ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-528/95-15
50-529/95-15
50-530/95-15

Licenses: NPF-41
NPF-51
NPF-74

Licensee: Arizona Public Service Company
P.O. Box 53999
Phoenix, Arizona

Facility Name: Palo Verde Nuclear Generating Station, Units 1, 2, and 3

Inspection At: Maricopa County, Arizona

Inspection Conducted: October 11-27, 1995

Inspectors: Ryan E. Lantz, Chief Examiner, Operations Branch
Division of Reactor Safety

David B. Pereira, Inspector, Plant Support Branch
Division of Reactor Safety

James Moorman, Examiner, Operator Licensing Branch
Division of Reactor Safety, Region II

Approved: _____ _____
Joseph Tapia, Acting Chief, Operations Branch Date
Division of Reactor Safety

Inspection Summary

Areas Inspected (Units 1, 2, and 3): Routine, announced inspection of the qualifications of applicants for senior operator licenses at the Palo Verde Nuclear Generating Station, which included an eligibility determination and administration of comprehensive written examinations and operating tests. The examination team also observed the performance of on-shift operators and plant conditions incident to the conduct of the applicant evaluations. The facility volunteered to participate in a pilot initial examination process which principally involved facility development and NRC administration of the initial examinations. Guidance for conduct of the pilot examinations was

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contained in Generic Letter 95-06 and Attachment 1 to Regional Office Interaction Memorandum 9525, which was used in addition to the guidance provided in NUREG-1021, "Operator Licensing Examiner Standards," Revision 7, Supplement 1, Sections 201-203, 301-303, 401-403, to develop and administer the examinations.

Results (Units 1, 2, and 3):

Operations

- Seven of the eight applicants for senior reactor operator licenses satisfied the requirements of 10 CFR 55.33(a)(2) (Section 1).

- Seven of the eight applicants passed the written examination. Scores ranged from a low of 75 percent to a high of 95 percent, with an average of 85 percent overall (Section 1.1).

- The team noted good procedure usage, plant awareness and ownership from the applicants (Section 1.2.2)

- The team observed generally good command and control, communications, safety awareness, and systems knowledge; however, one generic weakness was identified in operator knowledge of the steam bypass control system, and application of that knowledge in an emergency event (Section 1.2.1)

- One senior operator candidate, (a licensed reactor operator) was removed from licensed duties pending further evaluation by the licensee. His removal from licensed duties was based on poor performance during one scenario when the candidate was in the secondary operator position (Section 1.2.1)

- The facility developed examinations were challenging and discriminating, tested at the proper level, and required only mirror changes prior to NRC administration (Section 1.1, 1.2).

Summary of Inspection Findings:

- There were no findings that were assigned a tracking number identified during the course of this inspection.

Attachments:

- Attachment 1 - Persons Contacted and Exit Meeting
- Attachment 2 - Simulation Facility Report
- Attachment 3 - Written Examination and Answer Key

DETAILS

1 LICENSED OPERATOR APPLICANT INITIAL QUALIFICATION EVALUATION (NUREG-1021)

This inspection consisted of an evaluation of applicants for senior reactor operator licenses. The licensee volunteered to participate in a national pilot study for these examinations. The pilot study involved facility development of all initial examination material, facility administration of the written examination (not including the generic fundamentals examination,) and NRC administration of the operating examination. NRC coordinated with the facility using the pilot examination guidance contained in Regional Office Interaction Memorandum 9525, Attachment 1, and Generic Letter 95-06, and granted final approval of all initial examinations to be administered at the facility.

During the inspection, the examiners evaluated the qualifications of eight applicants for senior reactor operator licenses; seven currently licensed as reactor operators, and one whose senior reactor operator license had expired. The inspection assessed the eligibility and administrative and technical competency of the applicants to be issued licenses to direct the operation of the reactivity controls at the Palo Verde Units 1, 2, and 3 commercial nuclear power facility in accordance with 10 CFR Part 55 and NUREG-1021, "Operator License Examiner Standards," Revision 7, Supplement 1, Sections 200 (series), 300 (series), and 400 (series). Further, the inspection included evaluations of facility procedures and simulation capability used to support administration of the examinations. These areas were evaluated using the guidance provided in the areas of NUREG-1021 cited above. Finally, the examiners observed the performance of onshift operators and plant conditions during the conduct of inplant applicant evaluations.

After completion of the evaluations, the examiners determined that seven of the eight applicants for senior reactor operator licenses satisfied the requirements of 10 CFR 55.33(a)(2).

1.1 Written Examination

The facility developed and submitted the written examination to the chief examiner for review on September 20, 1995. On October 3, 1995, the chief examiner and a representative from the NRC headquarters operator licensing branch discussed revisions to the examination with the facility. The chief examiner found the as-developed examinations challenging and discriminating, testing at the proper level of knowledge, valid for a licensing decision and meeting the requirements of NUREG 1021, Revision 7, Supplement 1. Only minor revisions were required during the preadministration examination review. The chief examiner confirmed incorporation of the revisions into the final written examination on October 11, 1995, while onsite to conduct the review of the operating examinations.

The facility administered and graded the approved written examination on October 23, 1995. The chief examiner also graded the written examination, and reviewed the facility analysis of the examination results as required in the pilot guidance. The facility did not request any post-administration revisions to the examinations, and the chief examiner concurred with the facility grading and analysis.

One of the senior reactor operator upgrade applicants failed the written examination with a score of 75 percent. The highest grade on the examination was 95 percent, with an average overall of 87 percent. Eight questions (Nos. 13, 21, 45, 47, 51, 78, 91, 92) were missed by four or more of the candidates and were evaluated by the facility to be valid questions. The examiners concurred with this evaluation.

No generic training weaknesses were identified as a result of the written examination results.

1.2 Operating Examinations

The facility developed comprehensive operating tests in accordance with the pilot guidance and guidelines of NUREG-1021, Revision 7, Supplement 1, Section 301. The operating tests consisted of three parts: an administrative portion, a dynamic simulator scenario portion, and a control room/plant walkthrough portion. The chief examiner reviewed and validated the various portions of the operating tests at the Palo Verde, Unit 2, facility and dynamic simulator during the week of October 9, 1995. The as-developed operating examinations required only minor changes based on the NRC review. The licensee's personnel, under security agreement, assisted in the onsite validation. The examination team administered the operating tests during the week of October 23, 1995.

1.2.1 Dynamic Simulator Scenarios

The examiners evaluated three crews consisting of two to three applicants each on two to three scenarios using the Palo Verde plant-specific simulation facilities. The examiners evaluated the applicants' competencies by comparing actual performance during the scenarios against expected performance in accordance with the requirements in NUREG-1021, Revision 7, Supplement 1, Section 303.

One generic weakness was identified involving operator knowledge of the steam dump bypass control system, and application of that knowledge in an emergency event. During a steam generator tube rupture scenario, one crew failed to recognize that a main steam pressure transmitter had failed, which resulted in failure of the steam dump bypass control system to function in automatic. The steam dump bypass control system was still available for manual operation. The candidate acting as the secondary operator recognized the failure of the steam dump bypass control system to function in automatic, but was unable to operate the steam dump bypass control system in manual, and subsequently opened the steam generator atmospheric dump valves, which initiated an

unnecessary offsite release of reactor coolant from the ruptured steam generator. A second crew, during the same scenario, diagnosed the failure of the main steam pressure transmitter, but the candidate acting as the secondary operator recommended performing the reactor coolant system cooldown using the atmospheric dump valves. The candidate acting as the control room supervisor corrected the secondary operator candidate and directed manual operation of the steam dump bypass control system.

The team observed generally good command and control, communications, safety awareness and systems knowledge, however, one crew was a significant exception. During a steam generator tube rupture scenario, a senior reactor operator upgrade candidate, acting as the secondary operator performing reactor operator duties, failed to perform requiring actions in several instances. First, the candidate failed to reduce auxiliary feedwater flow to the steam generators following the manual reactor trip, which resulted in an excessive cooldown of the reactor coolant system. Next, while conducting his standard post-trip actions, the candidate recognized that the steam dump bypass control system had failed to operate in automatic, but did not inform the control room supervisor. The candidate did not diagnose the cause of the failure, which was a failed main steam pressure transmitter. The candidate attempted to take manual control of the steam dump bypass control system, but failed to manually initiate an arming signal and subsequently was unable to operate the steam dumps in manual. Manual operation of the steam dump bypass control system was available and was the preferred method of decay heat removal following a reactor trip and steam generator tube rupture. The candidate then opened the atmospheric dump valves, initiating an offsite release of reactor coolant from the ruptured steam generator. The candidate did not inform the control room supervisor of his actions. When the control room supervisor reached Step 9 of Procedure 40EP-9E004, "Steam Generator Tube Rupture," he appropriately directed that a cooldown be initiated to a hot-leg temperature less than 550 degrees F. At that point, the candidate informed the control room supervisor that the hot-leg temperature had already decreased to less than 550 degrees F. The control room supervisor did not stop to question the abnormally low hot-leg temperature, and the crew did not offer additional information as to the cause of the low temperature. An automatic main steam isolation then occurred while the control room supervisor was directing that the automatic setpoint for main steam isolation be adjusted to prevent an main steam isolation. The main steam isolation occurred due to the excessive cooldown initiated earlier by the candidate. This sequence of events forced the crew to continue the reactor coolant system cooldown using the atmospheric dump valves. The candidate's actions and failure of the crew to recognize the abnormal cooldown and availability of manual control of the steam dump bypass control system resulted in an unnecessary and unmonitored offsite release of reactor coolant for approximately 16 minutes.

During a low power scenario, which started at 3 percent power, the same crew failed to recognize that they were in a limiting condition for operation action statement in Mode 2 which did not permit entry into Mode 1. The crew increased power to greater than 5 percent, Mode 1, as directed in the turnover, but in violation of Technical Specification 3.7.1.2, Action a, and

Technical Specification 3.0.4. Later in the same scenario, the crew initiated a manual reactor trip when a loss of all main feedwater occurred at 5 percent power with auxiliary feed available. The manual reactor trip was not required due to the low power level and adequate inventory in the steam generators.

Use of the correct procedures would have directed the operators to reduce power to below 3 percent and start an auxiliary feedwater pump. Although the safety significance is low, the examiners considered this action to be nonconservative in that the reactor was unnecessarily tripped from power operation.

The facility licensee administratively removed the senior reactor operator upgrade candidate from licensed reactor operator shift duties pending further evaluation of his capabilities and completion of required remediation.

1.2.2 Walkthrough Examinations

The examination team evaluated each of the senior reactor operator applicants using system oriented job performance measures related to job tasks within the scope of their potential duties as appropriate in accordance with NUREG-1021, Revision 7, Supplement 1. This included nonlicensed operator tasks outside the control room and performance of some tasks in the simulator in the dynamic mode. Other job performance measures were simulated through discussion in the control room and at local plant stations. Each of the applicants was required to enter radiologically controlled access to complete one or more tasks. In addition to tasks, the examiners asked prescripted questions related to the task system. Facility administrative procedures and practices were also examined using job performance measures or questions.

