# U.S. NUCLEAR REGULATORY COMMISSION
# OPERATOR LICENSING EXAMINATION REPORT

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| Examination Report No.: | 92-22 (OL) |
| Facility Docket No.: | 50-293 |
| Facility License No.: | DPR-35 |
| Licensee: | Boston Edison Company<br>RFD #1 Rocky Hill Road<br>Plymouth, Massachusetts 02360 |
| Facility: | Pilgrim Nuclear Power Station |
| Examination Dates: | November 16 - 20, 1992 |
| Examiners: | T. Walker, Senior Operations Engineer<br>S. Hansell, Operations Engineer<br>C. Tyner, Examiner (EG&G)<br>J. Hanek, Examiner (EG&G) |

Chief Examiner: T. E. Walker 1/6/93
T. Walker, Senior Operations Engineer Date

Approved by: D. Florek for 1/6/93
Richard J. Conte, Chief, BWR Section Date
Operations Branch, DRS

## EXECUTIVE SUMMARY

Initial examinations were administered to three Senior Reactor Operator (SRO) instant and seven Reactor Operator (RO) applicants. Six of seven RO and all of the SRO applicants passed both portions of the examinations. One RO did not pass the written examination. Requalification retake examinations were administered to one licensed SRO (walk-through examination only) and one licensed RO (written examination only). Both licensed operators passed the requalification retake examinations. The examinees were well prepared for the examinations. The applicants' communications during the simulator portion of the initial examinations were a noted strength. The facility staff was very cooperative during the examination preparation and administration.

During the examination process, discrepancies were identified between the Emergency Plan training materials and procedures and management expectations which indicate a weakness in the interface between Operations and Emergency Planning departments in Emergency Plan training for licensed operators.

## DETAILS

### 1.0 INTRODUCTION

The NRC administered initial examinations to three Senior Reactor Operator (SRO) instant and seven Reactor Operator (RO) applicants. The examinations were administered in accordance with NUREG 1021, Examiner Standards, proposed Revision 7.

The NRC administered requalification retake examinations to one licensed SRO (walk-through portion only) and to one licensed RO (written portion only) who did not pass the NRC administered requalification examinations in May 1992. The examinations were administered in accordance with NUREG 1021, Examiner Standards, Revision 6.

### 2.0 PREEXAMINATION ACTIVITIES

Several problems related to learning objectives were identified during preparation of the initial examinations. The reference materials that were initially submitted for examination preparation did not contain any learning objectives related to the use of procedures or administrative topics. At the NRC's request, the facility provided the on-shift training materials, which included student handouts and task lists. The student handouts contained objectives related to procedures; however, these objectives were of a generic nature and did not contain specific conditions and standards of performance. An administrative task list was provided, but no learning objectives related to administrative topics were available. The materials that were initially submitted also did not contain learning objectives related to Technical Specifications for ROs. Technical Specification training materials, including learning objectives, for ROs were also provided at the NRC's request. Problems were also identified with the Emergency Plan training materials and learning objectives. These problems are discussed in Section 3.4 of this report.

The facility reviewed the written examinations in the facility training center during the week of November 2, 1992. A large number of changes had to be made to the initial examinations as a result of this review, including 15 questions (out of 130) that had to be replaced or revised significantly. Problems with learning objectives or lack of specific learning objectives were the cause of the majority of the significant revisions and replacements. The licensee agreed to review and revise the faulty learning objectives.

The simulator scenarios and Job Performance Measures (JPMs) were validated during the week of November 2, 1992, on the facility's simulator and in the plant. Several changes had to be made due to the limitations of the simulator as noted in Attachment 5. One JPM had to be replaced because of inconsistencies between the Emergency Plan training materials and management expectations as discussed in Section 3.4.

The facility staff who were involved with these reviews signed security agreements to ensure that the initial and requalification examinations were not compromised. The personnel involved in the preexamination reviews are identified in Attachment 1.

## 3.0 EXAMINATION RESULTS AND RELATED FINDINGS, OBSERVATIONS AND CONCLUSIONS

### 3.1 Examination Results

The results of the examinations are summarized below:

| Initial Examinations | SRO Pass/Fail | RO Pass/Fail |
|---|---|---|
| Written | 3/0 | 6/1 |
| Operating | 3/0 | 7/0 |
| Overall | 3/0 | 6/1 |

| Requalification Retake Examinations | SRO Pass/Fail | RO Pass/Fail |
|---|---|---|
| Written | N/A | 1/0 |
| Operating | 1/0 | N/A |
| Overall | 1/0 | 1/0 |

### 3.2 Generic Strengths and Weaknesses

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

Written Examination

Strengths:

- Knowledges and abilities associated with emergency and abnormal events (RO only)

- Ability to interpret Technical Specifications (SRO only)

Weaknesses:

The following subjects were missed by at least forty percent of the applicants that were evaluated on the subject indicating a generic weakness in the subject:

- Knowledge of minimum shift crew composition in accordance with Technical Specifications (SRO only)

- Ability to determine composition of a radioactive release (SRO only)

- Knowledge of restrictions on HPCI operation during implementation of the EOPs

- Ability to identify and predict the plant response for an APRM flow reference mismatch condition

- Ability to predict the system response to a failed control rod reed switch

- Ability to determine the cause and predict the plant response to a reactor feed pump low net positive suction head condition

- Ability to predict the system response to a Standby Gas Treatment train heater trip

- Ability to predict the HPCI system response to an increase in torus water level (RO only)

- Ability to predict the response of the MSIVs to a loss of instrument air and DC power

- Knowledge of the requirement for tripping the turbine on high vibration during a startup (RO only)

- Ability to determine the correct method for reactor pressure control following a failure to scram and MSIV closure (SRO only)

- Ability to determine the appropriate actions for primary containment hydrogen control (SRO only)

- Ability to determine the correct actions for venting primary containment with radioactive release rates above Technical Specification limits (SRO only)

- Ability to identify secondary containment conditions which would require Alternate RPV Depressurization (SRO only)

Operating Tests (Simulator and Walk-through)

Strengths:

- SRO briefings to the crew during abnormal and emergency events

- Crew communications and team skills (also observed in the Control Room by actual operating crews)

- Response to JPMs which required use of an alternate procedure or procedure path (alternate path JPMs)

- Applicant familiarity with the examination process

Weaknesses:

- Ability to perform immediate actions for abnormal and emergency events without the use of the procedure; specifically, the immediate actions for a Control Room evacuation

- Ability to assume the responsibilities of a Control Room communicator during an event which requires implementation of the Emergency Plan; specifically, ability to use the Plant Data Phone (RO only)

- Lack of familiarity with the procedure for a loss of Shutdown Cooling (SRO only)

- Knowledge of the interlock that causes a Recirculation Pump trip if the pump discharge valve is taken to the close position (RO only)

### 3.3 Procedures

During the preparation of the examinations, the NRC identified several problems with facility procedures. The following are examples of procedure discrepancies that were identified during examination preparation.

- PNPS Proc. 5.3.26, "RPV Injection During Emergencies," provides direction for cross-tieing the Firewater system to the Feedwater system for use when implementing the EOPs. PNPS 5.3.26 states that the methods for RPV injection may only be used when specifically directed by the EOPs. None of the EOP flowcharts provide direction to cross-tie Firewater to Feedwater for RPV injection. It is not clear whether Firewater cross-tied to Feedwater can be used when implementing the EOPs.

- Section 4.3 of PNPS Proc. 2.4.54, "Loss of All Fire Suppression Pumps or Loss of Redundancy in the Fire Water Supply System," which provides direction for isolating a leak on the Firewater ring header contains several technical errors. Portions of the procedure do not accomplish the intended function and/or fail to restore the Firewater system to an operable status following isolation.

The licensee was very responsive when the problems were identified and agreed to review and correct the procedures accordingly.

### 3.4 Emergency Plan

During the preparation of the written examinations, the NRC identified that the instructions for designation of an Assembly Area in the procedures responding to a Site Area Emergency and a General Emergency were not consistent with Emergency Planning department expectations. EP-IP-130, "Site Area Emergency," and EP-IP-140, "General Emergency," provide guidance on designating Assembly Areas for site evacuation. If a release is in progress, the Chiltonville Training Center or the Kingston Warehouse should be designated based on wind direction. In accordance with EP-IP-130 and EP-IP-140, if the event is being upgraded from an Alert with no release in progress, the I&S Building should be the designated Assembly Area. According to a PNPS emergency planning manager, I&S Building is the primary Assembly Area and should be designated regardless of the timing of the event classification. Chiltonville or Kingston would only be used as the Assembly Area in the case of a ground level release or an elevated release with a release path that would preclude evacuation to the I&S Building. Procedures that are inconsistent with management expectations and training could cause confusion during an emergency that requires evacuation of onsite personnel. The emergency planning manager indicated that the procedures would be revised as necessary to coincide with management expectations.

The training materials, specifically, the learning objectives related to Emergency Plan training for SRO licensing provided for examination preparation, were not consistent with Emergency Planning and Operations management expectations. The PNPS expectations for SRO knowledge and abilities were not clearly defined as indicated by the following discrepancies in the Emergency Plan training material for SROs.

- Unit Guide O-RO-06-03, "Emergency Plan Training for Senior Reactor Operators," references an abbreviated version of the Emergency Public Information Organization and Position Training module (T-ER-01-01-19), but the detailed module (T-ER-01-01-13) was provided with the examination materials. It was not clear which module is applicable to SRO license training.

- Unit Guide O-RO-06-03 references module T-ER-01-01-80 for dose assessment and Protective Action Recommendation (PAR) training. This module was provided with the examination materials; however, it is not the module that is used for SRO training. According to an Emergency Planning manager, module T-ER-01-01-81, the overview version of dose assessment and PAR training, is used for SRO training. This discrepancy was identified during the on-site JPM validation and resulted in replacement of a JPM. Based on the learning objectives in the training materials that were provided, the NRC prepared a JPM to perform an off-site dose calculation. The Operations and Training department personnel participating in the examination preparation indicated that this was not an appropriate task for an SRO. Task 91 of Unit Guide O-RO-04-10, "SRO On-Shift Tasks," specifies that an SRO must estimate offsite release rates. The NRC agreed to replace the JPM for this examination; however, the NRC expects an SRO to understand the dose assessment process thoroughly in order to perform the responsibilities of the Emergency Director in approving offsite notifications and determining PARs.

The discrepancies in the Emergency Plan training materials and unclear management expectations indicate a weakness in Operations involvement in the Emergency Plan portion of the licensed operator training program. The identified weakness in the RO applicants' ability to assume the responsibilities of a Control Room communicator during an emergency event also indicated a weakness in Emergency Plan training.

### 3.5 In-Plant Observations

The examiners noted that the material condition of the plant was good. Access into the plant and through the radiological control areas was smooth. The control room atmosphere was conducive to the conduct of the examinations. The examiners also observed excellent communication by the control room crews.

## 4.0 EXIT MEETING

An exit meeting was conducted November 20, 1992, following the administration of the examinations. Attendees are listed in Attachment 1. The facility presented their comments on the written examinations (Attachment 4). The NRC discussed generic findings regarding the applicants performance on the operating tests, simulator fidelity problems (Attachment 6), and the problems encountered during preparation for the examinations (Sections 2.0, 3.3, and 3.4).

## ATTACHMENT 1

## PERSONS CONTACTED

<u>Licensee Personnel</u>

| | |
|---|---|
| E. T. Boulette | Vice President Nuclear Operations/Station Director (3) |
| E. S. Kraft, Jr. | Plant Manager (2), (3) |
| W. C. Rothert | Director Nuclear Engineering (3) |
| J. F. Alexander | Nuclear Training Manager (2), (3) |
| H. R. Balfour | Operations Training Section Manager (2), (3) |
| T. A. Sullivan | Operations Section Manager (1), (2), (3) |
| A. R. Shiever | Operator Training Division Manager - Initial (1), (2), (3) |
| T. S. Swan | Operator Training Division Manager - Requalification(1) |
| N. L. Desmond | Compliance Division Manager (2) |
| R. L. Cannon | Senior Compliance Engineer (2), (3) |
| W. J. Green | Senior Operator Training Specialist (3) |
| M. Santiago | Senior Operator Training Specialist (1) |
| E. Olsen | Senior Operator Training Specialist (1) |
| P. V. Gallant | Operator Training Specialist (1) |
| R. F. Balduc | Operator Training Instructor (Protech) (1) |
| D. A. Whiting | Operator Training Instructor (Protech) (1) |
| D. Landahl | Emergency Planning Division Manager |

<u>NRC Personnel</u>

| | |
|---|---|
| R. Conte | BWR Section Chief |
| T. Walker | Senior Operations Engineer (1), (2), (3) |
| S. Hansell | Operations Engineer (1), (2) |
| A. C. Cerne | Resident Inspector (2) |

NOTES:
(1) Participated in Examination Preparation
(2) Attended Entrance Meeting on November 5, 1992
(3) Attended Exit Meeting on November 20, 1992

Attachment 2

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

| | |
|---|---|
| CANDIDATE'S NAME: | |
| FACILITY: | Pilgrim 1 |
| REACTOR TYPE: | BWR-GE3 |
| DATE ADMINISTERED: | 92/11/16 |

INSTRUCTIONS TO CANDIDATE:

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

| TEST VALUE | CANDIDATE'S SCORE | % | |
|---|---|---|---|
| 98.0 ~~100.00~~ | | % | TOTALS |
| | FINAL GRADE | | |

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

Candidate's Signature

deleted

QUESTION: 001 (1.00)

WHICH ONE of the following On Demand (OD) Programs would be used to obtain the effective readings from an LPRM string used in the most recent P1 power distribution calculation?

a. OD-6, Thermal Data in a Specified Bundle

b. OD-8, Present LPRM Readings

c. OD-9, Axial Interpolation in a Specified LPRM String

d. OD-17, Periodic Core Performance Logs

QUESTION: 002 (1.00)

Conditions in a recently surveyed area are:

25 mR/hr general area radiation
100 dpm/100 cm2 alpha loose surface
500 dpm/100 cm2 beta-gamma loose surface
0.20 MPC airborne beta-gamma radioactivity

WHICH ONE of the following describes the complete posting requirements for the area?

a. "CAUTION RADIATION AREA"

b. "CAUTION RADIATION AREA" and "CAUTION CONTAMINATED AREA"

c. "CAUTION RADIATION AREA" and "CAUTION AIRBORNE RADIOACTIVITY AREA"

d. "CAUTION RADIATION AREA," "CAUTION CONTAMINATED AREA," and "CAUTION AIRBORNE RADIOACTIVITY AREA"

QUESTION: 003 (1.00)

A 23 year old radiation worker with a current Form NRC-4 needs to perform work in an area with general radiation levels of 75 mR/hr. The worker's exposure history is:

| | |
|---|---|
| Lifetime: | 24.5 Rem |
| Current year: | 1400 mR |
| Current quarter: | 225 mR |

WHICH ONE of the following is the maximum time that the worker can stay in the area without exceeding any PNPS Exposure Control Levels? (Assume no special authorization.)

a. 20 minutes

b. 1 hour and 20 minutes

c. 6 hours and 40 minutes

d. 7 hours

QUESTION: 004 (1.00)

WHICH ONE of the following describes proper procedures for handling sodium pentaborate?

a. Respiratory protection is required to prevent inhaling dust containing boron.

b. Protective clothing is not required because sodium pentaborate cannot be absorbed through the skin.

c. Report to the Chemistry Department immediately after leaving the worksite to dispose of protective clothing and foot coverings.

d. Direct any liquid spillage to a floor drain, wipe and dry mop the flocr, and discard mops and wipes after use.

QUESTION: 005 (1.00)

WHICH ONE of the following situations is acceptable in accordance with PNPS Proc. No. 1.4.36, "High Pressure/Compressed Gas Cylinder Control?"

a. Argon cylinders on a cart are staged for a non-active Maintenance Request that has been rescheduled for later in the week. The cart is secured to a fixed support.

b. Reserve nitrogen cylinders are secured at approximately 3/4 height to a permanently installed cylinder holding station.

c. Acetylene cylinders on a cart are staged in the Cable Spreading Room for an active Maintenance Request. Work is expected to start next shift. The cart is secured to a fixed support.

d. Empty oxygen cylinders are secured at approximately 3/4 height to a fixed support during shift turnover. The cylinders will be removed next shift.

QUESTION: 006 (1.00)

WHICH ONE of the following situations would require independent verification in accordance with PNPS Proc. No. 1.3.34, "Conduct of Operations?" (All components are safety related.)

a. Fuses are pulled for maintenance in a cabinet in the Control Room.

b. An existing tagout is used to verify the position of an inaccessible component.

c. Verification of a component's position is expected to take 15 minutes. Radiation levels in the area are 120 mr/hr.

d. A large manually operated valve that requires two people to operate must be shut for maintenance.

QUESTION: 007 (1.00)

Breaker 103 (Bus A1 feeder from Unit Auxiliary Transformer) is open and has a white tag with a green border hanging on the control switch. WHICH ONE of the following describes the meaning of the tag?

a. A grounding device is installed on the breaker. The breaker may only be operated under the authority of the person for whom the tag was placed.

b. The breaker is not in its normal operable status. The breaker may be operated by qualified operators when the precautions listed on the tag are followed.

c. The breaker requires testing prior to restoring it to normal service. The breaker may be operated only under the authority of the person for whom the tag was placed.

d. The breaker is open to protect personnel from injury or equipment from damage. The breaker may not be operated until the tag is cleared.

QUESTION: 008 (1.00)

WHICH ONE of the following accurately describes the NRC Overtime Guidelines?

a. Operators may not work more than: 12 hours in 24 hours; 24 hours in 48 hours; or 84 hours in 7 days.

b. Operators may not work more than: 16 hours in 24 hours; 28 hours in 48 hours; or 84 hours in 7 days.

c. Operators may not work more than: 16 hours in 24 hours; 24 hours in 48 hours; or 72 hours in 7 days.

d. Operators may not work more than: 12 hours in 24 hours; 28 hours in 48 hours; or 72 hours in 7 days.

QUESTION: 009 (1.00) deleted

WHICH ONE of the following specifically describes the electrical equipment controlled by REMVEC?

a. All equipment $\geq$ 4.16 kV

b. All equipment $\geq$ 115 kV

c. All equipment $\geq$ 230 kV

d. All equipment $\geq$ 345 kV

QUESTION: 010 (1.00)

The paper in a Control Room chart recorder has jammed. WHICH ONE of the following describes the appropriate actions for replacing the recorder chart?

a. Initial and date the recorder chart that was removed and the new recorder chart to be installed. The removed recorder chart will be stored in the Control Room for at least a week.

b. Initial and date the recorder chart that was removed and the new recorder chart to be installed. The removed recorder chart will be sent to Records Management for storage at the earliest convenience.

c. Initial and date the new recorder chart to be installed and log the recorder chart replacement in the Operating Log. The removed recorder chart will be stored in the Control Room for at least a week.

d. Initial and date the new recorder chart to be installed and log the recorder chart replacement in the Operating Log. The removed recorder chart will be sent to Records Management for storage at the earliest convenience.

QUESTION: 011 (1.00)

WHICH ONE of the following describes the meaning of this icon:

when implementing the EOPs?

a. The CS/RHR pumps may not be operated below the vortex limit.

b. The CS/RHR pumps may be operated irregardless of the vortex limit.

c. The CS/RHR pumps may not be operated below the NPSH limit.

d. The CS/RHR pumps may be operated irregardless of the NPSH limit

QUESTION: 012 (1.00)

WHICH ONE of the following Emergency Action Levels (EALs) is the minimum action level at which the emergency facilities (TSC, OSC, and EOF) must be activated?

a. General Emergency

b. Site Area Emergency

c. Alert

d. Unusual Event

QUESTION: 013 (1.00)

WHICH ONE of the following communication systems can be used to transmit information from the Control Room directly to the EOF during an emergency?

a. Digital Voice Network (DVN)

b. Emergency Notification System (ENS)

c. Boston Edison Community Off-Site Notification System (BECONS)

d. Plant Data Phone (PDP)

QUESTION: 014 (1.00)

WHICH ONE of the following primary containment parameters does NOT have any associated Technical Specification requirements?

a. Drywell temperature

b. Torus water level

c. Primary containment oxygen concentration

d. Primary containment hydrogen concentration

QUESTION: 015 (1.00)

The plant was operating at 100% power when the main turbine first stage pressure transmitter (PT-652) failed downscale. WHICH ONE of the following describes the expected plant response?

a. The Rod Worth Minimizer will enforce adherence to the rod sequence due to pressure below the Low Power Setpoint.

b. The Main Condenser Low Vacuum Scram will be bypassed due to pressure below 600 psig.

c. The FW Control steam flow detectors will indicate lower than actual due to loss of density compensation.

d. The "MAIN STEAM LINE LEAKAGE" alarm will annunciate due to the main steam flow/turbine steam flow mismatch.

QUESTION: 016 (1.00)

During normal power operations, the operator at the controls accidently takes the control switch for the RWCU return isolation valve, MO-80, to the full closed position. WHICH ONE of the following describes the expected RWCU System response?

a. As soon as MO-80 leaves the full open position, MO-2 and MO-5 will close causing the RWCU pumps to trip.

b. Once MO-80 reaches the full closed position, the RWCU pumps will trip.

c. As soon as MO-80 leaves the full open position, the RWCU pumps will trip.

d. Once MO-80 reaches the full closed position, no flow will exist in the system causing the holding pump to start.

QUESTION: 017 (1.00)

The reference legs to all of the FWLC level instruments have broken. WHICH ONE of the following describes the signals that will be received due to the failure of the level instruments?

a. Reactor Feed Pump trip. RPV High Level alarm.

b. Main Turbine, HPCI, and RCIC trip.

c. Reactor Recirc Pump runback. RPV Low Level alarm.

d. HPCI and RCIC initiation. EDG start. PCIS isolations. Reactor Recirc Pump trip. Reactor scram.

QUESTION: 018 (1.00)

WHICH ONE of the following conditions could cause indicated level on the Fuel Zone instruments to be higher than actual reactor vessel water level?

a. Drywell temperature at 180°F.

b. A break in the variable leg.

c. No forced recirculation flow through the jet pumps.

d. A rapid reactor depressurization.

QUESTION: 019 (1.00)

All of the Scram Discharge Volume (SDV) vent and drain valves were manually operated during maintenance. The valves were returned to their normal position, but the manual operators for the SDV valves were not returned to the NEUTRAL position. WHICH ONE of the following describes the concern related to this operation?

a. The valves could fail to automatically reposition on a reactor scram, preventing drain down of the SDV.

b. The valves could fail to automatically reposition on a reactor scram, causing a direct discharge path from the RPV to the Reactor Building sump.

c. The valves could fail open after repositioning on a reactor scram, causing a breach of primary containment.

d. The valves could fail closed after repositioning on a reactor scram, preventing reset of the scram due to high SDV level.

QUESTION: 020 (1.00)

A scram has occurred and the mode switch has been placed in SHUTDOWN. Scram Discharge Volume (SDV) level is high, but the scram has been reset by use of the SDV High Level Scram Bypass. RPS bus A is then transferred to its alternate power supply. WHICH ONE of the following describes the expected response?

a. It is a dead bus transfer and a half scram will occur.

b. It is a dead bus transfer and a full scram will occur.

c. It is a dead bus transfer, but is rapid enough so that no scram will occur.

d. It is a live bus transfer, so no scram will occur.

QUESTION: 021 (1.00)

A plant startup was underway when annunciator "MSIV NOT FULLY OPEN SCRAM AT RX PRESSURE >600 PSI" (905R, D4) illuminated. A full scram occurred. WHICH ONE of the following describes the expected status of the plant?

a. One main steam line is isolated; reactor pressure is greater than 600 psig.

b. Two MSIVs have closed; reactor pressure is less than 600 psig.

c. Three main steam lines are isolated; reactor pressure is greater than 600 psig.

d. Four MSIVs have closed; reactor pressure is less than 600 psig.

QUESTION: 022 (1.00)

The plant was operating at 100% power when a loss of all feed caused reactor water level to decrease. The reactor scrammed and the mode switch was placed in SHUTDOWN. Reactor water level is -10 inches and reactor pressure is 900 psig. WHICH ONE of the following sets of valves should have received isolation signals?

a. Recirculation System process sample valves, Post Accident Sampling System isolation valves, and RHR reject to Radwaste valves

b. RWCU isolation valves, RBCCW to Drywell Coolers isolation valves, and Primary Containment Atmosphere Control makeup purge valves

c. MSIVs, Drywell Equipment Drain Sump isolation valves, and Radiation Leak Detection System isolation valves

d. Main Steam Line drains, Drywell Floor Drain Sump isolation valves, and Hydrogen/Oxygen Analyzer System isolation valves

QUESTION: 023 (1.00)

The RCIC system is running following a valid initiation signal received 10 minutes ago. The following conditions develop:

| | |
|---|---|
| RCIC Steam Line Flow: | 230% |
| RCIC Area Temperature: | 205°F |
| RCIC Pump Suction Pressure: | 10" Hg vacuum |
| RCIC Turbine Exhaust Pressure: | 38 psig |
| RPV Pressure: | 150 psig |
| RPV Water Level: | +40 inches |

WHICH ONE of the following describes the expected response of the RCIC system?

a. Only the RCIC inboard and outboard isolation valves close.

b. The RCIC inboard and outboard isolation valves close and RCIC trip throttle valve closes.

c. Only the RCIC trip throttle valve closes.

d. Only the RCIC steam to turbine supply valve closes.

QUESTION: 024 (1.00)

WHICH ONE of the following describes the restrictions on HPCI operation while implementing the EOPs?

a. HPCI may NEVER be operated below 1000 RPM because this is the minimum speed required to maintain adequate cooling and lubrication.

b. HPCI may NEVER be operated below 2000 RPM because this is the minimum speed required to generate sufficient control oil pressure for control valve operation.

c. HPCI may be operated below 1000 RPM only at NOS/NWE direction because low turbine exhaust pressure could create a cyclic steam hammer which could damage the exhaust check valve.

d. HPCI may be operated below 2000 RPM, but greater than 1000 RPM only as directed by the EOPs because operation at these speeds could cause excessive turbine vibration or oscillating flow rates.

QUESTION: 025 (1.00)

With the plant operating at 100% power, the Core Spray Loop A line break differential pressure indication on Rack 2207 reads +4.5 psid. WHICH ONE of the following conditions is indicated?

a. No core spray line break has occurred. This indication is normal for full power operations.

b. A core spray line break has occurred inside the reactor vessel shroud.

c. A core spray line break has occurred inside the reactor vessel, but outside the shroud.

d. A core spray line break has occurred inside the drywell or inside the reactor vessel, but outside the shroud.

QUESTION: 026 (1.00)

The reactor is shutdown and RHR loop B is in the shutdown cooling (SDC) mode. A reactor coolant system leak causes vessel level to decrease to 0 inches and drywell pressure to increase to 2.8 psig. Loop A jet pump riser pressure is less than loop B jet pump riser pressure. No operator action has been taken. WHICH ONE of the following describes the status of the Low Pressure Coolant Injection (LPCI) valves?

a. Outboard injection valve 28A is open
Inboard injection valve 29A is open
Outboard injection valve 28B is closed

b. Outboard injection valve 28A is closed
Inboard injection valve 29A is open
Outboard injection valve 28B is closed

c. Outboard injection valve 28B is open
Inboard injection valve 29B is closed
Outboard injection valve 28A is closed

d. Outboard injection valve 28B is open
Inboard injection valve 29B is open
Outboard injection valve 28A is closed

QUESTION: 027 (1.00)

WHICH ONE of the following RHR loop lineups assures that the design limits are met for adequate drywell sprays and RHR equipment operation?

a. RHR pump B is running, the RHR heat exchanger bypass valve is closed, and RHR heat exchanger flow is 4800 gpm.

b. RHR pump C is running, the RHR heat exchanger bypass valve is open, and RHR heat exchanger flow is 3750 gpm.

c. RHR pumps B and D are running, the RHR heat exchanger bypass valve is closed, and RHR heat exchanger flow is 9100 gpm.

d. RHR pumps A and C are running, the RHR heat exchanger bypass valve is open, and RHR heat exchanger flow is 5050 gpm.

QUESTION: 028 (1.00)

LPCI initiated on a valid initiation signal. RPV water level is -130 inches and increasing slowly. Drywell pressure is 2.0 psig and increasing. WHICH ONE of the following describes the actions necessary to open MO-34 and MO-37 to spray the torus under these conditions?

a. MO-34 and MO-37 cannot be opened until drywell pressure increases above 2.5 psig.

b. Place the keylock RPV level override switch in MANUAL OVERRIDE.

c. Place the pistol grip LPCI override switch in MANUAL OVERRIDE.

d. Place the keylock RPV level override switch in MANUAL OVERRIDE and place the pistol grip LPCI override switch in MANUAL OVERRIDE.

QUESTION: 029 (1.00)

An ADS blowdown was initiated on high drywell pressure and low-low reactor water level. The only low pressure CSCS pump running is Core Spray pump B. WHICH ONE of the following conditions will close the ADS valves?

a. The ADS Inhibit switches are placed in INHIBIT, drywell pressure decreases to 2.0 psig, and the high drywell pressure reset pushbuttons are depressed.

b. The ADS Inhibit switches are placed in INHIBIT, reactor water level increases to -35 inches, and Core Spray pump B trips.

c. Drywell pressure decreases to 1.0 psig, the high drywell pressure reset pushbuttons are depressed, and the timer reset pushbuttons are depressed.

d. Core Spray pump B trips, drywell pressure decreases to 1.8 psig, and reactor water level increases to -40 inches.

QUESTION: 030 (1.00)

The control switch for a safety relief valve is in the REMOTE position on alternate shutdown panel 156. WHICH ONE of the following describes operation of the safety relief valve in this condition?

a. The valve will only operate in the safety mode.

b. The valve will only operate automatically in the ADS mode.

c. The valve will only operate automatically in the ADS mode or the safety mode.

d. The valve will operate automatically in the ADS mode or the safety mode and can be manually operated from the control room.

QUESTION: 031 (1.00)

Refueling operations are in progress. A fuel assembly is being removed from location 19-22 in the core. Technical Specifications require an operable SRM in the quadrant where the fuel assembly is being moved and an operable SRM in an adjacent quadrant. WHICH ONE of the following situations meets the Technical Specification core monitoring requirements? (A core map is attached.)

a. SRM 'A' is bypassed.
SRM 'B' is fully inserted and reading 5 cps.
SRM 'C' is fully inserted and reading 2 cps.
SRM 'D' is fully inserted and reading 8 cps.

b. SRM 'A' is bypassed.
SRM 'B' is fully inserted and reading 10 cps.
SRM 'C' is bypassed.
SRM 'D' is fully inserted and reading 4 cps.

c. SRM 'A' is fully inserted and reading 6 cps.
SRM 'B' is bypassed.
SRM 'C' is bypassed.
SRM 'D' is fully inserted and reading 9 cps.

d. SRM 'A' is fully inserted and reading 12 cps.
SRM 'B' is fully inserted and reading 4 cps.
SRM 'C' is fully inserted and reading 7 cps.
SRM 'D' is bypassed.

QUESTION: 032 (1.00)

A normal plant startup is in progress with the mode switch in STARTUP. IRM A is failed downscale and bypassed with its function switch on panel 936 in STBY. All the IRM range switches are on Range 2. WHICH ONE of the following describes the expected plant response if IRM A is taken out of bypass?

a. No action

b. Rod block

c. Rod block and half scram

d. Rod block and full scram

QUESTION: 033 (1.00)

The Transverse Incore Probe (TIP) detector was performing an automatic scan when power was lost to 120 V lighting panel 17L. Simultaneous with the loss of power, the reactor scrammed and RPV water level dropped to 0 inches. WHICH ONE of the following describes the response of the TIP System?

a. The TIP detector will remain in the core, the ball valve will remain open, and the shear valve must be fired manually.

b. The TIP detector will remain in the core, the ball valve will close, and the shear valve will fire automatically.

c. The TIP detector will retract, the ball valve will remain open, and the shear valve must be fired manually.

d. The TIP detector will retract, the ball valve will close, and it is not necessary to fire the shear valve.

QUESTION: 034 (1.00)

With the plant operating at 90% power, the APRM Calibration section of the OD-3 Printout provided the following results:

| APRM | 1(A) | 2(C) | 3(E) | 4(B) | 5(D) | 6(F) |
|---|---|---|---|---|---|---|
| READING | 91.1 | 90.3 | 88.2 | 90.0 | 88.7 | 89.4 |
| AGAF | 0.975 | 1.102 | 0.994 | 1.003 | 0.987 | 1.058 |

WHICH ONE of the following identifies all the APRMs that require calibration?

a. APRM A, APRM D, and APRM E

b. APRM D, APRM E, and APRM F

c. APRM A, APRM C, and APRM F

d. APRM B, APRM C, and APRM F

QUESTION: 035 (1.00)

The plant is operating at 95% power. The following are the indications received when the APRM meter function switches on Panel 937 are placed in the AVERAGE, COUNT, and FLOW positions:

| | AVERAGE | COUNT | FLOW |
|---|---|---|---|
| APRM A | 98% | 70% | 85% |
| APRM B | 94% | 60% | 97% |
| APRM C | 96% | 80% | 85% |
| APRM D | 99% | 55% | 97% |
| APRM E | 93% | 55% | 85% |
| APRM F | 95% | 70% | 97% |

WHICH ONE of the following describes the expected plant response for these conditions?

a. No action

b. Rod block

c. Rod block and half scram

d. Rod block and full scram

QUESTION: 036 (1.00)

The plant was operating at 100% power when the 'A' Recirculation pump tripped. All of the appropriate actions were taken in response to the recirc pump trip. The following annunciators are lit:

"ROD WITHDRAW BLOCK" (C905L, A1)
"ROD BLOCK MONITOR DOWNSCALE" (C905R, F3)

The RBM indicates 75% for the selected rod. WHICH ONE of the following describes the cause of these alarms?

a. A voltage transient during the recirc pump trip caused the RBM to momentarily lower below 5/125 of scale causing the downscale trip.

b. As core power decreased, the local power around the selected rod also decreased. This caused the RBM output signal to decrease below 94% of the reference signal causing the downscale trip.

c. Prior to the transient, the RBM High Power Trip Setpoint was activated for the selected rod. The transient caused power to decrease prior to leveling out at 75%. As reactor power lowered into the Low Power Trip Setpoint, the downscale trip was actuated.

d. The recirc pump trip caused 'A' recirc loop flow to decrease. As a result, the 'A' flow converter input a downscale signal into the 'A' RBM causing the downscale trip.

