U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos: License Nos:	50-387, 50-388 NPF-14, NPF-22
Report Nos:	50-387/99-302, 50-388/99-302
Licensee:	Pennsylvania Power and Light Company
Facility:	Susquehanna Steam Electric Station
Location:	Berwick, Pennsylvania
Date:	September 21, 1999 (Administration) September 21-30, 1999 (Grading)
Chief Examiner:	John Caruso, Operations Engineer
Approved By:	Richard J. Conte, Chief Human Performance and Emergency Preparedness Branch Division of Reactor Safety

EXECUTIVE SUMMARY

Susquehanna Steam Electric Station
Examination Report Nos. 50-387/99-302 and 50-388/99-302

Operations

One instant SRO and one upgrade SRO applicant were administered an initial written retake licensing exam. Both applicants successfully passed the written retake examination.

Overall, the as-submitted written examination met the guidance of NUREG 1021. Eleven questions required replacement and several changes were also made to question stems to make the question easier to understand and to distractors to make them more plausible.

Report Details

I. Operations

05 Operator Training and Qualifications

05.1 Senior Reactor Operator Initial Written Retake Exam

a. Scope

The NRC Chief examiner reviewed the written retake initial examination submitted by the facility staff to ensure that the examination was prepared and developed in accordance with the guidelines of the "Operator Licensing Examination Standards for Power Reactors" (NUREG-1021, Revision 8). The review was conducted in the Region 1 office. On September 21, 1999, the written exam was administered by the facility's training organization. Grading was completed on September 30, 1999.

b. Observations and Findings

Grading and Results

The results of the exams are summarized below:

	SRO Pass	Fail
Written	2	0
Overall	2	0

There were no facility post examination comments.

Examination Preparation and Quality

The written exam was developed by the facility and their contractor representative using the guidelines of the examination standards. The exam development team was comprised of training and operations representatives. All individuals signed onto a security agreement once the development of the exam commenced. The NRC subsequently reviewed and validated the proposed exam. Some changes and/or additions to the proposed exams were requested by the NRC. Personnel subsequently incorporated the agreed to comments and finalized the exams.

The NRC review of the written examination resulted in the significant revision and/or replacement of eleven questions. Several changes were also made to question stems to make the question easier to understand and to distractors to make them more plausible.

Written Test Administration and Performance

The facility training department performed an analysis of questions missed on the written exam for generic and individual weaknesses. There were eight questions that were missed by both of the applicants. Discussions with the licensee indicate that these questions' subject areas will be discussed with all applicants prior to assumption of any licensed duties. As a result of the licensee's analysis, all eight of these questions were determined to be valid and no post exam changes or comments were requested. The licensee's action was determined to be acceptable.

c. Conclusions

One instant SRO and one upgrade SRO applicant were administered an initial written licensing exam. Both applicants successfully passed the written retake examination.

Overall, the as-submitted written examination met the guidance of NUREG 1021. Eleven questions required replacement and several changes were also made to question stems to make the question easier to understand and to distractors to make them more plausible.

O8 Miscellaneous Operations Issues

The eligibility requirements for both senior reactor operator (SRO) applicants were reviewed and determined to be met.

V. Management Meetings

X1 Exit Meeting Summary

On September 30, 1999, the Chief Examiner provided exam results to a Susquehanna training management representative via telephone.

The Chief Examiner also expressed appreciation for the cooperation and assistance that was provided during the preparation of the exam by the licensee's examination team.

Attachments:
1. SRO Written Exam w/Answer Key

PARTIAL LIST OF PERSONS CONTACTED

<u>FACILITY</u>

R. DeVore	Simulator Instructor
A. Fitch	Supervisor, Operations Instruction

<u>NRC</u>

J. Caruso	Operations Engineer

Attachment 1

SRO WRITTEN EXAM W/ANSWER KEY

NAME Answer Key 9/21/99 Susquehanna

Senior Reactor Operator Answer Sheets

1.	A	26.	A
2.	A	27.	A
3.	C	28.	B
4.	B	29.	D
5.	B	30.	D
6.	D	31.	A
7.	A	32.	D
8.	A	33.	D
9.	A	34.	D
10.	A	35.	D
11.	A	36.	A
12.	C	37.	D
13.	B	38.	A
14.	D	39.	B
15.	C	40.	B
16.	D	41.	A
17.	D	42.	C
18.	C	43.	A
19.	A	44.	B
20.	A	45.	B
21.	A	46.	D
22.	C	47.	B
23.	B	48.	B
24.	B	49.	B
25.	A	50.	C

NAME Answer Key 9/21/99 Susquehanna

Senior Reactor Operator Answer Sheets

51.	C	76.	C
52.	B	77.	B
53.	D	78.	C
54.	B	79.	C
55.	C	80.	D
56.	A	81.	A
57.	B	82.	D
58.	C	83.	B
59.	A	84.	B
60.	A	85.	D
61.	A	86.	C
62.	C	87.	B
63.	B	88.	C
64.	D	89.	C
65.	D	90.	A
66.	D	91.	D
67.	A	92.	B
68.	D	93.	B
69.	D	94.	C
70.	A	95.	A
71.	D	96.	B
72.	C	97.	D
73.	B	98.	C
74.	C	99.	D
75.	A	100.	A

1.A Unit 1 Reactor Shutdown is in progress. Power is at 18%.

- Control Rod 30-35 is being inserted to its Insert Limit of 12.
- During insertion, the position 14 reed switch fails to provide indication.

What is the expected result with respect to Rod Worth Minimizer?

A. Withdraw Block and Insert Block

B. Withdraw Block and Withdraw Error

C. Insert Error and Insert Block

D. Insert Error and Withdraw Error

2.Which condition is the Rod Worth Minimizer designed to limit?

A. Energy deposition to ≤280 cal/gm for an in-sequence Rod Drop Accident.

B. Integral Rod Worth to <1% Δk/k/notch for in-sequence Rod Withdrawals.

C. Fuel clad failure to ≤0.01% for an out-of-sequence Rod Drop Accident.

D. Local to Average Flux ratio to <0.1% for out-of-sequence Rod Withdrawals.

3.A Unit 2 Reactor Startup/Heatup is in progress when a LOCA occurs. Parameters achieved were:

Lowest RPV level (actual)	-55"
Highest drywell pressure	14 psig

Which reactor water coolant sampling flowpath is still available without the reset or override of any NSSS isolations?

A. Reactor Recirc Pump suction via the RWCU System through the F001 and F004 valves.

B. Reactor Recirc Pump discharge via the Recirc sample valves, F019 and F020.

C. Jet Pump discharge via the sample valve SV 12374.

D. RHR Pump discharge through RHR sample valves F079 and F080.

4.Unit 1 is at 75% power.

- A loss of speed control signal exists on the "B" Reactor Recirculation Pump. I&C is investigating.
- A momentary transient on the selected Narrow Range Level transmitter now causes the output to go downscale and return to the present level (35") 4 seconds later.

Predict the response of the Reactor Recirculation System.

A. Both Recirc Pumps will runback to 30% speed 10 seconds after the level transmitter transient.

B. "A" Recirc Pump will runback to 30% speed immediately upon receipt of downscale trip.

C. Both Recirc Pumps remain as is.

D. "A" Recirc Pump will runback to 45% speed 15 seconds after the level transmitter transient.

5.Unit 1 is at 100% when the "A" Reactor Recirc MG Set tachometer output fails downscale. Predict the response to this condition and the required actions if this condition is not corrected.

A. A Scoop Tube Lockup will occur. This will require local manual operation of the Scoop Tube Positioner if a Recirc runback signal is received.

B. Reactor Recirc Pump speed will increase to the upper electrical stop at a rate of 10% per second. This will require local manual operation of the Scoop Tube Positioner to effect a change in pump speed.

C. A Reactor Recirculation Pump trip will occur. This will require entry to ON-164-002 Loss of Reactor Recirc Flow.

D. Reactor Recirc Pump Speed will decrease to minimum speed of 30%. This will require operator action to reduce the speed mismatch between the "A" and "B" Recirc Pumps.

6.Unit 1 at 100%.
Unit 2 Startup in progress, RPV pressure 300 psig. All systems operable on both units.

- T= 0 Loss of Offsite Power (LOOP) occurs.
- T= 3 minutes Fault Lockout occurs on the "1B" ESS Bus.
- T= 5 minutes On Unit 1 a loss of feed results in RPV level decrease to -135"
- T= 7 minutes On Unit 2 a steam leak causes Drywell Pressure to exceed 1.72 psig.

At T= 8 minutes, which RHR pumps will be in service?

	UNIT 1	UNIT 2
A.	1A, 1C, 1D	2B
B.	1A	2C, 2D
C.	1A, 1C	2B, 2D
D.	1A	2B, 2C, 2D

7.Which condition will result in auto closure of RWCU Isolation Valve (HV-F001)?

A. Trip of the "B" RPS EPA breakers.

B. Inboard Isolation logic (RPV Level -38") transmitter (LITS N028C) fails downscale.

C. Standby Liquid Control System initiation.

D. Non-regenerative heat exchanger outlet high temperature (>140 °F).

8.Unit 2 Reactor scram occurred 6 days ago due to a Main Turbine High Vibration trip. .A faulty vibration transmitter has been replaced.

- The plant is in Mode 4.
- "B" Loop RHR is in SDC with the "D" RHR pump.
- RPV Level has been lowered to 35" in preparation for plant startup.
- RCS temperature is currently 110°F.

A "2D" ESS Bus Lockout occurs due to a faulty overcurrent relay. Following the trip, which action would prevent an inadvertant Mode Change?

A. Placing the "B" RHR in SDC within 2.25 hours after the loss of the ESS Bus.

B. Placing the "A" RHR loop in service 3 hours after the loss of the ESS Bus.

C. Placing the "A" Loop RHR in Suppression Pool Cooling and opening 3 SRV's 1.5 hours after the loss of the ESS Bus.

D. Placing the "B" Loop of RHR in SDC Mode from Remote Shutdown Panel 1 hour after the loss of the ESS Bus.

9. Unit 1 was at 50 % power when a Loss of Off Site Power occurred. All D/Gs started and energized the ESS buses.

- HPCI auto initiated and has been placed in Pressure Control Mode.
- RCIC auto initiated and is maintaining RPV level at +13" to +54".
- A steam leak develops in the HPCI PIPE ROUTING AREA.

Assuming the temperature reaches the isolation setpoint, predict the effects on HPCI and RCIC operation.

A. Both HPCI and RCIC will isolate in approximately 15 minutes.

B. HPCI will isolate immediately. RCIC will continue to operate indifinitely.

C. HPCI and RCIC will continue to operate indifinitely.

D. HPCI will isolate in approximately 15 minutes. RCIC will continue to operate indifinitely.

10. Unit 1 is at 100% power.

- During the performance of SO-152-004 Quarterly HPCI Valve Exercising, the power supply to HPCI Vacuum Breaker Isolation Valve HV-155-F079 tripped open immediately when the valve was being re-opened.
- The valve is assumed full closed at this time
- HPCI Vacuum Breaker Isolation Valve HV-155-F075 was tested successfully.

A plant transient occurs and RPV level decreases to the HPCI initiation setpoint. How will this condition affect HPCI operation ?

A. A subsequent restart of HPCI can result in severe water hammer in the steam exhaust line.

B. If HPCI is secured from operation, one vacuum breaker may still operate because HV-155-F075 is open.

C. HPCI will auto initiate, but will trip on Low Suction Pressure.

D. When Low RPV pressure occurs, the redundant HV-155-F075 will ensure vacuum breaker isolation.

11. Unit 1 power ascension was in progress at 30%. A Reactor Scram and MSIV isolation occurred for unknown reasons.

- RCIC is in Pressure Control Mode in accordance with OP-150-001.
- A Maintenance crew on Reactor Building Elevation 749 accidentally bumps into the 1C004 Instrument Rack resulting in Division I Wide Range Level instruments momentarily indicating -40".

Predict the RCIC system response to this event.

A. Valves automatically re-align. Injection will occur.

B. All valves re-align for injection <u>except</u> HV-149-F013 Injection Valve. It must be manually opened for injection to occur.

C. RCIC remains in Pressure Control Mode. Both divisions must be actuated for initiation.

D. RCIC remains in Pressure Control Mode. Initiation is received from Division II instrumentation only.

12. After a LOCA, the following conditions exist:

- RPV pressure is 700 psig and steady.
- Indicated Fuel Zone RPV water level is -180".
- 1A and 1C Core Spray Pumps are the only pumps running and lined up for injection.

Given these conditions when would Rapid Depressurization be required?

A. It is required at this time.

B. After a Table 5 Alternate Subsystem is lined up, pump(s) started, and injecting to max.

C. When RPV level cannot be restored and maintained >TAF.

D. When Minimum Zero Injection RPV Water Level is reached.

13.With Unit 1 at rated power 125 VDC 1D610 is lost. Which of the following will result due to this power loss?

A. HPCI aux oil pump will not start.

B. Div 1 Core Spray initiation logic will not actuate.

C. RCIC outboard injection valve will not close.

D. RHR SDC outboard isolation valve will not open.

14.Unit 2 is operating at rated power when a loss of stator cooling water occurs. If stator cooling water flow is <u>not</u> restored, which protective function will prevent exceeding a thermal limit?

A. The APRM Fixed Neutron Flux Upscale Trip

B. None. Power must be reduced at least 20% and the reactor manually scrammed to avoid a thermal limit violation.

C. An input from reactor vessel instrumentation will input to the RPS circuitry to initiate a scram on high RPV pressure.

D. An input from turbine stop valve position to the RPS scram circuitry initiates a reactor scram.

15. A loss of 125 VDC power occurs to the RPS B trip logic and RPS components supplied by that power. The trip system power loss will:

A. deenergize the "B" scram pilot solenoid valve associated with each Hydraulic Control Unit (HCU). No rod motion will occur directly from this event.

B. deenergize the Group 2 and Group 4 scram pilot solenoid groups. One-half of the control rods will scram.

C. prevent energizing the "B" Backup scram valve if a full scram signal occurs. No rod motion will occur directly from this event.

D. cause the "B" backup scram valve to fail closed. A scram signal will result in no rod motion.

16. During the overlap determination on a Unit 2 startup, IRM C did not show proper overlap. The C IRM was bypassed and the startup was continued.

I & C reports the C IRM has been repaired and can be returned to service. The following conditions exist:

APRM Power	2%
OPERABLE IRM channel status	
IRM readings	All between 20 and 40 / 125
Range	All on range 7

The US directs the PCOU to place the BYPASS switch for C IRM out of BYPASS. When this is performed a ROD OUT BLOCK occurs. No other alarms are received.

Assuming no other malfunctions have occurred, which of the following is the cause of the ROD OUT BLOCK?

