U. S. NUCLEAR REGULATORY COMMISSION

REGION V

Examination Report No. 50-275/OL-85-01

Docket Nos. 50-275, 50-323

Facility: Diablo Canyon 1 and 2

Examinations Administered at: Avila Beach, California

Chief Examiner: Gary W. Johnston 2-1-85
G. W. Johnston, Operator Licensing Examiner Date Signed

Approved: J. O. Elin 2/1/85
J. O. Elin, Acting Chief, Operations Section Date Signed

Summary:

Examinations on January 14-18, 1985 (Report No. 50-275/OL-85-01)

Four written and operating examinations were administered to one SRO candidate, and three instructor candidates. The SRO candidate and two of the instructor candidates passed the written and operating exams.

REPORT DETAILS

1. Persons Examined:

   SRO Candidate:

   Barkley, D. D.

   Instructor Certificates:

   Sawyer, L. R.
   Terrell, B. E.
   Welsch, J. M.

2. Examiners:

   *G. W. Johnston, RV
   A. J. Vinnola Jr., EG&G INEL

   *Lead Examiner

3. Persons Attending the Exit Meeting:

   T. Martin, PG&E
   J. Molden, PG&E
   A. Vinnola, EG&G INEL
   G. Johnston, RV

4. Written Examination and Facility Review:

   Written exams were administered as follows:

   3 Instructor Certification - January 14, 1985
   1 SRO exam - January 14, 1985

   At the conclusion of the exam the facility staff reviewed the exam and answer key. The facility staff comments are addressed in the enclosed Attachment (1). These comments were discussed with the facility staff and appropriate revisions to the master examination key were made by the lead examiner prior to grading the candidate responses.

5. Operating Examinations:

   Simulator exams and facility walkthroughs were conducted January 15-18, 1985. The examiners identified one general weakness; the candidates had difficulty aligning power to a dead bus after a loss of offsite power event occurred during the simulator phase of the operating examination. It is recommended that additional instruction in this area be conducted. This was not sufficient to justify failure of any of the candidates.

6. Exit Meeting:

On January 18, 1985, the lead examiner met with licensee representatives. Those individual candidates who clearly passed the operating exam were identified. Also, the weakness described in paragraph 5 was discussed.

FACILITY COMMENTS ON DIABLO CANYON SRO EXAM

JANUARY 15, 1984

1.0 Facility Comments:

5.1 (question): "We don't teach re-distribution, therefore my answer to the questions would be:

(1) Since at zero power there is no difference in temp (inlet/outlet) there is no difference in the reactivity added axially, when power increases to HFP the lower section of the core is colder thus adding more positive reactivity."

Resolution:

The reference material provided to the examiners discussed the effect, since the material was provided on the basis that it reflected what was taught to the candidates the examiner determined that the question asked is valid.

2.0 Facility Comments:

5.7 (key): "Another possible answer that could be received at 2335 psig is (high pressure alarm)."

Resolution:

This comment appears to be in error, as the reference material clearly states that the high pressure alarm comes in at 2310 psig.

3.0 Facility Comment:

5.10 (key): For part (4) the reviewer stated that emphasis should be on the change or shift of the onset of boiling.

Resolution:

The examiner agrees with this, and will grade the responses accordingly.

4.0 Facility Comment:

6.2 (key): The reviewer stated that for part (1) of the question the answer should be yes. The time stated for the load reduction of 90 sec. is less than the time constant for a 10 percent or greater load reduction for this plant which is 140 sec. The reviewer also expressed a concern that part (3) may not be yes, as it may be borderline.

Resolution:

The examiner agrees, for part (1). For part (3) if the candidate shows why the answer should be no he will get credit.

5.0 Facility Comments:

6.6 (key): The reviewer expressed the concern that the candidates might not know the precise sources of power, and inquired as to whether the examiner would give credit if they demonstrated that they know there are two busses and chargers.

Resolution:

Examiner Standard ES-402 A.2. states in part that "Candidates should be able to reproduce from memory, sketches or descriptions of various hydraulic, pneumatic, or electrical distribution systems...." Therefore, the examiner will only accept what is in the key.

6.0 Facility Comment:

6.7 (question): The reviewer expressed the concern that the question in part (2) was ambiguous. The answer in the key did not appear, for the second part, to be addressed in the question.

Resolution:

The examiner agrees, and will award 1.0 point for the first part of the answer in the key.

7.0 Facility Comment:

6.10 (question): The reviewer noted that the system in part (3) of the question had been deleted.

Resolution:

The examiner pointed out that the candidates should still be able to answer the question. The understanding of why piping systems should be full to prevent waterhammer events is germain to many systems in the plant and would reflect on the candidates general understanding of fluid systems. The question only asks why would it be desirable, which avoids any specifics about the system that was removed. The examiner sees no reason to delete the question.

8.0 Facility Comments:

6.11 (key): The reviewer pointed out that the definitions of Overpower Delta T and Overtemperature Delta T were reversed.

Resolution:

The examiner agrees, the key will be changed.

9.0 Facility Comments:

7.2 (key): The reviewer stated that the procedure EP OP-23 has been replaced, and the new procedure does not include symptoms.

Resolution:

The examiner agrees and will delete part (3).

10.0 Facility Comment:

7.4 (question): "Don't believe candidate should be responsible for classifying."

Resolution:

Examiner Standard ES-402 A.3. states in part that "The candidate must demonstrate complete knowledge and understanding of the symptoms, automatic actions, and immediate action steps specified by off-normal or emergency operating procedures." (Underlining added.)

11.0 Facility Comment:

7.7 (key): The reviewer noted that procedure AP-6 page 2 also requires emergency boration when:

1. Unexpected or increasing count rate when S/D.

2. Shutdown margin is less than acceptable minimum limits per Technical Specifications.

Resolution:

The question did not clearly indicate that the Technical Specifications were the basis of the response sought. Therefore, if the candidates' respond with the above points full credit will be given on the basis of any two responses will be 0.5 point each.

12.0 Facility Comment:

7.8 (key): "Air is isolated anytime >1200 lbs. is on load cell. (If he hangs up and load cell is <1200 lbs., then you could drop it.)"

Resolution:

The question only asks how the load cell prevents a fuel element from being dropped. No change will be made to key.

13.0 Facility Comment:

7.6 (key): The reviewer indicated a response could be that the P-250 computer could be used to address the point for Quadrant Power Tilt Ratio.

Resolution:

The question only inquires as to how the magnitude of a flux tilt is determined, the examiner sought to elicit a response about the relationship of the ratio of maximum upper or lower detector output to the respective upper or lower average detector outputs. No change will be made to the key.

14.0 Facility Comment:

7.9 (key): "Volume 9 Setpoints manual lists 550 psig as setpoint plus or minus accuracies."

Resolution:

The key does not place any requirement to recite the setpoint. The response sought is only that the oil lift pump must be running.

15.0 Facility Comment:

7.10 (key): The reviewer expressed the concern that the key seemed to require knowledge of the depth that would require memorization of procedural steps. This was indicated by the limits cited in the key.

Resolution:

The examiner agrees and will give full credit if the response includes just the mention of the excess letdown system.

16.0 Facility Comment:

8.7 (key): The Reviewer expressed the concern that the candidates would reply with a list of only the logs in the control room. The key lists several logs that are kept outside the control room.

Resolution:

Examiner Standard ES-402 A.4. States in part that "Questions may also cover ..., the types of records that must be maintained in the control room,..." If a candidate responds with the first three listed logs and provides either of the other two logs listed or another control room log full credit will be given.

17.0 Facility Comment:

8.9 (key): For part (1), "Add another answer: Adequate heat removal, (T.S. allows reduction of required loops of RHR if greater than 23 feet.)"

Resolution:

The question only asks about the bases cited in the Technical Specifications for the depth of 23 feet, and that is for Iodine gap activity removal. No change to the key.

18.0 Facility Comment:

8.10 (key): "There was a question in Section 8 of the exam which dealt with deciding whether or not 3.0.3. of Technical Specifications was in effect given 3 different situations. I request that the part which asked if 3.0.3. must be invoked when both RHR pump 1-1 and Cent. Charging pump 1-1 were inoperable be removed from the exam. This question on the surface would appear easy because the general assumption made is that since both are 1-1 pumps, they come off one ECCS subsystem therefore 3.0.3 does not apply. This however is not the case at Diablo Canyon. Centrifugal Charging Pump 1-1 is train A and RHR pump 1-1 is Train B."

Resolution:

The candidate should be aware of the electrical distribution (e.g., Question 6.6 resolution). No change to key.

19.0 Facility Comment:

8.2 (Question): "Several questions addressed Technical Specification requirements in a time period of more than 1 hour after LCO was exceeded.

Resolution:

Agree that expecting candidate to be fully aware of Technical Specifications with LCO requirements in excess of 1 hour is inappropriate. Therefore responses involving TS LCO's greater than one hour will be deleted.

20.0 Summary Change to question During Examination:

During the conduct of the examination it was brought to the attention of the examiner that a question on the test (question 7.3), concerned a procedure that had been superseded. The examiner reviewed the question, and determined that it was not valid. The question was restated as follows:

7.3: State the entry conditions of EP OP-7 for determining inadequate core cooling.