QUESTION: 037 (1.00)

Reactor power is at 16% during a reactor startup. Control rod 30-15 is at position 12 and has been selected to be withdrawn to position 48. The reed switch for Rod 30-15, position 32 is faulty and will not actuate. WHICH ONE of the following describes the expected response if the rod movement control switch is taken to the NOTCH OUT position simultaneously with the "Emergency in/notch override" switch being taken to the NOTCH OVERRIDE position?

a. No rod motion will occur.

b. A rod block will be initiated at position 32 and Rod 30-15 will settle at position 32.

c. A rod block will be initiated at position 32 and Rod 30-15 will settle at position 34.

d. Rod 30-15 will withdraw to position 48.

QUESTION: 038 (1.00)

Control rods are being withdrawn with reactor power on IRM range 4, in accordance with the attached Control Rod Sequence Sheet. Rod Group No. 8 is latched. WHICH ONE the following manipulations will result in a RWM select error?

a. Rod 14-39 is at position 08
   Rod 38-39 is at position 06
   Rod 38-15 is at position 04
   Rod 14-15 is at position 14
   Rod 14-39 is selected

b. Rod 14-39 is at position 10
   Rod 38-39 is at position 12
   Rod 38-15 is at position 06
   Rod 14-15 is at position 06
   Rod 38-39 is selected

c. Rod 14-39 is at position 04
   Rod 38-39 is at position 06
   Rod 38-15 is at position 04
   Rod 14-15 is at position 12
   Rod 38-15 is selected

d. Rod 14-39 is at position 14
   Rod 38-39 is at position 08
   Rod 38-15 is at position 10
   Rod 14-15 is at position 04
   Rod 14-15 is selected

QUESTION: 039 (1.00)

A 1st point feedwater heater was just removed from service. There has been no change in turbine steam flow. WHICH ONE of the following describes the effect on the plant?

a. Main generator output decreases. Plant efficiency increases.

b. Main generator output decreases. Plant efficiency decreases.

c. Main generator output increases. Plant efficiency increases.

d. Main generator output increases. Plant efficiency decreases.

QUESTION: 040 (1.00)

The plant was operating at 100% power when the "RFP A LOW NPSH" (C1L, A1) annunciator is received. The RFP sequential trip selector switch is "ON" and the reactor feed pump tripping sequence switch is in the "CAB" position. WHICH ONE of the following describes the potential cause of the alarm and the expected automatic actions?

a. The alarm was caused by a trip of one condensate pump. The condenser reject valves open. RFP A will trip if the low NPSH condition persists for 15 seconds.

b. The alarm was caused by a trip of one condensate pump. The condenser reject valves shut. RFP C will trip if the low NPSH condition persists for 15 seconds.

c. The alarm was caused by placing a condensate demineralizer in service. The condenser reject valves shut. RFP A will trip if the low NPSH condition persists for 15 seconds.

d. The alarm was caused by removing a condensate demineralizer from service. The condenser reject valves open. RFP C will trip if the low NPSH condition persists for 15 seconds.

QUESTION: 041 (1.00)

The plant is operating at 100% power with the FWLC setpoint set at +28 inches. WHICH ONE of the following describes the effect of a loss of one feedwater flow input to the FWLC System?

a. Feedwater flow decreases, reactor water level decreases, and the reactor scrams.

b. Feedwater flow decreases and reactor water level decreases, but the reactor should not scram.

c. Feedwater flow increases, reactor water level increases, and the reactor may scram.

d. Feedwater flow increases and reactor water level increases, but the reactor should not scram.

QUESTION: 042 (1.00)

The plant was operating at 100% power with the MHC System operating normally when the Bypass Opening Jack (BOJ) was taken to the RAISE position. WHICH ONE of the following describes the expected plant response if the BOJ cannot be returned to the OFF position?

a. The bypass valves will open fully. Reactor pressure will decrease and stabilize at a slightly lower value than before the transient.

b. The bypass valves will open fully, then the control valves will open until the control valve limit stop is reached. Reactor pressure will decrease and stabilize at a lower value than before the transient.

c. The control valves will open until the control valve limit stop is reached, then the bypass valves will open fully. Reactor pressure will decrease until the MSIVs close.

d. The control valves will open until the control valve limit stop is reached, then the bypass valves will open until the reactor flow limit is reached. Reactor pressure will decrease until the MSIVs close.

QUESTION: 043 (1.00)

A turbine runback was initiated due to high stator outlet coolant temperature at time zero. The following table shows the generator load and stator outlet coolant temperature response during the runback:

| TIME | STATOR AMPS | OUTLET TEMPERATURE |
|---|---|---|
| +0.5 minutes | 25,300 | 88°C |
| +1.0 minutes | 21,800 | 86°C |
| +1.5 minutes | 18,300 | 84°C |
| +2.0 minutes | 14,800 | 82°C |
| +2.5 minutes | 11,300 | 80°C |
| +3.0 minutes | 7,800 | 78°C |
| +3.5 minutes | 4,300 | 76°C |
| +4.0 minutes | 800 | 74°C |

WHICH ONE of the following correctly describes the plant response?

a. The turbine should have tripped at +2.0 minutes.

b. The turbine should have tripped at +3.5 minutes.

c. The runback should have stopped at +1.0 minutes.

d. The runback should have stopped at +3.5 minutes.

QUESTION: 044 (1.00)

The main condenser vapor valves (AO-3703, AO-3704, AO-3710, and AO-3711) have shut automatically while the plant was operating at 100% power. WHICH ONE of the following conditions could have caused the isolation?

a. Offgas high radiation caused by abnormal carbon vault temperature

b. Hydrogen concentration greater than 1 percent caused by a recombiner malfunction

c. Offgas high pressure caused by an explosion in the offgas system

d. High offgas flow caused by lowering condenser vacuum

QUESTION: 045 (1.00)

The plant is in single loop operation. For WHICH ONE of the following conditions can the idle recirculation pump be started? (Consider administrative and functional limitations.)

a. The idle recirculation pump suction and discharge valves are fully open. The scoop tube lock light is off and speed control is in manual with an output setpoint of zero.

b. The idle recirculation MG set generator field breaker is open. Lube oil pressure is 17 psig and lube oil temperature is 90°F.

c. The operating pump is running at 40% of rated speed. Core flow is 35 Mlb/hr. Core thermal power is 1050 MWT.

d. The idle pump suction temperature is 408°F. The operating pump suction temperature is 435°F. Bottom head drain temperature is 492°F. Vessel dome temperature is 548°F.

QUESTION: 046 (1.00)

The plant is operating at 50% power with both recirculation pumps in manual. WHICH ONE of the following describes the expected response if all FWLC System input were lost to Recirc Pump B's flow controller?

a. Recirc Pump B runback to 65% due to a Speed Limiter #2 runback.

b. Recirc Pump B runback to 26% due to a Speed Limiter #1 runback.

c. Recirc Pump B scoop tube will lockup due to a speed control signal failure.

d. There will be no effect on Recirc Pump B because reactor water level is normal and the pump discharge valve is full open.

QUESTION: 047 (1.00)

A LOCA has occurred concurrent with a loss of AC power. Both diesel generators (DGs) started, but the electric governor for DG B malfunctioned immediately after the start signal was received. WHICH ONE of the following describes the expected response of the DG B output breaker and DG B?

a. The DG output breaker will not close because the DG will not come up to speed and voltage. The DG will continue to run.

b. The DG output breaker will close because the mechanical governor will take over. The DG will continue to run.

c. The DG output breaker will close, but will reopen as DG speed increases due to overcurrent. The DG will continue to run.

d. The DG output breaker will close, but the DG will trip on overspeed causing the output breaker to trip open.

deleted

QUESTION: 048 (1.00)

The Standby Gas Treatment (SGT) System initiated on a valid initiation signal 3 minutes ago. Prior to the initiation, both SGT trains were in the normal standby lineup. No operator action has been taken and the initiation signal is still present. The SGT train A heater just tripped due to high temperature. WHICH ONE of the following describes the expected response of the SGT System?

a. SGT train A fan will trip. SGT train A inlet and outlet dampers will close. SGT train B fan will continue to run. SGT train B inlet and outlet dampers will remain open.

b. SGT train A fan will trip. SGT train A inlet and outlet dampers will close. SGT train B fan will start. SGT train B inlet and outlet dampers will open.

c. SGT train A fan will continue to run. SGT train A inlet and outlet dampers will remain open. SGT train B fan will continue to run. SGT train B inlet and outlet dampers will remain open.

d. SGT train A fan will continue to run. SGT train A inlet and outlet dampers will remain open. SGT train B fan will not start. SGT train B inlet and outlet dampers will remain closed.

QUESTION: 9 (1.00)

WHICH ONE of the following conditions will cause a secondary containment isolation?

a. Refuel floor rad monitor 1705-8A: 25 mR/hr
   Refuel floor rad monitor 1705-8B: 15 mR/hr
   Refuel floor rad monitor 1705-8C: 20 mR/hr
   Refuel floor rad monitor 1705-8D: 10 mR/hr

b. Refuel floor rad monitor 1705-8A: 10 mR/hr
   Refuel floor rad monitor 1705-8B: 15 mR/hr
   Refuel floor rad monitor 1705-8C: 20 mR/hr
   Refuel floor rad monitor 1705-8D: 0.1 mR/hr

c. Refuel floor rad monitor 1705-8A: 0.1 mR/hr
   Refuel floor rad monitor 1705-8B: 20 mR/hr
   Refuel floor rad monitor 1705-8C: 0.1 mR/hr
   Refuel floor rad monitor 1705-8D: 0.1 mR/hr

d. Refuel floor rad monitor 1705-8A: 15 mR/hr
   Refuel floor rad monitor 1705-8B: 20 mR/hr
   Refuel floor rad monitor 1705-8C: 0.1 mR/hr
   Refuel floor rad monitor 1705-8D: 0.1 mR/hr

QUESTION: 050 (1.00)

Reactor Building ventilation fans must be started in the proper order to maintain reactor building pressure and prevent the spread of contamination. WHICH ONE of the following describes the proper sequence for starting reactor building ventilation fans?

a. Exhaust fans should be started before supply fans. Contaminated exhaust (Zone 3) fans should be started before clean exhaust (Zone 2) fans.

b. Exhaust fans should be started before supply fans. Clean exhaust (Zone 2) fans should be started before contaminated exhaust (Zone 3) fans.

c. Supply fans should be started before exhaust fans. Clean exhaust (Zone 2) fans should be started before contaminated exhaust (Zone 3) fans.

d. Supply fans should be started before exhaust fans. Contaminated exhaust (Zone 3) fans should be started before clean exhaust (Zone 2) fans.

QUESTION: 051 (1.00)

Control room environmental control supply fan HS-77(VSF-103A) is in STBY and supply fan HS-78(VSF-103B) is in AUTO. WHICH ONE of the following describes the Control Room HVAC System response to a Halon initiation?

a. Both control room environmental control supply fans start. Normal supply and exhaust fans trip. Halon exhaust fan starts and damper AO N-141 shuts.

b. Control room environmental control supply fan HS-78 starts. Normal supply and exhaust fans trip. Cable spreading room supply and exhaust dampers shut.

c. Both control room environmental control supply fans start. Normal air intake dampers close and filtration system isolation dampers open. Cable spreading room supply and exhaust dampers shut.

d. Control room environmental control supply fan HS-78 starts. Normal air intake dampers close and filtration system isolation dampers open. Halon exhaust fan starts and damper AO N-141 shuts.

QUESTION: 052 (1.00)

If the Fuel Pool Cooling (FPC) System is unavailable, WHICH ONE of the following systems can be crosstied for pool cooling?

a. Fire Protection

b. Residual Heat Removal

c. Reactcr Building Closed Cooling Water

d. Condensate and Demineralized Water Storage and Transfer

QUESTION: 053 (1.00)

The reactor mode switch is in REFUEL. In WHICH ONE of the following conditions would control rod withdrawal be possible?

a. One control rod withdrawn to position 02. The withdrawn control rod is selected. The bridge is over the core. The main hoist is loaded with 600 lbs. The grapple is fully up.

b. All control rods are full in. The bridge is NOT over or near the core. The monorail hoist is loaded with 300 lbs. The grapple is NOT fully up.

c. One control rod withdrawn to position 02. A second control rod is selected. The bridge is over the core. The frame mounted hoist is loaded with 200 lbs. The grapple is fully up.

d. All control rods are full in. The bridge is NOT over or near the core. The service platform hoist is loaded with 500 lbs.

QUESTION: 054 (1.00)

WHICH ONE of the following Fire Protection Systems would be utilized to suppress a fuel oil fire in the DG fuel oil pipe trench?

a. Dry Chemical System

b. Halon System

c. Cardox System

d. Fire Water System

QUESTION: 055 (1.00)

IRM 'B' is indicating 10 on Range 7. WHICH ONE of the following values will IRM 'B' indicate if the range switch is placed on Range 6?

a. 10

b. 40

c. 100

d. 125

QUESTION: 056 (1.00)

A startup is in progress with the mode switch in STARTUP. All SRM detectors are fully inserted. WHICH ONE of the following describes the conditions when SRM detector withdrawal should commence?

a. When all IRMs are on Range 3

b. When the heating range is reached

c. When all the retract permissive indicators are illuminated

d. When all required SRM/IRM overlap data has been taken

QUESTION: 057 (1.00)

WHICH ONE of the following 120 VAC loads would receive uninterrupted power on a loss of all offsite power?

a. CRD flow control valves (Y-1)

b. Rod Position Information System (Y-2)

c. HPCI system control (Y-3)

d. APRMs and LPRMs (RPS)

QUESTION: 058 (1.00)

The plant is operating at 100% power with 4160 VAC busses A-1 through A-6 being supplied by the Unit Auxiliary Transformer. The auto transfer switches for busses A-3 and A-4 are in the OFF positions. The auto transfer switches for all the other busses are in the ON position. WHICH ONE of the following lists all of the busses that will automatically transfer to the Startup Transformer on a full reactor scram while the turbine is still on line?

a. A-1, A-2, A-3, A-4, A-5, and A-6

b. A-1, A-2, A-5, and A-6

c. A-5 and A-6

d. None

QUESTION: 059 (1.00)

WHICH ONE of the following correctly completes the following statement concerning the procedure for synchronizing the main generator to the grid and establishing initial loading?

-Turbine speed is adjusted so that the synchroscope is rotating slowly in the ________________ direction.
-The ACB control switch is taken to close when the synchroscope pointer is ________________ on the meter.
-Initial load is picked up by going to _____ on the governor speed control switch.

a. CLOCKWISE, AT 12 o'clock, LOWER

b. CLOCKWISE, 5 degrees BEFORE 12 o'clock, RAISE

c. COUNTERCLOCKWISE, 5 degrees BEFORE 12 o'clock, LOWER

d. COUNTERCLOCKWISE, AT 12 o'clock, RAISE

QUESTION: 060 (1.00)

An ATWS has occurred and SLC has been initiated. Reactor pressure is 1100 psig and reactor power is 25%. WHICH ONE of the following conditions would be indicative of a problem with SLC system operation?

a. Annunciator "LOSS OF CONTINUITY TO SQUIB VALVE" (C905R, B7)

b. Annunciator "STANDBY LIQUID CONT TANK HI/LO LEVEL" (C905R, A7)

c. SLC pump discharge pressure 1075 psig

d. Squib valve amber lights on C905 out

QUESTION: 061 (1.00)

WHICH ONE of the following conditions satisfies the logic for PCIS to recognize RHR in the Shutdown Cooling mode?

a. MO-47 is full open, MO-50 is full open, and reactor pressure is 115 psig.

b. MO-47 is full open, MO-50 is partially open, and reactor pressure is 105 psig.

c. MO-47 is partially open, MO-50 is partially open, and reactor pressure is 95 psig.

d. MO-47 is partially open, MO-50 is full closed, and reactor pressure is 85 psig.

QUESTION: 062 (1.00)

A transient occurred which caused RPV water level to decrease. The following conditions existed as the transient progressed:

| | |
|---|---|
| T=0 minutes | RPV water level dropped below the scram setpoint, but the reactor failed to scram |
| T= 7 minutes | RPV water level dropped below the low-low level setpoint |
| T= 15 minutes | RPV water level is -100 inches<br>RPV pressure is 800 psig<br>Drywell pressure is 2.0 psig<br>The Core Spray pump control switches are placed in the stop position and the control switches for the inboard and outboard injection valves are placed in the close position |

WHICH ONE of the following describes the expected status of the Core Spray System at T=15 minutes?

a. The Core Spray pumps were running and are now stopped. The inboard injection valves are open and the outboard injection valves are closed.

b. The Core Spray pumps were running and are now stopped. The inboard and outboard injection valves are closed.

c. The Core Spray pumps did not start. The inboard injection valves are open and the outboard injection valves are closed.

d. The Core Spray pumps did not start. The inboard and outboard injection valves are closed.

QUESTION: 063 (1.00)

WHICH ONE of the following describes the operation of the RCIC steam supply isolation valves, MO-16 and MO-17?

a. Both valves are full stroke valves for isolation signals and manual closing signals. MO-16 is a jog valve for manual opening and MO-17 is a full stroke valve for manual opening.

b. Both valves are full stroke valves for isolation signals and manual closing signals. MO-16 is a full stroke valve for manual opening and MO-17 is a jog valve for manual opening.

c. Both valves are full stroke valves for isolation signals and manual closing signals. Both valves are jog valves for manual opening.

d. Both valves are full stroke valves for isolation signals, manual closing signals, and manual opening.

QUESTION: 064 (1.00)

Following a loss of offsite power, HPCI was placed in service for pressure control. Prolonged HPCI operation has caused torus water level to increase to +6 inches. WHICH ONE of the following describes the expected response?

a. HPCI cannot be operated in pressure control mode because the HPCI/RCIC test return valve (MO-15) and HPCI full flow test valve (MO-10) automatically close when the HPCI torus suction valves (MO-35 and MO-36) automatically open.

b. The HPCI CST suction valve (MO-6) will automatically close, but the HPCI torus suction valves (MO-35 and MO-36) will not automatically open because there is no initiation signal present. HPCI will trip on low suction pressure.

c. The HPCI minimum flow valve (MO-14) will automatically close, if open, to prevent further increase in torus level. HPCI will continue to operate in pressure control mode.

d. The HPCI torus suction valves (MO-35 and MO-36) will automatically open and the HPCI CST suction valve (MO-6) will automatically close. HPCI will continue to operate in pressure control mode.

QUESTION: 065 (1.00)

No fuel movement is in progress. WHICH ONE of the following conditions would be a violation of Secondary Containment integrity?

a. Reactor Building differential pressure is greater than 0 inches of water in STARTUP.

b. Both reactor building to torus vacuum breakers are inoperable and stuck in the open position in RUN.

c. Both Standby Gas Treatment trains are inoperable in COLD SHUTDOWN with the reactor coolant system vented.

d. Both reactor building airlock doors are open to remove a large piece of equipment from the reactor building in HOT SHUTDOWN.

QUESTION: 066 (1.00)

WHICH ONE of the following conditions would be a safety limit violation?

a. While operating at 30% power, the MHC pressure regulators fail. Reactor pressure drops to 800 psig before the MSIVs close and the reactor scrams.

b. While operating at 75% power, a malfunction in the master recirculation flow controller causes the speed of both recirc pumps to increase to the high speed stop. The Minimum Critical Power Ratio is 1.05.

c. While operating at 90% power, a turbine trip occurs. The reactor fails to scram and the ATWS system logic trips the reactor feed pumps.

d. While operating at 100% power, an inadvertent MSIV closure causes reactor pressure to increase. The reactor scrams and reactor pressure increases until both safety valves lift.

QUESTION: 067 (1.00)

WHICH ONE of the following describes the expected indications if Jet Pump 5 failed?

a. Recirc Loop A flow increases. Total core flow decreases.

b. Recirc Loop A flow increases. Total core flow increases.

c. Recirc Loop A flow decreases. Total core flow decreases.

d. Recirc Loop A flow decreases. Total core flow increases.

QUESTION: 068 (1.00)

The plant was operating at 96% power on the 100% load line for several months when Recirc Pump A tripped due to operator error. Plant conditions following the trip are:

| | |
|---|---|
| Reactor Power: | 60% |
| Recirc Loop A Flow: (FI-263-107A) | 2.0 MLB/HR |
| Recirc Loop B Flow: (FI-263-107B) | 30.5 MLB/HR |
| Total Core Flow: (FR-263-110) | 32.5 MLB/HR |

WHICH ONE of the following describes the appropriate action in accordance with PNPS Proc. No. 2.4.17, "Recirculation Pump(s) Trip," (attached)?

a. Increase the speed of Recirc Pump B until loop flow is greater than 31.5 MLB/HR.

b. Insert control rods to decrease reactor power below the 80% load line.

c. Restart the idle pump in accordance with PNPS 2.2.84, "Reactor Recirculation System."

d. Scram the reactor and concurrently perform PNPS 2.1.6, "Reactor Scram."

QUESTION: 069 (1.00)

The plant was operating at 100% power when a break developed on the instrument air header. WHICH ONE of the following describes automatic actions that should occur before air header pressure drops below 75 psig?

a. Service air header isolates
Backup K104 air compressor starts
Feedwater control valves lock up

b. Non-essential instrument air isolates
Feedwater control valves lock up
Scram pilot valve air header depressurizes

c. Drywell pneumatic supply header isolates
Lagging compressor loads
Service air header isolates

d. Low pressure service air crossconnect isolates (if open)
Backup K104 air compressor starts
Non-essential instrument air isolates

QUESTION: 070 (1.00)

The plant was operating at 100% power when a complete loss of instrument air occurred. Power was also lost to the AC solenoid operated air cutoff valves for the inboard MSIVs and to the DC solenoid operated air cutoff valves for the outboard MSIVs. The primary containment is inerted. WHICH ONE of the following describes the response of the MSIVs?

a. Inboard MSIVs remain open. Outboard MSIVs remain open.

b. Inboard MSIVs close due to air pressure. Outboard MSIVs close due to air pressure.

c. Inboard MSIVs close due to spring pressure. Outboard MSIVs remain open.

d. Inboard MSIVs remain open. Outboard MSIVs close due to spring pressure.

QUESTION: 071 (1.00)

An inadvertent containment isolation signal was received and immediately cleared. No operator action was taken. WHICH ONE of the following sets of valves do NOT require a manual reset to reopen the valves if they isolated on the specified signal?

| Valves | Isolation Signal |
|---|---|
| a. Recirc sample lines | Main Steam Line low pressure |
| b. HPCI exhaust vacuum breaker isolation valves | High drywell pressure (w/ low reactor pressure) |
| c. RCIC inlet steam valves | Low reactor pressure |
| d. RHR/LPCI Injection valves | High reactor pressure (in Shutdown Cooling mode) |

QUESTION: 072 (1.00)

Three RBCCW pumps are unavailable. WHICH ONE of the following describes the appropriate action that must be taken with the specified pumps unavailable?

a. RBCCW pumps A, B, and C are unavailable. The RBCCW loops must be crosstied to prevent a reactor scram on high drywell pressure due to a loss of drywell cooling.

b. RBCCW pumps A, C, and E are unavailable. The RBCCW loops must be crosstied to prevent a trip of the recirculation pumps due to loss of cooling.

c. RBCCW pumps B, D, and F are unavailable. It is not necessary to crosstie the RBCCW loops because at least one pump in each loop is still available.

d. RBCCW pumps D, E, and F are unavailable. It is not necessary to crosstie the RBCCW loops because at least one pump in each loop is still available.

QUESTION: 073 (1.00)

The Salt Service Water (SSW) System was operating with SSW Pumps A, C, and D running and SSW Pumps B and E in AUTO. The loop selector switch is in its normal position. A complete loss of normal AC power occurs simultaneously with a loss of coolant accident (LOCA). WHICH ONE of the following describes the response of the SSW System with no operator action?

a. TBCCW heat exchanger outlet valves throttle 90% closed.
RBCCW heat exchanger outlet valves open fully.
SSW Pumps A and D restart after load shedding.

b. TBCCW heat exchanger outlet valves throttle 90% closed.
RBCCW heat exchanger outlet valves open fully.
SSW Pumps B and E start after load shedding.

c. TBCCW heat exchanger outlet valves open fully.
RBCCW heat exchanger outlet valves throttle 90% closed.
SSW Pumps A and C restart after load shedding.

d. TBCCW heat exchanger outlet valves open fully.
RBCCW heat exchanger outlet valves throttle 90% closed.
SSW Pumps B and C start after load shedding.

QUESTION: 074 (1.00)

The plant is operating at 800 MWt. Core flow is 35 MLB/HR. Condenser vacuum is 18" Hg. WHICH ONE of the following describes the appropriate actions to be taken under these conditions?

a. Reduce recirculation pump speed, maintaining core flow above 31.5 MLB/HR, until the vacuum decrease is terminated.

b. Reduce recirculation pump speed until the vacuum decrease is terminated OR the pumps reach minimum speed.

c. Insert control rods in reverse order of the pull sheet as necessary to stop the vacuum decrease.

d. Immediately scram the reactor and trip the turbine.

...ting at 95% power when the Train A 4th point
...isolated due to a tube rupture. Reactor power
...%. WHICH ONE of the following describes the appropriate
...ns?

...ck recirculation flow and insert control rods as necessary
...aintain reactor power below 95%.

...nback recirculation flow and insert control rods as necessary
...o maintain reactor power below 25%.

Runback recirculation flow until reactor power is below 75% or
core flow reaches 31.5 Mlb/hr.

d. Runback recirculation flow until reactor power reaches 70% or
core flow reaches 31.5 Mlb/hr.

...ON: 076 (1.00)

...ant is operating at 100% power when the in-service CRD flow
...l valve malfunctions causing CRD system flow to oscillate. WHICH
... the following subsequent CRD system failures would require an
...te reactor scram?

...Trip of both CRD pumps

...oss of all rod position indication

...re than one control rod in a 9-rod array drifts

...e than one CRD mechanism high temperature alarm in a 9-rod
...ay

QUESTION: 075 (1.00)

The plant was operating at 95% power when the Train A 4th point feedwater heater isolated due to a tube rupture. Reactor power increased to 100%. WHICH ONE of the following describes the appropriate immediate actions?

a. Runback recirculation flow and insert control rods as necessary to maintain reactor power below 95%.

b. Runback recirculation flow and insert control rods as necessary to maintain reactor power below 25%.

c. Runback recirculation flow until reactor power is below 75% or core flow reaches 31.5 Mlb/hr.

d. Runback recirculation flow until reactor power reaches 70% or core flow reaches 31.5 Mlb/hr.

QUESTION: 076 (1.00)

The plant is operating at 100% power when the in-service CRD flow control valve malfunctions causing CRD system flow to oscillate. WHICH ONE of the following subsequent CRD system failures would require an immediate reactor scram?

a. Trip of both CRD pumps

b. Loss of all rod position indication

c. More than one control rod in a 9-rod array drifts

d. More than one CRD mechanism high temperature alarm in a 9-rod array

QUESTION: 077 (1.00)

Core offloading is in progress. A spent fuel bundle is grappled over the core, when the Refueling Floor Area Radiation Monitor (ARM) alarms. No other alarms have been received on the refuel floor or in the Control Room. WHICH ONE of the following describes the appropriate actions?

a. Lower the fuel bundle into its designated location in the spent fuel pool.

b. Lower the fuel bundle into the nearest open location in the spent fuel pool.

c. Lower the fuel bundle into the nearest open location in the reactor vessel.

d. Lower the fuel bundle into the location from which it was removed in the reactor vessel.

QUESTION: 078 (1.00)

An electrical failure has occurred causing multiple alarms to annunciate. One of the illuminated alarms is annunciator "VITAL INST SYS LOSS OF DC POWER," (C3 Center, window D5). Based on this information, WHICH ONE of the following describes the emergency procedures that are applicable for this situation?

a. PNPS Proc. No. 5.3.6, "Loss of Vital AC (Y-2)" and PNPS Proc. No. 5.3.13, "Loss of Essential DC Bus D6"

b. PNPS Proc. No. 5.3.6, "Loss of Vital AC (Y-2)" and PNPS Proc. No. 5.3.30, "Loss of 250V DC Bus D-10"

c. PNPS Proc. No. 5.3.7, "Loss of Instrument Power Bus (Y-1)" and PNPS Proc. No. 5.3.13, "Loss of Essential DC Bus D6"

d. PNPS Proc. No. 5.3.7, "Loss of Instrument Power Bus (Y-1)" and PNPS Proc. No. 5.3.30, "Loss of 250V DC Bus D-10"

QUESTION: 079 (1.00)

A station blackout has occurred and the SBO diesel failed to start. RCIC and HPCI were being used to control RPV level and pressure when both systems tripped and could not be restarted. PNPS Proc. No. 5.3.31, "Station Blackout," directs use of PNPS Proc. No. 5.3.26, "RPV Injection During Emergencies," for alternate methods. WHICH ONE of the methods could be used for RPV injection in this situation?

a. SSW crosstied to RHR

b. Fire Water crosstied to RHR

c. Condensate Transfer crosstied to ECCS fill lines

d. Demineralized Water Transfer crosstied to SBLC

QUESTION: 080 (1.00)

The Control Room has been evacuated due to a fire in the Cable Spreading Room. At WHICH ONE of the following locations would you find an Alternate Shutdown toolbox?

a. 3′ Aux Bay

b. 23′ RPS MG Room

c. 37′ Switchgear Room

d. 51′ Turbine Building

QUESTION: 081 (1.00)

Following a reactor scram, the operator is required to determine whether or not all control rods are inserted to or beyond position 02 in order to determine the appropriate actions to be taken in accordance with the EOPs and PNPS Proc. No. 2.1.6, "Reactor Scram." WHICH ONE of the following is an acceptable method for determining whether all rods are fully inserted?

a. Observing that all the blue scram lights are illuminated on the full-core display.

b. Verifying that all APRMs are downscale and reactor power is trending down on the IRMS.

c. Prior to scram reset, verifying that the rod drift annunciator will not clear.

d. With the mode switch in REFUEL, taking the rod select power off, then back on, and observing if the Refuel Mode Select Permissive light illuminates.

QUESTION: 082 (1.00)

EOP-02, "RPV Control, Failure-to-Scram," directs actions based on whether or not the reactor is shutdown. For WHICH ONE of the following conditions would the reactor be considered shutdown in accordance with EOP-02?

a. Reactor pressure is 0 psig. The Hot Shutdown Boron Weight of sodium pentaborate has been injected into the RPV.

b. Reactor power is 20 on Range 5. Reactor period is +100 and stable.

c. Reactor power is 50 on Range 6. Reactor period is negative.

d. Reactor power is 30 on Range 9. Reactor period is negative.

QUESTION: 083 (1.00)

A reactor scram has occurred and all control rods are not inserted. WHICH ONE of the following indications would be expected if the failure of the rods to insert was due to an electrical failure of the Reactor Protection System (RPS)?

a. Alarm "SCRAM VALVE PILOT HEADER LO PRESSURE" (C905R, A6) illuminated.

b. Alarm "SCRAM DISCH VOLUME HI LEVEL SCRAM" (C905R, I4) illuminated.

c. Group Scram logic lights on C905 illuminated.

d. Blue Scram lights on C905 illuminated.

QUESTION: 084 (1.00)

A reactor scram has occurred and all control rods are not inserted. In accordance with PNPS Procedure No. 5.3.23, "Alternate Rod Insertion," WHICH ONE of the following methods for inserting control rods requires the scram to be reset?

a. Venting the overpiston areas of the control rod drives.

b. Individually scramming control rods from panel C916.

c. Venting the Scram air header and deenergizing the Scram solenoids.

d. Inserting a manual Scram from panel C905.

QUESTION: 085 (1.00)

WHICH ONE of the following actions should be taken to maximize drywell cooling in accordance with EOP-03, "Primary Containment Control?"

a. Initiate drywell spray.

b. Secure the recirculation pumps.

c. Increase the RBCCW system temperature.

d. Increase the RBCCW system flow rate.

QUESTION: 086 (1.00)

WHICH ONE of the following lineups would maximize torus cooling in accordance with EOP-03, "Primary Containment Control?"

a. Both RHR loops in service with one pump per loop operating and the RHR heat exchanger bypass valves shut.

b. One RHR loop in service with both pumps operating and the RHR heat exchanger bypass valve shut.

c. Both RHR loops in service with one pump per loop operating and the RHR heat exchanger bypass valves open.

d. One RHR loop in service with both pumps operating and the RHR heat exchanger bypass valve open.

QUESTION: 087 (1.00)

A station blackout and complete loss of instrument air has occurred. Plant conditions are as follows:

- -Drywell pressure is 3 psig and increasing slowly
- -Torus water level is 220 inches
- -Neither Standby Gas Treatment (SGTS) train is operable because the outlet dampers are stuck in the closed position

WHICH ONE of the following describes the reason that the Direct Torus Vent (DTV) cannot be used in this situation?

a. SGTS must have an operable vent path for use of the DTV.

b. Primary Containment conditions do not meet the criteria for use of the DTV.

c. The DTV path is not available during a station blackout.

d. The DTV path is not available during a loss of instrument air.

QUESTION: 088 (1.00)

Both torus water level indicators are upscale and the primary containment water level indicator is downscale. For WHICH ONE of the following conditions is this an accurate indication? (PNPS Proc. No. 5.3.27, "Determining Primary Containment Water Level," is attached.)

a. Drywell Wide Range Pressure: 25 psig
   Torus Bottom Pressure: 10 psig

b. Drywell Wide Range Pressure: 5 psig
   Torus Bottom Pressure: 35 psig

c. Drywell Wide Range Pressure: 20 psig
   Torus Bottom Pressure: 40 psig

d. Drywell Wide Range Pressure: 26 psig
   Torus Bottom Pressure: 20 psig

QUESTION: 089 (1.00)

Following a major loss of coolant accident and loss of off-site power, RPV level cannot be maintained above TAF. The operating crew determines that an alternate injection system needs to be aligned in order to recover RPV level. Reports from the field indicate that the aux. bay is inaccessible. WHICH ONE of the following lineups could be used to deliver a high capacity flowrate to the vessel given the current plant conditions?

a. SSW crosstied to RHR

b. Fire water crosstied to RHR

c. Fire water crosstied to Feedwater

d. Demineralized water transfer crosstied to SBLC

QUESTION: 090 (1.00)

A reactor startup is in progress. The turbine is at 600 RPM when the TURBINE HIGH VIBRATION (C2 Left, window D4) alarm is received. At WHICH ONE of the following vibration levels should the turbine be tripped?

a. 8 mils

b. 12 mils

c. 14 mils

d. 32 mils

QUESTION: 091 (1.00)

The plant was operating at 100% power when all service water was lost due to an unisolable pipe break. WHICH ONE of the following describes the appropriate immediate actions in accordance with PNPS Proc. No. 5.3.3, "Loss of All Service Water?"

a. Isolate the RWCU System and monitor CRD temperatures.

b. Reduce reactor power as necessary to maintain equipment temperatures within operating limits.

c. Reduce reactor power and trip all but one Feedwater pump.

d. Scram the reactor and trip all the Feedwater pumps.