A. The high voltage power supply to the IRM C detector is failed.

B. The IRM C MODE SWITCH was left in STANDBY.

C. The C APRM should be checked to ensure no downscale condition exists.

D. The C IRM is not fully inserted.

17. Unit 1 plant conditions during a reactor startup are:

Reactor Period	+200 seconds	
IRMs	FULL IN RANGE 1	Reading 3 / 125
SRMs A & B	FULL IN	Reading 2×10^3 cps
SRMs C & D	PARTIALLY WITHDRAWN	Reading 90 cps
ROD OUT BLOCK annunciator	lit	

What action, if any, must be taken to clear the ROD OUT BLOCK condition and continue rod withdrawal?

A. No action required. When the IRMs rise above 5 on range 1 the ROD OUT BLOCK will clear.

B. Withdraw SRMs A & B. When the indications fall below 10^3 cps rod withdrawal may continue.

C. Insert the in-sequence control rod. When period gets longer than 300 seconds, rod withdrawal will be permitted.

D. Reinsert the C and D SRMs. When the indications rise above 100 cps rod withdrawal is permitted.

18. Unit 1 is at 100% with the feedwater Narrow Range "B" RPV level channel surveillance in progress.

- "A" RPV Level Channel is currently selected.
- The indicating bulb above the "B" Level Select HS-C32-1S01 on panel 1C651 is illuminated.
- The indicating bulb above the "A" Level Select HS-C32-1S01 on panel 1C651 is extinguished.

The above indication informs the operator that the:

A. "B" level transmitter at instrument Rack 1C005 is tripped.

B. "A" level transmitter at instrument Rack 1C004 is bypassed.

C. "B" instrument "Out of Service" switch is in the "Out" position.

D. "A" indicating bulb is burned out.

19. On Unit 2 a small steam leak developed inside Primary Containment with the unit at rated conditions.

- Drywell temperature is 265°F and continues to rise.
- Actual RPV level remains constant.

How will the accuracy of the Extended Range level instrument be affected by this condition?

A. Net decrease in reference leg density will cause an increase in indicated level.

B. Net decrease in variable leg density will cause a decrease in indicated level.

C. Decrease in reference and variable leg density is equal. No change in indicated level.

D. Reg Guide 1.97 level indicators are temperature compensated. No change in indicated level.

20. Unit 1 Startup is in progress. Reactor pressure is 950 psig. Turbine startup preparations are being made.

The Reactor Building NPO discovers that the "B" Core Spray Pump discharge pressure switch has developed a leak. The Operator isolates the pressure switch as directed. No corrective maintanence has been performed at this time.

Which of the following statements would be applicable?

A. ADS remains OPERABLE for 8 days; if not repaired within 8 days declare ADS INOPERABLE.

B. Declare the "B" Core Spray Pump INOPERABLE. Restore the pressure switch to OPERABLE within 7 days or be in MODE 3 in 12 hours and MODE 4 in 36 hours.

C. Declare the "B" Core Spray LOOP INOPERABLE. Restore the loop to OPERABLE status in 7 days or be in MODE 3 in 12 hours and MODE 4 in 36 hours.

D. Declare Division 2 ADS INOPERABLE and place the Division 2 ADS trip system in the trip condition within 96 hours. Repair the pressure switch within 8 days or be in MODE 3 in 12 hours and MODE 4 in 36 hours.

21. A LOCA occurred three minutes ago. All systems responded per design. The following conditions exist:

Drywell Pressure	24 psig
RPV Level	-240" on Fuel Zone
RPV pressure	740 psig lowering rapidly

At this time a loss of offsite power occurs. All diesel generators restore power to their emergency busses. No operator action has been taken. The ADS valves will:

A. remain open throughout the event.

B. close and remain closed throughout the event.

C. close and reopen 3 seconds after power is restored to the ESS Busses.

D. close and reopen 102 seconds after power is restored to the ESS Busses.

22. If the air supply to the SLC level detector component(s) is greater than the required pressure:

A. pump operation will be inhibited.

B. sodium pentaborate may precipitate out of solution.

C. HSBW may be injected with control room indication >2800 gal.

D. the hi / lo tank level alarm will be inoperable.

23.Unit 1 startup is in progress. 1A RHR pump is in Suppression Pool Cooling to reduce pool temperature following SRV operability surveillance.

A small Loss of Coolant Accident occurs. Plant conditions are as follows:

RPV pressure	200 psig dropping slowly (10 psig / minute)
RPV level	45" rising slowly using condensate as inventory source
Drywell pressure	3.5 psig rising slowly
Drywell temperature	185°F steady
"A" RHR Loop flow	2000 gpm

Which of the following is the correct response for conditions currently existing within the "A" RHR loop?

A. Dispatch an auxiliary operator to manually open the discharge valve fully.

B. Verify the minimum flow valve F007A is open.

C. Dispatch an auxiliary operator to locate the leak in the RHR System because no flow should exist within the loop.

D. Close the A loop RHR Inboard injection valve (F015A) to remain in suppression pool cooling on the A RHR loop.

24.Which of the following Manual N4S Initiation Pushbuttons will result in a closure of the Reactor Water Cleanup Inboard Isolation Valve HV-1F001, if armed and depressed?

A. A

B. B

C. C

D. D

25. What action must be performed to minimize the potential for radioactive release from Primary Containment while in the Suppression Chamber Spray mode?

A. Place RHRSW radiation monitor in service.

B. Maintain Figure 4 PSL in the un-shaded area.

C. Ensure LOCA/TRIP ENABLE switch has been reset and white lamp extinguished.

D. Stop Suppression Chamber Sprays when Drywell pressure is reduced to < 1.72 psig.

26. Unit 1 Mode Switch is in REFUEL. The Refueling Platform is in the Spent Fuel Pool area raising a new fuel bundle for core placement. The Hoist Digital display indicates 1080 lbs.

How will the Unit 1 Refueling Platform respond if a control rod is withdrawn one notch?

A. Platform will not be allowed over the Reactor Cavity with this load.

B. The fuel grapple hoist interlock will automatically prevent raising or lowering the load.

C. Platform movement in X, Y, and Z direction will be prohibited.

D. Platform will be allowed over the core, but hoist cannot be raised or lowered.

27.Pressure control in a post-transient condition is being accomplished via SRVs. The G SRV control switch is taken to OPEN to maintain pressure between 800-1087 psig, but no response is observed using all available means. RPV pressure continues to rise and the SRV opens at approximately 1105 psig.

Which condition exists with the G SRV?

A. The G SRV control switch on 1C601 has failed.

B. The G SRV solenoid for manual operation has failed closed.

C. The internal bellows has failed.

D. CIG to the G SRV has been lost.

28.The "F" SRV tailpipe vacuum breaker is failed (stuck) in the position opposite its "normal" position. If the "F" SRV is used for pressure control, the vacuum breaker:

A. being in the OPEN position would result in pressurization of the suppression chamber airspace.

B. being in the OPEN position would result in pressurization of the drywell.

C. being in the CLOSED position may cause the SRV tailpipe to fail when the SRV is initially opened (never open before) because of excessive T quencher loading.

D. being in the CLOSED position may cause the SRV tailpipe to fail when the SRV is reopened several times because of excessive hydraulic loading.

29. Unit 1 is at 100% power. Control Rod Exercising is in progress. Upon selection of Control Rod 30-31, ROD OUT BLOCK and RBM INOP TRIP alarms are received.

Which of the following is the cause of the ROD OUT BLOCK?

A. All four (4) LPRM strings surrounding the control rod have the "A" level detectors bypassed.

B. APRM B has failed downscale.

C. The RBM meter function switch has been taken out of the "Average" position.

D. The RBM nulling sequence has not gone to completion.

30. While in operation at 80% power, the PCO observed the following sequenced conditions:

- Reactor Pressure HIGH
- APRM High Flux
- Reactor scram
- SRV lift to control pressure
- Turbine trip

Which of the following is the cause of this sequence of events?

A. MSIV "A" inadvertently closed.

B. Extraction steam to #3C Feedwater Heater isolated.

C. Master Recirculation Flow Controller failed upscale.

D. Maximum Combined Flow Limiter output failed to minimum.

31. While at 100% power, the Main Shaft Oil Pump (MSOP) discharge pressure slowly begins to decay from a normal operating pressure of approximately 240 psig. Assuming all automatic pump starts occur as a result of this event, if pressure continues to decay:

- What automatic action is expected first?
- What would be the main turbine's final status?

A. Turning Gear Oil Pump Auto Start. Main Turbine trip.

B. Lift Pumps Auto start. No Main Turbine trip.

C. Emergency Bearing Oil Pump Auto Start. Main Turbine trip.

D. Motor Suction Pump Auto Start. No Main Turbine trip.

32. Which condition would result in Main Turbine damage due to overspeed?

A. Main Generator Load Reject during Overspeed Trip Device Functional Test.

B. Intercept Valves fail open following a Main Turbine trip from 18% Load.

C. Main Generator Sync Breaker remains closed following a Main Turbine trip.

D. Extraction Steam Bleeder Trip Valves fail open following a Turbine trip at 50% load.

33. A dual seal failure on the "A" reactor recirculation pump seal has occurred. The Unit Supervisor has directed the Operator to trip and isolate the pump. A manual reactor scram has been initiated due to entry into Region I of the power to flow map. Level is -20" and slowly rising. Drywell pressure is rising.

Condensate and/or Feedwater can be used continuously for RPV level control:

A. until a high drywell pressure condition occurs. No restoration will be possible.

B. regardless of drywell pressure provided the UNDERVOLTAGE TRIP ENABLE switches on the local condensate pump breakers are in the ENABLE position.

C. if HPCI is controlled after its initiation on high drywell pressure to avoid reaching + 54" RPV water level.

D. if the main generator lockout is reset prior to receiving the high drywell pressure condition.

34. A Unit 1 startup is in progress. Automatic feedwater level control is being established per GO-100-002 using the A RFPT. The following conditions exist within the feedwater and feedwater level control systems:

A RFPT individual controller	AUTOMATIC
MASTER Feedwater Level controller	AUTOMATIC
1 Element	SELECTED
MASTER Controller Tapeset value	35"
Feedwater Lo Load Valve	AUTOMATIC
Feedwater Lo Load Valve Demand Signal	38"
RPV pressure	1005 psig
"A" RFP	
Discharge pressure	1065 psig
Discharge valve	CLOSED
Startup Isolation valve	OPEN

Actual level in this lineup is being maintained by:

A. varying the speed of A RFPT using the INCREASE and DECREASE pushbuttons on the MASTER CONTROLLER.

B. varying the speed of A RFPT using the INCREASE and DECREASE pushbuttons on the A RFPT INDIVIDUAL controller.

C. the FEEDWATER LEVEL CONTROL SYSTEM automatically controlling the LOW LOAD STARTUP valve position.

D. the FEEDWATER LEVEL CONTROL SYSTEM automatically controlling A RFPT speed.

35.With Unit 1 at 100% power the following annunciator is received:

INSTRUMENT AC UPS
1D130/PNL 1Y128
TROUBLE/ABNORMAL

Upon investigation at the UPS annunciation panel the NPO reports CB1, the Preferred AC input to 1D130 is tripped.

How long will the 1Y128 loads last following a Loss Of Offsite Power if no operator action is taken and all systems operate as designed?

A. 25 minutes

B. 4 hours

C. 8 hours.

D. >16 hours

36.In accordance with Liquid Radwaste Release (OP-069-050), all of the following activities can be performed with additional requirements EXCEPT:

A. Laundry tank release with the Liquid Radwaste Radiation Monitor inoperable.

B. Liquid Radwaste Sample Tank release with both cooling tower blowdown flowrate instruments inoperable.

C. Any tank release if total site blowdown flow instrumentation inoperable.

D. Distillate tank release with Liquid Radwaste Radiation Monitor inoperable.

37. A work package is being put together to work on a leaking valve in Radwaste. The valve is in close proximity to the Radwaste Collection Tanks.

- Unshielded general radiation field: 100 mr/hr
- Shielded general radiation field: 10 mr/hr

Which method below meets the requirements of NDAP-00-1191 ALARA PROGRAM AND POLICY?

A. One person does the job without shielding in 1 hour and 45 minutes.

B. One person installs shielding in 1 hour and 30 minutes; a different person does the job in 2 hours.

C. Two persons do the job without shielding in 45 minutes.

D. Two persons install shielding in 30 minutes; a different person does the job in 3 hours.

38. A small break in the RWCU system has occurred in the RWCU pump room.

- Power is 65%
- RWCU pump room area temperature is now 135°F
- RWCU area is now 36 mr/hr.
- Refuel Floor High Exhaust Radiation 14 mr/hr

Assuming NO operator actions, Zone 1 reactor building pressure will be maintained by:

A. ventilation exhaust fan discharge damper position.

B. ventilation supply fan discharge damper position.

C. SBGT fan inlet and fresh air inlet damper positions.

D. the number of equipment compartment exhaust fans in operation.

39. With both Units at 100% power and Reactor Buildings HVAC in a normal lineup. A Unit 1 Refuel Floor Exhaust High Radiation condition occurs when a Spent Fuel Assembly is damaged in the Unit 1 Spent Fuel Pool.

How will Reactor Building HVAC respond?

A. Zones 1 and 2 supply, exhaust and filtered exhaust fans stop.
Zones 1 and 2 supply and exhaust isolation dampers close.
Zone 3 remains in service.

B. Zone 3 supply, exhaust and filtered exhaust fans stop.
Zone 3 supply and exhaust isolation dampers close.
Zones 1 and 2 remain in service.

C. Zone 1 supply, exhaust and filtered exhaust fans stop.
Zone 1 supply and exhaust isolation dampers close.
Zones 2 and 3 remain in service

D. Zones 1 and 3 supply, exhaust and filtered exhaust fans stop.
Zones 1 and 3 supply and exhaust isolation dampers close.
Zone 2 remains in service.

40. A Fire in the Main Control Room has resulted in damage to the Control Structure HVAC wiring in the Control Room HVAC Panel 0C681. Control Room Evacuation was required. How is Control Structure HVAC re-established from outside the Control Room?

A. Transfer INSTRUMENT SET 1 to EMERG position at the Unit 1 Remote Shutdown Panel, then proceed with local breaker operations.

B. Place the "A" train of CS HVAC in service at panel 0C879, Area 21-783'.

C. Direct Electrical Maintenance to jumper the start relays at the Local Panel OC877A (B) for the train that was <u>not</u> in service.

D. Place both trains of CS HVAC in service by tripping the CREOASS Process Radiation Monitor in the Lower Relay Room.

41. All of the following require prompt verbal notification of Operations Management in accordance with OP-AD-001, Operations Policies and Work Practices **EXCEPT:**

A: Reactor power reduction initiated for Rod Sequence Exchange.

B. Evacuation of the Turbine Building due to main generator hydrogen leak.

C. A mispositioned control rod or unintended control rod motion.

D. Transporting a non-contaminated, injured person to the Berwick Hospital.

42. Unit 1 was at 100% power. Both Reactor Recirc Pumps were in Master Manual control. An inadvertent runback to the #1 Limiter of the "A" Reactor Recirc Pump has just occurred.