The team noted good procedure usage and system knowledge during the walkthroughs. The team also observed good plant housekeeping and material condition, and plant ownership from the candidates.

While conducting walkthroughs in the main control room, the examiners observed professional conduct from the licensed operators on shift, and good control of access to the main control room.

1.3 Simulator Facility

During the preparation and conduct of the operating examinations, the examination team observed some discrepancies in simulator fidelity, however, each of these had been identified previously by the licensee. The observed discrepancies did not impact examination validity.

ATTACHMENT 1

1 PERSONS CONTACTED

1.1 Licensee Personnel

*J. Levine, Vice President, Nuclear Production
*R. Nunez, Department Leader, Operations Training
*A. Krainik, Department Leader, Regulatory Affairs
*G. Overbeck, Vice President, Nuclear Support
*J. Velotta, Division Leader, Operations Training Support
*P. Wiley, Department leader, Operations Support
*B. Eklund, Consultant, Regulatory Affairs
*J. Hunter, Operations Training Liasion
*M. Baughman, Section Leader, Operations Training
*S. Zerkel, Operations Training Coordinator
*J. Brister, Operations Training Instructor
*W. Ide, Director, Operations
*W. Stewart, Executive Vice President, Arizona Public Service
*J. Taylor, Department Leader, Unit 3 Operations
*D. Marks, Section Leader, Nuclear Assurance
*R. Henry, Site Representative
*G. Box, Licensee Operator Initial Training Supervisor, Operations Training
*B. Dayyo, Senior Representative, Strategic Communications
*F. Riedel, Department Leader, Unit 2 Operations
*C. Zell, Department Leader, Unit 1 Operations
M. Sharp, Instructor
T. Mock, Instructor
J. Dennis, Operations Support Supervisor

1.2 NRC Personnel

*D. Garcia, Resident Inspector

In addition to the personnel listed above, the inspectors contacted other personnel during this inspection period.

* Denotes personnel that attended the exit meeting.

2 EXIT MEETING

An exit meeting was conducted on October 27, 1995. During this meeting, the inspectors reviewed the scope and findings of the report. The licensee did not express a position on the findings documented in this report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.

ATTACHMENT 2

SIMULATION FACILITY REPORT

Facility Licensee: Palo Verde Nuclear Generating Station

Facility Docket Nos.: 50-528
50-529
50-530

Operating Tests Administrated on: October 18, 1994

These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations. While conducting the simulator portion of the operating tests, the following unidentified discrepancies were observed:

ITEM	DESCRIPTION
None	

ATTACHMENT 3

WRITTEN EXAMINATION AND ANSWER KEY

US NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION
REGION 4

FACILITY: PALO VERDE

REACTOR TYPE: PWR-CE

DATE ADMINISTERED: 10/23/95

CANDIDATE: _____

INSTRUCTIONS TO CANDIDATE:

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. Points for each question are indicated in parenthesis after the question.

To pass this examination, you must achieve an overall grade of at least 80%. Examination papers will be picked up four hours after the examination starts.

RESULTS

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

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

_____
Candidate's Signature

POLICIES AND GUIDELINES FOR TAKING NRC WRITTEN EXAMINATIONS

1. Cheating on the examination will result in a denial of your application and could result in more severe penalties.

2. 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.

3. To pass the examination, you must achieve a grade of 80 percent or greater.

4. The point value for each question is indicated in parenthesis after the question number.

5. There is a time limit of 4 hours for competing the examination.

6. Use only black ink or dark pencil to ensure legible copies.

7. Print your name in the blank provided on the examination cover sheet and the answer sheet.

8. Mark your answers on the answer sheet provided and do not leave any question blank.

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

10. 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.

11. When you complete the examination, assemble a package including the examination questions, examination aids, and answer sheets and give it to the examiner or proctor. Remember to sign the statement on the examination cover sheet.

12. After you have turned in your examination, leave the examination area as defined by the examiner.

Question 1. (1.00)

Which ONE of the following is the Technical Specification basis for the Transient Insertion CEA Limits and the Shutdown CEA Insertion Limits?

The insertion limits ensure that...

a. the minimum SHUTDOWN MARGIN is maintained and the potential effects of a CEA ejection accident are limited to acceptable levels.

b. the minimum SHUTDOWN MARGIN is maintained and the potential effects of CEA Rod Shadowing is limited to acceptable levels.

c. the minimum SHUTDOWN MARGIN is maintained and the potential effects of a Steam Line Break Accident with a single stuck out full length CEA are limited to acceptable levels.

d. SHUTDOWN MARGIN is maximized for all analyzed accidents.

Question 2. (1.00)

The following plant conditions exist:

- The plant is operating at 100% power.
- NCW Pump 'A' is tagged out for bearing replacement.
- A loss of PBB-S04 occurs due to an 86 Lockout on the normal supply breaker.

WHICH ONE of the following is correct concerning the operation of the Reactor Coolant Pumps?

a. The Reactor Coolant Pumps must be tripped within ONE minute or motor winding damage will occur.

b. The Reactor Coolant Pumps may be run indefinitely provided seal injection flow is maintained and the seal bleedoff valves remain open.

c. Nuclear Cooling Water must be restored within 20 minutes or seal damage will occur.

d. The Reactor Coolant Pumps may be operated without Nuclear Cooling Water flow for a maximum of 10 minutes, provided seal injection flow is maintained.

Question 3. (1.00)

WHICH ONE of the following correctly describes the physical connection between the CVCS Charging Pumps and the Safety Injection System?

a. Upstream of the Charging Pump Discharge Valves to the Low Pressure Safety Injection lines to RC Loops 1A and 1B.

b. Downstream of the Charging Pump Discharge Valves to the High pressure safety injection lines to RC Loops 1A, 1B, 2A, 2B.

c. Downstream of the Charging Pump Discharge Valves to the Low Pressure Safety Injection lines to RC Loops 1A and 1B.

d. Upstream of the Charging Pump Discharge Valves to the High Pressure Safety Injection lines to RC Loops 1A, 1B, 2A, 2B.

Question 4. (1.00)

Given the following plant conditions:

- The reactor is tripped.
- SIAS is actuated.
- Pressurizer level is 10% and lowering.
- CSAS is NOT initiated.
- Containment pressure is rising slowly on all channels.
- Containment pressure is 9.0 psig on Channel 'A'; 8.5 psig on Channel 'B'; 8.0 psig on Channel 'C'; and 8.0 psig on Channel 'D'.

WHICH ONE of the following should be performed ?

a. When Channels C or D reach 8.5 psig, then manually initiate CSAS.

b. Manually initiate CSAS due to the CSAS setpoint being met or exceeded.

c. When channel B, C or D reach 9 psig, then verify CSAS automatically initiates.

d. CSAS may be initiated now based on trend, but is NOT required.

Question 5. (1.00)

The following plant conditions exist:

- The reactor is operating at 67% power. Power is being reduced at 20% per hour.
- All available charging pumps are operating.
- Letdown is isolated.
- Estimated RCS leakage is 92 gpm.
- Containment pressure is 2.8 psig.
- Containment temperature is 158F.
- Pressurizer level is 32% and lowering.
- RCS pressure is 1910 psia and lowering.

WHICH ONE of the following should be performed?

a. Continue to reduce power to 20% and then trip the reactor and perform the Standard Post Trip Actions.

b. Implement 41AO-1ZZ14, "EXCESSIVE RCS LEAKRATE" and maintain steady state power for 15 minutes to get an accurate leakrate determination.

c. Manually trip the reactor, and perform the Standard Post Trip Actions.

d. Perform 41AO-1ZZ56, "RAPID SHUTDOWN" to reduce power and place the plant in Mode 3 in less than one hour.

Question 6. (1.00)

WHICH ONE of the following will cause a CWP "CEA WITHDRAWAL PROHIBIT" alarm due to a CEA misalignment generated from the CPC/CEAC?

a. Subgroup deviation greater than 2.00 inches.

b. PLCEAs less than 30 inches withdrawn.

c. Greater than 7.5 inch deviation between groups.

d. CEA deviation within a subgroup less than or equal to 3.00 inches.

Question 7. (1.00)

WHICH ONE of the following describes ALL of the trips produced from the Excore Nuclear Instruments Safety Channels to the Plant Protection System?

a. LPD, DNBR, VOPT, Hi Log power trips.

b. LPD, DNBR, VOPT.

c. LPD and DNBR.

d. Hi Log power and VOPT.

Question 8. (1.00)

Given the following plant conditions:

- The reactor is tripped.
- A Small Break LOCA is in progress.
- Both HPSI pumps are inoperable.
- Containment temperature is 190F.
- RCS pressure is 1600 psia.

WHICH ONE of the following would indicate fuel uncovery on QSPDS CETs?

a. Highest CET temperature indicates 610 degrees F.

b. 30 degrees F superheat and steady.

c. 10 degrees F superheat and is now becoming more superheated.

d. 20 degrees F superheat and steady.

Question 9. (1.00)

Given the following plant conditions:

- The reactor is tripped.
- A feed line break has occurred inside containment.
- Containment pressure is 1.2 psig and rising slowly.
- 86 Lockout on PBB-S04K, "PBB-S04 Normal Supply Breaker".
- SIAS/CIAS were manually initiated.
- Train 'A' SIAS load shed panels are re-energized.

WHICH ONE of the following components is NOT available due to these conditions?

a. CTMT Normal ACU Fan, HCN-A01D.

b. Normal Chiller, WCN-E01C.

c. Non Essential Aux Feed Pump, AFN-P01.

d. Condensate Pump, CDN-P01B.

Question 10. (1.00)

WHICH ONE of the following is expected for a high alarm on RU-7, "AUXILIARY STEAM CONDENSATE RECEIVER TANK" monitor?

a. Post filter blower shifts to the "THRU FILTER MODE".

b. Auxiliary Steam Condensate diverts to the Liquid Radwaste System.

c. No AUTO functions are associated with this monitor.

d. Turbine Building Sump diverts to the Liquid Radwaste System.

Question 11. (1.00)

Given the following plant conditions:

- The reactor is tripped.
- SIAS is actuated.
- A steam line break is occurring in the containment.
- Containment pressure is 10 psig and rising.
- A loss of PBA-S03 has occurred.

WHICH ONE of the following is correct concerning the Containment Spray (CS) System?

a. CS Pump, SIA-P03 is on. CS Pump, SIB-P03 is on. SIA-UV-672 open. SIB-UV-671 open.

b. CS Pump, SIA-P03 is off. CS Pump, SIB-P03 is on. SIA-UV-672 closed. SIB-UV-671 open.

c. CS Pump, SIA-P03 is off. CS Pump, SIB-P03 is on. SIA-UV-672 open. SIB-UV-671 closed.

d. CS Pump, SIA-P03 is off. CS Pump, SIB-P03 is on. SIA-UV-672 open. SIB-UV-671 open.

Question 12. (1.00)

Given the following plant conditions:

- The plant is operating at 100% power.
- Level in Hotwell 1C is 29 inches.
- Level in Hotwell 2C is 35 inches.
- Condensate pump 'B' suction valve is open on Hotwell 2C.

