UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

ENCLOSURE 1

EXAMINATION REPORT - 50-338/OL-90-02

Facility Licensee: Virginia Power Company
P. O. Box 402
Mineral, VA 23117

Facility Name: North Anna Power Station

Facility Docket Nos.: 50-338, 50-339

Facility License Nos.: NPF-4 and NPF-7

Examinations were administered at North Anna Power Station near Mineral, Virginia.

Chief Examiner: for M.J. Morgan 12/11/90
Michael J. Morgan — Date Signed

Approved By: Lawrence L. Lawyer 12/12/90
Lawrence L. Lawyer, Chief — Date Signed
Operator Licensing Section 1

SUMMARY

Examinations were given on September 24-28, 1990.

Written examinations and operating tests were administered to eight Senior Reactor Operator (SRO) and eight Reactor Operator (RO) applicants. All passed.

REPORT DETAILS

1. Facility Employees Contacted During the Examination

*G. E. Kane, Station Manager
*R. D. Enfinger, Assistant Station Manager
*A. Stall, Supt. Operations
*L. L. Edmonds, Supt. Nuclear Training
*M. A. Allan, Training Supervisor
*M. D. Crist, Ops. Coord. - Training
M. Lesser, Senior Resident Inspector
*L. King, Resident Inspector
M. J. Buhrer

*Attended Exit Meeting

2. Examiners

*M. J. Morgan, Region II
R. F. Aiello, Region II
+J. F. Munro, Region II
K. L. Parkinson, Sonalysts
B. Haagensen, Sonalysts
G. Weale, Sonalysts
D. Lane, Sonalysts

*Chief Examiner
+Observer

3. Exit Meeting

At the conclusion of the site visit, the examiners met with representatives of the plant staff to discuss the results of the examinations. The examiners made the following observations concerning your facility and training program:

a. A generic weakness was found in the area of Emergency Diesel Generators. Candidates were not aware of all the emergency diesel generator shutdowns following an emergency start. In particular, Low lube oil Pressure.

b. Areas in which the examiners believe that the applicants exhibited good training and knowledge were procedure usage, communications, diagnosis of events, and control board operations.

c. Plant cleanliness was noticeable and commendable. The applicants displayed pride and a sense of "ownership" in the good appearance of the facility.

d. Several procedural problems were identified:

- Operations Procedure OP-6.1, "Unloading 1J DG", Step 5.5, Provides instructions for unloading the 1H DG instead.

- Periodic Test Procedure 91, "Containment Penetrations" did not invoke the true definition of containment integrity. In particular, neither containment leakage rates nor the operability of the penetration sealing mechanism were defined per the Technical Specifications.

- Abnormal Procedure 4.1, "Malfunction of the Source Range Nuclear Instrument" list the incorrect breaker for the motor generator in step 5.3.4.1.

- Administrative Procedure 16.5, "Work Request" was pulled by one candidate to complete a work request. Another candidate used VPAP 2002, "Work Request and Work Orders". VPAP 2002 was identified as the correct procedure, not Administrative procedure 16.5.

- Transparencies or handouts referenced in the text of the NCRODP training lessor plans were blank except for transparency number and title.

- There were numerous errors in reference material indexing. For example, NCRODP 93.7, "Pressurizer Pressure and Level Protection and Control" was labeled on the reference book as NCRODP 93.8.

- None of the even pages to the EPIPs were sent. This was later attributed to a two sided copy problem. Nevertheless, this is a result of inattention to detail during the pre-shipping materials check.

g. Security in-processing of visitors at the main gate was, for the most part, very efficient. However, there were a couple of times where long waits during in processing could have contributed to unnecessary stress on the candidate. Once entry was made, movement to various areas was relatively unimpeded.

The pre-examination facility technical review of the written examinations proved beneficial.

The cooperation given to the examiners was noted and appreciated. The licensee did not identify as proprietary any material provided to or reviewed by the examiners.

Nuclear Regulatory Commission
Operator Licensing
Examination

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Official Use Only category on
date of examination.

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION
REGION 2

FACILITY: North Anna 1 & 2

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 90/09/24

CANDIDATE: ____________________

INSTRUCTIONS TO CANDIDATE:

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

NUMBER QUESTIONS	TOTAL POINTS	CANDIDATE'S POINTS	CANDIDATE'S OVERALL GRADE (%)
100	97.00		

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

____
Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. After the examination has been completed, you must sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination. This must be done after you complete the examination.

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

4. Use black ink or dark pencil only to facilitate legible reproductions.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

22. When you are done and have turned in your examination, leave the examination area as defined by the examiner. If you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION: 001 (1.00)

Which ONE (1) of the following is the correct sequence of actions required upon discovery of a SMALL fire in a trash can in the auxiliary building?

a. Report the fire to the control room, attempt to extinguish the fire, notify the Station Safety and Loss Prevention Department to refill the extinguisher.

b. Report the fire to the Fire Brigade, remain at the scene of the fire, advise the Fire Brigade of any personnel who are in the area, leave the area.

c. Report the fire to the control room, remain at the scene of the fire, advise the Fire Brigade of work in progress and of hazards that may be in the area.

d. Report the fire to the control room, attempt to extinguish the fire, notify the Fire Brigade to refill the extinguisher.

QUESTION: 002 (1.00)

The concentration of oxygen in the waste gas decay tank is limited to a maximum of WHAT percent by volume when the hydrogen concentration is 4% to 96%?

a. 1 %

b. 2 %

c. 3 %

d. 4 %

QUESTION: 003 (1.00)

Which one of the following is an example of an area where a "Confined Space Entry Permit" must be in effect prior to entry?

a. The containment building.

b. A Cable tunnel

c. A Switchgear cabinet.

d. The condensate storage tank.

QUESTION: 004 (1.00)

If the atmosphere in a confined space contains 11% chlorine gas, 20.3% oxygen, 67.5% Nitrogen and 1.2% Carbon-dioxide, which ONE (1) of the following definitions correctly defines the atmosphere in accordance with ADM-20.10, Confined Area Entry Procedure?

a. Oxygen deficient

b. Toxic or nuisance

c. Non-toxic

d. Non-hazardous

QUESTION: 005 (1.00)

In accordance with ADM-20.30, caustic spills are reported to the Shift Supervisor. What reportable hazardous material is stored in the Auxiliary Building that would constitute a caustic spill?

a. Sodium Hydroxide.

b. Ammonium Hydroxide.

c. Aluminum Sulfate.

d. Potassium Hydroxide.

QUESTION: 006 (1.00)

Which ONE (1) of the following statements correctly describes the requirements for manipulation of systems and components outside the control room that may indirectly affect unit reactivity or power level?

a. A qualified operator with the knowledge and consent of an NRC licensed operator on duty in the control room.

b. An on-duty NRC licensed operator with the knowledge and consent of the SRO on shift.

c. Operations Department personnel with the knowledge and consent of an Operations Department supervisor.

d. An NRC licensed operator with the knowledge and consent of the Supervisor Shift Operations.

QUESTION: 007 (1.00)

How is a narrative log entry made if it was inadvertently omitted during the shift?

a. Record the current time and the actual event time in the "TIME" space then, following the log entry, record the words "LATE ENTRY".

b. Record the current time in the "TIME" space then, the actual event time preceding the log entry record and the words "LATE ENTRY".

c. Record the words "LATE ENTRY" in the "TIME" space then, the actual event time the event occurred preceding the log entry.

d. Record the actual event time in the "TIME" space then, preceding the log entry record the words "LATE ENTRY".

QUESTION: 008 (1.00)

When may an operator deviate from an EMERGENCY PROCEDURE without first obtaining an approved procedure deviation?

a. When written instructions in the procedure provide for a deviation otherwise verbatim compliance is required.

b. When conditions exist that warrant deviation from the procedure to mitigate the severity of an accident in progress.

c. Anytime, based on the judgment, knowledge and experience of the licensed operator if the existing conditions require a procedure deviation.

d. When the deviation to a procedure that changes the initial conditions is handled with a "prior to use" deviation.

QUESTION: 009 (1.00)

In accordance with VPAP-0501 who is responsible for verifying that operating procedures to be used are the correct revision?

a. The cognizant supervisor.

b. The procedure user.

c. The Manager O & M.

d. The on-call SRO.

QUESTION: 010 (1.00)

What is the maximum number of visitors a licensed operator may escort in the control room?

a. 1 person

b. 3 persons

c. 5 persons

d. 10 persons

QUESTION: 011 (1.00)

Which ONE (1) of the following is responsible for maintaining dose As Low As Reasonably Achievable (ALARA)?

a. The Assistant Station Manager - Operations and Maintenance, Chairman of the ALARA committee.

b. The ALARA Coordinator.

c. The Health Physics Department.

d. The individual radiation worker.

QUESTION: 012 (1.00)

Which ONE (1) of the following valve alignment verifications would be acceptable in accordance with VPAP 1405, "Independent Verification"

a. Fully close valves then reposition the valves to the desired position.

b. Fully open valves then reposition the valves to the desired position.

c. By Observation of valve position and process parameter verification.

d. Attempt to reposition open valves to open and closed valves to closed.

QUESTION: 013 (1.00)

Which ONE (1) of the following groups of functions does the PRODAC-250 computer perform during normal plant operations at 100% reactor power?

1. Determines AFD.

2. Calculates the value of Keff.

3. Performs daily calorimetric.

4. Determines average containment temperature.

5. Performs leak-rate calculations.

6. Determines rod height and compares it to the group position.

(Select the choice that includes ALL correct answers.)

a. 1, 2, 3, 4, 5

b. 2, 3, 4, 5, 6

c. 1, 3, 4, 5

d. 1, 3, 4, 5, 6

QUESTION: 014 (1.00)

Technical Specifications state that protection is provided to prevent DNB. Which one of the following reactor trips provide this protection.

a. Power range High Flux rate

b. Over-Power Delta-T

c. Over-Temperature Delta-T

d. Low Pressure

QUESTION: 015 (0.00)

DELETED

QUESTION: 016 (1.00)

A Loss of Coolant Accident (LOCA) has occurred causing RCS pressure to ramp down at a rate of 200 psi/minute. WHICH ONE (1) of the following lists the correct order of SI?

a. Charging pumps, SI accumulators, LHSI pumps.

b. SI accumulators, HHSI pumps, Charging pumps.

c. SI accumulators, Charging pumps, LHSI pumps.

d. Charging pumps, HHSI pumps, LHSI pumps.

QUESTION: 017 (1.00)

Which ONE (1) of the following correctly describes the expected impact on containment environment when defining a Design Basis Accident (DBA) Loss of Coolant Accident (LOCA).

a. Containment pressure, temperature and humidity will increase.

b. Containment temperature will decrease, pressure and humidity will increase.

c. Containment temperature and pressure will increase and humidity will decrease.

d. Containment pressure and humidity will increase and temperature will not change.

QUESTION: 018 (1.00)

If an abrupt steam line break upstream of the main steam line isolation valves occurs, which ONE (1) of the following choices correctly describes the response of the faulted steam generator water level.

a. Level would increase due to pressure decrease then hold steady at program level.

b. Level would increase due to swell then decrease as the generator boiled dry.

c. Level would decrease due to shrink and as the generator boiled dry.

d. Level would decrease due to pressure decrease then hold steady at program level.

QUESTION: 019 (1.00)

Which ONE (1) of the following alarm set-points is provided by the Tavg PROTECTION CHANNELS?

a. Alarm at 2 F comparison of loop Tavg and auctioneered Hi Tavg.

b. Alarm at 3 F comparison of loop Tavg and the Hi Tavg signal.

c. Alarm at 4 F comparison of loop Tavg and full load Tavg.

d. Alarm at 5 F comparison of Auctioneered Hi Tavg to Tref.

QUESTION: 020 (1.00)

Given the following conditions:

Tavg = Tref
Auto Rod Control is selected

How will the rod control system respond to a 1 F STEP INCREASE in Tref?

a. The control rods remain in position.

b. The control rods step out at 8 steps per minute.

c. The control rods step out at 48 steps per minute.

d. The control rods step out at 64 steps per minute.

QUESTION: 021 (1.00)

During a continuous rod withdrawal accident, how are steam generator LEVEL and PRESSURE affected?

a. Level will increase and pressure will increase due to Tavg increase.

b. Level will increase and pressure will decrease due to Tavg increase.

c. Level will decrease and pressure will increase due to Tavg decrease.

d. Level will increase and pressure will decrease due to Tavg decrease.

QUESTION: 022 (1.00)

Which ONE (1) of the following groups correctly state the sources of heat input which are included in the design of the AFW system?

a. Steam generator residual heat, Pressurizer heater input and reactor plant startup heat.

b. Core decay heat, Reactor Coolant Pump heat input, stored heat in RCS and steam generator metal.

c. Reactor vessel heat, RCS stored heat and pressurizer stored heat.

d. Main Turbine heat, steam generator residual heat and reactor plant stored heat.

QUESTION: 023 (1.00)

WHICH one of the following statements is NOT correct in reference to the turbine driven AFW pump steam supply valves?

a. Both 1-MS-TV-111A and 111B receives a signal from train A and B solid state protection.

b. Both 1-MS-TV-111A and 111B fail open on loss of air.

c. Both 1-MS-TV-111A and 111B fail open on loss of electrical power.

d. Both 1-MS-TV-111A and 111B can independently supply enough steam to run the terry turbine. (each is capable of supplying)

QUESTION: 024 (1.00)

Which one of the following statemen orrect concerning the motor driven and turbine driven auxiliary feedwat nps?

a. The turbine driven pump can be shutdown following an SI signal only after the SI signal has been reset.

b. A motor driven pump has no time delay following bus voltage restoration for auto start except SI.

c. The turbine driven pump's steam supply valves (TV-111A/B) fail open on a loss of power and fail as-is on a loss of air (but will drift open due to steam pressure).

d. A motor driven pump cannot be shutdown following a S/G LO-LO level start until the LO-LO level condition has cleared.

QUESTION: 025 (1.00)

Given that the highest incore thermo-couples indicate the following temperatures:

- 1210 F
- 1212 F
- 1212 F
- 1213 F
- 1214 F
- 1213 F

Which ONE (1) of the following conditions is indicated?

a. Core damage is imminent.

b. Core temperature at 100% reactor power.

c. Stuck control rod.

d. Core temperature below DNB.

QUESTION: 026 (1.00)

Which one of the following correctly describes the flowpath for relief of an overpressure condition in a waste gas decay tank?

a. Relief valve, rupture disc to "B" ventilation stack.

b. Relief valve, rupture disc, downstream of GW-101 to process vent system.

c. Rupture disc, relief valve, downstream of GW-101 to process vent system.

d. Rupture disc, relief valve to "B" ventilation stack.

QUESTION: 027 (1.00)

When venting from the high point vent connections, WHERE is liquid carryover entrained in the vent gases collected?

a. Primary Vent Pot.

b. Pressurizer Relief Tank (PRT).

c. Containment Sump.

d. Waste Gas Decay Tanks.

QUESTION: 028 (1.00)

Procedure 1-OP-22.11, Releasing Radioactive Liquid Waste, states:

"WHEN either of the following occurs, THEN push the CLOSE button for 1-LW-PCV-115.

- Clarifier discharge flow less than 125 gpm

OR

- CLARIFIER SURGE TANK A/B LO LEVEL alarm is received."

WHAT FUNCTION is provided by closing 1-LW-PCV-115?

a. Terminates the liquid waste discharge.

b. Prevents a vacuum from forming in the Clarifier Surge tank.

c. Re-directs flow to the Clarifier Holdup tank.

d. Re-directs flow to the Clarifier Surge tank.

QUESTION: 029 (1.00)

The Clarifier Liquid Waste Radiation Monitor LW-111 provides a signal to which ONE of the following valves?

a. LW-LCV-103A/B, Blowdown pump discharge valve(s).

b. LW-LCV-105A/B, Clarifier Discharge Pump discharge.

c. LW-FCV-110, Clarifier influent valve.

d. LW-PCV-257, Clarifier control valve.

QUESTION: 030 (1.00)

If the discharge check valve on a shutdown charging pump fails to close, HOW is injection flow affected?

a. Neither seal injection flow nor charging flow are affected.

b. Seal injection flow is reduced but, charging flow is unaffected.

c. Seal injection flow is unaffected but, charging flow is reduced.

d. Both seal injection flow and charging flow are reduced.

QUESTION: 031 (1.00)

WHY is valve HCV 1142, isolation valve for the RHR letdown to the CVCS, kept approximately 10% open?

a. To provide a flow path for the recirculation phase of emergency injection.

b. To provide a path to keep the RHR system full and allow for normal expansion and contraction.

c. To prevent over-pressurization of the RHR system during normal full power operations.

d. To provide a mixing flow path to maintain boron concentration in the RHR system approximately equal to the RCS.

QUESTION: 032 (1.00)

Given the following conditions:

- Normal charging is in service from the VCT.
- One charging pump is running.
- RCS pressure is maintained by CVCS pressure control valve, PCV-1145.

These conditions describe pressure control during WHICH ONE of the following plant operations?

a. Refueling

b. Hot Standby

c. Turbine startup

d. Solid plant

QUESTION: 033 (1.00)

An increase in Tavg causes an insurge to the pressurizer. HOW does the change in pressurizer level affect the CVCS system?

a. As level increases in the pressurizer a pressurizer level error signal is used to control LCV-1460A/B.

b. Pressurizer level and Lref, developed from auctioneered high Tavg control are compared to provide a level error signal to FCV-1122.

c. As level increases in the pressurizer a pressurizer level signal is sent to MOV-1289 A and B to control charging flow.

d. Pressurizer level and Tavg are compared to provide a level error signal to LCV-1115A which increases letdown flow.

QUESTION: 034 (1.00)

Select the choice that correctly matches the RADIATION MONITORS to the AUTOMATIC FUNCTIONS they perform upon a HIGH-HIGH radiation condition.

AUTOMATIC FUNCTIONS	RADIATION MONITORS
A. Closes the containment purge supply and exhaust MOV's.	1. RM-VG-103/104 "A" Vent Stack
B. Closes the containment vacuum pump discharge trip valves.	2. RM-GW-101 & 102 Process vent monitor
C. Closes the flow control valve from the waste gas decay tanks.	3. RM-VG-112/113 "B" Stack Monitor
D. Causes the containment vacuum pumps to trip off.	4. RM-RMS-159/160 Containment monitors
	5. RM-LW-111, Clarifier outlet sampler

a. A - 2, B - 1, C - 3, D - 4

b. A - 2, B - 4, C - 5, D - 4

c. A - 4, B - 2, C - 2, D - 2

d. A - 4, B - 3, C - 5, D - 4

QUESTION: 035 (1.00)

The plant is shut-down with the vessel head removed for refueling.

Which ONE (1) of the following conditions is the probable cause for an alarm actuated by radiation monitor RMS-162, Manipulator Crane.

a. Loss of letdown to the VCT.

b. Loss of refueling cavity level.

c. Incore detector movement.

d. RHR flow decreases to 3000 gpm.

QUESTION: 036 (1.00)

Which ONE (1) of the following conditions is indicated by an actuating of the control room area radiation monitor RMS-157 alarm?

a. A health hazard to control room personnel exists.

b. The control room ventilation system has isolated.

c. Notification must be made that an off-site radiation release is in progress.

d. A primary to secondary system leak has occurred.

QUESTION: 037 (1.00)

WHICH ONE (1) of the following choices describes North Anna's fire protection system in the main control room Computer Room?

a. Water Deluge System, actuated by a heat sensor set at a predetermined temperature.

b. Fire Suppression Foam System, coats the surface and surrounding structures to extinguish the fire.

c. High Pressure CO2, Carbon dioxide displaces oxygen and leaves no residue.

d. Halon 1301 System, halon displaces oxygen and leaves no residue.

QUESTION: 038 (1.00)

Which ONE (1) of the following statements correctly describes the fire extinguishing system associated with the Fuel Oil Pump house.

a. The H.P. CO_2 system is actuated by a heat detector with a temperature rate of rise exceeding 24 F/minute.

b. The Halon fire extinguishing system is actuated by either heat detectors at the charcoal filter or smoke detectors.

c. The L.P CO_2 system dumps to the HVAC filters, trips fan dampers and is actuated by either heat detectors or smoke detectors.

d. The Deluge system is actuated by an electrical signal from the heat detection system.

QUESTION: 039 (1.00)

If a pressurizer safety valve sticks open and NO operator action is taken, HOW will pressurizer level respond?

a. Decreases due to loss of inventory; when system pressure reaches saturation, pressurizer level will increase.

b. Remain steady then increase as system pressure reaches saturation pressure.

c. Increases due to swell; then decreases as inventory is lost; when pressure reaches saturation, level will increase.

d. Decreases as inventory is lost, then when pressure reaches saturation pressure, level will rapidly decrease.

QUESTION: 040 (1.00)

G. ven the followi conditions:

- Pressurizer vel controller LC-459F indicates that pressurizer level is less than p. gram level.
- Charging flow control valve FCV-1122 is ramping open in auto to increase charging flow.

Which ONE (1) of the following conditions correctly describe the functions of the Pressurizer level control system?

a. Reactor power is decreasing from 50% to 40% and pressurizer level control system is in automatic.

b. Reactor power is decreasing from 50% to 40% and pressurizer level control system is in manual.

c. Reactor power is increasing from 50% to 60% and pressurizer level control system is in manual.

d. Reactor power is increasing from 50% to 60% and pressurizer level control system is in automatic.

QUESTION: 041 (0.00)

DELETED

QUESTION: 042 (1.00)

Which ONE (1) of the following reactor trip signals is blocked or unblocked by the P-7 permissive?

a. Single Loop Low Coolant Flow Trip.

b. Two Loop Low Coolant Flow Trip.

c. Pressurizer High Pressure Trip.

d. Source Range Reactor Trip.

QUESTION: 043 (1.00)

Select the choice that correctly matches the REACTOR TRIP to the PROTECTION the trip provides.

REACTOR TRIP	PROTECTION
A. Pressurizer High Level	1. Protects against excessive fuel center line temperature (fuel melt).
B. Over-Temperature Delta T	2. Protection against reactivity changes too fast for temperature and pressure protective circuitry
C. Over-Power Delta T	3. Protects against DNB.
D. Neutron Flux Positive Rate	4. Prevents RCS over-pressurization
	5. Protects against loss of coolant.

a. A - 5, B - 1, C - 4, D - 3

b. A - 3, B - 1, C - 4, D - 2

c. A - 5, B - 3, C - 2, D - 4

d. A - 4, B - 3, C - 1, D - 2

QUESTION: 044 (1.00)

If RCS pressure decreases below 1275 psig during a LOCA/reactor shutdown, WHY are the charging pump recirc valves closed?

a. Recirc flow through the charging pumps below 1275 psig will cause the pumps to over heat.

b. Damage to the charging pumps is not a consideration below 1275 psig.

c. Stopping recirc flow at 1275 psig increases injection flow to the RCS.

d. Below 1275 psig the Charging Pumps begin to cavitate.

QUESTION: 045 (1.00)

If a small break in a main steam system header has raised containment pressure to 2.5 psig. How is containment isolation affected?

a. Phase A isolation has actuated and can be reset.

b. Phase B isolation has actuated and can be reset.

c. Phase A isolation has actuated and cannot be reset.

d. Phase B isolation has actuated and cannot be reset.

QUESTION: 046 (1.00)

Given the following conditions:

- LOCA inside containment has occurred.
- SI automatically initiated on LO-LO pressurizer pressure.
- Pressurizer pressure instruments were damaged after the SI automatically initiated.

What is the DESIGN FEATURE that assures reactor protection after the pressurizer pressure instrument is damaged?

a. Pressurizer pressure detectors and wiring runs are heavily constructed and designed withstand missile hazards.

b. SI logic circuitry seals-in (latches) when actuated and will be unaffected by pressurizer instrument damage.

c. The pressurizer pressure instruments are located away from all potential missile hazards.

d. Only one out of three signals is required to initiate an SI if the LOCA damages ONE (1) of the three instruments.

QUESTION: 047 (1.00)

WHICH ONE of the following correctly describes ALL the automatic actions that occur in the containment purge system upon a HI-HI radiation alarm in containment?

a. Containment purge supply and exhaust valves shut.

b. Containment purge supply and exhaust valves shut and fans stop.

c. Containment purge supply valves shut and fans stop.

d. Containment purge supply and exhaust fans stop.

QUESTION: 048 (1.00)

The Unit 1 Emergency Diesel Generators 1H and 1J have started due to the receipt of an SI signal. A phase imbalance of 150 amps picks up the 87X relay on the 1H Emergency Diesel Generator. The lubricating oil pressure instrument for 1J Emergency Diesel Generator fails low. HOW are the emergency diesel generators affected by these faults?

a. Both the 1H and 1J Emergency Diesel Generators will trip.

b. The 1H Emergency Diesel will trip and the 1J Emergency Diesel will continue to run.

c. The 1J Emergency Diesel will trip and the 1H Emergency Diesel will continue to run.

d. Both the 1H and 1J Emergency Diesel Generators will continue to run.

QUESTION: 049 (1.00)

Assume that the 1-II 120 VDC bus has been lost due to a failed battery charger and depleted battery. Breaker #1 has been closed to cross-tie the 1-II bus to the 1-I bus. WHAT component is the most limiting in this configuration per 1-AP-10.12, Restoration of DC buses?

a. Battery

b. Crosstie

c. Battery Charger

d. Inverter

QUESTION: 050 (1.00)

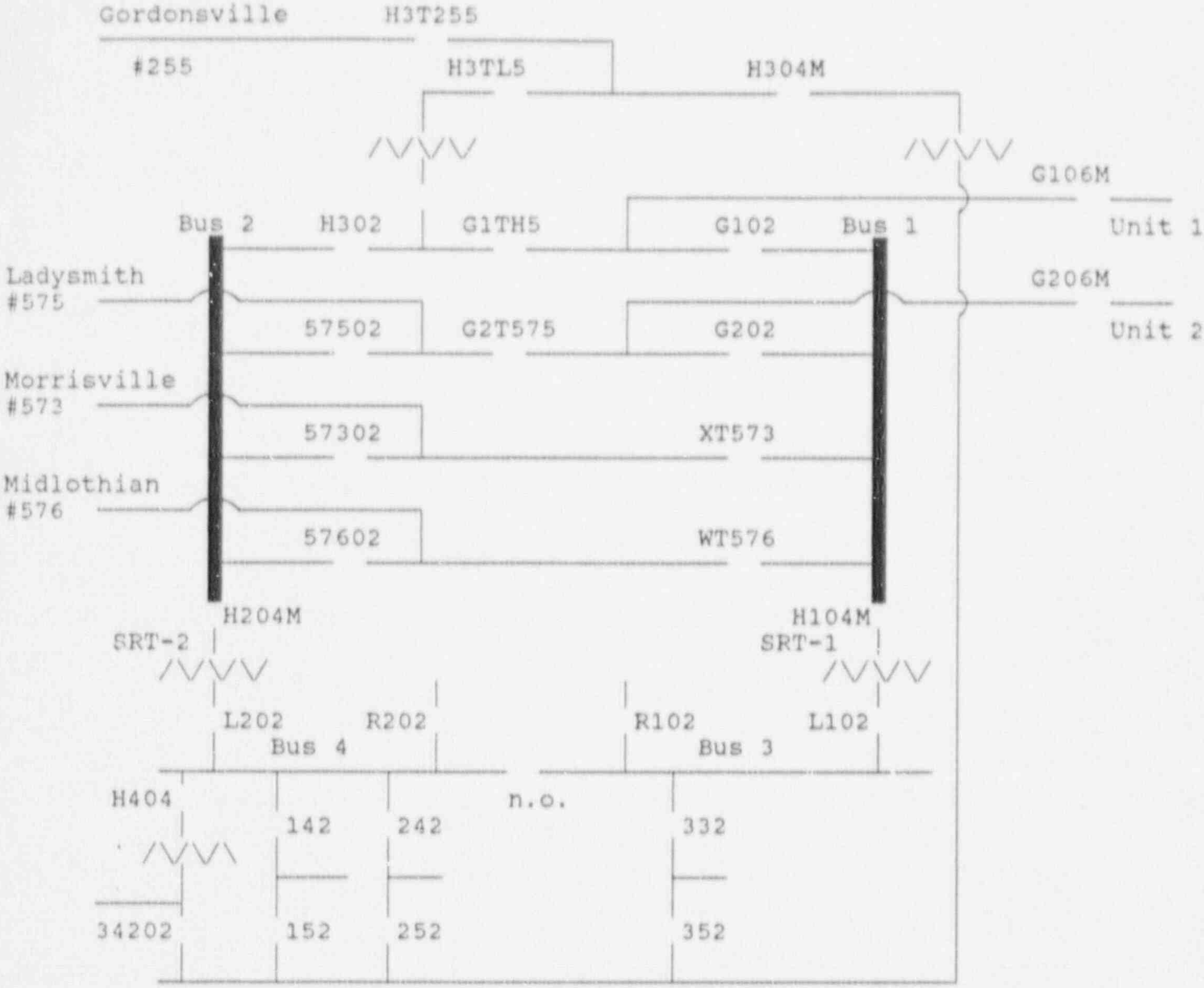
Using the drawing attached, if a fault exists on the Ladysmith # 575 line, HOW is electrical power distributed to Main Switchyard Buses 1 and 2 from Unit 2?

a. Power is supplied through breakers 57502 and G2T575.

b. Power is supplied through breakers G-202, 57502 and G2T575.

c. Power is supplied through breakers WT576 and 57602.

d. Power is supplied through breakers G-202, XT573 and 57302.



