U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.	50-443
Report No.	96-09
Licensee:	North Atlantic Energy Service Corporation
Facility:	Seabrook Station
Dates:	September 30 - October 4, 1996
Examiners:	Joseph D'Antonio, Operations Engineer Steve Barr, Operations Engineer
Approved by:	Glenn W. Meyer, Chief Operator Licensing and Human Performance Branch Division of Reactor Safety

EXECUTIVE SUMMARY

Seabrook Power Station
NRC Inspection Report 50-443/96-09

Operations

From September 30 - October 4, 1996, two examiners administered initial licensing examinations to four senior reactor operator upgrade (SROU) candidates and one senior reactor operator instant (SROI) candidate at Seabrook. This examination had been developed by the facility training department in accordance with NRC guidance. It was then approved, administered, and graded by NRC.

All candidates passed their examination.

The facility did a good job in developing the examination.

In the simulator, communications were generally excellent, STAR practices were evident, and the SROs assigned tasks effectively.

One weakness was noted in both the simulator and walkthroughs in that candidates had difficulty deciding that "local manual" action was appropriate when "local powered" actions were unsuccessful. Also, the lack of a procedure for a leak in a secondary system within containment appeared to delay effective mitigating actions.

Report Details

I. Operations

O5 Operator Training and Qualifications

O5.1 Operator Initial Examinations

a. Scope

The facility staff developed the written and operating examinations and submitted these proposed examinations for NRC review and approval. The NRC reviewed the proposed examinations, provided comments, and approved the examination for administration. The examinations administered reflected incorporation of the NRC comments.

b. Observations and Findings

The facility did a good job of exam development. The NRC had minor comments on approximately ten percent of the written exam. One proposed JPM (job performance measure) was not suitable as a system JPM, but was used in the administrative exam instead. Two JPM questions were replaced, and one event was added to two scenarios.

Summary of Results

	SRO Pass/Fail
Written	5 / 0
Simulator	5 / 0
Walk-through	5 / 0
Overall	5 / 0

Operating Examinations

Command and control and overall communications were strong. The crews did a generally excellent job of ensuring that communications were clearly delivered and acknowledged. When on the board, good STAR (stop, think, act, review) practices by the candidates were evident. When in the SRO position, the candidates did a good job of assigning particular tasks to a specific board individual to move activity along without confusion.

One area of procedural confusion was noted in a station blackout scenario in which the output breaker for an operating diesel was not able to be closed locally from the diesel control panel. The crew spent several minutes deciding to continue on in the

procedure, and never did order the breaker locally MANUALLY closed. The examiners expected that this would be attempted; however, the facility makes a distinction between "local" and "local manual" operations. The procedure directs a local, but not local manual breaker closure. This same confusion was apparent when one candidate took an excessive amount of time to complete a JPM requiring local operation of a diesel output breaker. From the time the candidate was directed to locally close the breaker, it took 10 minutes for the candidate to decide that local MANUAL closure was required.

Also, the examiners noted that in response to a feed leak in the containment, the facility had no procedure for a small secondary leak. In the scenario as run, the crew realized they had a secondary leak, but entered the procedure for a reactor coolant leak since this was the only procedure available with entry conditions matching plant symptoms. After executing this entire procedure with no useful results, the SRO decided to perform the expected action of shutting down the plant.

The above difficulties, resulting from lack of a procedure or lack of sufficiently explicit guidance, were discussed with the operations manager and training department following the scenarios, who agreed to further evaluate these difficulties.

Overall, walkthrough performance was good. The candidates performed the JPM tasks in a careful manner, checking and verifying the appropriate procedure section and components to be manipulated.

Written Examination

The following items were missed by three or more of the candidates tested and represented areas of generally weak understanding:

Question#	Topic
2	Conditions for placing RHR in service.
3	Rod control system component operations.
14	When to suspend core alterations.
40	Response to a dropped rod.
51	Automatic actions for radiation monitor alarm.
66	EDG response to SI and loss of offsite power.
85	Technical Specification immediate actions.
94	Radiation Work Permits.
95	Containment purge requirements.

Review of UFSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the updated final safety analysis report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR descriptions. While performing the activities discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to selected examination topics.

The NRC reviewed facility 18-month equipment response time surveillance procedures to verify that ECCS equipment loading times were consistent with the times listed in section 6 of the UFSAR. The inspectors verified that the loading sequence and times were consistent with the UFSAR statements.

c. Conclusions

The candidates were well prepared and generally performed well. All five candidates passed the examination and were issued licenses.

V. Management Meetings

XI Exit Meeting on May 10, 1994

The NRC expressed its appreciation for the facility examination development, validation efforts and facility accommodation of the needs of the examination process. Generic strengths and weaknesses observed in the operating examinations were discussed.

The following key facility personnel attended the exit meeting.

J. M. Grillo	Operations Manager
Tom Grew	Training Manager
D. Roy	Training Supervisor
L. Carlsen	Training Supervisor
T. Cassidy	Sr. Instructor

Attachments:
1. Written Examination and Answer Keys
2. Simulation Facility Report

ATTACHMENT 1

WRITTEN EXAMINATIONS AND ANSWER KEYS

Nuclear Regulatory Commission
Operator Licensing
Examination

This document is removed from
Official Use Only category on
date of examination

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

CANDIDATE'S NAME:	______
FACILITY:	SEABROOK
REACTOR TYPE:	PWR-WEC4
DATE ADMINISTERED:	9/30/96

INSTRUCTIONS TO CANDIDATE:

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

TEST VALUE	CANDIDATE'S SCORE	%	
100 POINTS	______ FINAL GRADE	____ %	TOTALS

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 applicant at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

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

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

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

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

8. If the intent of a question is unclear, ask questions of the examiner **ONLY**.

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

10. Ensure all information you wish to have evaluated as part of your answer is on your answer sheet. Scrap paper will be disposed of immediately following the examination.

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

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

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

QUESTION: 001

OS1202.05, "Reactor Coolant System High Activity", directs the operator to maximize letdown flow to reduce activity levels during a high RCS activity condition.

Under which of the following conditions would increasing letdown flow be most effective?

a. High RCS activity is caused by fission products from a leaking fuel assembly.

b. High RCS activity is due to non-soluble radioactive gasses.

c. High RCS activity is caused by high tritium levels.

d. High RCS activity is caused by high levels of non-ionic contaminants.

QUESTION: 002

PLANT CONDITIONS:

- MODE 4
- The crew is preparing to place RHR in service
- RCS Tavg is 349°F
- RCS pressure indication on PT-405 is 375 psig
- RCS pressure indication on PT-403 is 360 psig.

Which of the following statements describes the ability to place RHR in service for cooldown?

a. Both trains may be lined up for RHR cooldown.

b. RHR train 'A' may be lined up for cooldown.

c. RHR train 'B' may be lined up for cooldown.

d. Neither train may be lined up for cooldown.

QUESTION: 003

PLANT CONDITIONS:

- Operating at 90% power
- The operator receives a DRPI rod position deviation alarm for ROD H8.
- The crew determines that rod H8 has become misaligned 14 steps below the rest of CB D.

The crew is attempting to realign rod H8 per OS1210.06, "Misaligned Rod"

Which of the following actions will freeze the Bank Overlap Unit at its present count?

a. Placing the Rod Control Selector Switch to the CB D position.

b. Placing the Pulse to Analog converter AUTO-MAN switch to MANUAL.

c. Placing the unaffected CB D rods in the 'DISCONNECT' position.

d. Depressing the Rod Control alarm RESET Pushbutton.

QUESTION: 004

Reactor power is increasing. Which of the following describes the operation of Control Interlock C-2?

a. Energizes when 1 of 2 Intermediate Range detectors indicates a current equivalent to 20% power. Inhibits Rod withdrawal in Manual or Auto

b. Energizes when 1 of 4 Power Range detectors indicates 103%. Prevents Rod withdrawal in Man or Auto

c. Energizes when 1 of 2 Intermediate Range detectors indicates 20%. Prevents Rod withdrawl in AUTO only.

d. Energizes when 1 of 4 Power Range detectors indicates 103%. Prevents Rod withdrawl in AUTO only.

QUESTION: 005

Reactor power is 6% during a shutdown when intermediate range channel N-36 fails HIGH.

Which 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 NIs will re-energize when N-35 decreases to the proper setpoint.

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

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

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

QUESTION: 006

The following plant conditions exist:

- A rupture in the piping downstream of the Charging Cold Leg injection Valves (SI-V138/139) has occurred
- The check valves on the piping connecting to the RCS have failed causing a LOCA into the Containment penetration area of the PAB
- The Reactor has tripped and Safety Injection has actuated on Low PZR Pressure
- The Crew has entered E-0 and has completed the Immediate Actions
- RCS pressure is 1780 psig and decreasing
- Area and airborne high radiation alarms have actuated in the PAB
- All ECCS systems are operating per design
- Containment sump level is 0 feet on LI-2384 and LI-2385

Assuming plant conditions do not significantly change and the leak is unisolable, what is the expected flow path through the EOP Network for this accident?

a. E-0, "Reactor Trip or Safety Injection" to E-1, "Loss of Reactor or Secondary Coolant" to ES-1.2, "Post LOCA Cooldown & Depressurization".

b. E-0, "Reactor Trip or Safety Injection" to E-1, "Loss of Reactor or Secondary Coolant" to ECA-1.2, "LOCA Outside Containment" to ECA-1.1, "Loss of Emergency Coolant Recirculation".

c. E-0, "Reactor Trip or Safety Injection" to ECA-1.2, "LOCA Outside Containment" to E-1, "Loss of Reactor or Secondary Coolant".

d. E-0, "Reactor Trip or Safety Injection" to ECA-1.2, "LOCA Outside Containment" to ECA-1.1, "Loss of Emergency Coolant Recirculation".

QUESTION: 007

A plant event has resulted in implementation of ECA-2.1 'Uncontrolled Depressurization of All Steam Generators".

While attempting to control the RCS cooldown during this procedure, the operator throttles EFW flow which results in a RED path on the Heat Sink Critical Safety Function.

