Examination Report No. 50-206/OL-85-02

Facility Name: San Onofre Nuclear Generating Station, Unit 1

Docket No. 50-206

Examinations Administered at: San Onofre NGS, San Clemente, California, from July 9-10, 1985

Chief Examiner: [signature] for 8-23-85
G. W. Johnston, Operator License Examiner — Date Signed

Examiners: [signature] for 8-23-85
C. W. Shiraki, Operator License Examiner — Date Signed

[signature] for 8-23-85
L. F. Miller, Chief, Reactor Projects Section — Date Signed

Approved by: [signature] for 8-23-85
R. J. Pate, Chief, Operations Section — Date Signed

Summary:

Examinations on July 9-10, 1985

Written examinations were administered to three Reactor Operator candidates. Two operating examinations were administered to two of those candidates, all of whom were retakes, one receiving a waiver of the operating examination. Two candidates passed the written and operating examinations. One candidate failed the written examination and had received a waiver of the operating examination.

DETAILS

1. Persons Examined

   RO Candidates:

   Three candidates were examined. Two taking written and operating examinations, the other taking a written examination only with a waiver of the operating examination.

2. Examiners

   *G. W. Johnston, RV
   C. W. Shiraki, RV
   L. F. Miller, RV

   *Lead Examiner

3. Persons Attending the Exit Meeting

   Southern California Edison:

   J. R. Tate, Assistant Manager, Operations
   J. J. Wambold, Training Manager
   R. J. Mette, Supervisor of Operations Training
   M. J. Kirby, Nuclear Training Administrator

   NRC

   G. W. Johnston
   C. W. Shiraki

4. Written Examination and Facility Review

   Written exams were administered as follows:

   Three Reactor Operator - July 9, 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 candidates' responses.

5. Operating Examinations

   Facility walkthrough oral examinations were conducted July 10, 1985. No particular generic weaknesses were identified by the examiners during the course of the examinations. The sample size (three), and the fact that the candidates are retakes are factors in this evaluation.

6. Exit Meeting

   On July 11, 1985, the NRC representatives met with licensee personnel. Those individual candidates who clearly passed the oral examinations were identified. The NRC representatives discussed the overall performance of the candidates.

RESOLUTION OF FACILITY COMMENTS

REACTOR OPERATORS EXAMINATION

1.0 Facility Comment: Question 1.03

(a) The reviewer stated that the control group insertion limits were to maintain the reactor subcritical following a reactor trip.

Resolution:

(a) Comment accepted.

2.0 Facility Comment: Question 1.04

(c) The reviewer stated that axial offset would become more positive rather than negative.

Resolution:

(c) Comment accepted.

3.0 Facility Comment: Question 1.06

(c) The reviewer stated that the condition would be more accurately described as saturated rather than as subcooled.

Resolution:

(c) Comment Accepted.

4.0 Facility Comment: Question 1.08

(b) The reviewer stated that the present Technical Specification limits on cooldown rate for the pressurizer is 200 deg. F per hour.

Resolution:

(b) The author used the copy in place in Region V. Apparently this copy has not been updated. Comment will be accepted.

5.0 Facility Comment: Question 2.06

(b) The reviewer indicated that the reference used did not conform to the mode of operation that the switch position is used for. This is used primarily for heatup after condenser vacuum is established.

(c) Again, the reviewer indicated that this is the position normally used to allow heatup of the plant prior to establishing vacuum.

(d) This position is not normally used for normal heatup as was indicated by the reference, it would only be employed in unusual circumstances.

Resolution:

(b) Agreed. The key will be changed to reflect the procedures.

(c) Agreed. The key will be changed to reflect the procedures.

(d) For this case the examiner will accept the explanation that this position is not normally used.

6.0 Facility Comment: Question 2.09

(b) "Discusses reset and block, which is not the question. See SOI - 1.0 - 12 steps 1 and 2 for example of reset and terminate. Method of termination may vary depending on event."

Resolution:

(b) It appears to the examiner that the term "block" in the key has lead to confusion. The word will be changed to "reset" in the key. This should eliminate the confusion and conform to the procedure SOI - 1.0 - 12.

7.0 Facility Comment: Question 3.05

(c) The reviewer pointed out that the power range instruments utilize the intermediate range instruments compensated ionization chambers for some channels. It was also pointed out that the portion of the key for this question covered the topic of compensation that does not relate to the operation of the chamber.

Resolution:

(c) The examiner will accept both comments, the key will be adjusted for the point value.

8.0 Facility Comment: Question 3.06

(b) The reviewer commented that the term "low differential pressure" is not commonally used. The term typically used at the facility is "high reverse delta P".

Resolution:

(b) Comment accepted, will add "or high reverse delta P" to the key.

9.0 Facility Comment: Question 3.08

(a) "Should be 'Shutdown Rods Not Withdrawn'."

Resolution:

(a) The examiner recognizes that the term 'Shutdown bank deviation alarm' is not in the plant vernacular. Since this is so, the examiner will accept as an alarm function either case.

10.0 Facility Comment: Question 3.09

(a) For part 5 the reviewer felt that the feedback from Bistable TC-413B was appropriate to the question.

(b) Part 1 was also felt by the reviewer to indicate greater specificity than was appropriate as far as the Bistable TC-413B. In part 2 the reviewer felt the portion referring to the speed signal being fed through an auto/manual group selector switch to be more specific than required by the question.

Resolution:

For both (a) and (b) the comments are accepted.

11.0 Facility Comment: Question 3.10

(b) "The channel has no present control functions."

(c) The reviewer here again was concerned about the level of specificity of the question concerning the energization of a solenoid valve.

Resolution:

(b) The examiner will accept the answer of no control function.

(c) Agreed, will change key.

12.0 Facility Comment: Question 3.11

(a) The reviewers concern here was that the candidate may list each permissive and its purpose.

(b) For part 3 the reviewer stated that in this case the portion of the question pertaining to a decrease in power of 5% or greater being detected in any power range channel was not germain to the question.

Resolution:

(a) The examiner would concede that if a candidate answered in that fashion he would be answering the question appropriately, therefore will accept such a response.

(b) Agreed, will strike that portion.

13.0 Facility Comment: Question 4.01

(b) "For a pressure controller Shift Superintendent cannot approve exception."

Resolution:

(b) Agreed, will change key.

14.0 Facility Comment: Question 4.03

(a) The reviewer noted a typographical error.

Resolution:

(a) That will be corrected.

15.0 Facility Comment: Question 4.04

For all of the parts the reviewers felt the answers the candidates might give could be other events that would correspond to the symptoms listed.

Resolution:

The examiner agrees and will consider other events as answers so long as they correspond to the symptoms.

16.0 Facility Comment: Question 4.05

(b) The reviewer noted that there are other systems that are required by the T. S.

Resolution:

(b) If the candidate indicates that something in the T. S. would require the shutdown that will be sufficient as an answer.

U.S. Nuclear Regulatory Commission
Reactor Operator License Examination

Facility: SAN ONOFRE UNIT 1

Reactor Type: Westinghouse

Date Administered: July 9, 1985

Examiner: G. 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 sheets. Points for each question are indicated in parentheses 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 Category Value	Category
25.0	25.0			1. Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer and Fluid Flow
25.0	25.0			2. Plant Design Including Safety and Emergency Systems
25.0	25.0			3. Instruments and Controls
25.0	25.0			4. Procedures - Normal, Abnormal, Emergency, and Radiological Control
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)

$w = mg$

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

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

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

$\Delta E = 931\ \Delta m$

$I = I_o e^{-Ex}$

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

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

$Pwr = W_f \Delta n$

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

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

$TVL = -2.3/\mu$

$HVL = -0.693/\mu$

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

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

$SUR = 26.06/T$

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

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

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

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

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

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

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

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

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

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

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

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

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

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

$I_1 d_1 = I_2 d_2$

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

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

$\Sigma = \sigma N$

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

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)

SECTION ONE

REACTOR OPERATOR EXAMINATION

1.01 (2.5)

A refueling is being conducted following removal of many fuel assemblies to locate a foreign object in the bottom of the vessel. As the fuel assemblies are replaced, the neutron count rate increases as shown in Fig. 1.