WHICH ONE of the following describes the effect on the condensate system?

a. Condensate pump 'A' will trip.

b. Condensate pumps 'A' and 'B' will trip.

c. Condensate pumps 'A' and 'C' will trip.

d. All condensate pumps will trip.

Question 13. (1.00)

WHICH ONE of the following indicates a condition that will AUTOMATICALLY trip a Main Feedwater Pump Turbine?

a. A single pump discharge high pressure switch actuates.

b. Two pump suction low pressure switches actuate for 5 seconds.

c. A single turbine exhaust vacuum switch momentarily actuates.

d. A single bearing low oil pressure switch actuates for 5 seconds.

Question 14. (1.00)

The following plant conditions exist:

- The reactor has tripped on low Steam Generator Level in #1 SG.
- AFAS 1 and AFAS 2 initiated.
- SG 1 level - 27% WR.
- SG 2 level - 32% WR.
- SG 1 pressure - 940 psia.
- SG 2 pressure - 1150 psia.

WHICH ONE of the following is correct in regards to the Aux Feedwater Valves to the Steam Generators.

	SG 1 AFW Isolation Valves	SG 1 AFW Regulating Valves	SG 2 AFW Isolation Valves	SG 2 AFW Regulating Valves
a.	CLOSED	CLOSED	OPEN	OPEN
b.	OPEN	OPEN	OPEN	OPEN
c.	CLOSED	CLOSED	CLOSED	CLOSED
d.	OPEN	OPEN	CLOSED	CLOSED

Question 15. (1.00)

Given the following conditions:

- The plant is operating at 100% power.
- Diesel Generator 'A' is paralleled with offsite power.

WHICH ONE of the following indicates the effect on Diesel Generator 'A' to a loss of PKA-M41?

a. DG 'A' remains running, but its output breaker trips open.

b. DG 'A' trips and its output breaker trips open.

c. DG 'A' trips, but its output breaker remains closed.

d. DG 'A' remains running, and its output breaker remains closed.

Question 16. (1.00)

Given the following conditions:

- A planned Waste Gas (WG) Decay Tank release is in progress.
- A valid high alarm is received on RU-12, "WASTE GAS DECAY TANK" Discharge in the Control Room.

WHICH ONE of the following describes the automatic actions as a result of this alarm?

a. RDT to Waste Gas Valve, CH-UV-540 will close.

b. Waste Gas Discharge Valves, GRN-UV-34A and GRN-UV-34B close.

c. Initiates a FBEVAS.

d. Initiates a CREFAS.

Question 17. (1.00)

Given the following plant conditions:

- The plant is operating at 100% power.
- COLSS is out of service.
- Linear Heat Rate is 14 kW/ft.
- 15 minutes later, Linear Heat Rate is 14.5 kW/ft.

WHICH ONE of the following actions should be performed to comply with Technical Specifications?

a. Restore Linear Heat Rate to within the LCO limits within 4 hours.

b. Within 15 minutes, initiate corrective action to restore Linear Heat Rate to within the LCO limits within 1 hour.

c. Reduce thermal power to less than or equal to 20% RATED THERMAL POWER within the next 6 hours.

d. Within 1 hour, commence a shutdown to place the plant in MODE 2 over the next 6 hours and cold shutdown within the following 30 hours.

Question 18. (1.00)

The following plant conditions exist:

- The reactor was manually tripped.
- 3 Full Length CEAs are fully stuck out.
- RWT level is 70%.
- RCS pressure is 1950 psia.

WHICH ONE of the following is a correct Emergency Boration Flowpath per 41AO-1ZZ01, "EMERGENCY BORATION"?

a. CH-HV-536 to the Charging Pumps.

b. CH-V164 via CH-UV-514 to the Charging Pumps.

c. CH-V327 to the Charging Pumps.

d. RWT to the High Pressure Safety Injection Pumps.

Question 19. (1.00)

Given the following plant conditions:

- The reactor is at 100% power.
- An inadvertent CSAS Train 'A' has occurred.
- A SIAS has NOT actuated.
- Containment Spray Pump 'A' has been stopped.

WHICH ONE of the following would have to be performed if it is required to restart Containment Spray Pump 'A'?

a. Place the handswitch in the start position to override and then release the switch. The pump will auto start.

b. Place the handswitch in the start position to override and then release the switch. Place the handswitch in the start position again and the pump will start.

c. Place the handswitch in the stop position to override and then release the switch. The pump will auto start.

d. Momentarily turn off control power to the pump breaker. With control power restored, the pump will start.

Question 20. (1.00)

WHICH ONE of the following is the reason for maintaining pressurizer level below the upper limit of the Pressurizer level programmed band?

a. The water volume is limited to allow the reactor to be maneuvered at design rates at low load values.

b. The water volume is limited to ensure the heaters remain covered on a reactor trip.

c. The steam volume should be sufficient to accept a coolant insurge without the water level reaching the safety valve nozzles.

d. Sufficient steam volume is maintained to prevent a Safety Injection Actuation Signal (SIAS) on a reactor trip.

Question 21. (1.00)

Given the following plant conditions:

- A LOCA has occurred inside containment.
- Containment temperature is 205 degrees F.
- Containment pressure is 2.8 psig.
- RCS pressure is 1050 psia.
- Containment H_2 concentration is 0.7%.
- RU-16 is in Alert Alarm.

WHICH ONE of the following criteria must be met in order to reset the CIAS actuation and restore CIAS actuated equipment?

a. Containment H_2 concentration must be reduced to less than 0.3%.

b. Containment temperature must be reduced to less than 150 degrees F.

c. Containment pressure must be less than 2.5 psig.

d. RU-16 not in Alert or High alarm.

Question 22. (1.00)

Given the following plant conditions:

- The plant is operating at 100% power.
- RCS pressure is 2250 psia.
- Pressure control channel selector, RCN-HS-100 is positioned to "100X".
- Pressurizer heater control level trip channel selector, RCN-HS-100-3 is selected to "BOTH".

WHICH ONE of the following describes a consequence of PZR Pressure Transmitter 100Y failing high?

a. SBCS valves will receive an auto modulation signal.

b. SBCS valves will receive an auto permissive signal.

c. All pressurizer heaters will energize.

d. All pressurizer heaters will de-energize.

Question 23. (1.00)

Given the following plant conditions:

- The plant is operating at 100% power.
- PZR level control system is in Remote Auto.
- PZR level setpoint is 50%.
- PZR actual level reached 59% and is lowering slowly.
- Charging Pump Mode Selector Switch is in 2-3-1.

WHICH ONE of the following describes the response of the Pressurizer Level Control system to these conditions?

a. Charging Pump # 1 will remain stopped until actual PZR level gets to 50%.

b. Charging Pump # 2 will remain stopped until actual PZR level gets to 52.5%.

c. Charging Pump # 3 will remain stopped until actual PZR level gets to 53%.

d. PZR Backup Heaters will turn off if they were energized by Hi level deviation.

Question 24. (1.00)

WHICH ONE of the following describes the basis for the High Steam Generator Level Trip Setpoint?

a. Provided to protect the turbine from excessive moisture carryover.

b. Provides protection against a loss of feedwater flow incident.

c. Provides protection in the event of an increase in heat removal by the secondary system and subsequent cooldown of the reactor coolant.

d. Provides Steam Generator overpressure protection.

Question 25. (1.00)

WHICH ONE of the following states the power level at which the FWCS will shift from single element control to three element control?

a. 15%

b. 20%

c. 50%

d. 65%

Question 26. (1.00)

WHICH ONE of the following is the hydrogen concentration inside containment that would require operation of the hydrogen recombiner to reduce hydrogen levels following a LOCA?

a. 0.3%

b. 0.7%

c. 2.9%

d. 4.0%

Question 27. (1.00)

Given the following plant conditions:

- The plant is in mode 6.
- Containment Refueling Purge is in service.
- RU-37, "Power Access Purge - Train A" monitor is in alarm.

WHICH ONE of the following states the BOP ESFAS actuation(s) which should be automatically initiated for these conditions?

a. CPIAS with a cross trip to CREFAS.

b. CPIAS only.

c. CPIAS with a cross trip to FBEVAS.

d. CPIAS with a cross trip to CREFAS and FBEVAS.

Question 28. (1.00)

Given the following plant conditions:

- A low level alarm has occurred in the spent fuel pool.
- Level is being restored by makeup from the RWT.
- Spent Fuel Pool Level is 138 feet 2 inches.

WHICH ONE of the following is a consequence of allowing the spent fuel pool to be filled higher according to 41AO-1ZZ53, "LOSS OF REFUELING POOL AND/OR SPENT FUEL POOL LEVEL"?

a. Flooding of the fuel elevator cable tray causing damage to the elevator motor.

b. Dilution of the spent fuel pool to an unacceptable boron concentration.

c. Gravity flow to the RWT when the BAM Pump is stopped.

d. RWT level will fall below the minimum useful volume required per Technical Specifications.

Question 29. (1.00)

WHICH ONE of the following describes how to complete a fuel transfer if during automatic transfer of fuel, the sequence is halted due to winch overload?

Reset the winch overload and ...

a. restart the transfer in automatic to preserve the proper sequence.

b. complete the transfer through individual commands because carriage location will be reset in the automatic transfer mode.

c. restart the transfer in automatic to ensure all interlocks remain in effect.

d. complete the transfer through individual commands because the automatic sequence can not be restarted.

Question 30. (1.00)

Given the following plant conditions:

- The reactor is tripped following a Steam Generator Tube Rupture.
- RCS pressure is 895 psia.
- RCS subcooling is 55 degrees F.
- Steam Generator #1 pressure is 890 psia.
- RU-4 is in high alarm.
- Steam generator #1 is isolated.
- Steam Generator #1 level is 78% NR and rising slowly.
- Steam Generator #2 level is 50% NR and steady.

WHICH ONE of the following is the preferred method to control level in the isolated steam generator?

a. Bypass the MSIV and steam #1 Steam Generator to the condenser.

b. Steam #1 Steam Generator to atmosphere via the ADVs.

c. Lineup high rate blowdown to the condenser from #1 Steam Generator.

d. Lower RCS pressure below #1 Steam Generator pressure and allow backflow to the RCS.

Question 31. (1.00)

Given the following plant conditions:

- The plant tripped from 100% power due to a LOCA.
- Hot and cold leg injection are in service.
- Steam Generator #1 level 92% WR and steady.
- Steam Generator #2 level 91% WR and steady.
- Steam Generator #1 Pressure 600 psia and lowering.
- Steam Generator #2 Pressure 610 psia and lowering.
- Containment Pressure 8.5 psig and lowering.
- Containment H_2 Concentration 1.3% and steady.
- RCS Pressure 300 psia.
- RWT level 10% and lowering.

WHICH ONE of the following states the safety signals that should have initiated in addition to SIAS/CIAS?

a. CSAS

b. MSIS/AFAS

c. MSIS/CSAS

d. MSIS/RAS

Question 32. (1.00)

The following plant conditions exist:

- The reactor is operating at 75% power.
- Operators are performing the actions of 41AO-1ZZ08, "STEAM GENERATOR TUBE LEAK.
- RU-142, "MAIN STEAM LINE N-16" monitors, Channels 1,2,3,4 are in High Alarm.
- RU-141, "CONDENSER VACUUM/GLAND SEAL EXHAUST" monitor is in High Alarm.
- RU-4, "STEAM GENERATOR #1 BLOWDOWN" monitor is in Alert alarm and trending up.
- Air Removal Post Filter Blower Mode Select is in the "THRU FILTER MODE".