QUESTION: 092 (1.00)

WHICH ONE of the following would be a concern if Jet Pump 5 failed?

a. The cross-sectional flow area for blowdown following a double-ended recirculation line break would decrease.

b. It could preclude the capability of maintaining 2/3 core coverage during a loss of coolant accident.

c. Failure of the jet pump body could cause a subsequent failure of the jet pump nozzle.

d. Resulting inaccuracies in indicated core flow would affect thermal limit calculations nonconservatively.

QUESTION: 093 (1.00)

WHICH ONE of the following is the desired sequence for operation of the Safety Relief Valves (SRVs)?

a. A-B-C-D

b. B-C-D-A

c. A-C-B-D

d. B-D-C-A

QUESTION: 094 (1.00)

The plant was operating at 60% power when the MSIVs closed on low pressure. Several rods failed to insert and all the APRMs are downscale. WHICH ONE of the following describes the appropriate actions with respect to operation of the recirculation pumps?

a. Leave the recirc pumps operating at the present speed.

b. Runback the recirc pumps to minimum.

c. Runback the recirc pumps to minimum, then trip the recirc pumps.

d. Trip the recirc pumps immediately.

QUESTION: 095 (1.00)

The plant was operating at 100% power when the "MAIN STEAM LINE HI RADIATION SCRAM," (C905R, A5) alarm was received due to a valid signal. The MSIVs closed, but the reactor failed to scram. WHICH ONE of the following describes the appropriate actions for RPV pressure control?

a. Depressurize the RPV at less than 100°F/hr using the main turbine bypass valves.

b. Depressurize the RPV at less than 100°F/hr using HPCI in full flow test and the SRVs.

c. Maintain reactor pressure less than 1085 psig using the main turbine bypass valves.

d. Maintain reactor pressure less than 1085 psig using HPCI in full flow test and the SRVs.

QUESTION: 096 (1.00)

The isolation switches for DG A on the alternate shutdown panels are in the LOCAL position. A transient occurs which causes drywell pressure to increase to 5.0 psig. Subsequently, the DG A jacket water temperature increases to 200°F. WHICH ONE of the following describes the expected response of DG A to this event?"

a. DG A does not start.

b. DG A starts, then automatically trips on high jacket water temperature.

c. DG A starts, continues to run with high jacket water temperature, but can be tripped manually by depressing the local stop pushbutton on panel C103.

d. DG A starts, continues to run with high jacket water temperature, and cannot be tripped manually by depressing the local stop pushbutton on panel C103.

QUESTION: 097 (1.00)

Reactor Building HVAC is operating and no vacuum breakers are open. Current plant conditions are:

| | |
|---|---|
| - Drywell pressure (PID-5067A): | +1.7 psi |
| - Torus pressure (PID-5067B): | -0.5 psi |
| - Reactor Building pressure (manometer on Panel C-61): | 0 inches of water (pegged high) |
| - Reactor Building dp - north (DPI-8117 on Panel C-7): | 0 inches of water (pegged high) |
| - Reactor Building dp - south (DPI-8128 on Panel C-7): | 0 inches of water (pegged high) |

WHICH ONE of the following lists the documents that should be referenced for these conditions?"

a. EOP-03, "Primary Containment Control" and EOP-04, "Secondary Containment Control"

b. EOP-03, "Primary Containment Control" and suppression chamber to drywell vacuum breaker Tech Specs

c. EOP-04, "Secondary Containment Control" and reactor building to torus vacuum breaker Tech Specs

d. Suppression chamber to drywell vacuum breaker Tech Specs and reactor building to torus vacuum breaker Tech Specs

QUESTION: 098 (1.00)

WHICH ONE of the following alarms would require entry into EOP-04, "Secondary Containment Control?"

a. "STACK GAS HI-HI RADIATION" (903R, C2)

b. "REFUELING FLOOR AREA HI RADIATION" (903R, B4)

c. "BUILDING EXHAUST HI RADIATION" (904L, F3)

d. "STANDBY GAS TREATMENT DISCHARGE HI RADIATION" (904L, A3)

QUESTION: 099 (1.00)

WHICH ONE of the following systems would be considered a primary system when implementing EOP-04, "Secondary Containment Control?"

a. Fuel Pool Cooling

b. Reactor Building Closed Cooling Water

c. Reactor Water Cleanup

d. Primary Containment Cooling

QUESTION: 100 (1.00)

A fuel cladding failure has occurred, causing offgas radiation levels to increase. Keylock switch 17A-S12 on panel CP600 is in POSITION-2. The Offgas Adsorber System is in AUTO. WHICH ONE of the following describes the expected response of the Offgas System?

a. The charcoal adsorber bypass valve shuts and the inlet valve opens when the SJAE Offgas radiation monitors trip on high radiation. The Offgas isolation and holdup line drain valves close 13 minutes after the AOG post treatment radiation monitors trip on high radiation.

b. The charcoal adsorber bypass valve shuts and the inlet valve opens when the AOG post treatment radiation monitors trip on high radiation. The Offgas isolation and holdup line drain valves close 13 minutes after the SJAE Offgas radiation monitors trip on high radiation.

c. The charcoal adsorber bypass valve shuts and the inlet valve opens when the AOG post treatment radiation monitors trip on high radiation. The Offgas isolation and holdup line drain valves close when the Main Stack radiation monitors trip on high radiation.

d. The charcoal adsorber bypass valve shuts and the inlet valve opens when the SJAE Offgas radiation monitors trip on high radiation. The Offgas isolation and holdup line drain valves close 13 minutes after the Main Stack radiation monitors trip on high radiation.

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

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ANSWER: 001 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-11-01, "Control Room Computer System (EPIC/SPDS)," ELO 8.

[3.2/3.4]

294001A115 ..(KA's)

ANSWER: 002 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 6.1-024, "Radiological Posting of Areas of the Station."
2. Module C-GT-02-02-01, "Introduction to Radiation Protection," ELO 4.

[3.3/3.8]

294001K103 ..(KA's)

ANSWER: 003 (1.00)

b
~~c.~~

REFERENCE:

1. Module C-GT-02-02-02, "Exposure Limits," ELO 2.

[3.3/3.8]

294001K103 ..(KA's)

ANSWER: 004 (1.00)

a.

REFERENCE:

1. PNPS Proc. No. 1.4.9, "Storage, Handling, and Disposal of Sodium Pentaborate," page 6.
2. UG: O-RO-04-04, "Emergency Tasks," Task 91.

[3.1/3.4]

294001K110 ..(KA's)

ANSWER: 005 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 1.4.36, "High Pressure/Compressed Gas Cylinder Control."
2. IG: C-GT-01-01-03, "Industrial Safety," LO 30.

[3.4/3.8]

294001K109 ..(KA's)

ANSWER: 006 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 1.3.34, "Conduct of Operations," pages 38 and 39.

[3.7/3.7]

294001K101 ..(KA's)

ANSWER: 007 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 1.4.5, "PNPS Tagging Procedure."
2. UG: O-RO-04-02, "Administrative Tasks," Task 8.

[3.9/4.5]

294001K102 ..(KA's)

ANSWER: 008 (1.00)

c.

ANSWER: 011 (1.00)

d. or b.

REFERENCE:

1. IG: O-RO-03-04-02, "EOP Development and Use," ELO 25.

[4.2/4.2]

294001A102 ..(KA's)

ANSWER: 012 (1.00)

c.

REFERENCE:

1. IG: T-ER-01-01-11, "Emergency Plan Training for Licensed Operators and Shift Technical Advisors," ELO to be added (per A. Shiever).

[2.9/4.7]

294001A116 ..(KA's)

ANSWER: 013 (1.00)

d.

REFERENCE:

1. IG: T-ER-01-01-11, "Emergency Plan Training for Licensed Operators and Shift Technical Advisors," ELO 3.

[2.9/4.7]

294001A116 ..(KA's)

ANSWER: 014 (1.00)

d.

REFERENCE:

1. PNPS Technical Specifications 3.2.H and 3.7.A.
2. OJT Guide 10, Technical Specification Objective.

[3.3/4.2]

223001G011 ..(KA's)

ANSWER: 015 (1.00)

d.

REFERENCE:

1. IG: O-RO-02-04-02, "Condensate and Feedwater System," ELOs 61 and 86.
2. IG: O-RO-02-04-01, "Main Steam System," ELO 19.
3. IG: O-RO-02-06-03, "Control Rod Drive System," ELO 29.
4. IG: O-RO-02-07-02, "Reactor Protection System and Anticipated Transient Without Scram System," ELOs 14 and 23.
5. Modified question from 11/26/90 NRC Exam (modified/replaced distractors and reworded correct answer).

[3.2/3.0]

239003G008 ..(KA's)

ANSWER: 016 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-06-05, "Reactor Water Cleanup System," ELO 9.

[3.4/3.3]

204000G007 ..(KA's)

ANSWER: 017 (1.00)

a.

REFERENCE:

1. IG: O-RO-02-06-01, "Non-Nuclear Instrumentation and Reactor Vessel Internals," ELO 5.
2. Modified question from 11/26/90 NRC Exam (changed conditions).

[3.6/3.8]

216000K122 ..(KA's)

ANSWER: 018 (1.00)

d.

REFERENCE:

1. IG: O-RO-02-06-01, "Non-Nuclear Instrumentation and Reactor Vessel Internals," ELO 10.
2. Systems Reference Text: Nuclear Boiler Instrumentation.
3. LO Requalification IG: Reference Leg Perturbations Due to Non-Condensables.
4. Modified questions from 11/26/90 and 12/2/91 NRC Exams (combined concepts of questions and changed correct answer).
[3.3/3.5]

216000A210 ..(KA's)

ANSWER: 019 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-07-02, "Reactor Protection System and Anticipated Transient Without Scram System," ELO 22.

[3.8/4.2]

212000G010 ..(KA's)

ANSWER: 020 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 2.2.79, "Reactor Protection System," page 14.
2. IG: O-RO-02-07-02, "Reactor Protection System and Anticipated Transient Without Scram System," ELO 16.
3. Facility Question TYPA-13 (from Requal Retake Exam).

[3.9/4.1]

212000K412 ..(KA's)

ANSWER: 021 (1.00)

c.

REFERENCE:

1. PNPS: ARP-905R-D4, Rev. 8.
2. IG: O-RO-02-04-01, "Main Steam System," page IG-19-7/90, ELOs 18 and 211.
3. Modified question from 11/26/90 NRC Exam (changed conditions).

[4.0/4.1]

239001K127 ..(KA's)

ANSWER: 022 (1.00)

a.

REFERENCE:

1. IG: O-RO-02-08-01, "Primary Containment System," ELOs 33 and 34.
2. Modified questions from 11/26/90 and 12/02/91 NRC Exams (changed conditions, distractors, and correct answer).

[3.5/3.5]

223002A302 ..(KA's)

ANSWER: 023 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-09-04, "Reactor Core Isolation Cooling System," ELO 11.
2. Modified question from 11/26/90 NRC Exam (changed conditions, distractors and correct answer).

[3.8/3.8]

217000A215 ..(KA's)

ANSWER: 024 (1.00)

c.

REFERENCE:

1. IG: O-RO-03-04-02, "EOP Development and Use," ELO 22.

[3.9/3.8]

206000G010 ..(KA's)

ANSWER: 025 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-09-02, "Core Spray System," ELO 12.

[3.3/3.6]

209001A205 ..(KA's)

ANSWER: 026 (1.00)

c

REFERENCE:

1. IG: O-RO-02-09-01, "Low Pressure Coolant Injection and Residual Heat Removal," ELOs 9 and 11.
2. Modified questions from 11/26/90 and 12/2/91 NRC Exams (changed conditions).

[4.2/4.3]

203000A101 ..(KA's)

ANSWER: 027 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 2.2.19, "Residual Heat Removal," page 19.
2. UG: O-RO-04-04, "Emergency Tasks," Task 89c.
3. Modified question from 11/26/90 NRC Exam (changed values in distractors).

[3.2/3.4]

226001G010 ..(KA's)

ANSWER: 028 (1.00)

c.

REFERENCE:

1. PNPS Proc. 2.2.19, "Residual Heat Removal," page 29.
2. IG: O-RO-02-09-01, "Low Pressure Coolant Injection and Residual Heat Removal System," ELO 14.
3. Derived from Facility Questions 10-B and RHR/SDC-07.

[4.0/3.9]

230000A406 ..(KA's)

ANSWER: 029 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-09-05, "Automatic Depressurization System," ELOs 5 and 15.

[4.2/4.3]

218000A206 ..(KA's)

ANSWER: 030 (1.00)

d.

REFERENCE:

1. IG: 0-RO-02-09-05, "Automatic Depressurization System," ELO 10.

[3.6/3.7]

239002K405 ..(KA's)

ANSWER: 031 (1.00)

c.

REFERENCE:

1. PNPS Technical Specification 3.10.B.
2. OJT Guide No. 5, SRM System Objective 2.a.

[3.2/3.9]

215004G011 ..(KA's)

ANSWER: 032 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-07-01, "Neutron Monitoring Systems," ELOs 18, 23, and 24.

[3.5/3.7]

215003A202 ..(KA's)

ANSWER: 033 (1.00)

a.

REFERENCE:

1. IG: O-RO-02-07-01, "Neutron Monitoring Systems," ELOs 40, 42, 44, and 47.

[3.4/3.7]

215001A207 ..(KA's)

ANSWER: 034 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 2.1.15, "Daily Surveillance Log (Tech Specs and Regulatory Agencies)," Daily Log Test #22.
2. UG: O-RO-04-02, "Administrative Tasks," Task 4.
3. Modified question from 12/2/92 NRC Exam (changed conditions, distractors, and correct answer).

[3.0/3.4]

215005A107 ..(KA's)

ANSWER: 035 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-07-01, "Neutron Monitoring Systems," ELOs 63, 64, and 65.

[3.2/3.4]

215005A207 ..(KA's)

ANSWER: 036 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-07-01, "Neutron Monitoring Systems," ELO 50.
2. Facility Question PSU-24 (from Requal Retake Exam).

[3.3/3.3]

215002A202 ..(KA's)

ANSWER: 037 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-06-04, "Rod Position Information System," ELO 11.d.

[3.1/3.3]

214000A201 ..(KA's)

ANSWER: 038 (1.00)

a. or d.

REFERENCE:

1. IG: O-RO-02-06-03, "Control Rod Drive System," ELO 29.

[3.5/3.5]

201006K403 ..(KA's)

ANSWER: 039 (1.00)

d.

REFERENCE:

1. IG: O-RO-02-04-02, "Condensate and Feedwater System," ELO 29.

[3.3/3.4]

259001A204 ..(KA's)

ANSWER: 040 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-04-02, "Condensate and Feedwater System," ELO 15.

[3.6/3.7]

256000K304 ..(KA's)

ANSWER: 041 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-06-02, "Condensate and Feedwater System," ELO 83.
2. Modified question from 11/26/90 NRC Exam (modified distractors and correct answer).

[3.1/3.1]

259002K604 ..(KA's)

ANSWER: 042 (1.00)

d.

REFERENCE:

1. IG: O-RO-02-05-01, "Main Turbine System," ELOs 70, 79, and 80.

[4.1/4.1]

241000K306 ..(KA's)

ANSWER: 043 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-01-03, "Main Generator," ELO 19.
2. IG: O-RO-02-05-01, "Main Turbine System," page IG-34-5/89.
3. Modified questions from 11/26/90 NRC Exam (combined 2 questions).

[2.7/2.8]

245000A304 ..(KA's)

ANSWER: 044 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-04-03, "Main Condenser Vacuum and Augmented Off Gas Systems," ELOs 2, 9i, 18d, and 18e.

[3.5/3.9]

271000A206 ..(KA's)

ANSWER: 045 (1.00)

d.

REFERENCE:

1. IG: O-RO-02-06-02, "Recirculation System," ELOs 9, 14, 15, 18, and 20.

[3.3/3.4]

202001K410 ..(KA's)

ANSWER: 046 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-06-02, "Recirculation System," ELO 33.

[3.5/3.5]

202002K604 ..(KA's)

ANSWER: 047 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-09-06, "Diesel Generator System," ELO 24.

[3.8/3.7]

264000K408 ..(KA's)

deleted

ANSWER: 048 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-08-03, "Standby Gas Treatment System," ELOs 9, 13, and 15e.

[2.9/3.2]

261000A203 ..(KA's)

ANSWER: 049 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-08-05, "Plant Ventilation Systems," ELO 4.
2. Modified question from 11/26/90 NRC Exam (changed correct answer and conditions on all distractors).

[3.6/3.9]

272000K403 ..(KA's)

ANSWER: 050 (1.00)

a.

REFERENCE:

1. IG: O-RO-02-08-05, "Plant Ventilation Systems," ELOs 9, 10, and 11.
2. Expanded Previous Question from 12/02/91 NRC Exam.

[3.3/3.4]

290001G010 ..(KA's)

ANSWER: 051 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-08-05, "Plant Ventilation Systems," ELOs 5 and 9.
2. Modified question from 11/26/90 NRC Exam (added concept and changed answers).

[3.1/3.3]

290003A204 ..(KA's)

ANSWER: 052 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-09-01, "Low Pressure Coolant Injection and Residual Heat Removal System," ELO 15.
2. IG: O-NL-03-11-01, "Fuel Pool Cooling," ELO 8.

[2.9/3.0]

233000K102 ..(KA's)

ANSWER: 053 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-08-06, "Fuel Handling Equipment," ELO 8.

[3.3/4.1]

234000K402 ..(KA's)

ANSWER: 054 (1.00)

a.

REFERENCE:

1. IG: O-RO-02-10-01, "Fire Protection System," ELOs 15, 16, 17, and 18.

[3.8/3.9]

286000G004 ..(KA's)

ANSWER: 055 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-07-01, "Neutron Monitoring Systems," ELOs 16 and 19.

[3.6/3.4]

215003A403 ..(KA's)

ANSWER: 056 (1.00)

d.

REFERENCE:

1. IG: O-RO-02-07-01, "Neutron Monitoring Systems," ELO 9.
2. PNPS Proc. No. 2.2.64, "Source Range Monitoring System," page 9.
3. PNPS Proc. No. 2.1.1, "Startup From Shutdown," page 22.
[3.6/3.9]

215004G001 ..(KA's)

ANSWER: 057 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-01-02, "AC Electrical Distribution," ELOs 34, 35, and 37.

[2.7/2.9]

262002K103 ..(KA's)

ANSWER: 058 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-01-02, "AC Electrical Distribution," ELOs 14b, 15, and 16.

[3.2/3.3]

262001A302 ..(KA's)

ANSWER: 059 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-01-03, "Main Generator," ELO 7.

[3.1/2.9]

245000A402 ..(KA's)

ANSWER: 060 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-06-06, "Standby Liquid Control System," ELO 8.

[3.5/3.5]

211000A301 ..(KA's)

ANSWER: 061 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-08-01, "Primary Containment Systems," page 34.
2. IG: O-RO-02-09-01, "Low Pressure Coolant Injection and Residual Heat Removal System," ELO 14.
3. Modified question from 12/2/91 NRC Exam (changed distractors and correct answer).

[3.6/3.5]

205000A402 ..(KA's)

ANSWER: 062 (1.00)

d.

REFERENCE:

1. IG: O-RO-02-09-02, "Core Spray System," ELOs 4 and 5.
2. Modified question from 11/26/90 NRC Exam (changed conditions and answers).

[3.7/3.7]

209001A104 ..(KA's)

ANSWER: 063 (1.00)

a.

REFERENCE:

1. PNPS Proc. No. 2.2.22, "Reactor Core Isolation Cooling System."
2. IG: O-RO-02-09-04, "Reactor Core Isolation Cooling System," ELO 6.
3. Facility Question TYPA-4 (from proposed Requal Retake Exam).

[3.4/3.3]

217000A403 ..(KA's)

ANSWER: 064 (1.00)

a.

REFERENCE:

1. IG: O-RO-02-09-03, "High Pressure Coolant Injection," ELO 17.

[3.7/3.8]

206000A104 ..(KA's)

ANSWER: 065 (1.00)

d.

REFERENCE:

1. PNPS Technical Specifications, sections 1.0.N and 3.7.C.
2. IG: O-RO-06-01-01, "Technical Specification Definitions," ELO 3.
3. IG: O-RO-04-09, "Technical Specification Overview," ELO 3.
   [3.9/4.2]

295035K101 ..(KA's)

ANSWER: 066 (1.00)

c.

REFERENCE:

1. PNPS Technical Specifications, section 2.1.
2. IG: O-RO-06-01-02, "Safety Limits and Limiting Safety System Settings," ELO 1.

[3.5/4.3]

295025G003 ..(KA's)

ANSWER: 067 (1.00)

a.

REFERENCE:

1. PNPS Technical Specifications, 3.6.E and 4.6.E Bases, page 147a.
2. OJT Guide No. 4, Recirculation System Objective 2b.

[3.9/4.2]

295001G011 ..(KA's)

ANSWER: 068 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 2.4.17, "Recirculation Pump(s) Trip," page 4.
2. OJT Guide 4, Recirculation System Objective 2b.

[3.5/3.8]

295001A201 ..(KA's)

ANSWER: 069 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 5.3.8, Rev. 16, "Loss of Instrument Air," page 2.
2. IG: O-RO-02-02-04, "Instrument and High Pressure Air," ELO 6.
3. Modified question from 12/2/91 NRC Exam (changed conditions).
[3.6/3.7]

295019A202 ..(KA's)

ANSWER: 070 (1.00)

d.

REFERENCE:

1. IG: O-RO-02-04-01, "Main Steam System," ELO 24.

[3.6/3.7]

295020K201 ..(KA's)

ANSWER: 071 (1.00)

b.

REFERENCE:

1. IG: O-RO-0208-01, "Primary Containment System," ELO 36.

[3.6/3.6]

295020A101 ..(KA's)

ANSWER: 072 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 2.4.42, "Loss of RBCCW," pages 4 and 6.
2. IG: O-RO-02-02-06, "Reactor Building Closed Cooling Water," ELOs 2, 3, 8, and 11.
3. OJT Guide No. 2, RBCCW System Objective 2b.

[3.3/3.4]

295018K201 ..(KA's)

ANSWER: 073 (1.00)

a.

REFERENCE:

1. IG: O-RO-02-02-02, "Salt Service Water System," ELO 5.

[4.4/4.4]

295003A103 ..(KA's)

ANSWER: 074 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 1.3.4, "Procedure-," page 17.
2. PNPS Proc. No. 2.4.36, "Decreasing Condenser Vacuum," page 2.
3. IG: O-RO-03-04-02, "EOP Development and Use," ELOs 13 and 14.
4. IG: O-RO-02-07-02, "Reactor Protection System and Anticipated Transient Without Scram System," ELO 14.
5. IG: O-RO-02-05-01, "Main Turbine System," ELO 21.
6. Modified questions from 11/26/90 and 12/2/91 NRC Exams (changed conditions).

[3.8/3.7]

295002G010 ..(KA's)

ANSWER: 075 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 2.4.150, "Loss of Feedwater Heating," page 3.
2. OJT Guide No. 8, Feedwater Heating System Objective 2a.

[4.0/3.9]

295014G010 ..(KA's)

ANSWER: 076 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 2.4.4, "Loss of CRD Pumps."
2. PNPS Proc. No. 2.4.11, "Control Rod Positioning Malfunctions."
3. PNPS Proc. No. 2.4.11.1, "CRD System Malfunctions."
4. OJT Guide No. 3, CRDM System Objective 2a and CRD Hydraulic System Objective 2b.

[3.7/3.5]

295022G010 ..(KA's)

ANSWER: 077 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 5.4.3, Rev. 10, Refueling Floor High Radiation," page 2.
2. IG: O-RO-06-04-01, "Fuel Handling Operations and Supervision," ELO 9.

[3.8/3.9]

295023G010 ..(KA's)

ANSWER: 078 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 5.3.6, "Loss of Vital AC (Y-2)."
2. PNPS Proc. No. 5.3.30, "Loss of 250V DC Bus D-10."
3. OJT Guide No. 1, 120/240 VAC System Objective 2b and 250 VDC System Objective 4.

[3.5/3.5]

295004G005 ..(KA's)

ANSWER: 079 (1.00)

b

REFERENCE:

1. PNPS Proc. No. 5.3.31, "Station Blackout," page 2.
2. PNPS Proc. No. 5.3.26, "RPV Injection During Emergencies," page 2.
3. OJT Guide No. 1, 480/208 VAC System Objective 2b.

[4.4/4.4]

295003A103 ..(KA's)

ANSWER: 080 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 2.4.143, Rev. 12, "Shutdown from Outside Control Room," page 13.
2. OJT Guide O-RO-04-04, "Emergency Tasks," Task 62.

[4.1/4.1]

295016G006 ..(KA's)

ANSWER: 081 (1.00)

d.

REFERENCE:

1. UG: O-RO-04-04, "Emergency Tasks," Tasks 1, 2, and 3. (Derived from facility question PRO-1 from proposed requal retake exam.)

[4.3/4.4]

295006A202 ..(KA's)

ANSWER: 082 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-07-01, "Neutron Monitoring Systems," ELOs 14 and 16.
2. IG: O-RO-03-04-04, "EOP-02, Failure to Scram," ELO 16.

[4.2/4.3]

295037A201 ..(KA's)

ANSWER: 083 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 5.3.23, Rev. 9, "Alternate Rod Insertion," page 5.
2. OJTPG: O-RO-04-04, "Emergency Tasks," Task 4.

[4.0/4.1]

295015K204 ..(KA's)

ANSWER: 084 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 5.3.23, Rev. 9, "Alternate Rod Insertion," pages 6 - 10.
2. OJTPG: O-RO-04-04, "Emergency Tasks," Task 4.
3. Modified question from 12/2/91 NRC Exam (modified distractors).

[3.8/3.9]

295015A101 ..(KA's)

ANSWER: 085 (1.00)

d.

REFERENCE:

1. IG: O-RO-03-04-05, Primary Containment Control,", page IG-5, ELO 14.
2. OJTPG: O-RO-04-04, "Emergency Tasks," Task 89c.
3. Modified question from 12/2/91 NRC Exam (replaced 1 distractor). [3.6/4.3]

295012G012 ..(KA's)

ANSWER: 086 (1.00)

a.

REFERENCE:

1. IG: O-RO-03-04-05, Primary Containment Control,", page IG-11.
2. OJTPG: O-RO-04-04, "Emergency Tasks," Task 89c.

[3.8/4.5]

295026G012 ..(KA's)

ANSWER: 087 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-08-01, "Primary Containment System," ELO 39A.

[3.6/3.9]

295024G007 ..(KA's)

ANSWER: 088 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 5.3.27, "Determining Primary Containment Water Level."
2. OJT Guide No. 10, "Primary Containment System Structure Objective 2a.
3. IG: O-RO-02-08-01, "Primary Containment System," ELOs 13 and 17.

[3.4/3.5]

295029A203 ..(KA's)

ANSWER: 089 (1.00)

c

REFERENCE:

1. PNPS Proc. No. 5.3.26, "RPV Injection During Emergencies."
2. Facility Question 5.3.26-1 (Proposed Requal Retake Exam).

[4.1/3.9]

295031G006 ..(KA's)

ANSWER: 090 (1.00)

a.

REFERENCE:

1. PNPS Proc. No. 2.4.46, "Turbine Bearing Malfunction," page 2.
2. OJT Guide No. 7, Turbine Supervisory Instrumentation Objective 2a.

[3.8/3.6]

295005G010 ..(KA's)

ANSWER: 091 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 5.3.3, "Loss of All Service Water," page 2.
2. IG: O-RO-02-02-02, "Salt Service Water System," ELO 9.
3. OJT Guide No. 2, Service Water System Objective 2b.

[3.4/3.3]

295018G010 ..(KA's)

ANSWER: 092 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 2.4.23, "Jet Pump Flow Failure," page 3.
2. IG: O-RO-02-06-01, "Non-Nuclear Instrumentation and Reactor Vessel Internals," ELOs 11 and 12.

[3.3/3.6]

295001G007 ..(KA's)

ANSWER: 093 (1.00)

b.

REFERENCE:

1. IG: O-RO-03-04-03, "EOP-01, RPV Control," ELO 1d.

[4.4/4.4]

295025A103 ..(KA's)

ANSWER: 094 (1.00)

b.

REFERENCE:

1. IG: O-RO-03-04-04, "EOP-02, Failure to Scram," ELO 17.

[4.1/4.2]

295037K301 ..(KA's)

ANSWER: 095 (1.00)

d.

REFERENCE:

1. IG: O-RO-03-04-04, "EOP-02, Failure to Scram," pages IG-33 and IG-41, ELO 1.

[3.9/4.6]

295037G012 ..(KA's)

ANSWER: 096 (1.00)

a.

REFERENCE:

1. IG: O-RO-02-09-06, "Diesel Generator System," ELOs 9 and 22.

[3.7/3.7]

295024A106 ..(KA's)

ANSWER: 097 (1.00)

c.

REFERENCE:

1. IG: O-RO-03-04-06, "EOP-04, Secondary Containment Control," ELO 1.
2. IG: O-RO-02-08-01, "Primary Containment System," ELO 16.

[3.9/4.2]

295035G011 ..(KA's)

ANSWER: 098 (1.00)

c.

REFERENCE:

1. IG: O-RO-03-04-06, "EOP-04, Secondary Containment Control," ELO 1.

[4.2/4.3]

295034G011 ..(KA's)

ANSWER: 099 (1.00)

c.

REFERENCE:

1. IG: O-RO-03-04-06, "EOP-04, Secondary Containment Control," ELO 4.

[3.6/4.4]

295036G012 ..(KA's)

ANSWER: 100 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-03-02, "Process Radiation Monitoring System," ELOs 5 and 10.