(a) Define "M" in Figure 1. (0.5)

(b) What is the significance of the point where 1/M = 0? (0.5)

(c) Give one reasonable explanation for the concave downward shape of the plot. (0.5)

(d) The count rate after loading one fuel assembly is observed to increase from 100 cps to 125 cps. If the final keff was .99, what was the keff prior to loading the assembly? (Show your work) (0.5)

(e) At what count rate can the reactor go critical, in theory? (0.5)

(a) Multiplication ratio (fractional neutron increase per generation)

(b) Estimated fuel loading for criticality

(c) A larger percentage of neutrons are being detected as more fuel is loaded nearer the detector, or the detector was located close to an installed source assembly (either acceptable, may be others).

(d) keff prior to load = 0.9875 using C1(1-keff1) = C2(1-keff2)

(e) At any count rate above the installed source level

Ref: WNTC Lesson Notes, Phase 1, #A-8, Rev 1

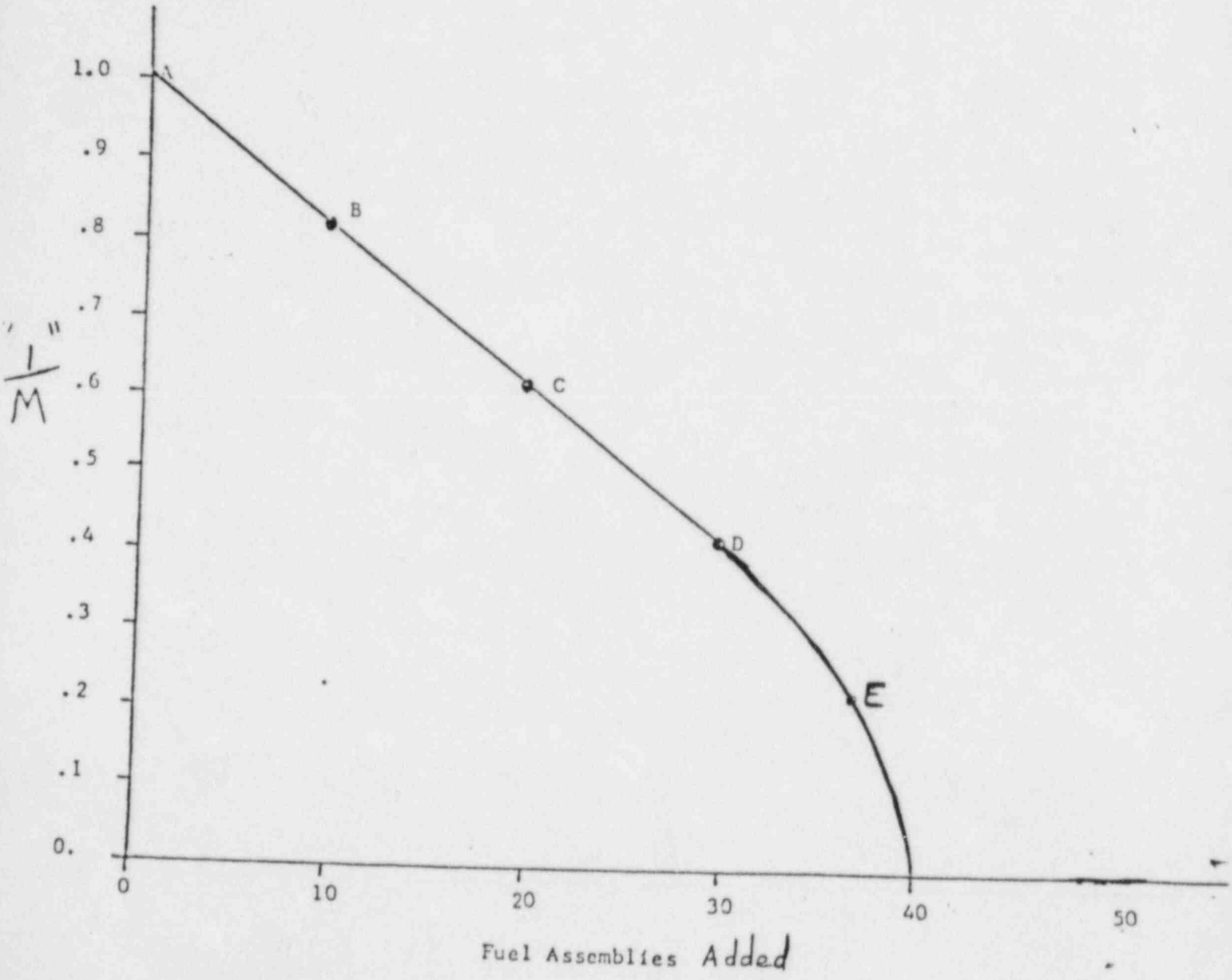


Figure 1: "1/M" Plot

1.02 (3.0)

Refer to the attached Figure 2

(a) The net amount of Xe-135 in the core is dependent on its rates of formation and removal. State the four most important processes which produce these rates for Xe-135. (2.0)

(b) The reactor trips after steady state operation at BOL from 75% power at 1200. When will the core be least reactive, assuming temperature and control rods remain unchanged after the trip? (0.5)

(c) How much reactivity will be added (or subtracted) due to Xe-135 at this time, compared to the time of the trip? (0.5)

(a) (1) production from fission (0.5), (2) decay from I-135 (and Te-135) (0.5), (3) decay to Cs-135 (0.5), and (4) neutron absorption (or burnout) (0.5)

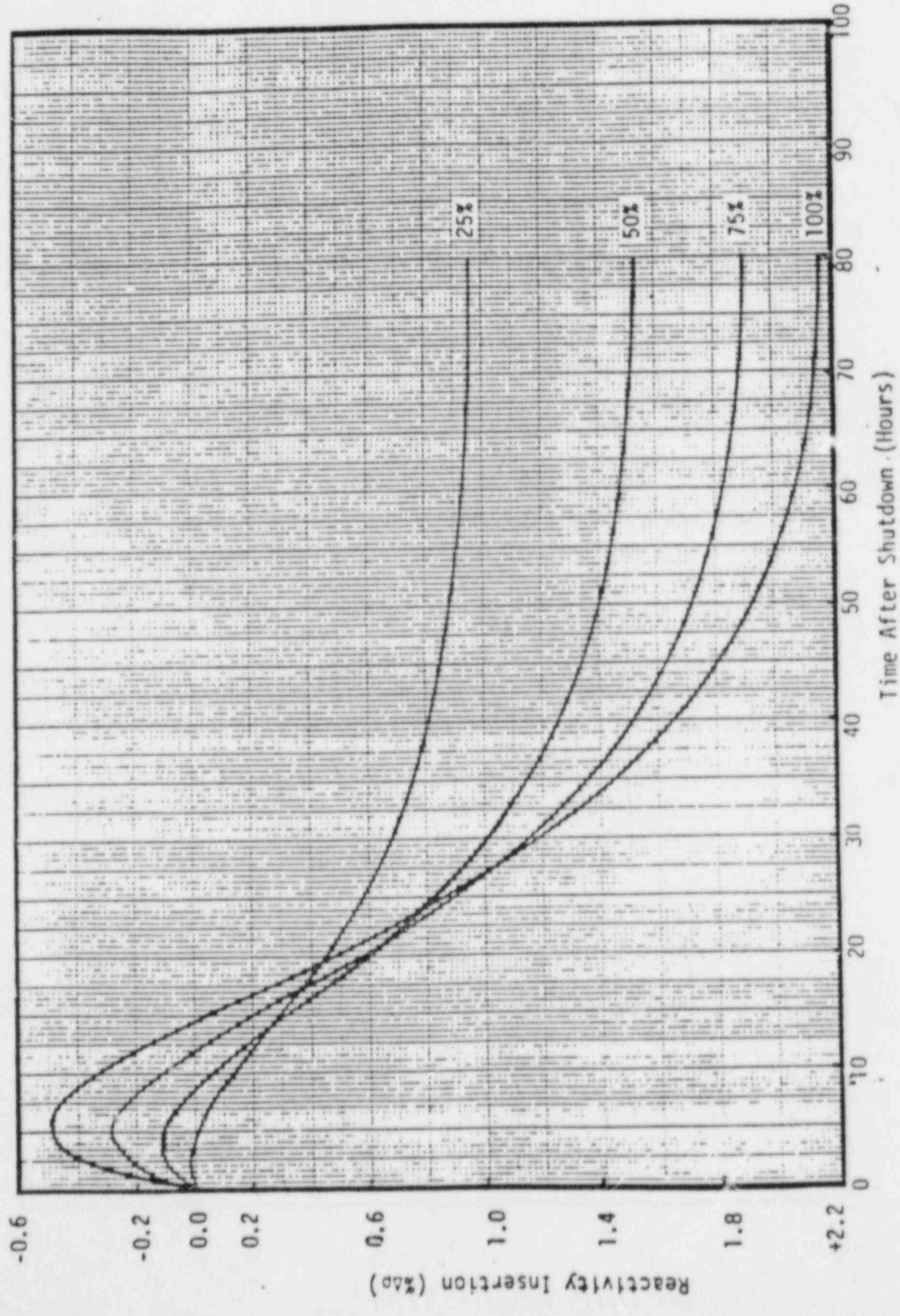
(b) 1700 (0.5)