Gordonsville
#255
H3T255
H3TL5
H304M
G106M
Unit 1
Bus 2
H302
G1TH5
G102
Bus 1
Ladysmith
#575
G206M
57502
G2T575
G202
Unit 2
Morrisville
#573
57302
XT573
Midlothian
#576
57602
WT576
H204M
SRT-2
H104M
SRT-1
L202
R202
R102
L102
Bus 4
Bus 3
H404
n.o.
142
242
332
34202
152
252
352

QUESTION. 051 (1.00)

Given the following conditions:

- Unit startup in MODE 5.
- RCS is filled solid.
- RCPs are running providing heat input to the RCS.

WHICH ONE of the following describes HOW the Pressurizer Pressure Control system is aligned during these conditions?

a. Spray valves are in auto, pressure is controlled by pressurizer heaters, PORV setpoints are reduced.

b. Pressurizer heaters are on, spray valves are shut and pressure is controlled by charging/letdown, PORV setpoints are reduced.

c. Pressure is controlled manually by pressurizer heaters, spray and CVCS flow control, PORV setpoints are reduced.

d. Pressurizer heaters are off, spray valves are open, pressure is controlled by charging/letdown, PORV setpoints are reduced.

QUESTION: 052 (1.00)

Which one of the following will cause actual pressurizer level to increase?

a. Pressurizer pressure level channel fails high.

b. Steam line leak

c. Increase in RCS temperature

d. RCS boration

QUESTION: 053 (1.00)

When filling and venting the RCS, RCPs are started with the RCS solid to allow opening of the Tc loop stop valves. Technical Specifications requires that SG temperatures be less than HOW many degrees above RCS cold temperature under these conditions.

a. 20 F

b. 30 F

c. 50 F

d. 70 F

QUESTION: 054 (1.00)

If the temperature control valve in the Emergency Diesel Generator Jacket Cooling system fails in the bypass position, HOW is the diesel affected?

a. Coolant would bypass the radiator resulting in an over-heated gear box.

b. Coolant would bypass the radiator resulting in an over-heated engine.

c. Coolant would bypass the lube oil cooler resulting in over-heated lube oil.

d. Coolant would bypass the engine resulting in an over-heated engine.

QUESTION: 055 (1.00)

Given the following conditions:

- Both Unit 1 and 2 are operating at 100%.
- Stormy weather has suddenly caused the Reserve Station Service (RSS) voltage to drop approximately 12 % below rated voltage, this reduced voltage condition is sustained for a 2 minute period.
- The 1J diesel generator mode selector switch is positioned to the "Automatic Remote" position while the 1H diesel generator's mode selector switch is positioned to the "Manual Remote" position.

HOW will each diesel generator react to the degraded grid voltage condition?

a. The 1J Diesel Generator will start automatically, the 1H Diesel Generator can only be started manually.

b. The 1J Diesel Generator will start automatically, the 1H Diesel Generator is locked out and cannot be started.

c. Both 1H and 1J Diesel Generators will start automatically.

d. Degraded RSS voltage has no effect on the Diesel Generators.

QUESTION: 056 (1.00)

Given the following conditions:

- 1J diesel generator is paralleled with the 1J bus.
- KVAR is adjusted from 50 KVAR to 250 KVAR out.

Which ONE (1) of the following statements describes how the 1J diesel generator parameters are affected?

a. Voltage is essentially constant, current increases.

b. Current is essentially constant, voltage increases.

c. Current increases, and voltage increases.

d. Voltage and current remains essentially constant.

QUESTION: 057 (1.00)

What are the IMMEDIATE ACTIONS to be taken in response to a high radiation alarm on RM-RMS 162 (Manipulator) and a decreasing level in the Refueling Cavity as per 1-AP-52, Loss of Refueling Cavity During Refueling?

a. [1.] RETURN ANY FUEL ASSEMBLIES IN THE CAVITY TO THE REACTOR VESSEL.
[2.] RETURN ANY FUEL ASSEMBLIES IN THE FUEL BUILDING TRANSFER CANAL TO THE SPF.
[3.] IF REQUIRED DUE TO HIGH RADIATION, THEN EVACUATE CONTAINMENT AND VERIFY CONTAINMENT INTEGRITY.

b. [1.] COMMENCE MAKEUP TO THE REFUELING CAVITY.
[2.] ISOLATE THE REFUELING CAVITY FROM THE SFP.
[3.] IF REQUIRED DUE TO HIGH RADIATION, THEN EVACUATE CONTAINMENT AND VERIFY CONTAINMENT INTEGRITY.

c. [1.] STOP ANY RUNNING SPF COOLING PUMP AND RP PUMP.
[2.] CHECK SFP LEVEL DECREASING.
[3.] IF REQUIRED DUE TO HIGH RADIATION, THEN EVACUATE CONTAINMENT AND VERIFY CONTAINMENT INTEGRITY.

d. [1.] COMMENCE MAKEUP TO THE REFUELING CAVITY.
[2.] CHECK SFP LEVEL DECREASING.
[3.] IF REQUIRED DUE TO HIGH RADIATION, THEN EVACUATE CONTAINMENT AND VERIFY CONTAINMENT INTEGRITY.

QUESTION: 058 (1.00)

In accordance with 1-OP-14.1 (Residual Heat Removal) Chemistry personnel have sampled RHR for boron concentration. The results snow that RHR concentration is 83 ppm less than the RCS. After performing the necessary calculations to compensate for the dilution of the RHR system what is the REQUIRED ACTION?

a. Borate the RCS if boron concentration is less than the Cold Shutdown boron concentration requirements before starting the RHR pump.

b. Borate the RHR system if boron concentration is less than the Cold Shutdown boron concentration requirements before starting the RHR pump.

c. Start the RHR pump and borate the RCS if boron concentration is less than the Cold Shutdown boron concentration requirements.

d. Start the RHR pump and borate the RHR if boron concentration is less than the Cold Shutdown boron concentration requirements.

QUESTION: 059 (1.00)

WHY is placing RHR in service following shutdown after a ONE (1) month period of operation at 100% power DELAYED for approximately 20 hours?

a. To provide sufficient time for operators to realign the RHR system.

b. To permit decay heat production to decrease to within the heat removal capacity of the RHR pumps.

c. To permit decay heat production to decrease to within the heat removal capacity of the RHR heat exchangers.

d. To permit RCS cooldown at a rate of 20 F per hour before placing RHR in operation.

QUESTION: 060 (1.00)

The on-shift Reactor Operator has noticed that Component Cooling Surge tank level has decreased to 59% (level normally maintained is 62%) and has been directed to refill the surge tank to the normal level. What is the SOURCE of makeup and how is level NORMALLY increased. (Assume no malfunction exists.)

a. Makeup is from the Condensate system and level is increased by placing LCV-100, CC surge tank level control valve, in Automatic.

b. Makeup is from Service water and level is increased by placing LCV-100, CC surge tank level control valve, in Automatic.

c. Makeup is from Service water and level is increased by manually bypassing the LCV-100,CC surge tank level control valve.

d. Makeup is from the Condensate system and level is increased by manually bypassing the LCV-100,CC surge tank level control valve.

QUESTION: 061 (1.00)

A reactor trip from 85% power has just occurred. The Reactor Operator notes the following parameters:

- Pressurizer Level - 42%
- Pressurizer Pressure - 2016 psig
- Tavg - 561 F
- Turbine Power - 0%

Which of the following represents the proper response of the steam dump system?

a. All steam dumps fully open.

b. Dumps A, B, C, and D open.

c. Dumps A, B, C, D, E, and F open.

d. All steam dumps fully shut.

QUESTION: 062 (1.00)

If the "condenser available permissive C-9" is satisfied, HOW is an "arming signal for the Steam Dumps provided?

a. 3/4 throttle valves shut and 3/3 ASO low pressures <45 psig, load rejection of 15% in 12 seconds, Mode selector switch in the "Steam Pressure" mode.

b. 4/4 throttle valves shut or 2/3 ASO low pressures <45 psig, load rejection of 15% in 120 seconds, Mode selector switch in the "Steam Pressure" mode.

c. 4/4 throttle valves shut and 2/3 ASO low pressures <45 psig, load rejection of 15% in 120 seconds, Mode selector switch in the "Tavg " mode.

d. 4/4 throttle valves shut or 3/3 ASO low pressures <45 psig, load rejection of 15% in 1200 seconds, Mode selector switch "Reset".

QUESTION: 063 (1.00)

Given the following conditions:

- The plant is operating at 100% power.
- The steam pressure controller, PC-464A, is set at 510 psig.
- The steam dump mode selector switch is in the "Tavg" position.

WHAT effect will placing the mode selector switch in the" Steam Pressure" position have?

a. Have no effect on the plant because the controller setting is well below normal operating steam pressure.

b. Open the steam dumps to about 50% because the controller setting is about 50% of normal setting of 1005 psig.

c. Arm the steam dumps but controller PC-464A will prevent them from opening.

d. Open all steam dump valves fully and cause a reactor trip.

QUESTION: 064 (1.00)

HOW is repositioning of the Fuel Building Bridge Crane at less than 2 feet-per-minute (fpm) accomplished?

a. By a manually operated cable winch.

b. By actuating the bridge interlock bypass switch.

c. By depressing the bridge push button lightly.

d. Not possible, the design does not permit operation at less than 2 fpm.

QUESTION: 065 (1.00)

Given the following conditions:

- Spurious reactor trip has occurred
- E-0, immediate actions are being performed
- Component Cooling water is lost due to a supply header pipe rupture

What operator ACTIONS are required as per 1-AP-15, Loss of Component Cooling?

a. Secure all charging pumps and terminated letdown.

b. Stop RCPs if charging pumps are running, or RCS subcooling is greater than 25 F.

c. Stop RCPs and manually initiate safety injection.

d. Stop RCPs if motor temperature exceeds 195 F. or pump bearing temperature exceeds 225 F.

QUESTION: 066 (1.00)

Concerning the Component Cooling Water system which ONE (1) of the following statements describes the automatic functions that occur as a result of an ESFAS due to a HI-HI containment pressure?

a. Phase "A" isolation closes containment penetration CCW valves and trips the running CCW pumps.

b. Phase "B" isolation closes containment penetration CCW valves and trips the running CCW pumps.

c. Phase "B" isolation closes containment penetration CCW valves running CCW pumps are not affected.

d. Phase "A" isolation closes containment penetration CCW valves running CCW pumps are not affected.

QUESTION: 067 (1.00)

WHAT ensures positive isolation of Service Water flow to the Component Cooling water heat exchangers during a CDA?

a. Redundant valves in series.

b. Redundant valves in parallel.

c. Redundant heat exchangers in parallel.

d. Redundant heat pumps in parallel.

QUESTION: 068 (1.00)

Which ONE (1) of the following statements correctly describes the basis for reducing THERMAL POWER when a control bank rod is stuck at a position greater than 12 steps from the remaining rods in the control bank?

a. Maintenance of power distribution limits, minimum shutdown margin, and limit potential effects on associated accident analysis.

b. Design core thermal criteria is met, limit peak linear power density during Condition I events and limits on peak local power density.

c. Maintenance of core thermal limits, limit peak linear power density and limit potential effects on associated accident analysis.

d. Design core power distribution limits are met and peak fuel clad temperature will not exceed the 2200 limit.

QUESTION: 069 (1.00)

Given the following conditions:

- The plant is shutdown.
- Tavg is 545 F
- Loop 2 RCP is out of service.
- Rod drop times are being tested.

Why are maximum plant operating power limits reduced (In accordance with Technical Specifications) when rod drop times tested under these conditions?

a. Reactor trip times may be slower than accident analysis calculations due to reduced RCS flow.

b. Accident analysis requires rod drop times to be performed with three RCPs operating.

c. Accident analysis assumes rod drop times with all RCPs operating and Tavg greater than 500 F.

d. Accident analysis assumes rod drop times with all RCPs operating and Tavg less than 500 F.

QUESTION: 070 (1.00)

What is the REASON for maintaining the control banks above the setpoint for the "ROD BANK (A/B/C/D) LO/LO-LO LIMIT" alarm?

a. Insures the maintenance of acceptable power distribution limits, maintains minimum shutdown margin, and limits the potential effects of rod misalignment on the associated accident analysis.

b. Insures adequate shutdown margin, maintain acceptable core thermal limits and limit the potential effects of rod misalignment.

c. Insures DNBR does not exceed its minimum limit during normal and short term transients, limit fission gas release, fuel pellet temperature & cladding mechanical properties to within design criteria.

d. Insures additional restrictions necessary to meet the original design criteria of rod insertion limits are not required to be implemented .

QUESTION: 071 (1.00)

During natural circulation conditions with the ERF computer not available, Abnormal Procedure 1-AP-10.1, "Natural Circulation Verification" Attachment 2, requires the operator to monitor the following parameters.

- RCS subcooling base on Core Exit TCs - GREATER THAN 30 F
- Steam Pressure - STABLE OR DECREASING
- RCS Hot Leg Temperature - STABLE OR DECREASING
- Core Exit TCs - STABLE OR DECREASING
- RCS Cold Leg temperature - AT SATURATION TEMPERATURE FOR SG PRESSURE

What CONDITION is indicated by a rapid decrease in steam generator pressures while steaming during natural circulation conditions?

a. Natural circulation flow has started.

b. Natural circulation flow has degraded to two-phase flow.

c. Natural circulation flow has increased due to two-phase flow .

d. Natural circulation flow has stopped.

QUESTION: 072 (1.00)

1-ECA-0.0 "Loss of All AC Power" step 16 directs the operator to depressurize intact SG(s) to 145 psig but not less than 120 psig. What is the BASIS for these limits?

a. Upper limit of 145 psig minimizes voiding in the core, lower limit of 120 psig maintains a density difference (delta p) in the RCS.

b. Upper limit of 145 psig minimizes loss of RCS inventory, lower limit of 120 psig prevents the injection of SI accumulator nitrogen into the RCS.

c. Upper limit of 145 psig minimizes the injection of accumulator nitrogen into the RCS, lower limit of 120 psig prevents voiding in the core.

d. Upper limit of 145 psig minimizes loss of RCS inventory, lower limit of 120 psig provides an adequate steam pressure head to the SG PORVs for cooldown.

QUESTION: 073 (1.00)

WHICH ONE (1) of the following actions is an IMMEDIATE operator ACTION for 1-ECA-0.0, "Loss of All AC Power"?

a. Place CCW pump switches in P-T-L (pull to lock).

b. Check if Letdown Isolation Valves are closed.

c. Energize AC Emergency bus with Emergency Diesel Generator.

d. Check main steamline isolation and bypass valves closed.

QUESTION: 074 (1.00)

1-AP-20, Operation from the Auxiliary Shutdown Panel, directs operators to trip the reactor and verify all rod bottom lights lit. IF ONE (1) control rod failed to fully insert, WHICH ONE (1) of the following actions is performed?

a. Emergency Borate until boron concentration is > 590 ppm.

b. Wait 15 hours and Emergency Borate for 21 minutes if the control room is still uninhabitable.

c. Calculate for a 930 ppm boron concentration and borate/dilute from the Auxiliary Shutdown Panel.

d. Emergency Borate from the Auxiliary Shutdown Panel for 21 minutes.

QUESTION: 075 (1.00)

Which ONE (1) of the following conditions requires the implementation of 1-AP-20, "Operation from the Auxiliary Shutdown Panel" when control room evacuation is necessary?

a. Smoke in the Control Room.

b. High Radiation in the Control Room.

c. Loss of Control Room air conditioning.

d. Loss of Emergency Switchgear Room air conditioning.

QUESTION: 076 (1.00)

The control room is being evacuated due to a large fire in the boron recovery panel. In accordance with 1-AP-50.1, Control Room Fire, where must all personnel proceed to, once the control room is evacuated? (Assume all personnel have been notified.)

a. Auxiliary Shutdown Panel

b. TSC Shift Supervisor's Office

c. TSC Fan Room

d. Emergency Switchgear Room by door to Control Room stairs

QUESTION: 077 (1.00)

WHICH ONE (1) of the following correctly describes the result of a loss of electrical power to RCP-1A Motor Component Cooling Water Supply Isolation Valve, TV-106-A?

a. TV-106-A will fail open causing condensation to develop in the stator.

b. TV-106-A will fail open causing a high temperature condition to develop in bearing lubrication system.

c. TV-106-A will fail shut causing a high temperature condition to develop in stator and motor bearing lubrication system.

d. TV-106-A will fail shut causing a high temperature condition to develop in pump bearing lubrication system.

QUESTION: 078 (1.00)

The reactor is operating at 100% power. In response to annunciator "RCP 1A-B-C SEAL LEAK LO FLOW", operators were unable to open the 1-CH-314 throttle valve in effort to increase seal injection flow to the 1B RCP. Then, annunciator "RCP 1A-B-C SEAL LEAK HI FLOW" is actuated and a check of the flow indication shows excessive leak-off for 1B RCP. SELECT the choice that lists the correct order of the following actions in accordance with 1-AP-33.

1. Shut seal leak-off valves HCV-1303A/C.
2. Ramp-down Unit at a rate greater than or equal to 3% per minute.
3. Manually trip the reactor then, stop the B RCP.
4. Verify Unit less than 25% power.

a. 2, 4, 3, 1

b. 4, 2, 3, 1

c. 1, 2, 4, 3

d. 1, 4, 2, 3

QUESTION: 079 (1.00)

WHICH ONE (1) of the following conditions requires an RCP to be stopped immediately?

a. Seal injection water increases to 130 F with RCS at 385 F.

b. Pump bearing temperature increases to 220 F.

c. Motor lower bearing temperature increases to 200 F.

d. RCP A seal leak-off temperature increases to 220 F.

QUESTION: 080 (1.00)

Due to an inadvertent RCP trip the reactor was tripped from a low operating power, which one of the following is a prerequisite for re-starting the RCP?

a. Oil lift pump has been operated for one (1) minute.

b. Reactor coolant system pressure is greater than 215 psig.

c. No.1 seal leakoff is above .2 gpm flow and delta-P above 200 psid.

d. The delta-P across No. 1 seal is greater than 150 psid.

QUESTION: 081 (1.00)

1-AP-16 directs the performance of 1-PT-52.2, Reactor Coolant System Leak Rate. What factors are used to determine the UNIDENTIFIED LEAKAGE rate?

a. RCS volume/temperature correction factor, pressurizer level, PDTT level, accumulator inleakage and the calculated identified leak rate.

b. RCS volume/temperature correction factor, pressurizer level, VCT level, and the calculated identified leak rate.

c. RCS volume/temperature correction factor, pressurizer level, PDTT level, and the calculated and other identified leakage.

d. RCS volume/temperature correction factor, pressurizer level, VCT level, accumulator inleakage, and the calculated identified leak rate.

QUESTION: 082 (1.00)

With voiding present in the core and no indicated RCS subcooling, what is the core exit thermo-couple temperature at which inadequate core cooling FIRST expected?

a. 700 F

b. 800 F

c. 900 F

d. 1000 F

QUESTION: 083 (1.00)

WHICH ONE (1) of the following BASES applies to the reactor coolant pump tripping criteria in response to a small break LOCA?

a. Maximizes two phase flow.

b. Faster transition to total single phase flow.

c. Reduces overall plant loads.

d. Prevents exceeding 10-CFR-100 criteria.

QUESTION: 084 (1.00)

WHICH ONE (1) of the following actions should be performed first upon entering 1-FR-S.1?

a. Reactor trip by tripping the Rod Drive MG breakers 14B1-4 and 14C2-12.

b. Actuating the mechanical turbine trip lever locally at the main turbine.

c. Control room operation of the Boric Acid Transfer Pump and Emergency Borate Valve.

d. Local operation of 52/RTA, 52/RTB, 52/BYA and 52/BYB reactor trip breakers.

QUESTION: 085 (1.00)

The total capacity of the VCT is 2,222 gallons. At what VCT level will the charging pump suction valves from the RWST open?

a. 20 %

b. 15 %

c. 10 %

d. 5 %

QUESTION: 086 (1.00)

Reactor power is 6% during a shutdown when intermediate range channel N-36 fails high. WHICH ONE (1) of the following statements describes how this failure affects the reactor shutdown and subsequent operation of the Nuclear Instrumentation system?

a. The reactor will trip on high IR flux, and source range NI's will re-energize when N-35 decreases to the proper setpoint.

b. The reactor will trip on high IR flux, and source range NI's will have to be manually re-energized.

c. The reactor will not trip, and source range NI's will re-energize when N-35 decreases to the proper setpoint.

d. The reactor will not trip, and source range NI's will have to be manually re-energized.

QUESTION: 087 (1.00)

While operating within normal plant parameters the pressurizer master pressure controller's output failed high and caused alarm annunciators "PRZ HI - LO PRESS" and "PRESSURIZER SAFETY VALVE OR PORV OPEN" to actuate. WHICH ONE (1) of the following actions is the appropriate immediate action to take in accordance with 1-AP-44, Loss of Reactor Coolant System Pressure?

a. Trip the reactor and enter E-0.

b. Take manual control of the master pressure controller.

c. Manually de-energize all pressurizer heaters.

d. Manually close PORV, PCV-1456.

QUESTION: 088 (1.00)

The plant is operating at 100% power in a normal operating line-up when the Spray Valve Controller, PC-1444G, fails (assume no operator action). Which of the following are indications of this failure?

1. Pressurizer pressure decrease.
2. Charging and letdown flow decrease.
3. Proportional and backup heaters energized.
4. Pressurizer low pressure alarm.
5. Over-Temperature delta-T runback and/or reactor trip.
6. Low pressure safety injection.
7. Containment pressure increase.

a. 1, 2, 3, 4, 5, 6

b. 1, 3, 4, 5, 6

c. 1, 2, 3, 4, 6, 7

d. 1, 3, 4, 5, 7

QUESTION: 089 (1.00)

Assume a pressurizer spray line has ruptured and raises containment temperature from 100 F to 180 F. How will indicated pressuizer level compare to actual pressurizer level?

a. Indicated level will be lower than acutal level.

b. Indicated level will be equal to actual level.

c. Indicated level will decrease to zero.

d. Indicated level will be higher than acutal level.

QUESTION: 090 (1.00)

What is the primary OBJECTIVE of early diagnosis and isolation of a ruptured steam generator.

a. To prevent or mitigate the release of radioactive contamination to the environment.

b. To prevent or mitigate the loss of reactor coolant.

c. To permit the termination of SI before the pressurizer is filled solid.

d. To stop feeding the faulted SG and permit the rapid cooldown to a cold shutdown condition.

QUESTION: 091 (1.00)

Step 9 of 1-E-3, "Steam Generator Tube Rupture" directs the operator to align the condenser air ejector discharge to containment and remove air ejector radiation monitor instrument fuses. Which ONE (1) of the following statements is the BASES for this step?

a. To permit placing the COND AIR EJECTOR DIVERT CONT SI RESET switches to "Reset".

b. To maintain condenser vacuum and minimize activity releases to the environment.

c. To maintain an negative pressure in containment to minimize activity releases to the environment.

d. To maintain condenser vacuum and direct the ejector discharge to an area where high range radiation monitors are available.

QUESTION: 092 (1.00)

A steam generator tube rupture is in progress. E-3, Steam Generator Tube Rupture is being performed. The terry turbine steam supply isolation valve(s) for _____[1]_______ should be ______[2]_______. (Select the choice that correctly fills in the blanks.

a. [1] the non-ruptured steam generators ... [2] shut to prevent loss of inventory to atmosphere.

b. [1] the non-ruptured steam generators ... [2] shut to prevent excessive RCS cooldown.

c. [1] the ruptured steam generator ... [2] shut to prevent excessive RCS cooldown.

d. [1] the ruptured steam generator ... [2] shut to prevent activity release to atmosphere.

QUESTION: 093 (1.00)

RHR is being placed in service for cooldown for refueling, MOV-1700 will not open to start the warmup of the RHR system, what actions can be taken to open the valve.

a. Decrease letdown pressure by adjusting PCV-1145.

b. Decrease RCS pressure by adjusting PZR heaters/spray.

c. Increase letdown pressure by adjusting PCV-1145.

d. Increase RCS pressure by adjusting PZR heaters/spray.

QUESTION: 094 (1.00)

Given the following conditions:

- Solid plant operations.
- RHR system and CVCS purification in service

WHERE is excess RCS inventory routed?

a. The PRT via pressurizer PORVs.

b. The Gas Stripper via CVCS.

c. The RWST via the RHR system.

d. The CDT's via the CVCS system.

QUESTION: 095 (1.00)

A LOCA is in progress, SI has initiated, RCS pressure is 130 psig, RWST level is 20 %. The LHSI pumps are taking a suction on ______[1]_____ because the _____[2]_____. (Select the ONE (1) choice the correctly fills in the blanks.)

a. [1] Containment sump and discharging to the RWST . . . [2] RWST is below the Lo-Lo level set point and RCS pressure is above the LHSI pump shutoff head.

b. [1] RWST and discharging to the RCS . . . [2] RCS pressure is below the LHSI pump shutoff head and SI has entered the recirculation mode.

c. [1] Containment sump and discharging to the suction of HHSI pump . . [2] RWST is below the Lo-Lo level set point and the SI has entered the recirculation mode.

d. [1] RWST and recircing to the RWST . . . [2] SI has not entered the recirculation mode due to RCS pressure being above the LHSI pump shutoff head.

QUESTION: 096 (1.00)

According to 1-E-0, Reactor Trip or Safety Injection, which ONE (1) of the following conditions would require entry into 1-ES-1.1, SI Termination?

	RCS PRESS	SUBCOOLING	PZR LEVEL	S/G LVL	TOTAL AFW FLOW
a.	Increasing	33 F	~16%	~12%	~355 gpm
b.	Increasing	34 F	~ 7%	~11%	~335 gpm
c.	Decreasing	31 F	~18%	~12%	~345 gpm
d.	Stable	40 F	~15%	~9%	~300 gpm

QUESTION: 097 (1.00)

Given the following conditions:

- The reactor has automatically tripped.
- The reactor trip breakers have opened.
- All IRPIs indicate zero.
- All rod bottom lights are de-energized.
- Neutror flux has leveled off at 6%.

WHAT are the reactor operator's required actions?

a. Enter E-0 and trip the rod drive MG's.

b. Enter E-0 and trip RCPs and the rod drive MG's.

c. Enter FR-S.1 drive rods in and emergency borate.

d. Enter E-0 transition to FR-S.1 drive rods in and e[illegible]rgency borate.

QUESTION: 098 (1.00)

If a fuel handling accident occurs, which one of the following is an immediate action in accordance with AP-30?

a. Verifying containment isolation has taken place.

b. Verify the Control Room ventilation system has actuated.

c. Notify Fuel Resources Group.

d. Notify Health Physics Department.

QUESTION: 099 (0.00)

DELETED

QUESTION: 100 (1.00)

During loss of all AC power conditions RCS heat is removed by which ONE (1) of the following?

a. Forced RCS convection flow through the core.

b. Natural RCS convection flow through the core.

c. Natural RCS conduction flow through the core.

d. Forced RCS conduction flow through the core.

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

ANSWER: 001 (1.00)

a.

REFERENCE:

NORTH ANNA: Virginia Power, Nuclear Power Station GENERAL EMPLOYEE TRAINING, Page 5.
194001K116 ..(KA's)

ANSWER: 002 (1.00)

b.

REFERENCE:

NORTH ANNA: Operating Procedure 1-OP-23.2, para. 3.2 and NOTE of para. 4.1.5
194001K115 ..(KA's)

ANSWER: 003 (1.00)

d.

REFERENCE:

NORTH ANNA: Virginia Power, Nuclear Power Station, General Employee Training, page 10.
194001K114 ..(KA's)

ANSWER: 004 (1.00)

b.

REFERENCE:

NORTH ANNA: Virginia Power, Nuclear Power Station General Employee Training, page 11;
ADM-20.10, Confined Area Entry Procedure, Section 3.3
194001K111 ..(KA's)

ANSWER: 005 (1.00)

b.

