EXAMINATION REPORT

Docket No: 50-445 Construction Permit No.: CPPR-126

Licensee: Texas Utilities Electric Company
Skyway Tower
400 N. Olive Street
Lock Box 81
Dallas, TX 75201

Examinations administered at Comanche Peak Steam Electric Station (CPSES)

Chief Examiner: S L McCrory 8/9/84
S. L. McCrory, Examiner Date

Approved by: R. A. Cooley 8/13/84
R. A. Cooley, Section Chief Date

Summary

Examinations conducted on March 6-7 and April 5-7, 1984.

Written and/or oral examinations were administered to twelve (12) Senior (instant) Reactor Operators and thirteen (13) Reactor Operators. Seven (7) Senior and eight (8) Reactor Operators passed these examinations. All others failed.

CPSES EXAMINATION REPORT

Details

1. Examination Results

SRO Candidates				RO Candidates			
Total	Pass	Fail	%	Total	Pass	Fail	%
12	7	5	58	13	8	5	62

2. Examiners

S. L. McCrory, Chief Examiner, NRC
R. A. Cooley, NRC
R. Smith, NRC
J. Pellet, NRC
T. Burdick, NRC RIII
P. Isaksen, EG&G

3. Examination Report

This Examination Report is composed of the sections listed below.

A. Examination Review Meeting Comment Resolution

B. Exit Meeting Minutes

C. Evaluation Summary

D. Examination Master Copy (SRO/RO Questions and Answers)

Performance results for individual candidates are not included in this report because, as noted in the transmittal letter attached, examination reports are placed in NRC's Public Document Room as a matter of course.

A. Examination Review Meeting Comment Resolution

In general, editorial comments or changes made during the exam, the exam review, or subsequent grading reviews are not addressed by this resolution section. This section reflects resolution of substantive comments made during the exam review. The modifications discussed below are included in the master exam key which is provided elsewhere in this report as are all other changes mentioned above but not discussed herein. The following personnel were present for the exam review:

NRC	UTILITY
R. Cooley	W. Short
R. Smith	D. Hill
	S. Dyas
	W. Bird

COMMENTS

(1) 1.3/ 5.3 Candidates may indicate a higher value than 0.3% delta K/K.
Resp. REJECT

(2) 1.4/ 5.4 "Spectrum Hardening" should be accepted in addition to "Diffusion Length".
Resp. ACCEPT - Key Modified

(3) 1.6/ 5.6 Decreasing fluid temperature causes head loss to go up, pump differential pressure to go up, and volumetric flow rate to go down.
Resp. REJECT - For a centrifugal pump operating at constant speed (as most are designed to do), when fluid temperature decreases, pump work goes up but volumetric flow rate remains constant.

(4) 1.7/ 5.7 The actual enthalpy of the feedwater is slightly lower than that shown in the key due to subcooling.
Resp. ACCEPT - This factor was given credit during the grading when properly indicated in a candidate's answer.

(5) 1.10/ 5.10 When the same amount of reactivity is added again, the reactor will be slightly super critical.
Resp. ACCEPT - Key modified to credit either answer.

(6) 2.3 The candidates may draw a sketch of a compensated ion chamber.
Resp. ACCEPT - Drawings were given partial credit but required an operative explanation for full credit.

(7) 2.4 Nitrogen blanketing should be accepted as a means of excluding oxygen from the RCS during cold shutdown.
Resp. ACCEPT - Key modified

(8) 2.5/ 6.1 The steam dumps do not PREVENT the operation of the safety valves on a turbine/reactor trip.
Resp. REJECT - the steam dumps are designed to reduce the challenges to safety valves which may be better stated as preventing UNNECESSARY operation of the safety valves.

(9) 3.2 The answer should indicate an "increased" rather than a "reduced" steam generator level signal.
Resp. ACCEPT - Key modified

(10) 3.3/ 6.9 The grade point distribution should be indicated on the key drawing.
Resp. ACCEPT - Key modified

(11) 3.4/ 6.12 The alternate values of 1865# and 1765# should be deleted from the answer.
Resp. ACCEPT - Key modified

(12) 3.11 "Power Supply" should be substituted for "piece of equipment".
Resp. ACCEPT - Key modified

(13) 4.1 It is not clear if the question wants CPSES admin limits or 10 CFR 20 limits.
Resp. The question was graded on EITHER of CPSES or 10 CFR 20 limits but not a combination of both.

(14) 4.3/ 7.4 The intent of the word "steps" is not clear. Candidates may answer with either prerequisite steps or operational steps.
Resp. Conflicting interpretations of the meaning of "steps" was conveyed to different candidates during the exam. The key was modified to accept EITHER operational steps or conditional steps but not a combination of both.

(15) 4.7/ 7.2 The operability conditions of parts A and B are incorrect in the key.
Resp. ACCEPT - Key modified

(16) 4.10 The immediate action "Check if SI initiated" should be deleted from the required answer since the question states that SI has not occured.
Resp. ACCEPT - Key modified

(17) 7.1c Weekly, quarterly, and annual limits should be accepted.
Resp. ACCEPT - Key modified to accept several responses which were evaluated as equally correct. (See key)

B. <u>Exit Meeting Summary</u>

At the conclusion of each week of the exam period, examiners met with representatives of the plant staff to discuss the results of the examinations. The following personnel were present for the exit interviews:

First Week

NRC	UTILITY
R. Cooley	R. Seidel

This exit was held by telephone due to the late completion of examinations the previous day. All except one candidate were reported as clear passes on the oral examinations. There were no significant generic weaknesses identified during this part of the examination.

Second Week

NRC	UTILITY
R. Smith	B. R. Clements
J. Pellet	J. C. Krykendal
	K. A. Jones
	R. Seidel
	C. L. Turner

Mr. Smith detailed the results of the oral examinations and reported that several candidates' examinations would require further review in the region office to arrive at a final decision. The following generic areas were indentified as weaknesses during this week of examinations:

(1) Electrical distribution - There was a general lack of familiarity with the electrical distribution system particularly in the area of manual diesel generator operations.

(2) Nitrogen 16 Detectors - Several candidates demonstrated a poor understanding of the operation of the N-16 detectors.

(3) Thermodynamics - Some of the candidates were unable to adequately perform basic thermodynamic calculations.

C. Evaluation Summary

There is sufficient evidence that there may be a problem in the training program or its implementation to warrant investigation by facility personnel and further evaluation by NRC, particularly with regard to "Retake" candidates. Of the twenty candidates participating in this examination, eight (8) were retakes. Six (6) of these retakes failed the written examination, a 75% failure rate. Further breakdown shows that four (4) of the eight (8) written retake candidates were SROs. All four (4) of the SRO retake candidates failed the written examination. In contrast, first time candidates only had a 25% failure rate, three (3) of twelve (12).

During the currently scheduled September 1984 examination period at CPSES, the NRC will further evaluate the CPSES training program effectiveness to ascertain the need for program revision or upgrade.

D. Examination Master Copy

Date Administered: April 5, 1984

Exam Type: Reactor Operator and Senior Reactor Operator

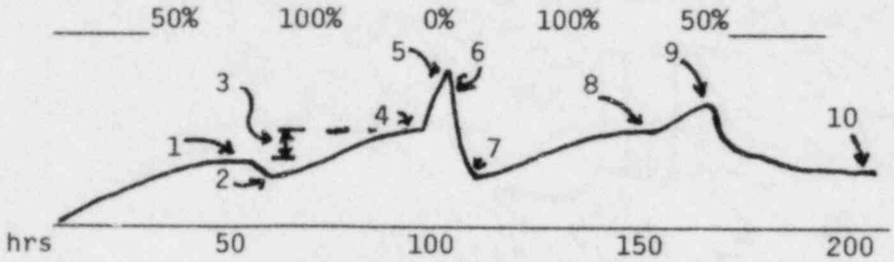
Comments: During the grading of the RO examination it was discovered that category 2 contained 27 points vice 25. The relative worth of category 2 was maintained by applying a factor of 0.926 (25 ÷ 27) to the points earned in category 2.

PRINCIPLES OF NUCLEAR POWER PLANT OPERATION,
THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW

1.1/ 5.1 Plot Xenon (Xe) for the following reactor conditions, assuming Xe free initially. (Show relative magnitude and curve shape - not absolute values) (3.0)

At time Zero Power step changes from 0% to 50%.
Stable Power level at 50% for 50 hrs.
Increase Power level 100% for next 40 hrs.
Reactor Trip power level 0% for next 8 hrs.
Startup Power level 100% for next 52 hrs.
Lower Power level 50% for next 50 hrs.