[3.6/3.8]

295038K202 ..(KA's)

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

A N S W E R   K E Y

MULTIPLE CHOICE

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

ANSWER KEY

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

A N S W E R   K E Y

| | |
|---|---|
| 092 | b |
| 093 | b |
| 094 | b |
| 095 | d |
| 096 | a |
| 097 | c |
| 098 | c |
| 099 | c |
| 100 | b |

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

RO Exam BWR Reactor

Organized by Question Number

| QUESTION | VALUE | REFERENCE | |
|---|---|---|---|
| ~~001~~ | ~~1.00~~ | ~~9000629~~ | deleted |
| 002 | 1.00 | 9000630 | |
| 003 | 1.00 | 9000631 | |
| 004 | 1.00 | 9000632 | |
| 005 | 1.00 | 9000633 | |
| 006 | 1.00 | 9000634 | |
| 007 | 1.00 | 9000635 | |
| 008 | 1.00 | 9000636 | |
| ~~009~~ | ~~1.00~~ | ~~9000637~~ | deleted |
| 010 | 1.00 | 9000638 | |
| 011 | 1.00 | 9000639 | |
| 012 | 1.00 | 9000640 | |
| 013 | 1.00 | 9000641 | |
| 014 | 1.00 | 9000651 | |
| 015 | 1.00 | 9000652 | |
| 016 | 1.00 | 9000653 | |
| 017 | 1.00 | 9000654 | |
| 018 | 1.00 | 9000655 | |
| 019 | 1.00 | 9000656 | |
| 020 | 1.00 | 9000657 | |
| 021 | 1.00 | 9000658 | |
| 022 | 1.00 | 9000659 | |
| 023 | 1.00 | 9000660 | |
| 024 | 1.00 | 9000661 | |
| 025 | 1.00 | 9000662 | |
| 026 | 1.00 | 9000663 | |
| 027 | 1.00 | 9000664 | |
| 028 | 1.00 | 9000665 | |
| 029 | 1.00 | 9000666 | |
| 030 | 1.00 | 9000667 | |
| 031 | 1.00 | 9000669 | |
| 032 | 1.00 | 9000670 | |
| 033 | 1.00 | 9000671 | |
| 034 | 1.00 | 9000672 | |
| 035 | 1.00 | 9000673 | |
| 036 | 1.00 | 9000674 | |
| 037 | 1.00 | 9000675 | |
| 038 | 1.00 | 9000676 | |
| 039 | 1.00 | 9000677 | |
| 040 | 1.00 | 9000678 | |
| 041 | 1.00 | 9000679 | |
| 042 | 1.00 | 9000680 | |
| 043 | 1.00 | 9000681 | |
| 044 | 1.00 | 9000682 | |
| 045 | 1.00 | 9000683 | |
| 046 | 1.00 | 9000684 | |
| 047 | 1.00 | 9000685 | |
| ~~048~~ | ~~1.00~~ | ~~9000686~~ | deleted |
| 049 | 1.00 | 9000687 | |

R O Exam B W R Reactor

Organized by Question Number

| QUESTION | VALUE | REFERENCE |
|---|---|---|
| 050 | 1.00 | 9000688 |
| 051 | 1.00 | 9000689 |
| 052 | 1.00 | 9000690 |
| 053 | 1.00 | 9000691 |
| 054 | 1.00 | 9000692 |
| 055 | 1.00 | 9000693 |
| 056 | 1.00 | 9000694 |
| 057 | 1.00 | 9000695 |
| 058 | 1.00 | 9000696 |
| 059 | 1.00 | 9000697 |
| 060 | 1.00 | 9000698 |
| 061 | 1.00 | 9000699 |
| 062 | 1.00 | 9000700 |
| 063 | 1.00 | 9000701 |
| 064 | 1.00 | 9000702 |
| 065 | 1.00 | 9000712 |
| 066 | 1.00 | 9000713 |
| 067 | 1.00 | 9000714 |
| 068 | 1.00 | 9000715 |
| 069 | 1.00 | 9000716 |
| 070 | 1.00 | 9000717 |
| 071 | 1.00 | 9000718 |
| 072 | 1.00 | 9000719 |
| 073 | 1.00 | 9000720 |
| 074 | 1.00 | 9000721 |
| 075 | 1.00 | 9000722 |
| 076 | 1.00 | 9000723 |
| 077 | 1.00 | 9000724 |
| 078 | 1.00 | 9000725 |
| 079 | 1.00 | 9000726 |
| 080 | 1.00 | 9000727 |
| 081 | 1.00 | 9000728 |
| 082 | 1.00 | 9000729 |
| 083 | 1.00 | 9000730 |
| 084 | 1.00 | 9000731 |
| 085 | 1.00 | 9000732 |
| 086 | 1.00 | 9000733 |
| 087 | 1.00 | 9000734 |
| 088 | 1.00 | 9000735 |
| 089 | 1.00 | 9000736 |
| 090 | 1.00 | 9000737 |
| 091 | 1.00 | 9000738 |
| 092 | 1.00 | 9000739 |
| 093 | 1.00 | 9000740 |
| 094 | 1.00 | 9000741 |
| 095 | 1.00 | 9000742 |
| 096 | 1.00 | 9000743 |
| 097 | 1.00 | 9000744 |
| 098 | 1.00 | 9000745 |

R O  E x a m  B W R  R e a c t o r

O r g a n i z e d  b y  Q u e s t i o n  N u m b e r

| QUESTION | VALUE | REFERENCE |
|---|---|---|
| 099 | 1.00 | 9000746 |
| 100 | 1.00 | 9000747 |
| | ------ | |
| | ~~100.00~~ 97.0 | |
| | ------ | |
| | ------ | |
| | ~~100.00~~ 97.0 | |

R O Exam BWR Reactor

Organized by KA Group

## PLANT WIDE GENERICS

| QUESTION | VALUE | KA | |
|---|---|---|---|
| 011 | 1.00 | 294001A102 | |
| 008 | 1.00 | 294001A103 | |
| 010 | 1.00 | 294001A106 | |
| ~~001~~ | ~~1.00~~ | ~~294001A115~~ | deleted |
| 012 | 1.00 | 294001A116 | |
| 013 | 1.00 | 294001A116 | |
| 006 | 1.00 | 294001K101 | |
| 007 | 1.00 | 294001K102 | |
| 003 | 1.00 | 294001K103 | |
| 002 | 1.00 | 294001K103 | |
| ~~009~~ | ~~1.00~~ | ~~294001K107~~ | deleted |
| 005 | 1.00 | 294001K109 | |
| 004 | 1.00 | 294001K110 | |
| PWG Total | ~~13.00~~ 11.00 | | |

## PLANT SYSTEMS

Group I

| QUESTION | VALUE | KA |
|---|---|---|
| 046 | 1.00 | 202002K604 |
| 026 | 1.00 | 203000A101 |
| 064 | 1.00 | 206000A104 |
| 024 | 1.00 | 206000G010 |
| 062 | 1.00 | 209001A104 |
| 025 | 1.00 | 209001A205 |
| 060 | 1.00 | 211000A301 |
| 019 | 1.00 | 212000G010 |
| 020 | 1.00 | 212000K412 |
| 032 | 1.00 | 215003A202 |
| 055 | 1.00 | 215003A403 |
| 056 | 1.00 | 215004G001 |
| 031 | 1.00 | 215004G011 |
| 034 | 1.00 | 215005A107 |
| 035 | 1.00 | 215005A207 |
| 018 | 1.00 | 216000A210 |
| 017 | 1.00 | 216000K122 |
| 023 | 1.00 | 217000A215 |
| 063 | 1.00 | 217000A403 |
| 029 | 1.00 | 218000A206 |
| 014 | 1.00 | 223001G011 |
| 022 | 1.00 | 223002A302 |
| 030 | 1.00 | 239002K405 |
| 042 | 1.00 | 241000K306 |

RO Exam BWR Reactor
Organized by KA Group

PLANT SYSTEMS

Group I

| QUESTION | VALUE | KA |
|---|---|---|
| 039 | 1.00 | 259001A204 |
| 041 | 1.00 | 259002K604 |
| ~~048~~ | ~~1.00~~ | ~~261000A203~~ deleted |
| 047 | 1.00 | 264000K408 |

PS-I Total ~~28.00~~ 27.00

Group II

| QUESTION | VALUE | KA |
|---|---|---|
| 038 | 1.00 | 201006K403 |
| 045 | 1.00 | 202001K410 |
| 016 | 1.00 | 204000G007 |
| 061 | 1.00 | 205000A402 |
| 037 | 1.00 | 214000A201 |
| 036 | 1.00 | 215002A202 |
| 027 | 1.00 | 226001G010 |
| 028 | 1.00 | 230000A406 |
| 021 | 1.00 | 239001K127 |
| 043 | 1.00 | 245000A304 |
| 059 | 1.00 | 245000A402 |
| 040 | 1.00 | 256000K304 |
| 058 | 1.00 | 262001A302 |
| 057 | 1.00 | 262002K103 |
| 044 | 1.00 | 271000A206 |
| 049 | 1.00 | 272000K403 |
| 054 | 1.00 | 286000G004 |
| 050 | 1.00 | 290001G010 |
| 051 | 1.00 | 290003A204 |

PS-II Total 19.00

Group III

| QUESTION | VALUE | KA |
|---|---|---|
| 033 | 1.00 | 215001A207 |
| 052 | 1.00 | 233000K102 |
| 053 | 1.00 | 234000K402 |
| 015 | 1.00 | 239003G008 |

PS-III Total 4.00

R O Exam BWR Reactor
Organized by KA Group

PLANT SYSTEMS

| QUESTION | VALUE | KA |
|---|---|---|
| PS Total | ~~51.00~~ 50.00 | |

EMERGENCY PLANT EVOLUTIONS

Group I

| QUESTION | VALUE | KA |
|---|---|---|
| 090 | 1.00 | 295005G010 |
| 081 | 1.00 | 295006A202 |
| 075 | 1.00 | 295014G010 |
| 084 | 1.00 | 295015A101 |
| 083 | 1.00 | 295015K204 |
| 096 | 1.00 | 295024A106 |
| 087 | 1.00 | 295024G007 |
| 093 | 1.00 | 295025A103 |
| 066 | 1.00 | 295025G003 |
| 089 | 1.00 | 295031G006 |
| 082 | 1.00 | 295037A201 |
| 095 | 1.00 | 295037G012 |
| 094 | 1.00 | 295037K301 |
| EPE-I Total | 13.00 | |

Group II

| QUESTION | VALUE | KA |
|---|---|---|
| 068 | 1.00 | 295001A201 |
| 092 | 1.00 | 295001G007 |
| 067 | 1.00 | 295001G011 |
| 074 | 1.00 | 295002G010 |
| 079 | 1.00 | 295003A103 |
| 073 | 1.00 | 295003A103 |
| 078 | 1.00 | 295004G005 |
| 085 | 1.00 | 295012G012 |
| 080 | 1.00 | 295016G006 |
| 091 | 1.00 | 295018G010 |
| 072 | 1.00 | 295018K201 |
| 069 | 1.00 | 295019A202 |
| 071 | 1.00 | 295020A101 |
| 070 | 1.00 | 295020K201 |
| 076 | 1.00 | 295022G010 |
| 086 | 1.00 | 295026G012 |
| 088 | 1.00 | 295029A203 |
| 098 | 1.00 | 295034G011 |

R O Exam BWR Reactor
Organized by KA Group

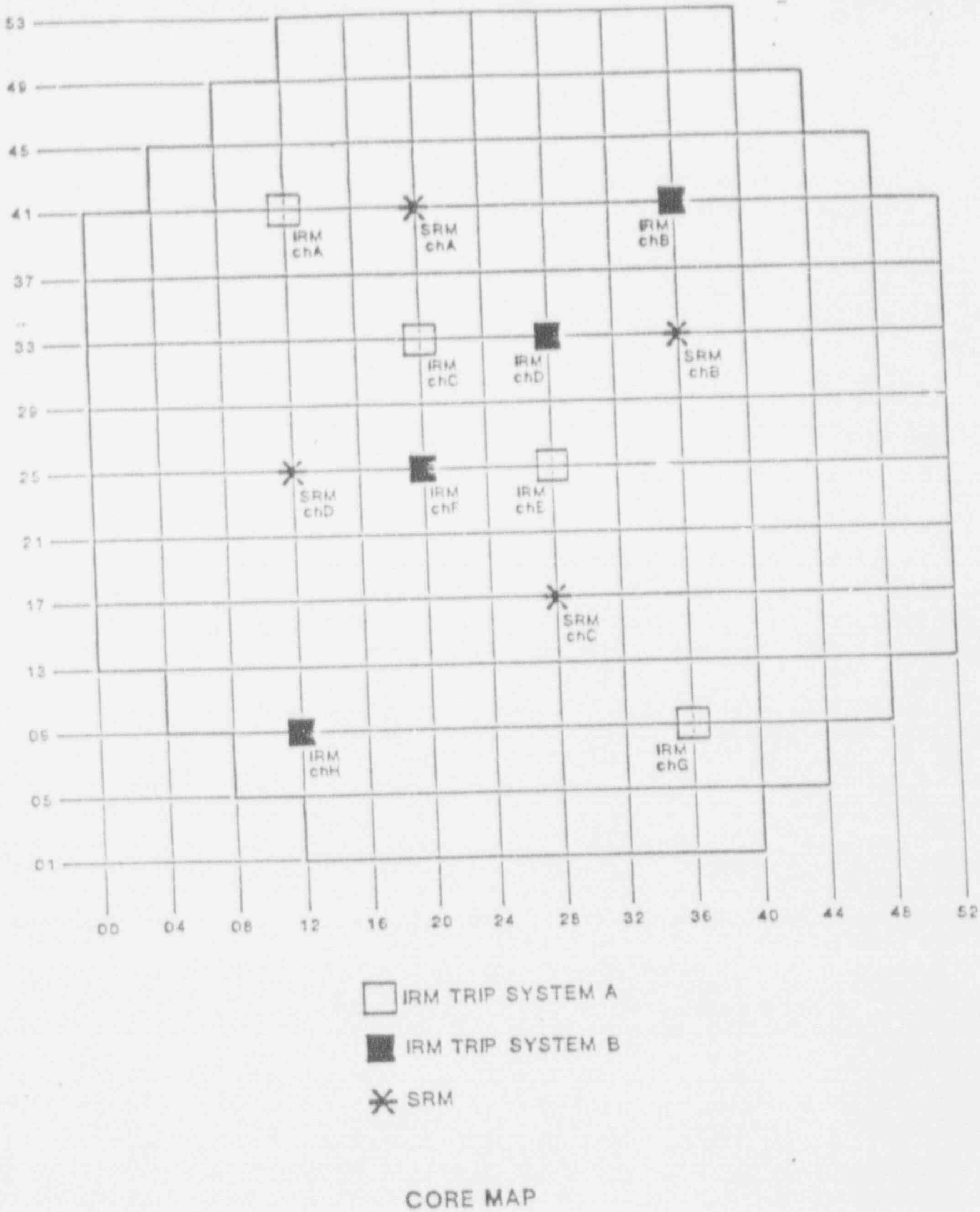
EMERGENCY PLANT EVOLUTIONS

Group II

| QUESTION | VALUE | KA |
|---|---|---|
| 100 | 1.00 | 295038K202 |
| EPE-II Total | 19.00 | |

Group III

| QUESTION | VALUE | KA |
|---|---|---|
| 077 | 1.00 | 295023G010 |
| 097 | 1.00 | 295035G011 |
| 065 | 1.00 | 295035K101 |
| 099 | 1.00 | 295036G012 |
| EPE-III Total | 4.00 | |
| EPE Total | 36.00 | |
| Test Total | ~~100.00~~ 97.00 | |




CORE MAP

FIGURE 4 REV. 1

# CONTROL ROD SEQUENCE SHEET
Figure 2



CONTROL ROD SEQ. A-1
V. NO. ____________
TE ____________

CHANGES

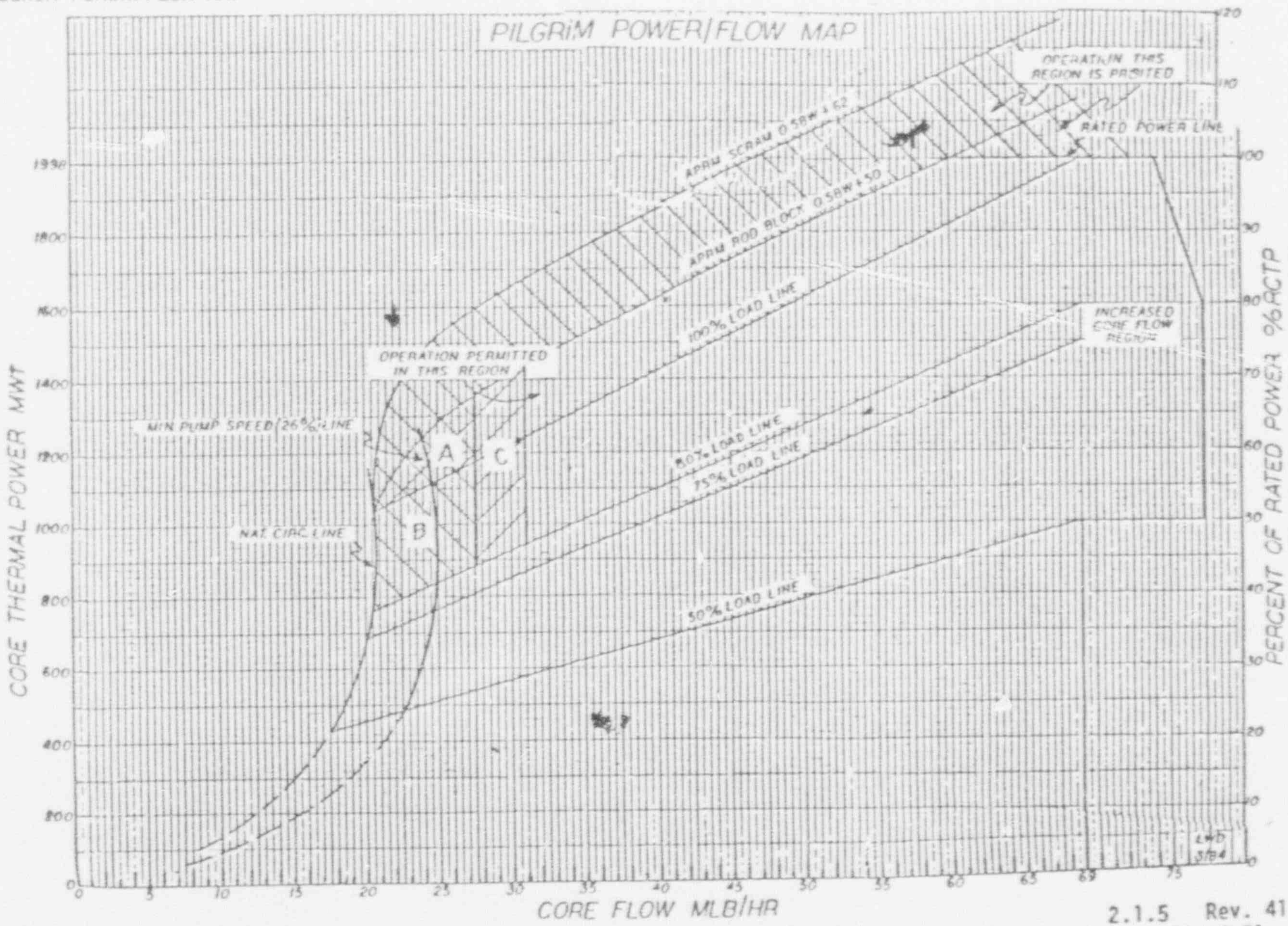
| RWM Step No. | Rod Grp No. | Rod Number | Mov Frm | Mov To | RWM Step No. | Rod Grp No. | Rod Numbr | Mov Frm | Mov To | Apr By | ROD MOVEMENT (✓) Withdraw | Verify | Insert | Verify | Withdraw | Verify | Insert | Verify | FLUX RESP (✓) | COUPLG CK (✓) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 26 | 7 | 22-47 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 46-31 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 30-07 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 06-23 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 06-31 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 30-47 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 46-23 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 22-07 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | |
| 27 | 8 | 14-39 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 38-39 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 38-15 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 14-15 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | |
| 28 | 9 | 22-31 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 30-31 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 30-23 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 22-23 | 08 | 12 | | | | | | | | | | | | | | | | |

Flux Response and/or Coupling Check performed if applicable
(Surveillance Requirement 4.3 B.1.a/4.3.B.1.b)

____________ NUCLEAR OPERATIONS SUPERVISOR ____________ DATE 17 PAGE NO.

SECTION G (Continued)

13. PILGRIM POWER/FLOW MAP



BOSTON EDISON

PILGRIM NUCLEAR POWER STATION

Procedure No. 2.4.17

RECIRCULATION PUMP(S) TRIP

INFORMATION ONLY

Use restricted to reference

REVIEWERS AND APPROVERS

| | |
|---|---|
| Eric W. Olson | 2/20/92 |
| Procedure Writer | Date |
| [illegible] | 3/2/92 |
| Technical Reviewer | Date |
| Eric W. Olson | 3/2/92 |
| Validator | Date |
| Thomas A. Graino LR PDS | 3/2/92 |
| Procedure Owner | Date |
| N/A | |
| OAD Manager | Date |
| JA Seery | 3/4/92 |
| ORC Chairman | Date |
| [illegible] | 3/5/92 |
| Plant Manager | Date |

SAFETY REVIEW REQUIRED

ORC REVIEW REQUIRED

Effective Date: 3-6-92

## 1.0 SYMPTOMS

### 1.1 ALARMS

| | ANNUNCIATOR | PANEL | WINDOW |
|---|---|---|---|
| [1] | RECIRC. MG SET GENERATOR DIFF. OVERCURRENT | 904C<br>904R | G4<br>A3 |
| [2] | RECIRC. MG SET DRIVE MOTOR TRIP | 904C<br>904R | B4<br>E2 |
| [3] | RECIRC. MG SET DRIVE MOTOR OVERLOAD | 904C<br>904R | C4<br>F2 |
| [4] | RECIRC. MG SET GENERATOR LOCKOUT | 904C<br>904R | E4<br>H2 |
| [5] | RECIRC. PUMP LOCKED ROTOR TRIP | 904C<br>904R | A4<br>D2 |
| [6] | RECIRC. MG SET LUBE OIL LOW PRESS | 904C<br>904R | B3<br>E1 |
| [7] | RECIRC. MG SET FLUID DRIVE HI OIL TEMP. | 904C<br>904R | D3<br>G1 |

### 1.2 PLANT INDICATIONS

[1] Reduction in recirculation flow.

[2] Sudden decrease in reactor power.

## 2.0 AUTOMATIC ACTIONS

None

## 3.0 IMMEDIATE OPERATOR ACTIONS

[1] MONITOR alarms and instrumentation AND DETERMINE the type of system malfunction that has occurred.

[2] IF both recirculation pumps trip, THEN MANUALLY SCRAM the reactor AND CONCURRENTLY PERFORM PNPS 2.1.6, "Reactor Scram", with this Procedure.

## 4.0 SUBSEQUENT OPERATOR ACTIONS

[1] TURN to the page of this Procedure for the malfunction that has occurred AND PERFORM the indicated steps:

| | Page | Section |
|---|---|---|
| (a) Trip of one recirculation pump | 3 | 4.1 |
| (b) Trip of both recirculation pumps | 6 | 4.2 |

### 4.1 TRIP OF ONE RECIRCULATION PUMP

---

NOTE

The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the Plant shall be placed in a Hot Shutdown condition within 24 hours unless the loop is sooner returned to service.

---

*******************************************************************************

CAUTION

If power level is less than 30%, stratification may occur; refer to PNPS 2.4.24, "Reactor Vessel Cold Water Stratification".

*******************************************************************************

[1] CLOSE affected MO-202-5A or B, PUMP DISCH VLV.

(a) WHEN 5 minutes have elapsed, THEN REOPEN the discharge valve.

[2] CHECK speed on the in-service pump to ensure it has not increased.

[3] DETERMINE Total Core Flow (TCF) [NRC Inspection Report 91-25]

(a) DETERMINE direction of flow through idle jet pumps.

(1) ADD in-service and idle jet pump loop flow rate:

| ________ | + | ________ | = | ________ |
|---|---|---|---|---|
| In-Service FI-263-107A(B) | | Idle FI-263-107A(B) | | Summed Value |

(2) MULTIPLY idle jet pump loop flow by 0.95 <u>AND</u> SUBTRACT this multiple from in-service jet pump loop flow:

| ________ | - [0.95 X | ________ ] | = | ________ |
|---|---|---|---|---|
| In-Service FI-263-107A(B) | | Idle FI-263-107A(B) | | Subtracted Value |

(3) USE current reactor power <u>AND</u> PLOT both of the calculated flow values on the Power-To-Flow Map.

---

<u>NOTE</u>

The change in reactor power due to the recirc pump trip will result in a Xenon transient. This transient will cause the previous load line to lower. The amount the load line shifts is dependent on the time after the recirc pump trip and previous equilibrium reactor power conditions.

---

(4) <u>IF</u> the Subtracted Value falls to the left of the expected load line (i.e., on or to the left of the minimum pump speed line) <u>AND</u> the Summed Value falls approximately on the expected Load Line, <u>THEN</u> Forward Flow exists through the idle jet pump loop.

(5) <u>IF</u> the Subtracted Value falls approximately on the expected load line <u>AND</u> the Summed Value falls below and to the right, <u>THEN</u> Reverse Flow exists through the idle jet pump loop.

(b) CALCULATE Total Core Flow (TCF)

(1) <u>IF</u> Forward Flow through the idle jet pump loop exists, <u>THEN</u> Total Core Flow equals the Summed Value.

(2) <u>IF</u> Reverse Flow through the idle jet pump loop exists, <u>THEN</u> Total Core Flow equals the Subtracted Value.

[4] IF the reactor is operating AT OR ABOVE the 80% load line AND total core flow decreases below 31.5 Mlb/hr, THEN PERFORM the following steps: [IEB 88-07, SUPP 1: BWROG 8879]

(a) MONITOR the APRMs and LPRMs for neutron flux instability oscillations.

(b) INCREASE speed of the operating recirculation pump until any flux instability ceases and core flow is greater than 31.5 Mlb/hr.

(c) INSERT control rods to decrease reactor power below the 80% load line.

(d) IF APRM oscillations of greater than 10% peak-to-peak OR periodic LPRM upscale or downscale alarms are observed, THEN SCRAM the reactor AND CONCURRENTLY PERFORM PNPS 2.1.6 with this Procedure.

[5] AFTER the recirculation pump is secured, ADJUST total core flow to greater than 27.6 Mlb/hr. [GE SIL 517]

(a) VERIFY that the recirculation loop flow of the active loop on FI-107A or B is less than 36.9 Mlb/hr. [GE SIL 517]

[6] SEND an operator to the 4kV breaker and to the MG Set Room to record all relay targets to determine cause of trip.

[7] ENSURE that power is available as follows:

(a) Instrument power Panel Y1

(b) Vital service Panel Y2

(c) 4160V load centers A3 or A4

(d) Power Centers B17, B18, B20 and D9

(e) Power Panels D4, D5 and D6

[8] IF the cause of the trip can be determined and corrected AND the reactor is operating outside of the area of the power flow map bounded by the 80% load line and 31.5 Mlb/hr, THEN the pump may be restarted in accordance with PNPS 2.2.84, "Reactor Recirculation System". [IEB 88-70, Supp 1; BWROG 8879]

[9] REFER to EPIP-100, "Emergency Classification", to determine whether an Emergency Action Level (EAL) has been exceeded.

4.2 TRIP OF BOTH RECIRCULATION PUMPS

[1] CLOSE both MO-202-5A and B, PUMP DISCH VLVs.

(a) WHEN 5 minutes have elapsed, THEN REOPEN the discharge valves.

[2] SEND an operator to the 4kV breaker and to the MG Set Room to record all relay targets to determine cause of trip.

[3] ENSURE that power is available as follows:

(a) Instrument power Panel Y1

(b) Vital service Panel Y2

(c) 4160V load centers A3 or A4

(d) Power Centers B17, B18, B20 and D9

(e) Power Panels D4, D5 and D6

[4] IF the cause of the pump trips can be identified and corrected, THEN RESTART the pumps in accordance with PNPS 2.2.84, "Reactor Recirculation System".

[5] REFER to PNPS EPIP-100, "Emergency Classification", to determine whether an Emergency Action Level (EAL) has been exceeded.

BOSTON EDISON

PILGRIM NUCLEAR POWER STATION

Procedure No. 5.3.27

DETERMINING PRIMARY CONTAINMENT WATER LEVEL

INFORMATION ONLY

Use restricted to reference

Approved Ed Kraft, Jr. for RAA 6/26/90
Plant Manager Date

## 1.0 PURPOSE

This procedure provides instructions for determining Primary Containment Water Level when flooding of the Primary Containment is directed by the Emergency Operating Procedures (EOPs).

- Primary Containment (PC) Water Level values are referenced to plant elevation.

## 2.0 ACTIONS

### 2.1 FOR CONTAINMENT LEVEL LESS THAN 47 FEET

[1] CALCULATE the existing differential pressure (sensed) between the drywell air space and the bottom of the torus, as follows (Figure 1):

| | | |
|---|---|---|
| | TORUS BOTTOM PRESS<br>(PI-1001-69)<br>(Panel C903) | ____________ psig |
| minus: | DRYWELL WIDE<br>RANGE PRESSURE<br>(PI-1001-600A/B)<br>(Panel C170/171) | ____________ psig |
| equals: | CONTAINMENT TO<br>TORUS BOTTOM dP | ____________ psig |

[2] For the calculated CONTAINMENT TO TORUS BOTTOM dP, the corresponding Primary Containment Water Level is obtained from the Primary Containment Water Level curve (Figure 2).

### 2.2 FOR CONTAINMENT LEVEL GREATER THAN OR EQUAL TO 47 FEET

[1] READ containment water level on DRYWELL LEVEL Gauge LI-5008 on Panel C903.

| CONTAINMENT TO TORUS BOTTOM dP CALCULATION | | |
|---|---|---|
| | TORUS BOTTOM PRESS (PI-1001-69) (Panel C903) | ______ psig |
| − | DRYWELL WIDE RANGE PRESSURE (PI-1001-600A/B) (Panel C170/171) | ______ psig |
| = | CONTAINMENT TO TORUS BOTTOM dP | ______ psig |

FIGURE 1

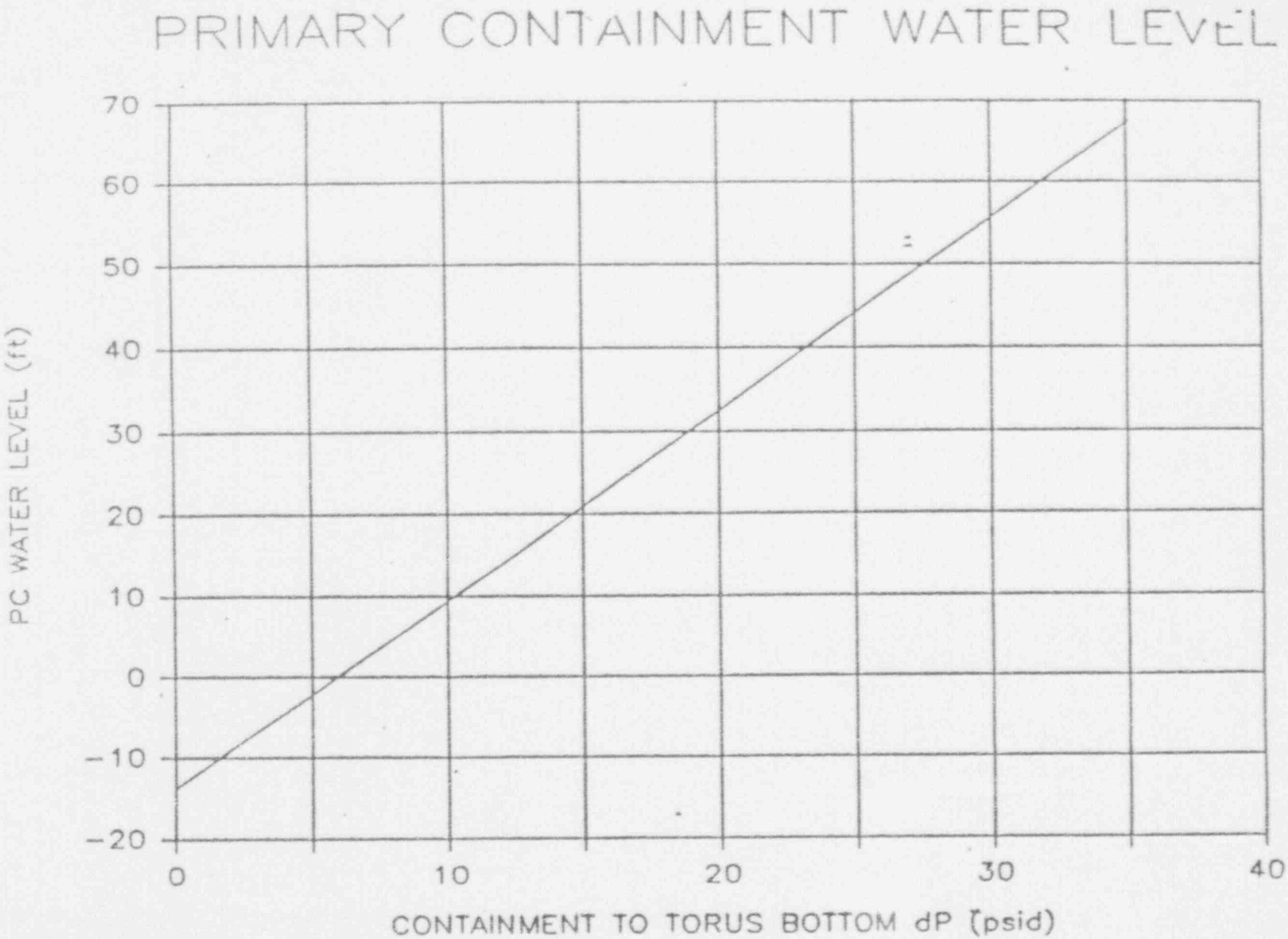

PRIMARY CONTAINMENT WATER LEVEL
PC WATER LEVEL (ft)
70
60
50
40
30
20
10
0
-10
-20
0
10
20
30
40
CONTAINMENT TO TORUS BOTTOM dP (psid)

FIGURE 2

Attachment 3

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

CANDIDATE'S NAME:

FACILITY: Pilgrim 1

REACTOR TYPE: BWR-GE3

DATE ADMINISTERED: 92/11/16

INSTRUCTIONS TO CANDIDATE:

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

| TEST VALUE | CANDIDATE'S SCORE | % | |
|---|---|---|---|
| 98.0 ~~100.00~~ | | % | TOTALS |
| | FINAL GRADE | | |

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

Candidate's Signature

deleted

QUESTION: 001 (1.00)

~~WHICH ONE of the following On Demand (OD) Programs would be used to obtain the effective readings from an LPRM string used in the most recent P1 power distribution calculation?~~

~~a. OD-6, Thermal Data in a Specified Bundle~~

~~b. OD-8, Present LPRM Readings~~

~~c. OD-9, Axial Interpolation in a Specified LPRM String~~

~~d. OD-17, Periodic Core Performance Logs~~

QUESTION: 002 (1.00)

Conditions in a recently surveyed area are:

25 mR/hr general area radiation
100 dpm/100 cm2 alpha loose surface
500 dpm/100 cm2 beta-gamma loose surface
0.20 MPC airborne beta-gamma radioactivity

WHICH ONE of the following describes the complete posting requirements for the area?

a. "CAUTION RADIATION AREA"

b. "CAUTION RADIATION AREA" and "CAUTION CONTAMINATED AREA"

c. "CAUTION RADIATION AREA" and "CAUTION AIRBORNE RADIOACTIVITY AREA"

d. "CAUTION RADIATION AREA," "CAUTION CONTAMINATED AREA," and "CAUTION AIRBORNE RADIOACTIVITY AREA"

QUESTION: 003 (1.00)

A 23 year old radiation worker with a current Form NRC-4 needs to perform work in an area with general radiation levels of 75 mR/hr. The worker's exposure history is:

| | |
|---|---|
| Lifetime: | 24.5 Rem |
| Current year: | 1400 mR |
| Current quarter: | 225 mR |

WHICH ONE of the following is the maximum time that the worker can stay in the area without exceeding any PNPS Exposure Control Levels? (Assume no special authorization.)

a. 20 minutes

b. 1 hour and 20 minutes

c. 6 hours and 40 minutes

d. 7 hours

QUESTION: 004 (1.00)

WHICH ONE of the following describes proper procedures for handling sodium pentaborate?

a. Respiratory protection is required to prevent inhaling dust containing boron.

b. Protective clothing is not required because sodium pentaborate cannot be absorbed through the skin.

c. Report to the Chemistry Department immediately after leaving the worksite to dispose of protective clothing and foot coverings.

d. Direct any liquid spillage to a floor drain, wipe and dry mop the floor, and discard mops and wipes after use.

QUESTION: 005 (1.00)

WHICH ONE of the following situations is acceptable in accordance with PNPS Proc. No. 1.4.36, "High Pressure/Compressed Gas Cylinder Control?"

a. Argon cylinders on a cart are staged for a non-active Maintenance Request that has been rescheduled for later in the week. The cart is secured to a fixed support.

b. Reserve nitrogen cylinders are secured at approximately 3/4 height to a permanently installed cylinder holding station.

c. Acetylene cylinders on a cart are staged in the Cable Spreading Room for an active Maintenance Request. Work is expected to start next shift. The cart is secured to a fixed support.

d. Empty oxygen cylinders are secured at approximately 3/4 height to a fixed support during shift turnover. The cylinders will be removed next shift.

QUESTION: 006 (1.00)

WHICH ONE of the following situations would require independent verification in accordance with PNPS Proc. No. 1.3.34, "Conduct of Operations?" (All components are safety related.)

a. Fuses are pulled for maintenance in a cabinet in the Control Room.

b. An existing tagout is used to verify the position of an inaccessible component.

c. Verification of a component's position is expected to take 15 minutes. Radiation levels in the area are 120 mr/hr.

d. A large manually operated valve that requires two people to operate must be shut for maintenance.

QUESTION: 007 (1.00)

Breaker 103 (Bus A1 feeder from Unit Auxiliary Transformer) is open and has a white tag with a green border hanging on the control switch. WHICH ONE of the following describes the meaning of the tag?

a. A grounding device is installed on the breaker. The breaker may only be operated under the authority of the person for whom the tag was placed.

b. The breaker is not in its normal operable status. The breaker may be operated by qualified operators when the precautions listed on the tag are followed.

c. The breaker requires testing prior to restoring it to normal service. The breaker may be operated only under the authority of the person for whom the tag was placed.

d. The breaker is open to protect personnel from injury or equipment from damage. The breaker may not be operated until the tag is cleared.

QUESTION: 008 (1.00)

WHICH ONE of the following accurately describes the NRC Overtime Guidelines?

a. Operators may not work more than: 12 hours in 24 hours; 24 hours in 48 hours; or 84 hours in 7 days.

b. Operators may not work more than: 16 hours in 24 hours; 28 hours in 48 hours; or 84 hours in 7 days.

c. Operators may not work more than: 16 hours in 24 hours; 24 hours in 48 hours; or 72 hours in 7 days.

d. Operators may not work more than: 12 hours in 24 hours; 28 hours in 48 hours; or 72 hours in 7 days.