REFERENCE:

NORTH ANNA: ADM-20.30, Reportable Hazardous Material Spill Reporting, Containment and Cleanup Procedure, page 4.
194001K110 ..(KA's)

ANSWER: 006 (1.00)

a.

REFERENCE:

NORTH ANNA: VPAP-1401, Conduct of Operations, Section 6.1.16
194001A112 ..(KA's)

ANSWER: 007 (1.00)

d.

REFERENCE:

NORTH ANNA: ADM-19.1, Operations Records Administration, Section 5.1.1.g
194001A106 ..(KA's)

ANSWER: 008 (1.00)

b.

REFERENCE:

NORTH ANNA: Operations Standards, Tab P, Section 1.
194001A102 ..(KA's)

ANSWER: 009 (1.00)

b.

REFERENCE:

NORTH ANNA: VPAP-0501, Procedure Administrative Control Program, Section 6.7.1
194001A101 ..(KA's)

ANSWER: 010 (1.00)

c.

REFERENCE:

NORTH ANNA: Virginia Power, Nuclear Power Station, General Employee Training, page 18.
194001K105 ..(KA's)

ANSWER: 011 (1.00)

d.

REFERENCE:

NORTH ANNA: Virginia Power, Nuclear Power Station, General Employee Training, pages 103 & 104.
194001K104 ..(KA's)

ANSWER: 012 (1.00)

c.

REFERENCE:

NORTH ANNA: VPAP-1405, Independent Verification, Section 6.1.6.b.
194001K101 ..(KA's)

ANSWER: 013 (1.00)

d.

REFERENCE:

NORTH ANNA: 1-AP-42, Loss of PRODAC-250 computer
194001A115 ..(KA's)

ANSWER: 014 (1.00)

c.

REFERENCE:

NORTH ANNA: Technical Specifications, Sections: 2.1.1 and 2.2.1
015020K509 ..(KA's)

ANSWER: 015 (0.00)

DELETED

REFERENCE:

DELETED
015020K507 ..(KA's)

ANSWER: 016 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP-91.1, Engineered Safety Features, Section 2.
013000A101 ..(KA's)

ANSWER: 017 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP-91.1, Engineered Safety Features, Section 2, Section Objective B, pages 2.7 and 2.8.
013000A102 ..(KA's)

ANSWER: 018 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP-89.1, Main Steam System.
013000A104 ..(KA's)

ANSWER: 019 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP-88.1, Reactor Coolant System, Section 6, Instrumentation and Controls, para. C.6.a.
001000A101 ..(KA's)

ANSWER: 020 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 93.5, Rod Control system, Section 2, Para. A.

001000A102 ..(KA's)

ANSWER: 021 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 93.5 Rod Control, Section 2, para. A.2.b.(2); E.2.(3)
NCRODP - 89.1, Section 1, para. A.
001000A103 ..(KA's)

ANSWER: 022 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 89.4, Feedwater Systems, Section 2, Auxiliary Feedwater system, para. A.
061000K502 ..(KA's)

ANSWER: 023 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 89.4, Feedwater Systems, Section 2, Auxiliary Feedwater System, para. B.4.
061000K501 ..(KA's)

ANSWER: 024 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 89.4, Feedwater Systems, Section 2, Auxiliary Feedwater System,
059000A201 ..(KA's)

ANSWER: 025 (1.00)

a.

REFERENCE:

NORTH ANNA: EPIP - 1.05, Response to General Emergency.
017020K503 ..(KA's)

ANSWER: 026 (1.00)

d. (1.00)

REFERENCE:

NORTH ANNA:
068000A204 ..(KA's)

ANSWER: 027 (1.00)

a. (1.00)

REFERENCE:

NORTH ANNA: NCRODP - 92.4 Gaseous Waste, VA System Handout, H.1.5
068000A302 ..(KA's)

ANSWER: 028 (1.00)

a.

REFERENCE:

NORTH ANNA: Drawing No. 11715 - 087C, Sheet 3 of 4
Operating procedure 1-OP-22.11.
068000A402 ..(KA's)

ANSWER: 029 (1.00)

c.

REFERENCE:

NORTH ANNA: Drawing No. 11715 - 087C, Sheet 3 of 4
068000A403 ..(KA's)

ANSWER: 030 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP _ 88.3, Chemical Volume and Control System, Section 1, Charging and Letdown System, LER 87-002-000, Charging Pump Discharge Check Valve Hangs Open Due to Insoluble Granular Type Substance in Hanger Bracket Bushing. (March 23, 1987), page 1.37
004020K607 ..(KA's)

ANSWER: 031 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP 88.3, Section 1, Chemical Volume and Control System, Section 1, Charging and Letdown System, C.1.h., page 1.11
004020K603 ..(KA's)

ANSWER: 032 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP 88.3, Section 1, Chemical Volume and Control System, Section 1, Charging and Letdown System, page 1.20
004020K602 ..(KA's)

ANSWER: 033 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP 88.3, Section 1, Chemical Volume and Control System, Section 1, Charging and Letdown System
Dwg. NA-DW-108D014 sheet 4 of 17, Pressurizer Level Control and Protection Block Diagram.
004020K511 ..(KA's)

ANSWER: 034 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 93.1, Radiation Monitoring, pages 1.25 thru - 1.28
072000K401 ..(KA's)

ANSWER: 035 (1.00)

b.

REFERENCE:

NORTH ANNA: 1-AP-5.1, Radiation Monitoring System, Section 6.8.3.
072000K402 ..(KA's)

ANSWER: 036 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 93.1, Radiation Monitoring, Section 1, para, A.1.b, page 1.3
072000K403 ..(KA's)

ANSWER: 037 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 92.1, Fire Protection System, Page 1.84
086000K503 ..(KA's)

ANSWER: 038 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 92.1, Fire Protection System, Page 1.80 and 1.82,
086000K406 ..(KA's)

ANSWER: 039 (1.00)

c.

REFERENCE:

NORTH ANNA: NAPS-MRC-21, Malfunction Cause and Effects, page 253.
011000K605 ..(KA's)

ANSWER: 040 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP 93.8; Section 2, Pressurizer Level control and Protection, page 2.11
011000K604 ..(KA's)

ANSWER: 041 (0.00)

DELETED

REFERENCE:

DELETED
016000K312 ..(KA's)

ANSWER: 042 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 93.10, Reactor Protection, Section 1, Reactor Protection Trips and Interlocks, page 1.18.
012000K406 ..(KA's)

ANSWER: 043 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 93.10, Reactor Protection, Section 1, Reactor Protection Trips and Interlocks, Section D. Reactor Trip Signals.
012000K402 ..(KA's)

ANSWER: 044 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP 91.1, Engineered Safety Features, Section 2, page 2.20
Emergency Procedure 1-E-1.
006050K402 ..(KA's)

ANSWER: 045 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 91.2 Containment and Containment Systems, page 1.28
006050K403 ..(KA's)

ANSWER: 046 (1.00)

b.

REFERENCE:

NORTH ANNA: Dwg. NA-Dw-5655D33, Sheet 8 of 16, Functional Diagrams of Safeguard Actuation Signals, Note 5.
006050K401 ..(KA's)

ANSWER: 047 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP 93.1 Radiation Monitoring, Section 1, page 1.27
NCRODP 91.2 Containment and Containment Systems, page 1.14.
029000K102 ..(KA's)

ANSWER: 048 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 90.4, Section 2, Emergency Diesel Operations, page 2.31
062000K102 ..(KA's)

ANSWER: 049 (1.00)

b.

REFERENCE:

NORTH ANNA: 1-AP-10.12, Restoration of DC Buses, Section 6.2; Tech. Spec. 3.8.2.3, D.C. Distribution - Operations.
062000K103 ..(KA's)

ANSWER: 050 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 90.1 Basic Electrical Distribution, Section 1, Switchyard Electrical Distribution System, Section Objectives A. and B.
062000K104 ..(KA's)

ANSWER: 051 (1.00)

d.

REFERENCE:

NORTH ANNA: 1-OP-1.1, Unit Start-up from a Mode 5 with RCS Drained and at 140 degrees or less To a Mode 5 at Less than 200 degrees with a Bubble in the Pressurizer
Tech. Spec. 3.4.9.3 Reactor Coolant System Overpressure Protection
010000A104 ..(KA's)

ANSWER: 052 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 93.8, Objective A, page 1.4
010000A106 ..(KA's)

ANSWER: 053 (1.00)

c.

REFERENCE:

NORTH ANNA: 1-OP-5.2, Reactor Coolant Pump Start-up and Shutdown; 1-OP-1.1, Unit Start-up from a Mode 5 with RCS Drained and at 140 degrees or less To a Mode 5 at Less than 200 degrees with a Bubble in the Pressurizer, page 8 of 11; NCRODP - 94.2 Objective I.
010000A108 ..(KA's)

ANSWER: 054 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 90.4 Emergency Diesel Generator, Section 1, Jacket Cooling System, page 1.30; Transparency T-1.14, Jacket Cooling System
064000A201 ..(KA's)

ANSWER: 055 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 90.4, Emergency Diesel Generators, Section II, Emergency Diesel Operations, Section Objective B. page 2.2
064000A401 ..(KA's)

ANSWER: 056 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 90.4, Emergency Diesel Generators, Section I, Emergency Diesel Operations, Section Objective D. page 1.2
064000A402 ..(KA's)

ANSWER: 057 (1.00)

a.

REFERENCE:

NORTH ANNA: 1-AP-52, Loss of Refueling Cavity During Refueling. Operations Standards, Use of Procedures, Section 7.A, Use of Abnormal Procedures.
034000G015 ..(KA's)

ANSWER: 058 (1.00)

a.

REFERENCE:

NORTH ANNA: 1-OP-14.1, Residual Heat Removal System, Section 5.3
NCRODP - 88.2, Residual Heat Removal System, Section 2, RHR System Operation, Section Objective C.
005000K509 ..(KA's)

ANSWER: 059 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 88.2, Residual Heat Removal System, Section 2, RHR System Operation, Section Objective B; and Presentation Section A, page 2.5
005000K502 ..(KA's)

ANSWER: 060 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 92.6, Component Cooling System, Section II Instrumentation and Control, page 2.31
008000A104 ..(KA's)

ANSWER: 061 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 93.11, Steam Dumps, Section Objective C, page 1.12
041020K105 ..(KA's)

ANSWER: 062 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 93.11, Steam Dumps, Section 1 Steam Dump Control and Protection, Section Objective E., page 1.8
041020A102 ..(KA's)

ANSWER: 063 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 93.11, Steam Dumps, Section 1, Steam Dump Control and Protection, Section Objective C., Section B.4.a
NCRODP - 93.10, Reactor Protection, Section 1, Reactor Protection Trips and Interlocks, Section Objective D.
041000G015 . (KA's)

ANSWER: 064 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 92.9, Fuel Handling and Refueling Operations, Section III, Fuel Handling Equipment, Section, C.1.h. page 3.20
034000G009 ..(KA's)

ANSWER: 065 (1.00)

d.

REFERENCE:

NORTH ANNA: 1-AP-15, Loss of Component Cooling, Step 5.3
000026K303 ..(KA's)

ANSWER: 066 (1.00)

b.

REFERENCE:

NORTH ANNA: Tech. Spec. 3/4.3.2, Engineered Safety Feature Actuation System Instrumentation, Tech. Spec. 3/4.6.3, Containment Isolation Valves, Tech. Spec. Bases, 3/4.6.3
000026K302 ..(KA's)

ANSWER: 067 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 92.2, Service Water System, page 1.5
000026K301 ..(KA's)

ANSWER: 068 (1.00)

a.

REFERENCE:

NORTH ANNA: Tech Spec. Bases, 3/4.1.3
000005K305 ..(KA's)

ANSWER: 069 (1.00)

c.

REFERENCE:

NORTH ANNA: Tech. Spec. Bases, 3/4.1.3
000005K304 ..(KA's)

ANSWER: 070 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 93.5, Rod Control and Rod Position Indication System, Section 2, Rod Control, page 2.76 & 2.77; Tech. Spec. 3/4.1.3 Movable Control Assemblies Bases.
000005K302 ..(KA's)

ANSWER: 071 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 95.2, Mitigating Core Damage, Section I, Post Accident Cooling, Section C.7; 1-AP_10.1, Natural Circulation Verification, Attach. 2.
000055K102 ..(KA's)

ANSWER: 072 (1.00)

b.

REFERENCE:

NORTH ANNA: 1-ECA-0.0 "Loss of All AC Power" Caution: and Note: preceding step 16.
000055K302 ..(KA's)

ANSWER: 073 (1.00)

b.

REFERENCE:

NORTH ANNA: 1-ECA-0.0, "Loss of All AC Power", Step [3]
000055G010 ..(KA's)

ANSWER: 074 (1.00)

d.

REFERENCE:

NORTH ANNA: 1-AP-20, Operation from the Auxiliary Shutdown Panel, Step 5.2.
000068K317 ..(KA's)

ANSWER: 075 (1.00)

a.

REFERENCE:

NORTH ANNA: 1-AP-20, Operation from the Auxiliary Shutdown Panel, Step 3.0
000068G011 ..(KA's)

ANSWER: 076 (1.00)

c.

REFERENCE:

NORTH ANNA: 1-AP-50.1 Control Room Fire, page 3 of 18
000068K318 ..(KA's)

ANSWER: 077 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 92.6 Component Cooling, Section 2 Instrumentation and Controls, page 2.19; NCRODP - 88.8, Reactor Coolant System, Section 3, Reactor Coolant Pumps, page 3.18
000015K302 ..(KA's)

ANSWER: 078 (1.00)

c.

REFERENCE:

NORTH ANNA: 1-AP-33, Reactor Coolant Pump Seal Failure.
000015K303 ..(KA's)

ANSWER: 079 (1.00)

c.

REFERENCE:

NORTH ANNA: Annunciator Response, AR-1C-H4, 1-OP-5.2, Reactor Coolant Pump Start-up and Shut-down, Section 4.3.
000015A208 ..(KA's)

ANSWER: 080 (1.00)

c.

REFERENCE:

NORTH ANNA: 1-OP-5.2, Reactor Coolant Startup and Shutdown, page 7 of 24
000015K304 ..(KA's)

ANSWER: 081 (1.00)

b.

REFERENCE:

NORTH ANNA: 1-PT-52.2, Reactor Coolant System Leak Rate, Sections 4.9 thru 4.12
000009K321 ..(KA's)

ANSWER: 082 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 95.2, Section 3, Accident Response of Incore Instrumentation, Objective D., Section D.1
000009K326 ..(KA's)

ANSWER: 083 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 95.4, Emergency Operating Procedures, Section 5, Loss of Reactor or Secondary Coolant, Section Objective D., page 5.11
000009K323 ..(KA's)

ANSW (1.00)

REFERENCE:

NORTH ANNA: NRCODP - 95.6, Functional Restoration Procedures, Section I, FR-S.1 Response to Nuclear Power Generation ATWS, Section Objective C., Section C.2, page 1.4
000029A101 ..(KA's)

ANSWER: 085 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 88.3, Chemical and Volume Control System, Section II, Makeup System, Section Objective D., page 2.16
000029A102 ..(KA's)

ANSWER: 086 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 93.2, Excore NIS, Section 2, Instrument Operation, Section Objectives D. and K.
000033A208 ..(KA's)

ANSWER: 087 (1.00)

b.

REFERENCE:

NORTH ANNA: 1-AP-44, Loss of Reactor Coolant System Pressure
000008A101 ..(KA's)

ANSWER: 088 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 93.8, Pressurizer Pressure and Level Protection and Control, Section I, Pressurizer Pressure Protection and Control, Section Objective G.
1-AP-44, Loss of Reactor Coolant System Pressure
(Simulator: Malfunction Cause & Effects, Pressurizer Spray Valve Stuck Open)
000008A219 ..(KA's)

ANSWER: 089 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 93.8, Presurizer Pressur and Level Protection and Control, Section II, Objective B. page 2.6 and 2.8
000008A226 ..(KA's)

ANSWER: 090 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 95.4, Emergency Operating Procedures Section 11, Steam Generator Tube Rupture, Section Objective A, page 11.2
000037A101 ..(KA's)

ANSWER: 091 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 95.4, Emergency Operating procedures, Section 11, Steam Generator Tube Rupture, Section Objective E, Step 9, page 11.8
000037K307 ..(KA's)

ANSWER: 092 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 95.4, Emergency Operating procedures, Section 11, Steam Generator Tube Rupture, Section Objective E, Step 3, page 11.6
000037G012 ..(KA's)

ANSWER: 093 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 88.2, Residual Heat Removal System, Section 2, RHR System Operation, Section Objective, A.3 and page 2.4

350 F - 140 F = 210 F delta T
210 F / 16 hours = 13.125 F/hr cooldown rate (2 RHR pumps and 2 coolers)

With only 1 RHR pump available cooldown rate = 6.5625 F/hr

325 F - 135 F = 190 F delta T
190 F / 6.5625 F/hr = 28.95238 hours
000025A101 ..(KA's)

ANSWER: 094 (1.00)

o.

REFERENCE:

NORTH ANNA: NCRODP - 88.2, Residual Heat Removal System, Section 2, RHR System Operation, Section Objective, F and page 2.10
000025A102 ..(KA's)

ANSWER: 095 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 91.1, Engineered Safety Features, Section 2, SI or ECCS as a Subset of the ESF Systems, Section Objective E, page 2.24
000025A103 ..(KA's)

ANSWER: 096 (1.00)

a.

REFERENCE:

NORTH ANNA: 1-E-0, Reactor Trip of Safety Injection, Step 26 and 27; 1-ES-1.1, SI Termination, Step 1.
000009A201 ..(KA's)

ANSWER: 097 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 95.4, Emergency Operating Procedures, Section 1, E-0, Reactor Trip/Safety Injection, Section Objective C, page 1.4, Step 1; 1-FR-S.1, Response to Nuclear Power Generation/ATWS, immediate action steps [1] thru [4]
000007K203 ..(KA's)

ANSWER: 098 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 92.9, Fuel Handling and Refueling Operation, Objective E; 1-AP-30, Fuel Failure During Handling
000036K303 ..(KA's)

ANSWER: 099 (0.00)

DELETED

REFERENCE:

DELETED
000036K301 ..(KA's)

ANSWER: 100 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 83, Thermodynamics, Fluid Flow, and Heat Transfer; Section 9, Objective D.3, page 9.27;
000056K101 ..(KA's)

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

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)

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

001	a	b	c	d	______
002	a	b	c	d	______
003	a	b	c	d	______
004	a	b	c	d	______
005	a	b	c	d	______
006	a	b	c	d	______
007	a	b	c	d	______
008	a	b	c	d	______
009	a	b	c	d	______
010	a	b	c	d	______
011	a	b	c	d	______
012	a	b	c	d	______
013	a	b	c	d	______
014	a	b	c	d	______
015	a	b	c	d	______
016	a	b	c	d	______
017	a	b	c	d	______
018	a	b	c	d	______
019	a	b	c	d	______
020	a	b	c	d	______
021	a	b	c	d	______
022	a	b	c	d	______
023	a	b	c	d	______
024	a	b	c	d	______
025	a	b	c	d	______

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)

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

026	a	b	c	d	______
027	a	b	c	d	______
028	a	b	c	d	______
029	a	b	c	d	______
030	a	b	c	d	______
031	a	b	c	d	______
032	a	b	c	d	______
033	a	b	c	d	______
034	a	b	c	d	______
035	a	b	c	d	______
036	a	b	c	d	______
037	a	b	c	d	______
038	a	b	c	d	______
039	a	b	c	d	______
040	a	b	c	d	______
041	a	b	c	d	______
042	a	b	c	d	______
043	a	b	c	d	______
044	a	b	c	d	______
045	a	b	c	d	______
046	a	b	c	d	______
047	a	b	c	d	______
048	a	b	c	d	______
049	a	b	c	d	______
050	a	b	c	d	______

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

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

051	a	b	c	d	______
052	a	b	c	d	______
053	a	b	c	d	______
054	a	b	c	d	______
055	a	b	c	d	______
056	a	b	c	d	______
057	a	b	c	d	______
058	a	b	c	d	______
059	a	b	c	d	______
060	a	b	c	d	______
061	a	b	c	d	______
062	a	b	c	d	______
063	a	b	c	d	______
064	a	b	c	d	______
065	a	b	c	d	______
066	a	b	c	d	______
067	a	b	c	d	______
068	a	b	c	d	______
069	a	b	c	d	______
070	a	b	c	d	______
071	a	b	c	d	______
072	a	b	c	d	______
073	a	b	c	d	______
074	a	b	c	d	______
075	a	b	c	d	______

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)

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

076	a	b	c	d	____
077	a	b	c	d	____
078	a	b	c	d	____
079	a	b	c	d	____
080	a	b	c	d	____
081	a	b	c	d	____
082	a	b	c	d	____
083	a	b	c	d	____
084	a	b	c	d	____
085	a	b	c	d	____
086	a	b	c	d	____
087	a	b	c	d	____
088	a	b	c	d	____
089	a	b	c	d	____
090	a	b	c	d	____
091	a	b	c	d	____
092	a	b	c	d	____
093	a	b	c	d	____
094	a	b	c	d	____
095	a	b	c	d	____
096	a	b	c	d	____
097	a	b	c	d	____
098	a	b	c	d	____
099	a	b	c	d	____
100	a	b	c	d	____

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)

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

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

A N S W E R K E Y

001	a
002	b
003	d
004	b
005	b
006	a
007	d
008	b
009	b
010	c
011	d
012	c
013	d
014	c
015	D
016	a
017	a
018	b
019	c
020	a
021	a
022	b
023	a
024	b
025	a

A N S W E R   K E Y

026	d
027	a
028	a
029	c
030	d
031	b
032	d
033	b
034	c
035	b
036	a
037	d
038	a
039	c
040	d
041	D
042	b
043	d
044	c
045	a
046	b
047	b
048	b
049	b
050	d

A N S W E R K E Y

051	d
052	c
053	c
054	b
055	c
056	a
057	a
058	a
059	c
060	d
061	b
062	b
063	d
064	c
065	d
066	b
067	a
068	a
069	c
070	a
071	d
072	b
073	b
074	d
075	a

ANSWER KEY

076	c
077	c
078	c
079	c
080	c
081	b
082	a
083	b
084	c
085	d
086	b
087	b
088	b
089	d
090	a
091	b
092	d
093	c
094	b
095	c
096	a
097	d
098	d
099	D
100	b

A N S W E R   K E Y

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

QUESTION	VALUE	REFERENCE
001	1.00	9000001
002	1.00	9000002
003	1.00	9000003
004	1.00	9000004
005	1.00	9000005
006	1.00	9000006
007	1.00	9000007
008	1.00	9000008
009	1.00	9000009
010	1.00	9000010
011	1.00	9000011
012	1.00	9000012
013	1.00	9000013
014	1.00	9000014
015	0.00	9000015
016	1.00	9000016
017	1.00	9000017
018	1.00	9000018
019	1.00	9000019
020	1.00	9000020
021	1.00	9000021
022	1.00	9000022
023	1.00	9000023
024	1.00	9000024
025	1.00	9000025
026	1.00	9000026
027	1.00	9000027
028	1.00	9000028
029	1.00	9000029
030	1.00	9000030
031	1.00	9000031
032	1.00	9000032
033	1.00	9000033
034	1.00	9000034
035	1.00	9000035
036	1.00	9000036
037	1.00	9000037
038	1.00	9000038
039	1.00	9000039
040	1.00	9000040
041	0.00	9000041
042	1.00	9000042
043	1.00	9000043
044	1.00	9000044
045	1.00	9000045
046	1.00	9000046
047	1.00	9000047
048	1.00	9000048
049	1.00	9000049
050	1.00	9000050
051	1.00	9000051
052	1.00	9000052
053	1.00	9000053
054	1.00	9000054

QUESTION	VALUE	REFERENCE
055	1.00	9000055
056	1.00	9000056
057	1.00	9000057
058	1.00	9000058
059	1.00	9000059
060	1.00	9000060
061	1.00	9000061
062	1.00	9000062
063	1.00	9000063
064	1.00	9000064
065	1.00	9000065
066	1.00	9000066
067	1.00	9000067
068	1.00	9000068
069	1.00	9000069
070	1.00	9000070
071	1.00	9000071
072	1.00	9000072
073	1.00	9000073
074	1.00	9000074
075	1.00	9000075
076	1.00	9000076
077	1.00	9000077
078	1.00	9000078
079	1.00	9000079
080	1.00	9000080
081	1.00	9000081
082	1.00	9000082
083	1.00	9000083
084	1.00	9000084
085	1.00	9000085
086	1.00	9000086
087	1.00	9000087
088	1.00	9000088
089	1.00	9000089
090	1.00	9000090
091	1.00	9000091
092	1.00	9000092
093	1.00	9000093
094	1.00	9000094
095	1.00	9000095
096	1.00	9000096
097	1.00	9000097
098	1.00	9000098
099	0.00	9000099
100	1.00	9000100

	97.00	

	97.00	

ENCLOSURE 2-SRO

Nuclear Regulatory Commission
Operator Licensing
Examination

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date of examination.

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION
REGION 2

FACILITY: North Anna 1 & 2

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 90/09/24

CANDIDATE:

INSTRUCTIONS TO CANDIDATE:

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

NUMBER QUESTIONS	TOTAL POINTS	CANDIDATE'S POINTS	CANDIDATE'S OVERALL GRADE (%)
100	97.00		

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

Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. After the examination has been completed, you must sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination. This must be done after you complete the examination.

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

4. Use black ink or dark pencil only to facilitate legible reproductions.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

22. When you are done and have turned in your examination, leave the examination area as defined by the examiner. If you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION: 001 (1.00)

Which ONE (1) of the following is the correct sequence of actions required upon discovery of a SMALL fire in a trash can in the auxiliary building?

a. Report the fire to the control room, attempt to extinguish the fire, notify the Station Safety and Loss Prevention Department to refill the extinguisher.

b. Report the fire to the Fire Brigade, remain at the scene of the fire, advise the Fire Brigade of any personnel who are in the area, leave the area.

c. Report the fire to the control room, remain at the scene of the fire, advise the Fire Brigade of work in progress and of hazards that may be in the area.

d. Report the fire to the control room, attempt to extinguish the fire, notify the Fire Brigade to refill the extinguisher.

QUESTION: 002 (1.00)

The concentration of oxygen in the waste gas decay tank is limited to a maximum of WHAT percent by volume when the hydrogen concentration is 4% to 96%?

a. 1 %

b. 2 %

c. 3 %

d. 4 %

QUESTION: 003 (1.00)

Which one of the following is an example of an area where a "Confined Space Entry Permit" must be in effect prior to entry?

a. The containment building.

b. A Cable tunnel

c. A Switchgear cabinet.

d. The condensate storage tank.

QUESTION: 004 (1.00)

If containment pressure is 9.5 psia, which ONE (1) of the following choices correctly describes the requirements for containment entry, in accordance with ADM 20.9, Containment Entry and Exit Under Sub-atmospheric Conditions?

a. A minimum of three (3) persons on the entry team.

b. Wear breathing devices with 100% oxygen.

c. At least six hours of rest within the previous 24 hours.

d. Must consume eight ounces of water within 2 hours of entry.

QUESTION: 005 (1.00)

If the atmosphere in a confined space contains 11% chlorine gas, 20.3% oxygen, 67.5% Nitrogen and 1.2% Carbon-dioxide, which ONE (1) of the following definitions correctly defines the atmosphere in accordance with ADM-20.10, Confined Area Entry Procedure?

a. Oxygen deficient

b. Toxic or nuisance

c. Non-toxic

d. Non-hazardous

QUESTION: 006 (1.00)

In accordance with ADM-20.30, caustic spills are reported to the Shift Supervisor. What reportable hazardous material is stored in the Auxiliary Building that would constitute a caustic spill?

a. Sodium Hydroxide.

b. Ammonium Hydroxide.

c. Aluminum Sulfate.

d. Potassium Hydroxide.

QUESTION: 007 (1.00)

Which ONE (1) of the following statements correctly describes the requirements for manipulation of systems and components outside the control room that may indirectly affect unit reactivity or power level?

a. A qualified operator with the knowledge and consent of an NRC licensed operator on duty in the control room.

b. An on-duty NRC licensed operator with the knowledge and consent of the SRO on shift.

c. Operations Department personnel with the knowledge and consent of an Operations Department supervisor.

d. An NRC licensed operator with the knowledge and consent of the Supervisor Shift Operations.

QUESTION: 008 (1.00)

How is a narrative log entry made if it was inadvertently omitted during the shift?

a. Record the current time and the actual event time in the "TIME" space then, following the log entry, record the words "LATE ENTRY".

b. Record the current time in the "TIME" space then, the actual event time preceding the log entry record and the words "LATE ENTRY".

c. Record the words "LATE ENTRY" in the "TIME" space then, the actual event time the event occurred preceding the log entry.

d. Record the actual event time in the "TIME" space then, preceding the log entry record the words "LATE ENTRY".

