists and injection rate must be increased.

QUESTION: 084 (1.00)

While implementing EMG-3200.02 "Primary Containment Control," Emergency RPV Depressurization is required if plant conditions cannot be maintained below the Pressure Suppression Pressure limit.

Which one of the following combination of parameters defines the Pressure Suppression Pressure limit?

a. Torus temperature and RPV pressure

b. Torus pressure and Torus water level

c. Drywell temperature and Drywell pressure

d. Torus water level and RPV pressure

QUESTION: 085 (1.00)

EMG-3200.12, "Radioactivity Release Control", shall be entered whenever the off-site radioactivity release rate requires the declaration of which one of the following emergency classifications?

a. Unusual Event

b. Alert

c. Site Emergency

d. General Emergency

QUESTION: 086 (1.00)

Step RR-1 of EMG-3200.12, "Radioactivity Release Control", directs the operator to isolate specific primary systems with the exception of those systems required to shutdown the reactor. Which one of the following is the reason why these systems are exempted from isolation?

a. Additional off-site releases from these systems are highly unlikely.

b. Loss of hot shutdown equipment is a Site Area Emergency by IMP-1300.01, "Classification of Emergency Conditions".

c. Isolation of these systems may result in much larger releases as the transient continues.

d. These systems are required to assure adequate core cooling if Emergency RPV Depressurization becomes necessary.

QUESTION: 087 (1.00)

While operating in EMG-3200.02, "Primary Containment Control", Step PC/P-3 directs the operator to: "Irrespective of the off-site radioactivity release rate, vent the primary containment...". Which one of the following is the bases for this action?

a. The Torus is in danger of exceeding the Torus load limit.

b. The Drywell has exceeded the limitations of the Containment Spray Initiation Limit Curve.

c. The limitations of the Heat Capacity Temperature Limit curve have been exceeded preventing depressurization to the Torus.

d. The containment is in danger of exceeding all safety margins and the pressure yield point.

QUESTION: 088 (1.00)

EMG-3200.01, "RPV Control", Step R/P-1 requires manual opening of EMRVs if "any EMRV is cycling". Which one of the following is the justification for this action?

a. Stabilizing RPV pressure by manually opening EMRVs decreases the containment loading transients.

b. Stabilizing RPV pressure by manually opening EMRVs minimizes the unrecoverable coolant inventory loss.

c. Stabilizing RPV pressure reduces the RPV level perturbations from pressure transients on the reference legs.

d. Stabilizing RPV pressure below the High Pressure Scram setpoint which allows the scram to be reset when required.

QUESTION: 089 (1.00)

The plant is in a condition requiring RPV Level to be lowered per Step LP-2 of EMG-3200.09, "Level/Power Control." The operator is directed to continue the level reduction until RPV water level reaches 0 inches. Which one of the following is the bases for this level limit?

a. Level reductions below 0 inches can cause significant power oscillations.

b. Level reductions below 0 inches will not decrease the natural circulation core flow any further.

c. Level reductions below 0 inches will adversely affect adequate core cooling.

d. Level reductions below 0 inches requires Emergency RPV Depressurization if power is not below 2%.

QUESTION: 090 (1.00)

EMG-3200.02, "Primary Containment Control", step DW/T-2 directs initiation of containment sprays only if the Containment Spray Initiation Limit is not exceeded. Which one of the following is NOT correct with respect to the Containment Spray Initiation Limit?

a. The Containment Spray Initiation Limit is not of concern in a dry gas atmosphere because there is not steam that could be condensed when containment sprays are initiated.

b. The pressure decrease due to evaporative cooling is more limiting than condensation of steam if the Containment Spray Initiation Limit is violated.

c. The Containment Spray Initiation Limit insures that pressure in the Primary Containment will not decrease below 0 psig when containment sprays are initiated.

d. The Containment Spray Initiation Limit insures that torus to drywell differential pressure limits will not be exceeded when containment sprays are initiated.