WHICH ONE of the following is the cause of the Post Filter Blower automatically shifting modes?

a. Alert alarm on RU-141.

b. High alarm on RU-142.

c. Alert alarm on RU-4.

d. High alarm on RU-141.

Question 33. (1.00)

Given the following plant conditions:

- The reactor is operating at 100% power.
- A loss of offsite power occurs.
- NBN-X03 Sudden pressure fault.

WHICH ONE of the following correctly describes the Sequencer mode and the effect on the Diesel Generator 'A' output breaker?

a. Mode 3, and the output breaker will close.

b. Mode 2, and the output breaker will remain open.

c. Mode 1, and the output breaker will remain open.

d. Mode 4, and the output breaker will close.

Question 34. (1.00)

Given the following plant conditions:

- DG 'A' has automatically started due to AFAS 1.
- Jacket Water Temperature is 450 degrees F and rising.
- Turbocharger oil pressure is 1 psig and lowering.
- The engine is vibrating excessively.
- Generator Differential Alarm is actuated.

WHICH ONE of the following describes the effect on the Diesel Generator?

a. Continue to run until a high priority alarm is received.

b. Immediately trip due to the Generator Differential.

c. Immediately trip due to low Turbocharger Oil Pressure.

d. Immediately trip due to high Jacket Water Temperature.

Question 35. (1.00)

Given the following plant conditions:

- A fire is occurring on the 100' level of the Control Bldg. in the 'A' Class battery room.
- Automatic fire suppression has actuated.

WHICH ONE of the following states the method of fire protection in the affected battery room and the hazard to personnel?

a. Wet pipe sprinkler system will actuate causing a shock hazard.

b. H_2O Deluge system will actuate causing a shock hazard.

c. CO^2 will flood the battery room causing a respiratory hazard.

d. Halon will flood the battery room causing a respiratory hazard.

Question 36. (1.00)

WHICH ONE of the following is the basis for maintaining <u>containment vessel structural</u> integrity?

a. Ensures that the containment will withstand the maximum pressure of 49.5 psig in the event of a LOCA.

b. Ensures that the containment will withstand the maximum of 60 psig in the event of a Steam Line Break Accident.

c. Ensures that Site Boundary Dose Limits will not exceed 10 CFR 100 limits regardless of penetration leakage.

d. Ensures that combustible gases will not exceed explosive concentrations in the event of a large break LOCA.

Question 37. (1.00)

Given the following plant conditions:

- The plant is in Mode 5.
- RCS temperature is 205 degrees F.

WHICH ONE of the following is the minimum required to provide overpressure protection for the RCS?

a. One operable SDC suction line relief valve.

b. Two operable SDC suction line relief valves.

c. One operable pressurizer code safety valve with a lift setting of 2500 psia ± 1%.

d. All pressurizer code safety valves are operable with a lift setting of 2500 psia ± 1%.

Question 38. (1.00)

Given the following plant conditions:

- The plant is at 100% power.
- PZR safety valve, PSV-203 has seat leakage.
- RDT level is rising.
- RDT pressure is 9.8 psig and rising slowly.

WHICH ONE of the following automatic actions will occur if no operator action is taken?

a. The RDT vent to waste gas header valve, CHN-UV-540 will close and the RDT rupture disc, CHN-PSE-12 will rupture.

b. The RDT vent to waste gas header valve, CHN-UV-540 will open and the RDT outlet containment isolation valve, CHA-UV-560 will close.

c. The RDT vent to waste gas header valve, CHN-UV-540 will close and the RDT outlet containment isolation valve, CHA-UV-560 will close.

d. The RDT vent to waste gas header valve, CHN-UV-540 will open and the RDT rupture disc, CHN-PSE-12 will rupture.

Question 39. (1.00)

Given the following plant conditions:

- The Main Turbine is at load with the Load Limiter limiting and the Load Set Reference at SET LOAD.
- The operator moves the Load Limiter in the increase direction.
- RCS temperature begins to decrease and reactor power begins to increase.

WHICH ONE of the following statements describes what has occurred?

a. The Load Set Reference is at SET LOAD, therefore an EHC malfunction is indicated.

b. The Load Set Reference value was greater than the Load Limit value and the rate of increase is determined by the rate the operator moved the Load Limit Potentiometer.

c. The Load Set Reference value was greater than the Load Limit value and the rate of increase is determined by the rate set in the Loading Rate Limit.

d. The Load Set Reference value was equal to the Load Limit value and the rate of increase is determined by the rate the operator moved the Load Limit Potentiometer.

Question 40. (1.00)

Given the following plant conditions:

- INST/SERV AIR SYS TRBL alarm is in.
- INST AIR HDR PRESS LO alarm is in.
- INST AIR N_2 BKUP VLV OPEN alarm is in.
- Instrument Air Header Pressure is 60 psig and lowering.

WHICH ONE of the following is correct concerning these indications?

a. Main Steam Isolation Valves fail AS-IS. The Backup Accumulator will allow 5 fast closure valve operations.

b. Steam Generator Economizer Feedwater Control Valves fail AS-IS. The Backup Accumulator will allow 5 fast closure valve operations.

c. Feedwater Economizer Isolation Valves fail AS-IS. Manual operation of these valves is available, as necessary.

d. Feedwater Downcomer Control Valves fail AS-IS. Manual operation of these valves is available, as necessary.

Question 41. (1.00)

While performing 41AO-1ZZ35, "CONTINUOUS CEA WITHDRAWAL", the Control Room Supervisor (CRS) directs that the MODE SELECT at the CEDM control panel be placed in SB (standby). The CEA motion stops. Operators determine that 2 CEAs have an inward subgroup deviation of 10 inches. WHICH ONE of the following actions should now be performed?

a. The CRS should perform 41AO-1ZZ11, "DROPPED OR SLIPPED CEA" to recover the rods.

b. Emergency borate to ensure adequate Shutdown Margin.

c. Manually trip the reactor and GO TO 40EP-9EO01, "SPTAs".

d. Commence a rapid shutdown to be within Cold Shutdown in the next 6 hours.

Question 42. (1.00)

Given the following plant conditions:

- The plant is at 95% power.
- A CEA has dropped to the bottom of the core.

WHICH ONE of the following is the reason for a slow recovery of a CEA that has been misaligned greater than one (1) hour per 41AO-1ZZ11, 'DROPPED OR SLIPPED CEA"?

a. To prevent unequal power levels in different loops.

b. To ensure that peaking induced by xenon will not cause a high power reactor trip.

c. To prevent fuel damage from occurring.

d. To ensure that an ASI pretrip is not received from the excore channel nearest the dropped CEA.

Question 43. (1.00)

WHICH ONE of the following specifies the MAXIMUM allowed misalignment limit for CEAs within its group according to Technical Specification 3.1.3, "MOVABLE CONTROL ASSEMBLIES"?

a. 6.6 inches.

b. 7.5 inches.

c. 9.9 inches.

d. 19 inches.

Question 44. (1.00)

WHICH ONE of the following states the reason for cooling down the Steam Generators in a large break LOCA?

a. Minimizes two phase flow.

b. Prevents the offsite dose rate from exceeding 10CFR100 criteria.

c. Improves RCS heat removal by enhancing natural circulation and reflux boiling.

d. Prevents the RCS from exceeding P/T limits.

Question 45. (1.00)

Given the following plant conditions:

- The reactor is at 100% power.
- RCP 1A Controlled Bleedoff flow is 6.0 gpm.
- RCP 1A Seal injection flow is 6.8 gpm.
- RCP 1A HP Cooler Inlet Temperature is 225 degrees F.
- RU-6 is in Alert Alarm.
- NCW supply temperature is 95 degrees F.

WHICH ONE of the following actions is required per 41AO-1ZZ29, "RCP AND MOTOR EMERGENCY"?

a. Trip the reactor, stop the affected RCP and perform the Standard Post Trip Actions (SPTAs).

b. Trip the reactor, concurrently perform the Standard Post Trip Actions, stop all RCPs.

c. Adjust NCW supply temperature by means of heat exchanger bypass valve NC-HCV-102.

d. Unlock and close the power supply breakers for the high pressure cooler isolation valves on the affected RCP.

Question 46. (1.00)

WHICH ONE of the following states when boration may be discontinued per 41AO-1ZZ01, "EMERGENCY BORATION"?

a. Anytime pressurizer level will exceed the program band.

b. When RWT level drops to 73%.

c. A plant cooldown has been directed by an Emergency Operating Procedure.

d. When the required boron concentration is reached.

Question 47. (1.00)

Given the following plant conditions:

- The reactor is at 100% power.
- NCN-P01A is Red Tagged Out for repairs.
- NCN-P01B has an overcurrent relay and 86 device actuated.
- Essential Cooling Water (EW) Train 'A' is cross connected to Nuclear Cooling Water.
- Spent Fuel Pool temperature is 125 degrees F.

WHICH ONE of the following actions should be performed per 41AO-1ZZ05, "LOSS OF NUCLEAR COOLING WATER"?

a. Align Train 'B' of EW to the Fuel Pool Heat Exchanger.

b. Close EW to NCW cross connect valves EWA-HV-65 and EWA-HV-145.

c. Align Train 'A' of EW to the Fuel Pool Heat Exchanger.

d. Open Train 'B' EW to NCW cross connect valves EWB-HCV-66 and EWB-HCV-146.

Question 48. (1.00)

Given the following plant conditions:

- The reactor is at 100% power.
- PPS Channel 'A' low Steam Generator 2 level is in Bypass due to a failed Channel 'A' level detector.
- PPS Channel 'B' low Steam Generator 2 level is in Trip.
- PPS Channel 'B' DNBR is in Trip.
- PPS Channel 'C' low Steam Generator 2 level is in Trip.
- PPS Channel 'D' low Steam Generator 2 level is in pretrip.

WHICH ONE of the following states the correct action for these conditions?

a. Manually trip the reactor.

b. Perform a rapid shutdown.

c. Maintain stable power while the correct Technical Specifications are entered.

d. Maintain power and determine if the plant auto trips when Channel 'D' goes into trip.

Question 49. (1.00)

WHICH ONE of the following discriminates between a steam line rupture inside containment and a small break LOCA?

a. RCS temperature.

b. RCS pressure.

c. Containment temperature.

d. Containment pressure.

Question 50. (1.00)

Given the following plant conditions:

- The reactor is at 100% power.
- Condenser backpressure is increasing rapidly.

WHICH ONE of the following is the condenser absolute pressure at which a Steam Bypass Control System Condenser Interlock will occur?

a. 4.0 inches HgA

b. 5.0 inches HgA

c. 7.5 inches HgA

d. 13.5 inches HgA

Question 51. (1.00)

Given the following plant conditions:

- A Station Blackout has occurred.
- Offsite power is not available.
- Steaming is being controlled by the ADVs.
- Steam Generators are being fed via AFA-P01.