(c) About .28% reactivitiy is subtracted at 1700 relative to the time of the trip. (0.5)

Ref: Academic program for Nuclear Power Personnel, Vol II, Chap. 4; Figure A-13, "Curve Book"

FIGURE 2

Reactivity Insertion Due to Xenon at BOL vs Time Following Plant Trip After Steady State Operation at Various Power Levels



1.03 (2.5)

(a) State three reasons for control group insertion limits. (1.5)

(b) State one reason why flux redistribution following a reactor trip from full power to hot zero power conditions may cause reactivity to increase. (0.5)

(c) Explain why the Technical Specifications limit the average burnup of the core to 21,000 MWD/MTU. (0.5)

(a) (1) To maintain an acceptable power distribution during normal operation (0.5), (2) Limit effect of a rod ejection accident (0.5), and (3) Maintain reactor subcritical during design accident (steam break) (0.5).

(b) At zero power, the moderator in the top of the core is relatively cooler than at full power (0.25), so with negative MTC (0.25), reactivity is higher.

(c) Reactivity addition accident analyses assume a maximum negative value of MTC. This value may be exceeded (0.25) if the burnup specification is exceeded because MTC becomes more negative with increasing burnup (0.25). 01

Ref: Tech. Spec. 3.9, 3.5.2, "Curve Book" pp. 44-45, 66-67

1.04 (2.75)

(a) Do the Technical Specifications allow operation at full power with control bank 2 eighty per cent withdrawn for a month in the middle of the core cycle? Explain your answer. (1.0)

(b) How will axial offset change immediately after this bank is rapidly borated out while maintaining power constant? Explain your choice. (1.0)

(c) Four hours after the rod motion, what changes, if any, do you expect to have occurred to axial offset? Explain your answer. (0.75)

(a) No. (0.5) Technical Specifications (3.5.2) require that the average position of CB 2 be at least 90% withdrawn (0.25) after 20% of the core cycle (0.25)

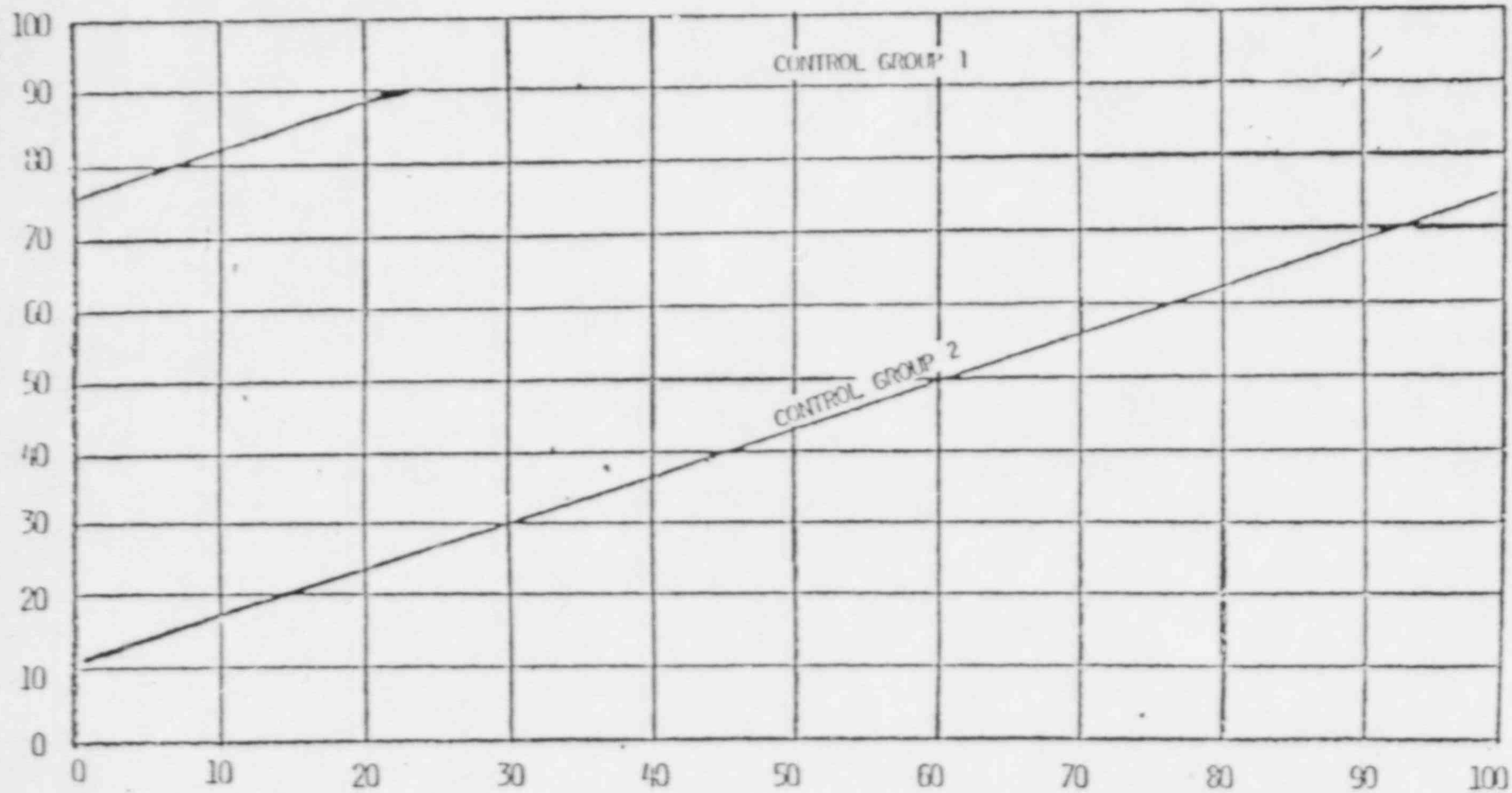
(b) Axial offset should become less negative or, perhaps, slightly positive (0.34), because the power distribution will shift from a peak in the lower two thirds of the core to a more symmetrical distribution (0.33) and because axial offset is (PTOP - PBOTTOM)/(PTOP + PBOTTOM). (0.33)

(c) Axial offset should become more negative (0.25) because xenon will build up in the reduced flux in the lower two thirds of the core (.25), and burnout in the increased flux in the upper one third of the core (.25) during this time period.

Ref: "Curve Book", TS 3.5.2

FIGURE 3

LIMITING CONDITION FOR OPERATION - CONTROL GROUP INSERTION LIMITS



Amendment No. 56

1.05 (2.5)

(a) The plant is operating at equilibrium steady state zero power conditions, all rods out, 2934 MWD/MTU when an alert operator notices that TAVE has increased 5 F in the last twenty four hours. Calculate, using Table 1, what change in boron concentration could have caused this increase. Show your work. (1.0)

(b) Considering each case independently, does differential boron worth increase or decrease as (1) Boron concentration increases, (2) Control rods are inserted, and (3) Moderator temperature increases. (1.5)

(a) Change in Boron Conc. = MTC X Change in Temperature X Boron Worth (0.5), so Boron concentration must have decreased by 7.0 X 5 X 0.15 = 52.5 ppm (0.5)

(b) (1) Differential boron worth decreases (less negative) as boron concentration increases (0.5)
(2) Differential boron worth decreases as control rods are inserted (0.5)
(3) Differential boron worth decreases as moderator temperature increases (0.5)

Ref: Academic Prog. for Nucl. Plant Personnel, p. 4-119, West. Nuclear Trng. Ctr. Reactor Theory Notes, Rev. III, p. 65

TABLE 1

END-POINT BORON CONCENTRATIONS, BORON WORTH, AND MODERATOR TEMPERATURE COEFFICIENTS

(No Redistribution Effects)

HZP

2934 MWD/MTU

Rod Condition	C_B (ppm)	Boron Worth (ppm/1000 pcm)	Moderator Temperature Coefficient (pcm/°F)
ARO	1355	150.36	-7.0
Control Group 2 In	991	147.43	-12.8
Control Groups 2+1 In	747	144.13	-17.2

Doppler Temperature Coefficient = -1.9 pcm/^{O}F

1.06 (1.5)

(a) The reactor is shutdown, and pressurizer pressure is 1425 psig. At what temperature will the reactor coolant system be 35 F subcooled, according to the Steam Tables? (0.5)

(b) A primary to containment atmosphere steam leak is occurring from the pressurizer with the pressurizer at 1600 psig. If containment pressure is 14.7 psia, what kind of steam (saturated, subcooled, or superheated) should be expected as soon as the steam from the break has depressurized to containment pressure. (0.5)

(c) For a leak identical to that in (b) above, if containment pressure is 50 psia, what kind of steam (saturated, subcooled, or superheated) should be expected as soon as the steam from the break has depressurized to containment pressure? (0.5)

(a) 553 F

(b) At 14.7 psia, the steam will be superheated (0.5),

(c) At 50 psia, it will be subcooled (0.5).