QUESTION: 091 (1.00)

The plant experiences high drywell pressure followed by a loss of Off Site Power, approximately 5 minutes later. Which one of the following describes the expected response of the #1 EDG? (Do not consider any other system's response in your answer.)

a. -- DG #1 fast starts
-- Runs unloaded for 1.5 minutes and does an automatic shutdown
-- Restarts (fast start) on the loss of off-site power
-- Within 15 seconds of the Loss of off-site power the output breaker shuts

b. -- DG #1 fast starts
-- Runs unloaded until the loss of off-site power
-- Within 15 seconds of the loss of off-site power the output breaker shuts

c. -- DG #1 idle starts
-- Runs at 450 rpm until the loss of off-site power
-- Within 15 seconds of the loss of off-site power speed increases to 900 rpm and the output breaker shuts

d. -- DG #1 idle starts
-- Runs at 450 rpm for 1.5 minutes and does an automatic shutdown
-- Restarts (idle start) on the loss of off-site power
-- Within 90 seconds of the loss of off-site power speed increases to 900 rpm and the output breaker shuts

QUESTION: 092 (1.00)

The plant has experienced a loss of all AC power (Station Blackout) with no time estimate on restoration of Off-Site power or either of the Emergency Diesel Generators. The Isolation Condensers have initiated and are operating normally. All other plant conditions are stable.

Based on the above plant conditions, which one of the following is the approximate time the Isolation Condensers can operate before makeup water is required and the source of that makeup water?

a. 3/4 hour, Condensate Transfer System

b. 3/4 hour, Fire Protection System

c. 1 2/3 hours, Fire Protection System

d. 1 2/3 hours, Condensate Transfer System

QUESTION: 093 (1.00)

The following plant conditions exist:

- Plant startup in progress
- Reactor is critical, heating up to Hot Standby conditions
- Reactor pressure is 180 psig and the Mechanical Pressure Regulator (MPR) is set at 250 psig
- HP turbine warmup is in progress IAW 315.1, "Main Turbine Operation"

A problem occurs and Turbine Exhaust Hood temperature reaches 230 degrees F and is increasing. Which one of the following is the required operator action?

a. The operator should immediately use the bypass valve jack to close the bypass valves.

b. The operator should adjust reactor power to reduce pressure by insertion of control rods.

c. The operator should reduce the pressure regulator setpoint until the first bypass valve is controlling pressure.

d. The operator should run the load limit very slowly to its "close" position.

QUESTION: 094 (1.00)

Which one of the following Control Rod Drive Hydraulic System conditions requires an immediate manual reactor scram?

a. Reactor pressure is 750 psig, charging water pressure is 1450 psig and four (4) CRD High Temperature alarms are confirmed.

b. Reactor pressure is 750 psig, the running CRD pump trips, the standby CRD pump is out of service and three (3) accumulator trouble alarms are in.

c. Reactor pressure is 900 psig, charging water pressure is 1450 psig and four (4) accumulator trouble alarms are in.

d. Reactor pressure is 900 psig, the running CRD pump trips, the standby CRD pump is out of service and one (1) accumulator trouble alarm is in.

QUESTION: 095 (1.00)

Which one of the following situations would REQUIRE a reactor scram in accordance with EMG-3200.11 "Secondary Containment Control," steps SC/L-2, SC/L-3 and SC/L-4? (See attached steps)

a. NO primary system discharging into secondary containment
Rx Bldg SE Corner Room level at 10 inches
RX Bldg SW Corner Room level at 20 inches

b. NO primary system discharging into secondary containment
Rx Bldg NE Corner Room level at 18 inches
RX Bldg SE Corner Room level at 24 inches

c. A primary system IS discharging into secondary containment
Rx Bldg NW Corner Room level at 0 inches
RX Bldg SW Corner Room level at 20 inches

d. A primary system IS discharging into secondary containment
Rx Bldg SE Corner Room level at 6 inches
RX Bldg SW Corner Room level at 12 inches

QUESTION: 096 (1.00)

Evacuation of the Control Room is complete. The following individuals are at the Remote Shutdown Panel: Radiological Control Technician, Control Room Operator, Shift Technical Advisor, and Equipment Operator.

Which one of the following personnel is NOT allowed to leave the Panel area after it has been activated?

a. Radiological Control Technician

b. Equipment Operator

c. Shift Technical Advisor

d. Control Room Operator

QUESTION: 097 (1.00)

While performing ABN-3200.01 "Reactor Scram" procedure the following plant conditions are noted.

- Reactor Power is in the IRM range decreasing
- Reactor Pressure is 1053 psig increasing
- Reactor Water Level is 161 inches increasing

Which one of the following is the action that must be taken?

a. Continue performing ABN-3200.01 and concurrently perform ABN 3200.09 "Electric Pressure Regulator Malfunction"

b. Exit ABN-3200.01 and enter EMG-3200.01 "RPV Control"

c. Continue performing ABN-3200.01 and concurrently perform EMG.01 "RPV Control"

d. Exit ABN-3200.01 and enter ABN-3200.17 "Feedwater System Flow Control Failure"

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

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.