WHICH ONE of the following actions should be performed to minimize loads on NKN-M46?

a. Open the supply breaker to the plant computer and allow it to automatically transfer to the backup power supply.

b. Open the breaker to the Main Generator Primary Protection Unit Tripping Power.

c. Open the breaker to the Emergency Seal Oil Pump.

d. Open the breaker to FWPT 'A' Emergency Oil Pump.

Question 52. (1.00)

WHICH ONE of the following is the reason for setting the CEAC 1 inoperable flag in ALL CPC channels prior to restoring power to PNA-D25 according to 41AO-1ZZ15, "LOSS OF CLASS IE INSTRUMENT AC POWER"?

a. Prevent a possible reactor trip due to erroneous high penalty factors from the CEAC.

b. Prevent an actual reactor trip due to valid high penalty factors from the CEAC.

c. CEAC 1 and CEAC 2 are inoperable due to the loss of PNA-D25.

d. Ensures that RTSG breakers 'A' and 'C' remain closed, preventing a reactor trip.

Question 53. (1.00)

Given the following plant conditions:

- The plant is in Mode 3 following a steam line break.
- 40EP-9EO05, Excess Steam Demand is in progress.
- SG #2 is isolated.
- SG #2 is completely blowndown.
- Loop 1 T_C is 510 degrees F and stable.
- Loop 2 T_C is 500 degrees F and stable.
- All Safety Injection Valves are throttled closed.
- PZR level is 40% and lowering.
- The Primary Operator reports that SI flow has to be re-initiated.
- RU-5 is in alert alarm.
- RU-16 is in alert alarm.
- SG #2 pressure is 20 psig and rising.

WHICH ONE of the following events is indicated?

a. PZR steam space LOCA

b. SGTR induced by the ESD.

c. Letdown line break inside containment.

d. Re-initiation of feed flow to #2 SG.

Question 54. (1.00)

Given the following plant conditions:

- A fire has occurred in the control room.
- The control room has been evacuated.

WHICH ONE of the following must be performed within 12 minutes of a fire in the control room to limit the loss of RWT inventory and/or initiation of an undesired RAS according to 41AO-1ZZ44, "CONTROL ROOM FIRE".

a. Open HPSI Pump 'A' Breaker, PBA-S03E.

b. Open CS Pump 'A' Breaker, PBA-S03D.

c. Locally close LPSI Pump 'A' Suction Valve, SIA-HV-683.

d. Locally close CS Pump 'A' Suction Valve, SIA-V105.

Question 55. (1.00)

Given the following plant conditions:

- The reactor is at 90% power.
- The control room is being evacuated due to threat of toxic gas.

WHICH ONE of the following actions must be performed prior to leaving the control room and transferring control to the remote shutdown panel according to 41AO-1ZZ27, "SHUTDOWN OUTSIDE CONTROL ROOM"?

a. Trip all RCPs.

b. Trip RCPs 1A and 2A.

c. Place the letdown control valve selector switch to the "BOTH" position.

d. Shift Charging Pump suction to the RWT by opening CHN-HV-536 and closing the VCT outlet valve, CHN-UV-501.

Question 56. (1.00)

Given the following plant conditions:

- The reactor has been manually tripped.
- A SGTR has occurred in SG 1 at 300 gpm.
- NAN-S01 and NAN-S02 failed to fast transfer to offsite power.
- Safety valves lifted on BOTH Steam Generators.
- A safety valve on SG 1 has failed to reseat.
- All other systems have responded normally.
- The CRS is performing Safety Function Tracking Page of 40EP-9EO09, "Functional Recovery"

WHICH ONE of the following correctly indicates which safety functions are in jeopardy?

a. MVAC, HR, CI.

b. HR, CI

c. HR

d. CI

Question 57. (1.00)

WHICH ONE of the following indicates how decay heat is removed from the core during a large break LOCA?

a. Single phase natural circulation.

b. Break flow and two phase natural circulation.

c. Steam Generator steaming.

d. Break flow and single phase natural circulation.

Question 58. (1.00)

WHICH ONE of the following is the Technical Specification basis for the limits of primary coolant specific activity per 3.4.7, "RCS SPECIFIC ACTIVITY"?

a. Ensure that site boundary doses will not exceed a small fraction of 10 CFR 100 limits in the event of a SGTR with concurrent Loss of Offsite Power.

b. Ensure that site boundary doses will not exceed a small fraction of 10 CFR 100 limits in the event of a SGTR with concurrent failed open ADV.

c. Ensure that the cumulative radiation exposure to personnel during normal daily operations is minimized to meet ALARA considerations.

d. Ensure that personnel access to Safe Shutdown Equipment is not hampered due to excessive radiation levels.

Question 59. (1.00)

WHICH ONE of the following is the reason for placing all Charging Pumps in "PULL TO LOCK" when RWT level lowers to 7.4% during a large break LOCA?

a. There is insufficient NPSH to run the Charging Pumps in this condition.

b. The Charging Pumps will interfere with hot and cold leg injection flowpaths.

c. This minimizes the amount of radioactive containment sump inventory pumped outside the containment wall.

d. This ensures there is adequate suction available for the LPSI Pumps.

Question 60. (1.00)

Given the following plant conditions:

- The reactor is tripped.
- 40EP-9EO09, Excess Steam Demand is in use.
- SG 1 is isolated.

WHICH ONE of the following indicates when the first Main Steam Safety Valve will lift on the operating steam generator if heat removal is not maintained?

a. 1200 psig.

b. 1250 psig.

c. 1290 psig.

d. 1315 psig.

Question 61. (1.00)

WHICH ONE of the following is the reason that the GTGs are NOT used to energize switchyard loads in the event of a blackout if thunderstorm activity is present in the vicinity of PVNGS.

a. Ensure the continued availability of ESF transformer NBN-X03.

b. Ensure the continued availability of ESF transformer NBN-X04.

c. Ensure that switchyard breaker control power remains available.

d. Ensure that Startup transformer NAN-X01 remains available for offsite power restoration.

Question 62. (1.00)

WHICH ONE of the following is the reason that L03 and L10 Breakers are opened for a MINIMUM of 5 seconds when performing the Reactivity Control Safety Function of Standard Post Trip Actions if the reactor did not trip automatically and the manual trip pushbuttons on B05 were not successful?

a. Allows time for the trip coils to actuate to open L03 and L10 breakers.

b. Allows time for the motor generator contactors to open.

c. Allows time for the motor generator stop contact to close.

d. Allows time for the effects of the flywheel to taper off interrupting power to the CEAs.

Question 63. (1.00)

Given the following plant conditions:

- The reactor is at 28% power.
- Steam bypass control valves are all closed.
- Condenser backpressure is 5.5 inches HgA.

WHICH ONE of the following indicates the actions that should be taken according to 41AO-1ZZ07, "LOSS OF CONDENSER VACUUM"?

a. Decrease main turbine load until backpressure stabilizes at < 5 inches HgA.

b. Trip the main turbine and REFER to 4XAO-XZZ02, "LOAD REJECTION".

c. Trip the reactor and main turbine and Perform 40EP-9EO01, "SPTAs".

d. Reduce power to prevent a steam bypass control system actuation.

Question 64. (1.00)

Given the following plant conditions:

- A large break LOCA is in progress.
- RCS pressure is 40 psia.
- REP CET is 380 degrees F.
- T_h is 220 degrees F.
- Containment temperature is 195 degrees F.

WHICH ONE of the following is correct concerning the status of the Core Heat Removal Safety Function?

a. Met due to T_h being less than 610 degrees F.

b. Met due to adequate subcooling.

c. NOT met, due to inadequate subcooling.

d. NOT met, due to containment temperature > 170 degrees F.

Question 65. (1.00)

WHICH ONE of the following states the required feedflow while performing 40EP-9EO01, "Standard Post Trip Actions"?

The minimum required feedwater flow is...

a. at least a total feedwater flow of 250 gpm.

b. at least a total feedwater flow of 500 gpm.

c. a minimum of 200 gpm to each Steam Generator.

d. a minimum of 500 gpm to each Steam Generator.

Question 66. (1.00)

WHICH ONE of the following is the reason that the RCS is checked for subcooling of 24 degrees F during the RCS Inventory Control Safety Function of the Standard Post Trip Actions?

a. Ensures that adequate RCS fluid in an appropriate state is available for removal of decay heat.

b. Ensures that the heat removal safety function is satisfied.

c. To,determine if Safety Injection flow should be throttled.

d. To ensure that single phase natural circulation flow is adequate.

Question 67. (1.00)

Given the following plant conditions:

- The reactor has been manually tripped.
- A PZR steam space LOCA is in progress.
- PZR level is 70% and increasing.
- RCS pressure is 1750 psia and decreasing.
- CET Subcooling is 25 degrees F and decreasing.
- All RCPs are off.
- SG levels are 10% NR and increasing, being fed by AFB-P01.
- SIAS/CIAS are initiated.
- Containment temperature is 160 degrees F.
- RVUH level indicates 67%.

WHICH ONE of the following indicates what should be done in regards to Safety Injection flow?

a. Safety Injection flow should be throttled because all throttle criteria are met and indications are that they will continue to be met.

b. Safety Injection flow should be throttled because all throttle criteria are met and PZR level is greater than the Technical Specification limit.

c. Safety Injection flow should NOT be throttled because Steam Generator levels are less than the throttle criteria.

d. Safety Injection flow should NOT be throttled because it will be necessary to fill the PZR solid to maintain subcooling.

Question 68. (1.00)

Given the following plant conditions:

- A small break LOCA is in progress.
- The leak is NOT isolable.
- 2 hours have elapsed from the start of the LOCA.

WHICH ONE of the following conditions would NOT require simultaneous hot and cold leg injection to be established?

a. One RCP running in each loop.

b. At least one SG available for RCS heat removal.

c. Subcooled margin is 20 degrees F and lowering.

d. Pressurizer level 10% and steady.

Question 69. (1.00)

WHICH ONE of the following would require tripping the RCP(s) according to 40EP-9EO03, "LOCA"?

a. RCS subcooling of 38 degrees F.

b. RCP motor amps of 450A.

c. RCP Controlled Bleedoff flow of 8.5 gpm.

d. RCP Upper Thrust Bearing temperature of 275 degrees F.

Question 70. (1.00)

WHICH ONE of the following is the reason that all RCPs are stopped if subcooling is inadequate during a LOCA?

a. Ensures that the RCS will not be depressurized by Main Spray.

b. Prevents damage to the RCP seals.

c. Minimizes heat input to the RCS.

d. Ensures that RCS heat removal capability is not challenged.

Question 71. (1.00)

Given the following plant conditions:

- The reactor is at 100% power.
- Pressurizer parameters are normal.
- No radiation alarms.
- CHG HDR SYS TRBL alarm is lit.
- Charging Pumps to Regenerative Heat Exchanger Pressure Low computer alarm is in.

WHICH ONE of the following is the cause of this situation?

a. Regenerative Heat Exchanger Tube Leak.

b. Charging Header Leak.

c. RCS Leak.

d. Letdown Heat Exchanger Leak.

Question 72. (1.00)

Given the following plant conditions:

- The plant is in Mode 5.
- RCS inventory is being lost.
- SDC flow oscillations are occurring at intervals less than one minute apart.