ANS: (0.3 per point)



REF: CPSES Vol 1 Figure CP-35 page 1-6.75

1.2/ 5.2 What are the two basic conditions required to establish natural circulation. (1.0)

ANS:

1. Heat sink (.25) higher (.25) than heat source. (0.5)

2. Heat sink cooler (.25) than heat source. (.25) (0.5)

REF: Basic Thermodynamics

1.3/ Fill in the blanks
5.3

For a negative reactivity insertion greater than __________% $\Delta K/K$ the resulting period will always be ------- seconds. This is due to formation of neutrons with half-lives up to ------- seconds. (1.0)

ANS	0.3% $\Delta K/K$	(0.33)
	-80 seconds	(0.34)
	55 seconds	(0.33)

REF: Basic LWR Theory

1.4/ 5.4 For each of the following, choose the situation for which INDIVIDUAL rod worth will be greater. Briefly explain your choice.

a. Tave equal to 150 degrees F or 500 degrees F? (1.0)

b. Early in core life OR late in core life? (Assume ARO then 1 individual rod inserted.) (1.0)

c. Adjacent to a rodded assembly from which the rod is WITHDRAWN or INSERTED? (1.0)

ANS:

a. At 500 degrees F - at higher temperatures, the diffusion length (or spectrum hardening) is greater, allowing neutrons to reach control rods from further away and enhancin control rod effectiveness. (1.0)

b. Late in life - reduced boron concentration increases diffusion length for the same effect as higher temperature. (1.0)

c. Next to withdrawn rods - the withdrawn rod implies a higher flux making the rod in question more effective. (1.0)

REF: Basic Reactor Theory CAF

1.5 Explain what is meant by a prompt critical reactor and calculate the reactor period for such a reactor. SHOW ALL WORK AND ASSUMPTIONS FOR FULL CREDIT. (2.0)

ANS: defin: reactor critical on prompt neutrons alone (1.0)
calc: assume n life = 10EE-5 seconds +10% (given)
BETA = 0.007 +0/-0.002 (0.5)
period = n life/BETA = (0.0014) sec. (0.5)

REF: Basic LWR Theory
Verify n life and BETA @ facility.

1.6/ 5.6 Describe how and why centrifugal pump discharge flow (gpm) changes when: <u>(2.00)</u>

A. Throttling the suction valve (0.50)

B. Decreasing the fluid temperature (0.50)

C. Throttling the discharge valve, and (0.50)

D. Increasing pump speed. (0.50)

ANS:

A. Flow <u>decreases</u> due to decreased pump inlet pressure.

B. <u>No change</u> in volumetric flow rate (mass flow and pump work increase) because the pump is a volumetric device.

C. Flow <u>decreases</u> as the system resistance <u>(discharge pressure)</u> increases.

D. Flow <u>increases</u> with speed (until onset of cavitation). (0.2 change; 0.3 explanation)

REF: Basic Thermo

1.7/ 5.7 At full power, feedwater flow into each S.G. is 3.785×10^6 lbm/hr at 440° and 975 psia. Steam leaving has 99.75% quality. How much heat is added per lbm? How much total heat is transferred per S.G.? Show work. (2.5)

$Q = m\,\Delta h$

ANS:

Using the steam tables

h(sat liq at 975 psia) = 538.7 BTU/lbm (0.5)

h(sat vapor at 975 psia) = 1193.8 BTU/lbm (0.5)

h_{out} = (.0025 x 538.7) + (.9975 x 1193.8) = 1192.2 BTU/lbm (0.5)

h_{in} (subcooled water 440°, 975 psia) = 419 BTU/lbm (0.5)

$q = h_{out} - h_{in}$ = 1192.2 - 419 = <u>773.2</u> BTU/lbm (0.5)

$Q = qm = 773.2 \frac{BTU}{Lbm}\ 3.785 \times 16^6$ lbm/hr = 2927×10^6 Btu/hr

Note that $2929 \times 10^6 \frac{BTR}{hr} \cdot \frac{1\ mw}{3.412x106\ \frac{BTU}{BTU/hr}} = .858 \times 10^6$ Kw = 858 MW

and the plant has 4 S.g. 858 MW x 4 = 3432 MW which is the approximate thermal output for one unit of CPSES.

May work all 4 SGs together.

REF: CPSES Vol 5 Section XI page X1-2.27

1.8/ 5.8 The value of the core delayed neutron fraction changes between the beginning of core life and the end of core life.

a. Does this value increase or decrease? (0.5)

b. Briefly explain why the two values are different. (1.5)

ANS:

a. Decrease (0.5)

b. About 30%-45% of the fissions are caused by Pu-239 at EOL. Since the delayed neutron fraction for Pu-239 is considerably less than that for U-235, the core delayed neutron fraction decreases with an increase in burnup. (accept similar) (1.5)

REF: Basic LWR Theory

1.9/ 5.9 Which reactivity coefficient would provide a significant negative reactivity insertion during an increasing power excursion early in core life with a high RCS boron concentration? Explain. (1.0)

a. FTC (fuel temperature or doppler coefficient)

b. MTC (moderator temperature coefficient)

c. MVC (moderator void coefficient)

d. MPC (moderator pressure coefficient)

ANS:

A-MVC/MPC small, MTC LT FTC @ BOC (0.25 ans; 0.75 expl)

REF: Basic PWR Reactor Theory

1.10/ 5.10 During a routine startup the reactor is subcritical with a stable count rate of 200 CPS on all source range instruments and the shutdown groups are fully withdrawn with a Keff of 0.95. The operator withdraws control banks until the count rate is 400 CPS then stops rod motion.

a. What will be the new Keff? Show all work. (1.0)

b. What will happen if the same amount of +reactivity is added again? Show all work. (1.0)

ANS:

a. Since the count rate has doubled, the margin to criticality has been halved. Keff = 0.975. Can also be shown by using the formula $C1\ (1-K1) = C2\ (1-K2)$

b. If the same amount of reactivity is added again the margin to criticality = 0 and Keff = 1.0 (or slightly greater). The reactor is critical.

REF: Reactor Theory

1.11/ The core is operating in the nucleate boiling region.
5.11 Reactor coolant pressure is increased. What effect does this have on heat transfer at the clad/coolant interface? Explain. (1.5)

ANS: Heat transfer at the clad/coolant interface decreases (0.5)

When the pressure is increased nucleate boiling rate is decreased due to saturation temperature increasing. This reduction in nucleate boiling reduced heat transfer. (1.0)

REF: CPSES Volume IV Section XI

1.12/ 5.12 After operating in natural circulation for 2 hours, a complete loss of natural circulation flow occurs. How will the following parameters change (increase, decrease, or remain the same)? Briefly explain your answer. (assume no further operator action) (3.0)

a. Core delta T

b. Core thermocouple temperature

c. Steam generator pressure

d. Reactor coolant system pressure

ANS:

a. Increase (0.25) as boiling occurs in the core. Th (0.25) will increase while Tc (0.25) remains relatively constant.

b. Increase (0.25) boiling in the core and a lesser means available to remove the heat (0.5).

c. Decrease (0.25) less primary to secondary heat transfer (0.5).

d. Increase (0.25), as core temperature increases water exponds → pressure goes up. (0.5)

REF: Heat transfer, Thermodynamics, and Fluid Flow, General Physics. CAF

1.13/ 5.13 Explain the importance of delayed neutrons in light water reactor technology. (1.0)

ANS:

Greatly extend avg neutron generation time/allow stable control by increasing period.