001	a	b	c	d	______
002	a	b	c	d	______
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 Answer the two parts of question 012

1.	a	b	c	d	______
2.	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	______

ANSWER SHEET

Multiple Choice (Circle or X your choice)

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

021 Answer the two parts of question 021

1. a b c d ______

2. a b c d ______

022 a b c d ______

023 a b c d ______

024 match with selected number in the blank

a ______

b ______

c ______

d ______

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 ______

036 a b c d ______

ANSWFR SHEET

Multiple Choice (Circle or X your choice)

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

037	a	b	c	d	______
038	a	b	c	d	______
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	______
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	______
054	a	b	c	d	______
055	a	b	c	d	______
056	a	b	c	d	______
057	a	b	c	d	______
058	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.

059	a	b	c	d	______
060	a	b	c	d	______
061	a	b	c	d	______
062	a	b	c	d	______
063	a	b	c	d	______
064	a	b	c	d	______
065	a	b	c	d	______
066	a	b	c	d	______
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	______
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	______

ANSWER SHEET

Multiple Choice (Circle or X your choice)

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

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	______
089	a	b	c	d	______
090	a	b	c	d	______
091	a	b	c	d	______
092	a	b	c	d	______
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 **********)

ANSWER: 001 (1.00)

b.

REFERENCE:

Oyster Creek Nuclear Generating Station Procedure 107 "Procedure Control" Rev. 36 page 27.0
OC GOS Qual Standard 8.A.2 pg 17
[/4.2]

294001A102 ..(KA's)

ANSWER: 002 (1.00)

c.

REFERENCE:

Oyster Creek Nuclear Generating Station Procedure 108 "Equipment Control Rev 48, page 22.0, 23.0, 27.0 & 105.0
OC GOS Qual Standard 9.A.3 pg 18
[/4.5]

294001K102 ..(KA's)

ANSWER: 003 (1.00)

b.

REFERENCE:

Oyster Creek Nuclear Generating Station Administrative Procedure 108 "Equipment Control" Rev 47 page 57.0 & 58.0
OC GOS Qual Standard 9.A.5 pg 18
[/4.5]

294001K102 ..(KA's)

ANSWER: 004 (1.00)

a.

REFERENCE:

Oyster Creek Nuclear Generating Station Administrative Procedure 108 "Equipment Control" Rev 47 page 104.0
OC GOS Qual Standard 9.A.8 pg 18
[/3.7]

294001K101 ..(KA's)

ANSWER: 005 (1.00)

c.

REFERENCE:

Oyster Creek Nuclear Generating Station Administrative Procedure 120.1 "Welding, Burning and Grinding Administrative Procedure " Rev 10 page 4.0
OC GOS Qual Standard 12.A.4 pg 22
[/3.8]

294001K116 ..(KA's)

ANSWER: 006 (1.00)

b.

REFERENCE:

Oyster Creek Nuclear Generating Station Administrative Procedure 120 "Fire Hazards" Rev 13 page 4.0
Oyster Creek Nuclear Generating Station Technical Specifications Section 3.12.E.2 pg 3.12-3
LER 90-001
OC GOS Qual Standard 12.A.3 pg 22
[/3.8]

294001K116 ..(KA's)

ANSWER: 007 (1.00)

b.

REFERENCE:

10 CFR 55.53(e) Issued 1-1-90
[/3.7]

294001A103 ..(KA's)

ANSWER: 008 (1.00)

c.

REFERENCE:

Oyster Creek Emergency Preparedness Implementing Document 9473-IMP-1300.01 "Classification of Emergency Conditions" Rev 8 Section 3.4 & 3.5 page 5.0
OC GOS Qual Standard Attachment No. 5 Emergency Plan 1.A.1. pg 65
[/4.7]

294001A116 ..(KA's)

ANSWER: 009 (1.00)

a.

REFERENCE:

Oyster Creek Nuclear Generating Station Procedure 122 "Security Guidelines for Plant Personnel" Rev 17 Section 5.2 pg 5.0 & 6.0
OC GOS Qual Standard Attachment No. 2 Procedure 122 15.A.1. pg 25
[/3.7]

294001K105 ..(KA's)

ANSWER: 010 (1.00)

b.