What action should be taken?

a. Raise at least one steam generator narrow range level to greater than 5%, then continue with ECA-2.1.

b. Increase EFW flow to 500 gpm to clear the "RED" condition and continue on in ECA-2.1.

c. Increase EFW flow to 500 gpm to clear the "RED" condition and transition to FR-H.1, "Response to Loss of Secondary Heat Sink".

d. Continue with ECA-2.1.

QUESTION: 008

Due to a Small Break LOCA, a Plant Trip and Safety Injection has occurred.

The following conditions exist:

- All automatic equipment responds as expected
- Containment pressure is 3.2 psig and slowly increasing
- Containment High Range Radiation Monitors RM-RI-6576-A & B read ~ 3 R/hr
- RCS pressure is 1750 psig and slowly decreasing
- Subcooling margin is 32 degrees F and slowly decreasing
- Pressurizer level is 22% and slowly decreasing

Assuming conditions do not significantly change, in which of the following procedures would you expect to be directed to stop one charging pump?

a. In ES-0.2, "Natural Circulation Cooldown".

b. In E-1, "Loss of Reactor or Secondary Coolant".

c. In ES-1.2. "Post-LOCA Cooldown and Depressurization".

d. In ES-1.1, "SI Termination".

QUESTION: 009

PLANT CONDITIONS:

- The crew is responding to a small break LOCA
- One charging pump has been stopped
- All other high head ECCS pumps are running
- No RCPs are running
- Pressurizer level is 30%

Sufficient subcooling exists and the operator stops the "A" Safety Injection pump. Immediately after stopping the pump, RCS pressure and PZR level begin to decrease.

What action should the crew take in response to the decrease in RCS pressure and PZR level?

a. Immediately restart the "A" Safety Injection pump to restore RCS pressure to its previous value.

b. Immediately reinitiate Safety Injection.

c. Monitor RCS subcooling, and pressurizer level to assure these parameters stabilize above their ECCS reinitiation values before continuing with ECCS flow reduction.

d. Monitor RCS pressure and subcooling. If they stabilize above their SI reinitiation values, place normal Charging and Letdown in service.

QUESTION: 010

A small break LOCA has occurred, all RCPs are running, and the operating crew is in ES-1.2, "Post-LOCA Cooldown and Depressurization." An RCS cooldown has been initiated by dumping steam to the condenser.

Which of the following statements describes the optimum reactor coolant pump configuration, and the basis for this configuration?

a. All RCPs should be stopped to minimize RCS inventory loss following break uncovery, and minimize heat input to the RCS.

b. Only one RCP should be run to allow for a normal RCS cooldown and provide PZR spray, yet minimize RCS heat input.

c. Only one RCP should be run to produce effective heat transfer and RCS pressure control, yet minimize RCS inventory loss.

d. Two RCPs should be run to ensure symmetric heat transfer to the intact SGs, to enhance RCS pressure control, and to prevent steam voiding in the reactor vessel head on the subsequent RCS depressurization.

QUESTION: 011

Which of the following describes the operation of the Emergency bus first level undervoltage protection scheme?

a. 2 normally energized undervoltage relays. When one of 2 relays sense bus voltage less than 70% of nominal for 1.2 seconds (RAT available), it initiates a sequence of load stripping and subsequent bus reenergization by the DG.

b. 2 normally energized undervoltage relays. When bus voltage drops below 25% of nominal, they deenergize, initiating auto closure of the RAT supply breaker.

c. 2 normally energized undervoltage relays. When both relays sense bus voltage less than 70% of nominal for 1.2 seconds (RAT available), they initiate a sequence of load stripping and subsequent bus reenergization by the DG.

d. 2 normally energized undervoltage relays. When both relays sense bus voltage less than 95% of nominal coincident with an SI existing for greater than 10 seconds, they initiate a sequence of load stripping and subsequent bus reenergization by the DG.

QUESTION: 012

PLANT CONDITIONS:

- MODE 5,
- RHR train 'A' is in service
- RHR letdown is in service
- T_{avg} is 190°F
- Pressurizer is solid.
- RCS pressure is being controlled at 300 psig by the use of the Letdown backpressure control valve, (CS-PCV-131) in AUTO

A bus fault causes a loss of 120 VAC Vital Instrument Panel PP-1E

Before any operator action is taken, what is the effect on plant operation?

a. RCS temperature will decrease, and RHR letdown flow will increase.

b. RCS temperature will increase, and RHR letdown flow will increase.

c. RCS temperature will decrease, and RHR letdown flow will decrease.

d. RCS temperature will increase, and RHR letdown flow will decrease.

QUESTION: 013

The plant is at 80% power, steady state.

Which of the following describes the effect that a loss of 125 VDC bus 11A has on DG-1A?

a. Loss of DG-1A breaker remote open/close capability from the Control Room only.

b. Loss of all engine protective tripping capability.

c. Loss of all normal and emergency engine start circuits.

d. Loss of all engine operating parameter indication.

QUESTION: 014

The plant is in MODE 6, Refueling is in progress.

Under which of the following circumstances would CORE ALTERATIONS be suspended?

a. Both doors of one personnel airlock are opened by a dedicated door operator during personnel entry into Containment.

b. The equipment hatch is in place but not bolted.

c. A manual containment isolation valve is open in preparation for Containment leak testing. The responsible Test Engineer is stationed at the valve.

d. All CAP valves are open for refueling purge operations.

QUESTION: 015

The plant is at 100 % power, all control systems are in their normal automatic alignment.

The following sequence of events occur:

- Control Bank D rods begin to step out.
- After checking board indications, the operator places the Rod Control Selector Switch to MANUAL.
- All Control Bank D rods stop moving with the exception of one Control Rod, which continues to step.

Which of the following actions should be taken?

a. Place the Control Bank Selector Switch in the CBD position and verify no rod movement.

b. Manually insert control rods to restore program Tavg.

c. Check PT-505 Turbine Impulse Pressure indication - NORMAL.

d. Trip the reactor and go to E-0, Reactor Trip or Safety Injection, Step 1.

QUESTION: 016

While operating at 92% power, a manual boration is initiated due to a malfunction in the makeup control system. All control system are in AUTO.

While performing the lineup the operator is momentarily distracted and when he completes the lineup he inadvertently opens the emergency boration valve (CS-V426) instead of the boration flow control valve (CS-FCV-110A).

Which of the following symptoms would indicate that the emergency boration valve (CS-V426) was inadvertently opened instead of flow control valve (CS-FCV-110A)?

a. T_{avg} begins to fall, the boric acid batch integrator advances more rapidly than normal, flow indicator (CS-FI-183A) reads 70 GPM, and control rods begin to step out.

b. T_{avg} remains steady, the boric acid batch integrator does not advance, flow indicator (CS-FI-183A) reads 70 GPM, and control rods do not move.

c. T_{avg} begins to fall, the boric acid batch integrator advances more rapidly than normal, flow indicator (CS-FI-183A) reads 70 GPM, and control rods begin to step in.

d. T_{avg} begins to fall, the boric acid batch integrator does not advance, flow indicator (CS-FI-183A) reads 70 GPM, and control rods begin to step out.

QUESTION: 017

The plant is at 100% power when the following VAS alarm is received:

- D5734 Vital UPS 1E INV PWR FUSE BLOWN

Locally at EDE-CP-1E (static transfer switch), the operator observes that the reverse transfer light is lit.

Which of the following events is the likely cause of this alarm/indication?

a. PP-1E is being supplied from it's DC supply.

b. PP-1E has transferred to it's alternate supply.

c. PP-1E maintenance supply breaker has tripped open.

d. PP-1E is de-energized.

QUESTION: 018

A loss of 120 VAC Vital Instrumentation Panel PP-1A has occurred. The crew is using OS1247.01, Loss of a 120 VAC Vital Instrument Panel, to stabilize the plant.

At step 2, the secondary operator notices that the Steam Dumps are open and closes the dumps using the steam dump interlock control switch as directed by the procedure.

Which of the following plant conditions would have to exist for the Steam Dumps to open when PP-1A was lost?

a. The C-7A signal from a prior load rejection was not reset.

b. The Steam Dumps were in the "Steam Pressure" mode of operation.

c. FW-PT-506 failed high.

d. C-16 was actuated.

QUESTION: 019

The plant is at 50% power when a total loss of Main Feedwater occurs. The reactor does not trip, and the crew enters FR-S.1.

What function, if any, will the ATWS Mitigation System provide under these conditions?

a. The ATWS Mitigation System is not armed under these conditions.

b. The ATWS Mitigation System will send a start signal to the EFW pumps when 1/4 SG NR levels are less than 5%.

c. The ATWS Mitigation System will send a start signal to the EFW pumps when 2/4 detectors on 1/4 SGs are less than 14%.

d. The ATWS Mitigation System will send a start signal to the EFW pumps when 3/4 SG NR levels are less than 5%.

QUESTION: 020

E-3, "Steam Generator Tube Rupture", directs the operators to cooldown the RCS using the intact SG's and then to depressurize the RCS below ruptured SG pressure.

Why doesn't the ruptured SG depressurize as a result of the RCS cooldown?

a. The tube rupture provides an energy input to the RCS, from both the high temperature water and mechanical compression of the steam space.

b. The ruptured SG is isolated and ruptured SG water level is maintained above the U-tubes so that a layer of hot water insulates the steam space from the U-tubes as the RCS cools.

c. The ruptured SG is isolated preventing energy removal.

d. The RCP in the ruptured loop is stopped reducing the heat transfer rate in that steam generator.

QUESTION: 021

Step #12 of ECA-3.1, "SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired", provides a check to determine if a subcooled recovery is appropriate. Why is a transition to ECA-3.2, "SGTR With Loss of Reactor Coolant - Saturated Recovery Desired" directed if RWST level is less than 290,000 gallons and Containment sump level is not within the "expected region" of Figure ECA-3.1-1?

a. Maintaining saturated RCS conditions facilitates PZR pressure control and allows RCP operation without concern for pump damage.

b. Reducing pressure to saturate the RCS and thereby reduce RCS leakage is appropriate if significant leakage is occurring outside Containment as RCS makeup water supply may be inadequate.

c. It is advantageous to maintain saturated RCS conditions in order to provide an indication of coolant inventory trends and ensure significant margin to core uncovery.

d. With subcooled conditions in the RCS no indication of RCS inventory trends exists until the core uncovers.