The answer for the inquiry is as follows:

(1) Any core exit thermocouple greater than 1200°F. (1.0)

(2) Or, a core exit thermocuple greater than 700 °F and RVLIS indicating less than 33 percent. (1.5)

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR EXAMINATION

Master Key

Facility: Diablo Canyon Units 1 and 2

Reactor Type: Westinghouse

Date Administered: ____________

Examiner: Gary W. Johnston

Candidate: ____________

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheet. Points for each question are indicated in parenthesis after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

Category Value	% of Total	Candidate's Score	% of Cat. Value	Category
25.0				5. Theory of Nuclear Power Plant Operation, Fluids, and Thermodynamics
25.0				6. Plant Systems Design, Control and Instrumentation
25.0				7. Procedures - Normal, Abnormal, Emergency and Radiological Control
25.0				8. Administrative Procedures, Conditions, and Limitations
100.0				TOTALS

Final Grade ______ %

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

____
Candidate's Signature

EQUATION SHEET

$f = ma$

$v = s/t$

Cycle efficiency = (Network out)/(Energy in)

$w = mg$

$s = V_o t + 1/2\ at^2$

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

$A = A_o e^{-\lambda t}$

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

$\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$

$t_{1/2}eff = \dfrac{[(t_{1/2})(t_b)]}{[(t_{1/2}) + (t_b)]}$

$\Delta E = 931\ \Delta m$

$I = I_o e^{-\Sigma x}$

$\dot{Q} = \dot{m}Cp\Delta t$

$\dot{Q} = UA\Delta T$

$I = I_o e^{-\mu x}$

$Pwr = W_f \dot{m}$

$I = I_o\ 10^{-x/TVL}$

$TVL = 1.3/\mu$

$P = P_o 10^{sur(t)}$

$HVL = -0.693/\mu$

$P = P_o e^{t/T}$

$SUR = 26.06/T$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$SUR = 26\rho/\ell^* + (\beta - \rho)T$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$T = (\ell^*/\rho) + [(\bar{\beta} - \rho)/\bar{\lambda}\rho]$

$M = 1/(1 - K_{eff}) = CR_1/CR_o$

$T = \ell/(\rho - \beta)$

$M = (1 - K_{effo})/(1 - K_{eff1})$

$T = (\beta - \rho)/(\lambda\rho)$

$SDM = (1 - K_{eff})/K_{eff}$

$\rho = (K_{eff} - 1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$\ell^* = 10^{-4}$ seconds

$\bar{\lambda} = 0.1\ \text{seconds}^{-1}$

$\rho = [(\ell^*/(T\ K_{eff})] + [\bar{\beta}_{eff}/(1 + \bar{\lambda}T)]$

$I_1 d_1 = I_2 d_2$

$P = (\Sigma\phi V)/(3 \times 10^{10})$

$I_1 d_1{}^2 = I_2 d_2{}^2$

$\Sigma = \sigma N$

$R/hr = (0.5\ CE)/d^2(\text{meters})$

Water Parameters

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.48 gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.

Miscellaneous Conversions

1 curie = 3.7×10^{10} dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1 in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)

DIABLO CANYON SRO EXAM

5. Theory of Nuclear Power Plant Operations, Fluids and Thermodynamics

5.1 (2.0)

Figure 5.1 shows the axial power distribution at beginning of life (BOL), hot zero power (HZP) conditions. Figure 5.2 shows the hot full power (HFP), end of life (EOL) and beginning of life (BOL) axial power distribution.

(1) For the case of BOL/HZP conditions versus BOL/HFP conditions there is a shift of the peak from approximately 50% to 40% for the axial power distribution. Why does the peak shift toward the core inlet at HFP from essentially a symetrical shape at HZP? 1.0

(2) For the case of BOL/HFP conditions versus EOL/HFP conditions discuss the flattening of the axial distribution over the life of the core. 1.0

5.1 Ans:

(1) At BOL the core contains all fresh fuel and at HZP the moderator temperature coefficient (MTC) is essentially constant axially. (Therefore the power distribution is approximately symetrical about the core). (0.5)

At HFP the coolant temperature rise across the core increases the MTC across the core from bottom to top, this increase shifts the power toward the core inlet. (0.5)

(2) In this case as the core depletes:

The higher power production within center of the core depletes the center of the core depletes the center faster. (0.5)

This results in a non-uniform axial burnup and consequently flattens the shape of the curve. (0.5)

Ref: WCAP-8408 "Reactor Theory Supplement"