QUESTION: 009 (1.00)

An initial entry is being made into Primary Containment in accordance with PNPS Proc. No. 1.4.12, "Primary Containment Entry." A nitrogen pocket causes the oxygen analyzer drywell entrance light to illuminate. WHICH ONE of the following describes the entry team's awareness of this situation?

a. They would not be aware that the light had illuminated.

b. The alarm at the drywell entrance is relayed to a siren in the drywell which would alert the team.

c. The backup team stationed outside the drywell entrance would alert the entry team of the condition by the plant paging system.

d. H.P. stationed outside the drywell entrance would alert the entry team of the condition by walkie talkie.

QUESTION: 010 (1.00)

The plant is operating at 100% with the Technical Specification Minimum Operating Shift Crew Composition on duty. The STA is filling a licensed SRO position. A fire has been reported in the turbine building. WHICH ONE of the following lists the maximum number and composition of operations personnel that can respond to the fire? (All operations personnel on duty are qualified to be Fire Brigade members.)

a. 2 unlicensed operators and the STA

b. 1 unlicensed operator, 1 licensed SRO, and the STA

c. 1 unlicensed operator and 1 licensed RO

d. 2 unlicensed operators

QUESTION: 011 (1.00)

The Control Room has been evacuated due to a fire in the Cable Spreading Room. There is a shortage of licensed operators. WHICH ONE of the following systems can be operated by a non-licensed operator from the Alternate Shutdown Panel?

a. HPCI

b. RBCCW

c. ADS

d. RHR

QUESTION: 012 (1.00)

WHICH ONE of the following situations would require an entry into the Lifted Lead/Jumper (LLJ) Log?

a. A jumper is installed and leads are lifted in an emergency condition in accordance with PNPS Proc. No. 5.3.21, "Bypassing Selected Interlocks." The emergency condition is expected to last longer than the current NWE working shift.

b. Motor leads are lifted for motor replacement. The leads fall within the Maintenance Request (MR) tagout boundary and are identified with colored tape. The MR provides for relanding the leads prior to clearing the isolation boundary. The motor replacement is expected to last longer than the current NWE working shift.

c. A jumper is installed for a surveillance test. Installation and removal of the jumper are signoff steps in the procedure. The jumper will be removed prior to the end of the current NWE working shift.

d. A gagging device is installed for maintenance in accordance with an approved station procedure. The gagging device is within the Maintenance Request (MR) tagout boundary and the MR provides for removing the gagging device prior to clearing the isolation boundary. The gagging device will not be removed prior to the end of the current NWE working shift.

QUESTION: 013 (1.00)

The plant is at 100% power with IRM G and APRM B inoperable. An I&C supervisor reports, after reviewing surveillance data, that the power supplies for APRM D and IRM A need replacement and the instruments are technically inoperable. WHICH ONE of the following describes the entries that should be made in the Limiting Condition for Operation (LCO) Log?

a. A tracking LCO is entered for APRM D and a tracking LCO is entered for IRM A.

b. A tracking LCO is entered for APRM D and an active LCO is entered for IRM A.

c. An active LCO is entered for APRM D and a tracking LCO is entered for IRM A.

d. An active LCO is entered for APRM D and an active LCO is entered for IRM A.

QUESTION: 014 (1.00)

WHICH ONE of the following conventions for annotating flowcharts is preferred when executing the EOPs?

a. Place an 'X' through any entry conditions that have been met.

b. When a step is completed, place an 'X' through the step and draw a line to the next step.

c. While performing steps in another EOP or another flow chart branch, draw arrows to indicate the stopping point in the current branch.

d. Periodically update values of the parameters controlled by the EOP next to the step that is being performed when the update is provided.

QUESTION: 015 (1.00)

WHICH ONE of the following describes the appropriate action to be taken when an Alarm Statement is reached when implementing the EOPs?

a. Review the overrides of all flowpaths being performed. Take the action specified in any applicable overrides.

b. Review the overrides of the flowpath containing the Alarm Statement that was reached. Take the action specified in any applicable overrides.

c. Exit the flowpath being executed when the Alarm Statement was reached and enter the contingency procedure applicable to the Alarm Statement.

d. Exit all the flowpaths containing the Alarm Statement that was reached and enter the contingency procedure applicable to the Alarm Statement.

QUESTION: 016 (1.00)

A double ended shear of Recirculation System piping has occurred. Secondary containment has isolated and SBGTS is operating. WHICH ONE of the following describes the expected release?

a. SBGTS removes most of the noble gases and halogens. Discharge of SBGTS is an elevated release.

b. SBGTS removes most of the halogens, but almost none of the noble gases. Discharge of SBGTS is an elevated release.

c. SBGTS removes most of the noble gases, but almost none of the halogens. Discharge of SBGTS is a ground level release.

d. SBGTS removes most of the halogens and noble gases. Discharge of SBGTS is a ground level release.

QUESTION: 017 (1.00)

A LOCA has occurred and a General Emergency has been declared. Emergency venting is in progress. Dose projection is not possible at this time. Plant conditions are:

| | |
|---|---|
| Drywell radiation level: | 3.0EE4 R/hr and increasing |
| Torus radiation level: | 2.8EE4 R/hr and increasing |
| Drywell hydrogen conc: | 7.2% and decreasing |
| Drywell oxygen conc: | 5.4% and decreasing |
| Torus hydrogen conc: | 5.8% and decreasing |
| Torus oxygen conc: | 4.5% and decreasing |
| Drywell pressure: | 44 psig and decreasing |
| Torus bottom pressure: | 40 psig and decreasing |

WHICH ONE of the following describes the appropriate Protective Action Recommendation (PAR) for these conditions? (EP-IP-400, Attachment 1 is attached.)

a. Consider evacuation of 2 mile ring and shelter 5 miles downwind.

b. Consider evacuation of 2 mile ring and shelter 5 miles downwind. Shelter other affected subareas.

c. Consider evacuation of 5 mile ring and 10 miles downwind.

d. Evacuate 5 mile ring and 10 miles downwind.

QUESTION: 018 (1.00)

WHICH ONE of the following conditions could cause indicated level on the Fuel Zone instruments to be higher than actual reactor vessel water level?

a. Drywell temperature at 180°F.

b. A break in the variable leg.

c. No forced recirculation flow through the jet pumps.

d. A rapid reactor depressurization.

QUESTION: 019 (1.00)

All of the Scram Discharge Volume (SDV) vent and drain valves were manually operated during maintenance. The valves were returned to their normal position, but the manual operators for the SDV valves were not returned to the NEUTRAL position. WHICH ONE of the following describes the concern related to this operation?

a. The valves could fail to automatically reposition on a reactor scram, preventing drain down of the SDV.

b. The valves could fail to automatically reposition on a reactor scram, causing a direct discharge path from the RPV to the Reactor Building sump.

c. The valves could fail open after repositioning on a reactor scram, causing a breach of primary containment.

d. The valves could fail closed after repositioning on a reactor scram, preventing reset of the scram due to high SDV level.

QUESTION: 020 (1.00)

A scram has occurred and the mode switch has been placed in SHUTDOWN. Scram Discharge Volume (SDV) level is high, but the scram has been reset by use of the SDV High Level Scram Bypass. RPS bus A is then transferred to its alternate power supply. WHICH ONE of the following describes the expected response?

a. It is a dead bus transfer and a half scram will occur.

b. It is a dead bus transfer and a full scram will occur.

c. It is a dead bus transfer, but is rapid enough so that no scram will occur.

d. It is a live bus transfer, so no scram will occur.

QUESTION: 021 (1.00)

A plant startup was underway when annunciator "MSIV NOT FULLY OPEN SCRAM AT RX PRESSURE >600 PSI" (905R, D4) illuminated. A full scram occurred. WHICH ONE of the following describes the expected status of the plant?

a. One main steam line is isolated; reactor pressure is greater than 600 psig.

b. Two MSIVs have closed; reactor pressure is less than 600 psig.

c. Three main steam lines are isolated; reactor pressure is greater than 600 psig.

d. Four MSIVs have closed; reactor pressure is less than 600 psig.

QUESTION: 022 (1.00)

The plant was operating at 100% power when a loss of all feed caused reactor water level to decrease. The reactor scrammed and the mode switch was placed in SHUTDOWN. Reactor water level is -10 inches and reactor pressure is 900 psig. WHICH ONE of the following sets of valves should have received isolation signals?

a. Recirculation System process sample valves, Post Accident Sampling System isolation valves, and RHR reject to Radwaste valves

b. RWCU isolation valves, RBCCW to Drywell Coolers isolation valves, and Primary Containment Atmosphere Control makeup purge valves

c. MSIVs, Drywell Equipment Drain Sump isolation valves, and Radiation Leak Detection System isolation valves

d. Main Steam Line drains, Drywell Floor Drain Sump isolation valves, and Hydrogen/Oxygen Analyzer System isolation valves

QUESTION: 023 (1.00)

The RCIC system is running following a valid initiation signal received 10 minutes ago. The following conditions develop:

| | |
|---|---|
| RCIC Steam Line Flow: | 230% |
| RCIC Area Temperature: | 205°F |
| RCIC Pump Suction Pressure: | 10" Hg vacuum |
| RCIC Turbine Exhaust Pressure: | 38 psig |
| RPV Pressure: | 150 psig |
| RPV Water Level: | +40 inches |

WHICH ONE of the following describes the expected response of the RCIC system?

a. Only the RCIC inboard and outboard isolation valves close.

b. The RCIC inboard and outboard isolation valves close and RCIC trip throttle valve closes.

c. Only the RCIC trip throttle valve closes.

d. Only the RCIC steam to turbine supply valve closes.

QUESTION: 024 (1.00)

WHICH ONE of the following describes the restrictions on HPCI operation while implementing the EOPs?

a. HPCI may NEVER be operated below 1000 RPM because this is the minimum speed required to maintain adequate cooling and lubrication.

b. HPCI may NEVER be operated below 2000 RPM because this is the minimum speed required to generate sufficient control oil pressure for control valve operation.

c. HPCI may be operated below 1000 RPM only at NOS/NWE direction because low turbine exhaust pressure could create a cyclic steam hammer which could damage the exhaust check valve.

d. HPCI may be operated below 2000 RPM, but greater than 1000 RPM only as directed by the EOPs because operation at these speeds could cause excessive turbine vibration or oscillating flow rates.

QUESTION: 025 (1.00)

With the plant operating at 100% power, the Core Spray Loop A line break differential pressure indication on Rack 2207 reads +4.5 psid. WHICH ONE of the following conditions is indicated?

a. No core spray line break has occurred. This indication is normal for full power operations.

b. A core spray line break has occurred inside the reactor vessel shroud.

c. A core spray line break has occurred inside the reactor vessel, but outside the shroud.

d. A core spray line break has occurred inside the drywell or inside the reactor vessel, but outside the shroud.

QUESTION: 026 (1.00)

The reactor is shutdown and RHR loop B is in the shutdown cooling (SDC) mode. A reactor coolant system leak causes vessel level to decrease to 0 inches and drywell pressure to increase to 2.8 psig. Loop A jet pump riser pressure is less than loop B jet pump riser pressure. No operator action has been taken. WHICH ONE of the following describes the status of the Low Pressure Coolant Injection (LPCI) valves?

a. Outboard injection valve 28A is open
   Inboard injection valve 29A is open
   Outboard injection valve 28B is closed

b. Outboard injection valve 28A is closed
   Inboard injection valve 29A is open
   Outboard injection valve 28B is closed

c. Outboard injection valve 28B is open
   Inboard injection valve 29B is closed
   Outboard injection valve 28A is closed

d. Outboard injection valve 28B is open
   Inboard injection valve 29B is open
   Outboard injection valve 28A is closed

QUESTION: 027 (1.00)

WHICH ONE of the following RHR loop lineups assures that the design limits are met for adequate drywell sprays and RHR equipment operation?

a. RHR pump B is running, the RHR heat exchanger bypass valve is closed, and RHR heat exchanger flow is 4800 gpm.

b. RHR pump C is running, the RHR heat exchanger bypass valve is open, and RHR heat exchanger flow is 3750 gpm.

c. RHR pumps B and D are running, the RHR heat exchanger bypass valve is closed, and RHR heat exchanger flow is 9100 gpm.

d. RHR pumps A and C are running, the RHR heat exchanger bypass valve is open, and RHR heat exchanger flow is 5050 gpm.

QUESTION: 028 (1.00)

LPCI initiated on a valid initiation signal. RPV water level is -130 inches and increasing slowly. Drywell pressure is 2.0 psig and increasing. WHICH ONE of the following describes the actions necessary to open MO-34 and MO-37 to spray the torus under these conditions?

a. MO-34 and MO-37 cannot be opened until drywell pressure increases above 2.5 psig.

b. Place the keylock RPV level override switch in MANUAL OVERRIDE.

c. Place the pistol grip LPCI override switch in MANUAL OVERRIDE.

d. Place the keylock RPV level override switch in MANUAL OVERRIDE and place the pistol grip LPCI override switch in MANUAL OVERRIDE.

QUESTION: 029 (1.00)

An ADS blowdown was initiated on high drywell pressure and low-low reactor water level. The only low pressure CSCS pump running is Core Spray pump B. WHICH ONE of the following conditions will close the ADS valves?

a. The ADS Inhibit switches are placed in INHIBIT, drywell pressure decreases to 2.0 psig, and the high drywell pressure reset pushbuttons are depressed.

b. The ADS Inhibit switches are placed in INHIBIT, reactor water level increases to -35 inches, and Core Spray pump B trips.

c. Drywell pressure decreases to 1.0 psig, the high drywell pressure reset pushbuttons are depressed, and the timer reset pushbuttons are depressed.

d. Core Spray pump B trips, drywell pressure decreases to 1.8 psig, and reactor water level increases to -40 inches.

QUESTION: 030 (1.00)

The control switch for a safety relief valve is in the REMOTE position on alternate shutdown panel 156. WHICH ONE of the following describes operation of the safety relief valve in this condition?

a. The valve will only operate in the safety mode.

b. The valve will only operate automatically in the ADS mode.

c. The valve will only operate automatically in the ADS mode or the safety mode.

d. The valve will operate automatically in the ADS mode or the safety mode and can be manually operated from the control room.

QUESTION: 031 (1.00)

Refueling operations are in progress. A fuel assembly is being removed from location 19-22 in the core. WHICH ONE of the following situations meets the Technical Specification core monitoring requirements? (A core map is attached.)

a. SRM 'A' is bypassed.
SRM 'B' is fully inserted and reading 5 cps.
SRM 'C' is fully inserted and reading 2 cps.
SRM 'D' is fully inserted and reading 8 cps.

b. SRM 'A' is bypassed.
SRM 'B' is fully inserted and reading 10 cps.
SRM 'C' is bypassed.
SRM 'D' is fully inserted and reading 4 cps.

c. SRM 'A' is fully inserted and reading 6 cps.
SRM 'B' is bypassed.
SRM 'C' is bypassed.
SRM 'D' is fully inserted and reading 9 cps.

d. SRM 'A' is fully inserted and reading 12 cps.
SRM 'B' is fully inserted and reading 4 cps.
SRM 'C' is fully inserted and reading 7 cps.
SRM 'D' is bypassed.

QUESTION: 032 (1.00)

A normal plant startup is in progress with the mode switch in STARTUP. IRM A is failed downscale and bypassed with its function switch on panel 936 in STBY. All the IRM range switches are on Range 2. WHICH ONE of the following describes the expected plant response if IRM A is taken out of bypass?

a. No action

b. Rod block

c. Rod block and half scram

d. Rod block and full scram

QUESTION: 033 (1.00)

The Transverse Incore Probe (TIP) detector was performing an automatic scan when power was lost to 120 V lighting panel 17L. Simultaneous with the loss of power, the reactor scrammed and RPV water level dropped to 0 inches. WHICH ONE of the following describes the response of the TIP System?

a. The TIP detector will remain in the core, the ball valve will remain open, and the shear valve must be fired manually.

b. The TIP detector will remain in the core, the ball valve will close, and the shear valve will fire automatically.

c. The TIP detector will retract, the ball valve will remain open, and the shear valve must be fired manually.

d. The TIP detector will retract, the ball valve will close, and it is not necessary to fire the shear valve.

QUESTION: 034 (1.00)

With the plant operating at 90% power, the APRM Calibration section of the OD-3 Printout provided the following results:

| APRM | 1(A) | 2(C) | 3(E) | 4(B) | 5(D) | 6(F) |
|---|---|---|---|---|---|---|
| READING | 91.1 | 90.3 | 88.2 | 90.0 | 88.7 | 89.4 |
| AGAF | 0.975 | 1.102 | 0.994 | 1.003 | 0.987 | 1.058 |

WHICH ONE of the following identifies all the APRMs that require calibration?

a. APRM A, APRM D, and APRM E

b. APRM D, APRM E, and APRM F

c. APRM A, APRM C, and APRM F

d. APRM B, APRM C, and APRM F

QUESTION: 035 (1.00)

The plant is operating at 95% power. The following are the indications received when the APRM meter function switches on Panel 937 are placed in the AVERAGE, COUNT, and FLOW positions:

| | AVERAGE | COUNT | FLOW |
|---|---|---|---|
| APRM A | 98% | 70% | 85% |
| APRM B | 94% | 70% | 97% |
| APRM C | 96% | [illegible]0% | 85% |
| APRM D | 99% | 55% | 97% |
| APRM E | 93% | 55% | 85% |
| APRM F | 95% | 70% | 97% |

WHICH ONE of the following describes the expected plant response for these conditions?

a. No action

b. Rod block

c. Rod block and half scram

d. Rod block and full scram

QUESTION: 036 (1.00)

The plant was operating at 100% power when the 'A' Recirculation pump tripped. All of the appropriate actions were taken in response to the recirc pump trip. The following annunciators are lit:

"ROD WITHDRAW BLOCK" (C905L, A1)
"ROD BLOCK MONITOR DOWNSCALE" (C905R, F3)

The RBM indicates 75% for the selected rod. WHICH ONE of the following describes the cause of these alarms?

a. A voltage transient during the recirc pump trip caused the RBM to momentarily lower below 5/125 of scale causing the downscale trip.

b. As core power decreased, the local power around the selected rod also decreased. This caused the RBM output signal to decrease below 94% of the reference signal causing the downscale trip.

c. Prior to the transient, the RBM High Power Trip Setpoint was activated for the selected rod. The transient caused power to decrease prior to leveling out at 75%. As reactor power lowered into the Low Power Trip Setpoint, the downscale trip was actuated.

d. The recirc pump trip caused 'A' recirc loop flow to decrease. As a result, the 'A' flow converter input a downscale signal into the 'A' RBM causing the downscale trip.

QUESTION: 037 (1.00)

Reactor power is at 16% during a reactor startup. Control rod 30-15 is at position 12 and has been selected to be withdrawn to position 48. The reed switch for Rod 30-15, position 32 is faulty and will not actuate. WHICH ONE of the following describes the expected response if the rod movement control switch is taken to the NOTCH OUT position simultaneously with the "Emergency in/notch override" switch being taken to the NOTCH OVERRIDE position?

a. No rod motion will occur.

b. A rod block will be initiated at position 32 and Rod 30-15 will settle at position 32.

c. A rod block will be initiated at position 32 and Rod 30-15 will settle at position 34.

d. Rod 30-15 will withdraw to position 48.

QUESTION: 038 (1.00)

Control rods are being withdrawn with reactor power on IRM range 4, in accordance with the attached Control Rod Sequence Sheet. Rod Group No. 8 is latched. WHICH ONE the following manipulations will result in a RWM select error?

a. Rod 14-39 is at position 08
Rod 38-39 is at position 06
Rod 38-15 is at position 04
Rod 14-15 is at position 14
Rod 14-39 is selected

b. Rod 14-39 is at position 10
Rod 38-39 is at position 12
Rod 38-15 is at position 06
Rod 14-15 is at position 06
Rod 38-39 is selected

c. Rod 14-39 is at position 04
Rod 38-39 is at position 06
Rod 38-15 is at position 04
Rod 14-15 is at position 12
Rod 38-15 is selected

d. Rod 14-39 is at position 14
Rod 38-39 is at position 08
Rod 38-15 is at position 10
Rod 14-15 is at position 04
Rod 14-15 is selected

QUESTION: 039 (1.00)

A 1st point feedwater heater was just removed from service. There has been no change in turbine steam flow. WHICH ONE of the following describes the effect on the plant?

a. Main generator output decreases. Plant efficiency increases.

b. Main generator output decreases. Plant efficiency decreases.

c. Main generator output increases. Plant efficiency increases.

d. Main generator output increases. Plant efficiency decreases.

QUESTION: 040 (1.00)

The plant was operating at 100% power when the "RFP A LOW NPSH" (C1L, A1) annunciator is received. The RFP sequential trip selector switch is "ON" and the reactor feed pump tripping sequence switch is in the "CAB" position. WHICH ONE of the following describes the potential cause of the alarm and the expected automatic actions?

a. The alarm was caused by a trip of one condensate pump. The condenser reject valves open. RFP A will trip if the low NPSH condition persists for 15 seconds.

b. The alarm was caused by a trip of one condensate pump. The condenser reject valves shut. RFP C will trip if the low NPSH condition persists for 15 seconds.

c. The alarm was caused by placing a condensate demineralizer in service. The condenser reject valves shut. RFP A will trip if the low NPSH condition persists for 15 seconds.

d. The alarm was caused by removing a condensate demineralizer from service. The condenser reject valves open. RFP C will trip if the low NPSH condition persists for 15 seconds.

QUESTION: 041 (1.00)

The plant is operating at 100% power with the FWLC setpoint set at +28 inches. WHICH ONE of the following describes the effect of a loss of one feedwater flow input to the FWLC System?

a. Feedwater flow decreases, reactor water level decreases, and the reactor scrams.

b. Feedwater flow decreases and reactor water level decreases, but the reactor should not scram.

c. Feedwater flow increases, reactor water level increases, and the reactor may scram.

d. Feedwater flow increases and reactor water level increases, but the reactor should not scram.

QUESTION: 042 (1.00)

The plant was operating at 100% power with the MHC System operating normally when the Bypass Opening Jack (BOJ) was taken to the RAISE position. WHICH ONE of the following describes the expected plant response if the BOJ cannot be returned to the OFF position?

a. The bypass valves will open fully. Reactor pressure will decrease and stabilize at a slightly lower value than before the transient.

b. The bypass valves will open fully, then the control valves will open until the control valve limit stop is reached. Reactor pressure will decrease and stabilize at a lower value than before the transient.

c. The control valves will open until the control valve limit stop is reached, then the bypass valves will open fully. Reactor pressure will decrease until the MSIVs close.

d. The control valves will open until the control valve limit stop is reached, then the bypass valves will open until the reactor flow limit is reached. Reactor pressure will decrease until the MSIVs close.

QUESTION: 043 (1.00)

A turbine runback was initiated due to high stator outlet coolant temperature at time zero. The following table shows the generator load and stator outlet coolant temperature response during the runback:

| TIME | STATOR AMPS | OUTLET TEMPERATURE |
|---|---|---|
| +0.5 minutes | 25,300 | 88°C |
| +1.0 minutes | 21,800 | 86°C |
| +1.5 minutes | 18,300 | 84°C |
| +2.0 minutes | 14,800 | 82°C |
| +2.5 minutes | 11,300 | 80°C |
| +3.0 minutes | 7,800 | 78°C |
| +3.5 minutes | 4,300 | 76°C |
| +4.0 minutes | 800 | 74°C |

WHICH ONE of the following correctly describes the plant response?

a. The turbine should have tripped at +2.0 minutes.

b. The turbine should have tripped at +3.5 minutes.

c. The runback should have stopped at +1.0 minutes.

d. The runback should have stopped at +3.5 minutes.

QUESTION: 044 (1.00)

The main condenser vapor valves (AO-3703, AO-3704, AO-3710, and AO-3711) have shut automatically while the plant was operating at 100% power. WHICH ONE of the following conditions could have caused the isolation?

a. Offgas high radiation caused by abnormal carbon vault temperature

b. Hydrogen concentration greater than 1 percent caused by a recombiner malfunction

c. Offgas high pressure caused by an explosion in the offgas system

d. High offgas flow caused by lowering condenser vacuum

QUESTION: 045 (1.00)

The plant is in single loop operation. For WHICH ONE of the following conditions can the idle recirculation pump be started? (Consider administrative and functional limitations.)

a. The idle recirculation pump suction and discharge valves are fully open. The scoop tube lock light is off and speed control is in manual with an output setpoint of zero.

b. The idle recirculation MG set generator field breaker is open. Lube oil pressure is 17 psig and lube oil temperature is 90°F.

c. The operating pump is running at 40% of rated speed. Core flow is 35 Mlb/hr. Core thermal power is 1050 MWT.

d. The idle pump suction temperature is 408°F. The operating pump suction temperature is 435°F. Bottom head drain temperature is 492°F. Vessel dome temperature is 548°F.

QUESTION: 046 (1.00)

The plant is operating at 50% power with both recirculation pumps in manual. WHICH ONE of the following describes the expected response if all FWLC System input were lost to Recirc Pump B's flow controller?

a. Recirc Pump B runback to 65% due to a Speed Limiter #2 runback.

b. Recirc Pump B runback to 26% due to a Speed Limiter #1 runback.

c. Recirc Pump B scoop tube will lockup due to a speed control signal failure.

d. There will be no effect on Recirc Pump B because reactor water level is normal and the pump discharge valve is full open.

QUESTION: 047 (1.00)

A LOCA has occurred concurrent with a loss of AC power. Both diesel generators (DGs) started, but the electric governor for DG B malfunctioned immediately after the start signal was received. WHICH ONE of the following describes the expected response of the DG B output breaker and DG B?

a. The DG output breaker will not close because the DG will not come up to speed and voltage. The DG will continue to run.

b. The DG output breaker will close because the mechanical governor will take over. The DG will continue to run.

c. The DG output breaker will close, but will reopen as DG speed increases due to overcurrent. The DG will continue to run.

d. The DG output breaker will close, but the DG will trip on overspeed causing the output breaker to trip open.

QUESTION: 048 (1.00)

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The Standby Gas Treatment (SGT) System initiated on a valid initiation signal 3 minutes ago. Prior to the initiation, both SGT trains were in the normal standby lineup. No operator action has been taken and the initiation signal is still present. The SGT train A heater just tripped due to high temperature. WHICH ONE of the following describes the expected response of the SGT System?

a. SGT train A fan will trip. SGT train A inlet and outlet dampers will close. SGT train B fan will continue to run. SGT train B inlet and outlet dampers will remain open.

b. SGT train A fan will trip. SGT train A inlet and outlet dampers will close. SGT train B fan will start. SGT train B inlet and outlet dampers will open.

c. SGT train A fan will continue to run. SGT train A inlet and outlet dampers will remain open. SGT train B fan will continue to run. SGT train B inlet and outlet dampers will remain open.

d. SGT train A fan will continue to run. SGT train A inlet and outlet dampers will remain open. SGT train B fan will not start. SGT train B inlet and outlet dampers will remain closed.

QUESTION: 049 (1.00)

WHICH ONE of the following conditions will cause a secondary containment isolation?

a. Refuel floor rad monitor 1705-8A: 25 mR/hr
   Refuel floor rad monitor 1705-8B: 15 mR/hr
   Refuel floor rad monitor 1705-8C: 20 mR/hr
   Refuel floor rad monitor 1705-8D: 10 mR/hr

b. Refuel floor rad monitor 1705-8A: 10 mR/hr
   Refuel floor rad monitor 1705-8B: 15 mR/hr
   Refuel floor rad monitor 1705-8C: 20 mR/hr
   Refuel floor rad monitor 1705-8D: 0.1 mR/hr

c. Refuel floor rad monitor 1705-8A: 0.1 mR/hr
   Refuel floor rad monitor 1705-8B: 20 mR/hr
   Refuel floor rad monitor 1705-8C: 0.1 mR/hr
   Refuel floor rad monitor 1705-8D: 0.1 mR/hr

d. Refuel floor rad monitor 1705-8A: 15 mR/hr
   Refuel floor rad monitor 1705-8B: 20 mR/hr
   Refuel floor rad monitor 1705-8C: 0.1 mR/hr
   Refuel floor rad monitor 1705-8D: 0.1 mR/hr

QUESTION: 050 (1.00)

Reactor Building ventilation fans must be started in the proper order to maintain reactor building pressure and prevent the spread of contamination. WHICH ONE of the following describes the proper sequence for starting reactor building ventilation fans?

a. Exhaust fans should be started before supply fans. Contaminated exhaust (Zone 3) fans should be started before clean exhaust (Zone 2) fans.

b. Exhaust fans should be started before supply fans. Clean exhaust (Zone 2) fans should be started before contaminated exhaust (Zone 3) fans.

c. Supply fans should be started before exhaust fans. Clean exhaust (Zone 2) fans should be started before contaminated exhaust (Zone 3) fans.

d. Supply fans should be started before exhaust fans. Contaminated exhaust (Zone 3) fans should be started before clean exhaust (Zone 2) fans.

QUESTION: 051 (1.00)

Control room environmental control supply fan HS-77(VSF-103A) is in STBY and supply fan HS-78(VSF-103B) is in AUTO. WHICH ONE of the following describes the Control Room HVAC System response to a Halon initiation?

a. Both control room environmental control supply fans start. Normal supply and exhaust fans trip. Halon exhaust fan starts and damper AO N-141 shuts.

b. Control room environmental control supply fan HS-78 starts. Normal supply and exhaust fans trip. Cable spreading room supply and exhaust dampers shut.

c. Both control room environmental control supply fans start. Normal air intake dampers close and filtration system isolation dampers open. Cable spreading room supply and exhaust dampers shut.

d. Control room environmental control supply fan HS-78 starts. Normal air intake dampers close and filtration system isolation dampers open. Halon exhaust fan starts and damper AO N-141 shuts.

QUESTION: 052 (1.00)

If the Fuel Pool Cooling (FPC) System is unavailable, WHICH ONE of the following systems can be crosstied for pool cooling?

a. Fire Protection

b. Residual Heat Removal

c. Reactor Building Closed Cooling Water

d. Condensate and Demineralized Water Storage and Transfer

QUESTION: 053 (1.00)

The reactor mode switch is in REFUEL. In WHICH ONE of the following conditions would control rod withdrawal be possible?

a. One control rod withdrawn to position 02. The withdrawn control rod is selected. The bridge is over the core. The main hoist is loaded with 600 lbs. The grapple is fully up.

b. All control rods are full in. The bridge is NOT over or near the core. The monorail hoist is loaded with 300 lbs. The grapple is NOT fully up.

c. One control rod withdrawn to position 02. A second control rod is selected. The bridge is over the core. The frame mounted hoist is loaded with 200 lbs. The grapple is fully up.

d. All control rods are full in. The bridge is NOT over or near the core. The service platform hoist is loaded with 500 lbs.

QUESTION: 054 (1.00)

WHICH ONE of the following Fire Protection Systems would be utilized to suppress a fuel oil fire in the DG fuel oil pipe trench?

a. Dry Chemical System

b. Halon System

c. Cardox System

d. Fire Water System

deleted

QUESTION: 055 (1.00)

The plant is operating at 100% power. WHICH ONE of the following conditions would NOT require initiation of a plant shutdown in accordance with Technical Specifications?

a. One ADS valve is inoperable. The HPCI inverter tripped on high voltage. Voltage returned to normal, but the inverter failed to automatically reset. Manual reset of the inverter is expected to take 15 minutes.

b. RCIC is inoperable. The HPCI gland seal condenser is inoperable. Repairs are expected to take 12 hours.

c. HPCI is inoperable. RHR valve MO-1001-37A (Torus Spray Valve) is stuck in the closed position. Repairs are expected to take 2 days.

d. LPCI B is inoperable. The HPCI flow indicator controller ramp generator is inoperable. A replacement part had to be ordered and will not be available for 5 days.

QUESTION: 056 (1.00)

The plant is operating at 100% power. PNPS Proc. No. 8.4.1, "Standby Liquid Control Pump Oper and Flow Test has just been performed." The surveillance data sheets are attached. Current SLC System status is as follows:

| | |
|---|---|
| Boron Enrichment: | 55.0 Atom Percent |
| SLC Tank Concentration: | 8.72% by weight |
| SLC Tank Volume (C905): | 1900 gallons |
| SLC Tank Temperature: | 45°F |

WHICH ONE of the following describes the required actions in accordance with Technical Specifications?

a. Continued reactor operation is permitted for seven days.

b. The reactor shall be placed in COLD SHUTDOWN with all operable control rods inserted within the next 24 hours.

c. Flow test the SLC System to verify a flow path. If the test is successful, there are no restrictions on reactor operation.

d. There are no restrictions on reactor operation.

QUESTION: 057 (1.00)

The plant is operating at 100% power with the shutdown transformer out of service for maintenance. The offsite 345 kV transmission line from Bridgewater Station is unavailable due to a problem at Bridgewater Station. It has just been reported that Diesel Generator A failed its monthly operability test. WHICH ONE of the following is the appropriate action in accordance with Technical Specifications?

a. Continued reactor operation is permissible for 7 days.

b. Continued reactor operation is permissible for 72 hours.

c. Reduce reactor power level to 25% and noti.y the NRC within 1 hour.

d. Initiate an orderly shutdown and be in COLD SHUTDOWN within 24 hours.

QUESTION: 058 (1.00)

The plant is operating at 100% power. On the most recent P-1 computer printout MAPRAT is 1.02. WHICH ONE of the following is the appropriate action that should be taken in accordance with Technical Specifications?

a. Within two hours, restore MAPRAT > 1.04 and insert all insertable control rods.

b. Within two hours, restore MCPR within limits and insert all insertable control rods.

c. Within 15 minutes, initiate action to restore APLHGR within limits.

d. Within 15 minutes, initiate action to restore MCPR within limits.

QUESTION: 059 (1.00)

The Salt Service Water (SSW) System is operating with SSW Pumps A, C, and E running with the plant at 100% power. SSW Pump B is in AUTO and SSW Pump D is tagged out for maintenance. A fault on MCC B-10 causes SSW Pump C to trip. WHICH ONE of the following describes the required actions in accordance with Technical Specifications while the MCC is being repaired?

a. No action is required.

b. Reactor operation may continue for 7 days.

c. Reactor operation may continue for 72 hours.

d. Initiate a shutdown and be in COLD SHUTDOWN within 24 hours.

QUESTION: 060 (1.00)

Core offloading is in progress. Standby Gas Treatment (SGT) Train A has been inoperable for four days. A spent fuel bundle is grappled over the core, when it is reported that the SGT Train B fan is operating at 3500 cfm during surveillance testing. WHICH ONE of the following describes the appropriate actions?

a. Lower the fuel bundle into its designated location in the spent fuel pool, then cease fuel movement.

b. Lower the fuel bundle into the nearest open location in the reactor vessel, then cease fuel movement.

c. Fuel movement may continue for seven days provided all other components of SGT B are operable.

d. Fuel movement may continue for three days provided secondary containment integrity is maintained.