WHICH ONE of the following is the reason that RCS level should be monitored only if boiling has NOT occurred?

a. Boiling pressurizes the RCS under the RV head causing RWLIS inaccuracies.

b. Boiling results in RWLIS inaccuracies due to higher temperature.

c. Boiling flow at the level instrumentation making it inaccurate.

d. RCS level is not a priority with boiling occurring.

Question 73. (1.00)

Given the following plant conditions:

- The plant is in Mode 5.
- A total loss of SDC flow has occurred.
- PZR Once Through Cooling is selected as the heat removal method.

WHICH ONE of the following is the correct concerning where RCS temperature should be maintained according to 41AO-1ZZ22, "LOSS OF SHUTDOWN COOLING"?

a. CET temperature 190 - 200 degrees.

b. T_h temperature 200 - 210 degrees.

c. T_c less than 210 degrees.

d. T_h less than 200 degrees.

Question 74. (1.00)

WHICH ONE of the following signals will automatically deenergize the Pressurizer Proportional Heaters?

a. Pressurizer level of 25%.

b. Pressurizer level deviation of +3%.

c. Safety Injection Actuation Signal.

d. Pressurizer pressure at the AUTO setpoint.

Question 75. (1.00)

Given the following plant conditions:

- The reactor has automatically tripped.
- SUR is negative.
- Two full length CEAs failed to fully insert.
- Boration is in progress from the RWT via CH-HV-536.

WHICH ONE of the following satisfies the reactivity control safety function per 40EP-9EO04, "Standard Post Trip Actions"?

a. Reactor power stable at less than 1×10^{-5}%.

b. Adequate shutdown margin established.

c. Reactor power dropping.

d. L03 and L10 Supply Breakers opened.

Question 76. (1.00)

Given the following plant conditions:

- The plant is at 100% power.
- A Steam Generator Tube leak is occurring in SG #1.
- Primary to secondary leakrate is 15 gallons/day.

WHICH ONE of the following radiation monitors would detect this size break first?

a. RU-4, "SG Blowdown Monitor".

b. RU-139, "Main Steam Line Monitor".

c. RU-141, "Condenser Offgas Monitor".

d. RU-142, "Main Steam Line N-16 Monitor".

Question 77. (1.00)

WHICH ONE of the following is the reason for cooling down the RCS to a T_h of less than 550 degrees F prior to isolating the affected Steam Generator in 40EP-9EO04, "Steam Generator Tube Rupture"?

a. Ensure that the pressurizer safety valves do not lift following Steam Generator isolation.

b. Prevents SBCVs 1007 and 1008 from opening when using the SBCS.

c. Ensures Steam Generator Safety Valves do not lift following Steam Generator isolation.

d. Along with Steam Generator level of 40 - 60% NR, ensures that the core heat removal safety function is met.

Question 78. (1.00)

Given the following plant conditions:

- The plant has tripped from 100% power.
- 40EP-9EO06, "Loss of all Feedwater" is in progress.
- Feed has been restored to #1 SG via AFB-P01.
- SG 1 level is 40% WR.
- SG 2 level is 50% WR.

WHICH ONE of the following is the MAXIMUM feed limitation following restoration of feedwater?

a. 500 gpm.

b. 1000 gpm.

c. 1600 gpm.

d. 2000 gpm.

Question 79. (1.00)

Given the following plant conditions:

- The reactor is at 100% power.
- A loss of PKA-M41 has occurred.
- RDT level is increasing.

WHICH ONE of the following correctly describes why RDT level is increasing?

a. CHA-UV-516, "Letdown Containment Isolation Valve" fails closed causing letdown relief valve CHN-PSV-354 to lift.

b. CHA-UV-560, "RDT Outlet Valve" fails closed. CHA-UV-580, "RDT makeup valve" fails open.

c. CHE-FV-241,242,243,244, "Seal Injection Flow Control Valves" fail open on the loss of instrument air to containment causing seal injection relief valve CHN-PSV-865 to lift.

d. CHA-UV-506, "RCP Controlled Bleedoff to VCT Valve" fails closed causing RCP seal bleedoff relief valve CHN-PSV-199 to lift.

Question 80. (1.00)

Given the following plant conditions:

- The reactor is at 100% power.
- All instrument air compressors are running.
- IA header pressure is 88 psig and gradually decreasing.

WHICH ONE of the following states when the reactor should be tripped and the Emergency Operations procedures implemented according to 41AO-1ZZ06, "LOSS OF INSTRUMENT AIR"?

a. If instrument air pressure can not be regained before the plant exhibits unstable operations.

b. If instrument air pressure can not be maintained above 70 psig.

c. If instrument air pressure can not be maintained above 85 psig with the Nitrogen Backup Valve open.

d. If instrument air pressure decreases to 65 psig and the Front Standard Turbine Trip Air Relay Dump Valve actuates its low pressure switches.

Question 81. (1.00)

Given the following plant conditions:

- The reactor is at 100% power.
- "PZR TRBL" alarm is in.
- "PZR LVL HI-LO" alarm is in.
- "RCP SEAL INJ FLOW HI-HI OR LO-LO" alarm is in.
- PZR level (RCN-LI-110X) - Off Scale High.
- PZR level (RCN-LI-110Y) - 50%.
- PZR pressure - 2260 psia.
- ALL PZR Backup heaters are energized.
- Charging Header flow - 44 gpm.
- Letdown Header flow - 90 gpm and increasing.

WHICH ONE of the following operator actions is required to stabilize the plant?

a. Stop the running charging pump and decrease charging flow, then restore the running charging pump after PZR level is restored.

b. Switch to the unaffected channel and if unable to restore level in auto, then take manual control of PZR Level Control System with RCN-LIC-110 and restore level by decreasing letdown flow.

c. Switch to the unaffected channel and if unable to restore level in auto, then take manual control of PZR Level Control System with RCN-LIC-110 and restore level by increasing letdown flow.

d. Manually start another charging pump and take manual control of the RCP seal injection flow controllers and restore seal injection to the normal range.

Question 82. (1.00)

Given the following plant conditions:

- The plant is in Mode 6.
- Fuel movement is in progress.
- An irradiated fuel element is suspended above the core.
- Area radiation levels become HIGH and a Containment evacuation is necessary.
- Radiation levels allow placement of the fuel in a safe position.

WHICH ONE of the following is the preferred location for storing the irradiated fuel assembly?

The irradiated fuel assembly should be stored in the...

a. upender in the vertical position.

b. upender in the horizontal position.

c. core in any available location.

d. core in the designated location.

Question 83. (1.00)

Given the following plant conditions:

- A Loss of Offsite Power has occurred.
- All offsite transmission lines are damaged and deenergized.
- NBN-X03 and NAN-S06 are faulted to ground.
- DG 'A' has tripped and cannot be restored.
- No feedwater is available to the Steam Generators.

WHICH ONE of the following success paths must be implemented to restore feed capability using AFN-P01?

a. MVAC-1, "Offsite Power".

b. MVAC-2, "Diesel Generators".

c. MVAC-3, "Gas Turbine Generators".

d. MVAC-4, "Second Unit DG".

Question 84. (1.00)

WHICH ONE of the following is correct in regards to performance of electrical lineups?

a. Breakers/switches should be cycled to verify position.

b. All available indications of position and control power availability should be checked.

c. Lockouts and protective relay targets should be checked actuated.

d. Breakers should be visually checked only if in the racked-out position.

Question 85. (1.00)

WHICH ONE of the following is true in accordance with 40AC-9OP15, "STATION TAGGING AND CLEARANCE" procedure?

a. A yellow Caution tag may hang with a MIP tag provided there are no tagging conflicts. The MIP tag overrides the yellow Caution tag.

b. A yellow Caution tag may hang with a blue Men-At-Work tag provided there are no tagging conflicts. The yellow Caution tag overrides the blue Men-At-Work tag.

c. A yellow Caution tag may hang with a red Danger tag even if there are tagging conflicts. The red Danger tag overrides the yellow Caution tag.

d. A yellow Caution tag may hang with a blue Men-At-Work tag even if there are tagging conflicts, provided the instructions on the Caution tag are understood prior to operating the affected equipment.

Question 86. (1.00)

WHICH ONE of the following is a requirement of the Site Fire Team according to 40AC-9OP02, "CONDUCT OF SHIFT OPERATIONS"?

a. A Site Fire Team of at least 5 members shall be maintained on site at all times. Operations shall provide an advisor to the Site Fire Team and as a minimum will be an Auxiliary Operator.

b. A Site Fire Team of at least 4 members shall be maintained on site at all times. Operations shall provide an advisor to the Site Fire Team and as a minimum will be a Reactor Operator.

c. A Site Fire Team of at least 5 members shall be maintained on site at all times. Operations shall provide an advisor to the Site Fire Team and as a minimum will be a Reactor Operator.

d. A Site Fire Team of at least 4 members shall be maintained on site at all times. Operations shall provide an advisor to the Site Fire Team and as a minimum will be a Senior Reactor Operator

Question 87. (1.00)

WHICH ONE of the following responsibilities may be delegated by the Emergency Coordinator during performance of EPIP-02, "EMERGENCY CLASSIFICATION"?

a. Authorization for emergency workers to exceed 10CFR20 exposure limits.

b. Determination of the necessity for onsite evacuation.

c. Initiate activation of onsite and offsite emergency response organizations for an ALERT or higher level of classification.

d. Determination of site boundary dose assessments.

Question 88. (1.00)

Given the following plant conditions:

- The plant has tripped from 100% power.
- A large break LOCA has occurred.
- QSPDS Subcooled Margin indicates 2 degrees of superheat and steady.
- Containment Systems have responded normally.

WHICH ONE of the following is the MINIMUM required classification for this event per EPIP-02, "EMERGENCY CLASSIFICATION"?

a. Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

Question 89. (1.00)

WHICH ONE of the following correctly states the PVNGS Administrative Hold Points for Whole Body radiation exposure?

a. Total Effective Dose Equivalent (TEDE) at 1.5 rem/year, 2.0 rem /year, 2.5 rem/year, 4.0 rem/year.

b. Total Effective Dose Equivalent (TEDE) at 2.0 rem/year, 2.5 rem/year, 3.0 rem/year, 4.0 rem/year.

c. Deep Dose Equivalent (DDE) at 2.0 rem/year, 3.0 rem/year, 4.0 rem/year, 5.0 rem/year.

d. Committed Effective Dose Equivalent (CEDE) at 1.0 rem/year, 2.0 rem/year, 3.0 rem/year, 4.0 rem/year.

Question 90. (1.00)

A point source in the auxiliary building is reading 500 mrem/hr at distance of Two (2) feet. Two options exist to complete rework on a valve near this radiation source.

Option 1: Operator X can perform the assignment in thirty minutes working at a distance of four feet from the point source.

Option 2: Operators Y and Z, who have been trained in the use of a special extension tool can perform the same task in 75 minutes at a distance of eight feet from the point source.

WHICH ONE of the following options is preferable and consistent with the ALARA program?

a. Option 1 as X's exposure is 31.25 mrem.

b. Option 1 as X's exposure is 62.50 mrem.

c. Option 2 as the exposure per person is 39.06 mrem.

d. Option 2 as the exposure per person is 78.12 mrem.