RELATIVE AXIAL POWER DISTRIBUTION AT HZP, BOL

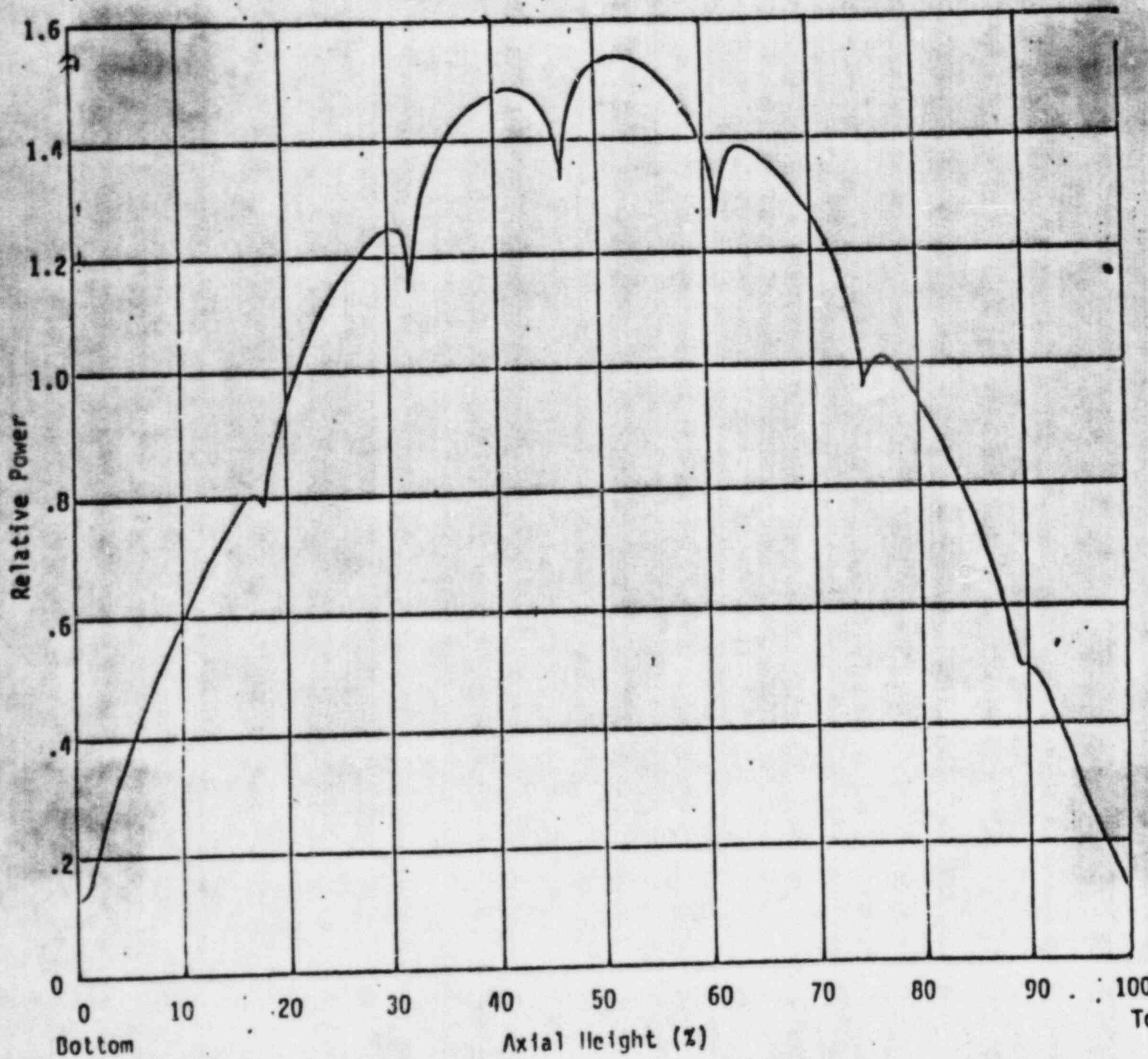


Figure 5.2

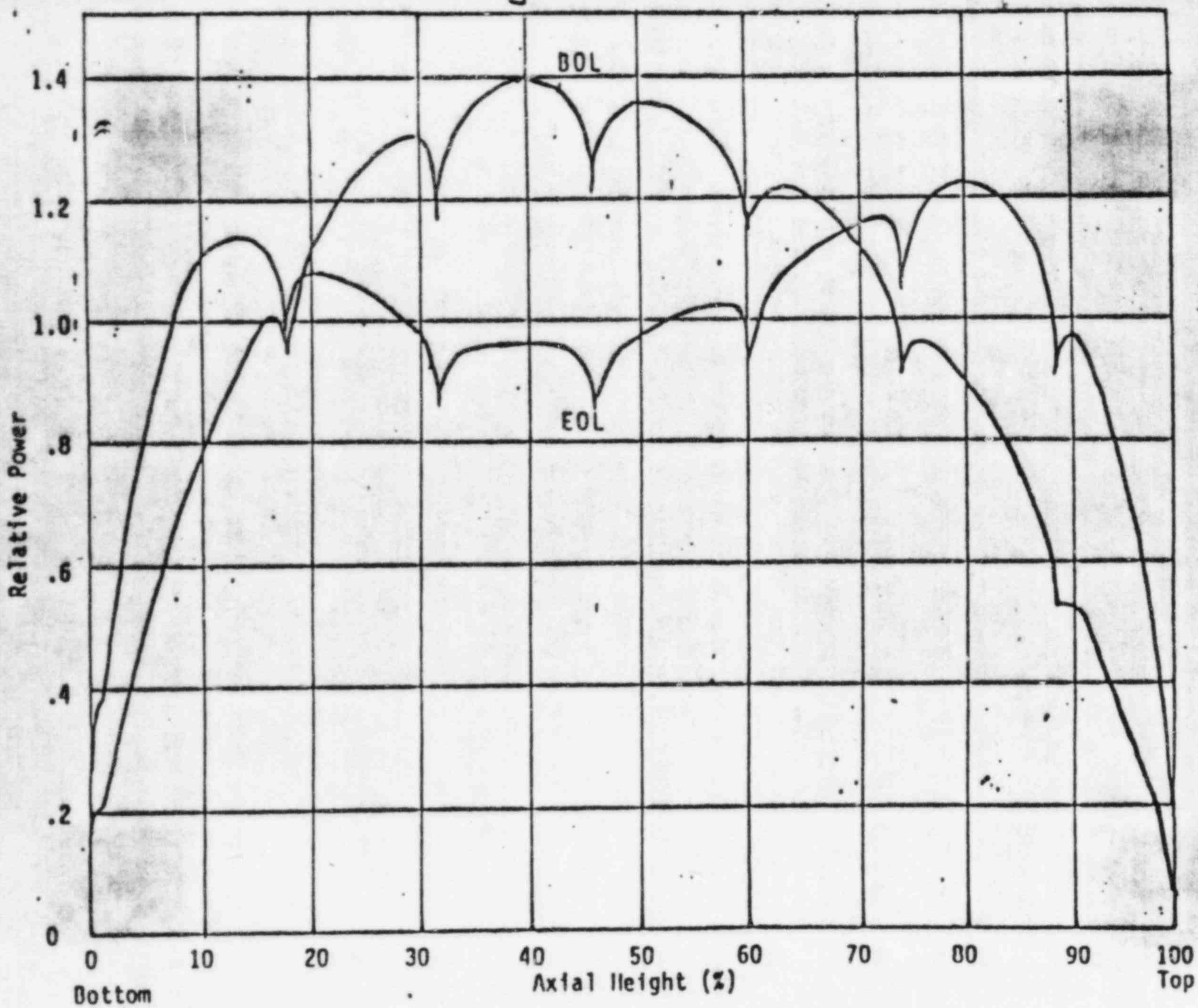


FIGURE 5.2 RELATIVE AXIAL POWER DISTRIBUTION AT HFP, BOL AND EOL

5.2 (1.5)

Several agents are periodically added to the Reactor Coolant system to control the chemical composition of the coolant. For each of the following describe the particular function each additive provides.

(1) hydrazine (0.5)
(2) hydrogen (0.5)
(3) lithium hydroxide (0.5)

5.2 Ans:

(1) Control of dissolved oxygen at low power and shutdown conditions (0.5)
(2) Control of dissolved oxygen at power (0.5)
(3) Control of RCS coolant pH at power (0.5)

Ref: Std

5.3 (1.0)

For a typical centrifugal pump the speed is increased by a factor of 3. For each of the following provide the factor by which they increase or decrease with relation to the speed of the pump.

(1) The volumetric flow rate (0.33)
(2) The brake horsepower (0.33)
(3) The head of the pump (0.33)

5.3 Ans:

(1) Increase by a factor of 3 (0.33)

(2) Increase by a factor of 3^3 = 27 (0.33)

(3) Increase by a factor of 3^2 = 9 (0.33)

Ref: Std ans.

5.4 (1.5)

List three of the four producers of intrinsic (not source) neutrons that generate neutrons that could be seen on source range instrumentation.

5.4 Ans:

Any 3 (0.5 each)

(1) Spontaneous fission

(2) Gamma-Deuterium

(3) Alpha-Oxygen

(4) Alpha-Boron

Ref: Operators Info. Manual, Pg. R-2-1

5.5 (2.5)

(1) With conditions in the pressurizer at 2235 psig describe the control room indications an operator would expect to see if a power operated relief valve was leaking by. (0.5)

(2) Calculate temperature in the tailpipe if the Pressurizer Relief Tank pressure is 20 psig. What is the state of the fluid in the tailpipe (i.e. saturated or superheated)? (1.5)

(3) What action(s) could be taken to limit or stop the leakage? (0.5)

5.5 Ans:

(1). Tailpipe temperature > ambient, PRT temperature and level increasing. (0.5)

(2) PRT pressure = 20 psig = 35 psia (0.25)
PZR pressure = 2235 psig = 2250 psia (0.25)
hg (2250 psia) = 1117.7
hg (35 psia) = 1117.7 (constant euthalpy) (0.25)
Tsat (35 psia) = 259°F (0.25)

The fluid in the pipe is saturated. (0.5)

(3) Close affected PORV block valve, and deenergize. (0.5)

Ref: Lesson Plan A-4a "Pressurizer and Pressure/Level Control System", Technical Specification 3.4.4.

5.6 (3.0)

Figure 5.3 shows the arrangement of the charging pumps in a typical CVCS system The plot of volumetric flow versus pressure shows the system with the positive displacement pump and one of the centrifugal pumps running, assuming HCV-142 is throttled:

(1) Show the change in the operating point from the initial conditions after HCV-142 is opened fully. **(1.0)**

(2) What is physically happening at point (a)? (High head, low volumetric flow rate). (1.0)

(3) What would happen if both FCV-128, HCV-142, and the seal injection isolation valve closed with both pumps running. (1.0)

5.6 Ans:

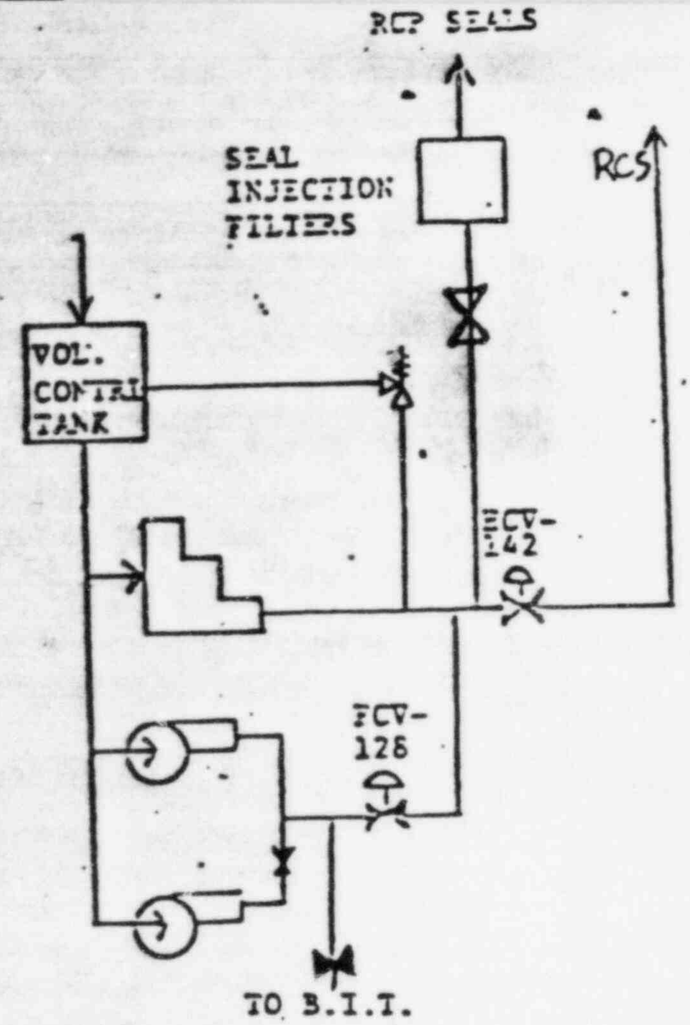
(1) Shift in operating curve. (1.0)

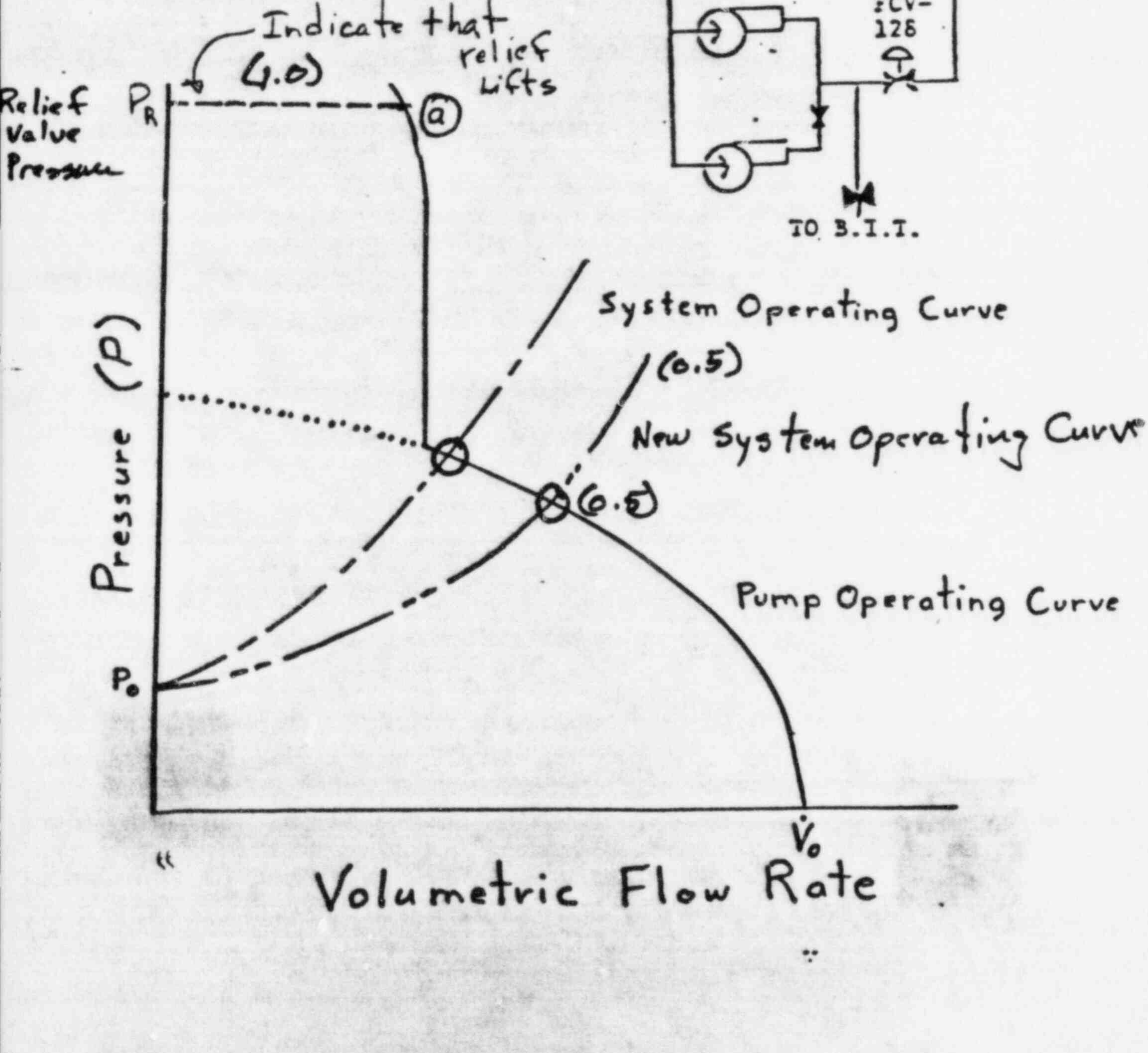
(2) Point (a) shows the "tail" as leakage past the moving parts starts to occur. (1.0)