REFERENCE:

Oyster Creek Nuclear Generating Station Procedure 120.1 "Welding, Burning and Grinding Administrative Procedure" Rev 11 Instructions for Fire Watches pg E2-1
OC GOS Qual Standard Attachment No. 2 Fire Protection 12.A.4. pg 22
[/3.8]

294001K116 ..(KA's)

ANSWER: 011 (1.00)

b.

REFERENCE:

Oyster Creek Nuclear Generating Station Technical Specifications Section 6.2.2.2 a. - d. page 6-1 & 6-2
OC GOS Qual Standard Attachment No. 7 "Technical Specifications Section 6, 4.A.1 pg 78
LER 89-022
[/3.7]

294001A103 ..(KA's)

ANSWER: 012 (2.00)

1. c. (1.0)
2. b. (1.0)

REFERENCE:

10CFR20.101.
Oyster Creek: Radiation Controls Policy and Procedure Manual, 9300-ADM-4000.01, 7.2, pp. 3.0 and 4.0.
[/3.8]

294001K103 ..(KA's)

ANSWER: 013 (1.00)

b.

REFERENCE:

Station Procedure 205.0 section 2.5.1 pg 4.0 Rev 30, & section 2.6.1.7 pg 6.0 Rev 30.
Learning Objective TCR 812.0 E,F,G
[/4.2]

294001A110 ..(KA's)

ANSWER: 014 (1.00)

a.

REFERENCE:

IMP-1300.02, pg. 2-5 section 4.0 rev 5
IMP-1300.03, pg. 5 section 4.6 rev 11
OC GOS Qual Standard Attachment No. 5 Emergency Plan 2.A.1. pg 66 & 3.A.1 pg 67.
[/4.7]

294001A116 ..(KA's)

ANSWER: 015 (1.00)

b.

REFERENCE:

OCNGS AP 106 section 4.3.2 page 17.0
[/4.2]

294001A109 ..(KA's)

ANSWER: 016 (1.00)

c.

REFERENCE:

OCNGS Administrative Procedure 106 section 4.4.9 and 4.4.10 pages 27.0 & 28.0
[/3.6]

294001A106 ..(KA's)

ANSWER: 017 (1.00)

d.

REFERENCE:

Oyster Creek Nuclear Generating Station Technical Specifications
Section 3.3.H page 3.3-4.
Lesson Objective X of SRO Upgrade 6231-PGM-2611 #832.0 #03
[/4.0]

202002K302 ..(KA's)

ANSWER: 018 (1.00)

c.

REFERENCE:

OCNGS OPM 23 "Isolation Condenser" Section 5 page 23-19 & 23-20
Lesson Objective #828.23 L
[/4.5]

207000K401 ..(KA's)

ANSWER: 019 (1.00)

a.

REFERENCE:

Oyster Creek: Station Procedure 307, 2.2.9, pg. 7.0.
Lesson Objective #828.23 Q
[/3.8]

207000K103 ..(KA's)

ANSWER: 020 (1.00)

a.

REFERENCE:

OCNGS SOP 346 step 7.3.2.3 page 14
Lesson Objective #828.23 L
[/3.7]

207000K608 ..(KA's)

ANSWER: 021 (2.00)

1. d. (1.0)
2. b. (1.0)

REFERENCE:

OCNGS OPM 46 "Standby Liquid Control" section III.A.1.a. page 46-6
OCNGS T.S. 3.2.C Bases pg 3.2.7
Lesson Objective #828.46 H
[/4.2]

211000G006 ..(KA's)

ANSWER: 022 (1.00)

d.

REFERENCE:

Oyster Creek Technical Specifications pg 2.3-5 through 2.3-7
Lesson Objective 850.90 D.
[/4.3]

223002G006 ..(KA's)

ANSWER: 023 (1.00)

c.

REFERENCE:

Oyster Creek Nuclear Generating Station Operations Plant Manual Module
37 Reactor Protection System pg 37-24 through 37-26
Lesson Objective 828.37 F.
[/3.5]
212000K502 ..(KA's)

ANSWER: 024 (2.00)

a. - 1.
b. - 4.
c. - 3.
d. - 3. (4 @ 0.5 ea)

REFERENCE:

OCNGS OPM LP 55 pg 55-10 through 55-19
Lesson Objective 828.55 C.
[/3.4]

216000K414 ..(KA's)

ANSWER: 025 (1.00)

d.