Question 91. (1.00)

Given the following plant conditions:

- The plant is in mode 5.
- Condensate Pump 'A' breaker is to be racked out.

WHICH ONE of the following is required personnel protective equipment to rack out this breaker?

a. Low voltage rubber gloves.

b. Hard hat.

c. Leather glove protectors.

d. Safety goggles covered by a face shield.

Question 92. (1.00)

WHICH ONE of the following is required for operation of the Main Generator according to 41OP-1GH01, "GENERATOR HYDROGEN"?

a. Hydrogen purity must be a minimum of 75%.

b. Hydrogen gas pressure must be a minimum of 30 psig.

c. Seal oil pressure must be a minimum of 8 psig greater than machine gas pressure.

d. Hydrogen temperature must be less than 30 degrees C.

Question 93. (1.00)

Given the following plant conditions:

- A Loss of Offsite Power has occurred.
- A Site Wide PA announcement is made that "An oxygen deficient atmosphere may exist in Unit 1 in valve galleries and rooms below the 100 ft elevation. Oxygen monitors shall be used in or near these rooms".

WHICH ONE of the following is the highest Oxygen concentration that would indicate an oxygen deficient atmosphere?

a. 15%

b. 17%

c. 19%

d. 21%

Question 94. (1.00)

Given the following plant conditions:

- The reactor is tripped.
- RCS pressure is 1830 psia.
- T_C is 560 degrees F.
- RCS temperature is being controlled by the ADVs.
- SG levels are 50% WR, being restored by AFB-P01 at 400 gpm to each SG.
- The CRS has completed Standard Post Trip Actions and diagnosed the SGTR procedure.
- The crew is stopping 1 RCP in each loop following a SIAS actuation as directed by the SGTR procedure.
- The STA reports to the CRS that the RCS heat removal safety function is not being met.

WHICH ONE of the following correctly describes what procedure steps the CRS should follow?

a. Go back to the SPTAs and rediagnose the event.

b. Exit the SGTR procedure and implement the Functional Recovery procedure due to the lost safety function.

c. Exit the SGTR procedure and implement the Functional Recovery procedure due a dual event in progress.

d. Continue the SGTR procedure because the SGTR procedure will recover the safety function.

Question 95. (1.00)

Given the following plant conditions:

- The plant tripped from 75% power due to a LOOP.
- Grid voltage was lost and is now restored.
- The CRS desires to restart RCPs.

WHICH ONE of the following is reset first in order to restart RCPs in accordance with 40EP-9EO07, "LOSS OF OFFSITE POWER"?

a. RCP breaker 286 Lockout Relay.

b. RCP Speed Switch Trip Relay 214 Slow Speed Alarm Target.

c. G5/186R-1 and G3/186R-2 Turb & Gen Aux Trip & Lockout Relays.

d. G7/186M Turb & Gen Aux Trip & Lockout Relay.

Question 96. (1.00)

During an abnormal event, the CRS directs an operator to place a controller in the MANUAL position.

WHICH ONE of the following is correct in regards to taking a controller from AUTO to MANUAL?

a. The operation should normally be performed at the individual component controller.

b. The operation should be done from the highest effective level available.

c. With the controller in manual, minimal operator attention is required to monitor the system response during the event.

d. If the event results in a plant trip, the operator should place the controller back in AUTO after the SPTAs are done.

Question 97. (1.00)

Given the following plant indications:

- The plant has tripped due to a Loss of All Feedwater event.
- The CRS directs the Area Operator to perform Appendix 40, "Local Operation of AFN-P01" to locally close the AFN-P01 Breaker.

WHICH ONE of the following is correct regarding the communication with the Area Operator?

a. The phonetic alphabet does not have to be used when communicating in the EOPs.

b. As a minimum, the Control Room should tell the Area Operator to "Perform Appendix 40".

c. It is the Area Operators responsibility to ensure he performs what the Control Room intends.

d. The Area Operator shall report back the component to be manipulated and the desired state.

Question 98. (1.00)

Given the following plant conditions:

- The plant is in Mode 3 following a Reactor Trip on Low DNBR.
- AFAS 1 actuated.
- AFAS 2 NOT actuated.
- AFB-P01 handswitch is in Red flag and the pump is not running.
- AFB-P01 Stop light indicates bright green on switch AFB-HS-10 on BO6.
- SG 1 level is 38% WR and increasing.
- SG 2 level is 30% WR and lowering.
- A white SEIS light is lit for AFB-P01.

WHICH ONE of the following is the cause of the white SEIS indication?

a. AFAS 1 is actuated and AFB-P01 is not in its required initiation condition.

b. Loss of control power to AFB-P01.

c. 786 Lockout on AFB-P01 breaker.

d. AFAS 2 failed to actuate as required.

Question 99. (1.00)

Given the following plant conditions:

- 77OP-9RJ04, "COLSS FUNCTIONAL VERIFICATION" is in progress.
- All rod positions need to be inserted.

WHICH ONE of the following is correct with regards to the restoration of rod positions and the reason for using this method?

a. Restoration should be performed in the UNSCHEDULED MODE because with COLSS running the unusual rod configuration during this evolution can cause problems with COLSS and other programs monitoring rod position.

b. Restoration should be performed in the SCHEDULED MODE because as the rod positions are individually restored, the Radial Peaking Factor will update so that when the rod position insertion is complete, COLSS will calculate valid power and Power Operating Limits sooner.

c. Restoration should be performed in the SCHEDULED MODE because it is important to keep COLSS running while inserting the rod positions to determine if the computer accepted the rod position by observing the Radial Peaking Factor Calculation Bypassed Flag (NKPRPFLG) reset.

d. Restoration should be performed in the UNSCHEDULED MODE because the COLSS program will not accept any changes to point Ids that are currently being used for calculation unless the calculation is halted during the change.

Question 100. (1.00)

Given the following plant conditions:

- The plant is operating at 98% power.
- Steam Generator Tube leakage is estimated at 90 gpm in #1 SG.
- The CRS directs a Reactor Trip, Manual SIAS/CIAS and stopping of two RCPs.

WHICH ONE of the following indicates the correct order in which these directions should be carried out?

a. Trip the reactor, verify the reactivity control safety function, initiate SIAS/CIAS, stop 2 RCPs, complete the remaining SPTAs.

b. Trip the reactor, initiate SIAS/CIAS, stop 2 RCPs, verify the reactivity control safety function, complete the remaining SPTAs.

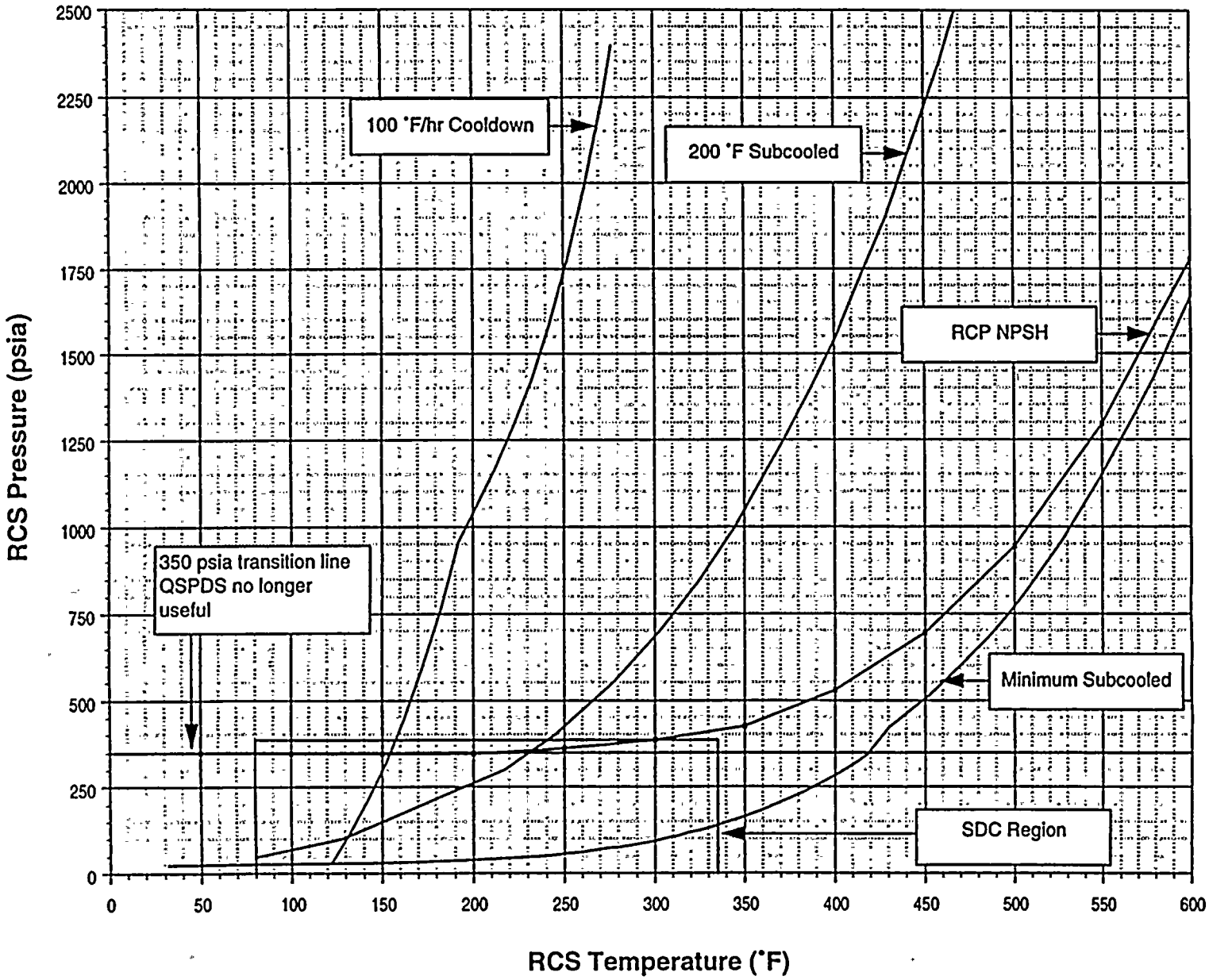
c. Trip the reactor, perform SPTAs in any order, initiate SIAS/CIAS, stop 2 RCPs.

d. Any order, as long as the manual reactor trip pushbuttons are depressed first.