REF: CPSES-Vol 1 page I-5.6

PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS

2.1 What THREE parameters are observed to determine the Reactor Core Safety Limits, during four loop operation? (1.5)

ANS:

a. Thermal power, RCS pressure and coolant temperature (0.5 ea)

REF: CPSES Technical Specifications; 2.1-1 and 6-14

2.2/ 6.2 Explain the operation of a CRDM by listing the withdrawal sequence. State the initial condition of the CRDM coils. (2.0)

ANS: Withdrawal Sequence

a. Stationary gripper coil is initially energized, others deenergized. (.50)

b. Movable gripper coil energized, gripper engages. (.25)

c. Stationary gripper deenergizes, load transfer. (.25)

d. Lift coil energized. (.25)

e. Stationary gripper coil energized, load transfer. (.25)

f. Movable gripper coil deenergized, latches withdraw. (.25)

g. Lift coil deenergized, ready for next step. (.25)

REF: CPSES Vol II page III-3.8

2.3 Explain how gamma compensation is accomplished in a compensated ion chamber. (2.0)

ANS:

The outer detector volume is sensitve to both gamma and neutron flux. (0.5)

The inner detector volume is sensitive to gamma only because it does not have a B coating. (0.5)

Compensation sums output of both volumes (subtracts inner from outer) yielding neutron proportional signal only. (1.0)

REF: Volume II CPSES pages 111-1.9 & 111-1.33

2.4 a. List the THREE chemicals added to the RCS via the CVCS to reduce corrosion. Indicate what chemistry parameter they are used to control. (2.0)

b. Define "Decontamination Factor (DF)" as it applies to the CVCS ion exchangers. (0.5)

ANS:

a. (any 3)
Hydrogen (H-2) (0.3) reduces Oxygen (0.3).
Hydrazene (N-2 H-4) (0.3) reduces Oxygen (0.3).
Lithium Hydroxide (LiOH) (0.3) pH control (0.3) (2.0)
Nitrogen (N-2)(0.3) displaces air to reduce Oxygen (0.3)

b. DF = ion exchange influent act./ion exchange effluent act. (0.5)

REF

CPSES System Information Manual; V-1.13-16

2.5/ 6.1 List the four functions of the steam dump system. (2.0)/

ANS: (0.5 ea) (1.5)

1. 50% Load Reject - Permit the plant to acc pt sudden large load decreases (up to 50%) without incurring a reactor trip or actuating the steam generator safety valves.

2. Prevent SV lift on TT/Rx trip - Remove stored energy and residual heat following a turbine/reactor trip to bring the plant to equilibrium no-load conditions without actuation of the steam generator safety valves.

3. Maintain the plant in hot standby conditions. Included in this function is acting as an artificial load for the reactor during turbine generator startup.

4. Permit a manually controlled cooldown of the plant to the point where the Residual Heat Removal System can be placed in service.

REF: CPSES Vol II page III-7.11

Note: As question 6.1, candidates are only required to answer any 3 of the 4.

2.6 a. List the automatic start signals for: (2.0)

1. MOTOR DRIVEN auxiliary feed pumps.

2. STEAM DRIVEN auxiliary feed pumps.

b. What valves are automatically shut with any of the auto start signals in "a" above? (1.0)

ANS:

a. 1. Low-low level on ONE S/G
Trip of BOTH main feedwater pumps
SI signal
Blackout signal

2. Low-low level in TWO S/G's
Blackout signal

b. Condensate Makeup/Reject valves (HV-2484 and 2485)
S/G blowdown valves
S/G sampling valves
(9 ans x 0.33 ea) (1.0)

REF:

CPSES System Information Manual; VIII-8.7 and 8.8

2.7 List the four design parameters the Rod Control System is designed to handle. (2.0)

ANS: 1. Maintain program T_{avg} within ± 3.5°F

2. ±10% step load change

3. 5% per min. ramp.

4. 50% step load reduction with steam dumps

REF: CPSES Vol II page III-3.2

2.8 Explain the advantage of controling T_{ave} on a ramped program versus a constant T_{ave} or constant steam pressure program. (2.0)

ANS: Constant T_{ave} program would cause steam pressure to drop too low for efficient operation of the secondary plant (turbine). (0.75)

Constant steam pressure program would require T_{ave} to be ramped too high from 0-100% causing problems with the primary plant, i.e. thermal limits and excessive reactivity associated with negative temperature coefficient. (0.75)

The use of a combination of both allow keeping steam pressure high without having to raise T_{ave} excessively. (0.5)

REF: CPSES Vol II page III-3.6

2.9 List three (3) basic design barriers against release of radioactive material offsite from the core. (1.0)

ANS: fuel clad (0.33ea)
RCS boundry
Containment

REF: CPSES Volume I page 11.7.2

2.10/ 6.4 The following questions pertain to the Fire Protection System at Comanche Peak. (2.0)/ (1.5)

a. Under normal operations (non-fire conditions) what pressure band is maintained on the firemain system and how is it maintained within this band?

b. Assume that a large load is placed on the firemain system such that firemain pressure is steadily decreasing. List all of the fire pumps that would automatically start and include the pressure at which they would start.

c. How are the pumps in part b above normally stopped when the pressure in the firemain begins to return to normal?

ANS:

a. Jockey pump starts @ 125 psig & stops @ 135 psig.

b. Electric fire pump starts @ 110 psig; diesel pump starts @ 90 psig, or after a preset time below 125 psig.

c. Electric pump stops @ 125 psig after a preset time; diesel pump stopped by local manual.

(5 ans @ 0.4 ea)

REF: Sys Desc Ch. IX-11, SOP 904

Note: As question 6.4; paragraph a. was not asked.

2.11 The following questions concern the Reactor Coolant Pumps (RCP's) and their support systems.

a. What conditions must be satisfied in the RCP starting circuitry to permit RCP startup? (1.0)

b. Injection water from the CVSC and primary makeup is sent to each RCP. Describe the flow path of water as it flows through the pump/seal arrangement. Include approximate individual component flow RATES and discharge collection POINTS. (2.0)

ANS: 6.04 (3.60)

a. RCP switch in start/run position (0.34)
Oil lift pump discharge pressure adequate (60 psig) (0.33)
(CAF) lift pump switch in run/start (0.33)
(locked rotor interlock accepted but <u>not</u> required)

b. 5 gpm flows past the THERMAL BARRIER and into the pump (0.3)

3 gpm flows past the RADIAL BEARING and THROUGH #1 SEAL TO THE CVSC SYSTEM (0.5)

CVSC BACKPRESSURE CAUSES 3 gph to flow THROUGH #2 SEAL TO THE RCDT (0.5)

STANDPIPE BACKPRESSURE causes flow THROUGH #3 SEAL (0.3)

400 cc/hr through the OUTER DAM SEAL TO THE CONTAINMENT SUMP (0.2) and 400 cc/hr through the INNER DAM SEAL TO THE RCDT (0.2)

(flowrates are valued at 0.1 points each) (2.0)

REF: CPSES System Information Manual;
II-1.4.13, 1.4.17, 1.4.5-11, Fig. RCP-9

2.12/ 6.7 Briefly describe the automatic <u>and</u> manual actions that take place to put the containment spray system into the recirculation mode. Include the initiating signals, valve actuations, and flowpaths. (1.5)

ANS:

1. Spray additive tank isolation valve shuts automatically on low and low-low (backup) level in spray additive tank. Also may be manually closed when going to recirc mode. (Deleted as a required answer)

2. On low-low level in RWST operator manually shift suction to containment sump. (0.75)

3. Flow is from the sump, suction of CS pump thru a CS heat exchanger, and on to the spray header (with no chemical addition.) (0.75)

REF: Sys. Des. Chap.II-9B, pgs. 2 & 9

Note: Part 1 does not apply to operation of the spray system in the recirc mode. The chemical tanks are isolated whenever the level is low regardless of whether the spray system is in recirc or not.

2.13/ 6.10

a. What signals will AUTOMATICALLY shut a Main Steam Isolation Valve (MSIV)? (Setpoints and logics are NOT required) (1.0)

b. The following are to be indicated on the attached Figure 6.1:

LIST OF COMPONENTS (1.3)

all pressure and flow transmitters.

all major valves (pneumatic, motor, and manual-gear operated - label as such).