(3) The relief valve opens at its's setpoint pressure. (1.0)

Ref: Westinghouse Thermodynamics book pages 10-32 through 10-52.

Ans: 5.7


RCP SEALS
SEAL INJECTION FILTERS
RCS
VOL. CONTRL TANK
FCV-142
FCV-128
TO B.I.T.


Indicate that relief lifts
(4.0)
Relief Valve Pressure
P_R
(a)
System Operating Curve
(0.5)
New System Operating Curve
(0.5)
Pump Operating Curve
Pressure (P)
P_o
$\dot{V}_o$
Volumetric Flow Rate

5.7 (2.0)

Describe the response of the pressurizer pressure control system as pressure rises from 2000 psig to 2450 psig.

5.7 Ans:

(2210 psig Backup heaters on)
2220 psig Backup heaters off, (proportional heaters at max) (0.4)
2250 psig Proportional heaters off (0.4)
2260 psig Spray valves start to modulate (full) open (0.4)
2310 psig Spray valves full open, (high pressure alarm) (0.4)
2335 psig PORV's open (0.4)

Ref: Lesson Plan A-4a "Pressurizer and Pressure/Level Control System"

5.8 (3.5)

A reactor is initially critical at a power level of 1 KW. At t = 0, the Keff of the reactor core was made to be 1.0020.

(1) Calculate the reactivity inserted in percent millirem (PCM). (1.0)

(2) Calculate the resulting period if $\bar{\beta}$eff = 0.0072 and $\bar{\lambda}$ = 0.1. (1.0)

(3) Calculate the resulting startup rate. (0.5)

(4) How long will it take for the reactor to reach 1 MW. (1.0)

5.8 Ans:

(1) PCM = 10^5

$P = \frac{(Keff-1)}{Keff} = \frac{1.0020-1}{1.0020} (10^5) = \underline{200\ PCM}$ (1.0)

(2) $T\ \frac{\beta eff-\rho}{\rho \bar{\lambda}} = \frac{(0.0072-.002)}{(0.002)\ (0.1)} = \underline{26}$ (1.0)

(3) $SUR = \frac{26.06}{26} = 1$ dpm (0.5)

(4) $P = P_o\ 10\ (SUR)^t$

1000 Kw = 1 MW = $(1\ Kw)10^{(1)t}$

$t = \log_{10}(1000) = \underline{3\ minutes}$ (1.0)

Ref: Westinghouse Training Notes, Neutron Kinetics

5.9 (2.0)

Give two reasons why at BOL nuclear instrumentation count rate will <u>increase</u> as plant temperature increases causing an RCS coolant density decrease. (Assume no rod movement on RCS boron concentration changes).

5.9 Ans:

(1) Neutrons travel farther, more leakage. (1.0)

(2) Lower boron density, less boron absorption. (1.0)

Ref: STD Ans.

5.10 (3.5)

Refer to Figure 5.4.; a sketch of Heat Flux vs Delta T (Twall-Tsat).

(1) Explain the effect of a pressure increase on the Nucleate Boiling Heat Transfer region of the curve. ~~(0.5)~~ (0.75)

(2) What would be the effect on Twall of an increase in pressure if heat transfer were initially occurring in the onset of the Nucleate Boiling Region of the curve? ~~(0.5)~~ (0.75)

(3) Explain the effect of a coolant flow increase on the Convection Heat Transfer region of the curve. ~~(0.5)~~ (1.0)

(4) Explain the effect of a coolant flow increase on the Nucleate Boiling Heat Transfer region of the curve. ~~(0.5)~~ (1.0)

5.10 Ans:

(1) As pressure is increased Nucleate boiling is suppressed. The saturation temperature of the water increases and more heat is required for nucleate boiling to occur.

(2) Since nucleate boiling is a much better means of heat transfer than convection, a sudden increase in pressure could cause a corresponding increase in the temperature of the heat source.

(3) As flow is increased the laminar layer in contact with the wall is reduced decreasing the delta t in the convection heat transfer region for a fixed heat flux.

(4) As flow is increased the steam bubbles formed in the nucleate boiling heat transfer region are "swept away" from the heat transfer surface so that a higher heat flux is required to form a steam film (steam blanket) so that the onset of boiling occurs at a higher heat flux.

Ref: Westinghouse Training Notes; Heat transfer

5.11 (2.5)

Refer to Figure 5.5 which shows an instantaneous, negative, reactivity insertion into an already critical reactor core (at time t = 0), followed by a removal of this negative reactivity after a stable reactor period is reached (at time t = 1), thus rendering the reactor critical once again. Assuming no source neutrons:

(1) Show the resulting startup rate (SUR) as a function of time (1.0) for this reactivity change.

(2) Show the reactor power level as a function of time for this (1.0) reactivity change.

(3) Explain the shape of the reactor power response immediately (0.5) after t = 0.

5.11 Ans: Refer to Key Figure 5.5.

Ref: Westinghouse Training Notes, Neutron Kinetics.