- Total Core flow indicates 71 Mlbm/hr.
- "A" Reactor Recirc Loop Jet pump flow indicates 3.6 Mlbm/hr.
- "B" Reactor Recirc Loop Jet pump flow indicates 67.4 Mlbm/hr.

Which of the following actions is required?

A. Immediately reduce the "B" Reactor Recirc Pump speed to ≤55 Mlbm/hr.

B. Correct flow mismatch within 24 hours or be in Mode 3 within the next 12 hours.

C. Declare the "A" Reactor Recirc Pump "not in operation" within 2 hours.

D. Immediately reduce power to below the 70% Rod Line of the Power/Flow Map.

43. Unit 1 returned to 100% power 1 hour ago following Feedwater Heater maintenance. Reactor Engineering reports that due to a problem with the Heat Balance program, calculated thermal power has been 70 MW_{th} lower than actual thermal power for the previous 2 hours.

Which action(s) are required?

A. Reduce thermal power to ≤100% immediately. Notify NRC Operations Center within 24 hours.

B. Reduce thermal power immediately to restore the hourly average to ≤100%. Nuclear Fuels group must ensure reload analysis has not been violated. Notify NRC Operations Center within 4 hours.

C. Reduce thermal power to ≤80% within 15 minutes. Commence a controlled shutdown. Notify NRC Operations Center within 1 hour.

D. Reduce power to ≤25% within 2 hours. Notify NRC Operations Center within 1 hour.

44. Unit 2 power ascension is in progress with reactor power currently at 18%. When Control Rod 18-19 is selected for motion from position 00 to 04, the Rod Sequence Control System locks up with WH BLK and INS BLK illuminated on the RSCS Display. Rod Worth Minimizer displays remain normal. When the Control Rod is de-selected, RSCS returns to normal status. No other abnormal alarms or indications are received.

Using the attached NDAP-QA-0338 attachment I, Reactivity Control System Bypass Authorization Form, determine the appropriate action.

A. Bypass RSCS in accordance with OP-156-002 using keylock switch on Control Room Panel 1C651 and apply appropriate Tech Specs. Unbypass RSCS when no longer selected to this rod.

B. Bypass the Control Rod in accordance with OP-156-002 using the Bypass Switch Card and apply appropriate Tech Specs. Unbypass the rod when condition corrected.

C. Bypass the Control Rod in accordance with OP-156-001 at the Rod Drive Control Cabinet and apply appropriate Tech Specs. Unbypass the rod when condition corrected.

D. Select substitute position and notify I&C.

45. Select the statement accurately describing <u>why</u> a significant difference exists between Unit 1 and Unit 2 Station Blackout coping strategies.

A. Unit 1 magenta colored instrumentation requires Blue Max for extended operation.

B. Unit 2 has a separate Non-Vital battery bank.

C. Unit 1 normally carries more Common Loads.

D. Unit 2 relies upon Unit 1 for ESW cooling.

46. Which activity can be authorized by the System Operating Representative (SO Rep) when it involves a Status Control or Red Tagged component?

A. Cycling of the breaker for a Red Tagged valve to ensure power leads have been re-terminated correctly.

B. Installation of a blank flange on a valve body flange to maintain boundary protection while permitting manual valve cycling.

C. Cycling of the Red Tagged valve to ensure no thermal binding of the valve stem has occurred.

D: Operation of a Status Control Tagged valve to drain a system prior to Active status of the permit.

47.While performing Core Alterations on Unit 1, a fuel assembly located in the core is engaged out of sequence. The error is identified just after the fuel assembly is raised above the Top Guide. Refueling activities are halted to notify the Unit Supervisor of the error.

SELECT the required response to this event.

A. Notify Reactor Engineer. If Shutdown Margin can be verified, obtain a FACCTAS Change Notice and return the fuel assembly to it's original core location.

B. Raise fuel assembly to full-up position and place the bundle into the Safe Setdown location in the Fuel Pool.

C. Note the error in the FACCTAS. With permission from Operations Manager and Reactor Engineer, return the fuel assembly to it's original core location.

D. Place the fuel assembly into the closest rodded cell. Notify Reactor Engineer to verify Shutdown Margin.

48.Which statement accurately describes the control rod programming of the Rod Sequence Control System?

A. After Group 1 rods are fully withdrawn to position 48, the next rod to be withdrawn may come from either Group 2, Group 3, or Group 4. Notch restraint will be enforced between notch position 00 and 12 for insert and withdraw.

B. With RSCS Groups 1 and 2 fully withdrawn to position 48, either RSCS Group 3 or Group 4 control rods may be withdrawn. Notch restraint will be enforced between 00 and 12 for insert and withdraw.

C. With RSCS Group 1 withdrawn to position 48, the second group to be moved must be RSCS Group 2. Notch restraint will be enforced between notch position 00 and 24 for withdraw only.

D. With RSCS Groups 1 and 2 fully withdrawn to position 48, RSCS Group 3 or Group 4 may be withdrawn to position 48. Notch restraint will be enforced between 00 and 24 for insert and withdraw.

49. The Liquid Radiation Release Monitor is determined to be INOPERABLE. Prior to the planned release from Sample Tanks "A" and "B":

A. the liquid radiation monitor must be returned to OPERABLE status.

B. the expected dose projection for the tank to be released must be manually, independently calculated.

C. interlocks associated with low blowdown flow effluent valve isolation must be bypassed.

D. the cooling tower blowdown valves must be fully opened to provide maximum dilution flow.

50. Unit 1 is at 30% power. A shutdown is in progress with preparations being made to purge Primary Containment.

Using the attached figure, SELECT purge lineup and condition which is permitted by NDAP-QA-0309 Primary Containment Control.

A. With either SBGT System train inoperable, purge the Drywell or Suppression Chamber through the 18" / 24" Vent Isolation Valves provided lineup does not exceed 90 hours per 365 days.

B. With both SBGT System trains operable, purge the Drywell and Suppression Chamber concurrently through the 18" / 24" Vent Isolation Valves to either train.

C. With both SBGT System trains operable, purge the Drywell or Suppression Chamber through the 2" Bypass Valve to either train.

D. With both SBGT System trains operable, purge the Drywell or Suppression Chamber through the 2" Bypass Valve to both trains.

51.An Alert has been declared due to a pipe break inside Secondary Containment. Elevated radiation levels exist. You, as Emergency Director, direct sending personnel into the area to attempt a valve closure that will terminate the event.

The individual assigned to the task has documented lifetime exposure. He has received 350 mREM Total Effective Dose Equivalent (TEDE) exposure this year and 1200 mREM lifetime TEDE. Assuming the exposure received during this evolution will also be whole body, the maximum dose in (mREM) this individual can receive <u>WITHOUT</u> a dose extension authorization is:

A. 23,800.

B. 8,500.

C. 1,650.

D. 800.

52.A Unit 2 Reactor Scram and MSIV isolation has occurred from a false high steam flow signal. Several SRVs were manually opened to maintain RPV pressure less than 1087 psig. Suppression Pool temperature has risen to 108°F. Drywell pressure has been slowly increasing in pressure and is currently at 1.3 psig. 1 hour has elapsed since the reactor scram. As unit Supervisor, you direct the PCO to vent Drywell using the "A" SBGT train in accordance with EO-100-103 step PC/P-1.

Which statement below meets the requirements for venting the Drywell?

A. If a valid SBGT System SPING HI HI alarm is received due to this evolution, vent release path must be terminated and execution of EO-100-104 is required.

B. Technical Requirements must be verified within 4 hours prior to establishing vent release path.

C. A Noble gas grab sample must be obtained and analyzed prior to vent release because SRV lifts have occurred.

D. Suppression Chamber to Drywell Vacuum Breakers must be successfully tested prior to vent release because SRV lifts have occurred.

53. With Unit 1 at rated power a valving error resulted in an introduction of water from a source external to primary containment. The suppression pool level rise was terminated at 24' 4".

Technical Specifications / Technical Requirements require:

A. declaring RHR and Core Spray subsystems inoperable. Enter LCO 3.0.3 immediately.

B. restoring suppression pool level to within limits within the next four hours or be in MODE 3 in 12 hours and MODE 4 in 36 hours.

C. declaring primary containment inoperable. Restore suppression pool water level to within limits within one hour or be in MODE 3 in twelve hours and MODE 4 in 36 hours.

D. restoring suppression pool water level to within limits within the next two hours or be in MODE 3 in 12 hours and MODE 4 in 36 hours.

54. During a failure to scram with initial ATWS power >5%, water level was lowered to -65 inches and is being maintained.

Which of the following is an advantage of maintaining level at -100 inches instead of -65 inches?

A. Core inlet subcooling is greater.

B. Power reduction is greater.

C. RPV level control is easier.

D. HSBW is achieved sooner.

55.EO-100-103, Primary Containment Control ensures the start of HPCI and RCIC if Suppression Pool level reaches 26'. What is the potential danger with such a high suppression pool level?

A. If level continues to rise, the HPCI and RCIC turbine exhaust vacuum breaker suppression chamber penetrations will be covered. Filling these lines will result in HPCI and RCIC exhaust line rupture diaphragms becoming flooded.

B. The HPCI and RCIC barometric condenser vacuum pump discharge lines will fill with water and provide a potential release path to the reactor building as the discharge will travel out the pump seals rather than the discharge line.

C. The elevated suppression chamber water level provides excessive exhaust backpressure. If either is started, exhaust pipe damage may result.

D. The increased suppression pool water volume will place undesirable loading on the suppression chamber walls resulting from HPCI and RCIC exhaust forces if either starts. This force may exceed suppression chamber hydrodynamic design pressures.

56.During a station blackout, the PCO reports the following:

- A Diesel is not running. There is less than 20 psig in the air receivers.
- C Diesel is running with no voltage. Electrical can fix the problem in 20 minutes.
- B Diesel tripped on a generator differential. Electrical is investigating.
- D Diesel tripped on low lube oil pressure due to a large leak.

Based on this information, which of the following actions is the highest priority?

A. Shutdown "C" Diesel.

B. Substitute "E" Diesel for "B" Diesel.

C. Reset the "B" Diesel Generator Differential Relay.

D. Substitute "E" Diesel for "A" Diesel.

57.Given the following plant parameters:

Rx Mode Switch Position:	SHUTDOWN
RPV pressure:	125 psig
Narrow Range level	+25"
Drywell Pressure	0.08 psig

All of the following alarms are consistent with these plant parameters EXCEPT:

A. RECIRC LOOP B
DISCHARGE VLVS AUTO CLS
LO PRESS PERMISSIVE

B. RHR LOOP A
DISCH/SHTDN SUCT HDR
HI PRESS

C. PRIMARY CONTAINMENT
HI-LO PRESS

D. MSIV CLOSURE
BYPASS

58.Which condition constitutes a thermal limit violation?

A. THERMAL POWER at 10% with reactor pressure at 650 psig.

B. Peak transient reactor pressure reached 1275 psig.

C. THERMAL POWER at 30% with core flow at 6 Mlbm/hr.

D. MCPR at a value greater than the applicable limit for the current power to flow conditions.
$MCPR_{actual} > MCPR_{limit}$

59.While at 100% power, a recirculation transient caused by a short in the control circuitry occurs. Immediately following the transient, the plant stabilizes with the following parameters:

- Reactor Power 50%
- "A" Recirc pump tripped
- "B" Recirc pump at 45% speed
- Loop "A" total jet pump flow is 20 Mlbm/hr
- Loop "B" total jet pump flow is 56 Mlbm/hr
- Total indicated core flow 36 Mlbm/hr

What is actual core flow, and how will the loss of the "A" Recirc pump affect the APRM Scram setpoint?

A. 36 Mlbm/hr. Setpoint needs to be adjusted to 0.58W + 57%.

B. 36 Mlbm/hr. Setpoint unaffected.

C. 76 Mlbm/hr. Setpoint needs to be adjusted to 0.58W + 57%.

D. 76 Mlbm/hr. Setpoint unaffected.

60.Unit 2 is operating at 100% power.. The following annunciators are received:

- CONDENSER LO VACUUM
- OFF-GAS HI/LO FLOW
- STEAM SEAL HDR HI/LO PRESS
- GEN VOLT REG AUTO TO MAN SETPOINT UNBALANCE

SELECT the initiating cause and the action(s) required.

A. Loss of steam seals. Break condenser vacuum if seals cannot be restored because steam seal / turbine damage may result.

B. Vacuum bellows failure. Reduce power to 70% and isolate affected condenser. If condenser vacuum continues to decay, perform Scram Imminent actions.

C. SJAE failure. Reduce power to restore condenser vacuum and reduce off gas steam dilution flow to prevent choking of the SJAE nozzle.

D. Steam Packing Exhauster failure. Place standby Steam Packing Exhauster in service. If condenser vacuum continues to decay, make preparations to remove Main Turbine from service.

61. A Station Blackout has been in progress for forty minutes. All actions and Emergency Support procedures required to be accomplished based upon the time requirements and systems required for safe shutdown during blackout are complete.

Plant conditions:

RPV	
Pressure	1005 psig rising slowly
Level	+25" steady
Containment	
Air temperature	185 °F rising slowly
Suppression Pool water temperature	105 °F rising slowly
Drywell pressure	1.6 psig rising slowly
Suppression Pool level	24' rising slowly

If operation of HPCI is being performed in accordance with EO-100-032, HPCI Operating Guidelines During Blackout and EO-100-030, Unit 1 Response to Station Blackout, the control room operator should:

A. altemate operation between Injection and Pressure Control modes as required.

B. ensure HPCI realigns to the injection mode when drywell pressure exceeds 1.72 psig and then secure HPCI to avoid RPV overfill.

C. ensure HPCI suction realigns to the suppression pool when suppression pool level reaches 23.75 '.

D. prevent HPCI injection and subsequent operation. . HPCI will not be required for continued operation during Station Blackout.

62. Unit 1 is normal at 100% power when the "1A" ESS Bus voltage decreases to 75% nominal value due to ESS Transformer trouble. 6 minutes later Unit 1 experiences a LOCA as evidenced by RPV level decreasing to -135 inches.

With the electrical lineup at the time of the LOCA, the 1A RHR and 1A Core Spray pumps will start per design:

A. and continue to run, drawing excessive current because Normal Supply voltage did not decrease to < 65% for 3 seconds. Analyzed response times will not be met for these 2 pumps only.

B. then will trip on undervoltage to prevent Bus damage from excessive current draw. Analyzed response times will not be met for these 2 pumps only.

C. with Alternate Supply maintaining bus voltage. Analyzed response times will be met.

D. when Plant Auxilliary Load Shedding is completed. Analyzed response times will be met.

63. Units 1 and 2 are in MODE 1. Electrcial Distribution is in a normal lineup when Electrical Maintenance reports one 0D595 battery terminal appears to have been damaged. Resistance taken on the terminal indicates 70.0 x E^{-6} ohms.