QUESTION: 061 (1.00)

PNPS Proc. No. 2.4.29, "Stuck Open Relief Valve," directs the operator to initiate a manual scram if a safety/relief valve (SRV) cannot be closed. WHICH ONE of the following describes the reason for this requirement?

a. Suppression pool temperature could increase and cause an increase in drywell temperature and pressure.

b. Rapid shutdown, cooldown, and depressurization of the reactor minimizes the coolant loss through the stuck open SRV.

c. Technical Specifications require that the self actuating relief mode for all SRVs be operable.

d. Sustained pressure and flow will damage the relief valve downstream piping.

QUESTION: 062 (1.00)

During preparations for refueling operations, with the plant in HOT SHUTDOWN, an unisolable leak in the Spent Fuel Pool caused fuel storage pool level to decrease below the top of the fuel bundles in the pool. General area radiation levels on the refuel floor are 250 mR/hr. Radiation levels in the FPC pump room and skimmer surge tank area are 750 mR/hr. Water levels in the SE and SW quadrants are 8 inches above the floor. WHICH ONE of the following is the appropriate emergency action level for this situation?

a. None

b. Unusual Event

c. Alert

d. Site Area Emergency

QUESTION: 063 (1.00)

The Control Room was evacuated due to a fire 5 minutes ago. All the immediate actions of PNPS Proc. No. 2.4.143, "Shutdown from Outside the Control Room," were performed with the exception of tripping the Reactor Feed Pumps. The Feed Pumps were tripped at their 4160 VAC breakers. Current plant conditions are:

- The reactor has been verified to be shutdown.
- RPV level is > +50 inches (pegged high).
- RPV pressure is 700 psig and decreasing slowly.
- The MSIVs are open and cannot be closed.
- The turbine bypass valves appear to be functioning normally.
- An operator is stationed to control RPV level and pressure with HPCI as necessary.

WHICH ONE of the following is the appropriate emergency action level for this situation?

a. Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

QUESTION: 064 (1.00)

An Anticipated Transient Without Scram (ATWS) has occurred. Reactor power is 5%. Reactor pressure is 1175 psig and reactor water level is +25 inches. Torus water temperature is 105°F and torus water level is 178 inches. WHICH ONE of the following is the appropriate emergency action level for this situation? (Assume no operator action.)

a. Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

QUESTION: 065 (1.00)

No fuel movement is in progress. WHICH ONE of the following conditions would be a violation of Secondary Containment integrity?

a. Reactor Building differential pressure is greater than 0 inches of water in STARTUP.

b. Both reactor building to torus vacuum breakers are inoperable and stuck in the open position in RUN.

c. Both Standby Gas Treatment trains are inoperable in COLD SHUTDOWN with the reactor coolant system vented.

d. Both reactor building airlock doors are open to remove a large piece of equipment from the reactor building in HOT SHUTDOWN.

QUESTION: 066 (1.00)

WHICH ONE of the following conditions would be a safety limit violation?

a. While operating at 30% power, the MHC pressure regulators fail. Reactor pressure drops to 800 psig before the MSIVs close and the reactor scrams.

b. While operating at 75% power, a malfunction in the master recirculation flow controller causes the speed of both recirc pumps to increase to the high speed stop. The Minimum Critical Power Ratio is 1.05.

c. While operating at 90% power, a turbine trip occurs. The reactor fails to scram and the ATWS system logic trips the reactor feed pumps.

d. While operating at 100% power, an inadvertent MSIV closure causes reactor pressure to increase. The reactor scrams and reactor pressure increases until both safety valves lift.

QUESTION: 067 (1.00)

WHICH ONE of the following describes the expected indications if Jet Pump 5 failed?

a. Recirc Loop A flow increases. Total core flow decreases.

b. Recirc Loop A flow increases. Total core flow increases.

c. Recirc Loop A flow decreases. Total core flow decreases.

d. Recirc Loop A flow decreases. Total core flow increases.

QUESTION: 068 (1.00)

The plant was operating at 96% power on the 100% load line for several months when Recirc Pump A tripped due to operator error. Plant conditions following the trip are:

| | |
|---|---|
| Reactor Power: | 60% |
| Recirc Loop A Flow: (FI-263-107A) | 2.0 MLB/HR |
| Recirc Loop B Flow: (FI-263-107B) | 30.5 MLB/HR |
| Total Core Flow: (FR-263-110) | 32.5 MLB/HR |

WHICH ONE of the following describes the appropriate action in accordance with PNPS Proc. No. 2.4.17, "Recirculation Pump(s) Trip," (attached)?

a. Increase the speed of Recirc Pump B until loop flow is greater than 31.5 MLB/HR.

b. Insert control rods to decrease reactor power below the 80% load line.

c. Restart the idle pump in accordance with PNPS 2.2.84, "Reactor Recirculation System."

d. Scram the reactor and concurrently perform PNPS 2.1.6, "Reactor Scram."

QUESTION: 069 (1.00)

The plant was operating at 100% power when a break developed on the instrument air header. WHICH ONE of the following describes automatic actions that should occur before air header pressure drops below 75 psig?

a. Service air header isolates
Backup K104 air compressor starts
Feedwater control valves lock up

b. Non-essential instrument air isolates
Feedwater control valves lock up
Scram pilot valve air header depressurizes

c. Drywell pneumatic supply header isolates
Lagging compressor loads
Service air header isolates

d. Low pressure service air crossconnect isolates (if open)
Backup K104 air compressor starts
Non-essential instrument air isolates

QUESTION: 070 (1.00)

The plant was operating at 100% power when a complete loss of instrument air occurred. Power was also lost to the AC solenoid operated air cutoff valves for the inboard MSIVs and to the DC solenoid operated air cutoff valves for the outboard MSIVs. The primary containment is inerted. WHICH ONE of the following describes the response of the MSIVs?

a. Inboard MSIVs remain open. Outboard MSIVs remain open.

b. Inboard MSIVs close due to air pressure. Outboard MSIVs close due to air pressure.

c. Inboard MSIVs close due to spring pressure. Outboard MSIVs remain open.

d. Inboard MSIVs remain open. Outboard MSIVs close due to spring pressure.

QUESTION: 071 (1.00)

An inadvertent containment isolation signal was received and immediately cleared. No operator action was taken. WHICH ONE of the following sets of valves do NOT require a manual reset to reopen the valves if they isolated on the specified signal?

| | Valves | Isolation Signal |
|---|---|---|
| a. | Recirc sample lines | Main Steam Line low pressure |
| b. | HPCI exhaust vacuum breaker isolation valves | High drywell pressure (w/ low reactor pressure) |
| c. | RCIC inlet steam valves | Low reactor pressure |
| d. | RHR/LPCI Injection valves | High reactor pressure (in Shutdown Cooling mode) |

QUESTION: 072 (1.00)

Three RBCCW pumps are unavailable. WHICH ONE of the following describes the appropriate action that must be taken with the specified pumps unavailable?

a. RBCCW pumps A, B, and C are unavailable. The RBCCW loops must be crosstied to prevent a reactor scram on high drywell pressure due to a loss of drywell cooling.

b. RBCCW pumps A, C, and E are unavailable. The RBCCW loops must be crosstied to prevent a trip of the recirculation pumps due to loss of cooling.

c. RBCCW pumps B, D, and F are unavailable. It is not necessary to crosstie the RBCCW loops because at least one pump in each loop is still available.

d. RBCCW pumps D, E, and F are unavailable. It is not necessary to crosstie the RBCCW loops because at least one pump in each loop is still available.

QUESTION: 073 (1.00)

The Salt Service Water (SSW) System was operating with SSW Pumps A, C, and D running and SSW Pumps B and E in AUTO. The loop selector switch is in its normal position. A complete loss of normal AC power occurs simultaneously with a loss of coolant accident (LOCA). WHICH ONE of the following describes the response of the SSW System with no operator action?

a. TBCCW heat exchanger outlet valves throttle 90% closed.
RBCCW heat exchanger outlet valves open fully.
SSW Pumps A and D restart after load shedding.

b. TBCCW heat exchanger outlet valves throttle 90% closed.
RBCCW heat exchanger outlet valves open fully.
SSW Pumps B and E start after load shedding.

c. TBCCW heat exchanger outlet valves open fully.
RBCCW heat exchanger outlet valves throttle 90% closed.
SSW Pumps A and C restart after load shedding.

d. TBCCW heat exchanger outlet valves open fully.
RBCCW heat exchanger outlet valves throttle 90% closed.
SSW Pumps B and C start after load shedding.

QUESTION: 074 (1.00)

The plant is operating at 800 MWt. Core flow is 35 MLB/HR. Condenser vacuum is 18" Hg. WHICH ONE of the following describes the appropriate actions to be taken under these conditions?

a. Reduce recirculation pump speed, maintaining core flow above 31.5 MLB/HR, until the vacuum decrease is terminated.

b. Reduce recirculation pump speed until the vacuum decrease is terminated OR the pumps reach minimum speed.

c. Insert control rods in reverse order of the pull sheet as necessary to stop the vacuum decrease.

d. Immediately scram the reactor and trip the turbine.

QUESTION: 075 (1.00)

The plant was operating at 95% power when the Train A 4th point feedwater heater isolated due to a tube rupture. Reactor power increased to 100%. WHICH ONE of the following describes the appropriate immediate actions?

a. Runback recirculation flow and insert control rods as necessary to maintain reactor power below 95%.

b. Runback recirculation flow and insert control rods as necessary to maintain reactor power below 25%.

c. Runback recirculation flow until reactor power is below 75% or core flow reaches 31.5 Mlb/hr.

d. Runback recirculation flow until reactor power reaches 70% or core flow reaches 31.5 Mlb/hr.

QUESTION: 076 (1.00)

The plant is operating at 100% power when the in-service CRD flow control valve malfunctions causing CRD system flow to oscillate. WHICH ONE of the following subsequent CRD system failures would require an immediate reactor scram?

a. Trip of both CRD pumps

b. Loss of all rod position indication

c. More than one control rod in a 9-rod array drifts

d. More than one CRD mechanism high temperature alarm in a 9-rod array

QUESTION: 077 (1.00)

Core offloading is in progress. A spent fuel bundle is grappled over the core, when the Refueling Floor Area Radiation Monitor (ARM) alarms. No other alarms have been received on the refuel floor or in the Control Room. WHICH ONE of the following describes the appropriate actions?

a. Lower the fuel bundle into its designated location in the spent fuel pool.

b. Lower the fuel bundle into the nearest open location in the spent fuel pool.

c. Lower the fuel bundle into the nearest open location in the reactor vessel.

d. Lower the fuel bundle into the location from which it was removed in the reactor vessel.

QUESTION: 078 (1.00)

An electrical failure has occurred causing multiple alarms to annunciate. One of the illuminated alarms is annunciator "VITAL INST SYS LOSS OF DC POWER," (C3 Center, window D5). Based on this information, WHICH ONE of the following describes the emergency procedures that are applicable for this situation?

a. PNPS Proc. No. 5.3.6, "Loss of Vital AC (Y-2)" and PNPS Proc. No. 5.3.13, "Loss of Essential DC Bus D6"

b. PNPS Proc. No. 5.3.6, "Loss of Vital AC (Y-2)" and PNPS Proc. No. 5.3.30, "Loss of 250V DC Bus D-10"

c. PNPS Proc. No. 5.3.7, "Loss of Instrument Power Bus (Y-1)" and PNPS Proc. No. 5.3.13, "Loss of Essential DC Bus D6"

d. PNPS Proc. No. 5.3.7, "Loss of Instrument Power Bus (Y-1)" and PNPS Proc. No. 5.3.30, "Loss of 250V DC Bus D-10"

QUESTION: 079 (1.00)

A station blackout has occurred and the SBO diesel failed to start. RCIC and HPCI were being used to control RPV level and pressure when both systems tripped and could not be restarted. PNPS Proc. No. 5.3.31, "Station Blackout," directs use of PNPS Proc. No. 5.3.26, "RPV Injection During Emergencies," for alternate methods. WHICH ONE of the methods would be used for RPV injection in this situation?

a. SSW crosstied to RHR

b. Fire Water crosstied to RHR

c. Condensate Transfer crosstied to ECCS fill lines

d. Demineralized Water Transfer crosstied to SBLC

QUESTION: 080 (1.00)

The Control Room has been evacuated due to a fire in the Cable Spreading Room. At WHICH ONE of the following locations would you find an Alternate Shutdown toolbox?

a. 3' Aux Bay

b. 23' RPS MG Room

c. 37' Switchgear Room

d. 51' Turbine Building

QUESTION: 081 (1.00)

Following a reactor scram, the operator is required to determine whether or not all control rods are inserted to or beyond position 02 in order to determine the appropriate actions to be taken in accordance with the EOPs and PNPS Proc. No. 2.1.6, "Reactor Scram." WHICH ONE of the following is an acceptable method for determining whether all rods are fully inserted?

a. Observing that all the blue scram lights are illuminated on the full-core display.

b. Verifying that all APRMs are downscale and reactor power is trending down on the IRMS.

c. Prior to scram reset, verifying that the rod drift annunciator will not clear.

d. With the mode switch in REFUEL, taking the rod select power off, then back on, and observing if the Refuel Mode Select Permissive light illuminates.

QUESTION: 082 (1.00)

EOP-02, "RPV Control, Failure-to-Scram," directs actions based on whether or not the reactor is shutdown. For WHICH ONE of the following conditions would the reactor be considered shutdown in accordance with EOP-02?

a. Reactor pressure is 0 psig. The Hot Shutdown Boron Weight of sodium pentaborate has been injected into the RPV.

b. Reactor power is 20 on Range 5. Reactor period is +100 and stable.

c. Reactor power is 50 on Range 6. Reactor period is negative.

d. Reactor power is 30 on Range 9. Reactor period is negative.

QUESTION: 083 (1.00)

A reactor scram has occurred and all control rods are not inserted. WHICH ONE of the following indications would be expect d if the failure of the rods to insert was due to an electrical failure of the Reactor Protection System (RPS)?

a. Alarm "SCRAM VALVE PILOT HEADER LO PRESSURE" (C905R, A6) illuminated.

b. Alarm "SCRAM DISCH VOLUME HI LEVEL SCRAM" (C905R, I4) illuminated.

c. Group Scram logic lights on C905 illuminated.

d. Blue Scram lights on C905 illuminated.

QUESTION: 084 (1.00)

A reactor scram has occurred and all control rods are not inserted. In accordance with PNPS Procedure No. 5.3.23, "Alternate Rod Insertion," WHICH ONE of the following methods for inserting control rods requires the scram to be reset?

a. Venting the overpiston areas of the control rod drives.

b. Individually scramming control rods from panel C916.

c. Venting the Scram air header and deenergizing the Scram solenoids.

d. Inserting a manual Scram from panel C905.

QUESTION: 085 (1.00)

WHICH ONE of the following actions should be taken to maximize drywell cooling in accordance with EOP-03, "Primary Containment Control?"

a. Initiate drywell spray.

b. Secure the recirculation pumps.

c. Increase the RBCCW system temperature.

d. Increase the RBCCW system flow rate.

QUESTION: 086 (1.00)

WHICH ONE of the following lineups would maximize torus cooling in accordance with EOP-03, "Primary Containment Control?"

a. Both RHR loops in service with one pump per loop operating and the RHR heat exchanger bypass valves shut.

b. One RHR loop in service with both pumps operating and the RHR heat exchanger bypass valve shut.

c. Both RHR loops in service with one pump per loop operating and the RHR heat exchanger bypass valves open.

d. One RHR loop in service with both pumps operating and the RHR heat exchanger bypass valve open.

QUESTION: 087 (1.00)

A station blackout and complete loss of instrument air has occurred. Plant conditions are as follows:

- -Drywell pressure is 3 psig and increasing slowly
- -Torus water level is 220 inches
- -Neither Standby Gas Treatment (SGTS) train is operable because the outlet dampers are stuck in the closed position

WHICH ONE of the following describes the reason that the Direct Torus Vent (DTV) cannot be used in this situation?

a. SGTS must have an operable vent path for use of the DTV.

b. Primary Containment conditions do not meet the criteria for use of the DTV.

c. The DTV path is not available during a station blackout.

d. The DTV path is not available during a loss of instrument air.

QUESTION: 088 (1.00)

Both torus water level indicators are upscale and the primary containment water level indicator is downscale. For WHICH ONE of the following conditions is this an accurate indication? (PNPS Proc. No. 5.3.27, "Determining Primary Containment Water Level," is attached.)

a. Drywell Wide Range Pressure: 25 psig
Torus Bottom Pressure: 10 psig

b. Drywell Wide Range Pressure: 5 psig
Torus Bottom Pressure: 35 psig

c. Drywell Wide Range Pressure: 20 psig
Torus Bottom Pressure: 40 psig

d. Drywell Wide Range Pressure: 26 psig
Torus Bottom Pressure: 20 psig

QUESTION: 089 (1.00)

Following a major loss of coolant accident and loss of off-site power, RPV level cannot be maintained above TAF. The operating crew determines that an alternate injection system needs to be aligned in order to recover RPV level. Reports from the field indicate that the aux. bay is inaccessible. WHICH ONE of the following lineups could be used to deliver a high capacity flowrate to the vessel given the current plant conditions?

a. SSW crosstied to RHR

b. Fire water crosstied to RHR

c. Fire water crosstied to Feedwater

d. Demineralized water transfer crosstied to SBLC

QUESTION: 090 (1.00)

A primary system leak has made the Reactor Building inaccessible. The following conditions exist:

- Reactor pressure is 200 psig
- RPV water level (indicated) is +5 inches
- Drywell temperature is 300°F

WHICH of the following instruments, if any, can be used to determine RPV water level?

a. Narrow Range, Fuel Zone, and FW Control

b. Narrow Range and FW Control

c. FW Control

d. None

QUESTION: 091 (1.00)

EOP-02, "Failure to Scram," Step L-10 directs you to stop and prevent all injection into the RPV except for boron, CRD, and RCIC. WHICH ONE of the following is the reason for securing injection in this step?

a. Securing injection sources helps to reduce the pressure in the reactor vessel while performing Alternate RPV Depressurization.

b. Securing injection sources prevents the addition of a large volume of cold, unborated water into the core.

c. The injection sources are not necessary, because adequate core cooling is assured by core submergence.

d. The injection sources are not necessary, because adequate core cooling is assured by steam cooling without injection.

QUESTION: 092 (1.00)

EOP-02, "RPV Control, Failure-to-Scram," is being implemented. The reactor is shutdown, but it has not yet been determined that it will remain shutdown under all conditions. WHICH ONE of the following describes the appropriate actions that should be taken if RPV level cannot be determined?

a. Enter EOP-16. Exit the RPV level leg of EOP-02.

b. Enter EOP-16. Exit the RPV level leg and the RPV pressure leg of EOP-02.

c. Enter EOP-26. Exit the RPV level leg of EOP-02.

d. Enter EOP-26. Exit the RPV level leg and the RPV pressure leg of EOP-02.

QUESTION: 093 (1.00)

The plant was operating at 100% power when the "MAIN STEAM LINE HI RADIATION SCRAM," (C905R, A5) alarm was received due to a valid signal. The expected automatic actions did NOT occur and have NOT been accomplished manually. Torus water temperature is 115°F and reactor power is 18%. Main Stack radiation levels are 75000 cps and have been steady for the last 20 minutes. WHICH ONE of the following describes the appropriate actions for RPV pressure control?

a. Perform Alternate RPV Depressurization using the SRVs.

b. Depressurize the RPV at less than 100°F/hr using the main turbine bypass valves.

c. Maintain reactor pressure less than 1085 psig using the main turbine bypass valves.

d. Maintain reactor pressure less than 1085 psig using HPCI in full flow test and the SRVs.

QUESTION: 094 (1.00)

Step TT-4 of EOP-03, "Primary Containment Control," directs concurrent performance of EOP-01, "RPV Control," before torus water temperature reaches the Boron Injection Initiation Temperature (BIIT). WHICH ONE of the following is the reason for performing EOP-01 before torus water temperature reaches the BIIT?

a. To reduce the potential for a condition that would compromise adequate core cooling which could preclude the use of RHR for torus cooling.

b. To ensure that the torus will be able to accept the heat addition from the RPV while the reactor is being shutdown without exceeding the Heat Capacity Temperature Limit.

c. To reduce the potential heat load on the torus by ensuring that the turbine bypass valves are used for pressure control and ADS is inhibited.

d. To assure cyclic condensation (chugging) does not occur when the RPV is depressurized by Alternate RPV Depressurization if the Heat Capacity Temperature Limit is exceeded.

QUESTION: 095 (1.00)

Following a LOCA, HPCI is the only injection source available. HPCI is maintaining RPV level at -50 inches with suction from the CST. The HPCI torus suction valves are stuck closed. For WHICH ONE of the following conditions would HPCI have to be secured irrespective of adequate core cooling in accordance with EOP-03, "Primary Containment Control?"

a. Torus water level: 80 inches
   RPV pressure: 600 psig

b. Torus water level: 250 inches
   RPV pressure: 1000 psig

c. Drywell water level: 30 feet
   RPV pressure: 400 psig

d. Torus water level: 200 inches
   RPV pressure: 800 psig

QUESTION: 096 (1.00)

WHICH ONE of the following conditions would require Alternate RPV Depressurization?

a. RPV Pressure: 1050 psig
Torus Bottom Pressure: 15 psig
Torus Water Temp: 120 °F
Torus Water Level: 100 inches
DW Temperature: 200 °F

b. RPV Pressure: 900 psig
Torus Bottom Pressure: 10 psig
Torus Water Temp: 140 °F
Torus Water Level: 85 inches
DW Temperature: 170 °F

c. RPV Pressure: 850 psig
Torus Bottom Pressure: 45 psig
Torus Water Temp: 175 °F
Torus Water Level: 220 inches
DW Temperature: 275 °F

d. RPV Pressure: 1000 psig
Torus Bottom Pressure: 20 psig
Torus Water Temp: 160 °F
Torus Water Level: 190 inches
DW Temperature: 225 °F

QUESTION: 097 (1.00)

The following conditions exist in primary containment:

- Torus bottom pressure is 50 psig
- Torus water level is 200 inches
- Drywell hydrogen concentration is 5%
- Drywell oxygen concentration is 7%
- Torus hydrogen concentration is 3%
- Torus oxygen concentration is 4%

WHICH ONE of the following describes the app opriate actions in accordance with EOP-03, "Primary Containment Control?"

a. Vent the drywell and establish a nitrogen purge.

b. Vent the torus and establish a nitrogen purge.

c. Vent the drywell and establish an air purge.

d. Vent the torus and establish an air purge.

QUESTION: 098 (1.00)

Following a Loss of Coolant Accident (LOCA), current plant conditions are:

- Torus bottom pressure is 10 psig
- Hydrogen concentration is 5%
- Oxygen concentration is 6%

Chemistry reports that projected offsite dose rates are 2 R/yr whole body. WHICH ONE of the following describes the correct action that should be taken to lower containment pressure and hydrogen and oxygen levels? (Attachment 1 of PNPS Proc. No. 5.4.6, "Primary Containment Venting and Purging Under Emergency Conditions," is attached.)

a. Do not vent the torus.

b. Vent the torus through SGTS using the 2" torus exhaust valves only.

c. Vent the torus through SGTS using the 2" torus exhaust valves and the 8" torus exhaust valves.

d. Vent the torus, bypassing SGTS.

QUESTION: 099 (1.00)

The reactor is shutdown with all control rods inserted. WHICH ONE of the following conditions would require Primary Containment Flooding?

a. RPV Pressure: 800 psig
RPV Water Level: -170 inches
Torus Pressure: 10 psig
No SRVs Open

b. RPV Pressure: 80 psig
RPV Water Level: -140 inches
Torus Pressure: 40 psig
1 SRV Open

c. RPV Pressure: 90 psig
RPV Water Level: Cannot Be Determined
Torus Pressure: 45 psig
2 SRVs Open

d. RPV Pressure: 60 psig
RPV Water Level: Cannot Be Determined
Torus Pressure: 25 psig
3 SRVs Open

QUESTION: 100 (1.00)

A primary system is discharging into secondary containment. WHICH ONE of the following conditions would require Alternate RPV Depressurization?

a. HPCI Turbine Area temperature: 200 °F
HPCI Piping Area - 23 ft El. temperature: 310 °F
HPCI Torus Piping Area temperature: 200 °F

b. RCIC Turbine Area temperature: 230 °F
RCIC Piping Area - 23 ft El. temperature: 200 °F
RCIC Torus Piping Area temperature: 270 °F

c. RWCU Backwash Tank Area temperature: 210 °F
RWCU Heat Exchanger Area temperature: 220 °F
RB West Area - 51 ft El. radiation level: 1200 mr/hr

d. RWCU & RHR Piping Area - 23 ft El. temp: 260 °F
CRD Quadrant water level: 8 in. above floor
CRD Pump Room - 17 ft 6 in El. rad level: 1200 mr/hr

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

deleted

ANSWER: 001 (1.00)

c.

REFERENCE:

1. IG: Ø-RO-02-11-01, "Control Room Computer System (EPIC/SPDS)," ELO 8.

[3.2/3.4]

294001A115 ..(KA's)

ANSWER: 002 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 6.1-024, "Radiological Posting of Areas of the Station."
2. Module C-GT-02-02-01, "Introduction to Radiation Protection," ELO 4.

[3.3/3.8]

294001K103 ..(KA's)

ANSWER: 003 (1.00)

b

~~e.~~

REFERENCE:

1. Module C-GT-02-02-02, "Exposure Limits," ELO 2.

[3.3/3.8]

294001K103 ..(KA's)

ANSWER: 004 (1.00)

a.

REFERENCE:

1. PNPS Proc. No. 1.4.9, "Storage, Handling, and Disposal of Sodium Pentaborate," page 6.
2. UG: O-RO-04-04, "Emergency Tasks," Task 91.

[3.1/3.4]

294001K110 ..(KA's)

ANSWER: 005 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 1.4.36, "High Pressure/Compressed Gas Cylinder Control."
2. IG: C-GT-01-01-03, "Industrial Safety," LO 30.

[3.4/3.8]

294001K109 ..(KA's)

ANSWER: 006 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 1.3.34, "Conduct of Operations," pages 38 and 39.

[3.7/3.7]

294001K101 ..(KA's)

ANSWER: 007 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 1.4.5, "PNPS Tagging Procedure."
2. UG: O-RO-04-02, "Administrative Tasks," Task 8.

[3.9/4.5]

294001K102 ..(KA's)

ANSWER: 008 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 1.3.67, "Control of Overtime."
2. UG: O-RO-04-10, "SRO On-Shift Tasks," Task 21.

[2.7/3.7]

294001A103 ..(KA's)

ANSWER: 009 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 1.4.12, "Primary Containment Entry."
2. Modified Facility Question ADMIN-12 (from Requal Retake Exam).

[3.2/3.6]

294001K113 ..(KA's)

ANSWER: 010 (1.00)

a.

REFERENCE:

1. PNPS Technical Specifications, Table 6.2-1.
2. IG: O-RO-06-01-04, "Technical Specification Design Factors and Admin Controls," ELO 5.

[3.5/3.8]

294001K116 ..(KA's)

ANSWER: 011 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 2.4.143, Rev. 12, "Shutdown from Outside Control Room," page 5.
2. OJT Guide O-RO-04-10: "SRO On-Shift Tasks," Task 77.

[3.6/4.2]

294001A110 ..(KA's)

ANSWER: 012 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 1.5.9.1, "Lifted Leads and Jumpers," page 6.
2. Modified question from 12/2/91 NRC Exam (replaced correct answer, reworded distractors).

[3.3/3.6]

294001K107 ..(KA's)

ANSWER: 013 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 1.3.34.2, "Limiting Condition for Operations Log," page 4.
2. PNPS Technical Specifications, Table 3.1.1.
3. UG: O-RO-04-10, "SRO On-Shift Tasks," Task 81.

[3.4/3.6]

294001A106 ..(KA's)

ANSWER: 014 (1.00)

b.

REFERENCE:

1. IG: O-RO-03-04-02, "EOP Development and Use," ELO 9.

[3.4/3.6]

294001A106 ..(KA's)

ANSWER: 015 (1.00)

a.

REFERENCE:

1. IG: O-RO-03-04-02, "EOP Development and Use," page IG-20, ELO 3n.

[4.2/4.2]

294001A102 ..(KA's)

ANSWER: 016 (1.00)

b.

REFERENCE:

1. IG: T-ER-01-01-80, "Dose Assessment/Protective Action Recommendations," ELOs 4 and 5.

[2.9/4.7]

294001A116 ..(KA's)

ANSWER: 017 (1.00)

d.

REFERENCE:

1. IG: T-ER-01-01-80, "Dose Assessment/Protective Action Recommendations," ELO 10.

[2.9/4.7]

294001A116 ..(KA's)

ANSWER: 018 (1.00)

d.

REFERENCE:

1. IG: O-RO-02-06-01, "Non-Nuclear Instrumentation and Reactor Vessel Internals," ELO 10.
2. Systems Reference Text: Nuclear Boiler Instrumentation.
3. LO Requalification IG: Reference Leg Perturbations Due to Non-Condensables.
4. Modified questions from 11/26/90 and 12/2/91 NRC Exams (combined concepts of questions and changed correct answer).
   [3.3/3.5]

216000A210 ..(KA's)

ANSWER: 019 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-07-02, "Reactor Protection System and Anticipated Transient Without Scram System," ELO 22.

   [3.8/4.2]

212000G010 ..(KA's)

ANSWER: 020 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 2.2.79, "Reactor Protection System," page 14.
2. IG: O-RO-02-07-02, "Reactor Protection System and Anticipated Transient Without Scram System," ELO 16.
3. Facility Question TYPA-13 (from Requal Retake Exam).

[3.9/4.1]

212000K412 ..(KA's)

ANSWER: 021 (1.00)

c.

REFERENCE:

1. PNPS: ARP-905R-D4, Rev. 8.
2. IG: O-RO-02-04-01, "Main Steam System," page IG-19-7/90, ELOs 18 and 211.
3. Modified question from 11/26/90 NRC Exam (changed conditions).

[4.0/4.1]

239001K127 ..(KA's)

ANSWER: 022 (1.00)

a.

REFERENCE:

1. IG: O-RO-02-08-01, "Primary Containment System," ELOs 33 and 34.
2. Modified questions from 11/26/90 and 12/02/91 NRC Exams (changed conditions, distractors, and correct answer).

[3.5/3.5]

223002A302 ..(KA's)

ANSWER: 023 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-09-04, "Reactor Core Isolation Cooling System," ELO 11.
2. Modified question from 11/26/90 NRC Exam (changed conditions, distractors and correct answer).

[3.8/3.8]

217000A215 ..(KA's)

ANSWER: 024 (1.00)

c.

REFERENCE:

1. IG: O-RO-03-04-02, "EOP Development and Use," ELO 22.

[3.9/3.8]

206000G010 ..(KA's)

ANSWER: 025 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-09-02, "Core Spray System," ELO 12.

[3.3/3.6]

209001A205 ..(KA's)

ANSWER: 026 (1.00)

c

REFERENCE:

1. IG: O-RO-02-09-01, "Low Pressure Coolant Injection and Residual Heat Removal," ELOs 9 and 11.
2. Modified questions from 11/26/90 and 12/2/91 NRC Exams (changed conditions).

[4.2/4.3]

203000A101 ..(KA's)

ANSWER: 027 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 2.2.19, "Residual Heat Removal," page 19.
2. UG: O-RO-04-04, "Emergency Tasks," Task 89c.
3. Modified question from 11/26/90 NRC Exam (changed values in distractors).

[3.2/3.4]

226001G010 ..(KA's)

ANSWER: 028 (1.00)

c.

REFERENCE:

1. PNPS Proc. 2.2.19, "Residual Heat Removal," page 29.
2. IG: O-RO-02-09-01, "Low Pressure Coolant Injection and Residual Heat Removal System," ELO 14.
3. Derived from Facility Questions 10-B and RHR/SDC-07.

[4.0/3.9]

230000A406 ..(KA's)

ANSWER: 029 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-09-05, "Automatic Depressurization System," ELOs 5 and 15.

[4.2/4.3]

218000A206 ..(KA's)

ANSWER: 030 (1.00)

d.

# 1

## IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

# 1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

# 1

## IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

REFERENCE:

1. IG: O-RO-02-09-05, "Automatic Depressurization System," ELO 10.

[3.6/3.7]

239002K405 ..(KA's)

ANSWER: 031 (1.00)

c.

REFERENCE:

1. PNPS Technical Specification 3.10.B.
2. OJT Guide No. 5, SRM System Objective 2.a.

[3.2/3.9]

215004G011 ..(KA's)

ANSWER: 032 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-07-01, "Neutron Monitoring Systems," ELOs 18, 23, and 24.

[3.5/3.7]

215003A202 ..(KA's)

ANSWER: 033 (1.00)

a.

REFERENCE:

1. IG: O-RO-02-07-01, "Neutron Monitoring Systems," ELOs 40, 42, 44, and 47.

[3.4/3.7]

215001A207 ..(KA's)

ANSWER: 034 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 2.1.15, "Daily Surveillance Log (Tech Specs and Regulatory Agencies)," Daily Log Test #22.
2. UG: O-RO-04-02, "Administrative Tasks," Task 4.
3. Modified question from 12/2/92 NRC Exam (changed conditions, distractors, and correct answer).

[3.0/3.4]

215005A107 ..(KA's)

ANSWER: 035 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-07-01, "Neutron Monitoring Systems," ELOs 63, 64, and 65.

[3.2/3.4]

215005A207 ..(KA's)

ANSWER: 036 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-07-01, "Neutron Monitoring Systems," ELO 50.
2. Facility Question PSU-24 (from Requal Retake Exam).

[3.3/3.3]

215002A202 ..(KA's)

ANSWER: 037 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-06-04, "Rod Position Information System," ELO 11.d.

[3.1/3.3]

214000A201 ..(KA's)

ANSWER: 038 (1.00)

a or d

REFERENCE:

1. IG: O-RO-02-06-03, "Control Rod Drive System," ELO 29.

[3.5/3.5]

201006K403 ..(KA's)

ANSWER: 039 (1.00)

d.

REFERENCE:

1. IG: O-RO-02-04-02, "Condensate and Feedwater System," ELO 29.

[3.3/3.4]

259001A204 ..(KA's)

ANSWER: 040 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-04-02, "Condensate and Feedwater System," ELO 15.

[3.6/3.7]

256000K304 ..(KA's)

ANSWER: 041 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-06-02, "Condensate and Feedwater System," ELO 83.
2. Modified question from 11/26/90 NRC Exam (modified distractors and correct answer).