QUESTION: 009 (1.00)

When may an operator deviate from an EMERGENCY PROCEDURE without first obtaining an approved procedure deviation?

a. When written instructions in the procedure provide for a deviation otherwise verbatim compliance is required.

b. When conditions exist that warrant deviation from the procedure to mitigate the severity of an accident in progress.

c. Anytime, based on the judgment, knowledge and experience of the licensed operator if the existing conditions require a procedure deviation.

d. When the deviation to a procedure that changes the initial conditions is handled with a "prior to use" deviation.

QUESTION: 010 (1.00)

In accordance with VPAP-0501 who is responsible for verifying that operating procedures to be used are the correct revision?

a. The cognizant supervisor.

b. The procedure user.

c. The Manager O & M.

d. The on-call SRO.

QUESTION: 011 (1.00)

What is the maximum number of visitors a licensed operator may escort in the control room?

a. 1 person

b. 3 persons

c. 5 persons

d. 10 persons

QUESTION: 012 (1.00)

Which ONE (1) of the following is responsible for maintaining dose As Low As Reasonably Achievable (ALARA)?

a. The Assistant Station Manager - Operations and Maintenance, Chairman of the ALARA committee.

b. The ALARA Coordinator.

c. The Health Physics Department.

d. The individual radiation worker.

QUESTION: 013 (1.00)

For the performance of a Work Order requiring a RWP, which ONE (1) of the following responsibilities is assigned to the control room SRO by VPAP-2102, "Station ALARA Program"?

a. Review and approve Man-Rem goals.

b. Participate in ALARA work order reviews.

c. Provide technical evaluation support.

d. Review and approve the Work Order.

QUESTION: 014 (1.00)

VPAP-1402, Control of Equipment Tag-outs and Tags, states that there is two type of danger tags, these two types of tags can be identified by color. WHAT color are these danger tags?

a. Red and Yellow

b. Both are Red

c. Red and Orange

d. Yellow and Orange

QUESTION: 015 (1.00)

Which ONE (1) of the following valve alignment verifications would be acceptable in accordance with VPAP 1405, "Independent Verification"

a. Fully close valves then reposition the valves to the desired position.

b. Fully open valves then reposition the valves to the desired position.

c. By Observation of valve position and process parameter verification.

d. Attempt to reposition open valves to open and closed valves to closed.

QUESTION: 016 (1.00)

Given the following conditions:

- A LOCA event is in progress.
- RCS leakage is determined to be between 250 and 300 gpm.
- Pressurizer level cannot be maintained with ONE (1) charging/SI pump.
- In accordance with EPIP - 1.01 EAL tables the event has been classified as an ALERT.
- The In-Plant Monitoring Team has taken surveys at the site boundary and determined the whole body dose rate to be 650 mrem/hr.

Which ONE (1) of the following choices correctly describes how the Station Emergency Manager should respond in accordance with EPIP - 1.01.

a. Continue to classify the event as an "ALERT".

b. Re-Classify the event as a "SITE AREA EMERGENCY".

c. Re-Classify the event as a "GENERAL EMERGENCY".

d. Re-Classify the event as an "UNUSUAL EVENT".

QUESTION: 017 (1.00)

Which ONE (1) of the following groups of functions does the PRODAC-250 computer perform during normal plant operations at 100% reactor power?

1. Determines AFD.
2. Calculates the value of Keff.
3. Performs daily calorimetric.
4. Determines average containment temperature.
5. Performs leak-rate calculations.
6. Determines rod height and compares it to the group position.

(Select the choice that includes ALL correct answers.)

a. 1, 2, 3, 4, 5

b. 2, 3, 4, 5, 6

c. 1, 3, 4, 5

d. 1, 3, 4, 5, 6

QUESTION: 018 (1.00)

Which ONE (1) of the following formulas correctly defines "Quadrant Power Tilt Ratio" (QPTR)?

a. PT - PB (divided by) PT + PB

b. max. upper(lower) NIS detector current (divided by) avg. upper(lower) NIS detector currents

c. (PT - PB) (divided by) 100% power

d. delta - flux (divided by) reactor power

QUESTION: 019 (1.00)

Technical Specifications state that protection is provided to prevent DNB. Which one of the following reactor trips provide this protection.

a. Power range High Flux rate

b. Over-Power Delta-T

c. Over-Temperature Delta-T

d. Low Pressure

QUESTION: 020 (0.00)

DELETED

QUESTION: 021 (1.00)

A Loss of Coolant Accident (LOCA) has occurred causing RCS pressure to ramp down at a rate of 200 psi/minute. WHICH ONE (1) of the following lists the correct order of SI?

a. Charging pumps, SI accumulators, LHSI pumps.

b. SI accumulators, HHSI pumps, Charging pumps.

c. SI accumulators, Charging pumps, LHSI pumps.

d. Charging pumps, HHSI pumps, LHSI pumps.

QUESTION: 022 (1.00)

Which ONE (1) of the following correctly describes the expected impact on containment environment when defining a Design Basis Accident (DBA) Loss of Coolant Accident (LOCA).

a. Containment pressure, temperature and humidity will increase.

b. Containment temperature will decrease, pressure and humidity will increase.

c. Containment temperature and pressure will increase and humidity will decrease.

d. Containment pressure and humidity will increase and temperature will not change.

QUESTION: 023 (1.00)

If an abrupt steam line break upstream of the main steam line isolation valves occurs, which ONE (1) of the following choices correctly describes the response of the faulted steam generator water level.

a. Level would increase due to pressure decrease then hold steady at program level.

b. Level would increase due to swell then decrease as the generator boiled dry.

c. Level would decrease due to shrink and as the generator boiled dry.

d. Level would decrease due to pressure decrease then hold steady at program level.

QUESTION: 024 (1.00)

Which ONE (1) of the following alarm set-points is provided by the Tavg PROTECTION CHANNELS?

a. Alarm at 2 F comparison of loop Tavg and auctioneered Hi Tavg.

b. Alarm at 3 F comparison of loop Tavg and the Hi Tavg signal.

c. Alarm at 4 F comparison of loop Tavg and full load Tavg.

d. Alarm at 5 F comparison of Auctioneered Hi Tavg to Tref.

QUESTION: 025 (1.00)

Given the following conditions:

Tavg = Tref
Auto Rod Control is selected

How will the rod control system respond to a 1 F STEP INCREASE in Tref?

a. The control rods remain in position.

b. The control rods step out at 8 steps per minute.

c. The control rods step out at 48 steps per minute.

d. The control rods step out at 64 steps per minute.

QUESTION: 026 (1.00)

During a continuous rod withdrawal accident, how are steam generator LEVEL and PRESSURE affected?

a. Level will increase and pressure will increase due to Tavg increase.

b. Level will increase and pressure will decrease due to Tavg increase.

c. Level will decrease and pressure will increase due to Tavg decrease.

d. Level will increase and pressure will decrease due to Tavg decrease.

QUESTION: 027 (1.00)

Given that a safety injection has occurred but, the safety injection signal has NOT been reset, HOW can an AFW pump be stopped?

a. Place the AFW pump switch in "STOP".

b. Place the AFW pump switch in "PTL".

c. Remove power from the D/F transfer bus (Unit 1).

d. Remove power from the H/J emergency bus.

QUESTION: 028 (1.00)

Which ONE (1) of the following groups correctly state the sources of heat input which are included in the design of the AFW system?

a. Steam generator residual heat, Pressurizer heater input and reactor plant startup heat.

b. Core decay heat, Reactor Coolant Pump heat input, stored heat in RCS and steam generator metal.

c. Reactor vessel heat, RCS stored heat and pressurizer stored heat.

d. Main Turbine heat, steam generator residual heat and reactor plant stored heat.

QUESTION: 029 (1.00)

WHICH one of the following statements is NOT correct in reference to the turbine driven AFW pump steam supply valves?

a. Both 1-MS-TV-111A and 111B receives a signal from train A and B solid state protection.

b. Both 1-MS-TV-111A and 111B fail open on loss of air.

c. Both 1-MS-TV-111A and 111B fail open on loss of electrical power.

d. Both 1-MS-TV-111A and 111B can independently supply enough steam to run the terry turbine. (each is capable of supplying)

QUESTION: 030 (1.00)

Which one of the following statements is correct concerning the motor driven and turbine driven auxiliary feedwater pumps?

a. The turbine driven pump can be shutdown following an SI signal only after the SI signal has been reset.

b. A motor driven pump has no time delay following bus voltage restoration for auto start except SI.

c. The turbine driven pump's steam supply valves (TV-111A/B) fail open on a loss of power and fail as-is on a loss of air (but will drift open due to steam pressure).

d. A motor driven pump cannot be shutdown following a S/G LO-LO level start until the LO-LO level condition has cleared.

QUESTION: 031 (1.00)

If a main feedwater regulating valve fails open (step change) while operating at 50% power, which ONE (1) of the following list of effects includes ALL of the expected effects if no operator action is taken? (All systems are in automatic)

1. The affected steam generator's level increases.
2. Automatic control rod withdrawal.
3. Turbine trip.
4. Main steam isolation.
5. Main feedwater isolation.
6. Reactor trip.

a. 1, 2, 3, 6

b. 1, 2, 3, 5, 6

c. 1, 2, 3, 4, 5

d. 1, 3, 4, 5

QUESTION: 032 (1.00)

1-OP-31.1, Main Feedwater System, step 5.3.8 states that the associated MFW pump discharge MOV (1-FW-MOV-150A/B/C) must be closed before attempting a start-up of a MFP. Assume no other MFPs are running.

Which ONE (1) of the following choices correctly states the reason for this action?

a. Pump start interlock prevents starting a MFP unless it's associated discharge MOV is closed and its recirc valve is in "AUTO".

b. Pump will auto start when the control switch is taken out of the P-T-L position and placed to the "AUTO AFTER STOP" position when no other MFP is running.

c. Pump start interlock prevents starting a MFP unless it's associated discharge MOV is shut and no other MFP pump is running.

d. Pump start interlock permits auto starting a MFP only when it's associated discharge MOV is closed and its recirc valve is open.

QUESTION: 033 (1.00)

Given that the highest incore thermo-couples indicate the following temperatures:

- 1210 F
- 1212 F
- 1212 F
- 1213 F
- 1214 F
- 1213 F

Which ONE (1) of the following conditions is indicated?

a. Core damage is imminent.

b. Core temperature at 100% reactor power.

c. Stuck control rod.

d. Core temperature below DNB.

QUESTION: 034 (1.00)

Select the choice that correctly matches the RADIATION MONITORS to the AUTOMATIC FUNCTIONS they perform upon a HIGH-HIGH radiation condition.

AUTOMATIC FUNCTIONS	RADIATION MONITORS
A. Closes the containment purge supply and exhaust MOV's.	1. RM-VG-103/104 "A" Vent Stack
B. Closes the containment vacuum pump discharge trip valves.	2. RM-GW-101 & 102 Process vent monitor
C. Closes the flow control valve from the waste gas decay tanks.	3. RM-VG-112/113 "B" Stack Monitor
D. Causes the containment vacuum pumps to trip off.	4. RM-RMS-159/160 Containment monitors
	5. RM-LW-111, Clarifier outlet sampler

a. A - 2, B - 1, C - 3, D - 4

b. A - 2, B - 4, C - 5, D - 4

c. A - 4, B - 2, C - 2, D - 2

d. A - 4, B - 3, C - 5, D - 4

QUESTION: 035 (1.00)

The plant is shut-down with the vessel head removed for refueling.

Which ONE (1) of the following conditions is the probable cause for an alarm actuated by radiation monitor RMS-162, Manipulator Crane.

a. Loss of letdown to the VCT.

b. Loss of refueling cavity level.

c. Incore detector movement.

d. RHR flow decreases to 3000 gpm.

QUESTION: 036 (1.00)

Which ONE (1) of the following conditions is indicated by an actuating of the control room area radiation monitor RMS-157 alarm?

a. A health hazard to control room personnel exists.

b. The control room ventilation system has isolated.

c. Notification must be made that an off-site radiation release is in progress.

d. A primary to secondary system leak has occurred.

QUESTION: 037 (1.00)

Below WHAT hydrogen percentage will the hydrogen recombiners maintain the containment atmosphere?

a. 2 %

b. 3 %

c. 4 %

d. 5 %

QUESTION: 038 (1.00)

What is the method NORMALLY used to maintain main fire loop pressure between 95 - 110 psig?

a. Jockey pump and hydropneumatic tank maintain pressure.

b. Motor driven fire pump recircs to maintain pressure.

c. Auto start of the motor driven fire pump below 95 psig.

d. Auto start of the diesel driven fire pump below 95 psig.

QUESTION: 039 (1.00)

WHICH ONE (1) of the following choices describes North Anna's fire protection system in the main control room Computer Room?

a. Water Deluge System, actuated by a heat sensor set at a predetermined temperature.

b. Fire Suppression Foam System, coats the surface and surrounding structures to extinguish the fire.

c. High Pressure CO2, Carbon dioxide displaces oxygen and leaves no residue.

d. Halon 1301 System, halon displaces oxygen and leaves no residue.

QUESTION: 040 (1.00)

Which ONE (1) of the following statements correctly describes the fire extinguishing system associated with the Fuel Oil Pump house.

a. The H.P. CO2 system is actuated by a heat detector with a temperature rate of rise exceeding 24 F/minute.

b. The Halon fire extinguishing system is actuated by either heat detectors at the charcoal filter or smoke detectors.

c. The L.P CO2 system dumps to the HVAC filters, trips fan dampers and is actuated by either heat detectors or smoke detectors.

d. The Deluge system is actuated by an electrical signal from the heat detection system.

QUESTION: 041 (1.00)

If a pressurizer safety valve sticks open and NO operator action is taken, HOW will pressurizer level respond?

a. Decreases due to loss of inventory; when system pressure reaches saturation, pressurizer level will increase.

b. Remain steady then increase as system pressure reaches saturation pressure.

c. Increases due to swell; then decreases as inventory is lost; when pressure reaches saturation, level will increase.

d. Decreases as inventory is lost, then when pressure reaches saturation pressure, level will rapidly decrease.

QUESTION: 042 (1.00)

Given the following conditions:

- Pressurizer level controller LC-459F indicates that pressurizer level is less than program level.
- Charging flow control valve FCV-1122 is ramping open in auto to increase charging flow.

Which ONE (1) of the following conditions correctly describe the functions of the Pressurizer level control system?

a. Reactor power is decreasing from 50% to 40% and pressurizer level control system is in automatic.

b. Reactor power is decreasing from 50% to 40% and pressurizer level control system is in manual.

c. Reactor power is increasing from 50% to 60% and pressurizer level control system is in manual.

d. Reactor power is increasing from 50% to 60% and pressurizer level control system is in automatic.

QUESTION: 043 (1.00)

Which ONE (1) of the following annunciator alarms indicates a failure of the flow feedback signal from charging flow detector FQ-1122?

a. CH PP SEAL/GEARBX COOLER INLET LO FLOW

b. CH PP TO REGEN HX HI-LO FLOW

c. CH PP TO REGEN HX LO PRESS

d. CH LINE FLOW LOCAL OPER

QUESTION: 044 (0.00)

DELETED

QUESTION: 045 (1.00)

Which ONE (1) of the following reactor trip signals is blocked or unblocked by the P-7 permissive?

a. Single Loop Low Coolant Flow Trip.

b. Two Loop Low Coolant Flow Trip.

c. Pressurizer High Pressure Trip.

d. Source Range Reactor Trip.

QUESTION: 046 (1.00)

Select the choice that correctly matches the REACTOR TRIP to the PROTECTION the trip provides.

REACTOR TRIP	PROTECTION
A. Pressurizer High Level	1. Protects against excessive fuel center line temperature (fuel melt).
B. Over-Temperature Delta T	2. Protection against reactivity changes too fast for temperature and pressure protective circuitry
C. Over-Power Delta T	3. Protects against DNB.
D. Neutron Flux Positive Rate	4. Prevents RCS over-pressurization
	5. Protects against loss of coolant.

a. A - 5, B - 1, C - 4, D - 3

b. A - 3, B - 1, C - 4, D - 2

c. A - 5, B - 3, C - 2, D - 4

d. A - 4, B - 3, C - 1, D - 2

QUESTION: 047 (1.00)

If RCS pressure decreases below 1275 psig during a LOCA/reactor shutdown, WHY are the charging pump recirc valves closed?

a. Recirc flow through the charging pumps below 1275 psig will cause the pumps to over heat.

b. Damage to the charging pumps is not a consideration below 1275 psig.

c. Stopping recirc flow at 1275 psig increases injection flow to the RCS.

d. Below 1275 psig the Charging Pumps begin to cavitate.

QUESTION: 048 (1.00)

If a small break in a main steam system header has raised containment pressure to 2.5 psig. How is containment isolation affected?

a. Phase A isolation has actuated and can be reset.

b. Phase B isolation has actuated and can be reset.

c. Phase A isolation has actuated and cannot be reset.

d. Phase B isolation has actuated and cannot be reset.

QUESTION: 049 (1.00)

Given the following conditions:

- LOCA inside containment has occurred.
- SI automatically initiated on LO-LO pressurizer pressure.
- Pressurizer pressure instruments were damaged after the SI automatically initiated.

What is the DESIGN FEATURE that assures reactor protection after the pressurizer pressure instrument is damaged?

a. Pressurizer pressure detectors and wiring runs are heavily constructed and designed withstand missile hazards.

b. SI logic circuitry seals-in (latches) when actuated and will be unaffected by pressurizer instrument damage.

c. The pressurizer pressure instruments are located away from all potential missile hazards.

d. Only one out of three signals is required to initiate an SI if the LOCA damages ONE (1) of the three instruments.

QUESTION: 050 (1.00)

If the temperature control valve in the Emergency Diesel Generator Jacket Cooling system fails in the bypass position, HOW is the diesel affected?

a. Coolant would bypass the radiator resulting in an over-heated gear box.

b. Coolant would bypass the radiator resulting in an over-heated engine.

c. Coolant would bypass the lube oil cooler resulting in over-heated lube oil.

d. Coolant would bypass the engine resulting in an over-heated engine.

QUESTION: 051 (1.00)

Given the following conditions:

- Both Unit 1 and 2 are operating at 100%.
- Stormy weather has suddenly caused the Reserve Station Service (RSS) voltage to drop approximately 12 % below rated voltage, this reduced voltage condition is sustained for a 2 minute period.
- The 1J diesel generator mode selector switch is positioned to the "Automatic Remote" position while the 1H diesel generator's mode selector switch is positioned to the "Manual Remote" position.

HOW will each diesel generator react to the degraded grid voltage condition?

a. The 1J Diesel Generator will start automatically, the 1H Diesel Generator can only be started manually.

b. The 1J Diesel Generator will start automatically, the 1H Diesel Generator is locked out and cannot be started.

c. Both 1H and 1J Diesel Generators will start automatically.

d. Degraded RSS voltage has no effect on the Diesel Generators.

QUESTION: 052 (1.00)

Given the following conditions:

- 1J diesel generator is paralleled with the 1J bus.
- KVAR is adjusted from 50 KVAR to 250 KVAR out.

Which ONE (1) of the following statements describes how the 1J diesel generator parameters are affected?

a. Voltage is essentially constant, current increases.

b. Current is essentially constant, voltage increases.

c. Current increases, and voltage increases.

d. Voltage and current remains essentially constant.

QUESTION: 053 (1.00)

What are the IMMEDIATE ACTIONS to be taken in response to a high radiation alarm on RM-RMS 162 (Manipulator) and a decreasing level in the Refueling Cavity as per 1-AP-52, Loss of Refueling Cavity During Refueling?

a. [1.] RETURN ANY FUEL ASSEMBLIES IN THE CAVITY TO THE REACTOR VESSEL.
   [2.] RETURN ANY FUEL ASSEMBLIES IN THE FUEL BUILDING TRANSFER CANAL TO THE SFP.
   [3.] IF REQUIRED DUE TO HIGH RADIATION, THEN EVACUATE CONTAINMENT AND VERIFY CONTAINMENT INTEGRITY.

b. [1.] COMMENCE MAKEUP TO THE REFUELING CAVITY.
   [2.] ISOLATE THE REFUELING CAVITY FROM THE SFP.
   [3.] IF REQUIRED DUE TO HIGH RADIATION, THEN EVACUATE CONTAINMENT AND VERIFY CONTAINMENT INTEGRITY.

c. [1.] STOP ANY RUNNING SPF COOLING PUMP AND RP PUMP.
   [2.] CHECK SFP LEVEL DECREASING.
   [3.] IF REQUIRED DUE TO HIGH RADIATION, THEN EVACUATE CONTAINMENT AND VERIFY CONTAINMENT INTEGRITY.

d. [1.] COMMENCE MAKEUP TO THE REFUELING CAVITY.
   [2.] CHECK SFP LEVEL DECREASING.
   [3.] IF REQUIRED DUE TO HIGH RADIATION, THEN EVACUATE CONTAINMENT AND VERIFY CONTAINMENT INTEGRITY.

QUESTION: 054 (1.00)

In accordance with 1-OP-14.1 (Residual Heat Removal) Chemistry personnel have sampled RHR for boron concentration. The results show that RHR concentration is 83 ppm less than the RCS. After performing the necessary calculations to compensate for the dilution of the RHR system what is the REQUIRED ACTION?

a. Borate the RCS if boron concentration is less than the Cold Shutdown boron concentration requirements before starting the RHR pump.

b. Borate the RHR system if boron concentration is less than the Cold Shutdown boron concentration requirements before starting the RHR pump.

c. Start the RHR pump and borate the RCS if boron concentration is less than the Cold Shutdown boron concentration requirements.

d. Start the RHR pump and borate the RHR if boron concentration is less than the Cold Shutdown boron concentration requirements.

QUESTION: 055 (1.00)

WHY is placing RHR in service following shutdown after a ONE (1) month period of operation at 100% power DELAYED for approximately 20 hours?

a. To provide sufficient time for operators to realign the RHR system.

b. To permit decay heat production to decrease to within the heat removal capacity of the RHR pumps.

c. To permit decay heat production to decrease to within the heat removal capacity of the RHR heat exchangers.

d. To permit RCS cooldown at a rate of 20 F per hour before placing RHR in operation.

QUESTION: 056 (1.00)

Given the following conditions:

- 100% power
- Receipt of alarm annunciator "EXC LTDN HX CC OUTLET HI TEMP"
- Receipt of alarm annunciator "CC PP SUCTION HI TEMP"
- The VCT level trending lower
- Charging flow has decreased
- CC Head tank level is trending higher
- Readings on RM-CC-120 are increasing

Which ONE (1) of the following conditions exist?

a. The component cooling water system temperature has increased.

b. The non-regenerative heat exchanger is leaking to component cooling.

c. The CVCS is leaking from the VCT to component cooling.

d. The service water heat exchanger is leaking to component cooling.

QUESTION: 057 (1.00)

The on-shift Reactor Operator has noticed that Component Cooling Surge tank level has decreased to 59% (level normally maintained is 62%) and has been directed to refill the surge tank to the normal level. What is the SOURCE of makeup and how is level NORMALLY increased. (Assume no malfunction exists.)

a. Makeup is from the Condensate system and level is increased by placing LCV-100, CC surge tank level control valve, in Automatic.

b. Makeup is from Service water and level is increased by placing LCV-100, CC surge tank level control valve, in Automatic.

c. Makeup is from Service water and level is increased by manually bypassing the LCV-100,CC surge tank level control valve.

d. Makeup is from the Condensate system and level is increased by manually bypassing the LCV-100,CC surge tank level control valve.

QUESTION: 058 (1.00)

An event has occurred causing a reactor trip and a containment pressure of 19 psia. Emergency procedure 1-E-0 Immediate Action Step [12].a states Annunciator "D" panel E-3 - LIT OR Containment pressure - HAS EXCEEDED 18 PSIA ON 1-LM-PR-110B. What CONDITION is indicated by a "YES" response to these "ACTION/EXPECTED RESPONSE" items?

a. Main Steam Line isolation signal actuates

b. LOCA with adverse containment

c. Faulted Steam Generator with adverse containment

d. Containment depressurization signal actuated

QUESTION: 059 (1.00)

What combination of the following signals will cause an ESF actuation signal?

1. High Steam flow signals from "B" Steam Generator protection channels III and IV.
2. High Steam flow signal from "C" Steam Generator protection channel IV.
3. Lo-Lo Tavg signal from "B" Steam Generator protection channel II.
4. Lo Steam Pressure from "B" Steam Generator protection channel III and "C" Steam Generator protection channel IV
5. High containment pressure signal from ONE (1) protection channel.

a. 1, 2, 5

b. 1, 3

c. 1, 4

d. 1, 2, 4

QUESTION: 060 (1.00)

Which ONE (1) of the following major action categories is correct for FR-P.1, "Response to Imminent Pressurized Thermal Shock".

a. Continue cooldown and maintain pressure and RCS inventory.

b. Identify and isolate the steam leak and cooldown the plant.

c. Stop cooldown, depressurize and stabilize conditions in RCS.

d. Identify and depressurize the steam generator to reduce the PTS.

QUESTION: 061 (1.00)

Given the following conditions:

- plant is operating at 100% power.
- The following alarm annunciators ALL actuated simultaneously:
  - HI STM LINE DELTA-P - SG 1B LO SI-RX TRIP
  - STM GEN 1A LO LEV/STM FWF MISMTCH
  - STM GEN 1B LO LEV/STM FWF MISMTCH
  - STM GEN 1C LO LEV/STM FWF MISMTCH
  - PZR LO LO PRESS SI-RX TRIP
  - TURBINE TRIPPED RX TRIP

What condition exists?

a. Major steam line break upstream the main steam isolation valve.

b. Major steam line break downstream the main steam isolation valve.

c. Excessive turbine load increase causing an accidental depressurization of the main steam system.

d. Loss of main feedwater flow to "B" Steam Generator.

QUESTION: 062 (1.00)

Given the following conditions:

- Spurious reactor trip has occurred
- E-0, immediate actions are being performed
- Component Cooling water is lost due to a supply header pipe rupture

What operator ACTIONS are required as per 1-AP-15, Loss of Component Cooling?

a. Secure all charging pumps and terminated letdown.

b. Stop RCPs if charging pumps are running, or RCS subcooling is greater than 25 F.

c. Stop RCPs and manually initiate safety injection.

d. Stop RCPs if motor temperature exceeds 195 F. or pump bearing temperature exceeds 225 F.

QUESTION: 063 (1.00)

Concerning the Component Cooling Water system which ONE (1) of the following statements describes the automatic functions that occur as a result of an ESFAS due to a HI-HI containment pressure?

a. Phase "A" isolation closes containment penetration CCW valves and trips the running CCW pumps.

b. Phase "B" isolation closes containment penetration CCW valves and trips the running CCW pumps.

c. Phase "B" isolation closes containment penetration CCW valves running CCW pumps are not affected.

d. Phase "A" isolation closes containment penetration CCW valves running CCW pumps are not affected.

QUESTION: 064 (1.00)

WHAT ensures positive isolation of Service Water flow to the Component Cooling water heat exchangers during a CDA?

a. Redundant valves in series.

b. Redundant valves in parallel.

c. Redundant heat exchangers in parallel.

d. Redundant heat pumps in parallel.

QUESTION: 065 (1.00)

Given the following:

- The reactor is operating at 100% power, steady state.
- Component Cooling Water valve TV-CC-104A fails closed.

WHICH one of the following components, if left without Component Cooling, would require manually tripping the reactor within approximately 12 minutes?

a. Non-Regenerative Heat Exchanger

b. "A" RCP Thermal Barrier Heat Exchanger

c. Shroud cooling coils

d. "A" RCP Lower Oil Reservoir

QUESTION: 066 (1.00)

Which ONE (1) of the following statements correctly describes the basis for reducing THERMAL POWER when a control bank rod is stuck at a position greater than 12 steps from the remaining rods in the control bank?

a. Maintenance of power distribution limits, minimum shutdown margin, and limit potential effects on associated accident analysis.

b. Design core thermal criteria is met, limit peak linear power density during Condition I events and limits on peak local power density.

c. Maintenance of core thermal limits, limit peak linear power density and limit potential effects on associated accident analysis.

d. Design core power distribution limits are met and peak fuel clad temperature will not exceed the 2200 limit.

QUESTION: 067 (1.00)

Given the following conditions:

- The plant is shutdown.
- Tavg is 545 F
- Loop 2 RCP is out of service.
- Rod drop times are being tested.

Why are maximum plant operating power limits reduced (In accordance with Technical Specifications) when rod drop times tested under these conditions?

a. Reactor trip times may be slower than accident analysis calculations due to reduced RCS flow.

b. Accident analysis requires rod drop times to be performed with three RCPs operating.

c. Accident analysis assumes rod drop times with all RCPs operating and Tavg greater than 500 F.

d. Accident analysis assumes rod drop times with all RCPs operating and Tavg less than 500 F.