DO NOT include small pipes and valves (drain, vent, etc.). Valve numbers are NOT required.

c. Indicate the containment boundary for both steam and feed piping. (0.2)

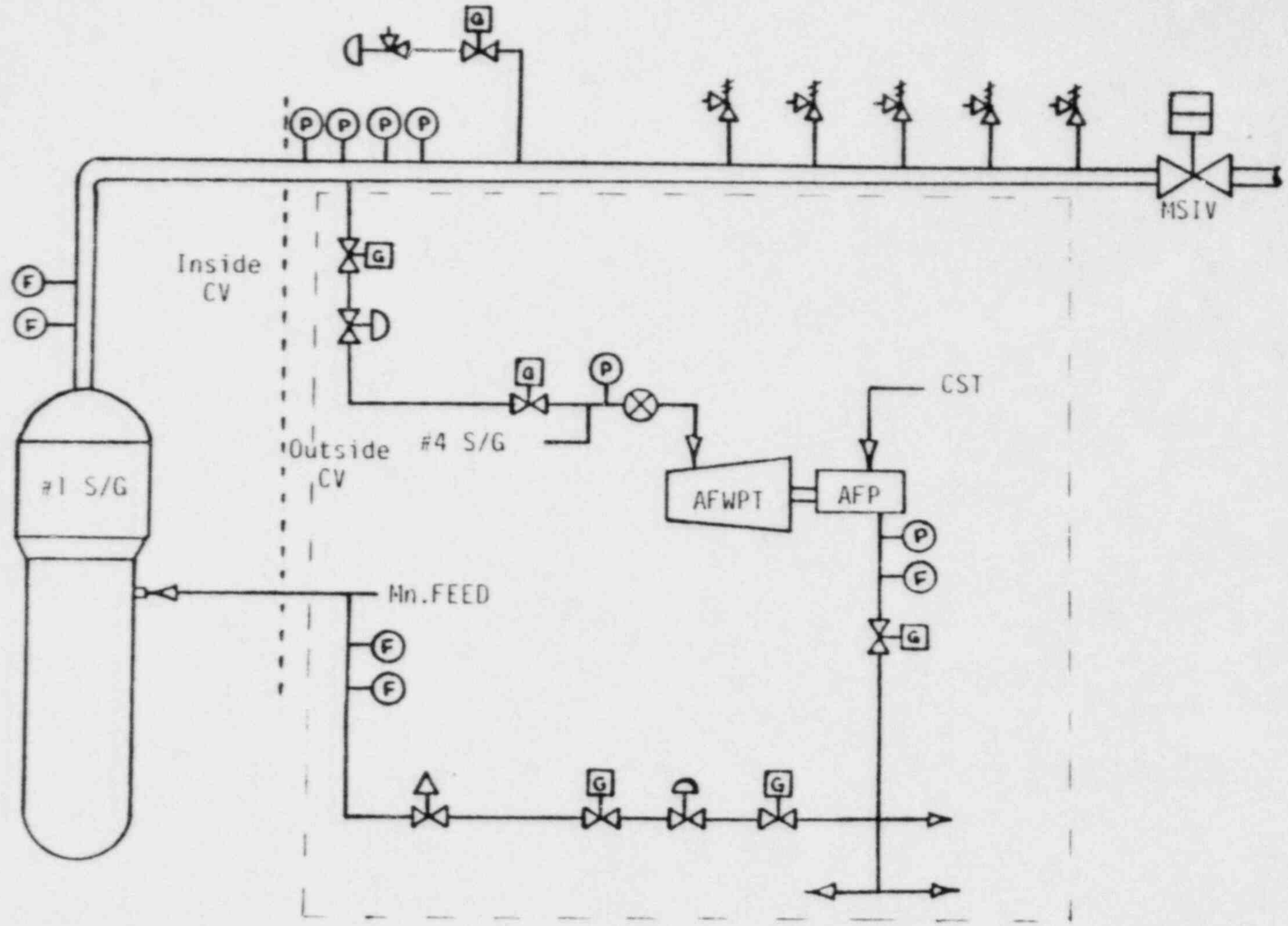
ANS:

a. Low pressure any steamline (0.25).
High steam pressure rate in any steamline (0.25), when below P-11 and SI blocked (0.25).
Hi-Hi containment pressure (0.25). (1.0)

b.&c. (On attached answer sheet for Figure 6.1)

REF: CPSES SIM: VIII-1.7 p:10 #2323-M1-0202 and 0206

U.S. NUCLEAR REGULATORY COMMISSION
OPERATOR/SENIOR OPERATOR LICENSING EXAMINATION
ANSWERS



(Please include this sheet with your answer sheets)

Q 2.13

Figure 6.1

26 items - 0.02 ea

26 labels - 0.02 ea

5 branch arrangements - 0.04 ea

INSTRUMENTS AND CONTROLS

3.1 List the signals which cause an automatic Safety Injection to occur? Include setpoints, coincidence and design basis accident. (2.0)

ANS: 1. Low pressurizer pressure--1810 psig--2/4--steam break, LOCA

2. Low steamline pressure--$\geq$605 psig--2/3 in any steam line--steam break (1)

3. High--1 Containment pressure--$\leq$3.7 psig--2/3--LOCA (1)

(Signal @ 0.2 ea; St Pt @ 0.1 ea; coincidence @ 0.2 ea; DBA @ 0.16 ea.)

REF: CPSES Vol II pages III-10.5, 6 & 7., T.S. 3.3-4, pg 3/4.3-25

3.2 a. On a 10% step load reduction from 100% power, steam generator level is observed to decrease 4%. Explain how the feedwater control system reduces feedwater flow when the level decrease is telling it to increase feedwater flow. Address answer to feedwater control valve response. (1.0)

b. How would the feedwater control system respond if pressure compensation to the steam flow signal were lost (failed low) at 100% power. (1.0)

ANS:

a. The feedwater control system is a "3-element" system using feed flow, steam flow and a level signal (filtered network to dampen oscillations) as inputs. On a load reduction the decreased steam flow signal causes a flow error signal to close down the feed control valve while the decrease in level is delayed by the lag (filter network) circuit, thereby giving an increase flow signal.

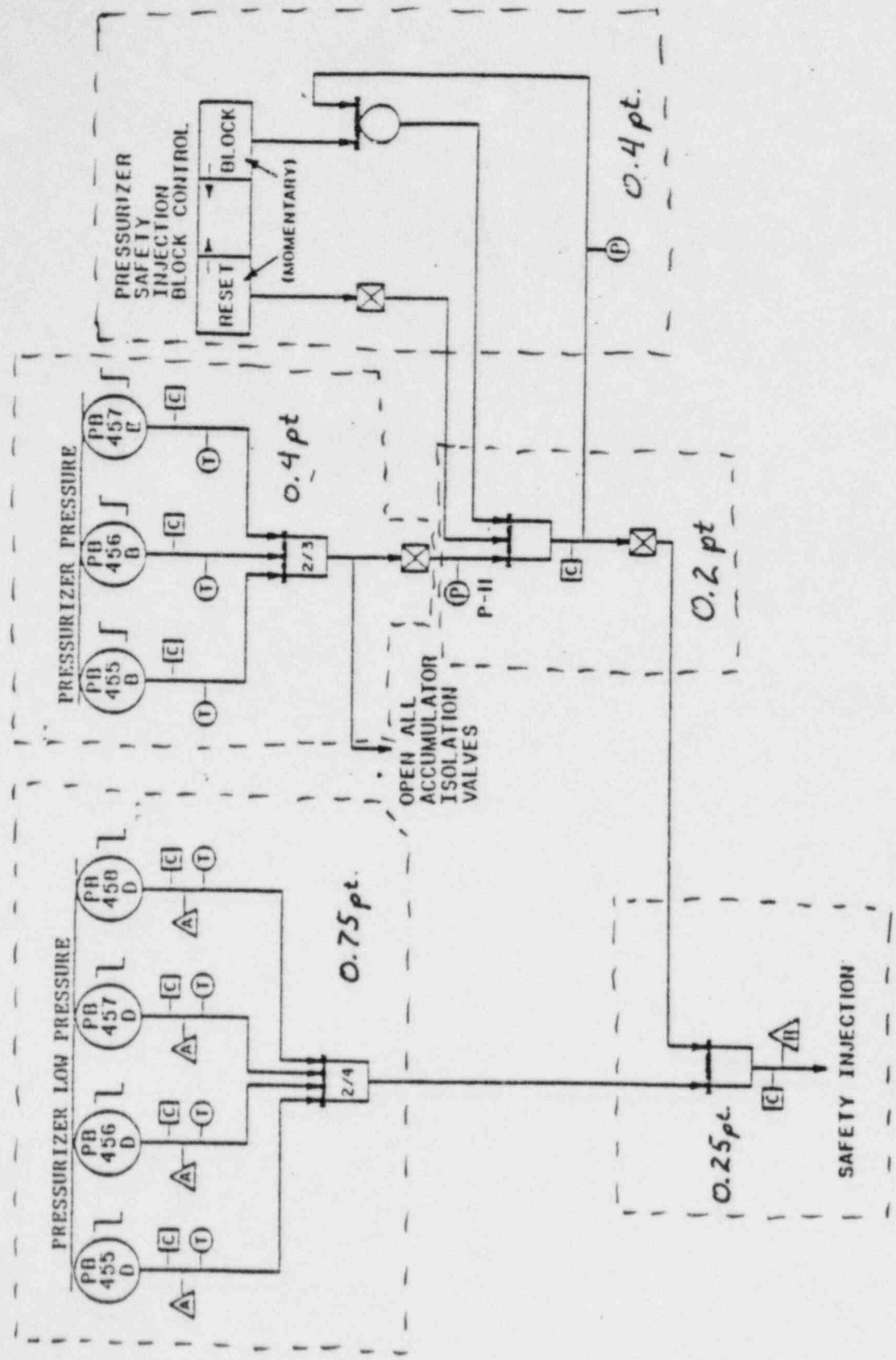
b. Pressure compensation to the steam flow signal adjusts the lbs/hr flow rate caused by density changes. When the pressure signal goes to zero, the decreased steam density indication causes the steam flow signal to be reduced. This would cause an accompanying decrease in feed flow until the level error signal compensates to cause feed to flow to increase (open control valve more).