Ref: Steam Tables, Mollier Diagram

1.07 (2.5)

(a) Define net positive suction head (NPSH). (0.5)

(b) Explain why, for many centrifugal pumps, it is recommended that NPSH be some specified amount greater than zero. (0.5)

(c) Th~ reactor is shutdown with only one reactor coolant pump operating. Then, a second reactor coolant pump is started. Describe what happens to loop flow in every loop and to flow through the core, once flow stabilizes. Calculations are not required. (1.5)

(a) NPSH is the difference between the suction pressure and the saturation pressure of the fluid being pumped (Mathematical explanation also OK)

(b) To prevent cavitation.

(c) Loop flow in the originally operating loop drops slightly (0.25), loop flow in the newly operating loop rises to the same value (0.25), loop flow in the idle loop is zero or slightly reversed (0.5), and flow through the reactor increases to slightly less than twice its original value (0.5).

Ref: General Physics HTFF Fundamentals Sect III, Pt. B, Chap 1

1.08 (3.0)

Refer to Figure 4 (from Tech. Spec. 3.1.3)

(a) Explain why, in theory, the curves in the figure should be expected to shift to the right for the second 6.0 effective full power years, the third 6.0 effective years, etc. (1.5)

(b) What are the pressurizer heatup and cooldown rate limits set by the Technical Specifications? (0.5)

(c) The reactor coolant system is at 1600 psig and 300 F due to a rapid cooldown transient. In what direction should pressure and temperature be changed to minimize the chance of overstressing the reactor vessel? Explain. (1.0)

(a) The shift in the curves corresponds to a change in RTNDT (a measure of the fracture toughness of the vessel, or of its degree of embrittlement) during that period. (0.75) RTNDT is expected to increase with reactor vessel fluence, according to the Tech. Specs., which will steadily increase during these intervals, since fluence is neutron flux times time. So the curves shift to the right. (0.75)

(b) 100 F/ hr heatup (0.25) and 200 F/ hr cooldown (0.25)

(c) Pressure should be decreased while maintaining temperature constant (0.5). This will minimize thermal stress, and therefore total stress on the reactor vessel (0.5).

Ref: TS 3.1.3

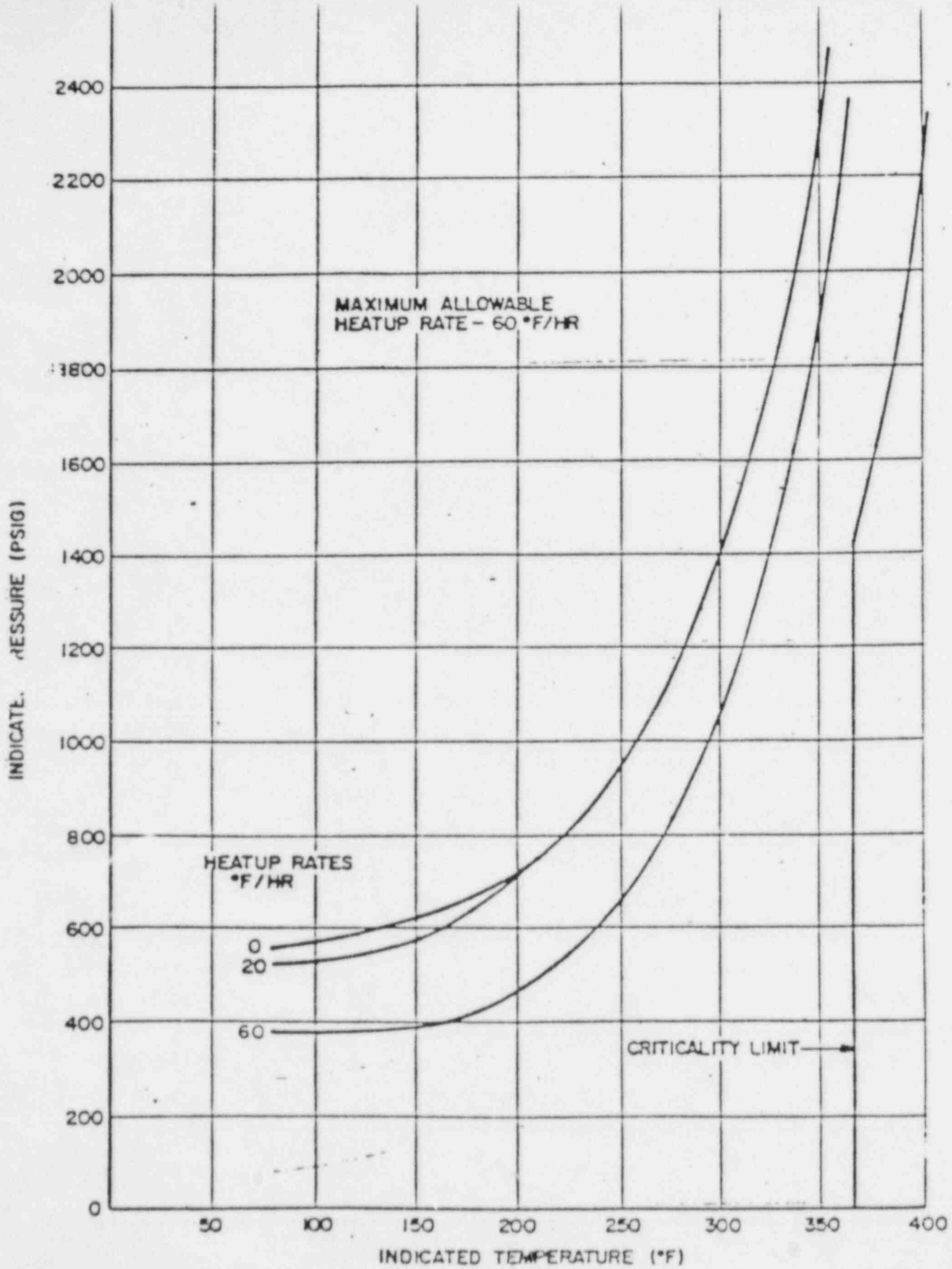


FIGURE 4 . SCE REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR FIRST 6.0 EFFECTIVE FULL POWER YEARS.