QUESTION: 068 (1.00)

Which of the following methods are used by the operator to maintain the heat flux and nuclear enthalpy hot channel factors (FQ(Z) and FN delta-H) within their Tech. Spec. limits?

1. Maintain Quadrant Power Tilt Ratio < 1.02
2. Maintain AFD within limits.
3. Maintain control rod insertion limits.
4. Maintain shutdown bank control rods at the top of the core.
5. Maintain all control rods within a group within + 12 steps of the demand counter.
6. Maintain control rod groups properly overlapped and sequenced.

a. 1, 3, 4, 5

b. 2, 3, 4, 5

c. 2, 3, 5, 6

d. 1, 2, 3, 5

QUESTION: 069 (1.00)

What is the REASON for maintaining the control banks above the setpoint for the "ROD BANK (A/B/C/D) LO/LO-LO LIMIT" alarm?

a. Insures the maintenance of acceptable power distribution limits, maintains minimum shutdown margin, and limits the potential effects of rod misalignment on the associated accident analysis.

b. Insures adequate shutdown margin, maintain acceptable core thermal limits and limit the potential effects of rod misalignment.

c. Insures DNBR does not exceed its minimum limit during normal and short term transients, limit fission gas release, fuel pellet temperature & cladding mechanical properties to within design criteria.

d. Insures additional restrictions necessary to meet the original design criteria of rod insertion limits are not required to be implemented .

QUESTION: 070 (1.00)

Given the following conditions:

- Station battery capacity is 1650 ampere-hours
- Batteries are charged to 98% capacity
- Station blackout occurs
- the following loads are connected to the station battery

Load 1 = 12.2 amps	Load 6 = 31.6 amps
Load 2 = 3.6 amps	Load 7 = 1.4 amps
Load 3 = 7.7 amps	Load 8 = 13.3 amps
Load 4 = 16.4 amps	Load 9 = 41.2 amps
Load 5 = 8.1 amps	Load 10 = 24.5 amps

How much TIME REMAINS before the battery is depleted?

a. 7 hours, 50 minutes, 24 seconds

b. 8 hours, 0 minutes, 0 seconds

c. 10 hours, 6 minutes, 22.5 seconds

d. 10 hours, 18 minutes, 45 seconds

QUESTION: 071 (1.00)

During natural circulation conditions with the ERF computer not available, Abnormal Procedure 1-AP-10.1, "Natural Circulation Verification" Attachment 2, requires the operator to monitor the following parameters.

- RCS subcooling base on Core Exit TCs - GREATER THAN 30 F
- Steam Pressure - STABLE OR DECREASING
- RCS Hot Leg Temperature - STABLE OR DECREASING
- Core Exit TCs - STABLE OR DECREASING
- RCS Cold Leg temperature - AT SATURATION TEMPERATURE FOR SG PRESSURE

What CONDITION is indicated by a rapid decrease in steam generator pressures while steaming during natural circulation conditions?

a. Natural circulation flow has started.

b. Natural circulation flow has degraded to two-phase flow.

c. Natural circulation flow has increased due to two-phase flow .

d. Natural circulation flow has stopped.

QUESTION: 072 (1.00)

1-ECA-0.0 "Loss of All AC Power" step 16 directs the operator to depressurize intact SG(s) to 145 psig but not less than 120 psig. What is the BASIS for these limits?

a. Upper limit of 145 psig minimizes voiding in the core, lower limit of 120 psig maintains a density difference (delta p) in the RCS.

b. Upper limit of 145 psig minimizes loss of RCS inventory, lower limit of 120 psig prevents the injection of SI accumulator nitrogen into the RCS.

c. Upper limit of 145 psig minimizes the injection of accumulator nitrogen into the RCS, lower limit of 120 psig prevents voiding in the core.

d. Upper limit of 145 psig minimizes loss of RCS inventory, lower limit of 120 psig provides an adequate steam pressure head to the SG PORVs for cooldown.

QUESTION: 073 (1.00)

WHICH ONE (1) of the following actions is an IMMEDIATE operator ACTION for 1-ECA-0.0, "Loss of All AC Power"?

a. Place CCW pump switches in P-T-L (pull to lock).

b. Check if Letdown Isolation Valves are closed.

c. Energize AC Emergency bus with Emergency Diesel Generator.

d. Check main steamline isolation and bypass valves closed.

QUESTION: 074 (1.00)

1-AP-20, Operation from the Auxiliary Shutdown Panel, directs operators to trip the reactor and verify all rod bottom lights lit. IF ONE (1) control rod failed to fully insert, WHICH ONE (1) of the following actions is performed?

a. Emergency Borate until boron concentration is > 590 ppm.

b. Wait 15 hours and Emergency Borate for 21 minutes if the control room is still uninhabitable.

c. Calculate for a 930 ppm boron concentration and borate/dilute from the Auxiliary Shutdown Panel.

d. Emergency Borate from the Auxiliary Shutdown Panel for 21 minutes.

QUESTION: 075 (1.00)

A control room evacuation has been directed by the Shift Supervisor. An RO has been directed to proceed to the Main Steam Valve House to control cooldown rate. How does the Operator control the cooldown rate from the Main Steam Valve House?

a. Verify SG PORVs are not isolated then make small adjustments to the SG PORVs using the override feature.

b. Verify the SG PORV isolation valves are shut, fully open the SG PORVs and throttle with the isolation valve.

c. Place the SG PORV selector switches in LOCAL and adjust the controller to control cooldown.

d. By unisolating and using the decay heat release valve to control cooldown.

QUESTION: 076 (1.00)

Which ONE (1) of the following conditions requires the implementation of 1-AP-20, "Operation from the Auxiliary Shutdown Panel" when control room evacuation is necessary?

a. Smoke in the Control Room.

b. High Radiation in the Control Room.

c. Loss of Control Room air conditioning.

d. Loss of Emergency Switchgear Room air conditioning.

QUESTION: 077 (1.00)

The control room is being evacuated due to a large fire in the boron recovery panel. In accordance with 1-AP-50.1, Control Room Fire, where must all personnel proceed to, once the control room is evacuated? (Assume all personnel have been notified.)

a. Auxiliary Shutdown Panel

b. TSC Shift Supervisor's Office

c. TSC Fan Room

d. Emergency Switchgear Room by door to Control Room stairs

QUESTION: 078 (1.00)

WHICH ONE (1) of the following correctly describes the result of a loss of electrical power to RCP-1A Motor Component Cooling Water Supply Isolation Valve, TV-106-A?

a. TV-106-A will fail open causing condensation to develop in the stator.

b. TV-106-A will fail open causing a high temperature condition to develop in bearing lubrication system.

c. TV-106-A will fail shut causing a high temperature condition to develop in stator and motor bearing lubrication system.

d. TV-106-A will fail shut causing a high temperature condition to develop in pump bearing lubrication system.

QUESTION: 079 (1.00)

The reactor is operating at 100% power. In response to annunciator "RCP 1A-B-C SEAL LEAK LO FLOW", operators were unable to open the 1-CH-314 throttle valve in effort to increase seal injection flow to the 1B RCP. Then, annunciator "RCP 1A-B-C SEAL LEAK HI FLOW" is actuated and a check of the flow indication shows excessive leak-off for 1B RCP. SELECT the choice that lists the correct order of the following actions in accordance with 1-AP-33.

1. Shut seal leak-off valves HCV-1303A/C.
2. Ramp-down Unit at a rate greater than or equal to 3% per minute.
3. Manually trip the reactor then, stop the B RCP.
4. Verify Unit less than 25% power.

a. 2, 4, 3, 1

b. 4, 2, 3, 1

c. 1, 2, 4, 3

d. 1, 4, 2, 3

QUESTION: 080 (1.00)

WHICH ONE (1) of the following conditions requires an RCP to be stopped immediately?

a. Seal injection water increases to 130 F with RCS at 385 F.

b. Pump bearing temperature increases to 220 F.

c. Motor lower bearing temperature increases to 200 F.

d. RCP A seal leak-off temperature increases to 220 F.

QUESTION: 081 (1.00)

Due to an inadvertent RCP trip the reactor was tripped from a low operating power, which one of the following is a prerequisite for re-starting the RCP?

a. Oil lift pump has been operated for one (1) minute.

b. Reactor coolant system pressure is greater than 215 psig.

c. No.1 seal leakoff is above .2 gpm flow and delta-P above 200 psid.

d. The delta-P across No. 1 seal is greater than 150 psid.

QUESTION: 082 (1.00)

Given the following conditions:

- The plant is operating at 100% power.
- Both Intermediate Range channels (N-35 and N-36) have failed low.
- In accordance with Tech. Specs. the Shift Supervisor has made the decision to shutdown to COLD SHUTDOWN as soon as possible.

WHAT ACTION should be taken before power is reduced below the P-10 setpoint?

a. Source range detectors must be energized and the reactor tripped.

b. Source range detectors must be de-energized manually.

c. Source range jumpers must be installed.

d. Trip the reactor and re-energize the source range 15 minutes later.

QUESTION: 083 (1.00)

WHICH ONE (1) of the following indications should be observed on the "COMPARATOR AND RATE" drawer of the excore NIS for intermediate range channel N-36 approximately 2 minutes after a reactor trip caused by intermediate range channel N-35 failure?

a. Approximately 10-5 amps Neutron Level.

b. Approximately 10-6 amps Neutron Level.

c. Approximately 0 Decades Per Minute.

d. Approximately -.3 Decades Per Minute.

QUESTION: 084 (1.00)

Reactor power is 6% during a shutdown when intermediate range channel N-36 fails high. WHICH ONE (1) of the following statements describes how this failure affects the reactor shutdown and subsequent operation of the Nuclear Instrumentation system?

a. The reactor will trip on high IR flux, and source range NI's will re-energize when N-35 decreases to the proper setpoint.

b. The reactor will trip on high IR flux, and source range NI's will have to be manually re-energized.

c. The reactor will not trip, and source range NI's will re-energize when N-35 decreases to the proper setpoint.

d. The reactor will not trip, and source range NI's will have to be manually re-energized.

QUESTION: 085 (1.00)

While operating within normal plant parameters the pressurizer master pressure controller's output failed high and caused alarm annunciators "PRZ HI - LO PRESS" and "PRESSURIZER SAFETY VALVE OR PORV OPEN" to actuate. WHICH ONE (1) of the following actions is the appropriate immediate action to take in accordance with 1-AP-44, Loss of Reactor Coolant System Pressure?

a. Trip the reactor and enter E-0.

b. Take manual control of the master pressure controller.

c. Manually de-energize all pressurizer heaters.

d. Manually close PORV, PCV-1456.

QUESTION: 086 (1.00)

The plant is operating at 100% power in a normal operating line-up when the Spray Valve Controller, PC-1444G, fails (assume no operator action). Which of the following are indications of this failure?

1. Pressurizer pressure decrease.
2. Charging and letdown flow decrease.
3. Proportional and backup heaters energized.
4. Pressurizer low pressure alarm.
5. Over Temperature delta-T runback and/or reactor trip.
6. Low pressure safety injection.
7. Containment pressure increase.

a. 1, 2, 3, 4, 5, 6

b. 1, 3, 4, 5, 6

c. 1, 2, 3, 4, 6, 7

d. 1, 3, 4, 5, 7

QUESTION: 087 (1.00)

Assume a pressurizer spray line has ruptured and raises containment temperature from 100 F to 180 F. How will indicated pressurizer level compare to actual pressurizer level?

a. Indicated level will be lower than actual level.

b. Indicated level will be equal to actual level.

c. Indicated level will decrease to zero.

d. Indicated level will be higher than actual level.

QUESTION: 088 (1.00)

What is the primary OBJECTIVE of early diagnosis and isolation of a ruptured steam generator.

a. To prevent or mitigate the release of radioactive contamination to the environment.

b. To prevent or mitigate the loss of reactor coolant.

c. To permit the termination of SI before the pressurizer is filled solid.

d. To stop feeding the faulted SG and permit the rapid cooldown to a cold shutdown condition.

QUESTION: 089 (1.00)

Step 9 of 1-E-3, "Steam Generator Tube Rupture" directs the operator to align the condenser air ejector discharge to containment and remove air ejector radiation monitor instrument fuses. Which ONE (1) of the following statements is the BASES for this step?

a. To permit placing the COND AIR EJECTOR DIVERT CONT SI RESET switches to "Reset".

b. To maintain condenser vacuum and minimize activity releases to the environment.

c. To maintain an negative pressure in containment to minimize activity releases to the environment.

d. To maintain condenser vacuum and direct the ejector discharge to an area where high range radiation monitors are available.

QUESTION: 090 (1.00)

A steam generator tube rupture is in progress. E-3, Steam Generator Tube Rupture is being performed. The terry turbine steam supply isolation valve(s) for _____[1]_______ should be ______[2]_______. (Select the choice that correctly fills in the blanks.

a. [1] the non-ruptured steam generators ... [2] shut to prevent loss of inventory to atmosphere.

b. [1] the non-ruptured steam generators ... [2] shut to prevent excessive RCS cooldown.

c. [1] the ruptured steam generator ... [2] shut to prevent excessive RCS cooldown.

d. [1] the ruptured steam generator ... [2] shut to prevent activity release to atmosphere.

QUESTION: 091 (1.00)

RHR is being placed in service for cooldown for refueling, MOV-1700 will not open to start the warmup of the RHR system, what actions can be taken to open the valve.

a. Decrease letdown pressure by adjusting PCV-1145.

b. Decrease RCS pressure by adjusting PZR heaters/spray.

c. Increase letdown pressure by adjusting PCV-1145.

d. Increase RCS pressure by adjusting PZR heaters/spray.

QUESTION: 092 (1.00)

Given the following conditions:

- Solid plant operations.
- RHR system and CVCS purification in service

WHERE is excess RCS inventory routed?

a. The PRT via pressurizer PORVs.

b. The Gas Stripper via CVCS.

c. The RWST via the RHR system.

d. The CDT's via the CVCS system.

QUESTION: 093 (1.00)

A LOCA is in progress, SI has initiated, RCS pressure is 130 psig, RWST level is 20 %. The LHSI pumps are taking a suction on ______[1]_____ because the _____[2]_____. (Select the ONE (1) choice the correctly fills in the blanks.)

a. [1] Containment sump and discharging to the RWST . . . [2] RWST is below the Lo-Lo level set point and RCS pressure is above the LHSI pump shutoff head.

b. [1] RWST and discharging to the RCS . . . [2] RCS pressure is below the LHSI pump shutoff head and SI has entered the recirculation mode.

c. [1] Containment sump and discharging to the suction of HHSI pump . . [2] RWST is below the Lo-Lo level set point and the SI has entered the recirculation mode.

d. [1] RWST and recircing to the RWST . . . [2] SI has not entered the recirculation mode due to RCS pressure being above the LHSI pump shutoff head.

QUESTION: 094 (1.00)

According to 1-E-0, Reactor Trip or Safety Injection, which ONE (1) of the following conditions would require entry into 1-ES-1.1, SI Termination?

	RCS PRESS	SUBCOOLING	PZR LEVEL	S/G LVL	TOTAL AFW FLOW
a.	Increasing	33 F	~16%	~12%	~355 gpm
b.	Increasing	34 F	~ 7%	~11%	~335 gpm
c.	Decreasing	31 F	~18%	~12%	~345 gpm
d.	Stable	40 F	~15%	~9%	~300 gpm

QUESTION: 095 (1.00)

There are four main categories of plant accident and event design classification (Condition I - IV). WHICH of the following is a basis used to determine the classification?

1. The frequency of occurrence.
2. Amount of core damage.
3. The peak containment pressure achieved.
4. The potential radiological consequences to the general public.
5. The initial plant power.

a. 1, 2, 3, 4

b. 2, 3, 4

c. 1, 2, 4

d. 2, 3, 4, 5

QUESTION: 096 (1.00)

Given the following plant conditions:

- The plant is in Mode 1 operating at a stable 100% reactor power.
- ONE (1) letdown orifice valve, 1-CH-HCV-1200B, is open.
- Charging flow indicator 1-CH-FT-1122 indicates 80 gpm.
- Seal injection total is 24 gpm.

WHAT is the plant status indicated by these conditions?

a. Normal plant operating parameters for these conditions.

b. Conditions indicate excessive RCP seal injection flow.

c. Conditions indicate a loss of RCP seal leak-off.

d. Conditions indicate an RCS leak of approximately 7 gpm.

QUESTION: 097 (1.00)

Given the following conditions:

- The reactor has automatically tripped.
- The reactor trip breakers have opened.
- All IRPIs indicate zero.
- All rod bottom lights are de-energized.
- Neutron flux has leveled off at 6%.

WHAT are the reactor operator's required actions?

a. Enter E-0 and trip the rod drive MG's.

b. Enter E-0 and trip RCPs and the rod drive MG's.

c. Enter FR-S.1 drive rods in and emergency borate.

d. Enter E-0 transition to FR-S.1 drive rods in and emergency borate.

QUESTION: 098 (1.00)

If a fuel handling accident occurs, which one of the following is an immediate action in accordance with AP-30?

a. Verifying containment isolation has taken place.

b. Verify the Control Room ventilation system has actuated.

c. Notify Fuel Resources Group.

d. Notify Health Physics Department.

QUESTION: 099 (0.00)

DELETED

QUESTION: 100 (1.00)

During loss of all AC power conditions RCS heat is removed by which ONE (1) of the following?

a. Forced RCS convection flow through the core.

b. Natural RCS convection flow through the core

c. Natural RCS conduction flow through the core.

d. Forced RCS conduction flow through the core.

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

ANSWER: 001 (1.00)

a.

REFERENCE:

NORTH ANNA: Virginia Power, Nuclear Power Station GENERAL EMPLOYEE TRAINING, Page 5.
194001K116 ..(KA's)

ANSWER: 002 (1.00)

b.

REFERENCE:

NORTH ANNA: Operating Procedure 1-OP-23.2, para. 3.2 and NOTE of para. 4.1.5
194001K115 ..(KA's)

ANSWER: 003 (1.00)

d.

REFERENCE:

NORTH ANNA: Virginia Power, Nuclear Power Station, General Employee Training, page 10.
194001K114 ..(KA's)

ANSWER: 004 (1.00)

c.

REFERENCE:

NORTH ANNA: ADM 20.9, Containment Entry and Exit Under Sub-atmospheric Conditions, Section 6.5.2.3
194001K113 ..(KA's)

ANSWER: 005 (1.00)

b.

REFERENCE:

NORTH ANNA: Virginia Power, Nuclear Power Station General Employee Training, page 11;
ADM-20.10, Confined Area Entry Procedure, Section 3.3
194001K111 ..(KA's)

ANSWER: 006 (1.00)

b.

REFERENCE:

NORTH ANNA: ADM-20.30, Reportable Hazardous Material Spill Reporting, Containment and Cleanup Procedure, page 4.
194001K110 ..(KA's)

ANSWER: 007 (1.00)

a.

REFERENCE:

NORTH ANNA: VPAP-1401, Conduct of Operations, Section 6.1.16
194001A112 ..(KA's)

ANSWER: 008 (1.00)

d.

REFERENCE:

NORTH ANNA: ADM-19.1, Operations Records Administration, Section 5.1.1.g
194001A106 ..(KA's)

ANSWER: 009 (1.00)

b.

REFERENCE:

NORTH ANNA: Operations Standards, Tab P, Section 1.
194001A102 ..(KA's)

ANSWER: 010 (1.00)

b.

REFERENCE:

NORTH ANNA: VPAP-0501, Procedure Administrative Control Program, Section 6.7.1
194001A101 ..(KA's)

ANSWER: 011 (1.00)

c.

REFERENCE:

NORTH ANNA: Virginia Power, Nuclear Power Station, General Employee Training, page 18.
194001K105 ..(KA's)

ANSWER: 012 (1.00)

d.

REFERENCE:

NORTH ANNA: Virginia Power, Nuclear Power Station, General Employee Training, pages 103 & 104.
194001K104 ..(KA's)

ANSWER: 013 (1.00)

d.

REFERENCE:

NORTH ANNA: ADM - 16.5, Work Requests, Attachment 4.9; VPAP-2102, Station ALARA Program
194001K103 ..(KA's)

ANSWER: 014 (1.00)

a.

REFERENCE:

NORTH ANNA: VPAP-1402, Control of Equipment Tag-outs and Tags, section 6.2.1
194001K102 ..(KA's)

ANSWER: 015 (1.00)

c.

REFERENCE:

NORTH ANNA: VPAP-1405, Independent Verification, Section 6.1.6.b.
194001K101 ..(KA's)

ANSWER: 016 (1.00)

b.

REFERENCE:

Open reference; Emergency Plan, EPIP - 1.01, with Attachment 1.
NORTH ANNA: Emergency Plan, page 6 5; EPIP - 1.01, Attachment 1, Tab E, Section 4, page 27 of 45.
194001A116 ..(KA's)

ANSWER: 017 (1.00)

d.

REFERENCE:

NORTH ANNA: 1-AP-42, Loss of PRODAC-250 computer
194001A115 ..(KA's)

ANSWER: 018 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 86.2, Reactor Operating Principles, Section 8, Power Operations, Section Objective D, Section D.3, page 8.21
015020K504 ..(KA's)

ANSWER: 019 (1.00)

c.

REFERENCE:

NORTH ANNA: Technical Specifications, Sections: 2.1.1 and 2.2.1
015020K509 ..(KA's)

ANSWER: 020 (0.00)

DELETED

REFERENCE:

DELETED
015020K507 ..(KA's)

ANSWER: 021 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP-91.1, Engineered Safety Features, Section 2.
013000A101 ..(KA's)

ANSWER: 022 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP-91.1, Engineered Safety Features, Section 2, Section Objective B, pages 2.7 and 2.8.
013000A102 ..(KA's)

ANSWER: 023 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP-89.1, Main Steam System.
013000A104 ..(KA's)

ANSWER: 024 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP-88.1, Reactor Coolant System, Section 6, Instrumentation and Controls, para. C.6.a.
001000A101 ..(KA's)

ANSWER: 025 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 93.5, Rod Control system, Section 2, Para. A.

001000A102 ..(KA's)

ANSWER: 026 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 93.5 Rod Control, Section 2, para. A.2.b.(2); E.2.(3)
NCRODP - 89.1, Section 1, para. A.
001000A103 ..(KA's)

ANSWER: 027 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 89.4, Feedwater Systems, Section 2, Auxiliary Feedwater System, Section Objective B, page 2.12
061000K402 ..(KA's)

ANSWER: 028 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 89.4, Feedwater Systems, Section 2, Auxiliary Feedwater system, para. A.
061000K502 ..(KA's)

ANSWER: 029 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 89.4, Feedwater Systems, Section 2, Auxiliary Feedwater System, para. B.4.
061000K501 ..(KA's)

ANSWER: 030 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 89.4, Feedwater Systems, Section 2, Auxiliary Feedwater System,
059000A201 ..(KA's)

ANSWER: 031 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 89.4, Feedwater Systems, Section 1, Main Feed System, Section Objectives F and L; page 1.12; NCRODP - 93.12, SGWLC and Protection, Section Objective B and E, page 1.24; Malfunction Cause and Effects, NAPS-MFW-18, page 163
059000A212 ..(KA's)

ANSWER: 032 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 89.4, Feedwater Systems, Section 1, Main Feed System, Section Objective F, page 1.10, para. D; 1-OP-31.1, Main Feedwater System, step 5.3.8
059000A306 ..(KA's)

ANSWER: 033 (1.00)

a.

REFERENCE:

NORTH ANNA: EPIP - 1.05, Response to General Emergency.
017020K503 ..(KA's)

ANSWER: 034 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 93.1, Radiation Monitoring, pages 1.25 thru - 1.28
072000K401 ..(KA's)

ANSWER: 035 (1.00)

b.

REFERENCE:

NORTH ANNA: 1-AP-5.1, Radiation Monitoring System, Section 6.8.3.
072000K402 ..(KA's)

ANSWER: 036 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 93.1, Radiation Monitoring, Section 1, para, A.1.b, page 1.3
072000K403 ..(KA's)

ANSWER: 037 (1.00)

c.

REFERENCE:

NORTH ANNA: Tech. Specs. Bases, 3/4.6.4, page B 3/4 6-3; NCRODP - 95.6, Section 18, FR-I.3, Response to Voids in Reactor Vessel, Section Objective D, Step 17, page 18.10
028000K501 ..(KA's)

ANSWER: 038 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 92.1, Fire Protection, Section Objective A. page 1.29 and 1.30
086000K402 ..(KA's)

ANSWER: 039 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 92.1, Fire Protection System, Page 1.84
086000K503 ..(KA's)

ANSWER: 040 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 92.1, Fire Protection System, Page 1.80 and 1.82,
086000K406 ..(KA's)

ANSWER: 041 (1.00)

c.

REFERENCE:

NORTH ANNA: NAPS-MRC-21, Malfunction Cause and Effects, page 253.
011000K605 ..(KA's)

ANSWER: 042 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP 93.8; Section 2, Pressurizer Level control and Protection, page 2.11
011000K604 ..(KA's)

ANSWER: 043 (1.00)

b.

REFERENCE:

NORTH ANNA: Annunciator Response 1C-C5; NCRODP - 93.8, Pressurizer Pressure and Level Control and Protection, Section 2, Pressurizer Level Control and Protection, Section Objective B, page 2.11
011000K301 ..(KA's)

ANSWER: 044 (0.00)

DELETED

REFERENCE:

DELETED
016000K312 ..(KA's)

ANSWER: 045 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 93.10, Reactor Protection, Section 1, Reactor Protection Trips and Interlocks, page 1.18.
012000K406 ..(KA's)

ANSWER: 046 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 93.10, Reactor Protection, Section 1, Reactor Protection Trips and Interlocks, Section D. Reactor Trip Signals.
012000K402 ..(KA's)

ANSWER: 047 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP 91.1, Engineered Safety Features, Section 2, page 2.20 Emergency Procedure 1-E-1.
006050K402 ..(KA's)

ANSWER: 048 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 91.2 Containment and Containment Systems, page 1.28
006050K403 ..(KA's)

ANSWER: 049 (1.00)

b.

REFERENCE:

NORTH ANNA: Dwg. NA-Dw-5655D33, Sheet 8 of 16, Functional Diagrams of Safeguard Actuation Signals, Note 5.
006050K401 ..(KA's)

ANSWER: 050 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 90.4 Emergency Diesel Generator, Section 1, Jacket Cooling System, page 1.30; Transparency T-1.14, Jacket Cooling System
064000A201 ..(KA's)

ANSWER: 051 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 90.4, Emergency Diesel Generators, Section II, Emergency Diesel Operations, Section Objective B. page 2.2
064000A401 ..(KA's)

ANSWER: 052 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 90.4, Emergency Diesel Generators, Section I, Emergency Diesel Operations, Section Objective D. page 1.2
064000A402 ..(KA's)

ANSWER: 053 (1.00)

a.

REFERENCE:

NORTH ANNA: 1-AP-52, Loss of Refueling Cavity During Refueling. Operations Standards, Use of Procedures, Section 7.A, Use of Abnormal Procedures.
034000G015 ..(KA's)

ANSWER: 054 (1.00)

a.

REFERENCE:

NORTH ANNA: 1-OP-14.1, Residual Heat Removal System, Section 5.3
NCRODP - 88.2, Residual Heat Removal System, Section 2, RHR System Operation, Section Objective C.
005000K509 ..(KA's)

ANSWER: 055 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 88.2, Residual Heat Removal System, Section 2, RHR System Operation, Section Objective B; and Presentation Section A, page 2.5
005000K502 ..(KA's)

ANSWER: 056 (1.00)

b.

REFERENCE:

NORTH ANNA: 1-AP-5.2, Common Radiation Monitoring System, Section 5.15; Annunciator Response Manual, 1G-E3, 1G-D3
008000A202 ..(KA's)

ANSWER: 057 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 92.6, Component Cooling System, Section II Instrumentation and Control, page 2.31
008000A104 ..(KA's)

ANSWER: 058 (1.00)

a.

REFERENCE:

NORTH ANNA: 1-E-0, Reactor Trip or Safety Injection, Step 12; Annunciator Response 1D-E3, "HI STM FLOW LO STM PRESS OR LO TAVG SI-RX TRIP", Probable Cause, 1.1
000040K304 ..(KA's)

ANSWER: 059 (1.00)

d.

REFERENCE:

NORTH ANNA: Dwg. NA-DW-5655D33 Sheets 5, 7 and 8; Tech. Specs, Bases. 3/4.3.1 and 3/4.3.2 page B 3/4 3-1
000040K301 ..(KA's)

ANSWER: 060 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 95.6, Functional Restoration Procedures Section 11, Section Objective E, page 11.12
000040K101 ..(KA's)

ANSWER: 061 (1.00)

a.