KEY

FIGURE 5.5

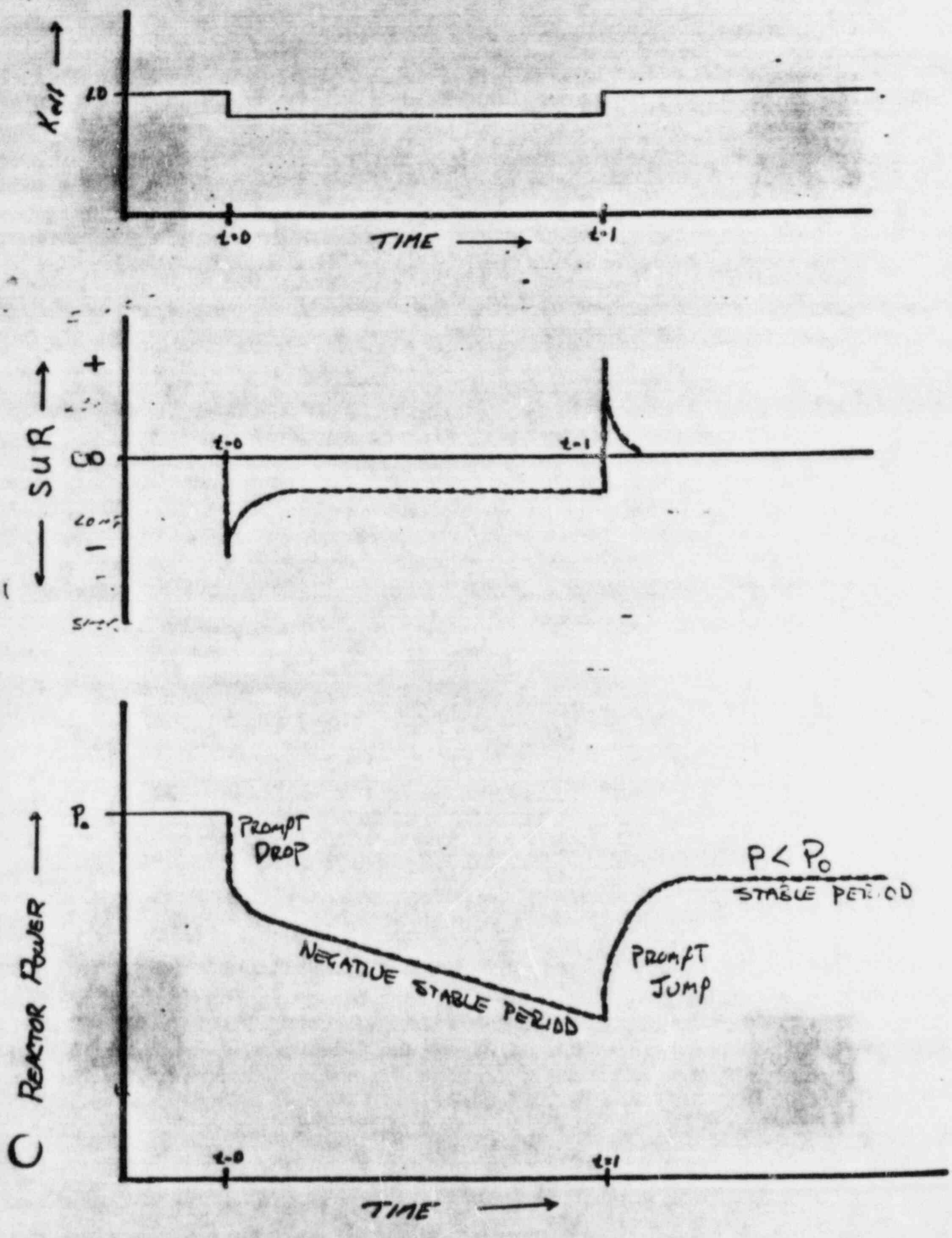
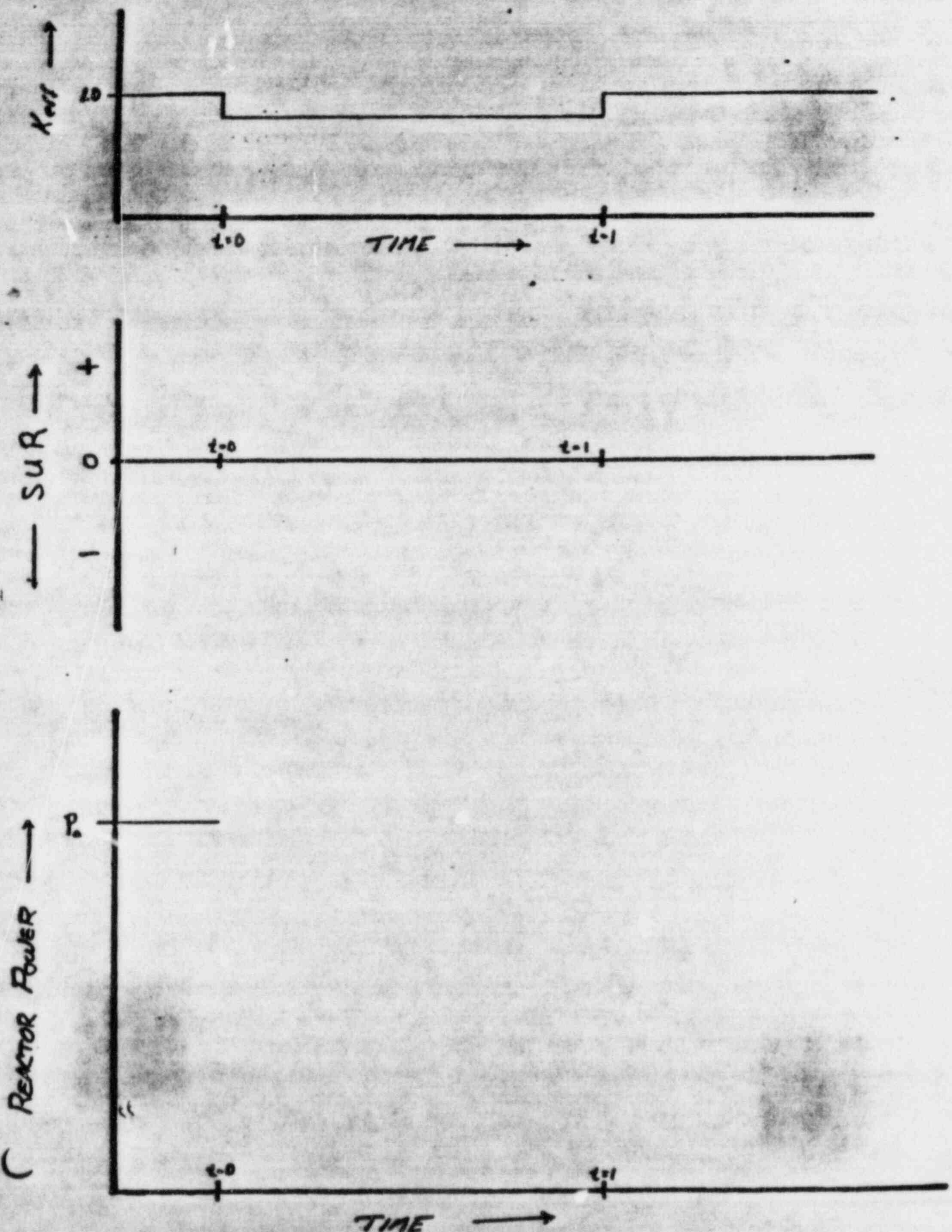


FIGURE 5.5



6. Plant System Design, Control and Instrumentation

6.1 (2.0)

Downstream of the non-regenerative heat exchanger in the CVCS letdown line there are two valves providing protective functions. Describe these functions for:

(1) Letdown temperature direct valve, TCV-149. (1.0)

(2) Low pressure letdown control valve, PCV-135 (1.0)

6.1 Ans:

(1) TCV-149 protects the demineralizers downstream by diverting flow around the demineralizers if temperature goes too high (136°F). (1.0)

(2) PCV-134 maintains a 350 psig backpressure on the letdown flow through the orifices to prevent the letdown flow from flashing and (erroding the orifices). (1.0)

Ref: Lesson Plan B-1a "Chemical and Volume Control System"

6.2 (2.5)

The plant is in stable operation at 50% power, the rods are in manual. Turbine load drops rapidly from 50% to 40% in 90 seconds due to a spurious EHC problem and remains at 40% of load.

(1) Will the Steam Dumps arm? (0.5)

(2) What is the purpose of Steam Dump System interlock C-9? (0.5)

(3) Will the steam line PORV's open? (0.5)

(4) What happens to reactor coolant average temperature (Tavc)? (0.5)

(5) Describe the two inputs to the C-9 interlock. (0.5)

6.2 Ans:

(1) ~~No.~~ yes (< 140 sec) (10% in 1.5 min) (0.5)

(2) Assures condenser available prior to allowing Steam Dump operation. (0.5)

(3) Yes. (Pressure should rise to H/C-516 setpoint). (0.5)

(4) Increases. (RCS heats up from differential in load vs Rx power). (0.5)

(5) Vacuum < 20 inches north and south (2/2 logic). (0.25)

At least one circulating water pump breaker is closed. (1/2 logic) (0.25)

Ref: Lesson Plan - "Steam Dump System" C-26

tt

6.3 (2.0)

Discuss briefly how the source and intermediate range nuclear instrumentation discriminate between neutron flux and gamma radiation, include in your discussion a description of the type of detector used. **(2.0)**

6.3 Ans:

Source range uses a pulse height (pulse amplifier) discriminator (0.25), the interaction of a neutron with the boron in the gas of the detector generates a stronger ionization event than a gamma would, the pulse height discriminator passes the higher pulse generated by the neutron (0.25). The detector is a Bf_3 proportional chamber, that utilizes the boron in the BF_3 gas as a neutron sensitive detector. (0.5)

Intermediate range uses a compensated ion chamber (0.5) consisting of two chambers in one case. One is coated with a neutron sensitive boron coating, the other is not coated. The two chambers are connected so their output currents are electrically opposed (0.5), (therefore the net electrical current output will be the algebraic sum of the currents).

Ref: STD, Lesson Plan B-4

6.4 (2.0)

Unit 1 is assumed to be at 50% power, with all controls in automatic, and with equilibrium conditions. Briefly discuss how each of the following events will affect control rod motion and why.

(1) Reactor coolant pump trip. (0.5)
(2) Atmospheric dump valve opening inadvertantly. (0.5)
(3) Isolation of a feedwater heater string. (0.5)
(4) 5% increase in turbine load. (0.5)

6.4 Ans:

(1) Rx coolant pump trip, Rx coolant pump breaker opens, Rx trip, rods in. (0.5)
(2) Steam demand increase causes Tave/Tref mismatch, rods move out. (0.5)
(3) Cooler feedwater drops Tc, Tave lowers, Tave-Tref mismatch rods move out. (0.5)
(4) 1st stage impulse pressure increases raising Tref, Tave-Tref mismatch rods move out. (0.5)

Ref: Lesson Plan A-3a "Full Length Rod Control"

6.5 (2.5)

The steam generator water level control system utilizes a programmed level from 0% power to 100% power. Figure 6.1 describes the program.

(1) From what reference signal is the program level derived? (1.0)

(2) Why is the S/G level ramped from 33% to 44% at low power levels (<20% power)? (1.5)

6.5 Ans:

(1) The program level is generated using main turbine first stage impulse pressure as a reference. (1.0)
(2) To limit the mass available in the S/G for consideration of steam line break accident (1.5). Lower power level results in lower steam volume inS/G bundle resulting in a greater mass. (0.5)

Ref: Lesson Plan C-8b

((

Level is programmed as follows:

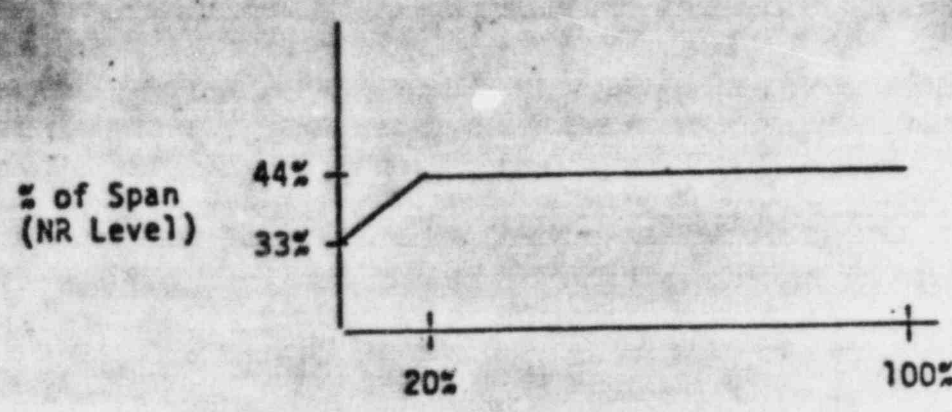


Fig. 6.1

6.6 (2.5)

Describe how the 125 VDC Vital Loads Panel 11 receives its sources of power. Include all sources from the 480 VAC source(s) through to the panel.