REF: CPSES, Nuclear Tng, Systems Manual, Vol III-Section III-8.

3.3/ 6.9 Sketch a diagram of the Pressurizer Pressure Safety Injection logic. (2.0)

ANS: see attached diagram

REF: CPSES Training Manual III-10.27


PRESSURIZER LOW PRESSURE
PB 455 D
PB 456 D
PB 457 D
PB 458 D
2/4
0.75 pt.
PRESSURIZER PRESSURE
PB 455 B
PB 456 B
PB 457 B
2/3
0.4 pt
OPEN ALL ACCUMULATOR ISOLATION VALVES
P-11
0.2 pt
PRESSURIZER SAFETY INJECTION BLOCK CONTROL
RESET
BLOCK
(MOMENTARY)
0.4 pt.
SAFETY INJECTION
0.25 pt.

-Q3.3

3.4/ 6.12 While at 100% power, the pressure instrument selected for pressurizer pressure control fails high. With no operator action, describe the control system response and the plant transient that occurs. Continue to steady state conditions. (3.0)

ANS: At the instant of instrument failure, one PORV opens, spray valves open and all heaters are deenergized. When actual pressure decreases to 2185 psig the PORV will be shut by an interlock signal from one of the remaining pressure instruments. However sprays will remain on and heater off. When pressure reaches 1910 a reactor trip occurs. At 1810 SIAS is received which will start all charging pumps. The RCS will eventually be charged solid and the PORVs (and possibly SVs) will open to stabilize pressure.

(10 undl. resp. @ 0.3 ea.)

REF: CPSES Training Manual section III-6.5
III-10.6, III-9 table RPS-1, IV-3.13

3.5/ 6.11 List and describe the "P" type permissives which receive <u>direct</u> input from the excore nuclear instrument system. (3.0)

ANS:

<u>P-6</u> permits <u>bypass</u> of <u>SR</u> high flux at <u>10^5 counts/sec</u> (IR input at 10^{-10}amp)

<u>P-8</u> permits continued reactor operation while <u><48%</u> power if a <u>single loop is lost</u> (PR input at 48%)

<u>P-10</u> permits blocking of the <u>IR and PR low range high power trips</u> (25%) (PR input at 10%)

Note: P-7 is NIS power dependent but gets its input from P-10 <u>not</u> the NIS directly

REF: CPSES Training Manual Section III-9

3.6 True or False: If power increases from 1×10^{-6}% to 2.2×10^{-6}% in 5 minutes a signal will be generated which, if not manually blocked, will cause the charging pump suction to shift from the VCT to the RWST. (0.5)

ANS:

True: (1×10^{-6}% to 2.2×10^{-6}% is equivalent to a SR power change of 1×10^{3}CPS to 2.2×10^{3}, a factor of 2.2. This meets the Flux Doubling shutdown criteria of $\geq$2x in $\leq$min.)

REF: CPSES Training Manual section III-1, figure EXC-4

3.7 During "normal" power operation state whether the four solenoid valves that control dump operation (labeled 1-4 on Figure 3-2 attached) are energized or de-energized. Answer for each valve separately. (1.0)

ANS:

1. EN
2. EN
3. DE
4. DE (0.25 ea)

REV: CPSES, Nuclear Training, Vol. 2 Steam Dump System, p.III-7.8, III-7.16

Steam Dump Solenoid Valve Operation

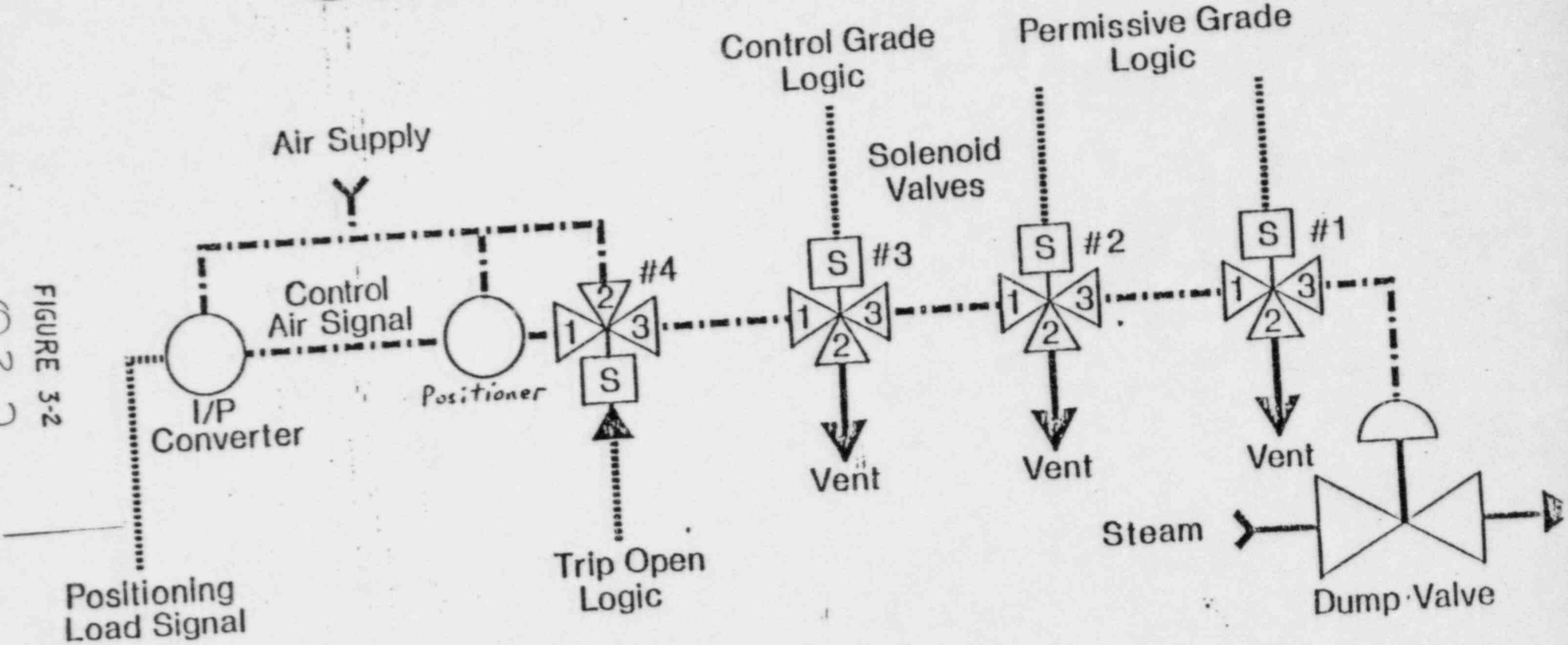


FIGURE 3-2

Q 3.7

1. Solenoids #1,2,3
   - When Deenergized, Port 1 Blocked, Port 2 & 3 Connected
   - When Energized, Port 2 Blocked, Port 1 & 3 Connected

2. Solenoid #4
   - When Deenergized, Port 2 Blocked, Port 1 & 3 Connected
   - When Energized, Port 1 Blocked, Port 2 & 3 Connected

00-8110

3.8/ 6.6 List the channels of excore nuclear instrumentation and indicate the number and type of detector used and the indicating range of each channel. (3.0)

ANS: (.25 pt per item)

channel	source range	intermediate range	power range
# of detector	2	2	16
type	BF_3	CIC	UIC
range	$1\text{-}10^6$CPS	$10^{-11}\text{-}10^{-3}$amp	0-120%or 0-200%

REF: CPSES Training Manual Section III-1

3.9 What would be the immediate symptoms if the selected pressurizer level instrument for level control failed low? (1.0)

ANS: (.033 ea)

charging flow increase
heater turn off (in auto)
letdown isolated

REF: S.D. IV-3.12, Vol II

3.10 What is the function the following incore operation selector switch positions. (a) off, (b) calibrate, (c) emergency, (d) common group, and (e) storage. (2.5)

ANS:

a) Off. When the operation selector switch is placed in the OFF position the five path transfer device is positioned to its normal ten path rotary device. Positioning the operation selector switch in the "OFF" position also prevents movement of the detector. (0.5ea)

b) Calibrate. When the operation selector switch is placed in the Calibrate position the five path rotary device is positioned to the Wye-units for the calibrate path (bypassing all ten path devices).

c) Emergency. When the operation selector switch is placed in the Emergency position the five path rotary device is positioned to the next sequentially lettered ten path rotary device. (Drive A to ten path rotary device C, etc.).

d) Common Group. When the operation selector switch is placed in the Common Group position the five path rotary device is positioned to ten path rotary device "C".

e) Storage. When the operation selector switch is placed in the Storage position the five path rotary device is positioned to shielded storage area, located in the seal table room.