T_{ERROR} = 10°F, P_{ERROR} = 60 PSIG

Change No. 1
Date: 4/12/

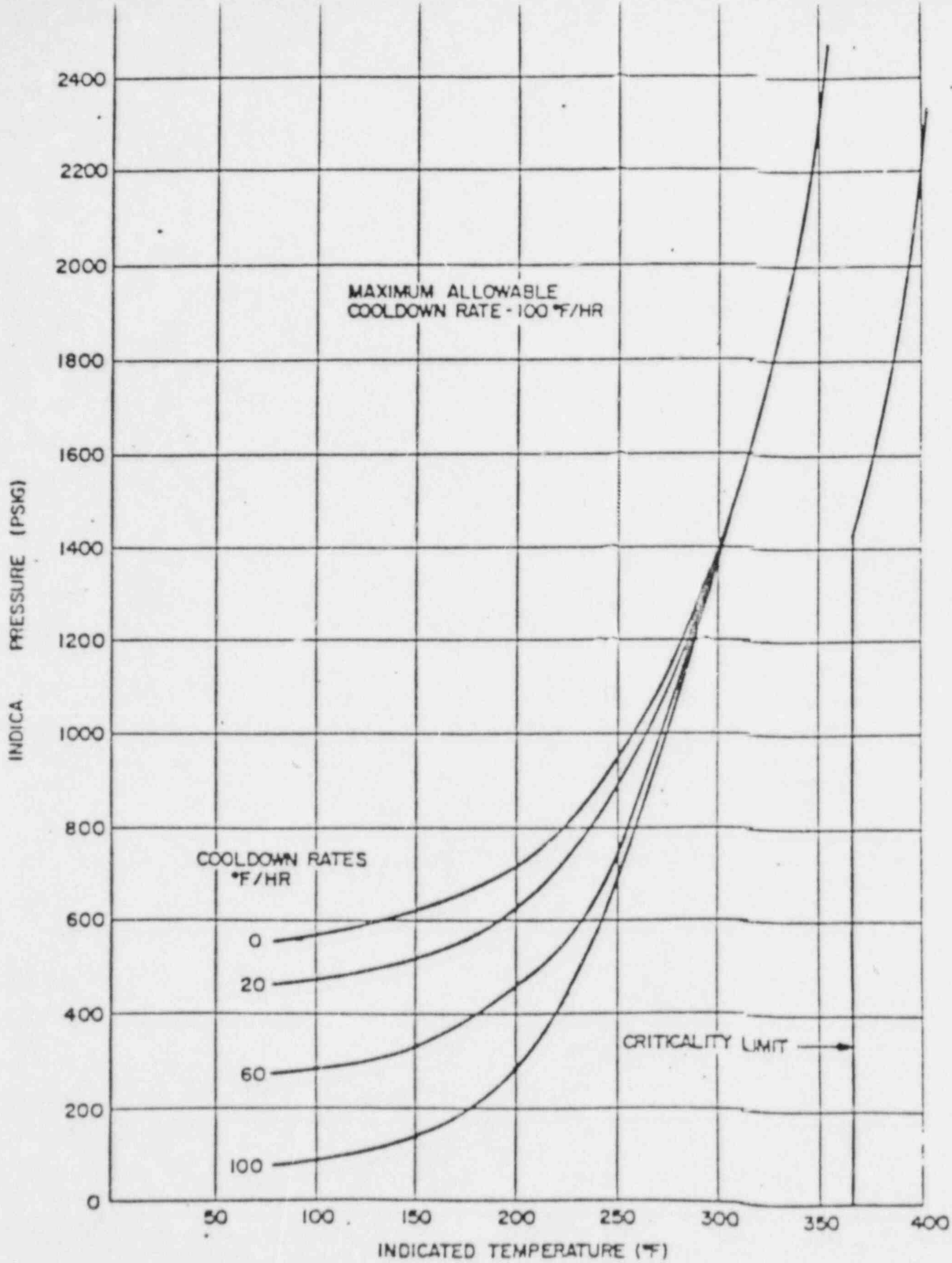


FIGURE 3.1.3b SCE REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR FIRST 6.0 EFFECTIVE FULL POWER YEARS.

T_{ERROR} = 10°F, P_{ERROR} = 60 PSIG

Change No. 14
Date: 4/12/74

1.09 (2.25)

(a) State three conditions which indicate that natural circulation of the reactor coolant system is occurring (assume no forced circulation). (1.75)

(b) If a DNB ratio of 1.00 exists in the core, what is the probable effect, if any? (0.5)

(a) (1) RCS DELTA T < or = to Full Load DELTA T (0.5)
(2) RCS or CET temperatures constant or decreasing (0.5)
(3) Steam generator pressures constant or decreasing at a rate equivalent to the rate of decrease of RCS temperatures (0.5) while maintaining steam generator level constant with continuous auxiliary feedwater (0.25).

(b) Failure of some fuel cladding (0.5)

Ref: TS 2.1,, Westinghouse Mitigating Core Damage Training Manual, p. 51

1.10 (2.5)

The reactor is operating at 85 % reactor power when Control Bank 2 is shimmed out 10 steps. Sketch (on the attached Figure 5) reactor power, fuel temperature, T HOT, T COLD, and T STEAM versus time until new final values are reached. Numerical estimates are not required, but carefully show the relative relationships of peaks and trends of these parameters. Assume no reactor trip occurs.

END OF SECTION ONE

See Attached Figure (0.5 pts. per variable)

Ref: Academic Program for Nuclear power Plant Personnel, Volume III, Nuclear Power Plant Technology, Chapter 3, Section C