REFERENCE:

OCNGS OPM #55 page 55-19 & table 55-1 pg 3 of 3.
Lesson Objective 828.55 B
[/4.1]

216000K107 ..(KA's)

ANSWER: 026 (1.00)

b.

REFERENCE:

OCNGS OPM 05 page 05-26 & Figure 05-14
Lesson Objective 828.05 J
[/4.2]

218000A403 ..(KA's)

ANSWER: 027 (1.00)

c.

REFERENCE:

OCNGS ABN-3200.01 Section 3.7 step 3.7.2 page 8.0
Lesson Objective 828.05 C
[/4.3]

218000A301 ..(KA's)

ANSWER: 028 (1.00)

a.

REFERENCE:

OCNGS OPM 09 pg 09-12 & 09-13
Lesson Objective 828.09 G
[/4.2]

226001G015 ..(KA's)

ANSWER: 029 (1.00)

b.

REFERENCE:

OCNGS OPM 51 pg 51-30 & 51-31
Lesson Objective 828.51 F
[/3.6]

241000K107 ..(KA's)

ANSWER: 030 (1.00)

c.

REFERENCE:

OCNGS OPM 51 pg 51-27 through 51-32
Lesson Objective 828.51 D
[/3.4]

241000A419 ..(KA's)

ANSWER: 031 (1.00)

a.

REFERENCE:

OCNGS NRC Exam Bank Question #006
[/4.1]

241000K102 ..(KA's)

ANSWER: 032 (1.00)

b.

REFERENCE:

OCNGS NRC Exam Bank Question 0032
OCNGS OPM 18 Figure 18-1
[/3.8]

259002K301 ..(KA's)

ANSWER: 033 (1.00)

d.

REFERENCE:

OCNGS NRC Exam Bank Question 0040
OCNGS OP 201.1
[/3.6]

215003K304 ..(KA's)

ANSWER: 034 (1.00)

d.

REFERENCE:

OCNGS NRC Exam Bank Question 0174
OCNGS GOP 201.2.3.10
Lesson Objective 828.39 H
[/3.3]

204000K101 ..(KA's)

ANSWER: 035 (1.00)

b.

REFERENCE:

OCNGS T.S. 3.13.G.3 page 3.13-3
Lesson Objective 828.033 B
[/3.9]

272000G005 ..(KA's)

ANSWER: 036 (1.00)

a.

REFERENCE:

OCNGS ABN 3200.15 table 2
Lesson Objective 801.001 B
[/3.3]

256000G014 ..(KA's)

ANSWER: 037 (1.00)

c.

REFERENCE:

OCNGS OP 317 ??? Facility review comment.
Lesson Objective 828.18 G
[/3.0]

259002A301 ..(KA's)

ANSWER: 038 (1.00)

a.

REFERENCE:

OCNGS SOP 330 Section 3.2.2 page 3.0 Rev 21.
Lesson Objective 828.42 K
[/3.7]

261000A213 ..(KA's)

ANSWER: 039 (1.00)

c.

REFERENCE:

OCNGS NRC Exam Bank Question # 0349
OCNGS RAP-3024.01 H-2-A
[/4.5]

201001K205 ..(KA's)

ANSWER: 040 (1.00)

b.

REFERENCE:

OCNGS OPM 13 pg 13-71 & 13-40 through 13-53
Lesson Objective 828.13 N
[/3.7]

264000K408 ..(KA's)

ANSWER: 041 (1.00)

b.

REFERENCE:

OCNGS OPM 13 pg 13-53
Lesson Objective 828.13 J
[/4.2]

264000K402 ..(KA's)

ANSWER: 042 (1.00)

a.

REFERENCE:

OCNGS OPM 29E pg 27
Lesson objective 828.29D I.E.2.
[/3.7]

215005K407 ..(KA's)

ANSWER: 043 (1.00)

~~d.~~ b.

REFERENCE:

OCNGS OPM 10 pg 10-22 & 10-23
Lesson objective 828.10 I.C
[/3.6]

209001A301 ..(KA's)

ANSWER: 044 (1.00)

a.

REFERENCE:

OCNGS GOP 201.3 P&L 3.4, 3.5.1, 3.6, & 3.8
No lesson objective identified.
[\3.7]

202001A401 ..(KA's)

ANSWER: 045 (1.00)

c.