SELECT the correct Technical Specification action.

A. Restore battery bank 0D595 to OPERABLE status within 2 hours or be in MODE 3 in 12 hours and MODE 4 in 36 hours.

B. Verify all ESW valves associated with "E" Diesel Generator are closed within 2 hours.

C. Restore battery bank 0D595 to OPERABLE status within 2 hours or declare "E" Diesel Generator INOPERABLE.

D. Declare "E" Diesel Generator INOPERABLE; perform of site circuit breaker alignment surveillance within 1 hour and once per 8 hours thereafter until the battery bank is OPERABLE.

64. While in operation at power, the Control Room Operator acknowledges the following Core Spray Division A BIS annunciator:

AR-154-001 PUMP C BKR POWER LOSS

Consequently, the amber "STOP" indication lamp for the "C" Core Spray pump is extinguished on panel 1C601. What is a consequence of this condition?

A. This Core Spray loop will not meet the criteria of a "TABLE 4 SUBSYSTEM".

B. Power has to be restored to this circuit by eliminating the malfunction and restoring power from that same circuit to restore pump operation.

C. The "C" Core Spray pump will not automatically start but can be manually started via control room control switch.

D. The "C" Core Spray pump can be returned to operation by manually transferring its control power supply to Unit 2 Division I DC power.

65. A Unit 1 shutdown is in progress. The crew is ready to remove the Main Turbine from the grid. The LOAD SELECTOR DECREASE pushbutton is depressed to reduce generator load to 5% but it sticks in the depressed position and all load from the main generator is removed. The synchronizing breaker opens on reverse power.

Assuming the main generator trip <u>FAILS TO CAUSE A TURBINE TRIP</u>

A. the Power / Load Unbalance circuit will trip the turbine.

B. the Control Valve Fast Closure logic will trip the turbine and scram the reactor.

C. the Overspeed Protection circuit will trip the turbine and the reactor will scram on high flux or pressure.

D. the Speed Regulation circuit will allow turbine speed to rise and then be controlled by the Control and Intercept valves.

66.Unit 2 shutdown is in progress because the #2 turbine stop valve would not close during testing. At 40% power high vibrations are experienced which exceed the trip setpoint. The plant responds as expected with the exception that the #2 Stop Valve remains open.

What is the condition of the main turbine and the reactor?

A. The main turbine is not tripped, and the reactor did not scram

B. The main turbine is not tripped and the reactor is scrammed.

C. The main turbine is tripped, and the reactor did not scram

D. The main turbine is tripped, and the reactor is scrammed.

67.A manual scram from 100% power has been initiated by placing the Mode Switch to SHUTDOWN. Which of the following would be a direct response?

A. Full Core Display has swapped to Full In-Full Out.

B. Alternate Rod Insertion block and vent valves have repositioned.

C. The turbine stop valve and control valve scram signals are bypassed.

D. Scram Group Solenoids indicating lamps in the relay rooms (1C609 and 1C611) have illuminated.

68.The Immediate Operator Actions of SCRAM procedure ON-100-101 accomplish all of the following actions EXCEPT:

A. Insertion of a manual RPS trip signal.

B. Verification as to whether or not the reactor will remain shutdown under all conditions without boron.

C. Verification of the system status to prevent potential radioactivity release.

D. Prevention of excessive RPV cooldown from steam usage by the main turbine.

69.Unit 1 is operating at 100% power. If the "INCREASE" Pressure Set pushbutton is maintained depressed, predict the response of the Main Turbine Bypass valves, RPV level and Reactor power.

	Bypass Valves	RPV Level	Reactor power
A.	Remain closed	Increase	Increase
B.	Valves open	Increase	Decrease
C.	Valves open	Decrease	Decrease
D.	Remain closed	Decrease	Increase

70.With Unit 2 at 45% power a Load Reject Without Bypass Valves occurred. A reactor scram has occurred. Plant conditions are listed below.

Highest RPV pressure	1130 psig
Lowest RPV level	-24"

SELECT the automatic protective functions that have occurred.

A. Reactor Recirculation pumps trip, ARI initiates, only 10 SRVs open.

B. Reactor Recirculation pumps trip, ARI initiates, only 6 SRVs open.

C. ARI initiates, RCIC initiates, only 8 SRVs open.

D. Reactor Recirculation pumps trip, only 2 SRVs open.

71.Following a reactor scram from loss of normal feedwater, HPCI and RCIC have automatically initiated on Low RPV water level signals. Assuming no operator action is taken to control RPV water level and these systems recover level, the final condition of the RCIC System valves listed will be:

	F007 Inbd Stm Iso	F008 Outbd Stm Iso	F045 Stm to RCIC Turb	HV15012 Turb T&T Valve	FV15013 Turb Gov Valve
A.	CLOSED	CLOSED	CLOSED	CLOSED	CLOSED
B.	OPEN	OPEN	OPEN	CLOSED	CLOSED
C.	CLOSED	CLOSED	OPEN	OPEN	OPEN
D.	OPEN	OPEN	CLOSED	OPEN	OPEN

72.With Unit 1 at 90% power in a normal lineup and no equipment out of service, the B Condensate pump trips. RPV level drops to the low level alarm setpoint.

Assuming <u>no</u> automatic plant response occurred, which action would you direct?

A. Attempt to restart the Condensate pump.

B. Perform scram imminent.

C. Reduce recirculation pump speeds to 45%.

D. Reduce reactor power to 80% via recirculation flow reduction.

73. With Unit 1 at 40% power, makeup to the drywell is in progress from the 1" nitrogen makeup line. A small steam leak develops within containment.

Assuming drywell pressure continues to rise, the first plant condition requiring automatic closure of the nitrogen makeup valves (SV 1537 and SV 1538) is:

A. RPV level +13".

B. drywell pressure of 1.0 psig.

C. RPV level -38".

D. drywell pressure of 1.72 psig.

74. EO-100-104, Secondary Containment Control, provides a table of radiation levels to be used to determine decision paths. A MAXIMUM SAFE radiation level is based on:

A. receipt of AR-100-001, A04, Refuel Floor Wall Exhaust High Radiation annunciator.

B. radiation levels which would exceed 10CFR20 limits for one hour exposure.

C. radiation levels which would exceed emergency exposure limits.

D. receipt of AR-100-001, B05, Rx Bldg Hi Rad , from any input to that annunciator.

75. A reactor scram and MSIV closure has occurred as a result of a Unit 1 transient. Plant conditions:

- HPCI is in pressure control mode
- RCIC is being used for RPV level control
- Suppression pool temperature 132°F rising
- Drywell pressure 0.5 psig steady
- Suppression pool level 23.6' rising

If RPV level reaches -40", extended operation without performing any Emergency Support Procedure actions would result in:

A. loss of HPCI from high lube oil temperature.

B. loss of RCIC due to high suppression pool level.

C. high radiation in the Reactor Building from the HPCI and RCIC barometric condensers.

D. cavitation of the HPCI and RCIC from elevated Suppression Pool temperature.

76. A reactor Startup was in progress following a 7 day forced outage. Criticality was achieved and heatup in progress. Problems with EHC delayed the rod withdrawals for approximately 1.5 hours. RPV pressure decreased from 360 psig to 325 psig during this delay. Control rod 14-19 was withdrawn one notch from 10 to 12 in order to re-establish a heatup rate. Reactor period continued to shorten and the operator re-inserted the rod to determine why period indication did not return to infinity just as the rods pulled prior to the delay.

Which condition explains the reactor period response?

A. A positive Moderator Temperature Coefficient exists.

B. The Void fraction had decreased due to the lower pressure.

C. Moderator temperature had decreased.

D. Xenon burnout in the high flux region had begun.

77. A reactor shutdown has been initiated following a report from the Reactor Engineer that a recent reactivity evaluation determined that the actual core value is 1.05% $\Delta k/k$ greater than the predicted value. Just after the shutdown was initiated, a loss of feedwater heating was experienced.

Why is there a concern about the above situation?

A. The effects of the transient on the fuel will be less severe than had been analyzed.

B. The effects of the transient on the fuel will be more severe than had been analyzed.

C. The reactivity addition rate for the core will be slower than analyzed.

D. The reactivity addition rate for the core will be faster than analyzed.

78. Refer to the attached prints.

While operating at full power a power excursion to 125% occurs and the following annunciators are received:

AR-104-A01 RPS CHANNEL B1/B2 AUTO SCRAM
AR-104-A04 NEUTRON MONITORING CHAN B SYSTEM TRIP

The immediate operator actions are executed per ON-100-101, Reactor Scram. <u>NO</u> control rods move.

The <u>MINIMUM</u> action(s) that will result in control rod insertion is(are):

A. arming and depressing the "A1" OR "B1" manual scram pushbutton.

B. arming and depressing the "A1" AND "B1" manual scram pushbuttons.

C. pulling fuse F14A OR F14C.

D. pulling fuses F14A AND F14C.

79.One hour has elapsed since a steam line break occured in the Turbine Building pipe tunnel. The transient has caused fuel damage, a reactor scram and MSIV isolation.

There is indication of flow on the "C" Main Steam Line flow instrument
RPV level is 45" steady
RPV pressure is 760 psig dropping slowly
OSCAR informs you steam is issuing from the Turbine Building blowout panels
U1 SPING readings

Noble Gas	9.2 E6	micro Curies / min
Iodine	3 E-4	micro Curies / min
Particulate	5 E2	micro Curies / min

Chemistry is sampling reactor coolant. No results are yet available.
30 minutes ago HP informed you the dose rate on Township Roadway 700 mRem / hr TEDE.

Classify the event.

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

80.A Unit 1 startup is in progress. The following conditions exist:

RPV

Temperature (coolant)	235°F
Pressure	5 psig
Power	IRM Range 8

Vacuum is being established using the mechanical vacuum pump.

A rod drop accident occurs causing fuel element failure. The resulting radiation release may produce automatically initiated protective functions. What action would be the direct result of this event?

A. Off-gas Isolation and Mechanical Vacuum Pump trip.

B. Turbine building HVAC trip and fan damper closure.

C. MSIV closure and Off-gas isolation.

D. Reactor scram, MSIV closure and Mechanical Vacuum Pump trip.

81. A leak in the Reactor Building Chilled Water (RBCW) System within the Reactor Building required its shutdown and isolation. What are the proper operator actions for this condition?

A. Verify Reactor Building Closed Cooling Water (RBCCW) automatically aligns to replace RBCW as the cooling water supply to the drywell coolers.

B. Reduce power in preparation for removing the unit from service due to loss of cooling water to the Reactor Recirculation pump seal and motor bearings.

C. Reduce power to reduce heat input to the primary containment. Cooling water flow has been lost to the drywell cooling units.

D. Place Emergency Service Water in service to supply RBCCW Heat Exchangers.

82. A loss of instrument air occurred on Unit 1. Its effects on the Reactor Building HVAC System will be:

A. RB HVAC will continue to operate in the same condition as prior to the instrument air loss. No isolation will occur if an initiation condition is received.

B. Supply and exhaust fans will trip. Recirculation system dampers fail open. Zone I pressure will be maintained negative as SBGT starts on equipment compartment low flow. Zone III will fluctuate with atmospheric and building conditions.

C. All dampers associated with RB HVAC components will close. Zone III pressure will remain negative as SBGT starts on equipment compartment low flow. Zone I will fluctuate with atmospheric and building internal conditions.

D. Supply and exhaust fans will trip. Recirculation system dampers fail open. SBGT will remain idle. Reactor building pressure will rise to ambient pressure.

83. While operating at rated power, an I&C technician inadvertently causes a Drywell Cooling Isolation. The PCO confirms Drywell Cooling header isolation valves are closed. Troubleshooting is in progress.

Shortly after the isolation the following annunciator is received:

CONTAINMENT DRWL CLG LOOP A HI TEMP

Assuming these valves cannot be immediately re-opened, predict the expected plant response to this event.

A. RBCW and RBCCW systems will swap and RBCCW will supply drywell cooling fan coils.

B. Drywell cooling fans in AUTO HIGH will auto start on high air temperature exiting the fan coil.

C. A power reduction will be required because of rising Reactor Recirculation Pump motor temperatures.

D. The standby RBCW chiller will start to supply additional cooling to the RBCW system.

84. During a plant startup RPV pressure is 850 psig when a loss of suction causes the "A" CRD pump to trip. The "B" CRD pump is out-of-service for maintenance.

If a reactor scram occurs, control rods will:

A. fully insert at normal speed.

B. fully insert at slower than normal speed.

C. partially insert at normal speed.

D. partially insert at slower than normal speed.

85. Unit 2 is in Shutdown Cooling using 2A RHR pump. Temperature control is being accomplished by throttling RHRSW flowrate via the RHRSW heat exchanger inlet valve and the RHR HX bypass valve.

An inadvertent high drywell pressure signal occurs. How will 2A RHR pump respond?

A. Trip and remain secured.

B. Trip, then restart when Shutdown Cooling suction valves are fully closed.

C. Remain running with no change in flowrate.

D. Remain running, with increase in flowrate

86. Unit 1 was operating at 100% power when the unit scrammed on High Drywell pressure. Immediate Operator Actions have been completed. The leak has been confined to the Drywell. ESW is supplying cooling water to the TBCCW system.

Which of the following is a correct statement, for this situation?

A. If <u>no</u> MSIVs are closed, it is due to a failure in the NSSSS logic.

B. If <u>all</u> MSIVs are closed, it is due to a loss of 90# nitrogen header to the MSIVs.

C. If <u>all</u> MSIVs are closed, it is due to a loss of condenser vacuum.

D. If <u>all</u> MSIVs are closed, it is due to leak detection isolation.

87. A Unit 2 transient has resulted in a loss of normal feedwater and a full reactor scram. EO-200-102 is being executed.

Plant conditions are:

RPV pressure	1000 psig rising at 10 psi / minute
RPV level	-20" rising approx. 3" / minute
MSIVs closed	
SRVs for pressure control	

HPCI has been placed into service in the injection mode to restore RPV level + 13 to + 54". Which of the conditions below related to the HPCI operation could result in the failure to restore RPV level to the normal band with HPCI.

A. "C" SRV sticks open.

B. The operator has placed the HPCI controller to MANUAL.

C. A suppression pool high level occurs.

D. Loss of Division I DC to the HPCI isolation circuitry.

88. Which of the following conditions would first require a reactor scram?

A. 100% Reactor Power
HPCI inadvertent initiation causes power to increase to 106%.
SPOTMOS reaches 105°F before overridden off.

B. Reactor Startup RPV pressure at 160 psig
HPCI 24 Month Flow Verification testing in progress
SPOTMOS reaches106°F when test is terminated.