6.6 Ans:

(1)	480 VAC Vital Bus "F"	(0.5)
	Through Battery charger 11 to the panel	(0.5)
(2)	480 VAC Vital Bus "H"	(0.5)
	Through battery charger 121 to the panel.	(0.5)
(3)	Battery 11	(0.5)

Ref: Operator Information Manual, Lesson Plan-DC Systems J-9.

6.7 (2.0)

No. 13 Diesel Generator can supply power to 4KV Bus "F" on either Unit 1 or 2.

(1) If No. 13 DG was in service on Unit 1's Bus F, but not as the result of an S.I. signal on that unit, what would occur if an S.I. signal came in on Unit 2? (1.0)

(2) If the No. 13 DG was answering an S.I. signal on Unit 2, and an S.I. signal came in on Unit 1, what would occur? (1.0)

6.7 Ans:

(1) An S.I. signal on Unit 2 would trip the feeder to Unit 1 (0.5) and allow transfer to Unit 2. (0.5)

(2) The DG would stay committed to Unit 2 (the first S.I.) ~~(0.5)~~ (1.0) and transfer only when the S.I. on Unit 2 was reset ~~(0.5)~~.

Ref: Lesson Plan J-6b "Diesel Generator and Auxiliary System"

6.8 (2.0)

For each of the following process radiation monitors describe the type of detector and the automatic function, if any, when an alarm is received:

1. Containment Air Particulate Monitor RE-11. (0.5)

2. Component Cooling Liquid Monitor RE-17A, 17B. (0.5)

3. Gas Decay Tank Discharge Gas Monitor RE-22. (0.5)

4. Control Room Air Particulate Monitor RE-21. (0.5)

6.8 Ans:

1. Scintillation counter (0.25), high radiation alarm initiates containment ventilation isolation (0.25).

2. Scintillation counter (0.25) high activity initiates closure of surge tank vent (RCV-16) (0.25).

3. G-M detector (0.25) high activity initiates closure of discharge from gas decay tanks (RCV-17) (0.25.

4. Scintillation counter (0.25) no automatic action (0.25).

Ref: Lesson Plans G-3 "Area Radiation Monitor System", and G-4 "Process Radiation Monitors"

6.9 (2.0)

The upper support plate of the reactor vessel intervals differ in configuration. Unit 2 has a flat plate and Unit 1 has a top-hat configuration. Briefly discuss the operational consideration this design difference imposes on Unit 2. (2.0)

6.9 Ans:

Operational Considerations

Soak times on natural circulation cooldowns w/o CRDM fans will be more restrictive for Unit 2 due to increased mass of water in plenum (1.0). It would also require a larger inventory of water in pressurizer to collapse a void in the plenum area. (1.0)

Ref: Unit Differences Manual Section 2.

6.10 (3.0)

For the Auxiliary Feedwater System:

(1) What interlock prevents level control valves LCV's 110, 111, 113, and 115 from opening to provide pump runout protection for auxiliary feedwater pumps 1-2 and 1-3? (1.0)

(2) What are the two automatic start signals for Auxiliary Feedwater pump 1-1? (1.0)

(3) The Auxiliary Feedwater pumps have a pump discharge pressurization system to keep the discharge piping full and pressurized. Why would it be desirable to do this? (1.0)

6.10 Ans:

(1) The level controller receives a level signal from the SGWLC transmitter (33%), a pressure transmitter in the discharge piping provides an output that modifies the signal to the positioner on a LCV (0.5). As pressure drops, the signal is biased to close the LCV (0.5).

(2) Undervoltage on both 12KV busses (0.5). Lo-Lo level on 2 of 4 steam generators (0.5).

(3) To reduce the potential of a water hammer event caused by a vapor pocket. (1.0)

Ref: Lesson Plan D-1 "Auxiliary Feedwater System"

"

6.11 (2.5)

The Reactor Protection System is afforded two loop temperature trip signals, Over Temperature Delta T and Over Power Delta T.

(1) What protective function does each of these provide for the Reactor Protection System? (1.0)

(2) Each of these signals provides an input to control interlocks. What do those interlocks functionally do to the rod control system and the Turbine Supervisory System? (1.0)

(3) What are the respective setpoints for those interlocks? (0.5)

6.11 Ans:

Terms Reversed

(1) Over Power Delta T - protects against exceeding DNBR limits. (0.5)
Over Temperature Delta T - protects against high fuel centerline temperatures (0.5)

(2) Rod stop (0.5)
Turbine runback (0.5)

(3) Both are 3% below trip setpoints (0.5)

Ref: Lesson Plan B-6a "Reactor Protection System"

7. <u>Procedures-Normal, Abnormal, Emergency, and Radiological Control</u>

7.1 (3.0)

In accordance with 10 CFR 20, "Standards for Protection Against Radiation":

(1) What are the Radiation Dose Standards for individuals in restricted areas per Calendar Quarter? (1.5)

(2) What are three requirements that must be met if the Whole Body Limits for a Calendar Quarter are to be met? (1.5)

7.1 Ans:

(1) 1.25 Rem - Whole body; head and trunk; active blood forming organs; lens of eyes; or gonads (0.5)
18.75 Rem - hands and forearms; feet and ankles (0.5)
7.5 Rem - Skin of the whole body (0.5)

(2) 3.0 Rem per Calendar Quarter maximum (0.5)
5 (N-18) total accumulated dose where N is the individuals age in years at his last birthday (0.5)
Form NRC-4 or equivalent (0.5)

Ref: 10 CFR 20

7.2 (2.5)

In accordance with EP OP-23, "Natural Circulation of Reactor Coolant"

(1) During a Natural Circulation Cooldown, what is the maximum cooldown rate that should be maintained? (0.5)

(2) What concern limits the cooldown rate? (1.0)

(3) List 4 symptoms that could indicate that a natural circulation cooldown may be necessitated? (1.0)

7.2 Ans:

(1) 25 degrees F per hour (0.5)

(2) To avoid void formation in the upper head which has little reactor coolant flow in natural circulation. (1.0)

(3) SYMPTOMS (0.25) for any 4

1. RTD bypass line low flow alarms.
2. Reactor coolant low flow protection bistable monitor lights on.
3. Reactor coolant flow indication decreases to near zero in all loops.
4. RCP breaker lights and motor ammeters indicate breakers tripped.
5. Possible RCP bus undervoltage or underfrequency.

Deleted

Ref: Emergency Operating Procedure EP-OP-23 "Natural Circulation of Reactor Coolant"

7.3 (2.5)

Appendix B of EP OP-0 "Determination of Adequate Core Cooling" provides guidance for the determination of inadequate cooling of the reactor core. What are the criteria for determination of inadequate core cooling:

(1) If the P-250 process computer is not available? (1.5)

(2) If the P-250 is available? (1.0)

7.3 Ans:

(1) 3 of 10 thermocouples exceed 700°F (pegged high) and SI flow to the RCS and AFW to the steam generators cannot be confirmed. (1.5)

(2) 5 or more P-250 thermocouple readings exceed 1200°F. (1.0)

Ref: Appendix B of EP OP-0 "Determination of Adequate Core Cooling".

Question was restated in course of Exam as:

7.3 (2.5).

State the entry conditions of EP OP-7 for determining inadequate core cooling.

7.3 ans:

(1) Any core exit thermocouple greater than (1.0) 1200 deg. F.

(2) Or, a core exit thermocouple greater than 700 deg F. (0.75) and RVLIS indicating less than 33%. (0.75)

7.4 (3.0)

Match the potential indicated condition with the appropriate Emergency Action Level.