REFERENCE:

OCNGS OPM Module 41 page 41-37
Lesson Objective 828.41.H
[\3.2]

201006K104 ..(KA's)

ANSWER: 046 (1.00)

d.

REFERENCE:

OCNGS OPM Module 41 page 41-35
Lesson Objective 828.41.G
[\3.5]

201006K402 ..(KA's)

ANSWER: 047 (1.00)

b.

REFERENCE:

CCNGS GOP 205.5 Sect. 6.1.9 page 7.0
Lesson Objective 828.36.K
[\3.7]

214000A303 ..(KA's)

ANSWER: 048 (1.00)

d.

REFERENCE:

OCNGS Lesson Plan 2610.828.36
Lesson Objective F.1
[/3.5]

201002K402 ..(KA's)

ANSWER: 049 (1.00)

b.

REFERENCE:

OCNGS OPM Module 16A Table 16A-1 page 1 of 2
No Lesson Objective Identified
[\3.3]

259001K201 ..(KA's)

ANSWER: 050 (1.00)

b.

REFERENCE:

OCNGS OPM Module 12 page 12-7
Lesson Objective 828.12.C
[\3.4]

263000K201 ..(KA's)

ANSWER: 051 (1.00)

a.

REFERENCE:

OCNGS OPM Module 26 page 26-27
Lesson Objective 828.26.I.F.4.
[\3.7]

239001A107 ..(KA's)

ANSWER: 052 (1.00)

a.

REFERENCE:

OCNGS ABN 3200.35
Lesson Objective 828.26.I.F.1.
[\3.2]

239001K602 ..(KA's)

ANSWER: 053 (1.00)

b.

REFERENCE:

OCNGS OPM Module 56 page 56-19
Lesson Objective 828.56.B & D
[\3.4]

262002K401 ..(KA's)

ANSWER: 054 (1.00)

a.

REFERENCE:

OCNGS OPM Module 42 page 42-35
Lesson Objective 828.42.G
[\3.8]

288000K402 ..(KA's)

ANSWER: 055 (1.00)

b.

REFERENCE:

OCNGS GOP 301 P&L 6.2.11 page 35.0
[\3.8]

295001A201 ..(KA's)

ANSWER: 056 (1.00)

a.

REFERENCE:

OCNGS ABN-3200.19 Section 3.1 page 4.0
Lesson Objective 801.01.A.7
[\3.3]

295018G010 ..(KA's)

ANSWER: 057 (1.00)

b.

REFERENCE:

OCNGS ABN-3200.20 step 3.3 page 4.0
[/3.3]

295018G010 ..(KA's)

ANSWER: 058 (1.00)

c.

REFERENCE:

OCNGS ABN-3200.14 Section 2.1 & 2.3 page 2.0
[\3.5]

295002K201 ..(KA's)

ANSWER: 059 (1.00)

d.

REFERENCE:

OCNGS ABN-3200.13 Section 3.1 pages 5.0 & 6.0
[\3.5]

295004K303 ..(KA's)

ANSWER: 060 (1.00)

a.

REFERENCE:

OCNGS 2000-RAP-3024.01 C-5-e
OCNGS Tech Specs Section 3.5.A.1 page 3.5-1 and Section 3.0.A page 3.0-1
[\4.4]

295029G008 ..(KA's)

ANSWER: 061 (1.00)

d.

REFERENCE:

OCNGS Tech Spec Section 2.3.C ,.I, .A.1, & .N page 2.3-2 & -3
[\3.8]

295005K301 ..(KA's)

ANSWER: 062 (1.00)

c.

REFERENCE:

OCNGS SBEOP Basis Review Handout 87.00 Step FLD-2.1 page 36 of 56
[\3.9]

295028K301 ..(KA's)

ANSWER: 063 (1.00)

d.

REFERENCE:

OCNGS Handout 87.12 EMG-3200.11, Section B.1 page 5 of 26
[\4.2]

295035G011 ..(KA's)

ANSWER: 064 (1.00)

c.

REFERENCE:

OCNGS Handout 87.12 Tables 11, 12, Section C.2.e page 13 & 14 of 26
[\3.4]

295032A104 ..(KA's)

ANSWER: 065 (1.00)

c.

REFERENCE:

OCNGS Handout No. 87.12 Section D.2.a. page 16 of 26
[\3.9]

295032K303 ..(KA's)

ANSWER: 066 (1.00)

c.