[3.1/3.1]

259002K604 ..(KA's)

ANSWER: 042 (1.00)

d.

REFERENCE:

1. IG: O-RO-02-05-01, "Main Turbine System," ELOs 70, 79, and 80.

[4.1/4.1]

241000K306 ..(KA's)

ANSWER: 043 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-01-03, "Main Generator," ELO 19.
2. IG: O-RO-02-05-01, "Main Turbine System," page IG-34-5/89.
3. Modified questions from 11/26/90 NRC Exam (combined 2 questions).

[2.7/2.8]

245000A304 ..(KA's)

ANSWER: 044 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-04-03, "Main Condenser Vacuum and Augmented Off Gas Systems," ELOs 2, 9i, 18d, and 18e.

[3.5/3.9]

271000A206 ..(KA's)

ANSWER: 045 (1.00)

d.

REFERENCE:

1. IG: O-RO-02-06-02, "Recirculation System," ELOs 9, 14, 15, 18, and 20.

[3.3/3.4]

202001K410 ..(KA's)

ANSWER: 046 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-06-02, "Recirculation System," ELO 33.

[3.5/3.5]

202002K604 ..(KA's)

ANSWER: 047 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-09-06, "Diesel Generator System," ELO 24.

[3.8/3.7]

264000K408 ..(KA's)

deleted

ANSWER: 048 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-08-03, "Standby Gas Treatment System," ELOs 9, 13, and 15e.

[2.9/3.2]

261000A203 ..(KA's)

ANSWER: 049 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-08-05, "Plant Ventilation Systems," ELO 4.
2. Modified question from 11/26/90 NRC Exam (changed correct answer and conditions on all distractors).

[3.6/3.9]

272000K403 ..(KA's)

ANSWER: 050 (1.00)

a.

REFERENCE:

1. IG: O-RO-02-08-05, "Plant Ventilation Systems," ELOs 9, 10, and 11.
2. Expanded Previous Question from 12/02/91 NRC Exam.

[3.3/3.4]

290001G010 ..(KA's)

ANSWER: 051 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-08-05, "Plant Ventilation Systems," ELOs 5 and 9.
2. Modified question from 11/26/90 NRC Exam (added concept and changed answers).

[3.1/3.3]

290003A204 ..(KA's)

ANSWER: 052 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-09-01, "Low Pressure Coolant Injection and Residual Heat Removal System," ELO 15.
2. IG: O-NL-03-11-01, "Fuel Pool Cooling," ELO 8.

[2.9/3.0]

233000K102 ..(KA's)

ANSWER: 053 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-08-06, "Fuel Handling Equipment," ELO 8.

[3.3/4.1]

234000K402 ..(KA's)

ANSWER: 054 (1.00)

a.

REFERENCE:

1. IG: O-RO-02-10-01, "Fire Protection System," ELOs 15, 16, 17, and 18.

[3.8/3.9]

286000G004 ..(KA's)

ANSWER: 055 (1.00) deleted

c.

REFERENCE:

1. IG: O-RO-02-09-03, "High Pressure Coolant Injection," page 33.
2. PNPS Technical Specifications, section 3.5.C.
3. IG: O-RO-06-01-03, "Limiting Conditions for Operations," ELO 3.

[3.6/4.3]

206000G005 ..(KA's)

deleted

ANSWER: 056 (1.00)

b.

REFERENCE:

1. PNPS Technical Specifications, Section 3.4.
2. IG: O-RO-06-01-03, "Limiting Conditions for Operations," ELO 3.

[3.6/4.4]

211000G005 ..(KA's)

ANSWER: 057 (1.00)

d.

REFERENCE:

1. PNPS Technical Specifications, Sections 3.9 and 3.5.F.
2. IG: O-RO-06-01-03, "Limiting Conditions for Operations," ELO 3.

[2.9/3.9]

262001G005 ..(KA's)

ANSWER: 058 (1.00)

c.

REFERENCE:

1. PNPS Technical Specifications, sections 2.0 and 3.11.
2. PNPS Proc. No. 1.3.6, "Technical Specification - Adherence and Clarification," page 6.
3. OJT Guide No. 13, Process Computer System Objecti\ 2a.

[3.5/4.3]

295014G003 ..(KA's)

ANSWER: 059 (1.00)

c.

REFERENCE:

1. PNPS Technical Specification 3.5.B.
2. IG: O-RO-06-01-03, "Limiting Conditions for Operations," ELOs 3 and 4.

[3.4/4.1]

295018G008 ..(KA's)

ANSWER: 060 (1.00)

a.

REFERENCE:

1. PNPS Technical Specification 3.7.B.
2. IG: O-RO-06-01-03, "Limiting Conditions for Operations," ELO 3.

[2.9/3.8]

295023G003 ..(KA's)

ANSWER: 061 (1.00)

a.

REFERENCE:

1. PNPS Proc. No. 2.4.29, "Stuck Open Safety/Relief Vavle," page 4.
2. OJT Guide No. 9, ADS System Objective 2b.

[2.9/4.1]

295026K305 ..(KA's)

ANSWER: 062 (1.00)

b.

REFERENCE:

1. EP-IP-100, "Emergency Classification," sections 4.1 and 4.3.
2. IG: T-ER-01-01-30, "Classifications," ELO 5.

[3.1/4.5]

295033G002 ..(KA's)

ANSWER: 063 (1.00)

c.

REFERENCE:

1. EP-IP-100, "Emergency Classification," sections 6.2, 7.1, and 7.2.
2. IG: T-ER-01-01-30, "Classifications," ELO 5.
[3.1/4.5]

295016G002 ..(KA's)

ANSWER: 064 (1.00)

c.

REFERENCE:

1. EP-IP-100, "Emergency Classification," sections 2.2, 2.3, 3.2, and 3.3.
2. IG: T-ER-01-01-30, "Classifications," ELO 5.

[2.9/4.2]

295029G002 ..(KA's)

ANSWER: 065 (1.00)

d.

REFERENCE:

1. PNPS Technical Specifications, sections 1.0.N and 3.7.C.
2. IG: O-RO-06-01-01, "Technical Specification Definitions," ELO 3.
3. IG: O-RO-04-09, "Technical Specification Overview," ELO 3.
   [3.9/4.2]

295035K101 ..(KA's)

ANSWER: 066 (1.00)

c.

REFERENCE:

1. PNPS Technical Specifications, section 2.1.
2. IG: O-RO-06-01-02, "Safety Limits and Limiting Safety System Settings," ELO 1.

[3.5/4.3]

295025G003 ..(KA's)

ANSWER: 067 (1.00)

a.

REFERENCE:

1. PNPS Technical Specifications, 3.6.E and 4.6.E Bases, page 147a.
2. OJT Guide No. 4, Recirculation System Objective 2b.

[3.9/4.2]

295001G011 ..(KA's)

ANSWER: 068 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 2.4.17, "Recirculation Pump(s) Trip," page 4.
2. OJT Guide 4, Recirculation System Objective 2b.

[3.5/3.8]

295001A201 ..(KA's)

ANSWER: 069 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 5.3.8, Rev. 16, "Loss of Instrument Air," page 2.
2. IG: O-RO-02-02-04, "Instrument and High Pressure Air," ELO 6.
3. Modified question from 12/2/91 NRC Exam (changed conditions).
[3.6/3.7]

295019A202 ..(KA's)

ANSWER: 070 (1.00)

d.

REFERENCE:

1. IG: O-RO-02-04-01, "Main Steam System," ELO .4.

[3.6/3.7]

295020K201 ..(KA's)

ANSWER: 071 (1.00)

b.

REFERENCE:

1. IG: O-RO-0208-01, "Primary Containment System," ELO 36.

[3.6/3.6]

295020A101 ..(KA's)

ANSWER: 072 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 2.4.42, "Loss of RBCCW," pages 4 and 6.
2. IG: O-RO-02-02-06, "Reactor Building Closed Cooling Water," ELOs 2, 3, 8, and 11.
3. OJT Guide No. 2, RBCCW System Objective 2b.

[3.3/3.4]

295018K201 ..(KA's)

ANSWER: 073 (1.00)

a.

REFERENCE:

1. IG: O-RO-02-02-02, "Salt Service Water System," ELO 5.

[4.4/4.4]

295003A103 ..(KA's)

ANSWER: 074 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 1.3.4, "Procedures," page 17.
2. PNPS Proc. No. 2.4.36, "Decreasing Condenser Vacuum," page 2.
3. IG: O-RO-03-04-02, "EOP Development and Use," ELOs 13 and 14.
4. IG: O-RO-02-07-02, "Reactor Protection System and Anticipated Transient Without Scram System," ELO 14.
5. IG: O-RO-02-05-01, "Main Turbine System," ELO 21.
6. Modified questions from 11/26/90 and 12/2/91 NRC Exams (changed conditions).

[3.8/3.7]

295002G010 ..(KA's)

ANSWER: C75 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 2.4.150, "Loss of Feedwater Heating," page 3.
2. OJT Guide No. 8, Feedwater Heating System Objective 2a.

[4.0/3.9]

295014G010 ..(KA's)

ANSWER: 076 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 2.4.4, "Loss of CRD Pumps."
2. PNPS Proc. No. 2.4.11, "Control Rod Positioning Malfunctions."
3. PNPS Proc. No. 2.4.11.1, "CRD System Malfunctions."
4. OJT Guide No. 3, CRDM System Objective 2a and CRD Hydraulic System Objective 2b.

[3.7/3.5]

295022G010 ..(KA's)

ANSWER: 077 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 5.4.3, Rev. 10, Refueling Floor High Radiation," page 2.
2. IG: O-RO-06-04-01, "Fuel Handling Operations and Supervision," ELO 9.

[3.8/3.9]

295023G010 ..(KA's)

ANSWER: 078 (1.00)

b.

REFERENCE:

1. PNPS Proc. No. 5.3.6, "Loss of Vital AC (Y-2)."
2. PNPS Proc. No. 5.3.30, "Loss of 250V DC Bus D-10."
3. OJT Guide No. 1, 120/240 VAC System Objective 2b and 250 VDC System Objective 4.

[3.5/3.5]

295004G005 ..(KA's)

ANSWER: 079 (1.00)

b

REFERENCE:

1. PNPS Proc. No. 5.3.31, "Station Blackout," page 2.
2. PNPS Proc. No. 5.3.26, "RPV Injection During Emergencies," page 2.
3. OJT Guide No. 1, 480/208 VAC System Objective 2b.

[4.4/4.4]

295003A103 ..(KA's)

ANSWER: 080 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 2.4.143, Rev. 12, "Shutdown from Outside Control Room," page 13.
2. OJT Guide O-RO-04-04, "Emergency Tasks," Task 62.

[4.1/4.1]

295016G006 ..(KA's)

ANSWER: 081 (1.00)

d.

REFERENCE:

1. UG: O-RO-04-04, "Emergency Tasks," Tasks 1, 2, and 3. (Derived from facility question PRO-1 from proposed requal retake exam.)

[4.3/4.4]

295006A202 ..(KA's)

ANSWER: 082 (1.00)

c.

REFERENCE:

1. IG: O-RO-02-07-01, "Neutron Monitoring Systems," ELOs 14 and 16.
2. IG: O-RO-03-04-04, "EOP-02, Failure to Scram," ELO 16.

[4.2/4.3]

295037A201 ..(KA's)

ANSWER: 083 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 5.3.23, Rev. 9, "Alternate Rod Insertion," page 5.
2. OJTPG: O-RO-04-04, "Emergency Tasks," Task 4.

[4.0/4.1]

295015K204 ..(KA's)

ANSWER: 084 (1.00)

d.

REFERENCE:

1. PNPS Proc. No. 5.3.23, Rev. 9, "Alternate Rod Insertion," pages 6 - 10.
2. OJTPG: O-RO-04-04, "Emergency Tasks," Task 4.
3. Modified question from 12/2/91 NRC Exam (modified distractors).

[3.8/3.9]

295015A101 ..(KA's)

ANSWER: 085 (1.00)

d.

REFERENCE:

1. IG: O-RO-03-04-05, Primary Containment Control,", page IG-5, ELO 14.
2. OJTPG: O-RO-04-04, "Emergency Tasks," Task 89c.
3. Modified question from 12/2/91 NRC Exam (replaced 1 distractor). [3.6/4.3]

295012G012 ..(KA's)

ANSWER: 086 (1.00)

a.

REFERENCE:

1. IG: O-RO-03-04-05, Primary Containment Control,", page IG-11.
2. OJTPG: O-RO-04-04, "Emergency Tasks," Task 89c.

[3.8/4.5]

295026G012 ..(KA's)

ANSWER: 087 (1.00)

b.

REFERENCE:

1. IG: O-RO-02-08-01, "Primary Containment System," ELO 39A.

[3.6/3.9]

295024G007 ..(KA's)

ANSWER: 088 (1.00)

c.

REFERENCE:

1. PNPS Proc. No. 5.3.27, "Determining Primary Containment Water Level."
2. OJT Guide No. 10, "Primary Containment System Structure Objective 2a.
3. IG: O-RO-02-08-01, "Primary Containment System," ELOs 13 and 17.

[3.4/3.5]

295029A203 ..(KA's)

ANSWER: 089 (1.00)

c

REFERENCE:

1. PNPS Proc. No. 5.3.26, "RPV Injection During Emergencies."
2. Facility Question 5.3.26-1 (Proposed Requal Retake Exam).

[4.1/3.9]

295031G006 ..(KA's)

ANSWER: 090 (1.00)

a.

REFERENCE:

1. IG: O-RO-03-04-02, "EOP Development and Use," ELO 21.
2. Modified question from 12/2/91 NRC Exam (changed conditions).

[3.8/4.3]

295028G012 ..(KA's)

ANSWER: 091 (1.00)

b.

REFERENCE:

1. IG: O-RO-03-04-02, "EOP-02, Failure to Scram," page IG-15, ELO 12.

[4.1/4.3]

295037K101 ..(KA's)

ANSWER: 092 (1.00)

d.

REFERENCE:

1. IG: O-RO-03-04-08, "EOP-16 and EOP-26, RPV Flooding," ELOs 2 and 10.

[3.9/4.6]

295037G012 ..(KA's)

ANSWER: 093 (1.00)

d.

REFERENCE:

1. IG: O-RO-03-04-04, "EOP-02, Failure to Scram," pages IG-33 and IG-41, ELO 22f.
2. IG: O-RO-03-04-07, "EOP-05, Radioactivity Release Control," ELO 6.

[3.9/4.6]

295037G012 ..(KA's)

ANSWER: 094 (1.00)

b.

REFERENCE:

1. IG: O-RO-03-04-05, "EOP-03, Primary Containment Control," page IG-11.
2. IG: O-RO-03-04-03, "EOP-01, RPV Control," ELO 1a.
3. Modified question from 12/2/91 NRC Exam (changed correct answer and modified distractors).

[3.9/4.1]

295026K305 ..(KA's)

ANSWER: 095 (1.00)

a.

REFERENCE:

1. EOP-03, "Primary Containment Control."
2. IG: O-RO-03-04-05, "Primary Containment Control," ELO 18.
3. Modified question from 12/2/91 NRC Exam (modified to require use of EOP flowchart).

[4.6/4.7]

295031K101 ..(KA's)

ANSWER: 096 (1.00)

b.

REFERENCE:

1. EOP-03, "Primary Containment Control."
2. IG: O-RO-03-04-05, "EOP-03, Primary Containment Control," ELOs 16 and 18.

[3.7/4.4]

295030G012 ..(KA's)

ANSWER: 097 (1.00)

d.

REFERENCE:

1. IG: O-RO-03-04-05, "EOP-03, Primary Containment Control," ELO 17.

[3.9/4.5]

295024G012 ..(KA's)

ANSWER: 098 (1.00)

a.

REFERENCE:

1. IG: O-RO-03-04-05, "EOP-03, Primary Containment Control," ELO 18.
2. IG: O-RO-03-04-07, "EOP-05, Radioactivity Release Control," ELOs 3 and 6.
3. LP-IP-100, "Emergency Classification," Attachment 1, section 5.0.

[3.9/4.5]

295038G012 ..(KA's)

ANSWER: 099 (1.00)

b. or c

REFERENCE:

1. EOP-01, "RPV Control."
2. EOP-16, "RPV Flooding."
3. IG: O-RO-03-04-03, "EOP-01, RPV Control," ELO 2f.
4. IG: O-RO-03-04-08, "EOP-16 and EOP-26, RPV Flooding," ELO 8.

[3.9/4.5]

295031G012 ..(KA's)

ANSWER: 100 (1.00)

a.

REFERENCE:

1. EOP-04, "Secondary Containment Control."
2. IG: O-RO-03-04-06, "EOP-04, Secondary Containment Control," ELO 10.

[3.6/4.4]

295032G012 ..(KA's)

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

# A N S W E R K E Y

MULTIPLE CHOICE

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

SRO Exam BWR Reactor
Organized by Question Number

| QUESTION | VALUE | REFERENCE | |
|---|---|---|---|
| ~~001~~ | ~~1.00~~ | ~~9000100~~ | deleted |
| 002 | 1.00 | 9000101 | |
| 003 | 1.00 | 9000102 | |
| 004 | 1.00 | 9000103 | |
| 005 | 1.00 | 9000104 | |
| 006 | 1.00 | 9000105 | |
| 007 | 1.00 | 9000106 | |
| 008 | 1.00 | 9000107 | |
| 009 | 1.00 | 9000113 | |
| 010 | 1.00 | 9000114 | |
| 011 | 1.00 | 9000115 | |
| 012 | 1.00 | 9000116 | |
| 013 | 1.00 | 9000117 | |
| 014 | 1.00 | 9000118 | |
| 015 | 1.00 | 9000119 | |
| 016 | 1.00 | 9000120 | |
| 017 | 1.00 | 9000121 | |
| 018 | 1.00 | 9000126 | |
| 019 | 1.00 | 9000127 | |
| 020 | 1.00 | 9000128 | |
| 021 | 1.00 | 9000129 | |
| 022 | 1.00 | 9000130 | |
| 023 | 1.00 | 9000131 | |
| 024 | 1.00 | 9000132 | |
| 025 | 1.00 | 9000133 | |
| 026 | 1.00 | 9000134 | |
| 027 | 1.00 | 9000135 | |
| 028 | 1.00 | 9000136 | |
| 029 | 1.00 | 9000137 | |
| 030 | 1.00 | 9000138 | |
| 031 | 1.00 | 9000139 | |
| 032 | 1.00 | 9000141 | |
| 033 | 1.00 | 9000142 | |
| 034 | 1.00 | 9000143 | |
| 035 | 1.00 | 9000144 | |
| 036 | 1.00 | 9000145 | |
| 037 | 1.00 | 9000146 | |
| 038 | 1.00 | 9000147 | |
| 039 | 1.00 | 9000148 | |
| 040 | 1.00 | 9000149 | |
| 041 | 1.00 | 9000150 | |
| 042 | 1.00 | 9000151 | |
| 043 | 1.00 | 9000152 | |
| 044 | 1.00 | 9000153 | |
| 045 | 1.00 | 9000154 | |
| 046 | 1.00 | 9000155 | |
| 047 | 1.00 | 9000156 | |
| ~~048~~ | ~~1.00~~ | ~~9000157~~ | deleted |
| 049 | 1.00 | 9000158 | |

S R O Exam B W R Reactor
Organized by Question Number

| QUESTION | VALUE | REFERENCE |
|---|---|---|
| 050 | 1.00 | 9000159 |
| 051 | 1.00 | 9000160 |
| 052 | 1.00 | 9000161 |
| 053 | 1.00 | 9000162 |
| 054 | 1.00 | 9000163 |
| ~~055~~ | ~~1.00~~ | ~~9000174~~ deleted |
| 056 | 1.0C | 9000175 |
| 057 | 1.00 | 9000176 |
| 058 | 1.00 | 9000177 |
| 059 | 1.00 | 9000178 |
| 060 | 1.00 | 9000179 |
| 061 | 1.00 | 9000180 |
| 062 | 1.00 | 9000181 |
| 063 | 1.00 | 9000182 |
| 064 | 1.00 | 9000183 |
| 065 | 1.00 | 9000184 |
| 066 | 1.00 | 9000185 |
| 067 | 1.00 | 9000186 |
| 068 | 1.00 | 9000187 |
| 069 | 1.00 | 9000188 |
| 070 | 1.00 | 9000189 |
| 071 | 1.00 | 9000190 |
| 072 | 1.00 | 9000191 |
| 073 | 1.00 | 9000192 |
| 074 | 1.00 | 9000193 |
| 075 | 1.00 | 9000194 |
| 076 | 1.00 | 9000195 |
| 077 | 1.00 | 9000196 |
| 078 | 1.00 | 9000197 |
| 079 | 1.00 | 9000198 |
| 080 | 1.00 | 9000199 |
| 081 | 1.00 | 9000200 |
| 082 | 1.00 | 9000201 |
| 083 | 1.00 | 9000202 |
| 084 | 1.00 | 9000203 |
| 085 | 1.00 | 9000204 |
| 086 | 1.00 | 9000205 |
| 087 | 1.00 | 9000206 |
| 088 | 1.00 | 9000207 |
| 089 | 1.00 | 9000208 |
| 090 | 1.00 | 9000220 |
| 091 | 1.00 | 9000221 |
| 092 | 1.00 | 9000222 |
| 093 | 1.00 | 9000223 |
| 094 | 1.00 | 9000224 |
| 095 | 1.00 | 9000225 |
| 096 | 1.00 | 9000226 |
| 097 | 1.00 | 9000227 |
| 098 | 1.00 | 9000228 |

SRO Exam BWR Reactor
Organized by Question Number

| QUESTION | VALUE | REFERENCE |
|---|---|---|
| 099 | 1.00 | 9000229 |
| 100 | 1.00 | 9000230 |
| | ------ | |
| | ~~100.00~~ 97.00 | |
| | ------ | |
| | ------ | |
| | ~~100.00~~ 97.00 | |

S R O  E x a m  B W R  R e a c t o r
O r g a n i z e d  b y  K A  G r o u p

WIDE GENERICS

| QUESTION | VALUE | KA |
|---|---|---|
| 015 | 1.00 | 294001A102 |
| 008 | 1.00 | 294001A103 |
| 013 | 1.00 | 294001A106 |
| 014 | 1.00 | 294001A106 |
| 011 | 1.00 | 294001A110 |
| ~~001~~ | ~~1.00~~ | ~~294001A115~~ deleted |
| 016 | 1.00 | 294001A116 |
| 017 | 1.00 | 294001A116 |
| 006 | 1.00 | 294001K101 |
| 007 | 1.00 | 294001K102 |
| 003 | 1.00 | 294001K103 |
| 002 | 1.00 | 294001K103 |
| 012 | 1.00 | 294001K107 |
| 005 | 1.00 | 294001K109 |
| 004 | 1.00 | 294001K110 |
| 009 | 1.00 | 294001K113 |
| 010 | 1.00 | 294001K116 |

PWG Total ~~17.00~~ 16.00

PLANT SYSTEMS

Group I

| QUESTION | VALUE | KA |
|---|---|---|
| 046 | 1.00 | 202002K604 |
| 026 | 1.00 | 203000A101 |
| ~~055~~ | ~~1.00~~ | ~~206000G005~~ deleted |
| 024 | 1.00 | 206000G010 |
| 025 | 1.00 | 209001A205 |
| 056 | 1.00 | 211000G005 |
| 019 | 1.00 | 212000G010 |
| 020 | 1.00 | 212000K412 |
| 031 | 1.00 | 215004G011 |
| 034 | 1.00 | 215005A107 |
| 035 | 1.00 | 215005A207 |
| 018 | 1.00 | 216000A210 |
| 023 | 1.00 | 217000A215 |
| 029 | 1.00 | 218000A206 |
| 022 | 1.00 | 223002A302 |
| 027 | 1.00 | 226001G010 |
| 030 | 1.00 | 239002K405 |
| 042 | 1.00 | 241000K306 |
| 041 | 1.00 | 259002K604 |
| ~~049~~ | ~~1.00~~ | ~~261000A203~~ deleted |

SRO Exam BWR Reactor
Organized by KA Group

## PLANT SYSTEMS

### Group I

| QUESTION | VALUE | KA |
|---|---|---|
| 057 | 1.00 | 262001G005 |
| 047 | 1.00 | 264000K408 |
| 050 | 1.00 | 290001G010 |
| PS-I Total | ~~23.00~~ 21.00 | |

### Group II

| QUESTION | VALUE | KA |
|---|---|---|
| 038 | 1.00 | 201006K403 |
| 045 | 1.00 | 202001K410 |
| 037 | 1.00 | 214000A201 |
| 036 | 1.00 | 215002A202 |
| 032 | 1.00 | 215003A202 |
| 028 | 1.00 | 230000A406 |
| 053 | 1.00 | 234000K402 |
| 043 | 1.00 | 245000A304 |
| 039 | 1.00 | 259001A204 |
| 044 | 1.00 | 271000A206 |
| 049 | 1.00 | 272000K403 |
| 054 | 1.00 | 286000G004 |
| 051 | 1.00 | 290003A204 |
| PS-II Total | 13.00 | |

### Group III

| QUESTION | VALUE | KA |
|---|---|---|
| 033 | 1.00 | 215001A207 |
| 052 | 1.00 | 233000K102 |
| 021 | 1.00 | 239001K127 |
| 040 | 1.00 | 256000K304 |
| PS-III Total | 4.00 | |

PS Total ~~40.00~~ 38.00

## EMERGENCY PLANT EVOLUTIONS

### Group I

SRO Exam BWR Reactor
Organized by KA Group

## EMERGENCY PLANT EVOLUTIONS

### Group I

| QUESTION | VALUE | KA |
|---|---|---|
| 079 | 1.00 | 295003A103 |
| 073 | 1.00 | 295003A103 |
| 081 | 1.00 | 295006A202 |
| 058 | 1.00 | 295014G003 |
| 075 | 1.00 | 295014G010 |
| 084 | 1.00 | 295015A101 |
| 083 | 1.00 | 295015K204 |
| 063 | 1.00 | 295016G002 |
| 080 | 1.00 | 295016G006 |
| 060 | 1.00 | 295023G003 |
| 077 | 1.00 | 295023G010 |
| 087 | 1.00 | 295024G007 |
| 097 | 1.00 | 295024G012 |
| 066 | 1.00 | 295025G003 |
| 086 | 1.00 | 295026G012 |
| 061 | 1.00 | 295026K305 |
| 094 | 1.00 | 295026K305 |
| 096 | 1.00 | 295030G012 |
| 089 | 1.00 | 295031G006 |
| 099 | 1.00 | 295031G012 |
| 095 | 1.00 | 295031K101 |
| 082 | 1.00 | 295037A201 |
| 092 | 1.00 | 295037G012 |
| 093 | 1.00 | 295037G012 |
| 091 | 1.00 | 295037K101 |
| 098 | 1.00 | 295038G012 |
| EPE-I Total | 26.00 | |

### Group II

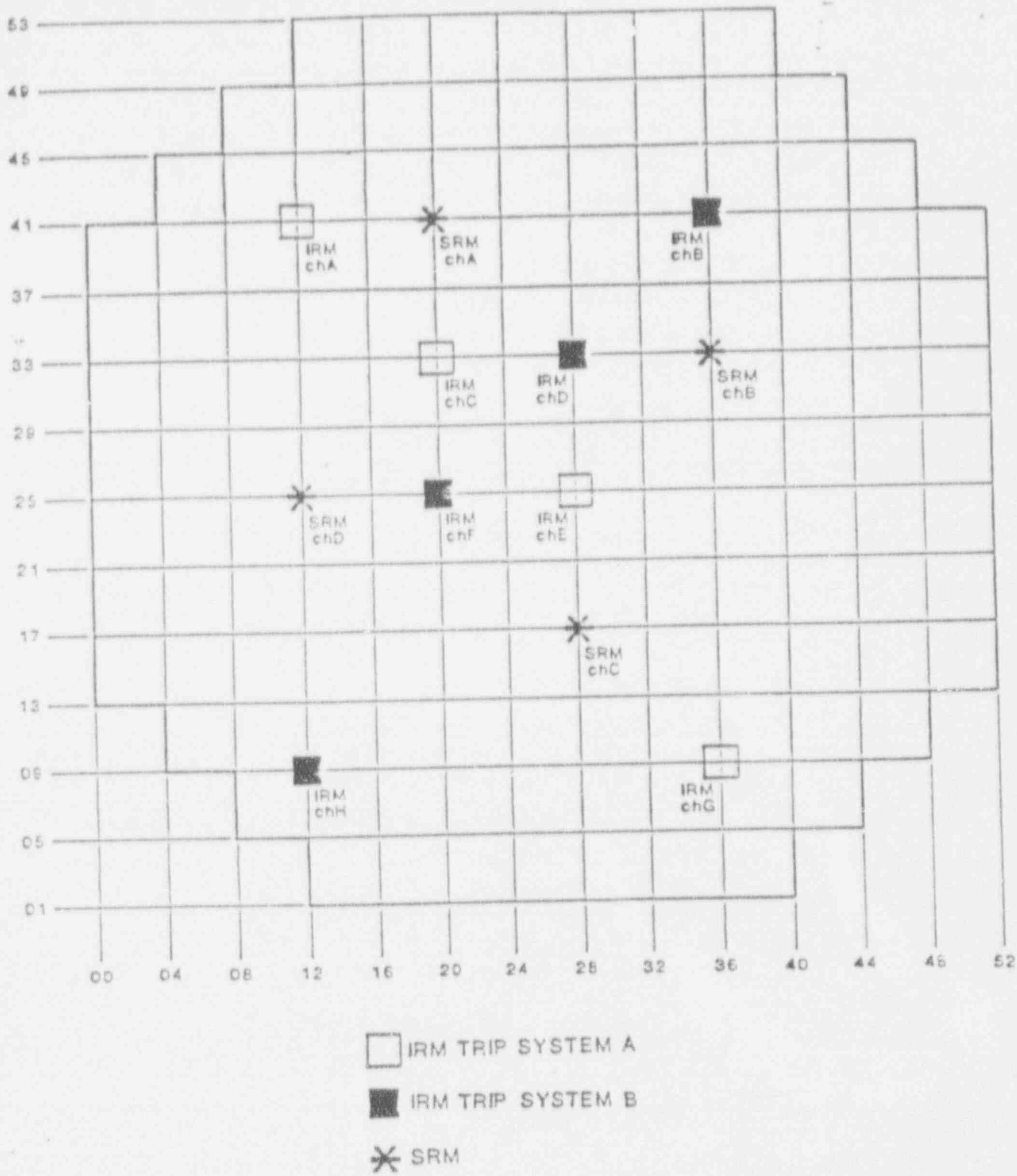
| QUESTION | VALUE | KA |
|---|---|---|
| 068 | 1.00 | 295001A201 |
| 067 | 1.00 | 295001G011 |
| 074 | 1.00 | 295002G010 |
| 078 | 1.00 | 295004G005 |
| 085 | 1.00 | 295012G012 |
| 059 | 1.00 | 295018G008 |
| 072 | 1.00 | 295018K201 |
| 069 | 1.00 | 295019A202 |
| 071 | 1.00 | 295020A101 |
| 070 | 1.00 | 295020K201 |
| 076 | 1.00 | 295022G010 |
| 090 | 1.00 | 295028G012 |

S R O Exam BWR Reactor
Organized by KA Group

EMERGENCY PLANT EVOLUTIONS

Group II

| QUESTION | VALUE | KA |
|---|---|---|
| 088 | 1.00 | 295029A203 |
| 064 | 1.00 | 295029G002 |
| 100 | 1.00 | 295032G012 |
| 062 | 1.00 | 295033G002 |
| 065 | 1.00 | 295035K101 |
| | ------ | |
| EPE-II Total | 17.00 | |
| | ------ | |
| | ------ | |
| EPE Total | 43.00 | |
| | ------ | |
| | ------ | |
| | ------ | |
| Test Total | ~~100.00~~ 97.00 | |




CORE MAP

FIGURE 4 REV. 1

# CONTROL ROD SEQUENCE SHEET
Figure 2

ATTACHMENT 3
Sheet 1 of 1

CONTROL ROD SEQ. A-1
V. NO. ____________
TE ____________

CHANGES

| RWM Step No. | Rod Grp No. | Rod Number | Mov Frm | Mov To | RWM Step No. | Rod Grp No. | Rod Numbr | Mov Frm | Mov To | Apr By | ROD MOVEMENT (✓) Withdraw | Verify | Insert | Verify | Withdraw | Verify | Insert | Verify | FLUX RESP (✓) | COUPLG CK (✓) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 26 | 7 | 22-47 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 46-31 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 30-07 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 06-23 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 06-31 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 30-47 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 46-23 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 22-07 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | |
| 27 | 8 | 14-39 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 38-39 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 38-15 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 14-15 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | |
| 28 | 9 | 22-31 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 30-31 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 30-23 | 08 | 12 | | | | | | | | | | | | | | | | |
| | | 22-23 | 08 | 12 | | | | | | | | | | | | | | | | |
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Flux Response and/or Coupling Check performed if applicable
(Surveillance Requirement 4.3 B.1.a/4.3.B.1.b)

____________________ NUCLEAR OPERATIONS SUPERVISOR

____________ DATE

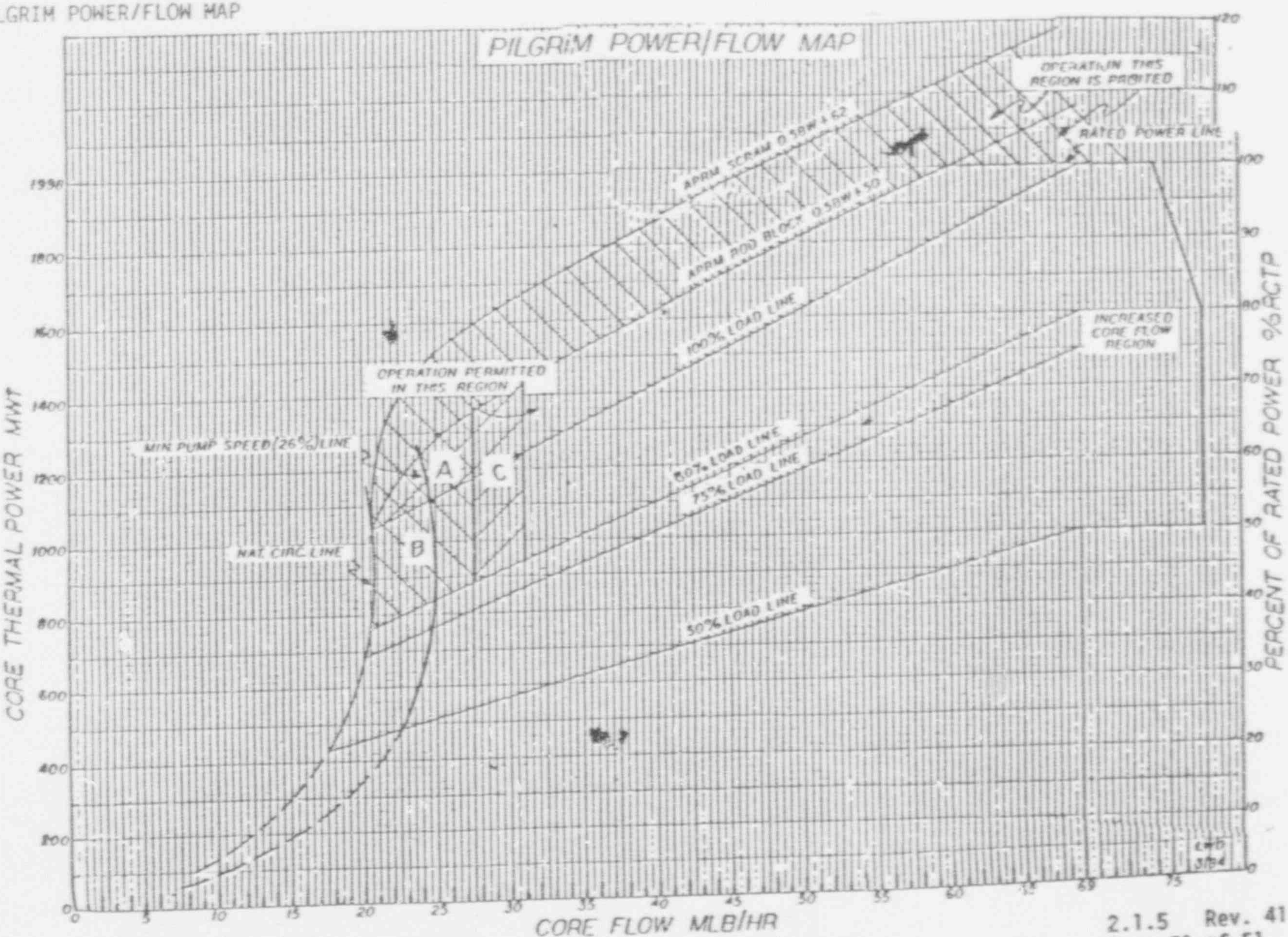
17
PAGE NO.