QUESTION: 022

While performing E-3, "Steam Generator Tube Rupture", prior to initiating the RCS cooldown, the USS directs you to determine if the ruptured steam generator pressure is greater than 225 psig.

If pressure is less than 225 psig, E-3 directs you to transition to ECA - 3.1, "SGTR with loss of Reactor Coolant - Subcooled Recovery Desired".

Which of the following describes the reason for transitioning to ECA-3.1?

a. ECA-3.1 is more appropriate because if ruptured SG pressure is this low it may be an indication of a ruptured-faulted SG.

b. ECA-3.1 is more appropriate because it ensures that the ruptured SG pressure is greater than the intact SG pressures for any subsequent cooldown.

c. ECA-3.1 is more appropriate because it contains steps that ensure that the subsequent cooldown does not cause a low steam line pressure Safety Injection actuation.

d. ECA-3.1 is more appropriate because it ensures that the subsequent cooldown does not decrease RCS pressure below accumulator injection pressure.

QUESTION: 023

The plant is in MODE 1, 100% power.

A loss of Off site Power occurs, and "A" DG fails to start.

Which of the following describes the expected electrical power flowpath to vital instrument power panels PP-1A, PP-1B, and PP-1E?

a. Battery B-1A → DC bus 11A → UPS-I-1A → **PP-1A**
   Bus E61 → MCC E612 → UPS-I-1B → **PP-1B**
   Battery B-1A → DC bus 11A → UPS-I-1E → **PP-1E**

b. Battery Charger BC-1A → DC bus 11A → UPS-I-1A → **PP-1A**
   Bus E61 → MCC E612 → UPS-I-1B → **PP-1B**
   Bus E61 → MCC E612 → UPS-I-1E → **PP-1E**

c. Battery B-1A → DC bus 11A → UPS-I-1A → **PP-1A**
   Battery Charger BC-1B → DC bus 11B → UPS-I-1B → **PP-1B**
   Battery B-1A → DC bus 11A → UPS-I-1E → **PP-1E**

d. Battery Charger BC-1A → DC bus 11A → UPS-I-1A → **PP-1A**
   Bus E61 → MCC E612 → UPS-I-1B → **PP-1B**
   Bus E53 → MCC E531 → 480/120v transformer via Static Transfer switch → **PP-1E**

QUESTION: 024

The plant is at 100% power at EOL with control bank D at 225 steps. A bank "D" rod drops into the core. No operator actions are taken.

Which of the following represents the expected plant response to the dropped rod?

a. Initially there will be a prompt drop in reactor power and then reactor power will slowly increase to a value equal to the initial power.

b. Reactor power will decrease to a new value that represents the negative reactivity worth of the dropped rod.

c. Initially there will be a prompt drop in reactor power and then reactor power will slowly increase to a value less than the initial power that represents the negative reactivity worth of the dropped rod.

d. Reactor power will increase slightly and then decrease to a value equal to the initial power.

QUESTION: 025

PLANT CONDITIONS:

- Condenser Vacuum is 22.5 inches Hg and slowly decreasing
- Load Reduction is in progress
- Turbine load is 350 MWE

Which of the following actions should be taken by the operating crew?

a. Immediately trip the turbine and verify all stop valves closed and the generator breaker opens.

b. Continue the load decrease to increase condenser vacuum to > 25 inches Hg.

c. Immediately trip the reactor and go to E-0.

d. Continue the load decrease and if vacuum remains greater than 22.4 inches Hg remove the turbine generator from service IAW OS1000.06, Power Decrease.

QUESTION: 026

Reactor trip and Safety Injection signals have actuated safeguards equipment. The Reactor Trip breakers failed to open and the crew has transitioned from E-0 to FR-S.1.

At the step requiring the crew to initiate Emergency Boration, what is the appropriate flowpath?

a. One boric acid pump running, charging flow maintained greater than 50 GPM, and letdown flow adjusted to maintain VCT level.

b. CS-V426 open, at least one boric acid pump running, charging flow control in Manual and at maximum, suction aligned to RWST with VCT isolated.

c. Charging flow maintained greater than or equal to 110 GPM, suction aligned to RWST, VCT suction isolated and letdown flow adjusted to maintain VCT level.

d. CS-V426 open, at least one boric acid pump running, charging flow control in auto and set at 120 GPM, suction aligned to RWST and VCT suction isolated.

QUESTION: 027

PLANT CONDITIONS:

- The unit has just returned to 100% power following a refueling outage.
- All Shutdown Banks are fully withdrawn.
- Control Banks are withdrawn with Control Bank D at 180 steps.
- Rod Control is operating in AUTO.
- A high stator cooling water temperature causes the turbine to begin running back.

During the runback D7762 "CNTL BK D INSERTION LIMIT LO-LO" is received.

What is the significance of this alarm?

a. The Relaxed Axial Offset Limits have been exceeded.

b. The MODE 1 Shutdown Margin Limit may have been exceeded.

c. The runback rate has exceeded the capabilities of the Control Rod Drive System.

d. The level of turbine runback has been excessive.

QUESTION: 028

PLANT CONDITIONS:

- The plant is operating at 100% steady state
- All Shutdown and Control Bank rods are at 229 steps
- The RO is performing rod exercises in accordance with the applicable surveillance procedure.
- As directed by the procedure, Control Bank "D" is exercised using "Bank Select" for CB D.

The procedure directs the operator to return the rods to their initial position but the operator accidentally leaves CB D rods at 224 steps following completion of surveillance testing.

What is the consequence of leaving CB D rods at 224 steps as opposed to returning them to their initial position of 229 steps?

a. The Rod Insertion Limit (RIL) computer will calculate a RIL that is lower than it should be for the given power level.

b. The Digital Rod Position Indication (DRPI) for the bank will read lower than the Bank Demand Position.

c. The Control Bank Overlap Unit will sense that CB D rods are still at 229 steps and overlap control banks based on this input.

d. A Logic Cabinet Non-urgent Failure will be generated the next time CB D rods are moved in AUTO or MANUAL.

QUESTION: 029

PLANT CONDITIONS

- 22% power.
- Pressurizer Pressure is stable at 2235 psig

Which of the following plant conditions would require the affected Reactor Coolant Pump to be tripped in accordance with OS1201.01, "RCP Malfunction"?

a. Seal Water Inlet Temperature is 200°F.

b. Upper Radial Bearing Temperature is 165°F.

c. RCP Frame Vibration is 2.5 mils and increasing at 0.2 mils per hour.

d. RCP Seal Leakoff Total Flow is greater than 9 gpm.

QUESTION: 030

Which of the following conditions will permit opening of the Regenerative Heat Exchanger Outlet Isolation Valve (CS-V-145)?

a. Pressurizer Level of 20%
   and
   Letdown Flow Isolation Valves (RC-LCV-460 and 459) - BOTH OPEN

b. Pressurizer Level of 20%
   and
   Letdown Flow Isolation Valves (RC-LCV-460 and 459) - EITHER ONE OPEN

c. Pressurizer Level of 20%
   and
   Letdown Line Isolation Valve (RC-V81) - OPEN

d. Pressurizer Level of 20%
   and
   Letdown Flow Isolation Valves (RC-LCV-460 and 459) - BOTH CLOSED

QUESTION: 031

Given the following:

- A Large Break LOCA has occurred
- All Safeguard Systems respond as designed
- The operating crew enters E-0, Reactor Trip or Safety Injection
- Based on observed plant conditions the crew transitions to E-1, Loss of Reactor or Secondary Coolant.
- In E-1 the crew resets the SI signal and shuts down the EDGs which have been running unloaded.
- When RWST Low-Low Level is reached at 125,000 gal. the Crew transitions to ES-1.3, Transfer To Cold Leg Recirculation.

How will the process of transferring to cold leg recirculation be affected with the SI signal having been reset?

a. Automatic opening of the Containment Sump Recirculation Valves (V-8 and V-14) will not occur. These valves will need to be manually opened in accordance with ES-1.3.

b. The 'S' signal must be reset to allow manual closure of the RWST suction valves (V-2 and V-5).

c. Automatic opening of the Containment Sump Recirculation Valves (V-8 and V-14) <u>and</u> automatic closure of the RWST Suction Valves (V-2 and V-5) will not occur. These valves will need to be manually operated in accordance with ES-1.3.

d. All valves reposition as designed; however, both Containment Building Spray pumps will trip when the RWST Low-Low Level setpoint is reached. The pumps will need to be manually started in accordance with ES-1.3.

QUESTION: 032

The following plant conditions exist:

- The plant was operating at 100% power when a large LOCA occured.
- A loss of offsite power has occurred, the EPS sequence is complete, and both Emergency Buses are being powered from their respective diesel generators.
- An 'A' train Containment Spray Actuation Signal (CSAS) was generated due to high containment pressure and 'A' train equipment is operating as required. A 'B' train CSAS failed to actuate and required Phase B status panel lights are not lit for Train 'B'.
- The operating crew is performing the RNO actions of step #14 of E-0 and have manually actuated both 'B' train CBS/P/CVI manual actuation switches but the 'B' train CBS pump has not started and automatic valves failed to reposition as required.
- The operator has attempted to manually start the 'B' train CBS pump but it will not start (yellow control switch light illuminated). The operator manually repositions other components in accordance with the status panel.

What action must the crew take to start CBS-P-9B?

a. The CBS pump cannot be started until off-site power is restored. The UAT or RAT breaker must be closed.

b. Since a 'B' train CSAS signal was not generated the CBS pump was not sequenced by the EPS. RMO must be reset before the pump can be started.

c. The crew must manually actuate stepping relay SR3 or relay HR8 at the EPS cabinet in the 'B' Train Essential Switchgear Room.

d. The RMO bypass switch must be held in the bypass position while the CBS pump breaker is locally closed onto Bus E6.

QUESTION: 033

Why is a Safety Injection signal generated by SSPS in response to a steamline rupture rather than just a reactor trip signal?

a. To ensure that EFW is available for subsequent plant cooldown.

b. To prevent the affected steam generator from dryout.

c. To ensure feedwater to the affected steam generator is isolated.

d. To prevent a return to criticality due to positive reactivity addition from the cooldown.