REF: CPSES page 3.-2.8

3.11 What is indicated by the following alarms. (a) Rod Control Urgent Failure (b) Rod Control non-urgent failure (c) Rod Drive Generator Breaker 1A or 1B tripped (d) Rod Control System Grounded. (2.0)

ANS:

a. ROD CONTROL URGENT FAILURE - Indicates that an internal failure has affected the ability of the system to move rods.

b. ROD CONTROL NON-URGENT FAILURE - Indicates that a redundant power supply has failed.

c. ROD DRIVE GENERATOR BREAKER 1A or 1B TRIPPED - Indicates the loss of one or both of the motor-generator set power sources.

d. ROD CONTROL SYSTEM GROUNDED - Indicates a phase ground fault on the (260-volt three phase) supply circuit.

REF: CPSES pages III-3.17 & 18

3.12/ 6.3 Describe how the reactor plant would respond to a Loop 1 Tc detector failed "high". Include response of all affected control systems (primary and secondary). Assume no operator action or reactor trip and the plant is at 75% power. Base answer on initial trend only. (3.0)

ANS:

Tc failed hi causes Tave failed hi.

ROD CONTROL: hi Tave causes rods in @ 72 spm which cause actual Tave decrease

PRZ PRESS CONT: decr. Tave causes decr Ppzr which causes spray valves to close & heaters energized

PRZ LEVEL CONT: decr Tave causes decr Lpzr also Lstpt incr to 60% (program level) so chg flow increases until Lpzr @ setpoint

STEAM DUMP CONT: hi Tave failure causes maximum output on both controllers-no dump until armed.

(0.75 ea CONTROL SYSTEM)

REF: CPSES Nuc Tr Man III-7, IV-3

PROCEDURES - NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL

4.1 Give the radiation dose rate limits for a Radiation Area and High Radiation Area. (1.0)

ANS: RA-GT 2.5 mr/hr or 5 mrem/hr or
HRA-GT 100 mr/hr 100 mrem in 5 consecutive days

2 ans x 0.5 ea

REF: CP RAD WORKER LP,p.24, 10 CFR 20.202(b)(2)

4.2/ 7.5 Give four (4) of the five (5) Reactor Vessel Vent Termination Criteria presented in FRI-0.3, "Response to Voids in Reactor Vessel". (2.0)

ANS:
1. Containment hydrogen conc GT 3% by VOLUME
2. RCS subcoolong LT 60 F
3. Pressurizer level LT 20%
4. RCS pressure decreases by 200 psi
5. Vent period time achieved

Any 4/5 @ 0.5 ea.

REF: CP PROC FRI-0.3, p.6

4.3/ Give four (4) of the five (5) steps required to initiate
7.4 EMERGENCY BORATION. (2.0)

ANS:
(based on interpretation that steps = operations = actions)
(any 4 of 6 at 0.5 ea)

1. Open emergency boration valve (1/1-8104)
2. Start <u>BOTH</u> boric acid transfer pumps (1/1-APBA1 & 1/1-APBA2)
3. Verify PD chg. pump running
4. Adjust letdown flow to 120 gpm
5. Verify boron addition per emergency borate flow indicator (1-FI-183)
6. Monitor the following parameters to determine adequate boration:
   Control rod bank position
   Tave
   Neutron flux level
   Emergency boration flow rate

(based on the interpretation that steps = conditions = reasons for)
(any 4 of 5 at 0.5 ea)
1. Failure of any rod control cluster assembly (RCCA) to fully insert following a reactor trip or shutdown.
2. Control rod height below the insertion limit.
3. Failure of the Reactor Makeup Control System to the extent that bypass is necessary to accomplish boration.
4. Uncontrolled Reactor Coolant System cooldown following a reactor trip or shutdown not requiring safeguards system actuation.
5. Unexplained or uncontrolled reactivity increase as indicated by abnormal control bank insertion, increasing temperature or nuclear power level or, increasing neutron flux level during shutdown.

Reference:

Case 1 FRS-0.1 pgs 2&3, ECA 1.0 pg 5
Case 2 CPSES SD II-3 pg II-3.15

(The question may be graded on EITHER case 1 or case 2 BUT NOT BOTH. This is the result of apparent contradiction in question explanation conveyed to different candidates).

4.4 FILL IN THE BLANKS

Fill in the blanks in the following sentences dealing with restarting a Reactor Coolant Pump (RCP).

a. After any period of running, a restart should not be made until the motor has been allowed to cool by ____________ for at least ________ minutes.

b. Within any __________ hour period, the number of restarts should be limited to ________ with a minimum _________ of _________ minutes prior to each restart.

c. The seal injection water must be maintained below _____ F.

d. Starts should not average more than ______ per day for the life of the ___________.

e. The RCP must be stopped when upper or lower bearing temperature has increased to _____ F.

f. The component cooling water to the RCP's must be maintained at or below _____ F. (2.5)

ANS: a. standing idle,30
b. 2,3,idle period,30
c. 130
d. 6,RCP <u>MOTOR</u>
e. 200
f. 105

8 # ans @ 0.2: 3 word ans @ 0.3: total 2.5

REF: CP PROC FRC-0.1, p.13 (Attach. 1)

4.5 List four (4) of the five (5) conditions required to establish CONTAINMENT INTEGRITY per technical specifications. (2.0)

ANS:

a. All penetrations required to be closed during accident conditions.
   1. Capable of being closed by an operable contain. auto. isol. valve.
   2. Closed by man. valve, blind flange, etc.

b. All equipment hatches closed and sealed.

c. Each airlock operable per TS.

d. Containment leak rates w/i TS.

e. Sealing mech. associated w/ ea. penet. OPERABLE.

any 4/5 ans @ 0.5 ea.

REF: CP TECH SPECS 1.7, p.1-2

4.6/ 8.2 Which of the following items are addressed by Technical Specifications? Answer YES if the items is or NO if the item is not addressed in the Technical Specifications.

a. Primary containment humidity,
b. Fuel oil storage quanity,
c. Steam dump control system,
d. Condensate storage tank chemistry,
e. Pressurizer heaters,
f. Plant Monitoring Computer,
g. Offsite power feeds, and
h. Incore muclear instrumentation. (2.0)

ANS:

a.	NO	e.	YES
b.	YES	f.	NO
c.	NO	g.	YES
d.	NO	h.	YES

REF: CP TECH SPECS,
b. 3.8.1.1, p. 3/4 8-1
e. 3.4.3, p. 3/4 4-9
g. 3.8.1.1, p. 3/4 8-1
h. 3.3.3.2, p.3/4 3-48

4.7/ 7.2 List the AFFECTED SYSTEM and whether it would be considered OPERABLE or INOPERABLE per Technical Specifications (i.e., entered an action statement or LCO) for each of the following conditions. Assume, unless otherwise specified, unit is in MODE 1.

a. Both boric acid tanks unavailable,
b. Refueling Water Storage Tank borated water volume equals 497,800 gallons,
c. One shutdown rod inserted 10 steps,
d. One pressurizer level instrument leaking at the level transmitter. Leakage is within RCS technical specification identified and unidentified limits. Level channel indicating correctly,
e. Both PORV's inoperable in MODE 5,
f. One RCS accumulator has a nitrogen cover pressure of 610 psia,
g. Both centrifugal charging pumps operable with Tc=290° F,
h. Primary containment pressure at 13.0 psia in MODE 5, and
i. Water level of the Squaw Creek Reservoir at 780 feet Mean Sea Level USGS datum. (3.0)