C. Reactor Startup, RPV pressure at 150 psig
ADS 24 Month Valve Manual Actuation surveillance in progress.
1 SRV sticks open for 1.5 minutes
SPOTMOS reaches 110°F maximum after the valve is closed.

D. 100% Reactor Power
SRV Inadvertent opening
SRV closes after 1 minute.
SPOTMOS reaches 106°F maximum while the valve is open.

89.A turbine trip and hydraulic ATWS occur. EO-100-113 and EO-100-103 are currently being executed. Current plant conditions:

RPV Parameters

Initial ATWS power	65%
Pressure	950 psig with BPVs controlling
Level	-80" with RCIC and CRD injecting
Power	IRM Range 8 50 / 125 (3% power) decreasing
SLC	injecting

Containment Parameters

Suppression Pool water temp.	192°F steady
Suppression Pool level	25' rising slowly

Which of the following actions will restore the required margin of safety?

A. Lower Suppression Pool level.

B. Open additional Bypass Valves.

C. Reduce Suppression Pool water temperature .

D. Rapidly Depressurize the RPV.

90.Which transient would be the most likely cause of a Pressure Suppression Pressure Limit violation?

A. A LOCA results in the breaking of one downcomer leg within the Suppression Chamber airspace above the water level in the suppression pool.

B. Full power ATWS with MSIV closure and inability to drive rods within the first fifteen minutes.

C. A rapid depressurization manually initiated as a result of EO-100-103 execution resulting from loss of Suppression Chamber inventory into the "B" Core Spray room.

D. When SRVs are opened for rapid depressurization, a vacuum relief valve on one of the SRV tailpipes fails open.

91. A LOCA has occurred on Unit 1. Actions have been taken in accordance with all applicable Emergency Operating Procedures. Present plant conditions are:

All control rods are fully inserted
Lowest RPV level was –135"
RHR "C" is the ONLY injection source available and is being used to maintain level.

RPV water level instrumentation is	OPERABLE
RPV Level	-110 inches rising ten (10) inches per minute
RHR flow rate	8500 gpm (maximum available)
Supression Chamber pressure	4 psig
Suppression Pool water temperature	185°F
Suppression Pool level	18.5 feet

Which of the following describes RHR operating requirements?

A. Secure the RHR pump.

B. Reduce and maintain RHR flowrate at < 6800 gpm.

C. Reduce RHR flow and attempt to control RHR flow less than 6800 gpm unless RPV level cannot be maintained >-129".

D. Maintain RHR flow at 8500 gpm and restore RPV level +13" to +54".

92. Unit 2 is at 100% power when a faulty relay causes a lockout on 13.8 KV Auxiliary Bus 12B.

ON-203-003, 13.8 KV BUS 12A and 12B LOSS OF BUS LOAD SHEDDING ON BUS UNDERVOLTAGE has been entered.

What is the expected plant response, and which operator actions would be the highest priority?

A. Loss of 2 Circ Water Pumps. Enter ON-243-001 Loss of Main Condenser Vacuum and reduce reactor power to maintain Main Condenser available.

B. Low RPV water level from loss of 2 Condensate Pumps. Enter EO-200-102 RPV Control and initiate HPCI and/or RCIC as needed.

C. Loss of 2 Service Water Pumps. Enter ON-211-001 Loss of Service Water and place ESW in service to supply RBCCW and TBCCW Heat Exchangers.

D. 2A Reactor Recirculation Pump trip. Enter ON-264-002 Loss Of Recirc Flow and begin an orderly shutdown because Single Loop Operation is not allowed.

93.Unit 2 has sustained a total loss of normal feedwater. Prior to the transient, RCIC was out of service. HPCI has received a low RPV water Level 2 start signal. HPCI initiates but subsequently trips. Review of the HPCI panel reveals the following HPCI System status:

HPCI LOW FLOW	In alarm
HPCI TURBINE TRIP SOL ENER	NOT in alarm
HPCI TURB BRG LO OIL PRESS	In alarm
HPCI OUT OF SERVICE	In alarm

RPV Parameters	
RPV Level	-20 lowering
RPV pressure	950 psig
HPCI Turbine speed	3200 rpm lowering

Assuming RPV level continues to lower, what actions must be taken to restore HPCI to the injection mode?

A. Nothing. HPCI will auto start at the low RPV level initiation signal (-38").

B. Investigate the low lube oil pressure condition. The auxiliary oil pump and oil system must be verified to operate before HPCI can be used again.

C. Wait until HPCI rpm is below 2200 rpm. Shut down the system and manually reinitiate it with the speed controller in MANUAL.

D. When low RPV level initiation signal (-38") is received again, restart HPCI component by component.

94. I&C has placed the Leak Detection Bypass Switch "HPCI DIV 1 ISOLATION TEST BYPASS" keylock switch on Control Room Panel 1C614 to the "BYPASS" position in order to perform required quarterly surveillances. The following alarm is received on Panel 1C601:

HPCI / RHR / RWCU LEAK DETECTION LOGIC "B" IN TEST BYPASS

With this configuration, how will a valid high HPCI ROOM area temperature affect the HPCI system?

A. Steam Isolation Valve F003 and Suppression Pool Suction Valve F042 will close and HPCI Turbine will trip

B. Steam Isolation Valve F003 and Suppression Pool Suction Valve F042 will stay open and HPCI Turbine will <u>not</u> trip.

C. Steam Isolation Valve F002 and Warmup Valve F100 will close. HPCI Turbine will trip.

D. Steam Isolation Valvel F002 and Warmup Valve F100 stay open. HPCI Turbine will <u>not</u> trip.

95. Which condition will most likely result in the damage of equipment necessary for safe shutdown of the reactor?

A. CS pump room water level at 4 feet.

B. HPCI pipe routing area temperature 195°F.

C. RHR B loop flow at 7200 gpm to maintain RPV level >TAF with suppression pool level 19'.

D. CRD area radiation level 15R /HR.

96.EO-100-104 Secondary Containment Control and EO-100-102 RPV Control are being executed due to a primary system discharging into the HPCI Room. RPV level never went less than -25", and RPV parameters are within the desired bands.

HPCI Room is verified to be at 33"
Core Spray A Room is at 15" and rising approximately 1" per minute.

How should RPV pressure be controlled?

A. Cool down less than 100°F/hr.

B. Open all Bypass valves.

C. Rapidly Depressurize the RPV.

D. Maintain RPV pressure between 800 psig and 1087 psig.

97.Plant conditions:

ATWS	12 rods not full in
Reactor power	IRMs R4 and lowering
Reactor pressure	800 psig
SLC	not injected
Suppression pool temperature	93°F

Subsequent to establishing these conditions, a steam break with failure to isolate develops in the RWCU pump room. Temperature in the room is 145°F rising slowly. No other EO-100-104 conditions are present. The STA recommends beginning a depressurization to lower RPV pressure and RWCU leak rate.

Which of the following statements is the correct response to the STA's recommendation?

A. Depressurization is not permitted. All rods must be full-in before a cooldown can begin.

B. Depressurization is not permitted. SLC should be initiated immediately to provide Cold Shutdown Boron Weight. A cooldown of <100°F/hr is not commenced until CSBW has been injected

C. Depressurization is permitted. Rising temperatures in the RWCU pump room warrant anticipation of Rapid Depressurization. Open all BPVs and cooldown in excess of 100°F / hr.

D. Depressurization is permitted. Depressurization and cooldown will be at <100°F/hr unless the reactor goes critical again.

98. With Unit 1 in a hydraulic ATWS condition, EO-100-113 is in progress. As part of the actions, the Unit Supervisor has directed implementation of ES-158-002, RPS and ARI Trip Bypass. Performing the actions related to Alternate Rod Insertion (ARI) in this procedure will:

A. prevent any scram signal being generated by the ATWS circuitry from affecting the position of the ARI and backup scram valves.

B. deenergize the ARI scram circuitry and seal-in the ARI trip signal to maintain the ARI valves in their present condition.

C. remove power (if any existed) from the ARI solenoid valves. This will realign the block and vent valves to permit repressurization of the scram air header when RPS is bypassed.

D. permit reset of the ARI logic circuits regardless of trip conditions (high pressure / low level), permitting repressurization of the scram air header when RPS is bypassed.

99. Specific guidelines are provided within the Emergency Operating Procedure EO-100(200)-105, Radioactivity Release Control. Among these is the requirement to " Rapidly Depressurize the RPV BEFORE EPB projected dose / dose rate reaches the General Emergency declaration criteria."

A release in excess of the limits specified as General Emergency declaration criteria would result in exposure of:

A. occupationally exposed personnel within the plume in excess of the Emergency Exposure limits set within 10CFR100 (25 R / 75 R for equipment manipulation and lifesaving actions, respectively).

B. occupationally exposed personnel within the site boundary to levels in excess of their annual NRC permissible whole body and thyroid exposure listed in 10CFR20.

C. members of the general public within or outside of the Site Boundary to radiation levels in excess of Technical Requirements Manual limits on annual total release (500 mr whole body / 3000 mr skin).

D. the general public to radiation levels in excess of those set by 10CFR100.

100. A plant transient has occurred resulting in fuel failure. An unisolable Main Steamline break exists in the Turbine Building. EO-100-102 is being executed. Based upon OSCAR Field data, EO-100-105 Radioactive Release Control is also being executed.

The dose assessment team reports the following DOSE RATES at the EPB:

800 mrem / hr TEDE and 4300 mrem / hr CDE.
No projected dose is yet available.

RPV water level is -45" slowly lowering, and no injection sources are available.

Rapid Depressurization should be delayed until:

A. the limits on EO-100-105 have been reached.

B. an injection source becomes available irrespective of projected off-site dose/dose rates.

C. -205" RPV level is reached irrespective of projected off-site dose/dose rates.

D. off-site dose/dose rates have been exceeded.

Distri35.txt
Distribution Sheet

Priority:'Normal

From: Esperanza Lomosbog

Action Recipients:	Copies:	
R Schaaf	1	Not Found
NRR/DLPM/LPD1-1	1	Not Found
Internal Recipients:		
RGN 1 .FILE 01	1	Not Found
NRR/DIPM/OLHP	1	Not Found
FILE CENTER	1	Not Found
ACRS	1	Not Found
External Recipients:		
NRC PDR	1	Not Found
NOAC	1	Not Found

Total Copies:	8	

Item: ADAMS Package
Library: ML_ADAMS^HQNTAD01
ID: 003671749

Subject:
Operator Licensing Examination Reports

Body:
PDR ADOCK 05000387 V

Docket: 05000387, Notes: N/A

Docket: 05000388, Notes: N/A

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

June 18, 1999

Mr. Robert G. Byram
Senior Vice President - Nuclear
PP&L, INC.
Susquehanna Steam Electric Station
2 North Ninth Street
Allentown, Pennsylvania 18101

SUBJECT: SUSQUEHANNA NUCLEAR GENERATING STATION SENIOR REACTOR OPERATOR INITIAL EXAMINATION REPORT NOS. 50-387/99-301 AND 50-388/99-301

Dear Mr. Byram:

This report transmits the results of the subject operator licensing examinations conducted by the NRC during the period of May 10 - 13, 1999 at your facility. These examinations addressed areas important to public health and safety and were developed and administered using the guidelines of NUREG-1021, Interim Revision 8, "Examination Standards for Power Reactors". Based on the results of the examinations, 3 of 5 Senior Reactor Operators (SRO) applicants passed all portions of the examinations. The findings and performance conclusions as a result of the examinations were discussed by Mr. L Briggs and me with Mr. H. Palmer and others of your staff via telephone conference call on June 3, 1999.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

No reply to this letter is required, but should you have any questions regarding this examination, please contact me at 610-337-5183, or by E-mail at RJC@NRC.GOV.

Sincerely,

Richard J. Conte

Richard J. Conte, Chief
Human Performance and
Emergency Preparedness Branch
Division of Reactor Safety

IE42

Docket Nos. 50-387, 50-388

Enclosure: Initial Examination Report No. 50-387/98-301 AND 50-388/98-301
w/Attachments 1, 2, and 3

003671737

PDR ADOCK

Distribution w/encl and w/Attachments 1 - 3:
DRS Master Exam File
PUBLIC
Nuclear Safety Information Center (NSIC)

Distribution w/encl; w/o Attachments 1 - 3:
Region I Docket Room (with concurrences)
W. Lanning, DRS
L. Briggs, Chief Examiner, DRS
NRC Resident Inspector
DRS OL Facility File
DRS File
H. Miller, RA/J. Wiggins, DRA
C. Cowgill, DRP
D. Florek, DRP
C. O'Daniell, DRP

Distribution w/encl; w/o Attachments 1-3: VIA E-MAIL
J. Shackelford, Acting RI EDO Coordinator
E. Adensam, PDI, NRR
V. Nerses, Project Manager, NRR
Inspection Program Branch, NRR (IPAS)
R. Correia, NRR
E. Brach, NMSS/SFPO
P. Eng, NMSS/SFPO
E. Loeds, NMSS/SFPO
S. Baggett, NMSS/SFPO
S. Shankman, NMSS/SFPO
DOCDESK

DOCUMENT NAME: G:\HPEPB\BRIGGS\SUS99301.WPD
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RI/DRS		RI/DRS	RI/DRS			
NAME	LBriggs		RConte				
DATE	06/?/99		06/ /99	06/ /99		06/ /99	

OFFICIAL RECORD COPY

EXECUTIVE SUMMARY

Susquehanna Steam Electric Station
Examination Report Nos. 50-387/99-301 and 50-388/99-301

Operations

Four instant SRO and one upgrade SRO applicants were administered initial licensing exams. Three applicants successfully passed all portions of the exam. Two applicants did not pass the written examination.

One performance problem exhibited by the applicants was identified concerning their knowledge of the function of the safety/relief valve (SRV) bellows and what affect its failure has concerning subsequent SRV function and operability.

Overall, the as-submitted written examination met the guidance of NUREG 1021. Only two questions required more than minor revision; however, subsequent analysis by the licensee resulted in comments, concerning technical subject matter, for 8 questions (see Attachments 1 and 2).

The JPM set developed by the licensee met the guidance of NUREG 1021.

Scenarios submitted were acceptable with one exception. The third scenario required the addition of an instrument failure that required operator action to prevent plant degradation, to meet the guidance of NUREG 1021.

development of the exam commenced. The NRC subsequently reviewed and validated all portions of the proposed exams. Some changes and/or additions to the proposed exams were requested by the NRC prior to and during the on-site review. Personnel subsequently incorporated the agreed to comments and finalized the exams.

The NRC review of the written examination resulted in the significant revision of only two questions. No questions required replacement. Several changes were also made to question stems to make the question easier to understand and to distractors to make them more plausible.

There were no JPMs replaced as a result of NRC review. There were several changes to JPM questions to enhance the level of difficulty and several open reference questions were changed to closed reference.