QUESTION: 034

PLANT CONDITIONS:

- The plant is in MODE 5
- The pressurizer is solid.
- Normal letdown is in service

Which of the following explains how RCS pressure is controlled in this situation.

a. Increase charging to increase pressure; increase letdown flow to lower pressure.

b. Energize pressurizer heaters to raise pressure; deenergize heaters or use aux.spray to lower pressure.

c. Throttle PCV-131 closed to raise pressure; throttle PCV-131 open to lower pressure.

d. Throttle CS-HCV-128 open to raise pressure; Throttle CS-HCV-128 closed to lower pressure.

QUESTION: 035

The plant is at 100%. The Backup Pressurizer Level Channel (RC-LT-460) rapidly fails low to 0%. No operator action is taken.

Which of the following describes the status for the given components/parameters 2 minutes after the level channel failure occurred?

	BACKUP HEATERS	CONTROL HEATERS	LCV-459	LCV-460	ACTUAL PZR LEVEL
a.	OFF	ON	OPEN	CLOSED	DECREASING
b.	OFF	OFF	CLOSED	CLOSED	DECREASING
c.	OFF	OFF	OPEN	CLOSED	INCREASING
d.	OFF	ON	CLOSED	CLOSED	INCREASING

QUESTION: 036

The time is 0300 and the plant is in MODE 3 at 549° F with EOL core conditions.

The following events occur:

- The PSO reports that Tavg has begun decreasing at a rate of approximately 1°F/min.
- The BOP reports that the low setpoint safety valve on the "C" SG has apparently failed partially open
- An NSO is dispatched and visually confirms one of the SG Safety Valves on the "C" SG is passing steam to the atmosphere
- The PSO confirms an RCS cooldown rate of 5°F in the last 5 minutes

Assuming no operator action is taken in response to the stuck open safety valve, which of the following conditions will exist by time 0400?

1. Main Steamline isolation.
2. The Tech. Spec. for RCS Cooldown Rate will be exceeded.
3. The actual amount by which the reactor is shutdown will decrease.
4. A PTS challenge to reactor vessel integrity will occur.

a. 1 and 3

b. 2 Only

c. 3 Only

d. 2 and 4

QUESTION: 037

What would be the initial effect on a SG's indicated pressure and level if its Main Steam Line Isolation Valve were to inadvertently close with the plant at full power?

a. Both pressure and level would initially increase.

b. Pressure would initially increase. Level would initially decrease.

c. Pressure would initially decrease. Level would initially increase.

d. Both pressure and level would initially decrease.

QUESTION: 038

The following plant conditions exist:

- A Reactor Startup following a refueling outage has just been completed.
- The Moderator Temperature Coefficient is + 2 pcm/°F
- Reactor power is currently stable below the Point Of Adding Heat (POAH) at $1X10^{-8}$ amps

An Atmospheric Steam Dump Valve fails open.

How will RCS Tavg and reactor power be affected by this failure?

a. Both T-avg and reactor power will increase.

b. T-avg will increase. Reactor power will decrease.

c. T-avg will decrease. Reactor power will increase.

d. Both T-avg and reactor power will decrease.

QUESTION: 039

A fire in the Train 'A' Electrical Penetration Area has been confirmed by the Fire Brigade, and the control room crew has entered the appropriate fire response procedure. Subsequently a reactor trip occurs.

The procedure flow path the operating crew will follow is:

a. E-0, "Reactor Trip or Safety Injection", OS1200.00, "Response to Fire or Fire Alarm Actuation", OS1200.00A, App. A to Fire Hazards Analysis for Affected Area/Zone, OS1200.01, "Safe Shutdown and Cooldown From the Main Control Room".

b. OS1200.00, "Response to Fire or Fire Alarm Actuation", OS1200.00A, App. A to Fire Hazards Analysis for Affected Area/Zone, OS1200.01, "Safe Shutdown and Cooldown From the Main Control Room".

c. E-0, "Reactor Trip and/or Safety Injection", ES-0.1, "Reactor Trip Response" and simultaneously OS1200.00, "Response to Fire or Fire Actuation".

d. OS1200.00, "Response to Fire or Fire Alarm Actuation", when directed by OS1200.00 enter E-0, "Reactor Trip or Safety Injection" and return to OS1200.00 after the immediate action steps, OS1200.00A, App. A to Fire Hazards Analysis for Affected Area/Zone.

QUESTION: 040

A reactor startup is in progress and the reactor has just achieved criticality.

A Rod Control System Urgent Failure Alarm is received and rod B6 trops into the core causing the reactor to go subcritical.

Which of the following describes the course of action to be taken?

a. Match Tavg-Tref by adjusting turbine load.

b. Trip the reactor and go to E-0, Reactor Trip or Safety Injection.

c. Refer to Technical Specifications and conduct QPTR and Shutdown Margin Calculations.

d. Conduct a plant shutdown using OS1000.03, Plant Shutdown From Minimum Load to Hot Standby.

QUESTION: 041

Given the following:

- ONE control rod is misaligned from its group by more than twelve (12) steps and determined to be INOPERABLE.
- The Technical Specification ACTION statement limits reactor power to 75% Rated Thermal Power.

Which of the following is the reason for this power limit?

a. Reduces the Rod Insertion Limit below the misaligned rod position.

b. Allows the plant to be operated without performing a re-evaluation of the safety analysis affected by a misaligned rod.

c. Relieves the operators of having to calculate Shutdown Margin every 12 hours.

d. Provides assurance of fuel rod integrity during continued operations.

QUESTION: 042

Which of the following describes the effect of one UNDERCOMPENSATED Intermediate Range Channel following a reactor trip?

a. Channel indicates HIGH preventing P-6 from automatically energizing the source range.

b. Channel indicates LOW prematurely energizing the source range.

c. Channel indicates HIGH, the source range will be energized by P-6 from the other IR channel.

d. Channel indicates LOW, the source range will NOT be energized until P-6 is supplied from the other IR channel.

QUESTION: 043

Which of the following describes how the Thermal Barrier Cooling Water System (CC) would be isolated from a tube rupture in an RCP thermal barrier?

a. Manual isolation valves on the supply and return lines to the RCPs.

b. Check valves in the supply lines and motor operated isolation valves on the return lines from the RCPs.

c. Motor operated isolation valves in the supply lines and check valves on the return lines from the RCPs.

d. Motor operated isolation valves on the supply lines and motor operated isolation valves on the return lines from the RCPs.

QUESTION: 044

Which of the following is a reason for isolating all feedwater to a faulted Steam Generator (SG)?

a. To reduce the probability of a SG tube rupture in the faulted SG.

b. To minimize the RCS cooldownand mass and energy release following a steamline break.

c. To prevent all feedwater flow from entering the faulted SG and filling the SG, causing the ASDV to lift.

d. To allow quicker identification of a SG tube rupture and thus limit the release to the environment below 10CFR100 limits during a design basis event.

QUESTION: 045

The following plant conditions exist:

- The plant is at 100% power
- The controlling SG water level channel to the 'C' SG has failed low
- The operator is unable to shift the affected feed water regulating valve to manual.

Which of the following methods is used to maintain steam generator water level?

a. Use the main feedwater pump master speed controller together with manual control of the unaffected SG feed water regulating bypass valves.

b. Take manual control of the unaffected SG feedwater regulating valves and use the feedwater regulating bypass valve for the affected SG.

c. Use the main feedwater pump master speed controller and locally control the affected SG feedwater regulating valve.

d. Immediately trip the applicable bistables so a shift to an unaffected channel can be made and control the affected SG level in auto.

QUESTION: 046

Given the following:

- The control room has been evacuated due to a fire.
- All remote safe shutdown system lineups have been completed.
- The local/remote switch on Bus E-5 for RHR pump RH-P-8A is in local
- A valid SI signal has just been received.

Which of the following describes the RHR pump response?

a. The pump will start and remains running until the "S" signal is reset, at which time the pump will stop.

b. The pump will start and remains running until its associated breaker is opened locally.

c. The pump will not automatically start, but the operator can start/stop the pump using the local control switch at the switchgear.

d. The pump will not automatically start, but the operator can start/stop the pump from the Train A remote safe shutdown panel.

QUESTION: 047

Given the following:

- A reactor startup is in progress with the reactor just critical.
- The operator has just stopped moving rods
- Power slowly increases to above the P-6 setpoint.

One source range (SR) channel fails LOW. The remaining power indications stabilize.

WHICH of the following actions is required?

a. Block the source range since it is not required above P-6.

b. Trip the reactor and enter E-0, Reactor Trip and/or Safety Injection.

c. Suspend all operations involving positive reactivity changes until both SR channels are restored to operability.

d. Conduct a reactor shutdown and restore both SR channels to operability prior to the next startup.

QUESTION: 048

The operating crew is establishing Feed and Bleed per FR-H.1, "Response to Loss of Secondary Heat Sink".

Why are the PORVs manually opened rather than allowing RCS pressure to rise and automatically open the valves at their lift setpoints?

a. RCP damage may occur due to inadequate seal injection at PORV setpoint.

b. Waiting for the PORVs to open at their lift setpoints may prevent adequate injection flow and lead to inadequate core cooling.

c. A solid water RCS at the PORV setpoint has a high potential to challenge the RCS pressure safety limit.

d. Depressurizing a SG for condensate feed with the RCS at the PORV setpoint will exceed SG U-tube delta-P limits.

QUESTION: 049

The plant is operating at 100% power.

The following events occur:

- Pressurizer level deviation alarm actuates
- Pressurizer backup heaters energize
- CS-FCV-121 output is slowly decreasing
- Pressurizer level is slowly decreasing on LT-460 and LT-461
- Pressurizer level reads 100% on LT-459

Which of the following will result if NO operator action is taken?

a. Pressurizer level will decrease and be controlled at a lower than normal level.

b. Pressurizer level will decrease, causing Pressurizer pressure to decrease until a reactor trip occurs on low pressurizer pressure.

c. Pressurizer level will decrease, then increase and be controlled at a higher than normal level.

d. Pressurizer level will decrease, then increase until a reactor trip occurs on high pressurizer level.