REFERENCE:

NORTH ANNA: Annunciator Response Manual, pages, 1D-C1, 1D-H1, 1D-F2, 1D-H2, 1D-B3, 1D-E3, 1D-A4,
000040G011 000040G009 ..(KA's)

ANSWER: 062 (1.00)

d.

REFERENCE:

NORTH ANNA: 1-AP-15, Loss of Component Cooling, Step 5.3
000026K303 ..(KA's)

ANSWER: 063 (1.00)

b.

REFERENCE:

NORTH ANNA: Tech. Spec. 3/4.3.2, Engineered Safety Feature Actuation System Instrumentation, Tech. Spec. 3/4.6.3, Containment Isolation Valves, Tech. Spec. Bases, 3/4.6.3
000026K302 ..(KA's)

ANSWER: 064 (1.00)

a.

REFERENCE:

NORTh ANNA: NCRODP - 92.2, Service Water System, page 1.5
000026K301 ..(KA's)

ANSWER: 065 (1.00)

d.

REFERENCE:

NORTH ANNA: 1-AP-15, Loss of Component Cooling Water, page 5 of 10
000026G012 ..(KA's)

ANSWER: 066 (1.00)

a.

REFERENCE:

NORTH ANNA: Tech Spec. Bases, 3/4.1.3
000005K305 ..(KA's)

ANSWER: 067 (1.00)

c.

REFERENCE:

NORTH ANNA: Tech. Spec. Bases, 3/4.1.3
000005K304 ..(KA's)

ANSWER: 068 (1.00)

c.

REFERENCE:

NORTH ANNA: Tech. Specs, Bases, 3/4.2.2 and 3/4.2.3, Heat Flux and Nuclear Enthalpy Hot Channel Factors FQ(Z) and FNdelta-H, page B 3/4 2-4
000005K303 ..(KA's)

ANSWER: 069 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 93.5, Rod Control and Rod Position Indication System, Section 2, Rod Control, page 2.76 & 2.77; Tech. Spec. 3/4.1.3 Movable Control Assemblies Bases.
000005K302 ..(KA's)

ANSWER: 070 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP 90.3, Vital and Emergency Distribution, Section 1, Vital Distribution System, page 1.21

1650 ampere-hours x 98% = 1617 ampere-hours

12.2 amps	31.6 amps	1617 amp-hrs = 10.10625 hrs
3.6 amps	1.4 amps	160 amps
7.7 amps	13.3 amps	
16.4 amps	41.2 amps	or 10 hrs, 6 min, 22.5 sec.
8.1 amps	24.5 amps	

Total = 160 amps
000055K101 ..(KA's)

ANSWER: 071 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 5.2, Mitigating Core Damage, Section I, Post Accident Cooling, Section C.7; 1-AP_10.1, Natural Circulation Verification, Attach. 2.
000055K102 ..(KA's)

ANSWER: 072 (1.00)

b.

REFERENCE:

NORTH ANNA: 1-ECA-0.0 "Loss of All AC Power" Caution: and Note: preceding step 16.
000055K302 ..(KA's)

ANSWER: 073 (1.00)

b.

REFERENCE:

NORTH ANNA: 1-ECA-0.0, "Loss of All AC Power", Step [3]
000055G010 ..(KA's)

ANSWER: 074 (1.00)

d.

REFERENCE:

NORTH ANNA: 1-AP-20, Operation from the Auxiliary Shutdown Panel, Step 5.2.
000068K317 ..(KA's)

ANSWER: 075 (1.00)

a.

REFERENCE:

NORTH ANNA: 1-AP-50.1, Control Room Fire, Attachment #2B
000068K306 ..(KA's)

ANSWER: 076 (1.00)

a.

REFERENCE:

NORTH ANNA: 1-AP-20, Operation from the Auxiliary Shutdown Panel, Step 3.0
000068G011 ..(KA's)

ANSWER: 077 (1.00)

c.

REFERENCE:

NORTH ANNA: 1-AP-50.1 Control Room Fire, page 3 of 18
000068K318 ..(KA's)

ANSWER: 078 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 92.6 Component Cooling, Section 2 Instrumentation and Controls, page 2.19; NCRODP - 88.8, Reactor Coolant System, Section 3, Reactor Coolant Pumps, page 3.18
000015K302 ..(KA's)

ANSWER: 079 (1.00)

c.

REFERENCE:

NORTH ANNA: 1-AP-33, Reactor Coolant Pump Seal Failure.
000015K303 ..(KA's)

ANSWER: 080 (1.00)

c.

REFERENCE:

NORTH ANNA: Annunciator Response, AR-1C-H4, 1-OP-5.2, Reactor Coolant Pump Start-up and Shut-down, Section 4.3.
000015A208 ..(KA's)

ANSWER: 081 (1.00)

c.

REFERENCE:

NORTH ANNA: 1-OP-5.2, Reactor Coolant Startup and Shutdown, page 7 of 24
000015K304 ..(KA's)

ANSWER: 082 (1.00)

c.

REFERENCE:

NORTH ANNA: 1-AP-4.2, Malfunction of Nuclear Instrumentation (Intermediate Range), step 5.1
000033A210 ..(KA's)

ANSWER: 083 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 93.2, Excore Instrumentation System, Section 2, Instrument Operation, Section Objective D, page 2.11 and transparency T-2.12, NCRODP -86.1, Reactor Physics, Section VIII, Section Objective B
000033A205 ..(KA's)

ANSWER: 084 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 93.2, Excore NIS, Section 2, Instrument Operation, Section Objectives D. and K.
000033A208 ..(KA's)

ANSWER: 085 (1.00)

b.

REFERENCE:

NORTH ANNA: 1-AP-44, Loss of Reactor Coolant System Pressure
000008A101 ..(KA's)

ANSWER: 086 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 93.8, Pressurizer Pressure and Level Protection and Control, Section I, Pressurizer Pressure Protection and Control, Section Objective G.
1-AP-44, Loss of Reactor Coolant System Pressure
(Simulator: Malfunction Cause & Effects, Pressurizer Spray Valve Stuck Open)
000008A219 ..(KA's)

ANSWER: 087 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 93.8, Pressurizer Pressure and Level Protection and Control, Section II, Objective B. page 2.6 and 2.8
000008A226 ..(KA's)

ANSWER: 088 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 95.4, Emergency Operating Procedures Section 11, Steam Generator Tube Rupture, Section Objective A, page 11.2
000037A101 ..(KA's)

ANSWER: 089 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 95.4, Emergency Operating procedures, Section 11, Steam Generator Tube Rupture, Section Objective E, Step 9, page 11.8
000037K307 ..(KA's)

ANSWER: 090 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 95.4, Emergency Operating procedures, Section 11, Steam Generator Tube Rupture, Section Objective E, Step 3, page 11.6
000037G012 ..(KA's)

ANSWER: 091 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 88.2, Residual Heat Removal System, Section 2, RHR System Operation, Section Objective, A.3 and page 2.4

350 F - 140 F = 210 F delta T
210 F / 16 hours = 13.125 F/hr cooldown rate (2 RHR pumps and 2 coolers)

With only 1 RHR pump available cooldown rate = 6.5625 F/hr

325 F - 135 F = 190 F delta T
190 F / 6.5625 F/hr = 28.95238 hours
000025A101 ..(KA's)

ANSWER: 092 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 88.2, Residual Heat Removal System, Section 2, RHR System Operation, Section Objective, F and page 2.10
000025A102 ..(KA's)

ANSWER: 093 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 91.1, Engineered Safety Features, Section 2, SI or ECCS as a Subset of the ESF Systems, Section Objective E, page 2.24
000025A103 ..(KA's)

ANSWER: 094 (1.00)

a.

REFERENCE:

NORTH ANNA: 1-E-0, Reactor Trip of Safety Injection, Step 26 and 27; 1-ES-1.1, SI Termination, Step 1.
000009A201 ..(KA's)

ANSWER: 095 (1.00)

c.

REFERENCE:

NORTH ANNA: NCRODP - 95 Section 1, Objective D, page 1.5, 1.6, and 1.7
000009A101 ..(KA's)

ANSWER: 096 (1.00)

a.

REFERENCE:

NORTH ANNA: NCRODP - 95.4, Emergency Operating Procedures, Section 5, E-1 Loss of Reactor or Secondary Coolant, Section Objective B, page 5.4
000009A104 ..(KA's)

ANSWER: 097 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 95.4, Emergency Operating Procedures, Section 1, E-0, Reactor Trip/Safety Injection, Section Objective C, page 1.4, Step 1; 1-FR-S.1, Response to Nuclear Power Generation/ATWS, immediate action steps [1] thru [4]
000007K203 ..(KA's)

ANSWER: 098 (1.00)

d.

REFERENCE:

NORTH ANNA: NCRODP - 92.9, Fuel Handling and Refueling Operation, Objective E; 1-AP-30, Fuel Failure During Handling
000036K303 ..(KA's)

ANSWER: 099 (0.00)

DELETED

REFERENCE:

DELETED
000036K301 ..(KA's)

ANSWER: 100 (1.00)

b.

REFERENCE:

NORTH ANNA: NCRODP - 83, Thermodynamics, Fluid Flow and Heat Transfer; Section 9, Objective D.3, page 9.27;
000056K101 ..(KA's)

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

ANSWER SHEET

Multiple Choice (Circle or X your choice)

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

001	a	b	c	d	______
002	a	b	c	d	______
003	a	b	c	d	______
004	a	b	c	d	______
005	a	b	c	d	______
006	a	b	c	d	______
007	a	b	c	d	______
008	a	b	c	d	______
009	a	b	c	d	______
010	a	b	c	d	______
011	a	b	c	d	______
012	a	b	c	d	______
013	a	b	c	d	______
014	a	b	c	d	______
015	a	b	c	d	______
016	a	b	c	d	______
017	a	b	c	d	______
018	a	b	c	d	______
019	a	b	c	d	______
020	a	b	c	d	______
021	a	b	c	d	______
022	a	b	c	d	______
023	a	b	c	d	______
024	a	b	c	d	______
025	a	b	c	d	______

ANSWER SHEET

Multiple Choice (Circle or X your choice)

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

026	a	b	c	d	______
027	a	b	c	d	______
028	a	b	c	d	______
029	a	b	c	d	______
030	a	b	c	d	______
031	a	b	c	d	______
032	a	b	c	d	______
033	a	b	c	d	______
034	a	b	c	d	______
035	a	b	c	d	______
036	a	b	c	d	______
037	a	b	c	d	______
038	a	b	c	d	______
039	a	b	c	d	______
040	a	b	c	d	______
041	a	b	c	d	______
042	a	b	c	d	______
043	a	b	c	d	______
044	a	b	c	d	______
045	a	b	c	d	______
046	a	b	c	d	______
047	a	b	c	d	______
048	a	b	c	d	______
049	a	b	c	d	______
050	a	b	c	d	______

A N S W E R   S H E E T

Multiple Choice   (Circle or X your choice)

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

051	a	b	c	d	______
052	a	b	c	d	______
053	a	b	c	d	______
054	a	b	c	d	______
055	a	b	c	d	______
056	a	b	c	d	______
057	a	b	c	d	______
058	a	b	c	d	______
059	a	b	c	d	______
060	a	b	c	d	______
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063	a	b	c	d	______
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065	a	b	c	d	______
066	a	b	c	d	______
067	a	b	c	d	______
068	a	b	c	d	______
069	a	b	c	d	______
070	a	b	c	d	______
071	a	b	c	d	______
072	a	b	c	d	______
073	a	b	c	d	______
074	a	b	c	d	______
075	a	b	c	d	______

ANSWER SHEET

Multiple Choice (Circle or X your choice)

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

076	a	b	c	d	______
077	a	b	c	d	______
078	a	b	c	d	______
079	a	b	c	d	______
080	a	b	c	d	______
081	a	b	c	d	______
082	a	b	c	d	______
083	a	b	c	d	______
084	a	b	c	d	______
085	a	b	c	d	______
086	a	b	c	d	______
087	a	b	c	d	______
088	a	b	c	d	______
089	a	b	c	d	______
090	a	b	c	d	______
091	a	b	c	d	______
092	a	b	c	d	______
093	a	b	c	d	______
094	a	b	c	d	______
095	a	b	c	d	______
096	a	b	c	d	______
097	a	b	c	d	______
098	a	b	c	d	______
099	a	b	c	d	______
100	a	b	c	d	______

ANSWER SHEET

Multiple Choice (Circle or X your choice)

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

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

A N S W E R K E Y

001	a
002	b
003	d
004	c
005	b
006	[illegible]
007	a
008	d
009	b
010	b
011	c
012	d
013	d
014	a
015	c
016	b
017	d
018	b
019	c
020	D
021	a
022	a
023	b
024	c
025	a

ANSWER KEY

026	a
027	d
028	b
029	a
030	b
031	b
032	b
033	a
034	c
035	b
036	a
037	c
038	a
039	d
040	a
041	c
042	d
043	b
044	D
045	b
046	d
047	c
048	a
049	b
050	b

ANSWER KEY

051	c
052	a
053	a
054	a
055	c
056	b
057	d
058	a
059	d
060	c
061	a
062	d
063	b
064	a
065	d
066	a
067	c
068	c
069	a
070	c
071	d
072	b
073	b
074	d
075	a

ANSWER KEY

076	a
077	c
078	c
079	c
080	c
081	c
082	c
083	d
084	b
085	b
086	b
087	d
088	a
089	b
090	d
091	c
092	b
093	c
094	a
095	c
096	a
097	d
098	d
099	D
100	b

ANSWER KEY

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

QUESTION	VALUE	REFERENCE
001	1.00	9000101
002	1.00	9000102
003	1.00	9000103
004	1.00	9000104
005	1.00	9000105
006	1.00	9000106
007	1.00	9000107
008	1.00	9000108
009	1.00	9000109
010	1.00	9000110
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015	1.00	9000115
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018	1.00	9000118
019	1.00	9000119
020	0.00	9000120
021	1.00	9000121
022	1.00	9000122
023	1.00	9000123
024	1.00	9000124
025	1.00	9000125
026	1.00	9000126
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039	1.00	9000139
040	1.00	9000140
041	1.00	9000141
042	1.00	9000142
043	1.00	9000143
044	0.00	9000144
045	1.00	9000145
046	1.00	9000146
047	1.00	9000147
048	1.00	9000148
049	1.00	9000149
050	1.00	9000150
051	1.00	9000151
052	1.00	9000152
053	1.00	9000153
054	1.00	9000154

QUESTION	VALUE	REFERENCE
055	1.00	9000155
056	1.00	9000156
057	1.00	9000157
058	1.00	9000158
059	1.00	9000159
060	1.00	9000160
061	1.00	9000161
062	1.00	9000162
063	1.00	9000163
064	1.00	9000164
065	1.00	9000165
066	1.00	9000166
067	1.00	9000167
068	1.00	9000168
069	1.00	9000169
070	1.00	9000170
071	1.00	9000171
072	1.00	9000172
073	1.00	9000173
074	1.00	9000174
075	1.00	9000175
076	1.00	9000176
077	1.00	9000177
078	1.00	9000178
079	1.00	9000179
080	1.00	9000180
081	1.00	9000181
082	1.00	9000182
083	1.00	9000183
084	1.00	9000184
085	1.00	9000185
086	1.00	9000186
087	1.00	9000187
088	1.00	9000188
089	1.00	9000189
090	1.00	9000190
091	1.00	9000191
092	1.00	9000192
093	1.00	9000193
094	1.00	9000194
095	1.00	9000195
096	1.00	9000196
097	1.00	9000197
098	1.00	9000198
099	0.00	9000199
100	1.00	9000200

	97.00	

	97.00	

No. 97887200
Rev. A

MTM 22

2450

VIRGINIA POWER
NORTH ANNA POWER STATION
EMERGENCY PLAN IMPLEMENTING PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
EPIP-1.01	EMERGENCY MANAGER CONTROLLING PROCEDURE (With 9 Attachments)	15
		PAGE 1 of 7

PURPOSE

To initially assess a potential emergency condition and initiate corrective actions.

USER

Shift Supervisor OR Station Emergency Manager (Steps 1-8)
Coordinator-Emergency Planning (Step 9)

ENTRY CONDITIONS

Any one of the following :

1. Another station procedure directs initiation of this procedure,
2. A potential emergency condition is reported to the Shift Supervisor.

SAFETY RELATED

REVISION RECORD

REV. 15 PAGE(S): ENTIRE PROCEDURE DATE: 3/21/90

RECOMMENDED APPROVAL	APPROVED	DATE
M. Bumberg CHAIRMAN STATION NUCLEAR SAFETY AND OPERATING COMMITTEE	[signature] STATION MANAGER	3/21/90

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 1.	MAKE INITIATE ASSESSMENT:	
	a) Perform initial assessment using EAL Tables:	
	b) IF EAL exceeded, initiate EPIP-1.01:	b) Return to procedure in effect.
	1) BY: ________ DATE: ________ TIME: ________	

<u>NOTE</u>: Continue through this and all further instructions unless otherwise directed to hold.

____ 2.	IDENTIFY EVENT:	
	a) Event-TRANSPORT OF CONTAMINATED INJURED PERSONNEL	a) <u>GO TO</u> Step <u>2.b</u>
	1) Initiate EPIP-5.01 <u>Transport of Contaminated Injured Personnel</u>	
	2) Verify initiation of EPIP-4.20, <u>H.P. Actions for Transport of Injured Contaminated Personnel</u>	
	3) Direct Health Physics to initiate EPIP-4.01, <u>Radiological Assessment Controlling Procedure</u>.	

(STEP 2 CONTINUED ON NEXT PAGE)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

____ 2. IDENTIFY EVENT (CONTINUED):

b) Event-Any of the following: — b) GO TO Step 3

Radiation Release
OR
Fuel Handling Incident
OR
Secondary Release
OR
S/G Tube Rupture
OR
LOCA

1) Direct Health Physics to initiate EPIP-4.01, Radiological Assessment Director Controlling Procedure.

* *

CAUTION: Declaration of the highest emergency class for which an Emergency Action Level is exceeded shall be made.

* *

NOTE: The SPDS is potentially unreliable in the event of an earthquake. Therefore, the SPDS should be evaluated for reliability of accident assessment should this situation occur.

____ 3. ASSESS AND CLASSIFY EVENT:

a) Use the SPDS to obtain indications of emergency conditions listed in the EAL Table, as necessary.

(STEP 3 CONTINUED ON NEXT PAGE)

No. 97887210

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 3.	ASSESS AND CLASSIFY EVENT: (CONTINUED)	
	b) Use Attachment 1, Emergency Action Level Table Index, to determine event category, AND Go To appropriate EAL Table.	
	c) Evaluate event to determine emergency classification	
	d) Go To Step 4	
___ 4.	PERFORM NOTIFICATION AND VERIFICATION:	
	a) Verify LEOF- NOT ACTIVATED	a) IF LEOF activated, announce to staff the transfer of Notification responsibility from TSC to LEOF and proceed to Step 4.b.
	b) Verify TSC - NOT ACTIVATED	b) IF TSC activated, GO TO Step 5.
	c) Direct STA to report to Control Room	
	d) Direct Emergency Communicators to report to Control Room and initiate EPIP-2.01 Notification of State and Local Governments.	
	e) Initiate ADM-16.3, Notification Requirements.	
	f) Verify H.P. has initiated EPIP-4.01, Radiological Assessment Director Controlling Procedure	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 5.	DETERMINE EPIPS:	
	a) Event classification - NOTIFICATION OF UNUSUAL EVENT	a) GO TO Step 5.b
	1) GO TO EPIP-1.02, Response to Notification of Unusual Event	
	b) Event classification - ALERT	b) GO TO Step 5.c
	1) GO TO EPIP-1.03, Response to Alert	
	c) Event classification - SITE AREA EMERGENCY	c) GO TO Step 5.d.
	1) GO TO EPIP-1.04, Response to Site Area Emergency	
	d) Event classification - GENERAL EMERGENCY	
	1) GO TO EPIP-1.05, Response to General Emergency	
____ 6.	MAKE OFFSITE TERMINATION NOTIFICATIONS:	
	a) Initiate termination notification to state and local governments IAW EPIP-2.01, Notification of State and Local Governments	
	b) Initiate termination notification to NRC IAW EPIP-2.02, Notification of NRC	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE: Termination announcement should, at a minimum, address the following:	
	a) Emergency Termination	
	b) Facility Deactivation	
	c) Selective release of personnel	
	d) Completion and collection of procedures.	
	e) Recovery	
____ 7.	TERMINATE EMERGENCY:	
	a) Make termination announcement to involved station personnel.	
	b) IF required, Go To EPIP-6.01, Re-entry Recovery Guidelines.	b) Go To Step 8.
____ 8.	TERMINATE EPIP 1.01:	
	a) COMPLETED BY: ______________	
	DATE: ______________	
	TIME: ______________	
	b) Forward completed EPIPs, forms and other applicable records to the Coordinator-Emergency Planning	

No. 97887210

NUMBER	PROCEDURE TITLE	REVISION
EPIP-1.01	EMERGENCY MANAGER CONTROLLING PROCEDURE	15
		PAGE 7 of 7

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: The following step is to be completed by the Coordinator-Emergency Planning.

_____ 9. COMPLETE REPORTING/DOCUMENTATION REQUIREMENTS:

a) Collect completed EPIPs

b) Complete written summary report

c) Provide written summary report to the State:

Date: ____________________

Time: ____________________

d) Attach copy of written summary report to this EPIP.

e) Completed By: ____________________

Date: ____________________

Time: ____________________

END

No. 97667220

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE	15
ATTACHMENT		PAGE
1	INDEX	1 of 45

****

CAUTION: Declaration of the highest emergency class for which an EAL is exceeded shall be made.

****

No. 97887320

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Non-transient ECCS initiation MODES 1, 2, 3 & 4	Non-spurious ECCS initiation as validated by Emergency Procedures	NOTIFICATION OF UNUSUAL EVENT
2. Mode reduction required by Tech Spec -LCO MODES 1, 2, 3, 4	Intentional reduction in power, load, or temperature because the unit has entered an Action Statement or will exceed an LCO NOTE: In the event other plant conditions require a shutdown, the NOUE must still be declared on the basis that a shutdown would have been required by the T.S.	NOTIFICATION OF UNUSUAL EVENT
3. Failure of a safety or relief valve to close after pressure reduction, which may affect the health and safety of the public ALL MODES	Either condition a) or b) exists: a) RCS Pressurizer safety or PORV flow as indicated by temperature monitoring equipment AND RCS pressure-LESS THAN 1600 psig	NOTIFICATION OF UNUSUAL EVENT

(Event 3 Continued Next Page)

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB A) SYSTEM SHUTDOWN, OR ASSESSMENT SYSTEM EVENT	15
ATTACHMENT		PAGE
1		3 of 45

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
	b) Main Steam Excessive Steam Generator Safety, PORV, or Decay Heat Release flow as indicated by rapid RCS cooldown rate AND MS pressure is GREATER THAN 100 psi below set point of affected valve.	
4. Total loss of function needed for unit CSD condition MODE 5 MODE 6	Condition A exists with B a) Secondary System cooling capability unavailable AND b) Loss of any of the following systems with RCS temperature greater than 140°F. 1) Service Water 2) Component Cooling 3) RHR	ALERT
5. Loss of Function needed for unit HSD condition MODES 1,2,3, & 4	Total loss of the following: a) Charging/SI System OR b) Main feedwater AND Auxiliary Feedwater System	SITE AREA EMERGENCY

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB A) SYSTEM SHUTDOWN, OR ASSESSMENT SYSTEM EVENT	15
ATTACHMENT		PAGE
1		4 of 45

CONDITION	INDICATION	CLASSIFICATION
6. Failure of the Reactor Protection System to complete a trip which takes the Rx Subcritcal MODES 1 & 2	a) Manual or Automatic RX trip-INITIATED AND b) Intermediate Range Monitor indicating a POSITIVE SUR	ALERT
7. Failure of the Reactor Protection System to initiate and complete a required trip while at power MODES 1 & 2	a) RX trip setpoint and coincidences-EXCEEDED AND b) Manual Rx trip-INITIATED AND c) Rx power indication NOT DECREASING	SITE AREA EMERGENCY
8. Indications or alarms on process or effluent parameters required for incident assessment NOT functional in the control room MODES 1,2,3 & 4	Intentional reduction in power, load, or temperature because the unit has entered an Action Statement or will exceed an LCO NOTE: In the event other plant conditions require a shutdown, the NOUE must still be declared on the basis that a shutdown would have been required by the T.S.	NOTIFICATION OF UNUSUAL EVENT

(Event 8 Continued Next Page)

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
	Any of the following exist: a) Containment Gaseous or Particulate Radiation Monitors- NOT OPERABLE AND Backup grab sample capability is inoperable per leak detection T.S.3.4.6.1 b) Post-accident instrumentation-LESS THAN minimum channels allowable per T.S. 3.3.3.6	
9. Failure of meteorological instrumentation required to perform-offsite dose calculations Modes 1, 2, 3 & 4	a) Meteorological monitoring instrumentation- LESS THAN minimum required to perform offsite dose calculations (wind speed, wind direction, and stability class)	NOTIFICATION OF UNUSUAL EVENT
10. Loss of communications capability ALL MODES	Complete failure of the following: a) Station PBX phone system AND b) Station Gai-Tronics system AND c) Station UHF radio system	NOTIFICATION OF UNUSUAL EVENT

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
11. All main board annunciator alarms and unit computer lost MODES 1 & 2	Simultaneous loss of all annunciator alarms on panels "A" to "K" with loss of unit computer	ALERT
12. All main board annunciator alarms and unit computer lost for more than 15 minutes during a unit transient MODES 1 & 2	a) Complete loss of all annunciator alarms on panels "A" to "K" AND b) Loss of unit computer for GREATER THAN 15 minutes AND c) Unit operational transient-IN PROGRESS	SITE AREA EMERGENCY
13. Evacuation of Main Control Room required ALL MODES	Evacuation of the Control Room with shut down control established within 15 minutes	ALERT
14. Evacuation of Main Control Room with control NOT established within 15 minutes ALL MODES	Evacuation of the Control Room with local shutdown control NOT established within 15 minutes	SITE AREA EMERGENCY

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Safety Limit-RCS Temperature/Pressure curve exceeded MODES 1 & 2	Limits of T.S. Fig 2.1-1 - EXCEEDED (See Att. 9)	NOTIFICATION OF UNUSUAL EVENT
2. RCS overpressure MODES 1,2,3,4 & 5	2735 psig RCS pressure limit-EXCEEDED	NOTIFICATION OF UNUSUAL EVENT
3. RCP locked rotor leading to fuel damage MODE 1	All the following exists: a) Flow in one or more RC loops LESS THAN 90% AND b) RCP trip caused by Phase Overcurrent Relay actuation AND c) High Range Letdown Radiation Monitor (RM-CH-128 or RM-CH-228) indication increases to - GREATER THAN 10^6 cpm	ALERT

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
4. RCS leak rate requiring plant shutdown IAW. T.S. 3.4.6.2 or 3.4.6.3 MODES 1, 2, 3, & 4	Intentional reduction in power, load, or temperature because the unit has entered an Action Statement or will exceed an LCO NOTE: In the event other plant conditions require a shutdown, the NOUE must still be declared on the basis that a shutdown would have been required by the T.S. Any of the following exist: a) Unidentified RCS leakage-GREATER THAN 1 gpm OR b) Identified leakage-GREATER THAN 10 gpm OR c) Controlled leakage from RCP seals-GREATER THAN 30 gpm TOTAL OR d) N16 Monitor indicates primary to secondary leakage greater than Tech. Spec. allowable limits. e) Any pressure boundary leakage	NOTIFICATION OF UNUSUAL EVENT

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
5. RCS leak rate exceeds 50 gpm MODES 1, 2, 3 & 4	a) Pressurizer Level cannot be maintained GREATER THAN 20% with one(1) Charging/SI Pump in operation AND b) RCS inventory balance indicates leakage-GREATER THAN 50 gpm	ALERT
6. RCS leak rate exceeds 300 gpm MODES 1, 2 , 3 & 4	a) Loss of Reactor Coolant in progress AND b) Pressurizer level can not be maintained with two (2) or more Charging/SI Pumps in operation	SITE AREA EMERGENCY
7. Primary to Secondary leakage-GREATER THAN 1 gpm MODES 1, 2, 3 & 4	Intentional reduction in power, load, or temperature because the unit has entered an Action Statement or will exceed an LCO. a) Primary to Secondary leakage Greater than 1 gpm. OR b) N16 Monitor indicates primary to secondary leakage greater than Tech. Spec. allowable limits. NOTE: In the event other plant conditions require a shutdown, the NOUE must still be declared on the basis that a shutdown would have been required by the T.S.	NOTIFICATION OF UNUSUAL EVENT