Written Test Administration and Performance

The facility training department performed an analysis of questions missed on the written exam for generic and individual weaknesses. There were 16 questions that were missed by more than 50 percent of the applicants. Discussions with the licensee indicate that these questions' subject areas will be discussed with all applicants prior to assumption of any licensed duties. The licensee's action was determined to be acceptable.

As a result of the licensee's analysis, comments concerning answer changes and multiple correct answers were provided to the NRC for evaluation and resolution for 8 or the 16 questions. The licensee's comments are detailed in Attachment 2 of this report. The NRC resolution is discussed in Attachment 1.

Operating Test Administration and Performance

During the preparation week one scenario required the addition of an instrument failure that would require operator actions to prevent plant degradation, as detailed in the examination standards, NUREG 1021, Interim Revision 8.

The applicants demonstrated satisfactory communications and teamwork during the simulator exercises in both the routine and emergency portions of the exercise. Briefings were routinely conducted by the candidates when in the control room supervisor position. The briefings were well controlled and ensured that all personnel knew the plant (simulator) status.

c. Conclusions

Four instant SRO and one upgrade SRO applicants were administered initial licensing exams. Three applicants successfully passed all portions of the exam. Two applicants did not pass the written examination.

Attachments:
1. NRC Resolution of Facility Comments
2. Facility Comments on the Written Examination
3. SRO Written Exam w/Answer Key

PARTIAL LIST OF PERSONS CONTACTED

FACILITY

K. Chambliss	Manager, Nuclear Operations
R. DeVore	Simulator Instructor
A. Fitch	Supervisor, Operations Instruction
M. Jacopetti	Simulator Instructor
H. Palmer	Manager Nuclear Training
J. Radishofski	Operations Supervisor

NRC

L. Briggs	Senior Operations Engineer
J. Caruso	Operations Engineer
R. Vogt-Lowell	Operations Engineer, NRR

Attachment 1

NRC RESOLUTION OF FACILITY COMMENTS

SRO Examination

Licensee comments in Attachment 2

Question 7: Summary of licensee's comment: Correct answer is "b" not "d" as initially indicated, based on testing on the plant reference simulator and the references identified on the review version of the question.

NRC response: Accept licensee's comment. Answer key changed to reflect "b" as the correct answer.

Question 76: Summary of licensee's comment: Correct answer is "a." For the stated conditions, level will not get to -38 inches so recirculation pumps will not trip. Procedure ON-100-109, Section 4.9 tells the operator to ensure Recirculation Pumps are at the minimum speed.

NRC response: Accept licensee's comment. Reviewed applicable portion of referenced procedure. Answer key changed to reflect "a" as the correct answer. The NRC also thinks that under certain accident or fire conditions there could be a loss of recirculation pumps and the cooling medium would be via natural circulation as stated in distractor "d."

The answer key was changed to accept either "a" or "d" as correct.

Question 86: Summary of licensee's comment: Correct answer is "a." The SRV relief setting of about 1200 psi would discharge into the suppression pool with a back pressure of about 75 psi, this would give a starting temperature of about 305F, making "a" the correct answer.

NRC response: Detailed review of the Mollier Diagram and the distractors for this question indicated that about 305F was the correct starting temperature for this question but the second half of the "a" distractor was incorrect because tailpipe temperature following initial lifting of the SRVs will increase until the reactor system pressure blows down to 500 psi then it will decrease with reactor system pressure. This occurs due to the isoenthalpic throttling process of the SRV.

There was no correct answer to question 86 therefore this question is being deleted.

downcomer flow lowers." Answer "a" indicates that the acoustic monitor indicated the SRV opened then lost indication as pressure decreased, which would not be the case as indicated in the licensee's comments. The open indication would never have been received making answer "a" incorrect.

The answer key was not changed for this question.

Question 87: Summary of licensee's comment: Accept answers "c" or "d" as correct. The licensee's comments indicate that if suppression pool temperature exceeds 110F then the plant must be placed in cold shutdown (mode 4) and therefore would be less than 110F prior to entering mode 2.

NRC response: Do not agree with the licensee's statement that the plant must be placed in cold shutdown. This statement is true only if pool temperature remains equal to or greater than 110F and cannot be reduced.

However the NRC does agree that the suppression pool must be less than 110F prior to entering mode 2 even if the plant was not placed in cold shutdown because it went to hot shutdown (mode 3) when the mode switch was placed in shutdown and the suppression pool must be less than 110F prior to the mode switch being placed in startup (mode 2).

The answer key was changed to indicate that "c" and "d" are correct answers.

May 12, 1999

1999 NRC WRITTEN EXAM REVIEW

A preliminary review of the written exam using the submitted answer key indicated 16 questions that should be reviewed. Those questions have been reviewed by two licensed operator training staff members. The following changes should be made to the answer key.

The following corrections should be made to the answer key:

Question # 7 The answer key is corrected to "b". The listed references support this answer. The situation was run on the simulator with the indicated result. The candidates were trained using the simulator.

Question # 76 The answer key is corrected to "a". ON-100-009 Section 4.9.8 has the operator ensure the Reactor Recirculation pumps are in service at minimum speed. Level would never get to -38", so the pumps would not trip.

Question # 86 The answer key is corrected to "a". Since the SRV's lifted on their safety setting and all of them lifted, the highest setting (1205 psi) was used. The SY017 C-4, Automatic Depressurization System unit of instruction gives the pressure sets. A science unit SC023 D-8 on steam throttling indicates a back pressure of about 75 psi would be experienced at the tail pipe elements. Using the Mollier Chart, this gives a starting temperature of about 305 °F, making "a" the correct answer.

The following questions have been determined to have two correct answers:

Question # 53 Answers "a" or "b" should be accepted. The OI-AD-013 section 4.1.5.e states that in a high rad area, the fire watch should step in and step out. This makes "a" correct. However, a high rad area is locked and HP is required to open the lock. So HP is required to go with the fire watch when the tour is performed. This makes "b" a correct answer also. The Site Fire Protection Engineer was called and stated that keys are NOT issued to fire watches which means that HP is required to accompany the fire watch.

Question # 55 Answers "a" or "b" should be accepted. The question is unclear as to whether the situation has been classified. The confusion comes with the use of the word "identify". The procedure uses the words "identify" and "verify". With this interpretation, the conditions are identified and 15 minutes is used to classify the situation based on these conditions. Then 15 minutes is used to notify the authorities. This gives a total of 30 minutes or answer "b". A second interpretation is that the condition being "identified" means that the operator realizes that an emergency condition exists. He has classified it. Only 15 minutes to notify are left. This makes "a" a correct answer.

Question # 63 Answers "b" or "d" should be accepted. Not all protective trips are lost on loss of DC. The mechanical overspeed trip and the pneumatic trips are still functional. Also the shutdown solenoid deenergizes and causes the diesel to trip on loss of DC. This was verified by running the transient in the simulator. The loss of control power will prevent the breaker from operating making "b" correct. Immediately after the loss of DC control power, the change suggested by "d" should be done so that the diesel can be controlled. ON-102-640 directs this.

Question # 82 Answers "a" or "d". The unit of instruction SY017 C-4, Automatic Depressurization System, indicates that flow will be too low to register on the acoustic monitors at this low pressure. This makes "a" correct. The suppression pool static head may

Attachment 3

SRO WRITTEN EXAM W/ANSWER KEY

PART B - WRITTEN EXAMINATION GUIDELINES

1. *[Read Verbatim]* After you complete the examination, 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.

2. To pass the examination, you must achieve a grade of 80.00 percent or greater. Every question is worth one point.

3. For an initial examination, the time limit for completing the examination is four hours.

   For a requalification examination, the time limit for completing both sections of the examination is three hours. If both sections are administered in the simulator during a single three-hour period, you may return to a section of the examination that was already completed or retain both sections of the examination until the allotted time has expired.

4. You may bring pens and calculators into the examination room. Use only black ink to ensure legible copies.

5. Print your name in the blank provided on the examination cover sheet and the answer sheet. You may be asked to provide the examiner with some form of positive identification.

6. Mark your answers on the answer sheet provided and do not leave any question blank. Use only the paper provided and do not write on the back side of the pages. If you decide to change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

7. If the intent of a question is unclear, ask questions of the NRC examiner or the designated facility instructor only.

8. Restroom trips are permitted, but only one applicant at a time will be allowed to leave. Avoid all contact with anyone outside the examination room to eliminate even the appearance or possibility of cheating.

9. When you complete the examination, assemble a package including the examination questions, examination aids, answer sheets, and scrap paper and give it to the NRC examiner or proctor. Remember to sign the statement on the examination cover sheet indicating that the work is your own and that you have neither given nor received assistance in completing the examination. The scrap paper will be disposed of immediately after the examination.

Senior Reactor Operator Answer Sheets

1. ___
2. ___
3. ___
4. ___
5. ___
6. ___
7. ___
8. ___
9. ___
10. ___
11. ___
12. ___
13. ___
14. ___
15. ___
16. ___
17. ___
18. ___
19. ___
20. ___
21. ___
22. ___
23. ___
24. ___
25. ___
26. ___
27. ___
28. ___
29. ___
30. ___
31. ___
32. ___
33. ___
34. ___
35. ___
36. ___
37. ___
38. ___
39. ___
40. ___
41. ___
42. ___
43. ___
44. ___
45. ___
46. ___
47. ___
48. ___
49. ___
50. ___

1. Given the following conditions:

   - Unit 1 was operating at 100% power
   - Following a valid reactor scram signal the Reactor Protection System was unable to de-energize the 185 individual Scram Pilot Valves
   - The Backup Scram Valves did function as designed and all control rods fully inserted

   Which of the following would be an indication that the Backup Scram Valves actually accomplished the scram?

   a. No hydraulic control unit accumulator fault alarms would be received on the full core display.

   b. The total elapsed time from the scram signal to all control rods fully inserted would be noticeably longer.

   c. The Scram Discharge Volume Vent and Drain Valves would not reposition.

   d. The individual control rod scram speeds would be slower.

2. Given the following CURRENT full core display parameters for control rod 22-35 that had been at Notch "48".

   - Full-In: Illuminated
   - Full-Out: NOT Illuminated
   - Drifting: Illuminated
   - Selected: NOT Illuminated
   - Accumulator: NOT Illuminated
   - Scram Valves: NOT Illuminated

   These conditions are the result of:

   a. the Scram Inlet Valve (126) opening.

   b. the Scram Outlet Valve (127) opening.

   c. the Scram Inlet Valve (126) and Scram Outlet Valve (127) both opening.

   d. the control rod being driven to Notch "00" using the "Insert Rod" pushbutton.

5. Given the following conditions:

- Unit 1 was operating at 80% power
- A logic failure has resulted in the "B" Recirculation Pump running back to the #2 Limiter
- Actual #2 Limiter Runback conditions do NOT exist

Which of the following describes the plant limitations required while operating under these conditions?

a. If the "B" Recirculation Pump runback cannot be reset in 2 hours it must be tripped within the next 12 hours.

b. Single loop operating restrictions and limitations must be in place within 2 hours.

c. The "B" Recirculation Pump runback must be reset and speed raised or the "A" Pump speed reduced to 45% within 2 hours.

d. Mismatched loop flow operation is not permitted and immediate action must be taken to be in Mode 3 within 12 hours.

6. Given the following conditions:

- Unit 2 is making preparations for a reactor and plant startup
- Reactor temperature is 120 degrees F
- The reactor is at atmospheric pressure
- The Condensate and Feedwater Systems are in Long Path Recirculation
- Reactor Water Cleanup is operating for reactor water level control

Following the start of the Reactor Recirculation Pumps, what is the MAXIMUM speed at which they can be operated for these conditions?

a. 20%

b. 30%

c. 40%

d. 45%

9. Given the following conditions:

- Unit 2 is in Mode 4 with Shutdown Cooling in service on the "B" Residual Heat Removal (RHR) loop
- A large leak has developed just upstream of the Shutdown Cooling Suction Outboard Isolation Valve (F008)
- Reactor water level rapidly reaches the Low Pressure Coolant Injection (LPCI) initiation setpoint
- All expected actions occur
- Core Spray is NOT available

Which of the following describes the expected affect on the leak and reactor water level for these conditions?

a. The leak will be stopped and reactor water level will stabilize but not recover unless operator action is taken to inject.

b. The leak will NOT be stopped. Operator action is required to isolate the leak and inject with RHR to recover level.

c. The leak will be stopped and reactor water level will rise due to the "B" Loop of RHR injecting in the LPCI mode.

d. The leak will NOT be stopped. Operator action is required to isolate the leak allowing automatic LPCI injection to recover level.

10. Given the following conditions:

- The Unit 1 High Pressure Coolant Injection (HPCI) system is running in the CST to CST mode
- The Flow Controller is in "Automatic" set for 3500 gpm
- System flowrate is 3500 gpm
- The Extra PCO reports that HPCI turbine speed is lowering

Which of the following would cause this response?

a. A relay failure has just transferred the Flow Controller from "Automatic" to "Manual"

b. The HPCI Test Line To CST Isolation Valve (F011) has just auto closed.

c. The HPCI ramp generator output just failed to its "low" limit.

d. The HPCI Minimum Flow To Suppression Pool Valve (F012) has just opened.

4. Given the following conditions:

- Unit 1 is operating at 100% power
- The "B" Standby Liquid Control (SLC) Pump was declared "Inoperable" 4 days ago
- The "Loss Of Continuity To Squib Valves" alarm has just been received
- Investigation reveals broken leads to the "A" SLC Squib Valve primers
- The "B" SLC Squib Valve primer continuity status has not changed

Select the required actions for these conditions.

a. Restore one subsystem to Operable status in 8 hours or be in Mode 3 within the next 12 hours.

b. Continue in the 7 day Required Action for one Inoperable subsystem, no further actions are required.

c. Enter a 7 day Required Action for the "A" SLC Subsystem, continue in the 7 day Required Action for the "B" SLC Subsystem.

d. Extend the current 7 day Required Action for one Inoperable subsystem not to exceed 10 days from the initial failure to meet the LCO.

5. Given the following conditions:

- Unit 2 is operating at 60% power
- A valid reactor scram signal occurs on high drywell pressure

Which of the following failures would PREVENT the Backup Scram Valves from venting the scram air header?

a. The solenoid on the upstream Backup Scram Valve (110B) does not de-energize.

b. The Alternate Rod Injection Scram Air Header Block Valves (SV 14799 & 147100) did not close on the scram.

c. Only one Reactor Protection System Trip System de-energized on the scram signal.

d. The check valve (111) bypassing the downstream Backup Scram Valve (110A) does not open.