QUESTION: 050

Under degraded core cooling conditions, one RCP is tripped by FR-C.2, "Response to Degraded Core Cooling".

Which of the following describes the basis for tripping only one RCP.

a. To reserve one pump from possible damage for future cooling needs yet still maintain sufficient core cooling flow.

b. To limit heat input to the RCS already in a degraded condition yet still maintain sufficient core cooling flow.

c. To conserve SG water inventory yet maintain sufficient boron mixing to prevent a reactor restart accident.

d. To reserve inventory in one SG for future cooling needs yet maintain sufficient loop flow to prevent a stagnant loop (PTS concerns).

QUESTION: 051

The following control room air intake radiation monitors are in "HIGH" alarm:

East Air Intake Train 'A' RM-6506A Channel A1	East Air Intake Train 'B' RM-6506B Channel B1
West Air Intake Train 'A' RM-6507A Channel A1	West Air Intake Train 'B' RM-6507B Channel B2

Which of the following describes the required control room ventilation alignment based upon the above radiation monitor alarms?

a. CBA-FN-27A and CBA-FN-27B makeup air fans - BOTH NOT RUNNING.

b. CBA-FN-27A and CBA-FN-27B makeup air fans - BOTH RUNNING.

c. CBA-FN-16A and CBA-FN-16B emergency makeup filter fans - BOTH RUNNING.

d. CBA-FN-16A and CBA-FN-16B emergency makeup filter fans - BOTH NOT RUNNING.

QUESTION: 052

The plant is at 100% power, 3410.9 MWth.

The crew has determined that feedwater heaters 25B and 26B must be removed from service to isolate a leaking 26B heater.

Which of the following actions will have to be taken by the crew?

a. Reduce turbine load to prevent exceeding licensed thermal power limits.

b. The 25A and 26A feedwater heaters will also have to be removed to maintain balanced feedwater heating through the feedwater strings.

c. Increase reactor power to ir .ease RCS T_{avg}.

d. Increase turbine load to compensate for the load decrease due to the partial loss of feedwater preheating.

QUESTION: 053

In response to a steamline break inside containment the crew has transitioned from E-2 to E-1 and the following conditions exist:

- Pressurizer level - 25%
- Intact SG narrow range levels - 52% and slowly increasing
- Total feed flow to intact SGs - 100 gpm
- RCS pressure - 1600 psig and increasing
- RCS temperature - 475°F
- Containment pressure - 16 psig

Which of the following MUST be increased before the operating crew can terminate SI?

a. Feed flow.

b. RCS subcooling.

c. Pressurizer level.

d. RCS pressure.

QUESTION: 054

The liquid radwaste test tank discharge radiation monitor (R-6509) has been declared INOPERABLE.

Which of the following describes the Technical Specification ACTION that will permit continued release from the liquid waste system?

a. Liquid waste discharge will not be permitted until the discharge radiation monitor is returned to operable status.

b. A temporary monitor may be used provided its alarm setpoint is more conservative than the R-6509 setpoint to allow the operator sufficient time to manually secure the discharge in the event an alarm condition occurs.

c. Two independent samples of the tank to be discharged must be analyzed, and two technically qualified staff members must independently verify the release rate calculations and the discharge line valve lineup.

d. Samples must be taken every 15 minutes while the discharge is in progress, to verify the effluent is within Technical Specification limits.

QUESTION: 055

Which of the following indications will the operator utilize to verify natural circulation during a post LOCA cooldown and depressurization?

a. Core exit TCs, RCS hot and cold leg temperatures, RCS pressure and RCS subcooling.

b. RCS subcooling, RCS hot leg temp., RCS pressure, PZR level and SG pressures.

c. Core exit TCs, RCS hot and cold leg temps. RCS subcooling and SG pressures.

d. RCS subcooling, RCS pressure, SG pressure, PZR level and RCS cold leg temperature.

QUESTION: 056

An LOP occurs.

While the Emergency Power Sequencer (EPS) is in the process of completing the stepping sequence, an automatic safety injection occurs.

Which of the following describes the operation of the Emergency Power Sequencer upon initiation of the Safety Injection?

a. Sequencer resets to step 0, diesel remains running, diesel output breaker remains closed, Containment structure cooling fans are tripped, all previously running loads remain running and the sequencer sequences the SI loads.

b. Diesel output breaker opens to strip the bus, sequencer resets to step 0 in a standby mode, "S" signal causes the diesel output breaker to close and the EPS to restart all the loads in the SI/LOP stepping sequence.

c. Sequencer stops at the step in progress, diesel remains running, diesel output breaker remains closed, the RA relay actuates to trip the Containment structure cooling fans and then the EPS completes the stepping sequence in progress.

d. Diesel output breaker opens to strip the bus, diesel remains running but the "S" signal prevents the EPS from actuating, the sequencer resets to step 0 in a standby mode and will restart the entire sequence when the operator closes the diesel output breaker.

QUESTION: 057

Which of the following will occur on a loss of Vital DC Bus 11B?

a. Both EFW pumps start and the MFRV and bypass valves fail open.

b. The steam driven EFW pump starts, however EFW flow can be throttled only with the "B" train throttle valves.

c. The "B" train P-14 solenoids on the MFRV and MFRV bypass valves are deenergized causing these valves to fail closed.

d. The "B" train P-12 solenoids on the steam dump valves are de-energized causing the steam dumps to fail cpen.

QUESTION: 058

The control room operators are responding to a large-break LOCA inside containment in accordance with E-1. Critical Safety Function Status is:

Subcriticality - GREEN
Core Cooling - YELLOW
Heat Sink - GREEN
Integrity - YELLOW

The Shift Manager checks containment conditions to determine if the Containment barrier is intact. The following conditions exist:

- Containment pressure: 42 psig and slowly increasing
- Containment sump level: 3.2 feet and slowly increasing
- Containment radiation: 8.0 R/hr and stable
- All Containment Phase A & B penetrations are isolated

Given these conditions the Shift Manager will direct the USS to:

a. Transition to FR-Z.1, "Response to High Containment Pressure".

b. Transition to FR-Z.2, "Response to Containment Flooding".

c. Transition to FR-Z.3, "Response to Containment High Radiation".

d. Remain in E-1, Yellow path FRPs are entered upon discretion only.

QUESTION: 059

The following plant conditions exist:

- The plant has just completed a 428 day full power run.
- The plant is cooling down in MODE 3
- RCS temperature is 490°F
- RCS pressure is 1800 psig (SI has been previously blocked)
- Excess letdown is in service

A leak develops in loop "B" of the PCCW system causing the loop "B" head tank level to INCREASE.

WHICH of the following could be the potential source of leakage into Loop "B"?

a. Excess letdown heat exchanger.

b. "B" RHR heat exchanger.

c. Letdown heat exchanger.

d. Seal Return Water heat exchanger.

QUESTION: 060

Which of the following describes plant protection provided for a steam line break accident?

a. If main steamline pressure drops to 585 psig on two of four detectors on any steamline after P-11 has been blocked, all the MSIVs will close and a safety injection will occur.

b. If main steamline pressure drops to 585 psig on two of three detectors on any steamline, the MSIV will close on that steamline isolating the break.

c. If main steamline pressure drops to 585 psig on two of four detectors on any steamline, the MSIVs will close and a safety injection will occur.

d. If main steamline pressure drops to 585 psig on two of three detectors on any steamline, the MSIVs will close and a safety injection will occur.

QUESTION: 061

Step 10 of E-1, "Loss of Reactor or Secondary Coolant", directs the operator to check all steam generator pressures stable or increasing.

If any steam generator pressure is decreasing, the operator is directed to return to step 1 of E-1.

Which of the following describes why the operator should not proceed past step 10 with a depressurizing steam generator?

a. SI termination criteria could not initially be met and more restrictive termination criteria would be encountered in ES-1.2, "Post LOCA Cooldown and Depressurization".

b. E-1 provides no guidance for faulted steam generator isolation past step 10.

c. The RCS cooldown rate must be under control in order for subsequent E-1 steps to be effectively implemented.

d. SI termination criteria do not exist in ES-1.2, "Post LOCA Cooldown and Depressurization". A loop back to step #1 ensures that SI termination criteria are met in E-1.

QUESTION: 062

The core exit thermocouples (CETCs) are used as the temperature input for subcooling margin calculation. Subcooling margin is one of the criteria for safety injection termination in all of the Emergency Operating Procedures (EOPs).

Which of the following explains why the core exit thermocouples are the preferred instrumentation for subcooling determination.

a. The core exit thermocouples are used because any effect due to an isolated steam generator will be limited to the core exit thermocouples in the vicinity of the isolated loop reactor vessel inlet.

b. The core exit thermocouples are used because they are automatically compensated for adverse Containment conditions.

c. The core exit thermocouples are used because they provide the most direct indication of conditions at the hottest point in the reactor coolant system.

d. The core exit thermocouples are used because they are retractable and can measure localized hot channels at various core heights.

QUESTION: 063

An inadvertent SI occurred and the operating crew is currently in ES-1.1, "SI Termination" at step 9, "Check if SI Pumps Should be Stopped".

The following conditions exist:

- PZR level is 35%
- PZR pressure is 2300 psig and increasing
- Both SI pumps are running.
- ONE charging pump is running providing 60 GPM flow through the normal charging header
- Both RHR pumps are running

Subsequently, one PORV opens and fails to reclose. Attempts to close the PORV's associated block valve fail.

Which of the following conditions will first require the operator to manually start the non-operating charging pump?

a. PZR level drops to less than 17%.

b. RCS Subcooling drops to less than 40°F.

c. RCS pressure drops to less than 1650 psig.

d. Containment pressure increases to greater than 4 psig.

QUESTION: 064

The pressure channel input to the master pressurizer pressure controller is slowly failing high.

Which of the following explains the pressurizer heater response to this failure?

a. The control group of heaters will go to zero output and remain there throughout the failure of the channel. The backup heaters will not energize.

b. The control group of heaters will go to zero output initially and then return to full output, followed by all sets of backup heaters turning on.

c. The control group of heaters will go to zero output and then turn back on to full output, followed by the C and D sets of backup heaters turning on.

d. The control group of heaters will go to zero output initially and then turn back on to full output, subsequently all sets of backup heaters will turn on followed by all heaters (control and backup) deenergizing.