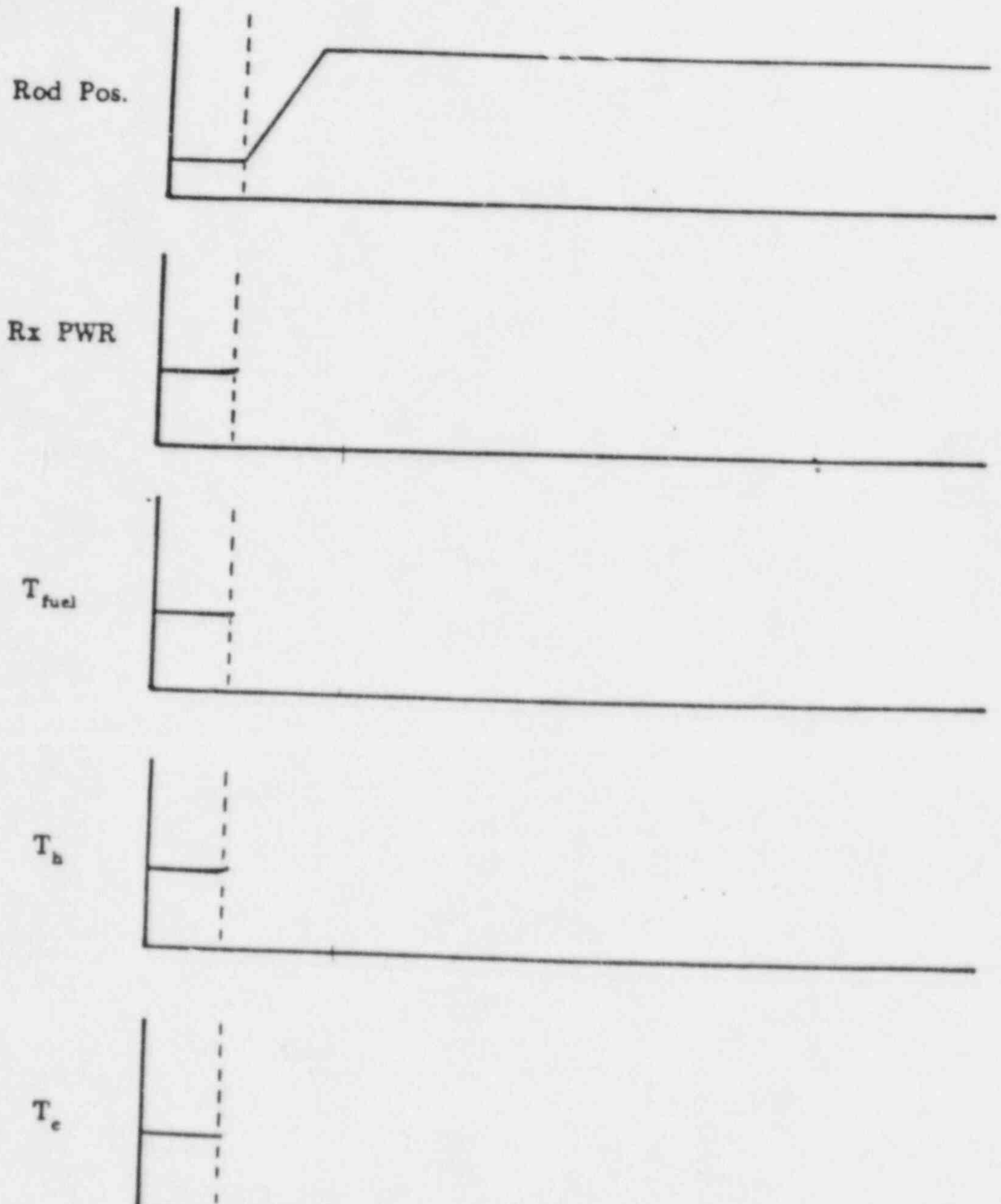


FIGURE 5

1.10

KEY

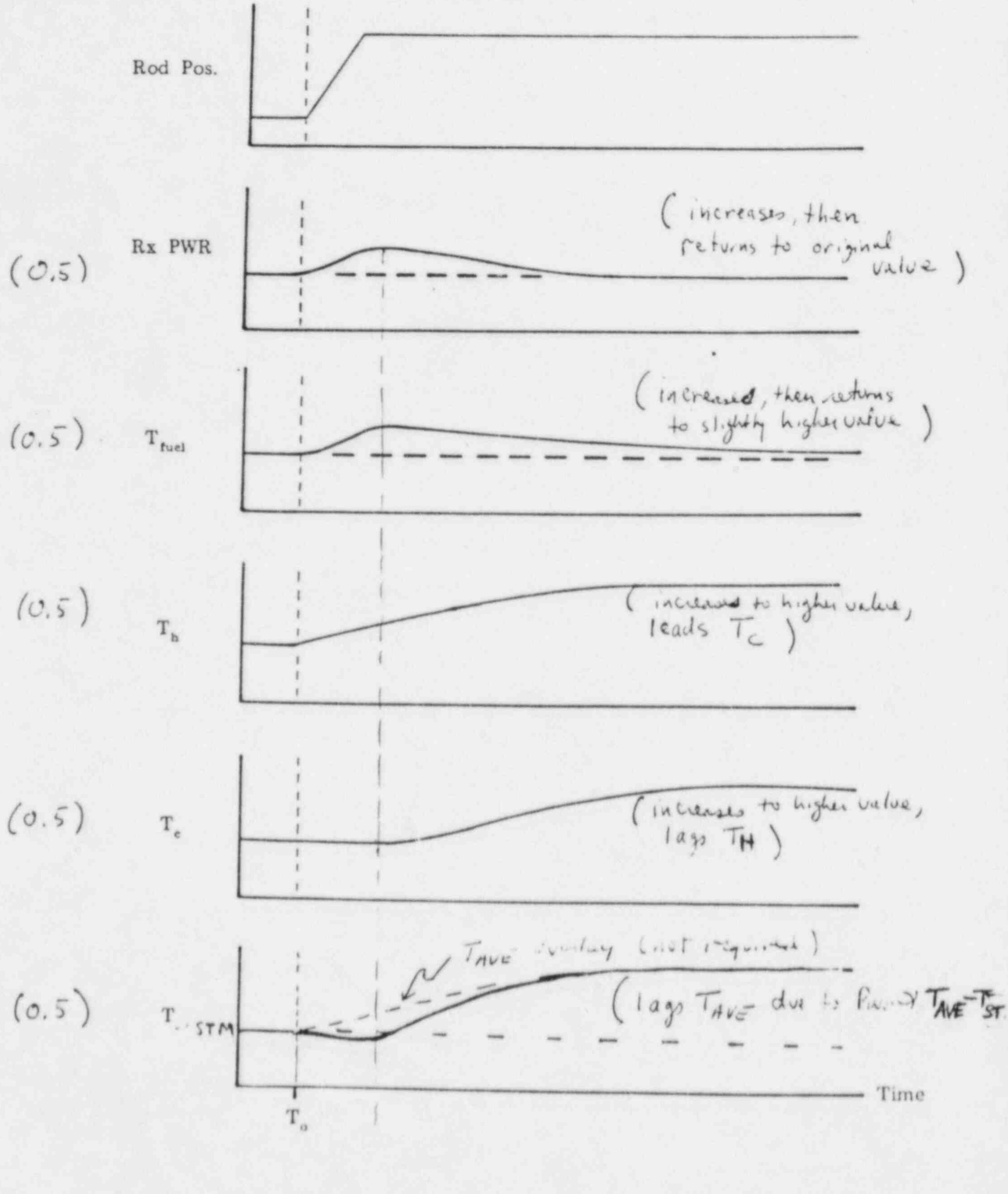


Figure Control Rod Withdrawal

SECTION 2

REACTOR OPERATORS EXAMINATION

Plant Design - Including Safety and Emergency Systems

2.01 (2.0)

For the Reactor Coolant Pump Seal Water Circuit:

(a) What is the primary function of the Reactor Coolant Pump Seal Water Circuit in conjunction with the pump seals it provides? (1.0)

(b) In the event of a failure (loss of seal water flow) how will seal water be supplied provided the loss of flow is not froma ruptured line? (1.0)

2.01 Answer:

(a) To effectively prevent the leakage of reactor coolant along the Reactor Coolant Pump shaft. (1.0)

(b) The Test pump in the CVCS is sized to supply adequate flow to all three RCP's. (1.0)

Reference: Study Guide No. 4, pages 13, 14, 17, and 24.

2.02 (2.0)

The pressurizer has two code safeties and two power operated relief valves.

(a) During operation while at power it is determined that there is leakage coming from a relief valve on the pressurizer. How can an operator determine which specific valve is leaking among the four valves? (1.0)

(b) If the leakage was determined to be past a power operated relief valve, and exceeded the Technical Specification limits, would the plant be required to be shutdown? Explain. (1.0)

2.02 Answer:

(a) Each safety valve has its own temperature element, an indication of higher than ambient temperature would allow the operator to determine the valve. (0.5)

The case for the power operated reliefs is different, they have a common temperature element. However they have individual isolation valves, which can be closed to determine which valve is leaking. (0.5)

(b) No (0.5). The power relief valves have block valves and can be isolated so that operation can continue (0.5).

Reference: Requalification Exam No. 0831, page 7.

2.03 (2.0)

Regarding the Pressurizer:

(a) Give three reasons why a 1 gpm flow rate is maintained around the spray valves into the Pressurizer. (1.5)

(b) The Pressurizer level program has a ramp function from 25% at no load to 37% at full load. Why is pressurizer level required to vary as a function of power? (0.5)

2.03 Answer:

(a) 1) To reduce thermal stresses and thermal shock when the spray valves open. (0.5)

2) To help maintain uniform water chemistry (boron concentration). (0.5)

3) And to maintain the desired temperature in the surge line. (0.5)

(b) To allow for surges into and out of the Pressurizer during load transients. (0.5)

Reference: Study Guide No. 16.

2.04 (3.0)

For the Diesel Generators:

(a) List six trips for the Diesel Generator engines that are in service for a manual start. (3.0)

2.04 Answer:

(a) Any 6 (0.5) each.

1) Engine Overspeed.
2) Generator Differential.
3) High Crankcase Pressure.
4) Low-low Engine Oil Pressure.
5) Low-low Turbo Oil Pressure.
6) High-high Jacket Water Temperature.
7) High Vibration Engine or Turbo.
8) High-high Lube Oil Temperature.
9) High Main Bearing Temperature.

Reference: Study Guide No. 98, page 9.

2.05 (3.5)

For the Reactor Coolant Pump No. 1 Seals:

(a) An operator is about to start a Reactor Coolant Pump, system pressure is 375 psig. An annunciator window is illuminated "No. 1 SEAL LO FLOW", can the pump be started? Explain. (1.0)

(b) What 5 conditions must be met to start a Reactor Coolant Pump? (2.5)

2.05 Answer:

(a) Yes (0.5). At pressures below 1000 psig the No. 1 seal flow may be too low to clear the alarm (a flow of at least 0.3 gpm is considered adequate for the condition) (0.5).

(b) 1) Oil lift pump permissive light on. (0.5)
2) No.1 seal delta-P greater than 275 psig. (0.5)
3) No.1 seal leakoff greater than 0.25 gpm. (0.5)
4) Thermal barrier delta-P about 20 inches. (0.5)
5) Reactor Coolant System Pressure greater than or equal to 350 psig. (0.5)

Reference: Study Guide No. 3, page 12.

2.06 (4.0)

On the "J" console there is a 5 position switch associated with the Steam Dump System. For the following switch positions under <u>what</u> conditions is each position used? And <u>how</u> does the system operate in each position?

(a) Position 2 "Automatic" (1.0)

(b) Position 3 "Pressure Control - Atmosphere and Condenser" (1.0)

(c) Position 4 "Pressure Control - Atmosphere Only" (1.0)

(d) Position 5 "Pressure Control - Condenser Only" (1.0)

2.06 Answer:

(a) Normal operating position when above 20% power (0.5). System will operate when a load reduction of 10% to 40% occurs and Tave is greater than Tref by 5 degrees F (0.5)

(b) The switch is placed here when Automatic control is not availiable (0.5). Strictly by pressure control only (from PC-418A pressure controller.) (0.5).