ANS: a. Boration Sys/inop; b. RWST/OP, c. RODS/INOP
d. RCS PB/PZR LVL/OP; e. PORV/LTPO/INOP; f. SIT/ACC/INOP
g. ECCS/LTCP/INOP; h. CONTAIN/OP; i. UHS/FLOOD PROT/INOP

(0.166 ea item)

REF: CP TECH SPECS
a. 3.1.2.1, p. 3/4 1-7; b. 3.1.2.6.b, p. 3/4 1_12
c. 3.1.3.5, p. 3/4 1-20; d. 3.3.3.10, p. 3.4 3-77
e. 3.4.9.3, p. 3/4 4-35 f. 3.5.1.d, p. 3/4 5-1
g. 3.5.3.a, p. 3/4 5-7; h. 3.6.1.5, p. 3/4 6-7
i. 3.7.6, p. 3/4 7-14

4.8/ 7.3 Give the name, reactivity limit, rated power limit, and average coolant temperature limit for the OPERATIONAL MODES listed below.

a. MODE 1
b. MODE 2
c. MODE 3
d. MODE 4
e. MODE 5
f. MODE 6 (3.0)

ANS:

a. Power Operation;GT/EQ 0.99;GT 5%;GT/EQ 350 F
b. Startup;GT/EQ 0.99;LT/EQ 5%;GT/EQ 350 F
c. Hot Standby;LT 0.99;0;GT/EQ 350 F
d. Hot Shutdown;LT 0.99;0;350 GT Tavg GT 200 F
e. Cold Shutdown;LT 0.99;0;LT/EQ 200 F
f. Refueling;LT/EQ 0.95;0;LT/EQ 140 F

6 ans @ 0.5 ea.; for ea. part a-f 4 ans @ 0.125 ea.

REF: CP TECH SPECS Table 1.2, p. 1-9

4.9 A. If a fuel failure is confirmed what actions are to be taken prior to referring to the Emergency action plan and Tech Specs? (1.0)

B. What other plant conditions or events may result in a gross failed fuel monitor (IRE-406) "alert" besides an actual failed fuel condition? (1.0)

ANS:

A.
1. Increase letdown to 120 gpm.
2. Increase charging as needed to maintain pzr level.
3. Notify Rad Protection of possible increased radiation levels in aux, fuel, and safeguards buildings.

b.
1. Monitor/detector malfunction.
2. Exhausted CVCS mixed bed ion exchanger. (low OF)
3. Crud burst.

REF: CPSES ABN-102A

4.10 For the symptoms given below identify the event which has occurred and give the immediate operator action(s) per emergency response guidelines. Assume SI has NOT occurred.

a. Rapid decrease in neutron flux level.
b. Reactor trip breakers open.
c. Rapid decrease in unit load to 0 power,

NOTE: Other symptoms may also be present. (2.0)

ANS Event: Reactor/Turbine trip (0.3)
IA's: Verify rx trip
-all rod bottom light LIT
-all rod posit. ind. ZERO
-N flux decr.
Verify TT
-all stop valves CLOSED
Verify AC emergency buses energized
-AC emer. bus volt. NORMAL

4 major @ 0.4 ea. 5 minor @ 0.1 ea.

REF: CPSES PROC EOP - 0.0, p. 2-3

4.11/ 7.7 Explain the difference in the purpose of a GENERAL ACCESS PERMIT (GAP) and a RADIATION WORK PERMIT (RWP). (2.0)

ANS: A GAP is a general-area type document that allows access to an area where the rad hazard is slight and slow to change. A RWP, on the other hand, is to allow access to a specific area for a specific task (or series of tasks) where hazards are significant. An RWP also allows specific accounting of exact exposure.

1.0 for GAP,1.0 for RWP

REF: CP RAD WORKER LP,p.8-9

4.12/ 7.11 Provide a list of items in the proper order for donning single layer ANTI-Cs (1.5)

a. rubber shoe covers
b. cotton glove liners
c. cloth overalls
d. required head cover
e. PVC shoe covers
f. rubber gloves
g. dosimetry

ANS: e,c,a,d,b,f,g

7 ans x 0.214 ea.

REF: CP RAD WORKER LP,p. 33-34

6.5 Injection water from the CVCS and primary makeup is sent to each RCP. Describe the flow path of water as it flows through the pump/seal arrangement. Include approximate individual component flow RATES and discharge collection POINTS. (2.0)

ANS: 6.04 (3.60)

5 gpm flows past the THERMAL BARRIER and into the pump (0.3)

3 gpm flows past the RADIAL BEARING and THROUGH #1 SEAL TO THE CVSC SYSTEM (0.5)

CVSC BACKPRESSURE CAUSES 3 gph to flow THROUGH #2 SEAL TO THE RCDT (0.5)

STANDPIPE BACKPRESSURE causes flow THROUGH #3 SEAL (0.3)

400 cc/hr through the OUTER DAM SEAL TO THE CONTAINMENT SUMP (0.2) and 400 cc/hr through the INNER DAM SEAL TO THE RCDT (0.2)

(flowrates are valued at 0.1 points each) (2.0)

REF: CPSES System Information Manual;
II-1.4.13, 1.4.17, 1.4.5-11, Fig. RCP-9

6.8 On a 10% step load reduction from 100% power, steam generator level is observed to decrease 4%. Explain how the feedwater control system reduces feedwater flow when the level decrease is telling it to increase feedwater flow. Address answer to feedwater control valve response. (1.0)

ANS:

The feedwater control system is a "3-element" system using feed flow, steam flow and a level signal (filtered network to dampen oscillations) as inputs. On a load reduction the decreased steam flow signal causes a flow error signal to close down the feed control valve while the decrease in level is delayed by the lag (filter network) circuit, thereby giving an increase flow signal.

REF: CPSES, Nuclear Tng, Systems Manual, Vol III-Section III-8.

7.1 a. What is the maximum permissible whole body occupational dose for an individual in a restricted area in accordance with 10CFR20? (0.5)

b. What are the 10 CFR 20 requirements if an individual is permitted to receive a whole body dose greater than that permissible in question a. above? (1.5)

c. What is the Comanche Peak Administrative whole body exposure limit AND how can this limit be raised? (1.0)

ANS:

a. 1.25 REM/quarter (0.5)

b.
1. During any calander quarter exposure shall not exceed 3 REM.
2. Accumulated dose shall not exceed 5(N-18).
3. Must have dose record recorded on Form NRC-4. (1.5)

c. 300 mrem/wk (.167 ea)
1000 mrem/qtr
3000 mrem/yr

or
1 rem/qtr raised to 2.5 rem/qtr (.25 ea)

or

3 rem/yr raised to 5 rem/yr (.25 ea)

Approval is from the Radiation Protection Engineer (0.5). (1.0)

REF:

CPSES Radiation Protection Lesson Plan; pp 11-13

7.5 Give four (4) of the five (5) Reactor Vessel Vent Termination Criteria presented in FRI-0.3, "Response to Voids in Reactor Vessel". (2.0)

ANS:
1. Containment hydrogen conc GT 3% by VOLUME
2. RCS subcoolong LT 60 F
3. Pressurizer level LT 20%
4. RCS pressure decreases by 200 psi
5. Vent period time achieved

Any 4/5 @ 0.5 ea.

REF: CP PROC FRI-0.3, p.6

7.6 List the three (3) basic factors that determine external radiation exposure and explain how exposure varies with each of them (assume a point source of gamma radiation) (3.0)

ANS: 1. TIME-exp directly proportional to time exposed
2. DISTANCE-exp falls off as the square of the distance
3. SHIELDING-exp falls off exponentially with increased shielding

1.0 ans x 3 ans 0.25 factor,0.75 expl.

REF: CP RAD WORKER LP,p.14-17

7.8 List four (4) components or systems which have parameters which must be verified to be within Technical Specification limits weekly per procedure OPT-104A, "OPERATIONS WEEKLY ROUTINE TESTS," while operating in MODE 1. (1.0)

ANS:

1. Boric Acid Tanks
2. Refueling Water Storage Tank
3. Spent Fuel Pool
4. AFD Monitor Alarm
5. Quad Power Tile Alarm
6. Turbine Overspeed Protection System
7. AC Power Sources
8. DC Power System
9. Locked Closed Fire Doors
10. Plant Vent Iodine Sampler
11. Plant Vent Particulate Sampler
12 Containment Press. Relief Ex. Isol. Valve

any 4 @ 0.25 ea.

REF: CPSES PROC OPT-104A, p. 5-6

7.9 The following alarms are received in the control room:

PRZR LVL DEV LO
PRZR LVL LO
ANY RCP SEAL WTR INJ FLO LO
VCT LVL HI
(Other related alarms may exist but are not necessary to identify the problem.)