REFERENCE:

OCNGS EPIP 9473-IMP-1300.02 Exhibit 2 Section 4 page E2-4
[\4.2]

295033K102 ..(KA's)

ANSWER: 067 (1.00)

d.

REFERENCE:

OCNGS D & R 2000-OPS-3024.27 section 4.0 page 9.0
[\3.7]

295021K104 ..(KA's)

ANSWER: 068 (1.00)

c.

REFERENCE:

OCNGS OPM Module 32 Section 2.a pages 32-54 through 32-57
[\3.2]

295020A102 ..(KA's)

ANSWER: 069 (1.00)

d.

REFERENCE:

OCNGS 341, Pages 4.0, 5.0 & 15.0
No Lesson Objective Identified
[4.2/4.3]

295003A102 ..(KA's)

ANSWER: 070 (1.00)

a.

REFERENCE:

OCNGS OPM Module 51 page 51-36
Lesson Objective 828 51.E
[/3.7]

295007K201 ..(KA's)

ANSWER: 071 (1.00)

c.

REFERENCE:

Symptom Based Emergency Operating Procedures - RPV Power Control,
Lesson 845.05, Page 9 of 10, Lesson Objective "A"
[3.8/3.9]

295015A101 ..(KA's)

ANSWER: 072 (1.00)

c.

REFERENCE:

Symptom Based Emergency Operating Procedures - Level/Power Control,
Lesson 845.19, Page 4 of 8, Lesson Objective "B"
[/4.2]

295037K106 ..(KA's)

ANSWER: 073 (1.00)

b.

REFERENCE:

OCNGS ABN-3200.30, Pages 3.0 & 4.0
No Lesson Objective Identified
[3.8/3.6]

295016G010 ..(KA's)

ANSWER: 074 (1.00)

c.

REFERENCE:

OCNGS T.S. Section 3.5.A.1
OCNGS 310, Page 16.0
No Lesson Objective Identified
[/4.4]

295013G008 ..(KA's)

ANSWER: 075 (1.00)

a.

REFERENCE:

EMG-3200.02, Primary Containment Control Flowchart, Figure F step TOR/T-4.
SBEOP - Containment Control Lesson 845.06,
Lesson Objective "H"
[/4.5]

295026G012 ..(KA's)

ANSWER: 076 (1.00)

d.

REFERENCE:

SBEOP - Emergency RPV Depressurization page 7 of 12,
Lesson 845.14, Lesson Objective "B"
[3.8/4.1]

295030K103 ..(KA's)

ANSWER: 077 (1.00)

b.

REFERENCE:

Oyster Creek Technical Specification 3.5 "A" Bases, Page 3.5-9
No Lesson Objective Identified
[2.7/4.2]

295030G004 ..(KA's)

ANSWER: 078 (1.00)

c.

REFERENCE:

SBEOP - Primary Containment Control, Lesson 845.09, Page 10 of 12
and 87.02 page 5 of 6, Lesson Objective "D"
[/4.8]

295031A204 ..(KA's)

ANSWER: 079 (1.00)

b.

REFERENCE:

OCNGS ABN-3200.16, "Loss of FW Heaters", Page 4.0
OCNGS ABN-3200.34, "Power Oscillations", Page 3.0
No Lesson Objective Identified
[4.1/4.2]

295014A201 ..(KA's)

ANSWER: 080 (1.00)

a.

REFERENCE:

NRC BULLETIN NO. 88-07, SUPPLEMENT 1: Power Oscillations in Boiling
Water Reactors
Lesson Objective
[3.7/4.2]

295014K105 ..(KA's)

ANSWER: 081 (1.00)

b.

REFERENCE:

OCNGS ABN-3200.07, "Unexplained Reactivity Change", Page 4.0
No Lesson Objective Identified
[/3.7]

295014K102 ..(KA's)

ANSWER: 082 (1.00)

b.

REFERENCE:

OCNGS Tech. Spec. Section 3.9 page 3.98-1 and 3.9-2
[/3.8]
295023G003 ..(KA's)

ANSWER: 083 (1.00)

b.

REFERENCE:

Symptom Based Emergency Operating Procedure: EMG-3200.08, "RPV Flooding", Handout #87.17, Page 15 of 21
Symptom Based Emergency Operating Procedures - RPV Flooding, Lesson 845.18, Lesson Objective "B"
[/4.7]

295031K101 ..(KA's)

ANSWER: 084 (1.00)

b.