QUESTION: 065

A total loss of instrument air has occurred.

Which of the following is the failed position for the listed components:

	CC-V122 PCCW ORC CTMT Isol	CS-LV-112A VCT divert	CS-V150 L/D HX ORC isol.	CS-HCV-128 RHR flow control
a.	CLOSED	VCT	CLOSED	CLOSED
b.	CLOSED	DIVERT	OPEN	CLOSED
c.	OPEN	VCT	OPEN	OPEN
d.	OPEN	DIVERT	CLOSED	OPEN

QUESTION: 066

The local/remote switch for the 'A' emergency diesel is in the "local" position.

An LOP occurs and 30 seconds later a safety injection occurs.

Which of the following describes the status of the "A" emergency diesel?

a. The diesel is running, the output breaker automatically closed and the sequencer is following the LOP/SI load sequence.

b. The diesel is running and the output breaker must be closed by an operator before the sequencer will commence the LOP/SI load sequence.

c. The diesel is running, the output breaker must be closed by an operator and the LOP/SI loads must be manually started.

d. The diesel generator will NOT start until the local/remote switch is returned to the "remote" position.

QUESTION: 067

In accordance with Attachment 'A' to OS1215.07, "Loss of Spent Fuel Pool Cooling or Level", which of the following describes the preferred order of EMERGENCY makeup water sources to the spent fuel pool?

a. Chemical and Volume Control System Makeup, Demineralized Water, Gravity Feed from the RWST.

b. Chemical and Volume Control System Makeup, Gravity Feed from the RWST, Gravity Feed from the CST.

c. Gravity Feed from RWST, Gravity Feed from CST, Demineralized Water.

d. Gravity Feed from RWST, Gravity Feed from CST, Fire Protection System.

QUESTION: 068

The plant is at 100% power and a loop 1 T_{hot} instrument has failed LOW. The operating crew has completed all of the actions in the applicable Abnormal operating procedure for this instrument failure, tripped applicable bistables, and all controls are back in automatic.

Subsequently, pressurizer pressure instrument PT-456 fails LOW.

Which of the following describes the expected plant response?

a. No effect on the plant.

b. Reactor trip will occur due to OTΔT trip coincidence being met.

c. Reactor trip will occur due to OPΔT trip coincidence being met.

d. "S" signal on PZR low pressure.

QUESTION: 069

Refueling is in progress when Channel "A" of the fuel manipulator crane radiation monitor (R-6535A) spuriously alarms HIGH. Channel "B" reads normal.

What operator actions are required in accordance with OS1252.03 "Area High Radiation"?

a. Perform an OPERABILITY surveillance on the RDMS channel to verify that the alarm is false.

b. Verify a containment ventilation isolation has occurred.

c. Manually close the valves and stop the fans for a containment ventilation isolation.

d. Ensure the containment purge supply and exhaust valves are open.

QUESTION: 070

PLANT CONDITIONS:

- Plant is at 100% power
- Pressurizer pressure channel, PT-455 failed high
- The operating crew carried out the actions of OS1201.06, all applicable bistables are tripped
- Channel PT-457 is now the controlling channel
- All systems have been returned to automatic control

A loss of 120 VAC vital instrument panel PP-1C has just occurred.

Which of the following describes the impact on the plant of the loss of PP-1C?

a. The plant will remain at 100% power. PZR pressure control will be in manual and the automatic actuation of the PORVs has been lost.

b. A safety injection will have occurred due to low pressurizer pressure logic coincidence being met.

c. The PORVs will open due to a high pressure signal and this will eventually lead to a safety injection on low pressurizer pressure.

d. The master pressure controller will cause the pressurizer control heaters to go to minimum output and close the spray valves.

QUESTION: 071

A reactor trip and safety injection have occurred.

Which of the following describes the status of RCP seal leakoff?

a. Seal leakoff flow thru the seal return heat exchangers to the suction of the charging pumps is retained for continued RCP operations.

b. Seal leakoff flow is diverted to the #2 seal when CS-V 167 and CS-V-168 go closed on an "T" signal.

c. Seal leakoff isolates when CS-V-167 and CS-V-168 close on a "T" signal; seal leakoff flow is provided via a relief valve to the PRT.

d. Seal leakoff flow isolates when CS-V-167 and CS -V-168 go closed on an "T" signal and all seal flow is directed back to the RCS via a relief valve on the return line.

QUESTION: 072

The unit has tripped due to a loss of all circulating water pumps. All other plant equipment has operated as designed, and no operator actions have been taken.

When the plant is stable, what should the average RCS temperature (T_{avg}) be?

a. 550°F

b. 557°F

c. 561°F

d. 564°F

QUESTION: 073

The plant is at 40% power.

The alarms for 'D' RCP high #1 seal leakoff flow is received. Upon checking applicable parameters the operators determine that the 'D' RCP should be tripped immediately due to seal leakoff problems.

Which of the following actions should be taken by the operating crew?

a. Trip the reactor and go to E-0. After step 4 of E-0, stop 'D' RCP and close the #1 seal leakoff valve after the pump has stopped.

b. Feed 'D' SG to 60-70%, trip 'D' RCP, close the #1 seal leakoff valve after the pump has stopped, and commence a plant Shutdown.

c. Trip the 'D' RCP, take manual control of 'D' SG feedwater control, and commence a plant shutdown.

d. Trip 'D' RCP, close the #1 seal leakoff valve after the pump has stopped, trip the reactor, and perform the immediate actions of E-0.

QUESTION: 074

The following alarms have been received:

- Instrument Air Pressure Low UA-54
- D4976 Inst Air Hdr Press A Low
- D4981 Inst Air Hdr Press B Low

When should the reactor be manually tripped per the Abnormal Operating Procedure?

a. When the ORC Containment PCCW isolation valves fail closed on loss of air.

b. When the Main Feed Regulating and Bypass valves fail open on loss of air.

c. When Instrument Air header pressure decreases to 90 psig.

d. When HCV-182 (RCP seal supply flow control valve) fails closed on loss of air.

QUESTION: 075

The plant is in MODE 5 on Shutdown Cooling. The following conditions exist:

- RCS pressure: 350 psig
- RCS temp: 160 ° F
- "B" RHR Pump is unavailable due to a "B" train electrical outage

"A" RHR Pump trips on overcurrent due to a seized pump shaft.

The crew has entered OS1213.01, "Loss of RHR during Shutdown Cooling".

If both RHR pumps will be out of service for an extended period of time and the plant is slowly heating up, what is the ultimate goal of OS1213.01?

a. Manually control RCS inventory and pressure. Monitor Containment integrity, and use at least 2 SG's for heat removal.

b. Isolate Containment, increase RCS pressure to increase subcooling, and use Feed and Bleed to control RCS temperature.

c. Manually control RCS inventory and pressure. Isolate Containment and use Feed and Bleed to control RCS temperature.

d. Manually control RCS inventory, retain automatic pressure control via letdown, decrease pressure and use any available SG's for heat removal

QUESTION: 076

From the list below, select the radiation monitor that is both a release path monitor and has an automatic action associated with it.

a. 1GM810, Condenser Air Evacuation.

b. 1LM805, Turbine Building Sump.

c. 1GA410, Fuel Storage Building Exhaust.

d. 1NG222, Plant Vent Lo Range Gas.

QUESTION: 077

A Safety Injection has occurred and the crew is at step 11 of E-0 when a loss of Off Site power occurs. The Emergency Diesel Generators start and their output breakers close to restore power to the vital busses.

What is the status of Containment Structure Cooling (CAH)?

a. The CAH fans will initially trip due to low CC flow and then restart at SR3 of the Emergency Power Sequencer on the LOP signal.

b. The CAH fans will initially trip due to the "S" signal and then restart at SR3 of the Emergency Power Sequencer on the LOP signal.

c. The CAH fans will initially trip due to the "S" signal and will not restart at SR3 of the Emergency Power Sequencer on the LOP signal.

d. The CAH fans will have tripped and will automatically restart when Remote Manual Override (RMO) is reset.

QUESTION: 078

A steam break in the 'C' SG resulted in a reactor trip and Safety Injection. The EFW system functioned normally. The control room team eventually restored an adequate heat sink with the following alignment for the EFW flow control valves:

- CS-4214-A1 Throttled Full closed
- CS-4214-B1 Auto-Full Open
- CS-4224-A1 Auto-Full Open
- CS-4224-B1 Throttled Full closed
- CS-4234-A1 Auto-Full closed
- CS-4234-B1 Auto-Full closed
- CS-4244-A1 Throttled Full closed
- CS-4244-B1 Auto-Full Open

Subsequently, power to MCC-615 is lost.

What is the effect on the operators ability to control steam generator level?

a. No effect since the loss of MCC affects only one of the two flow control valves to each steam generator.

b. The crew will be unable to initiate flow to the 'A' and 'D' SG's.

c. The crew will be unable to initiate flow to the 'B' and 'D' SG's.

d. The crew will be unable to initiate flow to the 'B' SG.

QUESTION: 079

While operating at 100% power the following VAS alarm is received:

- D4602 RCP A No 1 seal leakoff flow low.

When the operator checks the status of the A RCP on the computer he observes the following point:

- A4250 RCP A No 2 seal leakoff flow reads 1.2 gpm

Which of the following is the most likely cause of these indications?

a. Failure of the "A" RCP thermal barrier.

b. Failure of the "A" RCP #1 seal.

c. Failure of the "A" RCP #2 seal.

d. Failure of the "A" RCP #3 seal.

QUESTION: 080

Which of the following describes the eventual result of VCT level channel LT-112 failing HIGH? No operator actions are taken.

a. A low VCT level and auto swapover of the charging pump suction to the RWST.

b. A high VCT level due to continuous makeup from the blender to the VCT.

c. A low VCT level and loss of NPSH to the charging pumps.

d. A high VCT level and letdown diverted to the PDT.

QUESTION: 081

Refueling operations are in progress with a core off load being conducted in accordance with applicable plant procedures. A fuel assembly has just been placed in the fuel transfer car.