11

Form RE.12

SECTION G (Continued)

13. PILGRIM POWER/FLOW MAP



BOSTON EDISON

PILGRIM NUCLEAR POWER STATION

Procedure No. 2.4.17

RECIRCULATION PUMP(S) TRIP

**INFORMATION ONLY**

Use restricted to reference

REVIEWERS AND APPROVERS

| | |
|---|---|
| Eric W. Olson | 2/28/92 |
| Procedure Writer | Date |
| [illegible] | 3/2/92 |
| Technical Reviewer | Date |
| Eric W. Olson | 3/2/92 |
| Validator | Date |
| Thomas A. [illegible] LR POS | 3/2/92 |
| Procedure Owner | Date |
| N/A | |
| QAD Manager | Date |
| J A Seery | 3/4/92 |
| ORC Chairman | Date |
| [illegible] | 3/5/92 |
| Plant Manager | Date |

SAFETY REVIEW REQUIRED

ORC REVIEW REQUIRED

Effective Date: 3-6-92

## 1.0 SYMPTOMS

### 1.1 ALARMS

| | ANNUNCIATOR | PANEL | WINDOW |
|---|---|---|---|
| [1] | RECIRC. MG SET GENERATOR DIFF. OVERCURRENT | 904C<br>904R | G4<br>A3 |
| [2] | RECIRC. MG SET DRIVE MOTOR TRIP | 904C<br>904R | B4<br>E2 |
| [3] | RECIRC. MG SET DRIVE MOTOR OVERLOAD | 904C<br>904R | C4<br>F2 |
| [4] | RECIRC. MG SET GENERATOR LOCKOUT | 904C<br>904R | E4<br>H2 |
| [5] | RECIRC. PUMP LOCKED ROTOR TRIP | 904C<br>904R | A4<br>D2 |
| [6] | RECIRC. MG SET LUBE OIL LOW PRESS | 904C<br>904R | B3<br>E1 |
| [7] | RECIRC. MG SET FLUID DRIVE HI OIL TEMP. | 904C<br>904R | D3<br>G1 |

### 1.2 PLANT INDICATIONS

[1] Reduction in recirculation flow.

[2] Sudden decrease in reactor power.

## 2.0 AUTOMATIC ACTIONS

None

## 3.0 IMMEDIATE OPERATOR ACTIONS

[1] MONITOR alarms and instrumentation AND DETERMINE the type of system malfunction that has occurred.

[2] IF both recirculation pumps trip, THEN MANUALLY SCRAM the reactor AND CONCURRENTLY PERFORM PNPS 2.1.6, "Reactor Scram", with this Procedure.

## 4.0 SUBSEQUENT OPERATOR ACTIONS

[1] TURN to the page of this Procedure for the malfunction that has occurred AND PERFORM the indicated steps:

| | Page | Section |
|---|---|---|
| (a) Trip of one recirculation pump | 3 | 4.1 |
| (b) Trip of both recirculation pumps | 6 | 4.2 |

### 4.1 TRIP OF ONE RECIRCULATION PUMP

---

NOTE

The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the Plant shall be placed in a Hot Shutdown condition within 24 hours unless the loop is sooner returned to service.

---

*************************************************************************************

CAUTION

If power level is less than 30%, stratification may occur; refer to PNPS 2.4.24, "Reactor Vessel Cold Water Stratification".

*************************************************************************************

[1] CLOSE affected MO-202-5A or B, PUMP DISCH VLV.

(a) WHEN 5 minutes have elapsed, THEN REOPEN the discharge valve.

[2] CHECK speed on the in-service pump to ensure it has not increased.

[3] DETERMINE Total Core Flow (TCF) [NRC Inspection Report 91-25]

(a) DETERMINE direction of flow through idle jet pumps.

(1) ADD in-service and idle jet pump loop flow rate:

| ________ | + | ________ | = | ________ |
|---|---|---|---|---|
| In-Service FI-263-107A(B) | | Idle FI-263-107A(B) | | Summed Value |

(2) MULTIPLY idle jet pump loop flow by 0.95 AND SUBTRACT this multiple from in-service jet pump loop flow:

| ________ | - [0.95 X | ________ ] | = | ________ |
|---|---|---|---|---|
| In-Service FI-263-107A(B) | | Idle FI-263-107A(B) | | Subtracted Value |

(3) USE current reactor power AND PLOT both of the calculated flow values on the Power-To-Flow Map.

---

NOTE

The change in reactor power due to the recirc pump trip will result in a Xenon transient. This transient will cause the previous load line to lower. The amount the load line shifts is dependent on the time after the recirc pump trip and previous equilibrium reactor power conditions.

---

(4) IF the Subtracted Value falls to the left of the expected load line (i.e., on or to the left of the minimum pump speed line) AND the Summed Value falls approximately on the expected Load Line, THEN Forward Flow exists through the idle jet pump loop.

(5) IF the Subtracted Value falls approximately on the expected load line AND the Summed Value falls below and to the right, THEN Reverse Flow exists through the idle jet pump loop.

(b) CALCULATE Total Core Flow (TCF)

(1) IF Forward Flow through the idle jet pump loop exists, THEN Total Core Flow equals the Summed Value.

(2) IF Reverse Flow through the idle jet pump loop exists, THEN Total Core Flow equals the Subtracted Value.

[4] IF the reactor is operating AT OR ABOVE the 80% load line AND total core flow decreases below 31.5 Mlb/hr, THEN PERFORM the following steps: [IEB 88-07, SUPP 1: BWROG 8879]

(a) MONITOR the APRMs and LPRMs for neutron flux instability oscillations.

(b) INCREASE speed of the operating recirculation pump until any flux instability ceases and core flow is greater than 31.5 Mlb/hr.

(c) INSERT control rods to decrease reactor power below the 80% load line.

(d) IF APRM oscillations of greater than 10% peak-to-peak OR periodic LPRM upscale or downscale alarms are observed, THEN SCRAM the reactor AND CONCURRENTLY PERFORM PNPS 2.1.6 with this Procedure.

[5] AFTER the recirculation pump is secured, ADJUST total core flow to greater than 27.6 Mlb/hr. [GE SIL 517]

(a) VERIFY that the recirculation loop flow of the active loop on FI-107A or B is less than 36.9 Mlb/hr. [GE SIL 517]

[6] SEND an operator to the 4kV breaker and to the MG Set Room to record all relay targets to determine cause of trip.

[7] ENSURE that power is available as follows:

(a) Instrument power Panel Y1

(b) Vital service Panel Y2

(c) 4160V load centers A3 or A4

(d) Power Centers B17, B18, B20 and D9

(e) Power Panels D4, D5 and D6

[8] IF the cause of the trip can be determined and corrected AND the reactor is operating outside of the area of the power flow map bounded by the 80% load line and 31.5 Mlb/hr, THEN the pump may be restarted in accordance with PNPS 2.2.84, "Reactor Recirculation System". [IEB 88-70, Supp 1; BWROG 8879]

[9] REFER to EPIP-100, "Emergency Classification", to determine whether an Emergency Action Level (EAL) has been exceeded.

4.2 TRIP OF BOTH RECIRCULATION PUMPS

[1] CLOSE both MO-202-5A and B, PUMP DISCH VLVs.

(a) WHEN 5 minutes have elapsed, THEN REOPEN the discharge valves.

[2] SEND an operator to the 4kV breaker and to the MG Set Room to record all relay targets to determine cause of trip.

[3] ENSURE that power is available as follows:

(a) Instrument power Panel Y1

(b) Vital service Panel Y2

(c) 4160V load centers A3 or A4

(d) Power Centers B17, B18, B20 and D9

(e) Power Panels D4, D5 and D6

[4] IF the cause of the pump trips can be identified and corrected, THEN RESTART the pumps in accordance with PNPS 2.2.84, "Reactor Recirculation System".

[5] REFER to PNPS EPIP-100, "Emergency Classification", to determine whether an Emergency Action Level (EAL) has been exceeded.

BOSTON EDISON

PILGRIM NUCLEAR POWER STATION

Procedure No. 5.3.27

DETERMINING PRIMARY CONTAINMENT WATER LEVEL

INFORMATION ONLY

Use restricted to reference

Approved Ed Kraft, Jr. for RAA 6/26/90
Plant Manager Date

1.0 PURPOSE

This procedure provides instructions for determining Primary Containment Water Level when flooding of the Primary Containment is directed by the Emergency Operating Procedures (EOPs).

- Primary Containment (PC) Water Level values are referenced to plant elevation.

2.0 ACTIONS

2.1 FOR CONTAINMENT LEVEL LESS THAN 47 FEET

[1] CALCULATE the existing differential pressure (sensed) between the drywell air space and the bottom of the torus, as follows (Figure 1):

TORUS BOTTOM PRESS
(PI-1001-69)
(Panel C903) ____________________ psig

minus: DRYWELL WIDE
RANGE PRESSURE
(PI-1001-600A/B)
(Panel C170/171) ____________________ psig

______________________________________________

equals: CONTAINMENT TO
TORUS BOTTOM dP ____________________ psig

[2] For the calculated CONTAINMENT TO TORUS BOTTOM dP, the corresponding Primary Containment Water Level is obtained from the Primary Containment Water Level curve (Figure 2).

2.2 FOR CONTAINMENT LEVEL GREATER THAN OR EQUAL TO 47 FEET

[1] READ containment water level on DRYWELL LEVEL Gauge LI-5008 on Panel C903.

| CONTAINMENT TO TORUS BOTTOM dP CALCULATION | | |
|---|---|---|
| | TORUS BOTTOM PRESS<br>(PI-1001-69)<br>(Panel C903) | ____________psig |
| - | DRYWELL WIDE<br>RANGE PRESSURE<br>(PI-1001-600A/B)<br>(Panel C170/171) | ____________psig |
| = | CONTAINMENT TO<br>TORUS BOTTOM dP | ____________psig |

FIGURE 1

# PRIMARY CONTAINMENT WATER LEVEL

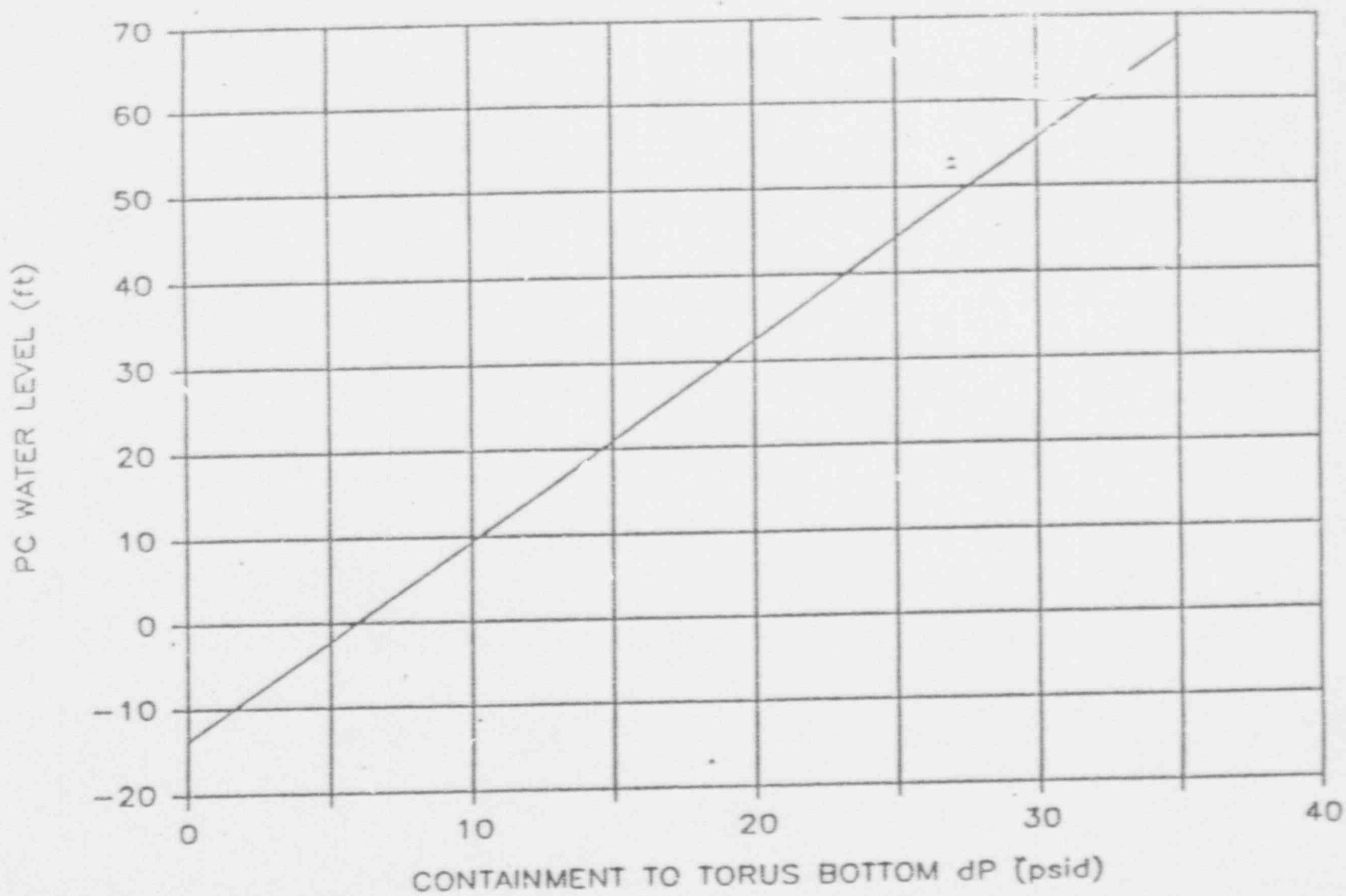
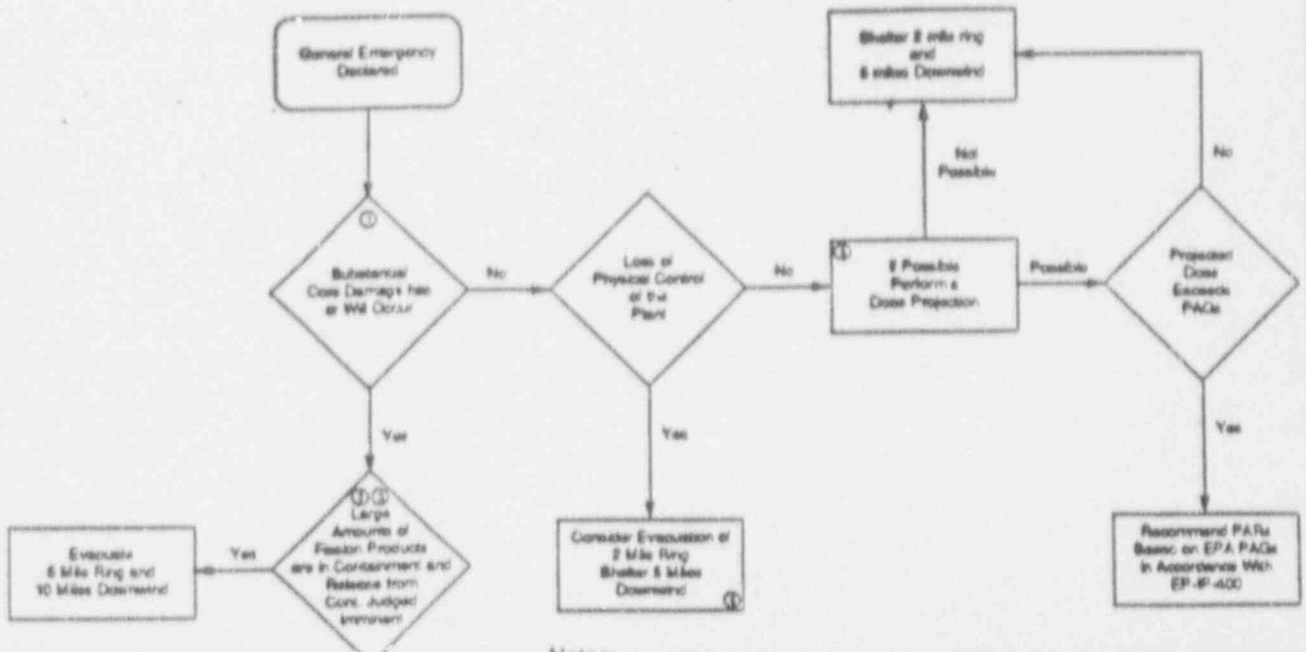
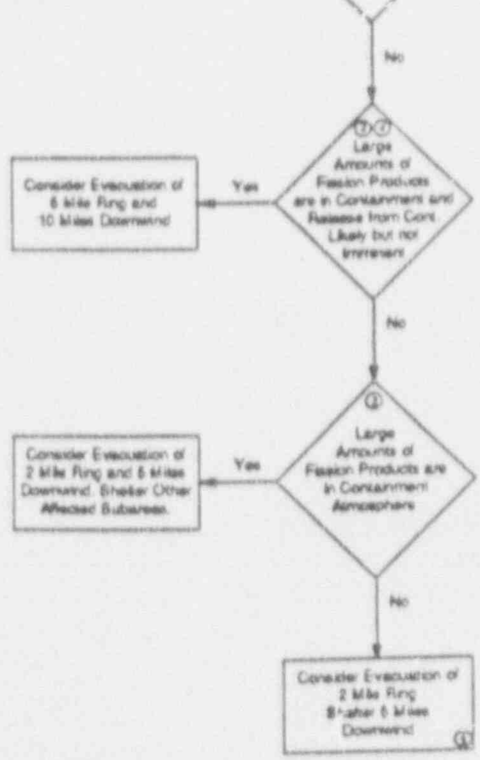



FIGURE 2

ATTACHMENT 1

## Protective Action Recommendations Based on Plant Conditions




Notes:

1) Indications of substantial core damage include:
   - Extremely high and increasing Off-Gas radiation levels.
   - High Main Steam Line radiation levels.
   - Elevated containment high range radiation monitor readings.
   - >20% clad failure as determined by EP-IP-330.
2) Indications of large amounts of fission products in containment include:
   - Containment high range radiation monitors reading >25,000 R/Hr.
   - LOCA with indications of core damage >100% gap as determined by EP-IP-330.
3) Releases from containment, intentional or unintentional, may be judged to be imminent based on indications such as:
   - Primary containment pressure approaching or in excess of the PCPL.
   - Primary containment combustible gas concentrations above deflagratable limits.
4) Releases from containment, intentional or unintentional, may be judged likely but not imminent based on indications such as:
   - Primary containment pressure high and increasing.
   - Primary containment combustible gas concentrations approaching deflagratable limits.
5) Dose projections can be performed using:
   - Effluent monitor readings.
   - Plant parameters.
   - Environmental measurements.
6) Minimum Protective Action Recommendation when in a General Emergency is shelter 2 mile ring and 5 miles downwind.

Before making the final decision on a recommendation consider the following:

- Would the population at risk be able to evacuate before the plume arrives?
- Are the winds variable or expected to shift and effect areas where PARs are not made?
- Would severe weather cause road conditions to be so dangerous as to negate the benefits of the dose saved?

INSERVICE PUMP TESTING DATA SHEET (P-207A)

| TEST PARAMETER | REFERENCE VALUE | | ACCEPTABLE RANGE | ALERT RANGE LOW | ALERT RANGE HIGH | REQ. ACTION RANGE LOW | REQ. ACTION RANGE HIGH | MEASURED VALUE |
|---|---|---|---|---|---|---|---|---|
| FLOW RATE (GPM) | 44.3 | | 41.6 to 45.6 | < 41.6 to 39.9 | > 45.6 to 46.5 | < 39.9 | > 46.5 | 42.8 |
| DISCH. PRESS. (PSI) | 1275 | | NA | NA | NA | < 1275 | NA | 1280 |
| TANK LEVEL AT START (INCH) | 10 1/2 | | NA | NA | NA | NA | NA | 12 5/8 |
| TANK LEVEL AT FINISH (INCHES) | 41 3/8 | | NA | NA | NA | NA | NA | 42 1/2 |
| VIB. DISPLACEMENT (MILS) | (H) NA | (V) 0.91 | 0 to 1.82 | NA | > 1.82 to 2.73 | NA | > 2.73 | 1.21 |
| VIB. VELOCITY (IN./SEC) | (H) NA | (V) 0.15 | ≤ 0.314 | NA | NA | NA | > 0.314 | 0.18 |
| LUBRICANT - STOPPED | VISIBLE | | VISIBLE | NA | NA | NOT VISIBLE | NA | VISIBLE |
| LUBRICANT - RUNNING * | NA | | NA | NA | NA | NA | NA | NA |

PERFORMED BY: (SIGNATURE) [signature] DATE 11/13/92 TIME 1430

IST REVIEWED BY: (SIGNATURE) [signature] DATE 11/13/92 TIME 1600

CALCULATIONS:

[1] (Final Tank Level [Step 17] 42 1/2 inches - Starting Tank Level [Step 11] 12 5/8 inches) x 4.3 gal/inch = 128.5 gallons

[2] 128.5 gallons/3 minutes = Flow rate 42.8 gpm

TEST EQUIPMENT:

| TEST PARAMETER | INST. NO. | CAL. DATE | CAL. DUE DA |
|---|---|---|---|
| DISCHARGE PRESS | P121A | 9/20/92 | 12/20/92 |
| FLOW RATE | F174D | 9/27/92 | 12/27/92 |
| VIBRATION | V36B | 10/12/92 | 1/12/93 |

NOTES:

*Lubricant Level Running cannot be observed due to pump design.

Reference values were obtained during performance of this Procedure on 2/20/90 (vibration) and 6/20/90 (hydraulic).

INSERVICE PUMP TESTING DATA SHEET (P-207B)

| TEST PARAMETER | REFERENCE VALUE (H) | REFERENCE VALUE (V) | ACCEPTABLE RANGE | ALERT RANGE LOW | ALERT RANGE HIGH | REQ. ACTION RANGE LOW | REQ. ACTION RANGE HIGH | MEASURED VALUE |
|---|---|---|---|---|---|---|---|---|
| FLOW RATE (GPM) | 43.7 | | 41.1 to 45.0 | < 41.1 to 39.3 | > 45.0 to 45.9 | < 39.3 | > 45.9 | 40.3 |
| DISCH. PRESS. (PSI) | 1275 | | NA | NA | NA | < 1275 | NA | 1250 |
| TANK LEVEL AT START (INCH) | 6 5/8 | | NA | NA | NA | NA | NA | 10 1/4 |
| TANK LEVEL AT FINISH (INCHES) | 37 1/8 | | NA | NA | NA | NA | NA | 37 3/8 |
| VIB. DISPLACEMENT (MILS) | NA | 0.26 | 0 to 1 | NA | > 1 to 1.5 | NA | > 1.5 | ( |
| VIB. VELOCITY (IN/SEC) | NA | 0.15 | ≤ 0.314 | NA | NA | NA | > 0.314 | ( |
| LUBRICANT - STOPPED | VISIBLE | | VISIBLE | NA | NA | NOT VISIBLE | NA | VISIBLE |
| LUBRICANT - RUNNING * | NA | | NA | NA | NA | NA | NA | NA |

PERFORMED BY: (SIGNATURE) ____________ DATE ________ TIME ________

IST REVIEWED BY: (SIGNATURE) ____________ DATE ________ TIME ________

CALCULATIONS:

[1] (Final Tank Level [Step 17] 37 3/8 inches - Starting Tank Level [Step 11] 10 1/4 inches) x 4.3 gal/inch = 120.9 gallons

[2] 120.9 gallons/3 minutes = Flow rate 40.3 gpm

TEST EQUIPMENT:

| TEST PARAMETER | INST. NO. | CAL. DATE | CAL. DUE DA |
|---|---|---|---|
| DISCHARGE PRESS | P121A | 9/20/92 | 12/20/92 |
| FLOW RATE | F174D | 9/27/92 | 12/27/92 |
| VIBRATION | V360 | 10/12/92 | 1/12/93 |

NOTES: ____________

*Lubricant Level Running cannot be observed due to pump design.

Reference values were obtained during performance of this Procedure on 2/20/90 (vibration) and 6/20/90 (hydraulic).

BOSTON EDISON RTYPE H6.05

PILGRIM NUCLEAR POWER STATION

Procedure No. 5.4.6

PRIMARY CONTAINMENT VENTING AND PURGING
UNDER EMERGENCY CONDITIONS

**INFORMATION ONLY**

**Use restricted to reference**

REVIEWERS AND APPROVERS

| | |
|---|---|
| Thomas A. Iuaina | 3/16/92 |
| Procedure Writer | Date |
| Robert W. Horacie | 3/16/92 |
| Technical Reviewer | Date |
| Thomas A. Iuaina | 3/16/92 |
| Validator | Date |
| Thomas Iuaina for PD Smith | 3/16/92 |
| Procedure Owner | Date |
| N/A | |
| QAD Manager | Date |
| Charles Watt | 3/18/92 |
| ORC Chairman | Date |
| [illegible] for GM | 3/14/92 |
| Plant Manager | Date |

SAFETY REVIEW REQUIRED

ORC REVIEW REQUIRED/
~~NOT REQUIRED~~

Effective Date: 3-20-92

TABLE OF CONTENTS

# 1.0 TORUS VENTING UNDER EMERGENCY CONDITIONS

[1] IF primary containment pressure is below 2.5 psig,

THEN VENT the torus through the SGTS System as follows:

(a) IF while performing Step 1.0[1], primary containment pressure increases above 2.5 psig,

THEN VENT the torus according to Step 1.0[2].

(b) INITIATE OR VERIFY INITIATED SGTS.

(c) OPEN the minimum number of the following valves required to maintain Primary Containment pressure below the Primary Containment pressure limit or to maintain hydrogen/oxygen below the limits of EOP-03, "Primary Containment Control":

---

NOTE

When RPV water level is above low-low level, isolation interlocks for AO-5041A and AO-5041B are defeated by placing valve control switches to the "EMERG OPEN" position.

---

(1) 2" Torus Normal Exhaust Valves (C7):

- AO-5041A, TORUS NORMAL EXHAUST ISOL VLV (Key: CR-33)
- AO-5041B, TORUS NORMAL EXHAUST ISOL VLV (Key: CR-32)

(2) Containment Atmospheric Dilution System vent valves (C170 and C171) Special Key CR-V.

- SV-5083A, TORUS ISOLATION VALVE
- SV-5084A, TORUS ISOLATION VALVE
- SV-5083B, TORUS ISOLATION VALVE
- SV-5084B, TORUS ISOLATION VALVE

(3) 8 in. Torus Purge Exhaust Valves (C7):

- AO-5042A, TORUS PURGE EXHAUST ISOL VLV
- AO-5042B, TORUS PURGE EXHAUST ISOL VLV

[2] WHEN primary containment pressure is at or above 2.5 psig,

THEN VENT the Torus through the SGTS as follows:

*******************************************************************************

CAUTION

Actions performed in this step have the potential to rupture the duct work between the torus vent valves and SGTS resulting in a direct release of primary containment atmosphere to the Reactor Building. All non-essential personnel should be evacuated from the Reactor Building.

*******************************************************************************

(a) INITIATE OR VERIFY INITIATED SGTS.

(b) DEFEAT isolation interlocks as necessary AND OPEN the minimum number of the following valves required to maintain Primary Containment Pressure below the Primary Containment Pressure Limit or to maintain hydrogen/oxygen below the limits of EOP-3, "Primary Containment Control":

---

NOTE

Isolation interlocks for AO-5041A and AO-5041B are defeated by:

1. Placing valve control switches in the "EMERG OPEN" position if RPV water level is greater than low-low level.

OR

2. In accordance with PNPS 5.3.21, "Bypassing Selected Interlocks", Attachment 5 if RPV water level is less than or equal to low-low level.

---

(1) 2" Torus Normal Exhaust Valves (C7):

- AO-5041A, TORUS NORMAL EXHAUST ISOL VLV (Key: CR-33)
- AO-5041B, TORUS NORMAL EXHAUST ISOL VLV (Key: CR-32)

(2) Containment Atmosphere Dilution System Torus Isolation valves (C170 and C171) Special Key CR-V.

- SV-5083A, TORUS ISOLATION VALVE
- SV-5084A, TORUS ISOLATION VALVE
- SV-5083B, TORUS ISOLATION VALVE
- SV-5084B, TORUS ISOLATION VALVE

**********************************************************************

CAUTION

Prior to performing the following, refer to Emergency Classifications IAW EP-IP-100. Reference EAL 3.4.1.4, EAL 3.5.1.4, and EAL 3.5.2.4.

**********************************************************************

(c) IF additional venting capacity is required,

AND primary containment pressure is ≥ 30 psig,

THEN VENT the Torus, bypassing SGTS as follows:

(1) CLOSE OR VERIFY CLOSED the following valves:

- AO-5042A, TORUS PURGE EXHAUST ISOL VLV
- AO-5041A, TORUS NORMAL EXHAUST ISOL VLV
- AO-5041B, TORUS NORMAL EXHAUST ISOL VLV
- SV-5083A, TORUS ISOLATION VALVE
- SV-5084A, TORUS ISOLATION VALVE
- SV-5083B, TORUS ISOLATION VALVE
- SV-5084B, TORUS ISOLATION VALVE

(2) PLACE the fan control switches for SGTS Exhaust Fans VEX-210A AND VEX-210B to "OFF".

(3) CLOSE OR VERIFY CLOSED the following valves:

- AO-N-112, TRAIN B OUTL DMPR
- AO-N-108, TRAIN A OUTL DMPR

(4) OPEN AO-5042B, TORUS PURGE EXHAUST ISOL VLV, bypassing isolation interlocks according to PNPS 5.3.21, "Bypassing Selected Interlocks", Attachment 12.

NOTE

The fuse holders for the two 3 amp fuses are designated "UQQ" in Panel C7.

(5) INSTALL two 3 amp fuses for valve AO-5025, DIRECT TORUS VENT ISOL VLV, in the back of Panel C7 (fuses are stored in the Control Room "Q" Fuse Box).

(6) OPEN AO-5025, DIRECT TORUS VENT ISOL VLV (Key: CR-W).

[3] WHEN venting is no longer required,

THEN TERMINATE Torus venting as follows:

(a) IF Torus venting is through AO-5025, DIRECT TORUS VENT ISOL VLV,

THEN CONTINUE with Step [4].

(b) IF not required to be operating by EOP-4, "Secondary Containment Control",

THEN SHUT DOWN SGTS in accordance with PNPS 2.2.50, "Standby Gas Treatment".

(c) CLOSE OR VERIFY CLOSED the following valves:

- AO-5042A, TORUS PURGE EXHAUST ISOL VLV
- AO-5042B, TORUS PURGE EXHAUST ISOL VLV
- AO-5041A, TORUS NORMAL EXHAUST ISOL VLV
- AO-5041B, TORUS NORMAL EXHAUST ISOL VLV
- SV-5083A, TORUS ISOLATION VALVE
- SV-5084A, TORUS ISOLATION VALVE
- SV-5083B, TORUS ISOLATION VALVE
- SV-5084B, TORUS ISOLATION VALVE

(d) IF control circuits for the following valves were bypassed, THEN RESTORE them to normal in accordance with PNPS 5.3.21, "Bypassing Selected Interlocks", Attachment 5:

- AO-5041A, TORUS NORMAL EXHAUST ISOL VLV
- AO-5041B, TORUS NORMAL EXHAUST ISOL VLV

---

NOTE

The following step (Step [4]) is only to be performed if SGTS has been bypassed and the Torus was vented through the Direct Torus Vent.

---

[4] WHEN venting through the Direct Torus Vent is no longer required,

THEN TERMINATE Torus venting as follows:

(a) CLOSE the following valves:

- AO-5025, DIRECT TORUS VENT ISOL VLV
- AO-5042B, TORUS PURGE EXHAUST ISOL VLV

(b) OPEN the Outlet Valve of the SGTS fan that will be placed in service.

- AO-N-108, TRAIN A OUTL DMPR
- AO-N-112, TRAIN B OUTL DMPR

(c) RETURN the fan control switch for the SGTS Exhaust Fan (VEX-210A OR VEX-210B) that will be placed in service to "AUTO" AND the fan control switch for the standby fan to "STANDBY".

(d) REMOVE the fuses from Panel C7 for valve AO-5025, DIRECT TORUS VENT ISOL VLV, AND RETURN the fuses to the NWE.

(e) RESTORE the control circuit for AO-5042B, TORUS PURGE EXHAUST ISOL VLV, to normal in accordance with PNPS 5.3.21, "Bypassing Selected Interlocks", Attachment 12.