(c) This is used when the condenser is not availiable (0.5). Pressure control is as in (b) above (0.5).

(d) Used for normal startup and cooldown, and is placed here for normal operation when in pressure control (0.5). Pressure control as in (b) above (0.5).

Reference: Study Guide No. 53, pages 7 and 8.

2.07 (3.5)

For the Pressurizer Level Control System supply the associated alarm and/or control actuations for the following setpoints.

(a) 70% level. (0.5)

(b) +4% of programmed level. (1.0)

(c) -4% of programmed level. (0.5)

(d) 10% level. (1.5)

2.07 Answer:

(a) High Pressurizer water level Reactor trip. (0.5)

(b) High Pressurizer water level alarm. (0.5)
Heaters off. (0.5)

(c) Pressurizer low level alarm. (0.5)

(d) Pressurizer low-low alarm. (0.5)
Heaters off. (0.5)
Isolation of letdown valve. (0.5)

Reference: Study Guide No. 11, page 16.

2.08 (2.0)

For the Containment Spray System and the Containment Spray Actuation System (CSAS):

(a) What is the sequence (order) of events when an actuation signal (CSAS) comes in to start Containment Spray? (1.0)

(b) What setpoints and logic must be satisfied to initiate an actuation of Containment Spray from the CSAS (automatic only)? (1.0)

2.08 Answer:

(a) Refueling Water Pumps start first. (0.25)

The Spray Header valves open. (0.25)

The Hydrazine injection pump stars. (0.25)

The Hydrazine valves open to admit hydrazine to the Containment Spray header. (0.25)

(b) 2/2 Safety Injection (1/1 sequencer per train) and 2/3 High Sphere pressure - 10psig. (1.0)

Reference: Study Guides No.s 16 and 17.

2.09 (3.0)

Concerning the Safeguards Load Sequencing System (SLSS):

(a) The SLSS recieves six signals from various sources including one from a test switch. What are the remaining five signals? (1.5)

(b) What actions must an operator take to terminate and reset an Automatic actuation of Safety Injection? (1.5)

2.09 Answer:

(a) 1) Pressurizer Pressure (0.3)

2) Containment Pressure (0.3)

3) Undervoltage signal from both 4160 busses (1C and 2C). (0.3)

4) Safety Injection Block (0.3)

5) Standby Diesel Generation (Voltage frequency signal and circuit breaker status signal.) (0.3)

(b) First he must block the automatic actuation signal by depressing the reset switches on the sequencer surveillance panel (0.5). and then in order turning off the feedwater pumps (0.5) and the safety injection pumps (0.5).

Reference: Study Guide No. 17 pages 11, 12, and 15.

END OF SECTION 2

SECTION 3

REACTOR OPERATORS EXAMINATION

Instrumentation and Controls

3.01 (2.0)

Regarding the steam generator level control system:

(a) The steam generator level control system is commonaly referred to as a three element control system, what are those three elements? (1.5)

(b) If there are three elements used for control in this system, why then is steam pressure (not one of the three 'elements') also an input to the system? (0.5)

3.01 Answer

(a) 1. Feedwater flow. (0.5)

2. Steam flow. (0.5)

3. Steam generator level. (0.5)

(b) To provide input for density compensation in the Steam Flow Computer. (0.5)

Reference: OT-1045 and OT-1064 "Steam Generator Water Level Control", Study Guide No. 62.

3.02 (1.5)

For the diagram (figure 3.1) on the following page describe:

(a) Which side of the manometer will rise when a fluid is flowing in the direction shown? (0.5)

(b) What relationship (i.e. equation) is used to determine the flow rate in the pipe? (0.5)

(c) What other type of instrumentation is normally provided in the plant to perform this function (of the manometer)? (0.5)

3.02 Answer

(a) The leg connected to the restriction of the venturi. (0.5)

(b) The Bernoulli equation (Flow rate is proportional to the square root of the differential pressure). (0.5)

(c) Typically a Bourdon tube type transmitter (D/P cell). (0.5)

Reference: General Physics "Heat Transfer Thermodynamics and Fluid Flow Fundamentals"

3.03 (3.0)

SECTION 3

REACTOR OPERATORS EXAMINATION

Instrumentation and Controls

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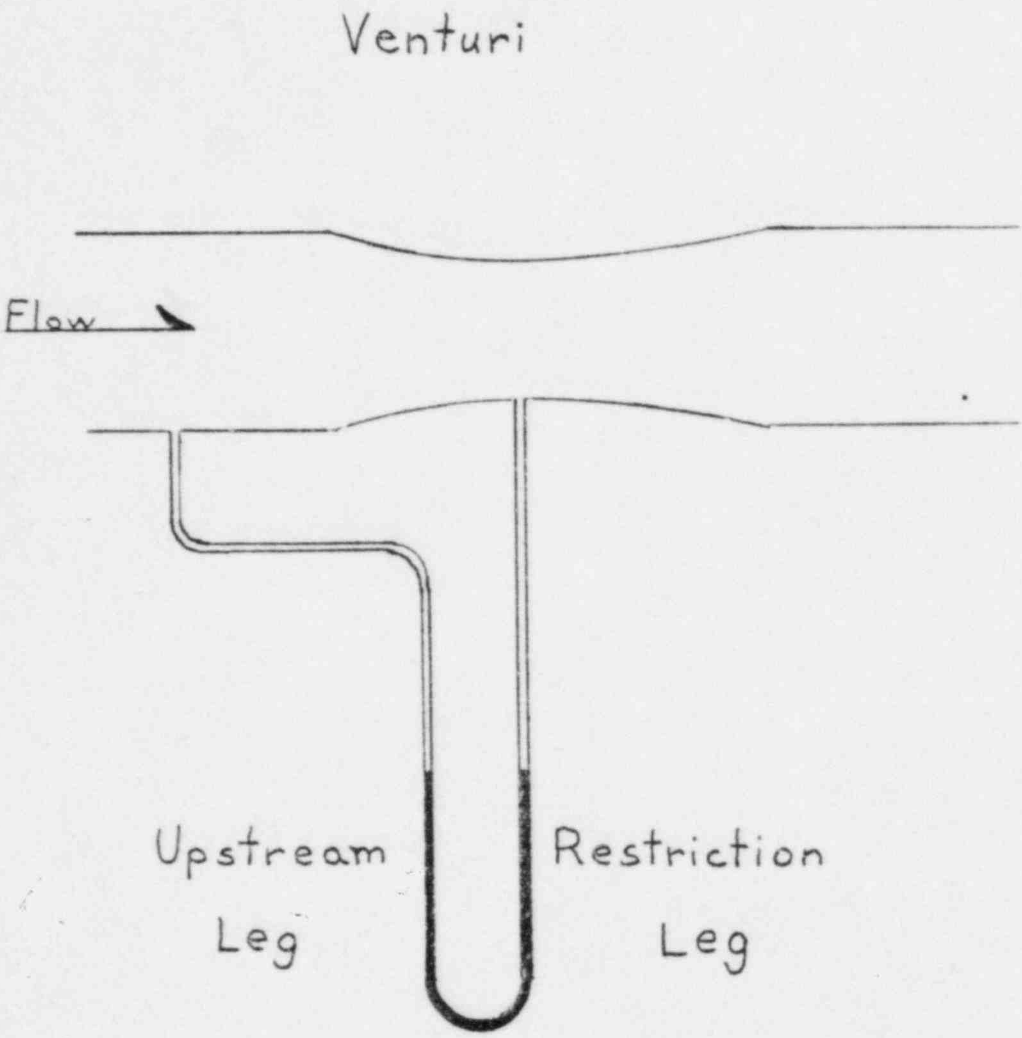


Figure 3.1

3.03 (3.0)

For the curve on the following page (Figure 3.2) identify the following parts of the curve:

(a) I (0.5)

(b) II (0.5)

(c) III (0.5)

(d) IV (0.5)

(e) V (0.5)

(f) VI (0.5)

3.03 Answer

(a) Recombination (0.5)

(b) Ionization (0.5)