A. What is the nature of the problem and what system of component is malfunctioning? (1.0)

B. What are your initial actions? Include all options. (1.5)

ANS:

A. Inadequate charging flow (.5) from CVCS (.5)

B. Restore PZR level (.5) by
   1. Taking manual level control and increase charging or
   2. Isolate letdown or
   3. Start additional charging pumps or
   4. Manually initiate SI (last resort)

REF: CPSES ABN 101A, 105A 1-ALB-5A 1.6

7.10 Give the immediate operator actions from the control room for an Anticipated Transient Without Trip (2.0)

ANS:

1. Trip reactor manually
2. Trip turbine manually
3. Check AFW running
   - -motor driven pump bkr. ind. lites LIT
   - -turbine driven pump steam supply valves OPEN
   - -verify AFW valve alignment
4. Initiate rapid boration of RCS
   - -open EMER BORATE VLV (1/1-APBA 1&2)
   - -start both BA XFER PUMPS (1/1-APBA 1&2)
   - -verify PD CHG PUMP running
   - -adjust LTDN to 120 GPM
   - -observe B_{10} additional/flow by EMER BORAT FLOW (1-FI-183)
5. Verify containment ventilation isolation

5 major @ 0.4 ea. minor not req'd

REF: CPSES PROC ECA - 1.0, p. 3-5

ADMINISTRATIVE PROCEDURES, CONDITIONS AND LIMITATIONS

8.1 a. What immediate actions must be taken per Technical Specifications if a Safety Limit violation occurs during power operation? (1.0)

b. What additional requirements are imposed by Technical Specifications for a Safety Limit violation? (1.5)

ANS:

a. 1. S/D w/in 1 hr
   2. Notify NRC ASAP (1 hour)

b. 1. Notify Mgr, Nuc Ops and ORC (24 hrs)
   2. Prepare & submit SLVR to NRC (SORC, ORC & Mgr, Nuc Ops) (14 days)
   3. NRC permission for critical ops

REF: CPSES TS, 2.1, p. 2-1; 6.7.7, p. 6-13

8.3 a. Who is responsible (position/title) for the preparation of Clearance Reports and Danger Tags? (0.5)

b. What is required if more than one task is to be performed on a particular component by different work groups and WHY is this necessary? (1.0)

c. Which personnel are authorized (by position/title) to attach Danger Tag stickers to the Main Control Board? (1.0)

d. 1. Under what condition can the second independent verification be omitted for a Clearance on a safety-related system, subsystem, or component? (0.5)

2. When would a second verification be required for a Clearance on a NON safety-related system or component? (0.5)

ANS:

a. Relief Reactor Operator (0.5)

b. A separate Clearance Report and Danger Tag shall be completed for each work group (0.5) to ensure the safety of each group (0.5) (1.0)

c. Shift Supervisor (SS); Assistant SS; Reactor Operator (1.0)

d. 1. Only in cases of significant radiation exposure. (0.5)

2. If the function, or loss of function if the system or component affects the safe operation of the plant. (0.5)

REF:

CPSES Station Administrative Manual; STA-605, pp 2-8

8.4 As a result of an administrative error, a monthly surveillance test was performed 35 days after the last surveillance rather than 30 days. Does this error constitute noncompliance with Technical Specifications?

EXPLAIN. (Assume previous surveillances have been completed once every thirty days.) (1.0)

ANS:

NO-- Meets 1.25x & 3.25x TS limits)

(1.0)

REF:

CPSES Technical Specifications; pp 3/4 0-2

8.5 a. If the Reactor Operator is to be "temporarily" relieved during the shift, who (by position/title) can relieve the Reactor Operator AND what must be done before the position can be assumed? (1.0)

b. What is required before an Auxiliary Operator may assume more than one watch station? (1.0)

c. What are the THREE plant conditions when two (an additional) licensed Reactor Operators should be in the control room for Unit 1 according to ODA-102 "Shift Compliment Responsibilities and Authorities"? (1.0)

ANS:

a. The Relief Reactor Operator (0.5) shall completely review the Reactor Operator checklist (0.5) before assuming the new position. (1.0)

b. The Auxiliary Operator must be qualified in the watchstation (0.5) and have Shift Supervisor permission (0.5) (1.0)

c. 1. During a startup
   2. Reactor trip recovery
   3. Scheduled shutdown (1.0)

REF:

CPSES Operation Department Admin.strative Procedures;
ODA-302,p 4
ODA-102,p 9

8.6 List four (4) of the five (5) conditions required to establish <u>CONTAINMENT</u> <u>INTEGRITY</u> per technical specifications. (2.0)

ANS:

a. All penetrations required to be closed during accident conditions.
   1. Capable of being closed by an operable contain. auto. isol. valve.
   2. Closed by man. valve, blind flange, etc.

b. <u>All</u> equipment hatches closed and sealed.

c. Each airlock operable per TS.

d. Containment leak rates w/i TS.

e. Sealing mech. associated w/ ea. penet. <u>OPERABLE</u>.

any 4/5 ans @ 0.5 ea.

REF: CP TECH SPECS 1.7, p.1-2

8.7 FILL IN THE BLANKS

Fill in the blanks in the following sentences dealing with restarting a Reactor Coolant Pump (RCP).

a. After any period of running, a restart should not be made until the motor has been allowed to cool by ______________ for at least __________ minutes.

b. Within any ___________ hour period, the number of restarts should be limited to _________ with a minimum __________ of __________ minutes prior to each restart.

c. The seal injection water must be maintained below _____ F.

d. Starts should not average more than _______ per day for the life of the _____________.

e. The RCP must be stopped when upper or lower bearing temperature has increased to ______ F.

f. The component cooling water to the RCP's must be maintained at or below ______ F. (2.5)

ANS: a. standing idle,30
b. 2,3,idle period,30
c. 130
d. 6,RCP <u>MOTOR</u>
e. 200
f. 105

8 # ans @ 0.2: 3 word ans @ 0.3: total 2.5

REF: CP PROC FRC-0.1, p.13 (Attach. 1)

8.8 List the four (4) Emergency Action Levels in order of increasing severity and classify each of the events below to the appropriate level per procedure EPP-201.

a. RCS leak rate exceeding Technical Specification limits,
b. 400 gpm steam generator tube failure with loss of offsite power,
c. Large steam line break (rapid depressurization),
d. Unisolable steam line break outside containment with 100 gpm steam generator tube leakage and high primary system radiation indicating fuel failure,
e. Loss of offsite power, and
f. Loss of both onsite and offsite AC power. (3.5)

ANS: 4 EAL's: Notice of Unusual Event
Alert
Site Area Emergency
General Emergency

a. NOUE
b. SAE
c. NOUE
d. GE
e. NOUE
f. A

EAL's 4 ans @ 0.125 ea.; a-f 6 ans @ 0.5 ea.
(for a-f accept 1 level more severe for 1/2 credit)

REF: CP PROC EPP-201, p. 2,3,8,9,10,12

8.9 What are the maximum time limits permitted after recognition of an Emergency Condition prior to initial notification of State and local authorities and the NRC Incident Response Center? (1.0)

ANS: 15 minutes State and local; 60 minutes NRC-IRC
0.5 ea. time x 2 times

REF: CP PROC EPP-203, p. 2

8.10 Explain the two types of temporary changes to procedures and give the four (4) conditions necessary to made a temporary change to an approved procedure per STA-205, "Temporary Changes to Procedures". (2.5)

ANS:

TYPES:
1. One use only (1 time)
2. Limited time, multiple use (extended)

CONDITIONS:
1. No change in intent of current procedure
2. Approved by 2 members of station management knowledgeable in the area affected by the change. One must have SRO license
3. Change is documented, reviewed by SORC, & approved by MGR, Plant Ops w/i 14 calendar days of implementation
4. Change does not lessen any QA req'm'nts in current procedure

TYPES: 2 @ 0.25 ea.; CONDITIONS: 4 @ 0.5 ea.

REF: CP PROC STA-205, p. 2,3

8.11 Who must approve a SPECIAL ORDER prior to issuance? (0.5)

ANS: Manager, Plant Operations

0.5 pts

REF: CP PROC STA-207, p. 3

8.12 Define REPORTABLE OCCURRENCE per CPSES procedures. (1.0)

ANS: An unscheduled event or incident which is significant from the standpoint of public health and safety

0.5 for ea. underlined segment

REF: CP PROC STA-501, p. 2