A loss of refueling cavity water occurs and the crew commences to carry out the actions of OS1215.05, "Loss of Refueling Cavity Water". The crew is unsuccessful in moving the fuel transfer car to the Fuel Storage Building.

Which of the following is the appropriate action for the crew to take for the fuel assembly that is out of the core.

a. Store it in the transfer canal with the refueling machine mast fully extended.

b. Store it in the RCCA change fixture.

c. Store it in the upender in a vertical position.

d. Store it in the transfer car in a horizontal position.

QUESTION: 082

Due to an operator error during SSPS testing at 100% power, a Safety Injection occurs.

Which of the following procedure steps terminates SI flow to prevent a pressurizer overfill situation?

a. E-0, "Reactor Trip or Safety Injection", step 15, Verify ECCS flow.

b. E-0, "Reactor Trip or Safety Injection", step 25, Check if ECCS Flow Should be Reduced.

c. ES-1.1, "SI Termination", step 7, Establish 60 GPM Charging Flow.

d. ES-1.1, "SI Termination", step 11, Verify ECCS flow is not required.

QUESTION: 083

Radiation monitor 1LM216, STM Gen 'A' Blowdown goes into HIGH alarm.

Which of the following describes the expected plant response to this alarm?

a. SG blowdown flash tank discharge valve SB-CV-6519 goes closed and the blowdown flash tank level control valve (LCV-1909-1) operates to maintain flash tank level at approximately 25%.

b. SG blowdown flash tank discharge valve SB-CV-6519 goes closed and flash tank pressure will increase until the blowdown outer Containment isolation valves go closed on high flash tank pressure.

c. SG blowdown flash tank discharge valve SB-CV-6519 goes closed and flash tank level will increase until the blowdown inner Containment isolation valves go closed on high flash tank level.

d. SG blowdown flash tank discharge valve SB-CV-6519 will remain open until radiation monitor 1LM215 Blowdown Flash Tank Outlet also goes into high alarm and satisfies the coincidence for closing CV-6519.

QUESTION: 084

Which of the following sets of conditions would require a plant shutdown as directed by a Seabrook Station Abnormal operating procedure?

a. RCP 'A' frame vibration is 2.5 mils and stable, shaft vibration is 15 mils and stable. Reactor power is currently at 30%.

b. Verification that sustained winds in excess of 65 MPH are expected to hit the site within the next six hours.

c. Control Bank D rod H8 cannot be moved in automatic or manual MODE of control. The control room has a power cabinet urgent VAS alarm. The plant is at 100% power.

d. Chemistry has been monitoring steam generator tube leakage; SG 'A' leakage is 25 gallons per day, SG 'C' leakage is 300 gallons per day. Other SG samples show no detectable activity (NDA). Plant power is 75%.

QUESTION: 085

Which of the following Technical Specification LCO's <u>DOES NOT</u> have Immediate ACTIONs associated with it?

a. 3.7.7, Snubbers (MODEs 1, 2, 3, and 4).

b. 3.4.2.1, Pressurizer Safety Valves (MODEs 4 and 5).

c. 3.1.1.1, Shutdown Margin (MODEs 1, 2, 3, and 4).

d. 3.9.2, Refueling Operations, Source Range Monitors (MODE 6).

QUESTION: 086

<u>PLANT CONDITIONS</u>:

- Train 'A' Service Water (SW) was transferred from the ocean to the cooling tower for quarterly surveillance testing.
- When transferring Train 'A' SW back to the ocean, the breaker for cooling tower pump 110A discharge valve, SW-V-54, tripped and the valve was found to be mechanically bound at 60% open.
- Cooling tower pump 110A discharge pressure is 60 psig
- Train 'A' system flow is 10,000 gpm.

What associated Tech Spec ACTION should the crew enter? (See reference material attached to exam package)

a. T.S. 3.7.4, ACTION a

b. T.S. 3.7.4, ACTION b.

c. T.S. 3.7.4, ACTION d.

d. No T.S. 3.7.4 ACTIONs apply, the crew should enter T.S. 3.0.3.

QUESTION: 087

OS1000.01, "Heatup From Cold Shutdown to Hot Standby", states the following Limitation:

"When changing RCS boron concentration, pressurizer sprays should be utilized to maintain the differential boron concentration between the pressurizer and reactor coolant loops to less than 50 ppm."

What is the reason for this Limitation?

a. Ensures proper mixing occurs such that RCS loop boron samples accurately reflect actual boron concentration.

b. Prevents boron stratification in the pressurizer spray nozzles.

c. Prevents an RCS dilution event from occurring during a pressurizer outsurge.

d. Ensures that the boron concentration used in the accident analyses remain in their analyzed range of values.

QUESTION: 088

Which of the following statements describe <u>satisfied</u> Technical Specification Limiting Conditions for Operation (LCOs) in MODE 1?

a. Accumulator 'D' pressure indicators read 550 psig and 560 psig respectively.

b. RWST level is 485,000 gallons and RWST boron concentration is 2750 ppm.

c. NG-V-13, Safety Injection Accumulator Nitrogen Outside Containment Isolation valve fails its CLOSE stroke time requirement during surveillance testing.

d. CBS-V-43, Spray Additive Tank Outlet MOV, fails to OPEN on a HI-3 pressure test signal during surveillance testing.

QUESTION: 089

In accordance with the North Atlantic Procedure Administration Manual (NAPA), which of the following is **NOT** true regarding the review and approval of non-intent procedure changes?

a. Changes to Station Procedures, EOPs, and AOPs require a SORC review within 14 days of the Effective Date.

b. Technical Specifications require that non-intent changes are reviewed and approved or canceled within 14 days of the effective date.

c. If the change is to a department procedure, and the change is needed immediately, then the four 10CFR50.59 applicability questions must be answered by a SORC member.

d. If after further review, it is found that the non-intent change should have been an intent change, then the Operating Experience Manual (SSOE) should be referred to for Adverse Condition Report (ACR) applicability.

QUESTION: 090

Which of the following statements is a tagging requirement of MA 4.2, "Equipment Tagging and Isolation", when applied to 480V Motor Control Center (MCCs) breakers?

a. For load work only - Open the breaker and hang the tag on the breaker operator.

b. For load work concurrent with breaker work - Rack the breaker to the connected position. Position the lockout pawls to the locked position. Place the tag on the cubicle door (Maintenance is allowed to remove the breaker from the cubicle).

c. For breaker work only - Rack the breaker to the disconnect position, position the lockout pawls to the locked position and lock the lower lockout pawl. No Tags are required.

d. For all work on an MCC breaker - rack the breaker to disconnect, remove breaker from the cubicle, attach proper grounding device, and hang tag on cubicle door.

QUESTION: 091

Which of the following statements is TRUE regarding Technical Specification 2.1, "Safety Limits"?

a. The basis for the reactor coolant system pressure safety limit is to protect the integrity of the reactor coolant system piping and components, which prevents the release of radionuclides contained in the RCS to the containment atmosphere.

b. The reactor core safety limit is bounded by a figure using a combination of Thermal Power, pressurizer pressure, and highest loop auctioneered coolant flow.

c. If the safety limit for reactor coolant system pressure is exceeded in MODE 3, 4, or 5, then the pressure must be reduced to within it's limit within 30 minutes, and a Safety Limit Violation report must be prepared.

d. The curves used in the reactor core safety limit are based on a Heat Flux Hot Channel Factor $F_Q(Z)$ at rated thermal power of 2.32.

QUESTION: 092

Which of the following correctly describes a Limitation & Setpoint of OS1000.09, "Refueling Operation"?

a. Allow no more than one irradiated assembly in the cavity and canal at any one time. Fuel in the transfer car is not counted as if it were in the canal until it is latched by the spent fuel handling tool.

b. The minimum RHR flow in MODE 6 is 2750 gpm.

c. Maintain at least 20 feet of water above the reactor vessel flange during core alterations.

d. When there is fuel in the reactor, maintain greater than 120°F in the RCS to ensure that RCS temperature is within its analyzed range for shutdown margin calculation.

QUESTION: 093

The fuel handling SRO is performing OS1000.09, "Refueling Operation"

Which of the following Technical Specification based procedure Prerequisites is stated correctly?

a. If the containment purge and exhaust isolation system is needed for CAP or COP, then both trains of SSPS are in OPERATE.

b. Both RHR loops should be OPERABLE and in operation (until water level in the reactor cavity is greater than or equal to 23 feet above the flange, then both RHR loops should be OPERABLE and one RHR loop in operation).

c. The spent fuel pool and fuel transfer area (in the Fuel Storage Building) should be filled to a level at least 23 feet above the bottom of the fuel racks in the spent fuel pool.

d. At least one source range NI should be operable and in operation including the control room and containment audible indications.

QUESTION: 094

Which of the following statements is correct regarding Radiation Work Permits?

a. Health Physics personnel may authorize deviation from RWP requirements in the field on a case-by-case basis provided that the following conditions are met:

    1. the deviation is documented, and
    2. the deviation permitted is not above the individual's normal approved authority.

b. An example of work requiring a Specific Radiation Work Permit would be:

    Entry into an area with removable contamination greater than 1,000 dpm/100 cm^2 (beta, gamma) or 20 dpm/100 cm^2 (alpha)

c. An example of work requiring a Routine Radiation Work Permit would be:

    Opening a potentially contaminated system

d. Routine RWPs will normally remain in effect for the duration of the job.

QUESTION: 095

When performing a containment on-line purge to reduce containment ammonia levels for a planned entry, what is the sequence of events necessary to place the COP system in service after an approved Gaseous Effluent Release permit is received?

a. Set the COP radiation monitor setpoints to 1×10^5 cpm, unlock and energize the circuits for COP valves 1-4, and place the COP system in service to maintain 15.2 to 15.3 psia.

b. Enter the Tech Specs for the COP radiation monitors, set the COP radiation monitor setpoints to 1×10^5 cpm, unlock and energize the circuits for COP valves 1-4, and place the COP system in service to maintain 15.2 and 15.3 psia.

c. Enter the Tech Specs for the COP radiation monitors, set the COP radiation monitor setpoints to 1×10^5 cpm, enter the Tech Spec for the COP valves, unlock and energize the circuits for COP valves 1-4, and place the COP system in service to maintain 15.2 and 15.3 psia.

d. Set the COP radiation monitor setpoints to 1×10^5 cpm, enter the Tech Spec for the COP valves, unlock and energize the circuits for COP valves 1-4, and place the COP system in service to maintain 15.2 to 15.3 psia.