(1) On-going security threat which may result in loss of physical control of facility. (0.5)

(2) Earthquake greater than 0.6g. (0.5)

(3) Effluent monitors levels corresponding to site boundary dose rates of 1 Rem/hr whole body or 5 Rem/hr thyroid. (0.5)

(4) Primary coolant leak rate in excess of 50 gpm. (0.5)

(5) Loss of offsite power or loss of onsite AC power capability. (0.5)

(6) Fire within the plant lasting more than 10 minutes. (0.5)

EAL's

a. Unusual Event
b. Alert
c. Site Area Emergency
d. General Emergency

7.4 Ans:

(1) C (0.5)
(2) C (0.5)
(3) D (0.5)
(4) B (0.5)
(5) A (0.5)
(6) A (0.5)

Ref: EP-G1 "Accident Classification and Emergency Plan Activation"

7.5 (3.0)

(1) Technical Specification Limiting Condition for Operation 3.4.8 establishes the maximum Reactor Coolant Activity Limits, what are they? (1.0)

(2) What are the bases for these limits? (1.0)

(3) The plant has been operating at a RCS coolant activity of 100 μCi/gm Dose equivalent I-131 for a period of 40 hours.

a. From figure 3.4-1 what is the maximum power the plant may continue to operate at? (0.5)

b. How much longer may it operate if the cumulative hours are 748 hours of operation? (0.5)

7.5 Ans:

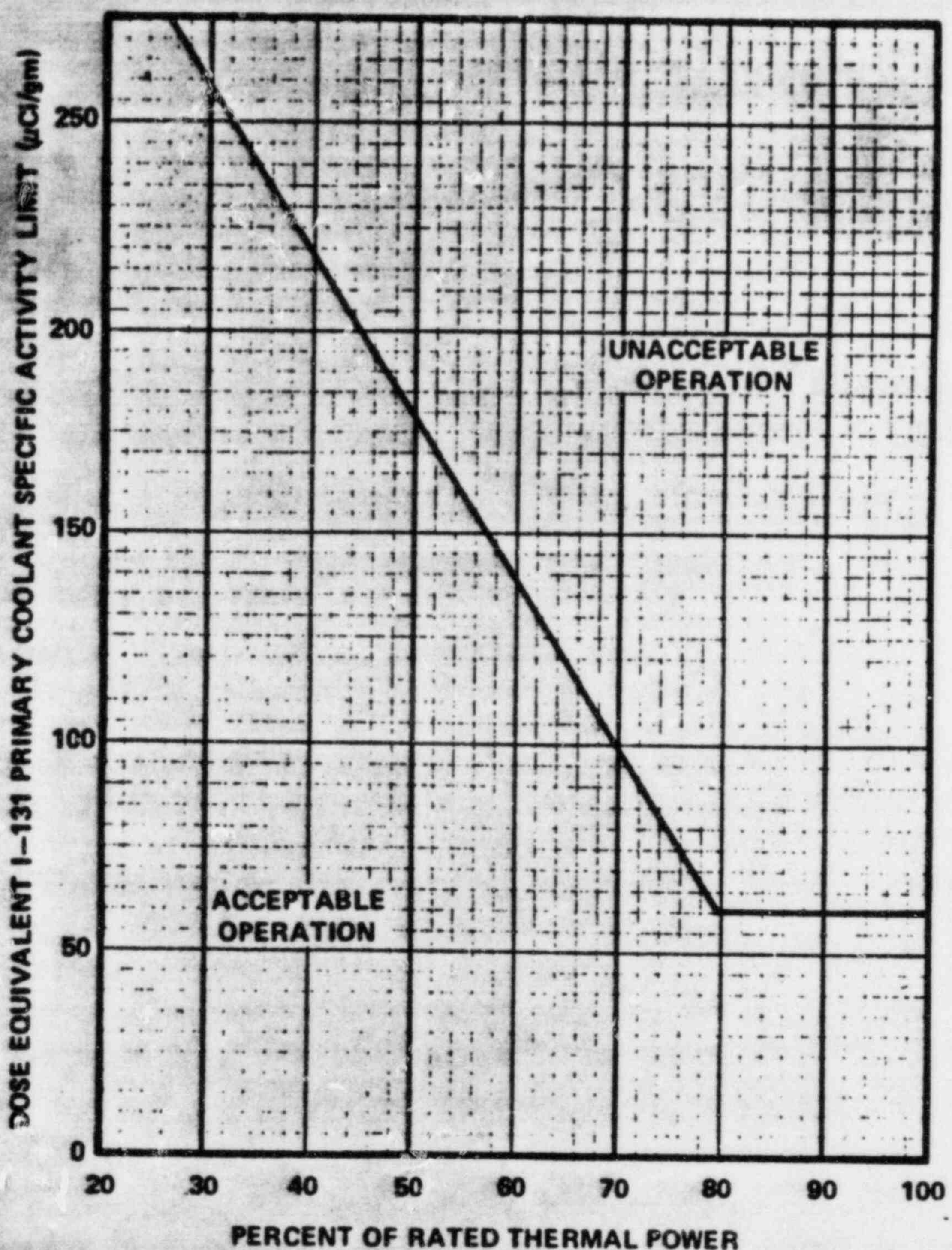
(1) 1.0 Ci/gm Dose Equivalent (0.5)
I-131 and,
100/E ¯Ci/gm specific activity (0.5)

$\bar{E}$ = the average of the beta and gamma energies per disintegration for isotopes other than Iodine with a half life greater than 15 minutes.

(2) The limits ensure the 2 hour dose at the site boundary will not exceed an appropriately small fraction of the 10 CFR 100 limits after a steam generator tube rupture in excess of 1 gpm. (1.0)

(3) a. 70% power (0.5)
b. 8 hours, can operate over 1.0 Ci/gm for a period not to exceed 48 hours if cumulative hours are less than 800 in last consecutive 12 months. (0.5)

Ref: Technical Specification 3/4.4.8. "Specific Activity"



FIGURL 3.4-1

DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1.0 μCi/GRAM DOSE EQUIVALENT I-131

7.6 (2.5)

While operating at full power, all rods out, equilibrium conditions, annunciator window "Power Range Flux Deviation" alarms indicating an apparent flux tilt.

(1) How is the magnitude of an apparent flux tilt determined? (1.0)

(2) Assuming the instrumentation is correct, what other event could be a probable cause of the flux tilt? (0.5)

(3) What two other instrumentation systems could be utilized to determine if a flux tilt exists. (1.0)

7.6 Ans:

(1) Ratio of maximum upper or lower detector output to the respective upper or lower average detector outputs. (1.0)

(2) Dropped rod (0.5)

(3) (any 2 - 0.5) each

(1) Thermocouples
(2) Incore detectors
(3) Rod position indication

Ref: Annunciator response Guide PK03 window 10, Technical Specifications

7.7 (2.0)

During refueling operations:

(1) What two conditions could require emergency boration during (1.0) Mode 6?

(2) Why must fuel movement not occur until the reactor has been (1.0) subcritical for 100 hours.

7.7 Ans:

(1) a. Keff > 0.95 ~~(0.5)~~
b. Boron concentration < 2000 ppm ~~(0.5)~~

(2) To allow for the decay of short lived fission products. (This time is consistent with accident analysis.) (1.0)

Ref: Technical Specifications 3.9.1 and 3.9.3

add to (1)
c. Unexpected or increasing count rate when shut down.
d. Shutdown Margin is less than acceptable minimum limits per Technical Specifications.

any 2 (0.5) points each

tt

7.8 (1.5)

(1) Operating Procedure OP B-8D "Core Loading Sequence Unit 1, Initial Core" has prescribed load limits for the insertion (+100, -300 lbs.) and withdrawal (+300, -100 lbs.) of fuel assemblies. Why is it desirable to have these limits? (0.5)

(2) How does the "Dillon" load cell of the manipulator crane prevent a fuel assembly from being dropped inadvertantly during insertion or withdrawal? (0.5)

(3) Prior to movement of fuel Attachment B of OP B-8D "Core Loading Valve Check List" must be filled out. Valve 8469 "Primary Water Supply to VCT" is verified to be closed, while Valve 8066 "TygonHose Connection to RCS Loop 4" is verified to be open. For each valve, discuss they they must be in these positions prior to fuel movement. (0.5)

7.8 Ans:

(1) To preclude possible damage to fuel spacer grids. (0.5)

(2) Isolates air to the gripper when (1200 lbs.) is sensed on withdrawal. The reverse is true on insertion. (0.5)

(3) Valve 8469 - Closed prevents or precludes inadvertant dilution. (0.25)
Valve 8066 - Open ensure that visual level indication of RCS can be seen in Tygon hose. (0.25)

Ref: Operating Procedure OPB-8D "Core Loading Sequence Unit 1, Initial Core"

7.9 (1.5)

(1) Procedure OP A-6:I "Reactor Coolant Pumps - Place In Service" provides restrictions on frequent starts, what are they? (1.0)

(2) An interlock is provided to prevent a Reactor Coolant pump from starting until what criteria are met? (0.5)

7.9 Ans:

(1) If the Reactor Coolant Pumps are to be started for reactor coolant system venting, or any other reason which may require frequent starts, the following shall be observed.

a. After any period of running or after any attempt to start a motor and it fails to achieve rated speed, the motor must be allowed to cool a minimum of 30 minutes before a restart is made. (0.5)

b. Within any 2-hour period, the number of starts shall be limited to three with a minimum idle period of 30 minutes between starts. When this limit is reached, a fourth start shall not be attempted until the motor has been allowed to cool a minimum of one hour. (0.5)

(2) An interlock prevents starting a Reactor Coolant Pump unless the oil Lift Pump has been running for (two minutes with a discharge pressure of at least 600 psig). (0.5)

Ref: Procedure OP A-6:I "Reactor Coolant Pumps - Place in Service"

7.10 (3.5)

(1) Per Operating Procedure OP L-1 "Plant Heatup from cold shutdown to hot standby" what two indications or steps are used to verify that a steam bubble has formed in the pressurizer? (1.5)

(2) Once the presence of a steam bubble has been verified, what three steps can be used to expedite lowering level in the pressurizer? (1.5)

(3) Prior to the formation of a bubble, how is system pressure controlled? (0.5)

7.10 Ans:

(1) a. Observe the letdown flow rate carefully. It should show an increase when steam formation begins. (0.5)

b. Carefully reduce the charging pump speed while observing system pressure. If a bubble is present there will be no change in system pressure. If the bubble has not formed yet, the system pressure will decrease. (1.0)

(2) a. Reduce the charging pump speed to the minimum value required for the coolant pump seals (approximately 32 GPM). (0.5)

b. Increase the letdown flow to its maximum value of 120 GPM. (0.5)

c. If necessary or desirable, place the excess letdown system in service. (Limit the flow through this system to a value that maintains the heat exchanger outlet temperature below 210°F) (about 20 GPM). (0.5)

(3) Letdown pressure control valve PCV-135. (0.5)

Ref: Operating Procedure OP L-1 "Plant Heatup from Cold Shutdown to Hot Standby"

8. Administrative Procedures, Conditions, and Limitations

8.1 (1.5)

What are the Technical Specification requirements for making temporary changes to operating procedures? (1.5)

8.1 Ans:

The essence of the underlined must be included.

a. The intent of the original procedure is not altered. (0.5)

b. The change is approved by two members (0.25) of the plant management staff, at least one of whom holds a Senior Reactor Operator's License (0.25) on the Unit affected.

c. The change is documented, reviewed by the PSRC and approved (0.25) by the Plant Manager within 14 days of implementation. (0.25)

Ref: Tech. Spec. 6.8.3.