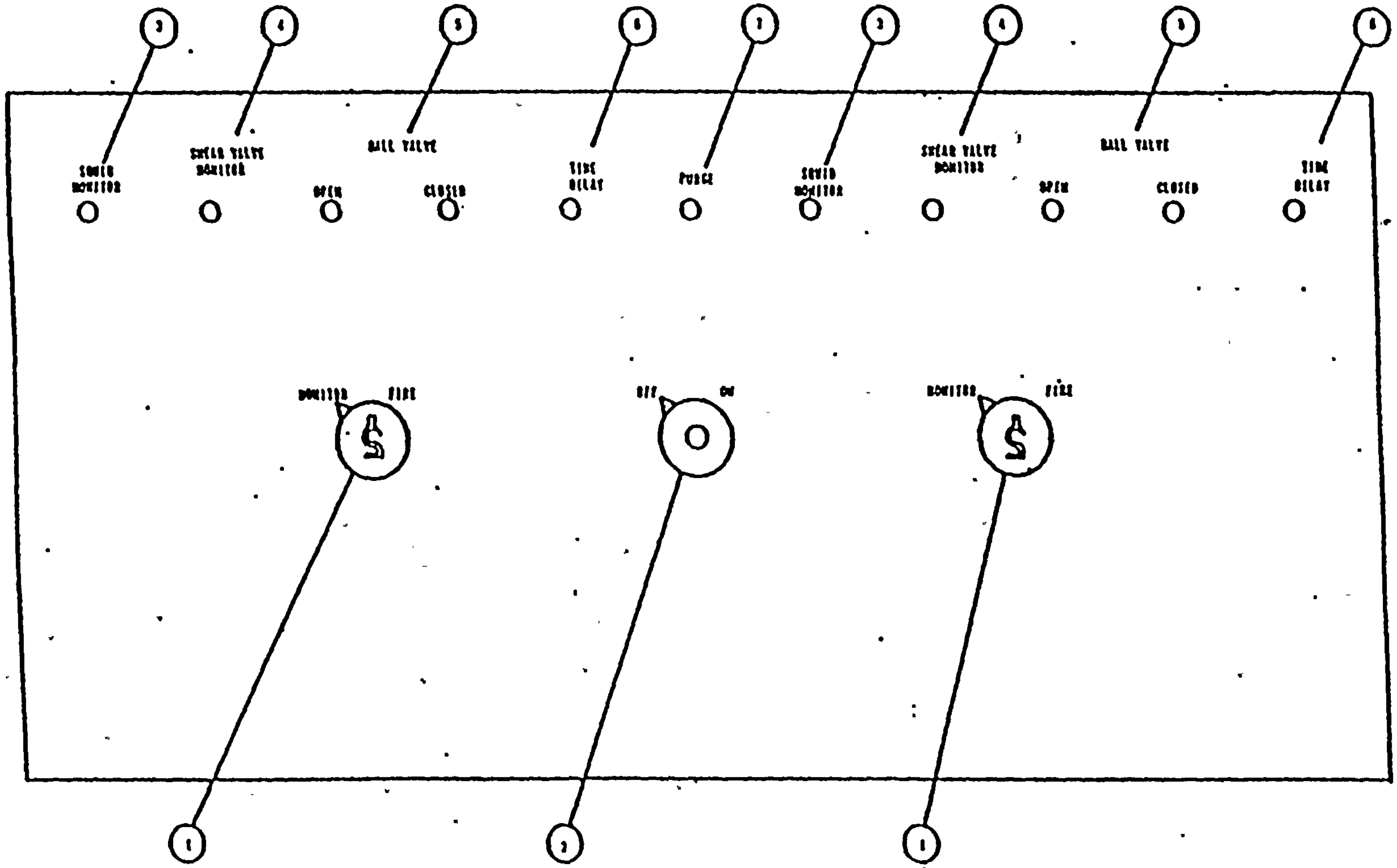

FIGURE 14
DUAL VALVE CONTROL MONITOR

20. Given the following conditions:

- Unit 1 is ready to load fuel following a complete core offload (no fuel assemblies are in the core)
- Source Range Monitor (SRM) channels "A" and "C" are Inoperable
- SRM detectors "B" and "D" are fully inserted with count rates of 1.2 cps and 2 cps respectively
- The Signal-To-Noise Ratio for SRM "B" and "D" is 3:1 for both channels

Which of the following describes what is required to commence core reload under these conditions?

a. One of the Operable SRMs must be in the quadrant where core alterations are being performed and the other in an adjacent quadrant. Minimum count rate is not required to be met until 4 fuel assemblies have been installed adjacent to the SRM.

b. SRM "A" or "C" must be returned to Operability to allow core alterations to begin. Minimum count rate is not required to be met until 4 fuel assemblies have been installed adjacent to the SRM.

c. One of the Operable SRMs must be in the quadrant where core alterations are being performed and the other in an adjacent quadrant. A neutron source must be installed adjacent to the SRMs to achieve a minimum 3 cps.

d. SRM "A" or "C" must be returned to Operability and a neutron source must be installed adjacent to the SRMs to achieve a minimum 3 cps to allow core alterations to begin.

21. Given the following conditions:

- Unit 1 is operating at 100% power
- Average Power Range Monitor (APRM) Channel "C" has been bypassed with the joystick for maintenance

A Gain Adjustment for APRM "C" will be required:

a. prior to taking it out of "Bypass".

b. if it differs by more than 2% from the average of the remaining 5 APRM channels.

c. prior to exceeding a gain adjustment factor (AGAF) of 1.00.

d. if its gain adjustment factor (AGAF) is less than 0.98.

24. Given the following conditions:

- Unit 2 has experienced a loss of off-site power
- The Reactor Core Isolation Cooling (RCIC) system automatically initiated as designed
- The Extra PCO has placed the RCIC Flow Controller in "Manual" to control flow at 350 gpm to maintain reactor water level
- While in these conditions a failure of the shaft driven lube oil pump results in a total loss of oil pressure (reading 0 psig)

Which of the following describes the expected response of RCIC?

a. RCIC will immediately trip on low lube oil pressure.

b. RCIC will decelerate as the governor valve strokes closed.

c. RCIC speed will remain constant until turbine bearing damage begins.

d. RCIC will accelerate and trip on overspeed.

25. Given the following conditions:

- The Automatic Depressurization System (ADS) Manual Initiation pushbuttons "A" and "C" (HS30A and HS30C) have been armed and pressed
- There is no response from the ADS safety relief valves

Which of the following electrical bus failures caused this system response?

a. A loss of 125 VDC Bus 1D624

b. A loss of 250 VDC Bus 1D652.

c. A loss of 125 VDC Bus 1D614.

d. A loss of 250 VDC Bus 1D662.

28. Given the following conditions:

- Unit 2 is shutdown with core alterations in progress
- While a fuel bundle is being raised out of the core the "Normal Up" light illuminates and the fuel hoist stops
- The Fuel Grapple position indicator (Z) reads 20
- The expected "Normal Up" position should be 16

Which of the following describes the use of the "Hoist Override" pushbutton for these conditions?

a. Hoist Override may be used to raise the grapple only to the "Normal Up" position of 16 with Refueling SRO explicit permission.

b. With the refueling bridge over the core, the Hoist Override pushbutton is bypassed and is unavailable for use.

c. With irradiated fuel on the hoist use of the Hoist Override pushbutton is procedurally prohibited.

d. Hoist Override may be used for raising the grapple one "Z" direction increment at a time if a second licensed operator is available for concurrent position verification.

31. Which of the following describes how the main turbine is protected from overspeed conditions if the generator output breaker trips open at 30% power? (See attached figure.)

The Electro-Hydraulic Control (EHC) system:

a. load reject circuit will initiate a fast closure of the Intercept Valves.

b. power/load unbalance circuit will initiate a fast closure of the Turbine Control Valves.

c. load reject circuit will throttle the Intercept Valves closed.

d. power/load unbalance circuit will throttle the Turbine Control Valves closed.

32. During a Unit 1 startup and heatup in accordance with GO-100-002, "Plant Startup, Heatup And Power Operation", the operator is directed to maintain turbine first stage pressure less than 120 psig during shell warming.

Which of the following would be expected to occur if this value is exceeded?

a. Main turbine Exhaust Hood Spray initiates.

b. Reactor scram.

c. Main turbine overspeed trip

d. Main Steam Isolation Valve closure.

33. Given the following conditions:

- Unit 1 experienced a reactor scram from 95% power
- Reactor water level reached 0 inches
- Feedwater level control remained in "Automatic" and reactor water level currently at +5 inches and is rising
- The Unit PCO has pressed the "Level Setpoint Setdown" pushbutton (HS-C32-1S08)
- All plant systems responded as designed

Reactor water level will:

a. return to +35 inches.

b. stabilize at +5 inches.

c. rise to +13 inches

d. stabilize at +18 inches

34. Given the following conditions:

- Both Units are operating at 100% power
- The Standby Gas Treatment System (SGTS) is in a normal, standby lineup
- A valid Unit 1 SGTS system initiation signal on high drywell pressure is received

Which of the following is the location from which SGTS will AUTOMATICALLY take suction for these conditions?

SGTS will begin to process the:

a. High Pressure Coolant Injection Barometric Condenser Vacuum Pump discharge.

b. drywell atmosphere via the drywell and suppression chamber purge connections.

c. discharge flow from the running Reactor Building Ventilation Exhaust Fans.

d. drywell atmosphere via the drywell vent connections.

35. Given the following conditions:

- Both Units are operating at 100% power
- All Startup Bus power sources are available
- All four Diesel Generators are available
- The Normal Source Breaker (1A201-01) to ESS Bus 1A is opened with its handswitch on Panel 0C653
- No other operator actions were taken

Which of the following describes what must occur to reenergize the ESS 1A Bus assuming all systems operate as designed?

a. The "A" Diesel Generator will start and the Emergency Source Breaker (1A201-04) will automatically close.

b. The Alternate Source Breaker (1A201-09) will automatically close.

c. The PCO will have to start the "A" Diesel Generator and close the Emergency Source Breaker (1A201-04).

d. The PCO will have to close the Alternate Source Breaker (1A201-09).

38. Given the following conditions:

- Unit 2 is operating at 55% power with the "C" Main Steam Line (MSL) isolated (Inboard and Outboard MSIVs are closed)
- A fuel failure results in rising main steam line radiation levels

Which of the following describes the Main Steam Line Radiation Monitor automatic MSIV closure functions under these conditions?

a. With the "C" MSL MSIVs closed, the "C" MSL Rad Monitor signal is removed from the circuitry and the isolation logic is modified to a one-out-of-three to close the remaining 6 MSIVs.

b. The physical location of the 4 MSL Rad Monitors allows each of them to "see" all four steam lines providing for a normal MSIV closure based upon rad levels in the operating steam lines.

c. With the "C" MSL MSIVs closed, the "C" MSL Rad Monitor will have a "downscale" signal present providing one of the two required "trips" for the isolation.

d. The physical location of the 4 MSL Rad Monitors upstream of the Inboard MSIV provides for continued monitoring of the "C" MSL even though it is isolated.

39. Given the following conditions:

- Unit 1 is operating at 90% power
- The functional test of the Standby Gas Treatment (SGTS) Fire Suppression System has just been performed
- The test results were UNSATISFACTORY for both SGTS trains

How does this failure impact continued plant operation?

a. Both trains of SGTS are Inoperable. If one cannot be restored to Operable status in 4 hours, the plant must be in Mode 3 within 12 hours.

b. The secondary containment is Inoperable. If one SGTS train cannot be restored to Operable status in 4 hours, the plant must be in Mode 3 within 12 hours.

c. Both trains of SGTS remain Operable with no restrictions on plant operations.

d. A continuous fire watch is required in the area when running either SGTS train.

43. Given the following information for a Unit 1 Technical Specification System:

- This System is Inoperable and must be restored to Operable status within 7 days
- If not returned to Operable status within 7 days, this System is required to:
  - Be in Mode 3 in 12 hours
    AND
  - Be in Mode 4 in 36 hours
- The 7 day return to Operable status requirement expired at 0300 on May 12th
- Unit 1 reached Mode 3 at 0900 May 12th

When is Unit 1 required to be in Mode 4?

a. 0300 May 13th

b. 1500 May 13th

c. 2100 May 13th

d. 0300 May 14th

44. Prior to placing the Reactor Mode Switch to "Startup/Hot Standby" during a reactor startup, the Shift Supervisor shall notify and obtain approval from the:

a. Supervisor - Reactor Engineering.

b. Operations Supervisor - Nuclear.

c. Manager - Nuclear Operations.

d. General Manager - Susquehanna.

45. During a Unit 1 evolution, a procedure must be removed from its Controlled Manual. The Operations Department Clerk is not available to provide the User Controlled copy required. The copy of the procedure was made at 1300 on May 11, 1999.

Which of the following is the maximum expiration date and time allowed for this procedure WITHOUT requiring User Controlled tracking from the Document Control Center?

a. 1900, May 11, 1999

b. 0700, May 12, 1999

c. 1300, May 12, 1999

d. 1300, May 13, 1999

48. Given the following information:

- Plant Systems "A" and "B" are required to support the operation of System "C"
- The completion times for restoration of these systems to Operable status are:
  - System "A" - 7 days
  - System "B" - 14 days
  - System "C" - 3 days
- System "A" became Inoperable 4 days ago at 0800
- System "B" became Inoperable today at 0800
- System "A" was restored to Operable status today at 1200

Assuming the "Maximum Out Of Service Time" criteria, when must System "B" be restored to Operable status?

At 0800:

a. 6 days from today.

b. 10 days from today.

c. 14 days from today.

d. 17 days from today.

49. Which of the following individuals has the procedural responsibility to authorize entry into a Unit 2 area with a dose rate of 750 mr/hour for investigation of abnormal equipment vibration WITHOUT a job specific RWP or specific work plan?

a. The on-shift Health Physics Foreman.

b. The Unit Supervisor.

c. The Radiological Operations Supervisor.

d. The Shift Supervisor

52. Given the following conditions:

    - The Unit 1 Extra PCO is writing a blocking permit for a plant component in an area with a 6.5 rem/hour dose rate requiring a Health Physics escort for entry
    - The component is already in the required position and has remote indication

    Which of the following describes how the component blocking is accomplished for these conditions?

    The Unit Supervisor will:

    a. approve red tag installation directly on the component by Operations Personnel and waive the Independent Verification requirement.
    b. direct Health Physics to hold the red tag and provide it as part of the Radiation Work Permit briefing to all personnel subsequently entering the area.
    c. direct the red tag be installed on the knob or handle of the door to the area where the component is located.
    d. approve the permit without the need for red tag installation if the Operations Lock is accounted for on the permit.

53. Due to Simplex Fire Protection sensor failure, an hourly firewatch is required in a High Radiation Area.

    Which of the following describes the restrictions on these firewatch tours?

    The Firewatch individual:

    a. shall step into the area, make an observation and exit the area.
    b. must be escorted by a Health Physics Technician.
    c. shall perform a normal walkthrough inspection of the area if total dose expected to be received is less than 10 mrem.
    d. may only be a Health Physics Technician.