QUESTION: 096

Which of the following describes the flow path for performing an air purge of containment to reduce hydrogen concentration while in the emergency operating procedures?

a. From the service air system into containment via the normal H_2 analyzer sample lines. Out of containment via CGC-V14 and CGC-V28 to the containment enclosure emergency exhaust filters and then out the plant vent.

b. From the service air system via normally locked closed valves into containment. Out of containment via CGC-V14 to the inlet of the Train A containment enclosure emergency exhaust filter and then out to PAH-F-16.

c. From the service air system via normally locked closed valves into containment. Out of containment via CGC-V14 and CGC-V28 to the inlet of the containment enclosure emergency exhaust filters and then out the plant vent.

d. From the service air system via normally locked open valves through a containment isolation check valve into containment. Out of containment via CGC-V14 or CGC-V28 to the containment enclosure emergency exhaust filter to PAH-F-16.

QUESTION: 097

The plant is at 100% power when the following events occur:

Power is lost to MCC-531, deenergizing DRPI indication and causing an indicated Red Path on the Subcriticality CSF Status tree. Soon afterwards, the Reactor Protection system generates an automatic reactor trip signal resulting in the reactor trip breakers opening.

Which of the following actions should be performed by the operating crew.

a. Rod bottom lights are lit, the primary side operator should attempt a manual trip using the manual trip switches. Reactor trip is then verified so the crew should proceed on to verify turbine trip.

b. Rod bottom lights are not lit, the primary side operator should attempt a manual trip using the manual trip switches. The crew should go to FR-S.1, "Response to Nuclear Power Generation/ATWS", Step 1.

c. Rod bottom lights are not lit, but reactor trip is verified by reactor trip breaker open indication and neutron flux levels decreasing. The crew should proceed on to verify turbine trip.

d. Rod bottom lights are not lit, the crew should exit the procedure to FR-S.1, "Response to Nuclear Power Generation/ATWS", Step 1, due to the red path that exists on the subcriticality CSF status tree.

QUESTION: 098

The operating crew is responding to a LOCA in containment and have exited E-0 to FR-P.1 due to an ORANGE path on the Integrity CSF status tree.

While performing this procedure the crew notices that there is an ORANGE path indicated on the Containment CSF status tree and the Core Cooling CSF status tree.

These indications are followed 10 minutes later by a RED path on the Heat Sink CSF status tree.

Which of the following procedure flowpaths should be used by the operating crew?

a. Complete the actions of FR-P.1, and proceed to FR-C.2. Suspend the actions of FR-C.2 when the red path on the heat sink CSF tree is noticed and proceed to FR-H.1.

b. Complete the actions of FR-P.1, and proceed to FR-Z.1. Suspend the actions of FR-Z.1 when the red path on the heat sink CSF tree is noticed and proceed to FR-H.1.

c. Suspend the actions of FR-P.1 and proceed for FR-C.2. Complete the actions of FR-C.2 and then proceed to FR-H.1

d. Suspend the actions of FR-P.1 and proceed to FR-C.2. Suspend the action of FR-C.2 when the red path on the Heat Sink CSF tree is noticed and proceed to FR-H.1.

QUESTION: 099

For an ATWS event where a loss of normal feedwater has occurred, FR-S.1, "Response to Nuclear Power Generation/ATWS", has the operator verify Turbine trip at step #2.

What is the basis for this requirement?

a. The turbine is tripped to prevent a controlled cooldown of the RCS due to the steam flow that the turbine would require.

b. Turbine trip is required (within 30 seconds) to maintain SG inventory.

c. The turbine is tripped to increase primary system pressure and temperature so that feedback mechanisms will add negative reactivity to the core

d. The turbine is tripped to decrease the probability of delayed operator action due to misdiagnosis of the ATWS event.

QUESTION: 100

The plant has experienced a large LOCA in containment and the operating crew is performing actions in E-1, Loss of Reactor or Secondary Coolant. The CSF status trees have the following indications;
S - Yellow; C - Orange; Z - Orange; P - Orange, H - Red.

What Emergency Plan classification should the Shift Manager make?

a. Site Area Emergency, based on P - Orange <u>or</u> S - Yellow.

b. General Emergency, based on C - Orange <u>or</u> H - Red.

c. Site Area Emergency, based on Z - Orange <u>and</u> S - Yellow

d. General Emergency, based on C - Orange <u>and</u> H- Red.

3/4.7.4 SERVICE WATER SYSTEM/ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.4 The Service Water System shall be OPERABLE with:

a. An OPERABLE service water pumphouse and two service water loops with one OPERABLE service water pump in each loop,

b. An OPERABLE mechanical draft cooling tower and two cooling tower service water loops with one OPERABLE cooling tower service water pump in each loop, and

c. A portable cooling tower makeup system stored in its design operational readiness state.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one service water loop inoperable, return the loop to OPERABLE status within 72 hours, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one cooling tower service water loop or one cooling tower cell inoperable, return the affected loop or cell to OPERABLE status within 7 days, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With two cooling tower service water loops or the mechanical draft cooling tower inoperable, return at least one loop and the mechanical draft cooling tower to OPERABLE status within 72 hours, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two loops (except as described in c) or the service water pumphouse inoperable, return at least one of the affected loops and the service water pumphouse to OPERABLE status within 24 hours, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With the portable tower makeup pump system not stored in its design operational readiness state, restore the portable tower makeup pump system to its required condition within 72 hours, or continue operation and notify the NRC within the following 1 hour in accordance with the requirements of 10 CFR 50.72 of actions to ensure an adequate supply of makeup water for the service water cooling tower for a minimum of 30 days.

TECHNICAL CLARIFICATION

SECTION I - REQUEST FOR CLARIFICATION

Originator: Gene St.Pierre Date: 6/6/96

Technical Clarification Title: Inoperable Service Water Loops

Technical Clarification No.: TS-66

Type of Clarification:

Tech Spec (TS) ☒ UFSAR (Excluding 17.2) (FS) ☐ UFSAR 17.2 (QS) ☐ Licensing (LS) ☐

REQUEST FOR CLARIFICATION: (Attempt to state the request as a question.)

What is the correct Action Statement to enter when one SW loop and one SW cooling tower loop in the same train are inoperable?

CONCURRENCE: Gene St. Pierre 7/31/96
Group Manager Date

SECTION II - INITIATION

RECEIVED BY REGULATORY COMPLIANCE: [signature] 7/2/96
Regulatory Compliance Supervisor Date

TECHNICAL CLARIFICATION
(Continued)

SECTION III - EVALUATION

The Seabrook Station Service Water System (SW) consists of 4 loops, Train A and B SW loops and Train A and B cooling tower SW loops.

When one SW loop and one SW cooling tower loop in the same Train are inoperable, the correct Action to enter is Technical Specification 3.7.4, Action d., which allows 24 hours to return at least one of the affected loops to operable status.

Action d. is also applicable for any combination of SW loops and cooling tower SW loops (except 2 cooling tower SW loops) provided the functional equivalent of one full Train i.e. 2 of 4 loops is operable. (Example: Train A SW loop and Train B SW cooling tower loop)

Prepared By: Michael O'Keefe 8/6/96 (Date) Concurrence: [signature] 8/6/96 Cognizant Group Manager (Date)

SECTION IV - REVIEW AND APPROVAL

(Check Appropriate Boxes)

☐ [signature] for AMCallendrello 8/6/96 Licensing Manager (Date)	☐ ______ (Date)
☐ [signature] 8/7/96 Station Director (Date)	☐ ______ (Date)
☐ ______ (Date)	☐ ______ (Date)
	☐ ______ (Date)

SORC MEETING NO.: 96-077 DATE: 8/7/96

ATTACHMENT 2

SIMULATION FACILITY REPORT

FACILITY LICENSEE: North Atlantic Energy Service Corporation

FACILITY DOCKET NO: 50-443

Operating Tests Administered: 10/01 - 10/02 1996

This form is to be used only to report observations. These observations do not constitute audit or inspection findings, and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information, which may be used in future evaluations. No licensee action is required in response to these observations.

During the validation and performance of the simulator examination scenarios and Job Performance Measures, the following item was observed.

- An invalid position indication was illuminated for a D moisture separator heater steam valve. This was overridden by the simulator operator for the scenario.

- An improvement in modelling capability was noted in a scenario involving FR-C.1. In this scenario, core exit thermocouple temperatures of 1650 deg F were achieved, then brought down with accumulator injection and SI. The simulator response appeared realistic to the examiners and allowed the running of a scenario resulting in more severe core conditions than had previously been attainable.

ATTACHMENT 3
ANSWER KEY

NRC LICENSE EXAM - SEABROOK SRO (EXAM DATE: 9/30/96)

ANSWER KEY

1.	a	26.	b	51.	d	76.	b
2.	d	27.	b	52.	a	77.	c
3.	a	28.	c	53.	c	78.	d
4.	b	29.	d	54.	c	79.	c
5.	b	30.	a	55.	c	80.	c
6.	d	31.	b	56.	a	81.	d
7.	d	32.	b	57.	c	82.	c
8.	c	33.	d	58.	a	83.	c
9.	c	34.	c	59.	a	84.	d
10.	b	35.	c	60.	d	85.	a
11.	c	36.	c	61.	a	86.	c
12.	c	37.	b	62.	c	87.	c
13.	c	38.	d	63.	b	88.	b
14.	b	39.	b	64.	a	89.	c
15.	d	40.	d	65.	a	90.	c
16.	d	41.	d	66.	b	91.	a
17.	b	42.	a	67.	d	92.	b
18.	a	43.	b	68.	b	93.	a
19.	d	44.	b	69.	b	94.	a
20.	b	45.	c	70.	b	95.	b
21.	b	46.	c	71.	c	96.	c
22.	a	47.	a	72.	c	97.	c
23.	a	48.	b	73.	b	98.	d
24.	a	49.	d	74.	a	99.	b
25.	c	50.	a	75.	a	100.	d