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
8. Gross Primary to Secondary leakage MODES 1, 2, 3 & 4	Steam Generator Tube Rupture, is in progress with SI in progress	ALERT
9. Excessive Primary to Secondary leakage with loss of offsite power MODES 1, 2, 3, & 4	a) Intentional reduction in power, load, or temperature because the unit has entered an Action Statement or will exceed an LCO for Primary to Secondary Leakage AND b) Condenser Air Ejector (RM-SV-121 or RM-SV-221) - GREATER THAN 1000% (10 times allowable) Tech. Specs - (See Att. 5) OR Steam Generator Blowdown Monitor Readings-GREATER THAN $1x10^5$ cpm Monitor designations: RM-SS-122 RM-SS-123 RM-SS-124 RM-SS-222 RM-SS-223 RM-SS-224 AND c) A subsequent loss of offsite power indicated by zero volts on voltmeters for 4160V buses D, E & F.	ALERT

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB B) REACTOR COOLANT SYSTEM EVENT	15
ATTACHMENT		PAGE
1		11 of 45

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
10. Gross Primary to Secondary leakage with loss of offsite power MODES 1, 2,3 & 4	a) Steam Generator Tube Rupture, is in progress with SI in progress AND b) Condenser Air Ejector (RM-SV-121 or RM-SV-221) - GREATER THAN Site Area Level (See Att. 5) OR Steam Generator Blowdown Monitor GREATER THAN $1x10^6$ cpm Monitor designations: RM-SS-122 RM-SS-123 RM-SS-124 RM-SS-222 RM-SS-223 RM-SS-224 AND c) A subsequent loss of offsite power indicated by zero volts on voltmeters for 4160V Buses D, E & F	SITE AREA EMERGENCY

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
11. Loss of 2 of 3 fission product barriers with potential loss of 3rd barrier ALL MODES	Any two of a),b) or c) exist and the third is imminent a) Fuel clad integrity failure as indicated by any of the following: 1) RCS specific activity - GREATER THAN OR EQUAL TO <u>300.0</u> uCi/Gram dose equivalent I-131. 2) 5 or more core exit thermocouples GREATER <u>THAN-1200°</u> F <u>OR</u> b) Loss of RCS integrity as indicated by any of the following: 1) RCS pressure-GREATER THAN <u>2735 psig</u> 2) A loss of Reactor Coolant is in progress <u>OR</u> c) Loss of containment integrity as indicated by any of the following: 1) Containment pressure -GREATER THAN <u>60</u> psia and <u>NOT</u> decreasing 2) Loss of containment integrity as defined in T.S. 1.6 (See Att. 7)	GENERAL EMERGENCY

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
12. Fuel failure with steam generator tube rupture ALL MODES	Any two of a),b) or c) exists and the third is imminent a) Fuel clad integrity failure as indicated by any of the following: 1) RCS specific activity-GREATER THAN <u>300</u> uCi/gram dose equivalent I-131 2) 5 or more core exit thermocouples -GREATER THAN <u>1200° F</u> <u>OR</u> b) S/G tube rupture as indicated by both of the following: 1) SI initiated by RCS Low Pressure 2) A Steam Generator Tube Rupture, is in progress <u>OR</u> c) Loss of Secondary integrity as indicated by any of the following: 1) Steam Generator Power Operated Relief Valve - OPEN 2) Main Steam Code Safety Valve - OPEN 3) A Loss of Secondary Coolant, is in progress	GENERAL EMERGENCY

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB C) FUEL FAILURE OR FUEL HANDLING ACCIDENT	15
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Fuel clad damage indication MODES 1,2,3 AND 4	a) Intentional reduction in power, load, or temperature because the unit has entered an Action Statement or will exceed LCO. NOTE: In the event other plant conditions require a shutdown, the NOUE must still be declared on the basis that a shutdown would have been required by the T.S. OR b) High Range Letdown Radiation Monitor (RM-CH-128 or RM-CH-228) indication increases GREATER THAN 10^5 cpm within 30 minutes AND remains for at least 15 minutes	NOTIFICATION UNUSUAL EVENT
2. Severe Fuel Clad Damage MODES 1,2,3 AND 4	a) RCS specific activity-GREATER THAN 300.0 uCi/gram dose equivalent I-131 OR b) High Range Letdown Radiation Monitor (RM-CH-128 or RM-CH-228) indication increases GREATER THAN 10^5 cpm within 30 minutes AND remains for at least 15 minutes	ALERT

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
3. Core damage with possible loss of coolable geometry MODES 1,2,3 AND 4	Condition a) exists with b) a) Fuel clad failure as indicated by any of the following: 1) RCS Specific activity GREATER THAN <u>60</u> uCi/gram dose equivalent I-131 2) High Range Letdown Radiation Monitor (RM-CH-128 or RM-CH-228) indication GREATER THAN $1x10^6$ cpm <u>AND</u> b) Loss of cooling as indicated by any of the following: 1) 5 confirmed core exit thermocouples - GREATER THAN <u>1200° F</u> 2) Core DT-ZERO 3) Core DT - RAPIDLY DIVERGING	SITE AREA EMERGENCY

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
4. Probable large radioactivity release iniated by LOCA with ECCS failure leading to core degradation ALL MODES	a) A Loss of Reactor Coolant is in progress AND RCS specific activity-GREATER THAN 300.0 uCi/gram dose equivalent I-131 AND b) High or Low Head ECCS flow are NOT being delivered to the core	GENERAL EMERGENCY
5. Probable large radioactivity release initiated by loss of heat sink leading to core degradation ALL MODES	a) Loss of Main FW system and Condensate System AND b) Loss of Auxiliary FW System	GENERAL EMERGENCY
6. Probable large radioactivity release initiated by failure of protection system to bring Rx subcritical and causing core degradation ALL MODES	Condition a) exists with b) or c) a) Rx nuclear power after a trip -GREATER THAN 5% AND b) RCS pressure GREATER THAN OR EQUAL TO 2485 psig OR c) Containment pressure AND temperature -RAPIDLY INCREASING	GENERAL EMERGENCY

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
7. Probable large radioactivity release initiated by loss of AC and all feedwater ALL MODES	a) Loss of all AC procedures implemented AND b) Turbine Driven Auxiliary Feedwater Pump NOT OPERABLE AND c) Restoration of a) or b) above not likely within 2 hours	GENERAL EMERGENCY
8. Probable large radioactivity release initiated by LOCA with loss of ECCS and containment cooling ALL MODES	Condition a) and b) exist with c) or d) a) A Loss of Reactor Coolant is in progress AND b) High or Low Head ECCS flow is NOT being delivered to the core AND c) Containment RS sump temperature-GREATER THAN 190° F AND NOT DECREASING OR d) Quench Spray and Recirculation Spray Systems- NOT OPERABLE	GENERAL EMERGENCY

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB C) FUEL FAILURE OR FUEL HANDLING ACCIDENT	15
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
9. Fuel damage accident with release of radioactivity to containment or fuel buildings ALL MODES	Condition a) exists with b) or c) a) Verified accident involving damage to irradiated fuel AND b) Health Physics confirms fission product release from fuel OR c) Readings on the Ventilation Vent B Gaseous Monitor (RM-VG-113) -GREATER THAN 1000% (10 times allowable) Tech. Specs. (See Att. 3)	ALERT
10. Potential for fuel damage to occur during refueling MODE 6	Continuing uncontrolled decrease of water level in Reactor Refueling Cavity OR Continuing uncontrolled decrease of water level in Spent Fuel Pool	ALERT

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
11. Major fuel damage accident with radio-activity release to containment or fuel buildings ALL MODES	Conditions a) or b) exist with c) a) Water level in Rx vessel during refueling -BELOW TOP OF CORE OR b) Water level in Spent Fuel Pool -BELOW TOP OF SPENT FUEL AND c) Verified damage to irrad-iated fuel resulting in readings on Ventilation Vent B Gaseous Monitor (RM-VG-113) -GREATER THAN Site Area Level (See Att. 3)	SITE AREA EMERGENCY

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Loss of Containment integrity MODES 1,2,3 & 4	Intentional reduction in power, load, or temperature because the unit has entered an Action Statement or will exceed an LCO. NOTE: In the event other plant conditions require a shutdown, the NOUE must still be declared on the basis that a shutdown would have been required by the T.S. The following exists: a) Loss of containment integrity as defined by T.S.1.6 (See Att. 7) and prescribed by T.S. 3.6.1.1	NOTIFICATION OF UNUSUAL EVENT
2. High Containment radiation, pressure and temperature MODES 1,2,3 & 4	Condition a) exists with b) or c) a) Containment High Range Radiation Monitor (RM-RMS-165 or RM-RMS-265) indicates GREATER THAN 8.15×10^{1} R/hr AND b) Containment pressure -GREATER THAN 17 psia OR c) Containment temperature -GREATER THAN 150° F	ALERT

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB D) CONTAINMENT EVENT	15
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
3. High-high Containment radiation, pressure, and temperature MODES 1,2,3 & 4	Condition a) exists with b) or c) a) Containment High Range Radiation Monitor (RM-RMS-165 or RM-RMS-265) indicates GREATER THAN 1.63×10^3 R/hr AND b) Containment pressure -GREATER THAN 27.75 psia AND is NOT decreasing OR c) Containment temperature indicates-GREATER THAN 200° F	SITE AREA EMERGENCY

No. 97667220

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB D) CONTAINMENT EVENT	15
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
4. Extremely high Containment radiation, pressure and temperature MODES 1,2,3 & 4	Condition a)exists with b) or c) a) Containment High Range Radiation Monitor (RM-RMS-165 or RM-RMS-265) indicates GREATER THAN 3.25×10^4 R/Hr AND b) Containment pressure -GREATER THAN 45 psia AND NOT DECREASING OR c) Containment temperature -GREATER THAN 280° F	GENERAL EMERGENCY

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB E) RADIOACTIVITY EVENT	15
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Effluent release GREATER THAN T.S. allowable Limit ALL MODES	Any of the following monitors indicate valid readings above the specified value a) Clarifer Effluent Monitor (RM-LW-111) -GREATER THAN 2.2×10^{5} cpm b) Vent Stack A Gaseous Monitor (RM-VG-104) -GREATER THAN Tech Specs. limit. (see Att. 2) c) Vent Stack B Gaseous Monitor (RM-VG-113) -GREATER THAN Tech Specs. limit (See Att. 3) d) Air Ejector Monitor(s)- (RM-SV-121 and/or RM-SV-221) GREATER THAN Tech Specs. limit (See Att. 5) e) Discharge Canal Monitor(s) (RM-SW-130 and/or RM-SW-230) -GREATER THAN 6.1×10^{2} cpm	NOTIFICATION OF UNUSUAL EVENT

Event 1 Continued on next page.

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NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB E) RADIOACTIVITY EVENT	15
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. (CONTINUED)	f) Process Vent Gaseous Monitor (RM-GW-102) -GREATER THAN $1x10^6$ cpm FOR LESS THAN 15 MINUTES g) Sample results indicate GREATER THAN T.S. allowable limit.	
2. Effluent Release GREATER THAN 10 TIMES T.S. instantaneous allowable limits ALL MODES	Any of the following monitors indicate valid readings above the specified values for GREATER THAN 15 minutes a) Clarifier Effluent Monitor (RM-LW-111) -GREATER THAN $1x10^6$ cpm b) Vent Stack A Gaseous Monitor (RM-VG-104) -GREATER THAN 1,000% (10 times limit) Tech. Specs. (See Att. 2) c) Vent Stack B Gaseous Monitor (RM-VG-113) -GREATER THAN 1,000% (10 times limit) Tech. Specs.(see Att. 3)	ALERT

Event 2 Continued on next page.

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
2. (CONTINUED)		
	d) Air Ejector Monitor(s)- (RM-SV-121 and/or RM-SV-221) GREATER THAN 1,000% (10 times limit) Tech. Specs. (see Att. 5)	
	e) Discharge Canal Monitor(s) (RM-SW-130 and/or RM-SW-230) GREATER THAN $\underline{6.1\times10^{3}}$ cpm	
	f) Process Vent Gaseous Monitor (RM-GW-102) GREATER THAN $\underline{1\times10^{6}}$ cpm for GREATER THAN <u>15</u> minutes.	
	g) Sample results indicate GREATER THAN <u>10 TIMES</u> T.S. allowable limit.	

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
3. High radiation or airborne contamination levels indicate a severe degradation in control of radioactive material ALL MODES	Valid readings on any of the following monitors have increased by a factor of 1000 a) Ventilation Vent Multi-sample Gaseous (RM-VG-106) and Particulate Monitor (RM-VG-105) b) Control Room Area Monitor (RMS-157) c) Auxiliary Building Control Area Monitor (RMS-154) d) Decontamination Building Area Monitor (RMS-151) e) Fuel Pit Bridge Area Monitor (RMS-153) f) New Fuel Storage Area Monitor (RMS-152) g) Laboratory Area Monitor (RMS-158) h) Sample Room Area Monitor (RMS-156)	ALERT

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB E) RADIOACTIVITY EVENT	15
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
4. Release imminent or in progress and projected or actual site boundary doses of 0.5 Rem to 2 REM W.B. or 1 Rem to 12 Rem thyroid exposure ALL MODES	Valid indications of any of the following exist: a) Any Main Steam Line High Range Monitor GREATER THAN 0.47 mR/hr Monitor Designations: RM-MS-170 RM-MS-171 RM-MS-172 RM-MS-270 RM-MS-271 RM-MS-272 OR b) Ventilation Vent A Gaseous Monitor (RM-VG-104) - GREATER THAN Site Area level (See Att. 2) or Kaman monitor (RM-VG-179) reading greater than 3.6 X 10^6 uCi/sec OR c) Ventilation Vent B Gaseous Monitor (RM-VG-113) - GREATER THAN Site Area level (See Att .3) or Kaman monitor (RM-VG-180) reading greater than 3.6 X 10^6 uCi/sec OR	SITE AREA EMERGENCY

Event 4 Continued on next page.

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
4. (CONTINUED)	d) Process Vent High Range Monitor (RM-GW-173) -GREATER THAN Site Area level (See Att. 6) or Kaman monitor (RM-VG-178) reading greater than 3.6 X 10^6 uCi/sec OR e) Monitoring Team samples indicate Site Boundary doses of 0.5 to 2.0 Rem W.B. or 1 to 12 Rem thyroid exposure	
5. Release imminent or in progress and projected or actual site boundary doses exceed 2 Rem W.B. or 12 REM thyroid exposure ALL MODES	a) Confirmed Health Physics assessments of actual or projected Site boundary doses -GREATER THAN 2 Rem WHOLE BODY OR 12 Rem THYROID EXPOSURE OR b) Any Main Steam Line High Range Monitor GREATER THAN 9.4 mR/hr Monitor Designations: RM-MS-170 RM-MS-171 RM-MS-172 RM-MS-270 RM-MS-271 RM-MS-272 OR	GENERAL EMERGENCY

Event 5 Continued on next page

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EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB E) RADIOACTIVITY EVENT	15
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
5. (CONTINUED)		
	c) Ventilation Vent A High Range Monitor (RM-VG-174) - GREATER THAN General Emergency level (See Att. 4) OR Kaman monitor (RM-VG-179) reading greater than 7.2×10^7 uCi/sec	
	OR	
	d) Ventilation Vent B High Range Monitor (RM-VG-175) GREATER THAN General Emergency level (See Att. 4) OR Kaman Monitor (RM-VG-180) reading greater than 7.2×10^7 uCi/sec	
	OR	
	e) Process Vent High Range Monitor (RM-GW-173) GREATER THAN General Emergency level (See Att. 6) OR Kaman Monitor (RM-GW-178) reading greater than 7.2×10^7 uCi/sec	

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Transportation of contaminated injured individual to off-site facility ALL MODES	Contaminated injured individual enroute to off-site facility for treatment	NOTIFICATION OF UNUSUAL EVENT

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB G) LOSS OF SECONDARY COOLANT	15
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Major Secondary line break MODES 1,2,3 & 4	Loss of Secondary Coolant, in progress and verified non-spurious	NOTIFICATION OF UNUSUAL EVENT
2. Major Secondary line break with significant Primary to Secondary leakage MODES 1,2,3 & 4	Condition a) exists with b),c) or d) a) Loss of Secondary Coolant, in progress and verified non-spurious AND b) Condenser Air Ejector Radiation Monitor (RM-SV-121 or RM-SV-221) -GREATER THAN 1,000% (10 times limit) Tech Specs. (See Att. 5) OR c) Steam Generator Blow-down Radiation Monitor -GREATER THAN 10^5 cpm Monitor designations: RM-SS-122 RM-SS-123 RM-SS-124 RM-SS-222 RM-SS-223 RM-SS-224 OR d) MS Line High Range Radiation Monitor -GREATER THAN 0.01 mR/hr Monitor designations: RM-MS-170 RM-MS-270 RM-MS-171 RM-MS-271 RM-MS-172 RM-MS-272	ALERT

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB G) LOSS OF SECONDARY COOLANT	15
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
3. Major Secondary line break with significant Primary to Secondary leakage and fuel damage indicated MODES 1,2,3 AND 4	Conditions a) and b) exist with c), d) or e) a) Loss of Secondary Coolant, in progress and verified non-spurious AND b) RCS specific activity exceeds limits of T.S. Figure 3.4.1 (See Att. 8) OR Letdown High Range Radiation Monitor (RM-CH-128 or RM-CH-228) -GREATER THAN 10^5 cpm AND c) Condenser Air Ejector Radiation Monitor (RM-SV-121 or RM-SV-221) -GREATER THAN Site Area level (See Att. 5) OR d) Steam Generator Blowdown Radiation Monitor -GREATER THAN 10^6 cpm Monitor Designations: RM-SS-122 RM-SS-123 RM-SS-124 RM-SS-222 RM-SS-223 RM-SS-224 OR	SITE AREA EMERGENCY

Event 3 Continued on Next Page

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
3. (CONTINUED)	e) MS Line High Range Radiation Monitor -GREATER THAN 0.47 mR/hr Monitor Designations: RM-MS-170 RM-MS-171 RM-MS-172 RM-MS-270 RM-MS-271 RM-MS-272	

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Loss of off-site power or on-site AC power capability ALL MODES	a) Unit Main Generator and both Emergency Diesel Generators out of service OR b) Loss of all 34.5KV Reserve Station Service Buses	NOTIFICATION OF UNUSUAL EVENT
2. Loss of all off-site and on-site AC power ALL MODES	a) Ammeters for 4160V Reserve Station Service Buses D, E, & F all indicate-ZERO (0) AMPS AND b) Ammeters for 4160V Station Service Buses A,B,& C all indicate-ZERO (0) AMPS AND c) Ammeters for 4160V Emergency Buses H and J both indicate-ZERO (0) AMPS	ALERT
3. Loss of off site and on-site AC power for more than 15 minutes ALL MODES	The following conditions exist for a period of-GREATER THAN 15 minutes a) Ammeters for 4160V Reserve Station Service Buses D,E,& F all indicate-ZERO (0) AMPS AND b) Ammeters for 4160V Station Service Buses A,B & C all indicate-ZERO (0) AMPS AND	SITE AREA EMERGENCY

Event 3 Continued on next page

	CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
3.	(CONTINUED)	c) Ammeters for 4160V Emergency Buses H & J both indicate-ZERO (0) AMPS	
4.	Loss of all on-site DC power ALL MODES	a) All Station Battery voltmeters indicate-ZERO (0) VOLTS AND b) No light indication available to Reserve Station Service Breakers 15D1, 15E1 and 15F1	ALERT
5.	Loss of all on-site DC power for-GREATER THAN 15 minutes ALL MODES	The following conditions exist for a period of-GREATER THAN 15 minutes: a) All Station Battery voltmeters indicate-ZERO (0) VOLTS AND b) No light indication available to Reserve Station Service Breakers 15D1, 15E1 and 15F1	SITE AREA EMERGENCY

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NUMBER	ATTACHMENT TITLE	REVISION
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Fire lasting-GREATER THAN 10 minutes ALL MODES	Fire within the Station which is not under control within 10 minutes after fire fighting efforts begin	NOTIFICATION OF UNUSUAL EVENT
2. Fire potentially affecting station safety systems MODES 1,2,3,& 4	Fire within the Station which has potential for causing a safety system NOT to be operable as defined by T.S.1.17 and 3.0.5	ALERT
3. Fire resulting in degradation of safety systems MODES 1,2,3 & 4	a) Fire within the Station which causes major degradation of a safety system function required for protection of the public AND b) Affected systems are caused to be NOT operable as defined by T.S.1.17 and 3.0.5	SITE AREA EMERGENCY

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB J) SECURITY EVENT	15
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Security threat, unauthorized attempted entry, or attempted sabotage ALL MODES	Security Shift Supervisor has recommended Shift Supervisor declare an Unusual Event IAW applicable Security Contingency Plan Implementing Procedure.	NOTIFICATION OF UNUSUAL EVENT
2. Ongoing Security compromise ALL MODES	Security Shift Supervisor has notified the Shift Supervisor of a confirmed unneutralized intrusion into the Protected Area	ALERT
3. Imminent loss of physical Station control ALL MODES	Security Shift Supervisor has notified the Shift Supervisor of imminent intrusion into a Vital Area	SITE AREA EMERGENCY
4. Loss of Station physical control ALL MODES	a) Shift Supervisor has been informed that the security force has been neutralized by attack, resulting in loss of physical control of station OR b) Shift Supervisor has been informed of intrusion into one or more Vital Areas which are occupied OR controlled by an aggressor	GENERAL EMERGENCY

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Aircraft crash or unusual aircraft activity ALL MODES	a) Confirmed notification of aircraft crash within the site boundary OR b) Unusual aircraft activity in the vicinity of the site as determined by the Shift Supervisor AND/OR Security Shift Supervisor	NOTIFICATION OF UNUSUAL EVENT
2. Aircraft crash on the facility ALL MODES	a) Aircraft crash within the Protected Area OR b) Aircraft crash in Station Switchyard	ALERT
3. Aircraft damage to vital plant systems MODES 1,2,3 & 4	Aircraft crash which affects vital structures by impact or fire	SITE AREA EMERGENCY
4. Train derailment on site ALL MODES	Confirmed report of train derailment onsite	NOTIFICATION OF UNUSUAL EVENT
5. Onsite explosion ALL MODES	Confirmed report of unplanned explosion onsite	NOTIFICATION OF UNUSUAL EVENT
6. Explosion damage to facility ALL MODES	Unplanned explosion resulting in damage to plant structure or equipment	ALERT

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
7. Severe explosive damage MODE 1,2,3 & 4	Explosion which results in severe degradation of any of the following systems required for safe shutdown: a) CVCS System OR b) ECCS System OR c) Main/Auxiliary Feedwater System	SITE AREA EMERGENCY
8. On or near site release of toxic or flammable liquids or gases ALL MODES	Notification of unplanned release of toxic OR flammable agents which may affect safety of station personnel OR equipment	NOTIFICATION OF UNUSUAL EVENT
9. Entry of toxic or flammable gases or liquids into plant facility ALL MODES	Notification of uncontrolled release of toxic OR flammable agent which cause: a) Evacuation of personnel from plant areas AND b) Safety related equipment is rendered inoperable	ALERT
10. Entry of toxic or flammable gases into plant vital areas MODE 1,2,3 & 4	a) Uncontrolled release of toxic OR flammable agents GREATER THAN life threatening or explosive limits in Vital Areas AND	SITE AREA EMERGENCY

Item 10 continued on next page

NUMBER	ATTACHMENT TITLE	REVISION
EPIP-1.01	EMERGENCY ACTION LEVEL TABLE (TAB K) HAZARD TO STATION OPERATION	15
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
10. (CONTINUED)	b) Evacuation of Vital Area required OR c) Significant degradation of plant safety systems resulting in loss of a safety system function required for protection of the public	
11. Turbine rotating component failure with no casing penetration MODES 1 & 2	Failure o' Turbine/ Generator rotating equipment resulting in immediate unit shutdown	NOTIFICATION OF UNUSUAL EVENT
12. Turbine failure or missile impact MODES 1 & 2	Failure of Turbine/ Generator rotating equipment resulting in casing penetration	ALERT
13 Missile damage to safety related equipment or structures MODES 1,2,3 & 4	Notification of missile impact causing damage to safety related equipment or structures	ALERT
14. Severe missile damage to safety systems MODES 1,2,3 & 4	Missile impact causing severe degradation of safety systems required for unit shutdown	SITE AREA EMERGENCY

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Earthquake detected ALL MODES	Confirmed earthquake which activates the Event Alarm on the Strong Motion Accelerograph	NOTIFICATION OF UNUSUAL EVENT
2. Earthquake greater than OBE levels ALL MODES	a) Confirmed earthquake which activates Event Alarm on the Strong Motion Accelerograph AND b) Alarms on the Peak Shock Annunciator indicate a horizontal motion of-GREATER THAN or EQUAL TO 0.06 g or a vertical motion of-GREATER THAN or EQUAL TO 0.04 g	ALERT
3. Earthquake greater than OBE levels MODES 1,2,3 & 4	a) Earthquake which activates the Event Alarm on the Strong Motion Accelerograph AND b) Alarms on the Peak Shock Annunciator indicates a horizontal motion of GREATER THAN or EQUAL TO 0.12 g or a vertical motion of-GREATER THAN or EQUAL TO 0.08 g	SITE AREA EMERGENCY
4. Tornado onsite ALL MODES	Tornado visually detected onsite	NOTIFICATION OF UNUSUAL EVENT
5. Tornado striking facility ALL MODES	Tornado visually detected striking within the Protected Area or Switchyard	ALERT

	CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION.
6.	Hurricane "WARNING" OR hurricane force winds projected onsite within 12 hours. ALL MODES	Confirmation by Air Quality/ Meteorological Dept. that Hurricane "WARNING" in effect for Louisa County OR hurricane force winds (GREATER THAN 73 mph) projected onsite within 12 hours.	NOTIFICATION OF UNUSUAL EVENT
7.	Hurricane "WARNING" AND hurricane force winds projected onsite within 6 hours. ALL MODES	Confirmation by AIR Quality/ Meteorological Dept. that Hurricane "WARNING" in effect for Louisa County AND hurricane force winds (GREATER THAN 73 mph) projected onsite within 6 hours.	ALERT
8.	Extreme winds above Design Basis Conditions OF 80 MPH. ALL MODES	Extreme winds confirmed onsite which exceed UFSAR Section 3.3.1 conditions (80 mph).	SITE AREA EMERGENCY
9.	50 year flood or low water level ALL MODES	a) Flood in the Lake Anna Reservoir with indicated level-GREATER THAN 254 feet MSL OR b) Low water level in the Lake Anna Reservoir with indicated level-LESS THAN 247 feet MSL	NOTIFICATION OF UNUSUAL EVENT

CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
10. Flood or low water level near design levels ALL MODES	a) Flood in the Lake Anna Reservoir with indicated level-GREATER THAN 263 feet MSL OR b) Low water level in the Lake Anna Reservoir with indicated level-LESS THAN 245 feet MSL	ALERT
11. Flood or low water level above design levels MODES 1,2,3 & 4	a) Flood in the Lake Anna Reservoir with indicated level-GREATER THAN 264 feet MSL OR b) Low water level in the Lake Anna Reservoir with indicated level-LESS THAN 244 feet MSL	SITE AREA EMERGENCY

NUMBER	ATTACHMENT TITLE	REVISION
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CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
1. Station conditions which warrant increased awareness of state and/or local authoriti ALL MODES	Shift supervisor judgement that any of the following exist: a) Intentional reduction in power, load, or temperature because the unit has entered an Action Statement or will exceed an LCO. NOTE: In the event other plant conditions require a shutdown, the NOUE must still be declared on the basis that a shutdown would have been required by the T.S. OR b) Unit shutdown is other than a controlled shutdown OR c) Unit is in an uncontrolled condition during operation OR d) A condition exists which has the potential for escalation and, therefore, warrants notification	NOTIFICATION OF UNUSUAL EVENT
2. Station conditions which warrant precautionary notification to the near-site public ALL MODES	Shift Supervisor/Station Emergency Manager judgement	ALERT

	CONDITION/APPLICABILITY	INDICATION	CLASSIFICATION
3.	Station conditions which warrant activation of emergency facilities monitoring teams or precautionary notification to the near-site public ALL MODES	Shift Supervisor/Station Emergency Manager judgement	SITE AREA EMERGENCY
4.	Any major internal or external events which singly or in combination cause massive damage to station facilities ALL MODES	Shift Supervisor/Station Emergency Manager judgement	GENERAL EMERGENCY

ENCLOSURE 3

Attachment
Page 1
09/27/90

NORTH ANNA POWER STATION
COMMENTS ON WRITTEN NRC EXAMINATIONS
ADMINISTERED ON SEPTEMBER 24, 1990

A. Reactor Operator Examination

1. QUESTION: 015 (1.00) (same as SRO 020)

Assuming no other changes to reactivity, select the ONE (1) choice that correctly fills the blanks describing the reactivity effects xenon has on reactor power at the times listed following a power increase after a two week period of operating at 50% reactor power.