56. Given the following conditions:

- During a transient Unit 2 momentarily met the conditions requiring a Site Area Emergency
- Prior to the actual classification being made, conditions continued to change such that an Alert is now the appropriate classification

What is the guidance for the classification of this event?

The event shall be classified as:

a. a Site Area Emergency and then downgraded to an Alert after the initial Emergency Notification Report has been acknowledged by the NRC, state and local agencies.

b. an Alert, but should make note of the momentarily Site Area Emergency conditions on the Emergency Notification Report.

c. a Site Area Emergency, make the appropriate notifications and then downgrade the classification to an Alert as soon as possible with management concurrence.

d. an Alert, but should consider upgrading to the Site Area Emergency once all emergency response facilities are activated.

57. Given the following conditions:

- Unit 1 is operating at 100% power
- An Electro-Hydraulic Control (EHC) malfunction has resulted in rapidly rising reactor pressure
- Reactor pressure has reached 1100 psig

What are the EXPECTED Unit PCO actions for these conditions?

a. Initiate a manual reactor scram and inform the Unit Supervisor of the condition and the action taken.

b. Immediately lower the setpoint of the Maximum Combined Flow Limiter to reduce reactor pressure.

c. Inform the Unit Supervisor of the condition and initiate a manual reactor scram when directed.

d. Immediately reduce reactor pressure by placing the Turbine Bypass Valves in "Test" and opening them.

61. Given the following conditions:

- A Station Blackout (SBO) has occurred
- Unit 1 Reactor water level control is via Reactor Core Isolation Cooling (RCIC)
- The Condensate Storage Tank is NOT available due to a tank rupture

Which of the following describes how RCIC operation for level control is accomplished with steadily rising suppression pool temperatures (and subsequent rising lube oil temperatures) during the SBO?

a. RCIC suction is lined up to the Refueling Water Storage Tank (RWST) for a source of cool water.

b. RCIC lube oil cooling water is supplied from the Fire Protection System.

c. All RCIC protective features and trips are bypassed by EO-100-033, "RCIC Operating Guidelines During Station Blackout".

d. RCIC will be run only as necessary to maintain reactor water level +13 to +54".

62. Given the following conditions:

- Unit 2 is performing a startup with the Reactor Mode Switch in "Startup/ Hot Standby"
- Main condenser vacuum has been established
- The Outboard Main Steam Isolation Valves have just been opened and steam line warming is in progress
- The "B" Reactor Protection System MG set has just tripped
- The alternate power supply is not available

How will this bus loss affect the plant assuming it is NOT restored as directed by ON-158-001, "Loss Of RPS"?

a. Main condenser vacuum will begin to degrade.

b. The Recirculation Pumps will immediately trip.

c. The Scram Discharge Volume will begin filling.

d. The Outboard Main Steam Isolation Valves will begin to drift closed.

66. Given the following conditions:

- Unit 1 has been scrammed
- A large coolant leak into the drywell is occurring
- In anticipation of rapid depressurization, all Bypass Valves have been opened
- Reactor pressure has been reduced to 175 psig
- Conditions worsen requiring entry into EO-100-112, "Rapid Depressurization"

Select the required actions for these conditions.

a. Open the 6 ADS valves and close the Bypass Valves.

b. Close the Bypass Valves and open the 6 ADS valves

c. Open the 6 ADS valves and leave the Bypass Valves open.

d. Complete the depressurization using only the Bypass Valves.

67. Given the following conditions:

- Unit 1 is operating at 100% power
- A loss of coolant accident occurs
- Reactor water level is -50 inches
- Drywell pressure is 2.4 psig
- All plant systems respond as designed

Using the attached Reactor Water Cleanup System diagram, determine the valves that REQUIRE operator action to be closed for completion of the system isolation for these plant conditions.

a. HV-144-F001 and HV-144-F004

b. HV-144-F042 and HV-144-F104

c. HV-14182A and HV-14182B

d. HV-144-F100 and HV-144-F106

68. Given the following conditions:

- Unit 1 has performed a manual reactor scram as directed by ON-100-101, "Scram"
- The reason for scramming was a trip of both Recirculation Pumps
- The Control Rod Drive Flow Controller has been lowered to "Minimum" as directed by the ON
- The delta T between the reactor bottom head drain and the steam dome is 156 degrees F

For these conditions the operator is required to:

a. establish natural circulation flow.

b. cooldown to Mode 4.

c. start at least one Recirc Pump.

d. ensure natural circulation flow will not occur.

69. A small Recirculation loop leak exists on Unit 2.

Which of the following describes the expected pressure relationship between the drywell and suppression chamber as the leak continues and pressure rises?

Drywell pressure:

a. will rise to about 4.5 to 5.0 psi above suppression chamber, the two pressures will equalize and remain nearly equal as the leak continues.

b. will rise to about 0.5 to 1.0 psi above suppression chamber and that differential will be maintained as the leak continues.

c. and suppression chamber pressure will initially equalize and maintain that equalization as the leak continues.

d. will rise to about 4.5 to 5.0 psi above suppression chamber and that differential will be maintained as the leak continues.

72. Given the following conditions AFTER a transient from 90% power on Unit 1:

- Reactor power (MWt) is slightly higher
- Generator megawatts (MWe) are slightly lower
- Indicated feedwater flow is greater than indicated steam flow (matched before the transient)
- Reactor water level is slightly lower

These conditions are being caused by:

a. isolation of extraction steam to one feedwater heater.

b. a stuck open Safety Relief Valve.

c. rising main condenser backpressure (degrading vacuum).

d. failure of the on-service EHC pressure regulator to a lower output.

73. Given the following conditions:

- Unit 1 has experienced a Main Steam Isolation Valve closure from 100% power
- The control rods did not insert
- EO-100-113, "Level/Power Control", has been entered
- The Safety Relief Valves have been manually opened to control pressure less than 965 psig
- Standby Liquid Control is not available

For these conditions, the Heat Capacity Temperature Limit:

a. will steadily become more restrictive.

b. will remain constant.

c. will steadily become less restrictive.

d. has been exceeded.

77. Given the following conditions:

- A Unit 1 fire has resulted in the closure of all Outboard Main Steam Isolation Valves from 100% power
- High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) both automatically initiated and are injecting
- The Immediate Operator Actions of ON-100-009, "Control Room Evacuation" were completed
- All Remote Shutdown Panel (RSP) Control Transfer Switches have been placed in "Emergency"
- The RSP operator trips RCIC when reactor water level reaches +54 inches

Reactor water level will:

a. lower until RCIC automatically re-initiates at -30 inches.

b. lower until HPCI automatically re-initiates at -38 inches.

c. lower until both HPCI and RCIC automatically re-initiate.

d. continue to rise due to HPCI injection.

78. Entry into EO-100-105, "Radioactivity Release Control", and completion of the required actions will limit the activity release from:

a. the reactor coolant into the primary containment.

b. the reactor coolant into areas outside the primary and secondary containment.

c. damaged fuel directly into the reactor coolant and plant primary systems.

d. the reactor coolant into the secondary containment.

92. Given the following conditions:

- Unit 2 has experienced a loss of Shutdown Cooling while in Mode 3
- Primary and Secondary Containment are established
- The plant has been shutdown for 36 hours
- Reactor water level is stable at +48 inches
- Reactor pressure is being maintained 20 to 98 psig by opening Non-ADS Safety Relief Valves (SRV) as needed
- Following opening of the "B" SRV, the Extra PCO is unable to close the valve

Which of the following describes the effect of this failure to close as the reactor depressurizes?

a. SRV "Open" position indication from the Acoustic Monitor will be lost as discharge downcomer flow lowers.

b. The reactor will reach saturation temperature with a subsequent reduction in the "time-to-boiling" value.

c. Adequate core cooling will not be maintained for these conditions.

d. The SRV discharge downcomer may begin to reflood with suppression pool water.

3. Given the following conditions:

- Unit 1 is performing a reactor startup
- Reactor pressure is 825 psig
- The Reactor Mode Switch is in "Startup/Hot Standby"
- Control Rods 30-15 and 46-47 (both at Notch "00") have accumulator alarms in on low pressure and are being recharged
- The "A" Control Rod Drive Pump is not available
- The "B" Control Rod Drive Pump has just tripped and cannot be restarted
- Charging header pressure has equalized with reactor pressure

Which of the following describes the plant conditions requiring the Reactor Mode Switch be placed in "Shutdown"?

a. Charging header pressure cannot be raised to or above 940 psig within 20 minutes.

b. An accumulator alarm is received on control rod 46-43 at Notch "00".

c. An accumulator alarm is received on a currently withdrawn control rod.

d. Control rods 30-15 and 46-47 cannot be returned to Operable status within 20 minutes.

86. Given the following conditions:

- Unit 1 was operating at 100% power
- A severe overpressure transient has resulted in the Safety Relief Valves (SRV) opening in their "Safety Valve" mode
- All valves, with the exception of one, have reseated (closed)
- The required actions of ON-183-001, "Stuck Open Safety Relief Valve" have been completed
- The reactor has been scrammed
- The SRV has NOT closed

#86 deleted. No correct ans. ΥΥΘ

As the reactor cools down and depressurizes through the stuck open SRV tail pipe temperature will:

a. start at 305 degrees F and will slowly fall following reactor pressure during the depressurization.

b. start at 270 degrees F, rise to approximately 300 degrees F and then will slowly fall following reactor pressure during the depressurization below 500 psig

c. start at 525 degrees F and will slowly fall following reactor pressure during the depressurization.

d. start at 285 degrees F, rise to approximately 325 degrees F and then will slowly fall following reactor pressure during the depressurization below 500 psig

87. The Unit 1 Reactor Mode Switch was placed in "Shutdown" due to suppression pool temperature being greater than the Technical Specification limit.

Suppression pool temperature must be at or below:

a. 110 degrees F for 36 hours prior to entering Mode 3.

b. 90 degrees F within 24 hours of placing the Reactor Mode Switch in "Shutdown".

c. 110 degrees F prior to entering Mode 2 on the ensuing startup.

d. 90 degrees F prior to reaching the point of adding heat on the ensuing startup.

90. Which of the following describes how the operator determines if water level in the containment is above the top of active fuel while flooding the primary containment?

Top of active fuel is determined by:

a. indicated drywell pressure versus containment level correlation if the drywell is vented to atmosphere.

b. a pressure and temperature corrected reading from Wide Range Suppression Pool Level indication.

c. a level calculated from the pressure differential between the drywell and the suppression chamber.

d. direct reading from the reactor water level Fuel Zone Level indicator if the drywell is vented to atmosphere.

91. Given the following conditions on Unit 1:

- A failure-to-scram (ATWS) condition exists
- Reactor power is 22%
- Standby Liquid Control is injecting
- The Scram Discharge Volume did NOT isolate
- Suppression pool level is 15 feet and lowering
- A greater than Max Safe Water Level exists in two (2) Reactor Building areas

Which of the following are the appropriate actions for these conditions?

a. Immediately open 6 ADS Safety Relief Valves.

b. Take no action until power is less than 5% or all rods are inserted.

c. Immediately open the Turbine Bypass Valves.

d. Take no action until suppression pool reaches 12 feet.

94. While operating in EO-100-104/EO-200-104, "Secondary Containment Control", the Max Safe Temperatures for the HPCI Equipment Areas are different between Unit 1 (300 degrees F) and Unit 2 (240 degrees F).

Which of the following describes the reason for this difference and how that difference will affect operation in Secondary Containment Control?

a. The Unit 2 HPCI Room room coolers are arranged differently and can be provided with cooling from both DX Units. This additional cooling capacity allows lower EO-200-104 temperature limits.

b. The Unit 2 safe shutdown analysis for HPCI equipment operability concerns during loss of off-site power was more restrictive than that done on Unit 1. Thus, EO-200-104 requires action earlier than EO-100-104.

c. On Unit 2, temperature instrumentation location for RCIC and HPCI is such that the rooms are considered one "area" for EO-200-104 purposes. Therefore, the more restrictive RCIC Max Safe Temperature is limiting.

d. Post loss of off-site power natural ventilation flow has more heat removal capabilities in the Unit 2 Reactor Building as opposed to Unit 1. Additional equipment operability analysis allows a higher temperature in EO-100-104.

95. With Unit 1 at power an EO-100-104, "Secondary Containment Control", entry condition has been received.

Which of the following EO-100-104 directed actions will NOT reduce any current and future Off-Site doses for these conditions?

a. "Go to RPV Control" - Step SC/R-5

b. "Rapid Depress is required" - Step SC/R-6

c. "Restart RB HVAC" - Step SC-3

d. "Isolate all systems discharging into area" - Step SC/R-1

98. While operating in accordance with EO-100-113, "Level/Power Control", the operator is directed to lower level to between -60 and -110 inches (Step LQ/L-6) utilizing Table 15 systems.

Which of the following describes why the Low Pressure Coolant Injection (LPCI) mode of Residual Heat Removal (RHR) system is the LEAST preferred Table 15 system for accomplishing this step?

a. Utilizing the other Table 15 systems first maintains RHR available for containment and/or suppression pool problems during the ATWS.

b. The LPCI injection flowpath receives minimal preheating and its use may result in power/flow instabilities as level is lowered.

c. The relatively low RHR Pump shutoff head limits the systems' ability to inject during an high power/pressure ATWS.

d. The high RHR Pump flow rates may result in sweeping any injected boron out of the core resulting in a power rise as level is lowered.

99. EO-100-105, "Radioactivity Release Control", directs isolation of all primary systems discharging into areas outside Primary Containment or Reactor Building except those systems required to support EOP/DSP actions.

These systems are specifically exempted from isolation because:

a. additional off-site releases from them are unlikely.

b. they are required to support alternate reactor depressurization methods.

c. their isolation may result in larger, uncontrolled releases as the transient continues.

d. these additional isolations would require an unnecessarily escalation of the emergency classification.

100. Unit 1 is operating in accordance with ON-113-001, "Response To Fire".

Select the specific conditions that direct the Unit Supervisor to EXIT ON-113-001 even with a fire still burning.

a. The fire is affecting Unit equipment required to reach and maintain "safe shutdown".

b. The Fire Brigade Leader has determined that off-site fire fighting assistance is required.

c. Any Emergency Operating Procedure entry condition is met.

d. ON-100-009, "Control Room Evacuation", entry is required.

51. b
52. c
53. a OR b. [illegible]
54. a
55. b OR a. [illegible]
56. c
57. a
58. c
59. a
60. d
61. b
62. a
63. ~~c~~ b. [illegible]
64. d
65. a
66. c
67. c
68. d
69. d
70. d
71. b
72. b
73. a
74. c
75. b
76. d OR a. [illegible]
77. b
78. b
79. c
80. a
81. d
82. d
83. c
84. d
85. c
~~86. d~~ deleted [illegible]
87. d OR C [illegible]
88. d
89. b
90. a
91. a
92. b
93. a
94. c
95. c
96. a
97. d
98. b
99. c
00. d