8.2 (3.0)

Match the event in Comumn 1 with the proper action statement in Column 2. Assume Mode 1 operation above 50% power.

Column 1

1. One pressurizer code safety out of service. (0.5)
2. A leak through the Reactor Coolant Pressure Boundary. (0.5)
3. An accumulator isolation valve closed, and cannot be opened. (0.5)
4. A Control Room ventilation system main supply fan out of service. (0.5)
5. A 125-volt D.C. battery out of service. (0.5)
6. An auxiliary saltwater train out of service. (0.5)

Column 2

a. Restore to operable status within 7 days or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

b. Restore to operable status within 15 minutes or be in HOT STANDBY within 6 hours and within 6 hours be in HOT SHUTDOWN.

c. Restore to operable status within 2 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. Be in HOT STANDBY in 1 hour and be in HOT SHUTDOWN within the next 12 hours.

e. Be in HOT STANDBY in 6 hours and in COLT SHUTDOWN within the following 30 hours.

f. Restore to operable status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

8.2 Ans:

(1) b. (0.5)
(2) e. (0.5)
(3) d. (0.5)
(4) a. (0.5)
(5) c. (0.5)
(6) f. (0.5)

Delete 2,4,5,6

Points (1.0)

Ref: Tech Specs 3.4.2.2, 3.4.6.2, 3.5.1, 3.7.4.1, 3.7.5.1, 3.8.3.1

8.3 (2.0)

Regarding procedure NPAP A-8 "Overtime and Emergency Relief Restrictions":

(1) How long may an individual work period be? (0.5)

(2) If an individual SRO or RO is scheduled to work in excess of 8 hours, how long may he be "at the controls" during that period without being relieved of primary duties? (0.5)

(3) How many hours shall be provided between work periods? (0.5)

(4) How many hours may an operator work in a seven day period? (0.5)

8.3 Ans: (0.5 each)

1. 12 hours
2. 4 hours
3. 12 hours
4. 72 hours

Ref: NPAP A-8 "Overtime and Emergency Relief Restrictions"

8.4 (3.0)

While the plant is in Mode 3 a routine startup is proceeding. (3.0) During surveillance testing on one of the Diesel Generators a diesel fuel oil transfer pump fails to operate. The Maintenance Supervisor states that the pump motor will have to be replaced, there is no motor onsite, but one can be procured and installed in no later than 48 hours. Explain why the startup can or cannot proceed. A copy of the Technical Specification pertaining to the emergency power sources is provided.

8.4 Ans:

A diesel generator out of service places you in an action statement. The startup cannot proceed until the LCO is met. The startup cannot proceed because entry onto an Operating Mode is not allowed if an action statement must be relied on to do so as per (Technical Specification) 3.0.4. (3.0)

Ref: Tech. Spec 3.0.4

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two independent circuits (one with delayed access) between the offsite transmission network and the onsite Class 1E distribution system, and

b. Three separate and independent diesel generators, each with:

1. A separate engine-mounted fuel tank containing a minimum volume of 200 gallons of fuel,

2. A common fuel storage system containing a minimum volume of 31,023 gallons of fuel, and two diesel fuel transfer pumps.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.2 within one hour and at least once per 8 hours thereafter; restore at least two offsite circuits and three diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.2 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and three diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*One inoperable fuel transfer pump is equivalent to one inoperable diesel generator.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With one diesel generator inoperable in addition to a. or b. above verify that:

   (1) All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generators as a source of emergency power are also OPERABLE, and

   (2) When in MODE 1, 2, or 3 that at least two auxiliary feed pumps are OPERABLE.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of three diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.2 within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two or more of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; restore at least two of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least three diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

8.5 (2.0)

ANSWER TRUE or FALSE

1. Temporary procedures, and changes to temporary procedures, shall receive the same review and approval as permanent procedures. (0.5)

2. A person can alter the sequence of steps of a procedure without the required approval(s) if the sequence of steps is not clearly arbitrary and the procedure would be significantly improved. (0.5)

3. A surveillance requirement, if conducted within 25% of the time interval required in the Technical Specifications, will meet the OPERABILITY requirements for an LCO. (0.5)

4. A supervisor is authorized to issue temporary instructions as he sees fit for any departmental matters which are outside the scope of approved procedures. (0.5)

8.5 Ans:

1. True (0.5)
2. False (0.5)
3. True (0.5)
4. True (0.5)

Ref: NPAP E-4 "Procedures", AP C-351 "Surveillance Testing and Inspection", Technical Specification 6.2.2.

8.6 (3.0)

In accordance with 10 CFR 55, "Operators' Licenses:

(1) The "Exemptions from License" provisions of the Code of Federal Regulations (10 CFR 55), allow what individuals to operate the reactor controls without a license? (1.0)

(2) As defined in 10 CFR 55, when is an individual deemed to be operating the controls of a nuclear facility? (1.0)

(3) What are the "controls" as defined in 10 CFR 55? (1.0)

8.6 Ans: (1) An individual may manipulate the controls as a part of his training to qualify for an operator license under the direction and in the presence of a licensed operator or senior operator.(1.0)

(2) An individual is deemed to operate the controls of a nuclear facility if he directly manipulates the controls or directs another to manipulate the controls. (1.0)

(3) "controls" - apparatus and mechanisms the manipulation of which directly affect the reactivity or power level of the reactor.(1.0)

Ref: 10 CFR 55

8.7 (2.0)

List 4 of the 5 logs required by NPAP E-6 "Plant Logs". (2.0)

8.7 Ans: Required Logs (any 4 - 0.5 each)

(1) A Shift Foreman's log shall be maintained for each plant or group of Units under the jurisdiction of a single Shift Foreman during normal power operation.

(2) A Control Operator's log shall be maintained for each unit.

(3) An Auxiliary Operator's log should be maintained if the Auxiliary Operator's normal duty station is outside of the control room.

(4) A Shift Chemical and Radiation Process Technician log or the equivalent shall be maintained.

(5) A Shift Control Technician log or the equivalent shall be maintained.

Ref: NPAP E-6 "Plant Logs"

If candidate responds with another C.R. log full credit, however responses (1), (2), (3) required.

8.8 (3.0)

Which of the following events require immediate notification (within a period of 1 hour or sooner) to the NRC Operations Center via the Emergency Notification System per 10 CFR 50.72 and/or 10 CFR 20.403?

1. Damage to property on the site in excess of $100,000. (0.5)
2. Vehicular accident on access road resulting in multiple injuries to passenger requiring evacuation to offsite medical facility. (0.5)
3. Declaration of an Unusual Event at Diablo Canyon Unit 1. (0.5)
4. Personnel exposure to an individuals hands of 350 Rem while performing maintenance in a steam generator. (0.5)
5. An event or condition that alone could have prevented the removal of residual heat. (0.5)
6. The facility receives warning of an impending Tsunami wave that could affect the intake structure. (0.5)

8.8 Ans:

1. no (0.5)
2. no (0.5)
3. yes (0.5)
4. no (0.5)
5. no (0.5)
6. yes (0.5)

Ref: 10 CFR 50.72, 10 CFR 20.403

8.9 (3.0)

(1) What are the bases for maintaining at least 23 feet of water over the top of the reactor vessel flange? (1.0)

(2) In Mode 6 operation with 23 feet of water above the vessel flange, the Technical Specifications allow all residual heat removal may be suspended for 1 hour, why? (1.0)

(3) When must direct communications between the control room and personnel at the refueling station be maintained? (1.0)

8.9 Ans:

(1) Removal of 99% of the 10% gap iodine activity if a fuel element is damaged. (1.0)

(2) To allow fuel movement near the hot leg nozzles. (1.0)

(3) During core alterations. (1.0)

Ref: Technical Specifications and Bases, 3.9.10, 3.9.8.1, 3.9.5

8.10 (2.5)

(1) What are the bases for Technical Specification 3.0.3 (TS 3.0.3 is provided). (1.0)

(2) Does Technical Specification 3.0.3 apply in each of the following cases:

a. Centrifugal charging pump 1-1 and RHR pump 1-1 inoperable. (0.5)

b. Both safety injection pumps. (0.5)

c. Charging pump 1-3 and charging pump 1-1. (0.5)

8.10 Ans:

(1) 3.0.3 This specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION Statements and whose occurrence would violate the intent of a specification. For example, Specification 3.5.2 requires two independent ECCS Subsystems to be OPERABLE and provides explicit ACTION requirements if one ECCS Subsystem is inoperable. Under the requirements of Specification 3.0.3, if both the required ECCS Subsystems are inoperable, within one hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours. (The essence of the underlined sections 1.0 pt).

(2) a. no (0.5)
b. yes (0.5)
c. no (0.5)

Ref: Technical Specification 3.0.3 and Bases.

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in a MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time to failure to meet the Limiting Condition for Operation. Exceptions of these requirements are stated in the individual Specifications.