REFERENCE:

EMG-3200.02, Primary Containment Control Flowchart, Figure H
[/4.5]

295024G012 ..(KA's)

ANSWER: 085 (1.00)

b.

REFERENCE:

Symptom Based Emergency Operating Procedure Basis Review, Handout #87.13, Page 5 of 7
Symptom Based Emergency Operating Procedures - Radioactivity Release Control, Lesson 845.12, Lesson Objective "B"
OCEPID, IMP-1300.01, "Classification of Emergency Conditions", Page E1-5
[3.7/4.7]

295038K205 ..(KA's)

ANSWER: 086 (1.00)

c.

REFERENCE:

Symptom Based Emergency Operating Procedure Basis Review, Handout #87.13, Page 6 of 7
Symptom Based Emergency Operating Procedures - Radioactivity Release Control, Lesson 845.12, Lesson Objective "D"
[3.9/4.2]

295038K302 ..(KA's)

ANSWER: 087 (1.00)

d.

REFERENCE:

Symptom Based Emergency Operating Procedure Basis Review, Handout #87.00, Pages 21 and 50 of 56
Symptom Based Emergency Operating Procedures - Primary Containment Pressure Control, Lesson 845.08, Lesson Objective "A"
[/4.0]

295024K307 ..(KA's)

ANSWER: 088 (1.00)

a.

REFERENCE:

Symptom Based Emergency Operating Procedure Basis Review, Handout #87.00, Page 11 of 56
Symptom Based Emergency Operating Procedures - RPV Pressure Control, Lesson 845.04, Lesson Objective "A"
[4.1/4.2]

295025K205 ..(KA's)

ANSWER: 089 (1.00)

c.

REFERENCE:

Symptom Based Emergency Operating Procedure Basis Review, Handout #87.18, Page 13 of 29
Symptom Based Emergency Operating Procedures - Level/Power Control, Lesson 845.19, Lesson Objective "B"
[/4.5]

295037K303 ..(KA's)

ANSWER: 090 (1.00)

a.

REFERENCE:

EMG-3200.02, "Primary Containment Control" flowchart, Step DW/T-2
SBEOP Handout 87.00 page 19 and 20 of 56
Objective Identified 845.07.I.A
[/3.9]

295028K301 ..(KA's)

ANSWER: 091 (1.00)

c.

REFERENCE:

OPM Module 13, "Emergency Diesel Generators", Page 13-62
No Lesson Objective Identified
[4.1/4.2]

295003K202 ..(KA's)

ANSWER: 092 (1.00)

c.

REFERENCE:

OPM Module 23, "Isolation Condenser System", Pages 23-5 & 23-7
No Lesson Objective Identified
[3.8/4.0]

295003K205 ..(KA's)

ANSWER: 093 (1.00)

a.

REFERENCE:

OCNGS 315.1, "Main Turbine Operation", Page 14.0
No Lesson Objective Identified
[3.6/3.7]

295005K207 ..(KA's)

ANSWER: 094 (1.00)

b.

REFERENCE:

OCNGS RAP-3024.0-1, H-7-c
No Lesson Objective Identified
[/3.5]

295022G010 ..(KA's)

ANSWER: 095 (1.00)

c.

REFERENCE:

Symptom Based Emergency Operating Procedure Basis Review, Handout #87.12, Pages 24 and 25 of 26
Symptom Based Emergency Operating Procedures - Secondary Containment Control, Lesson 845.11, Lesson Objective "B"
[/3.9]

295036G012 ..(KA's)

ANSWER: 096 (1.00)

d.

REFERENCE:

OCNGS ABN-3200.30, Page 2.0
No Lesson Objective Identified
[4.4/4.5]

295016K201 ..(KA's)

ANSWER: 097 (1.00)

b.

REFERENCE:

OCNGS ABN-3200.01 Step 3.0 page 3.0
OCNGS EMG-3200.01 "RPV Control" Entry Conditions
[/4.3]

295007G011 ..(KA's)

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

TEST CROSS REFERENCE

QUESTION	VALUE	REFERENCE
001	1.00	9000001
002	1.00	9000002
003	1.00	9000003
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005	1.00	9000005
006	1.00	9000006
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010	1.00	9000010
011	1.00	9000011
012	2.00	9000012
013	1.00	9000013
014	1.00	9000014
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