END OF EXAMINATION

Appendix 2,
Figures

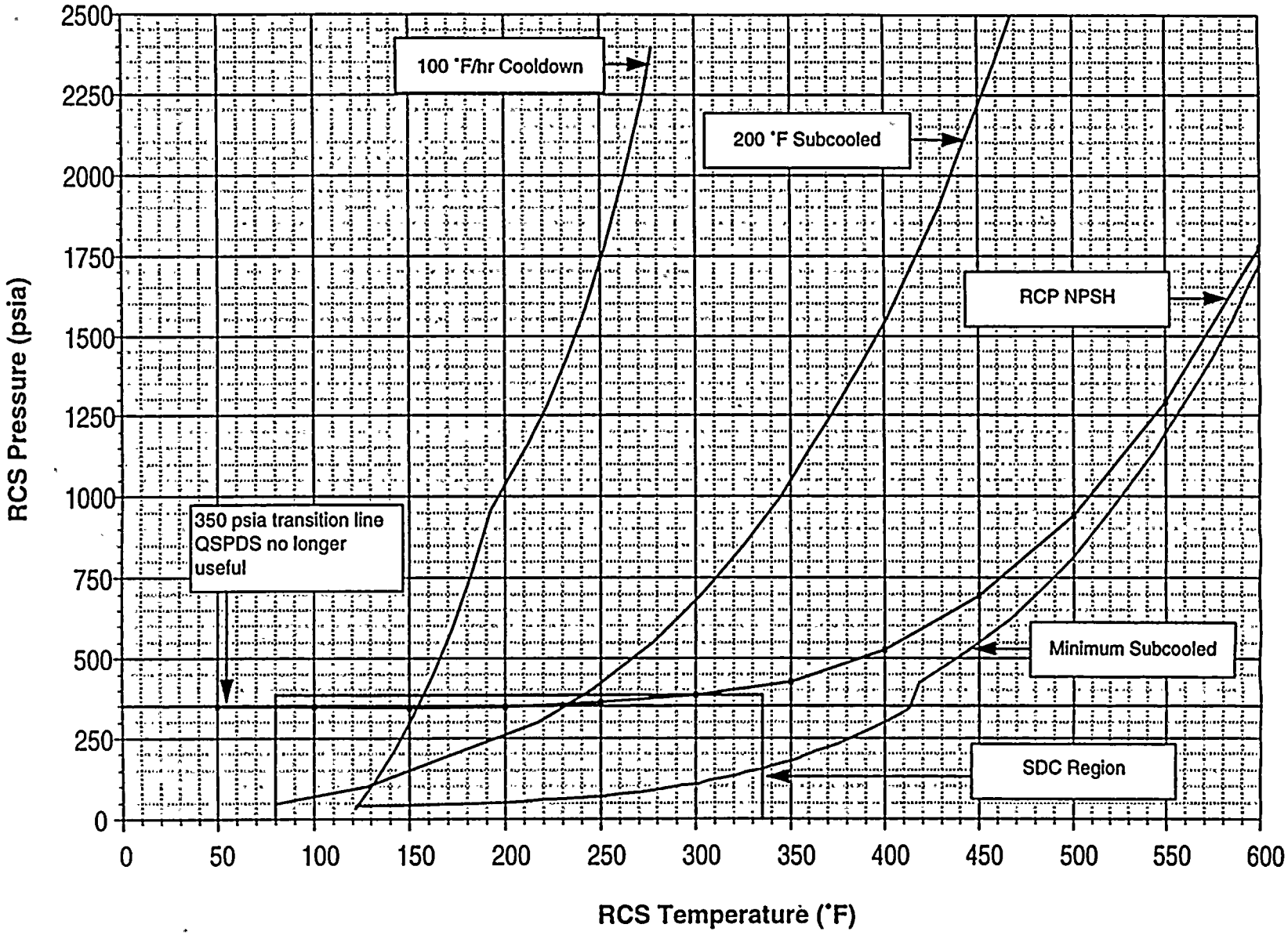
Pressure Temperature Limits NORMAL Containment Conditions



Forced Circulation - Th indication used

Natural Circulation - REP CET

Pressure Temperature Limits HARSH Containment Conditions



Forced Circulation - Th indication used

Natural Circulation - REP CET used

SEATING CHART

	PROCTOR TABLES	

EXAM START 0910
STOP 1310

RON OAKLEY FINISHED 10:42	PAT WILEY FINISHED 11:20
NICK POVIO FINISHED 11:15	DAVE CHARLES FINISHED 12:35
STEVE COATES FINISHED 12:41	DAN ARBUCKLE FINISHED 12:04
JORDAN JOHNSTON FINISHED 11:23	FRANK KUSLUCH FINISHED 12:30

Examination Quality Assurance Check

Question Number	Number Missed	Wiley	Charles	Arbuckle	Kusluch	Oakley	Povio	Coates	Johnston
1	1								X
2	2							X	X
3	0								
4	0								
5	0								
6	1					X			
7	2						X		X
8	2		X						X
9	1						X		
10	0								
11	0								
12	2		X			X			
13	6	X			X	X	X	X	X
14	1							X	
15	1					X			
16	0								
17	0								
18	0								
19	0								
20	1		X						
21	5	X				X	X	X	X
22	1	X							
23	0								
24	0								
25	1					X			
26	3		X			X		X	
27	0								
28	1								X
29	2							X	X
30	0								
31	0								
32	1			X					
33	3	X				X		X	
34	0								
35	0								
36	1					X			
37	3			X			X		X
38	3	X					X		X
39	1					X			

Question Number	Number Missed	Wiley	Charles	Arbuckle	Kusluch	Oakley	Povio	Coates	Johnston
40	2				X	X			
41	0								
42	0								
43	1					X			
44	0								
45	6	X	X		X	X		X	X
46	0								
47	4	X				X	X		X
48	0								
49	2					X	X		
50	1					X			
51	4					X	X	X	X
52	0								
53	0								
54	0								
55	1					X			
56	0								
57	0								
58	1					X			
59	3			X	X	X			
60	0								
61	0								
62	2		X		X				
63	2	X	X						
64	0								
65	0								
66	0								
67	0								
68	0								
69	0								
70	3	X				X		X	
71	3				X			X	X
72	0								
73	0								
74	0								
75	1	X							
76	1	X							
77	0								
78	4				X	X	X	X	
79	0								
80	1		X						

Question Number	Number Missed	Wiley	Charles	Arbuckle	Kusluch	Oakley	Povio	Coates	Johnston
81	0								
82	1					X			
83	0								
84	0								
85	0								
86	0								
87	1				X				
88	0								
89	0								
90	2			X	X				
91	6	X	X	X	X		X		X
92	4	X				X		X	X
93	0								
94	0								
95	1				X				
96	1						X		
97	1		X						
98	0								
99	2					X		X	
100	0								
TOTALS	106	13	10	5	11	25	12	14	16

The following questions were evaluated for validity due to being missed by half or more of the applicants:

Question 13

This is a valid question. No changes required.

Question 21

This is a valid question. No changes required.

Question 45

This is a valid question. No Changes required.

Question 47

This is a valid question. No changes required.

Question 51

This is a valid question. No changes required.

Question 78

This is a valid question. No changes required.

Question 91

This is a valid question. No changes required.

Question 92

This is a valid question. No changes required.

The following questions were asked during the examination by the examinees:

Jordan Johnston
Question 11
Does Bullets represent a time line?
Answer: yes

Frank Kusluch
Question 2
Which charging pumps are running?
Answer: Proctor Mark Sharp told him to wait until Proctor Tom Mock came back to the room to answer. Candidate said "Never mind".

Nick Povio
Question 11
When did you lose AC power?
Answer: Bullets represent time line.

Steve Coates
Question 11
Only SIAS occurred?
Answer: Assume all required actuation's occurred.

Dave Charles
Question 11
Where in sequence of events did this occur (loss of power)
Answer: Occurred prior to >8.5 psig in CTMT.

Class Brief on Question 11 to inform them that loss of AC power occurred prior to >8.5 psig in CTMT.
Specific follow-up to Johnston and Povio of "Do you understand that power was restored prior to >8.5 psig? They both responded yes.

Nick Povio
Question 49
Stated "We don't really use these parameters to discriminate between a steam line break and LOCA.
Answer: I can't comment on that.

Dan Arbuckle
Question 37
Is this a basis question or an LCO question?
Answer: I can't answer that.

Steve Coates
Question 88
Are we to assume a plenum level?
Answer: Answer the question based on the given conditions.

Nick Povio
Question 94
There is no indication that a safety function is lost.
Answer: Answer based on the given indications.

Steve Coates
Question 2
Do we have a loss of NC?
Answer: Based on the given conditions, answer the question.

ANSWER SHEET

Multiple Choice (Circle or X your choice)

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

MULTIPLE CHOICE

001	a	b	c	d	a	024	a	b	c	d	a
002	a	b	c	d	d	025	a	b	c	d	a
003	a	b	c	d	d	026	a	b	c	d	b
004	a	b	c	d	b	027	a	b	c	d	a
005	a	b	c	d	c	028	a	b	c	d	a
006	a	b	c	d	b	029	a	b	c	d	d
007	a	b	c	d	a	030	a	b	c	d	d
008	a	b	c	d	c	031	a	b	c	d	c
009	a	b	c	d	a	032	a	b	c	d	d
010	a	b	c	d	b	033	a	b	c	d	a
011	a	b	c	d	b	034	a	b	c	d	b
012	a	b	c	d	a	035	a	b	c	d	c
013	a	b	c	d	d	036	a	b	c	d	a
014	a	b	c	d	a	037	a	b	c	d	b
015	a	b	c	d	c	038	a	b	c	d	c
016	a	b	c	d	b	039	a	b	c	d	b
017	a	b	c	d	b	040	a	b	c	d	a
018	a	b	c	d	c	041	a	b	c	d	c
019	a	b	c	d	d	042	a	b	c	d	c
020	a	b	c	d	c	043	a	b	c	d	a
021	a	b	c	d	d	044	a	b	c	d	c
022	a	b	c	d	b	045	a	b	c	d	d
023	a	b	c	d	c	046	a	b	c	d	d

ANSWER SHEET

Multiple Choice (Circle or X your choice)

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

047	a	b	c	d	a	074	a	b	c	d	a
048	a	b	c	d	a	075	a	b	c	d	c
049	a	b	c	d	a	076	a	b	c	d	d
050	a	b	c	d	b	077	a	b	c	d	c
051	a	b	c	d	d	078	a	b	c	d	b
052	a	b	c	d	a	079	a	b	c	d	d
053	a	b	c	d	b	080	a	b	c	d	a
054	a	b	c	d	b	081	a	b	c	d	b
055	a	b	c	d	c	082	a	b	c	d	d
056	a	b	c	d	d	083	a	b	c	d	b
057	a	b	c	d	b	084	a	b	c	d	b
058	a	b	c	d	a	085	a	b	c	d	a
059	a	b	c	d	a	086	a	b	c	d	c
060	a	b	c	d	b	087	a	b	c	d	d
061	a	b	c	d	a	088	a	b	c	d	b
062	a	b	c	d	d	089	a	b	c	d	a
063	a	b	c	d	b	090	a	b	c	d	b
064	a	b	c	d	c	091	a	b	c	d	d
065	a	b	c	d	b	092	a	b	c	d	b
066	a	b	c	d	a	093	a	b	c	d	c
067	a	b	c	d	d	094	a	b	c	d	d
068	a	b	c	d	a	095	a	b	c	d	c
069	a	b	c	d	d	096	a	b	c	d	b
070	a	b	c	d	d	097	a	b	c	d	d
071	a	b	c	d	b	098	a	b	c	d	c
072	a	b	c	d	a	099	a	b	c	d	a
073	a	b	c	d	a	100	a	b	c	d	a