At 1 HOUR reactor power is [1] ______________.
At 5 HOURS reactor power is [2] ______________.
At 12 HOURS reactor power is [3] ______________.
At 20 HOURS reactor power is [4] ______________.

a. [1] increasing, [2] increasing, [3] decreasing, [4] decreasing

b. [1] increasing, [2] increasing, [3] increasing, [4] decreasing

c. [1] decreasing, [2] increasing, [3] increasing, [4] decreasing

d. [1] decreasing, [2] decreasing, [3] increasing, [4] increasing

ANSWER: d

COMMENT: In the power range, the overall power of the reactor is controlled by steam demand. This is due to the inherent stability of the reactor that is caused by the negative moderator temperature coefficient. For example, a xenon decrease will add positive reactivity to the reactor, that will momentarily cause a slight increase in reactor power. This will cause reactor power to momentarily be greater than steam demand, which will cause the moderator's temperature to increase. The increase in moderator temperature will add negative reactivity to offset the positive reactivity added by the decrease in xenon. Consequently, the power of the reactor does not noticeably change, but, instead, the reactor coolant's temperature rises. Xenon changes will not cause a change in the overall power of the reactor. In this respect, the question is flawed since it asks how the xenon reactivity change affects reactor power.

Furthermore, the question is still in error, even if one assumes that the moderator temperature coefficient is equal to zero. After one hour following an increase in reactor power, the xenon concentration will be decreasing. This will be adding positive reactivity to the core. This will tend to increase reactor power, which will also increase reactor coolant temperature, but will not, in this case, add negative reactivity since the moderator temperature coefficient is assumed to be zero. So, after one hour, reactor power would be increasing, not decreasing as specified by the answer key. Therefore, even if the moderator temperature were zero, the answer key is in error.

Lastly, the xenon transient depends upon the magnitude of the power change, and the core burnup. Neither of these quantities is specified by the question. Consequently, it is not possible to determine what the xenon concentration is doing at 5 hours after the power change. 5 hours is very close to the minimum xenon concentration that occurs following a large power increase. Whether or not the xenon concentration is increasing, or decreasing, or even at the minimum at 5 hours is dependent upon the magnitude of the power change and the core burnup, which are not specified.

RECOMMENDATION: Delete the question.

STATUS AFTER PRE-REVIEW: This was not identified by the utility during the preview.

REFERENCE: NCRODP-86.2 Section 4

2. QUESTION: 041 (1.00) (same as SRO 044)

If the turbine controlling first stage pressure detector fails LOW, HOW will this failure affect the RCS?

a. RCS delta-T across the Steam Generator tubes will decrease.

b. RCS Tavg and pressure will increase.

c. RCS temperature and pressure will not change.

d. RCS delta-T across the Steam Generator tubes will increase.

ANSWER: a

COMMENT: The stem is poorly worded. "the controlling first stage pressure" is too ambiguous. This could mean either PT-446, 447 or PT-132 (EHC). This could result in a, c, d being correct.

If PT-446 (selected channel) fails answer (a) or (d) could be correct depending on the response considered for rod control, SGWLC, and VPL position.

If PT-132 fails the answer would be (c). In "Imp In" the control would auto shift to "Imp Out" with no problems. In "Imp Out" no effects would occur.

RECOMMENDATIONS: Delete the question.

STATUS AFTER PRE-REVIEW: Corrections during prereview did not adequately clarify question.

REFERENCE: NCRODP-89.1; NCRODP-89.5

3. QUESTION: 089 (1.00) (same as SRO 087)

Assume a pressurizer spray line has ruptured and raises containment temperature from 100 F to 180 F. How will indicated pressurizer level compare to actual pressurizer level?

a. Indicated level will be lower than actual level.

b. Indicated level will be equal to actual level.

c. Indicated level will decrease to zero.

d. Indicated level will be higher than actual level.

ANSWER: c

COMMENT: The answer key is incorrect. Under these conditions PZR indicated level will be higher than actual level.

RECOMMENDATIONS: Change answer key to make (d) the correct answer.

STATUS AFTER PRE-REVIEW: Corrections during prereview did not adequately clarify question.

REFERENCE: NCRODP-95.2 Section 10
PWR OPERATOR GENERIC FUNDAMENTALS TEST ITEM CATALOG, Sensors/Detectors Questions #29

4. QUESTION: 099 (1.00) (same as SRO 099)

Which ONE (1) of the following is the correct action in response to a SFP level reduced 1 inch below normal that was caused by a refueling gate failure as per 1-AP-27.1, Loss of Spent Fuel Pool Level immediate actions?

a. Monitor radiation in the Fuel Building.

b. Evacuate the fuel building.

c. Initiate normal make-up to the SFP if level still decreasing.

d. Close the transfer canal gate valve.

ANSWER: c

COMMENT: There is no correct answer. According to 1-AP-27.1 immediate actions, normal make-up to the SFP is initiated if SFP level is NOT decreasing.

RECOMMENDATION: Delete the question.

STATUS AFTER PRE-REVIEW: Corrections during prereview did not adequately clarify question.

REFERENCE: 1-AP-27.1

NORTH ANNA POWER STATION
COMMENTS ON WRITTEN NRC EXAMINATIONS
ADMINISTERED ON SEPTEMBER 24, 1990

B. Senior Reactor Operator Examination

1. QUESTION: 020 (1.00) (same as RO 015)

Assuming no other changes to reactivity, select the ONE (1) choice that correctly fills the blanks describing the reactivity effects xenon has on reactor power at the times listed following a power increase after a two week period of operating at 50% reactor power.

At 1 HOUR reactor power is [1] ______________.
At 5 HOURS reactor power is [2] ______________.
At 12 HOURS reactor power is [3] ______________.
At 20 HOURS reactor power is [4] ______________.

a. [1] increasing, [2] increasing, [3] decreasing, [4] decreasing

b. [1] increasing, [2] increasing, [3] increasing, [4] decreasing

c. [1] decreasing, [2] increasing, [3] increasing, [4] decreasing

d. [1] decreasing, [2] decreasing, [3] increasing, [4] increasing

ANSWER: d

COMMENT: In the power range, the overall power of the reactor is controlled by steam demand. This is due to the inherent stability of the reactor that is caused by the negative moderator temperature coefficient. For example, a xenon decrease will add positive reactivity to the reactor, that will momentarily cause a slight increase in reactor power. This will cause reactor power to momentarily be greater than steam demand, which will cause the moderator's temperature to increase. The increase in moderator temperature will add negative reactivity to offset the positive reactivity added by the decrease in xenon. Consequently, the power of the reactor does not noticeably change, but, instead, the reactor coolant's temperature rises. Xenon changes will not cause a change in the overall power of the reactor. In this respect, the question is flawed since it asks how the xenon reactivity change affects reactor power.

Furthermore, the question is still in error, even if one assumes that the moderator temperature coefficient is equal to zero. After one hour following an increase in reactor power, the xenon concentration will be decreasing. This will be adding positive reactivity to the core. This will tend to increase reactor power, which will also increase reactor coolant temperature, but will not, in this case, add negative reactivity since the moderator temperature coefficient is assumed to be zero. So, after one hour, reactor power would be increasing, not decreasing as specified by the answer key. Therefore, even if the moderator temperature were zero, the answer key is in error.

Lastly, the xenon transient depends upon the magnitude of the power change, and the core burnup. Neither of these quantities is specified by the question. Consequently, it is not possible to determine what the xenon concentration is doing at 5 hours after the power change. 5 hours is very close to the minimum xenon concentration that occurs following a large power increase. Whether or not the xenon concentration is increasing or decreasing, or even at the minimum at 5 hours is dependent upon the magnitude of the power change and the core burnup, which are not specified.

RECOMMENDATION: Delete the question.

STATUS AFTER PRE-REVIEW: This was not identified by the utility during the preview.

REFERENCE: NCRODP-86.2 Section 4

2. QUESTION: 044 (1.00) (same as RO 041)

If the turbine controlling first stage pressure detector fails LOW, HOW will this failure affect the RCS?

a. RCS delta-T across the Steam Generator tubes will decrease.

b. RCS Tavg and pressure will increase.

c. RCS temperature and pressure will not change.

d. RCS delta-T across the Steam Gener. ~r tubes will increase.

ANSWER: a

COMMENT: The stem is poorly worded. "the controlling first stage pressure" is too ambiguous. This could mean either PT-446, 447 or PT-132 (EHC). This could result in a, c, d being correct.

If PT-446 (selected channel) fails answer (a) or (d) could be correct depending on the response considered for rod control, SGWLC, and VPL position.

If PT-132 fails the answer would be (c). In "Imp In" the control would auto shift to "Imp Out" with no problems. In "Imp Out" no effects would occur.

RECOMMENDATIONS: Delete the question.

STATUS AFTER PRE-REVIEW: Corrections during prereview did not adequately clarify question.

REFERENCE: NCRODP-89.1; NCRODP-89.5

3. QUESTION: 087 (1.00) (same as RO 089)

Assume a pressurizer spray line has ruptured and raises containment temperature from 100 F to 180 F. How will indicated pressurizer level compare to actual pressurizer level?

a. Indicated level will be lower than actual level.

b. Indicated level will be equal to actual level.

c. Indicated level will decrease to zero.

d. Indicated level will be higher than actual level.

ANSWER: c

COMMENT: The answer key is incorrect. Under these conditions PZR indicated level will be higher than actual level.

RECOMMENDATIONS: Change answer key to make (d) the correct answer.

STATUS AFTER PRE-REVIEW: Corrections during prereview did not adequately clarify question.

REFERENCE: NCRODP-95.2 Section 10
PWR OPERATOR GENERIC FUNDAMENTALS TEST ITEM CATALOG, Sensors/Detectors Questions #29

4. QUESTION: 099 (1.00) (same as RO 099)

Which ONE (1) of the following is the correct action in response to a SFP level reduced 1 inch below normal that was caused by a refueling gate failure as per 1-AP-27.1, Loss of Spent Fuel Pool Level immediate actions?

a. Monitor radiation in the Fuel Building.

b. Evacuate the fuel building.

c. Initiate normal make-up to the SFP if level still decreasing.

d. Close the transfer canal gate valve.

ANSWER: c

COMMENT: There is no correct answer. According to 1-AP-27.1 immediate actions, normal make-up to the SFP is initiated if SFP level is NOT decreasing.

RECOMMENDATION: Delete the question.

STATUS AFTER PRE-REVIEW: Corrections during prereview did not adequately clarify question.

REFERENCE: 1-AP-27.1

3. Other Xe-135 reactivity transients

Changing reactor power causes a change in xenon concentration and a change in reactivity. In this section we will discuss the effects of power level changes from other than 0% power.

a. Increase in power level

Immediately after a power level change, burnout of Xe-135 increases, and production directly from the fission process increases slightly. Production from I-135 does not increase instantaneously, since it takes some time for I-135 itself to increase its concentration. The result is that immediately following a power increase, Xe-135 concentration decreases.

As Xe-135 concentration decreases, xenon decay and burnout decrease. At the same time, I-135 is building up and Xe-135 production from I-135 decay increases. When this occurs, production exceeds removal and Xe-135 concentration increases.

As Xe-135 concentration increases, the burnout and decay rates increase until eventually production equals removal at a higher Xe-135 concentration.

Display transparency 4.8. This illustrates several Xe-135 reactivity transients.

Point out the following characteristics on the curve:

1) The minimum Xe-135 concentration reached is dependent on the magnitude of the power level change and initial power level.

2) The time to reach the minimum Xe-135 concentration is always less than 11 hours.

3) The time to reach equilibrium is also dependent on the magnitude of the power change and final power level.

4) The time to reach equilibrium is approximately 40-50 hours after the power change.

---

b. Decrease in power

Have trainees think about how Xe-135 reactivity changes when power decreases (not a reactor trip). Ask trainees to each draw a curve of the reactivity transient.

Display transparency 4.9. The behavior of Xe-135 is similar to that after a reactor trip, except now equilibrium Xe-135 will be established for the new power level after 40-50 hours.

---

c. Decrease followed by an increase in power

e. IMP-OUT/IMP-IN Bumpless transfer

Use T-2.5 for explanation of the following events.

(1) At approximately 15% power, Turbine impulse pressure is used to aid the Turbine control system in maintaining a smooth linear response to a desired load change. Impulse pressure is used to give a more linear response during load changes vs. the non-linear response using Governor valve position.

(2) Using Governor valve position discontinuities occurs at various valve points (where one valve approaches its steam flow capability and the next valve in sequence starts to open). As a result, in the vicinity of the valve points, an incremental change in load reference does not produce a corresponding incremental change in load. This is eliminated using impulse pressure.

Use T-2.5 and T-2.6 for discussion of use of impulse pressure.

(3) When it is desired to transfer to IMP-IN the operator depresses the pushbutton. At this point the Turbine control transfers to pseudo-manual mode.

(4) Pseudo-manual mode holds the Governor valves at the position they were at prior to depressing the

pushbutton while the Auto circuit places the impulse pressure in service.

(5) The additional of the impulse pressure changes the output of the speed error/reference counter comparator which is now being compared to the manual signal holding the valves in their original position. The reference counter adjusts until the Auto is matched with the manual.

(6) When the Auto matches manual the Auto takes over and valves stay where they are. This is called a bumpless transfer.

(7) The reference/setter are now indicating the first stage pressure in %.

(8) The use of the impulse pressure also makes the Governor valve controller a proportional + integral controller.

Use T-2.5 and 2.6 to describe the following events.

f. Load control (IMP-IN)

(1) Used from approximately 15% to 100% power.

(2) Describe how changes in system load and changes in impulse pressure effect system

C. Turbine Supervisory Panel

(b) If the load reference from the controller differs from the turbine impulse pressure by 20%.

(c) Shifts turbine control from oper auto to turbine manual.

4. Load Channel

(a) Loss of turbine first stage impulse pressure transducer.

(b) This does not shift the turbine control to manual but does shift from IMP-IN to IMP-OUT.

5. Overspeed Protection Control

(a) Comes in when the OPC solenoid valves operate.

(b) Failure of either of the following transducers:

(1) M.W. Transducer

(2) Reheat Pressure Transducer

(3) A loss of these would give a loss of necessary control inputs should a load rejection occur.

(c) If Reheat Pressure is MW light comes on indicating a partial load loss.

(d) This light is normally on when the turbine is shutdown because the transducers are out of operating range until unit assumes about 5% power.

6. Emergency Power Supply

2. P-13 (Turbine impulse pressure '10%) which supplies P-7. 2/2 with reactor power '10% block numerous Rx trips.

   "TURB PWR IMP PRESS INTLK '10% P13 CH III" (1M-B5,6)

   "TURB PWR IMP PRESS '10% POWER P13 PERM (1P-D1)

3. C-7, time derivative of PT-1447 only, decrease 10% in 2 minutes, send arming signal to steam dumps.

   "STEAM DUMP ARMED FROM LOSS OF LOAD C-7" (1P-F7)

b. Selected turbine first stage pressure (PT-446 or 447):

   1. Is used as an input to control feedwater regulating valves.

   2. Form C-5 permissive (Turbine impulse pressure '15% which blocks auto rod withdrawal)

      "TURB PWR 15% INTLK AUTO ROD BLOCK" (1P-E2)

   3. Used to control rod speed.

4. Used to generate T_{ref} for comparison with RCS temp. T_{ref} is displayed on the MCB (530°-630°F), a recorder (530°-630°f) and sent to the computer.

c. Analog signals are sent to meters and displayed on the MCB. PI-1446 and PI-1 7 both have 0-600 psig range.

13. H.P. Turbine First Stage Pressure (PT-MS-110-1,2)

a. Feeds a computer input and two 0-1000 psig recorders on the vertical board.

14. H.P. Turbine Impulse Chamber Pressure (PT-MS-132)

a. This signal is used by the EH Governor Controller.

15. H.P. Turbine Exhaust Pressure (PT-MS-105A-D)

a. Input to computer. Sensed on crossunder piping.

16. Anti-Motoring Trip (PS-MS-134 or PS-63AM1)

a. Pressure is sensed at the impulse chamber and H.P. turbine exhaust. When ΔP drops below 7.2 psid, a trip signal is generated, providing the generator breaker is closed.

17. Steam Dump Header Pressure (PT-MS-108A,B)

a. Is sensed off of each 18 inch line which feeds the steam dumps and reheat steam to determine the indicated pressurizer level verse actual level. Similarly, four case studies were performed for steam generator level.

(1) case 1 - normal containment conditions with no reference leg heatup.

(2) case 2 - normal containment conditions, but reference leg heated to ~~280~~ 200 F.

(3) case 3 - adverse containment conditions with no reference leg heating.

(4) case 4 - adverse containment conditions with reference leg heated to 280 F.

e. Our EPs utilize the values acquired in the case 2 and 4 studies, to be conservation. As an example, to verify that a water level is present in the pressurizer our EPs utilize values of 15 percent (normal containment) and 50 percent (adverse containment). The 15 percent is derived from case study 2, assuming pressurizer pressure equal to 2335 psig (the PORV setpoint). When actual level was 0 percent, indicated level was 12.4 percent. To allow for human factors the setpoint chosen was 15 percent. In case study 4, indicated level was 47 percent when actual level was 0 percent, again assuming pressure equal to 2335 psig. The setpoint was chosen as 50 percent for human factors considerations.

f. In general, the greatest non-conservative errors between indicated and actual pressurizer level, for

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NATIONAL ACADEMY FOR NUCLEAR TRAINING

PWR OPERATOR

GENERIC FUNDAMENTALS TEST ITEM CATALOG

Plant Area: Training, Operations
Key Words: Training, Operator, Examination

23. Two differential pressure level transmitters are installed in a large tank. Transmitter I is calibrated at 200°F. Transmitter II is calibrated at 100°F. Which transmitter will read lower at 150°F?

    A. transmitter I

    B. transmitter II

    C. must consult water density curve to accurately determine

    D. neither, they will read the same

24. The theory of operation of a differential pressure level detector using a wet reference leg is best described by which of the following statements? The pressure differential between a

    A. reference leg and a variable leg is directly proportional to the height of the variable leg

    B. reference leg and a variable leg is inversely proportional to the height of the variable leg

    C. variable leg and a reference leg is directly proportional to the density of the variable leg

    D. variable leg and a reference leg is inversely proportional to the temperature of the reference leg

25. A level differential pressure transmitter has been calibrated for a density of 62 lbm/ft^3. However, the actual tank liquid density is 40 lbm/ft^3. The indicated tank level will be

    A. more than actual level
    B. less than actual level
    C. the same as actual level
    D. erratic

26. The differential pressure level detector senses the differential pressure between a reference height of liquid and the pressure at the bottom of a tank. This differential pressure is ________ the level of liquid in the tank.

    A. an integral of
    B. a differential of
    C. directly proportional to
    D. inversely proportional to

27. The differential pressure type level detector senses the differential pressure between a ________ height of liquid and a column of liquid at a fixed height.

    A. programmed
    B. backup
    C. reference
    D. variable

28. A differential pressure type level detector senses the differential pressure between a reference height of liquid and

    A. atmospheric pressure
    B. programmed pressure
    C. the pressure at the top of a tank
    D. the pressure at the bottom of a tank

29. A high ambient temperature at its reference leg may cause a differential pressure type level indicator using a wet reference leg to read

    A. greater than actual level
    B. less than actual level
    C. at a constant level
    D. at a fluctuating lower level

SENSORS/DETECTORS
Answers

22. B

Reference 31, chapter 5, page 59.

23. B

Reference 31, chapter 5, page 59.

24. B

The D/P cell is the most commonly used level sensor. If the pressure generated by a known (fixed) height of liquid is compared to the pressure generated at the bottom of a tank, the difference of these pressures is inversely proportional to level of liquid in the tank.

Reference 53, chapter 2, page 87.

25. B

If the density of a liquid in a tank is less than one, then more water (volume) is needed in the tank as compared to liquid with a density of one to have the same level indication.

Reference 53, chapter 2, page 88.

26. D

The D/P cell is the most commonly used level sensor. If the pressure generated by a known (fixed) height of liquid is compared to the pressure generated at the bottom of a tank, the difference of these pressures is inversely proportional to level of liquid in the tank.

Reference 53, chapter 2, page 87.

27. D

The D/P cell is the most commonly used level sensor. If the pressure generated by a known (fixed) height of liquid is compared to the pressure generated at the bottom of a tank, the difference of these pressures is inversely proportional to level of liquid in the tank.

Reference 53, chapter 2, page 87.

28. D

The D/P cell is the most commonly used level sensor. If the pressure generated by a known (fixed) height of liquid is compared to the pressure generated at the bottom of a tank, the difference of these pressures is inversely proportional to level of liquid in the tank.

Reference 53, chapter 2, page 87.

29. A

An increase in ambient temperature will cause the density of the reference leg of a D/P cell to decrease. This results in a lower Δ P sensed by the D/P cell, which results in a higher indicated level.

Reference 53, chapter 2, page 87.

30. C

Increasing ambient pressure has no effect on D/P cell level instruments because they are sealed.

Reference 53, chapter 2, page 88.

RO/SRO 099
VIRGINIA POWER
NORTH ANNA POWER STATION

ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-27.1	LOSS OF SPENT FUEL POOL LEVEL	1
	(With one Attachment)	PAGE 1 of 5

PURPOSE

To provide instructions in the event of a loss of SFP level.

ENTRY CONDITIONS

This procedure is entered when any of the following conditions exists:

- Transition from Annunciator Response "E" Panel C-6 (Unit 1), or
- Transition from 1-AP-5.2, COMMON RADIATION MONITORING SYSTEM, upon receipt of high radiation readings, or
- Transition from 1-AP-27.2, LOSS OF SPENT FUEL COOLING SYSTEM, upon loss of Spent Fuel cooling, or
- Loss of SFP level as indicated by observation.

NOT A CONTROLLED DOCUMENT
JUL 10 1990
NOR NECESSARILY THE L[illegible] [illegible]VISION

APPROVAL RECOMMENDED	APPROVED	DATE
[signature]	[signature]	5/31/90
REVIEWED [signature]	CHAIRMAN STATION NUCLEAR SAFETY AND OPERATING COMMITTEE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1]	__STOP ANY RUNNING SFP COOLING PUMP	
[2]	__STOP ANY RUNNING RP PUMP	
[3]	__CHECK SFP LEVEL - DECREASING	Restore SFP level as follows: a) Determine cause for the low level. b) Make up to the SFP using 1-OP-16.5, MAKEUP FOR SPENT FUEL POOL FROM THE UNIT 1 BLENDER, <u>OR</u> 2-OP-16.5, MAKEUP FOR SPENT FUEL POOL FROM THE UNIT 2 BLENDER. c) Start the desired SFP Cooling pump using 1-OP-16, SPENT FUEL PIT COOLING AND PURIFICATION SYSTEM. d) Terminate this procedure. GO TO Step 7.
[4]	__CHECK FUEL HANDLING ACTIVITIES - IN PROGRESS	
[5]	__RETURN FUEL ASSEMBLY TO SFP: • Place assembly in nearest open storage location • Note location used:______	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: Closing the transfer tube gate valve with the transfer cart in the Fuel Building will result in damage to the transfer system.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	__LOCALLY CHECK UNIT 1 AND 2 TRANSFER TUBE GATE VALVES - CLOSED	Isolate the transfer tubes as follows: a) Verify OR send the transfer cart to containment. b) Close any open transfer gate valve.
7.	__LOCALLY VERIFY TRANSFER CANAL AND CASK AREA GATE SEALS - INTACT	Initiate 1-AP-27.3, PLACING NITROGEN BACKUP IN SERVICE TO SFP GATE SEALS, to place nitrogen backup system in service.
8.	__PERFORM THE FOLLOWING NOTIFICATIONS: • Health Physics • SRO on call	
9.	__CHECK HIGH VOLUME MAKEUP - REQUIRED	Make up to the SFP using 1-OP-16.5, MAKEUP FOR SPENT FUEL POOL FROM THE UNIT 1 BLENDER, OR 2-OP-16.5, MAKEUP FOR SPENT FUEL POOL FROM THE UNIT 2 BLENDER, AND GO TO Step 12.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.	__INITIATE HIGH VOLUME MAKEUP:	
	a) Obtain permission from Superintendent of Operations or SRO on call to initiate makeup using Fire Main	a) Make up to SFP from Unit 1 OR 2 RWST as directed by the Shift Supervisor.
	b) Locally Close 1-FP-56, PIV Isol. Valve Emergency Fill of Spent Fuel Pit	
	c) Locally Open 1-FP-54, PIV to Fuel Bldg. and Emergency Fill of Fuel Pit	
	d) Locally Open 1-FP-56, PIV Isol. Valve Emergency Fill of Spent Fuel Pit	
	e) Locally Verify Fire Main flow	e) Make up to SFP from Unit 1 OR 2 RWST as directed by the Shift Supervisor.
11.	__EVACUATE FUEL BUILDING	
12.	__MONITOR FUEL BUILDING RADIATION • 1-RMS-RI-152 • 1-RMS-RI-153	
13.	__INITIATE EPIP-1.01, EMERGENCY MANAGER CONTROLLING PROCEDURE	
14.	__IDENTIFY AND STOP THE CAUSE OF INVENTORY LOSS	
15.	__CHECK SFP LEVEL - STABLE OR INCREASING	RETURN TO Step 14.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16.	__CONTROL MAKEUP SOURCE TO MAINTAIN SFP LEVEL	
17.	__CHECK SFP LEVEL - ABOVE SFP COOLING PUMP SUCTION • Start the desired SFP Cooling Pump using 1-OP-16.1, SPENT FUEL PIT COOLING AND PURIFICATION SYSTEM • Return the RP system to service using 1-OP-16, SPENT FUEL PIT COOLING AND PURIFICATION SYSTEM	Do <u>NOT</u> start SFP Cooling Pumps <u>OR</u> RP Pumps. RETURN TO Step 16 when SFP level is above SFP Cooling Pump suction
18.	__CHECK SFP BORON CONCENTRATION - GREATER THAN 2300 PPM	Borate the SFP using 1-OP-16.5, MAKEUP FOR SPENT FUEL POOL FROM THE UNIT 1 BLENDER, <u>OR</u> 2-OP-16.5, MAKEUP FOR SPENT FUEL POOL FROM THE UNIT 2 BLENDER.
19.	__VERIFY INVENTORY LOSS - CORRECTED	RETURN TO Step 14.
20.	__SECURE SFP MAKEUP a) High volume makeup in service • Stop makeup from RWST <u>OR</u> • Stop makeup from Fire Main by closing 1-FP-54, PIV to Fuel Bldg. and Emergency Fill of Fuel Pit	a) Secure SFP makeup using 1-OP-16.5, MAKEUP FOR SPENT FUEL POOL FROM THE UNIT 1 BLENDER, <u>OR</u> 2-OP-16.5, MAKEUP FOR SPENT FUEL POOL FROM THE UNIT 2 BLENDER.
21.	__RETURN TO PROCEDURE IN EFFECT	

- END -

- 10-18-88 R.L. RASNIC MEMO, REVIEW OF NORTH ANNA FUEL POOL TRANSFER DOOR SEAL DESIGN
- UFSAR
- Tech Spec 3.9.11

ENCLOSURE 4

NRC Resolution of Facility Comments

RO Examination

Question: 015

NRC resolution: Comment accepted. Item deleted.

Question: 041

NRC resolution: Comment accepted. Item deleted.

Question: 089

NRC resolution: Comment accepted. Answer Key Changed to make (d) the correct answer.

Question: 099

NRC resolution: Comment accepted. Item deleted.

SRO Examination

Question: 020

NRC resolution: Comment accepted. Item deleted.

Question: 044

NRC resolution: Comment accepted. Item deleted.

Question: 087

NRC resolution: Comment accepted. Answer Key Changed to make (d) the correct answer.

Question: 099

NRC resolution: Comment accepted. Item deleted.

ENCLOSURE 5

SIMULATOR FACILITY FIDELITY REPORT

Facility Licensee: North Anna Nuclear Station

Facility Docket No.: 50-338, 50-339

Operating Tests Administered on: September 25 - 27, 1990

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating tests, the following item was observed:

> On the test run-through of scenario 1-1, a 43.5 gpd steam generator tube leak was set in as an initial condition. The simulator responded as expected and as desired. On the following day the same malfunction was inserted. This time, the setpoint drifted to 46 gpd. This subsequently locked in the N-16 alarm which resulted in an undesirable distractor. This prolonged the scenario due to the candidate's attention being drawn away. The intent of the scenario's first event was to conduct a smooth ramp down in power to 75 percent without being hampered.