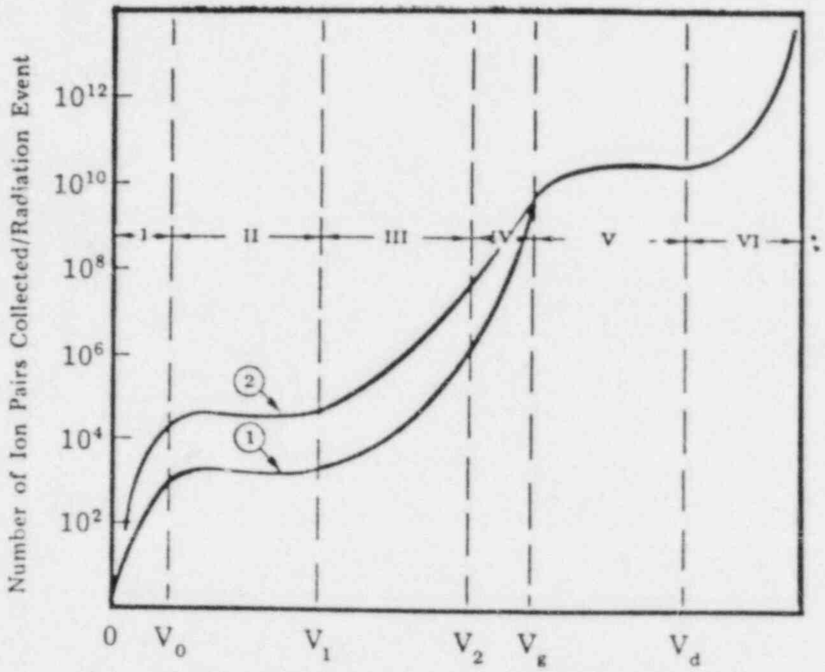
(c) Proportional (0.5)

(d) Limited Proportional (0.5)

(e) Geiger-Mueller (0.5)

(f) Continous Discharge (0.5)

Reference: General Physics Volume II, Chapter 3, Section E



Curve 1: Radiation event of lower specific ionization.
Curve 2: Radiation event of higher specific ionization.

Figure 3.2

3.04 (2.5)

During solid plant operations the following alarms come in:

"OMS Hi Pressure"

"Pressure Transient in Progress"

(a) What condition would actuate these alarms (i.e. System setpoints)? (0.5)

(b) What function does the Overpressure Mitigation System serve? (0.5)

(c) After securing charging flow as the first immediate action taken, what valves must an operator check to insure are open to complete the immediate actions in response to these alarms? (1.5)

3.04 Answer:

(a) The "OMS Hi Pressure" alarm comes in at 480 psig. (0.25)

"Pressure Transient in Progress" at 500 psig. (0.25)

(b) To protectect the RCS from potential overpressurization during solid operations. (0.5)

(c) Insure LCV-1112, letdown isolation, open. (0.5)

PCV-1105, letdown control valve open. (0.5)

And, CV-525 and CV-526, containment isolation valves, are open. (0.5)

Reference: Line Diagrams, SO1-2.1-11 "Overpressurization Mitigation System Actuation", and Study Guide No. 6

3.05 (3.0)

For the following nuclear instrumentation detectors discuss the operation of the <u>detector</u> <u>chamber</u>:

(a) Intermediate Range Instruments. (1.0)

(b) Source Range Instruments. (1.0)

(c) Power Range Instruments. (1.0)

3.05 Answer:

(a) The intermediate range instrument chamber is a Compensated Ion Chamber (0.5). This chamber is essentially two chambers, one coated on the inside with a Boron coating (0.25), the other not. The chambers are wired together such that the outputs cancel out the current that incident gamma radiation contributes leaving a current output representative of the neutron population (0.25)

(b) The chamber is filled with Boron Trifluoride gas (0.5). Thermal neutrons are absorbed by the Boron 10 atoms which then disintegrate into Lithium 7 atoms and emit high energy alpha particles (0.25). This generates an ionization that is collected by the electrodes producing a pulse output (0.25) proportional to the number of neutrons. (Discrimination is accomplished in the circuitry in the channel drawer.)

(c) This channel uses an Uncompensated Ion Chamber (0.5). Which is coated on the inside with a Boron coating (0.25). Because of the power range in which the instrumentation operates there is no need to compensate for gamma radiation because it presents a very minor contribution (0.25).

Reference: Study Guide No. 32: General Physics Volume II, Chapter 3, Section E.

3.06 (2.0)

For the following components how would the failure affect their indication and why?

(a) The failure of a Delta-P diaphragm on the pressurizer level detection system. (1.0)

(b) A rupture of the reference leg on the Steam Generator level detection system. (1.0)

3.06 Answer:

(a) The failure of a diaphragm in this case would present a condition where there is no indicated differential pressure (0.5). This would cause an indication of high level (0.5).

(b) If a reference leg ruptured the indication would be of a low differential pressure (0.5). This would also indicate a high level (0.5)

Reference: Requalification Exam 940, page 13, question 10.

3.07 (1.5)

For the Main Turbine:

(a) Besides the Overspeed trip that is provided what are three other trips associated with the Turbine. (1.5)

3.07 Answer:

(a) Any three (0.5) each.

Low bearing oil pressure.
Solenoid Trip.
Thrust bearing trip.
Low vacuum trip.

Reference: Study Guide 51, pages 29 and 30.

3.08 (3.0)

For the following alarm indications indicate what condition exists for the alarm to be in.

(a) Shutdown bank deviation alarm. (1.0)

(b) Control bank deviation alarm. (1.0)

(c) Rod bottom light on for one control rod. (1.0)

3.08 Answer:

(a) The alarm will come in whenever a rod in a shutdown bank deviates below a preset position. (1.0)

(b) In this case the alarm would come in because a rod has deviated by more than a preset amount from the bank position. The bank positon comes from the demanded indication circuit from the Digital Detection System. (1.0)

(c) This would come in if a rod dropped below approximately 25 steps. (1.0)

Reference: Study Guide 13, page 2.

3.09 (3.0)

For tne Control Rod Drive Summing Computer:

(a) The computer receives inputs from 5 sources, name 4 of those sources. (2.0)

(b) The output from the computer goes to two places, what are those places and what function does the signal that is provided perform? (1.0)

3.09 Answer:

(a) Any 4 (0.5) each:

1) Average Tavg.

2) Tref.

3) Primary pressure.

4) Nuclear flux.

5) Feedback from Bistable TC-413B indicating demanded rod postion.

(b) 1) Bistable 413B (0.25), to move rods in or out (0.25).

2) Rod speed controller (0.25). Which produces a demanded speed to be fed to the rod drive system through an auto/manual group selector switch (0.25).

Reference: Study Guide 12, page 4.

3.10 (1.5)

For the following Operational Radiation Monitoring System channels what type of detector is used, and what control funtions do they provide? (Do not provide alarms.)

(a) Channel R-1211 - continuous air sample from either the vapor container or the plant stack. (0.5)

(b) Channel R-1214 Main Stack Gas Monitor. (0.5)

(c) Channel R-1216 Steam Generator Blowdown-Liquid Sample monitor. (0.5)

3.10 Answer:

(a) Scintillation detector. (0.25)

No direct control function. (0.25)

(b) Geiger-Mueller detectors. (0.25)

High level alarm deenergizes SV-99 which closes to stop all gas discharge form gas decay tanks. (0.25)

(c) Scintillation detector. (0.25)

High radiation level alarm energizes solenoid valve (SV-84), causing CV-100, 100A, and 100B to close. This stops discharge to blowdown tank and to circulating water outfall. (Candidate only has to indicate that discharges to blowdown tank and outfall are isolated.) (0.25)

Reference: Study Guide 31; pages 2, 4, 8, 9, 11, and 13.

3.11 (2.0)

The reactor Protection System is provided with 8 "Permissive Circuits":

(a) What are the purposes of these permissive circuits? (0.5)

(b) What function do the following permissive circuits provide?

1) Permissive Circuit No. 5 - Shutdown Margin Low Alarm (0.5)

2) Permissive Circuit No. 4 - Steam Dump Automatic Mode Cutout (0.5)

3) Permissive Circuit No. 3 - Rod Drop Rod Stop. (0.5)

3.11 Answer:

(a) To provide increased reliability, and protection, and to provide compatibility between control systems. (0.5)

(b) 1) This would warn operators of the possibility of inadequate shutdown margin as determined by the shutdown margin computers. (0.5)

2) This would block automatic operation of the steam dumps when Tave-Tref setpoint deviations are reached unless the MWe load decrease is greater than 45 MWe in less than 10 seconds. (0.5)

3) This would prevent automatic rod withdrawal and would initiate a turbine load limit runback to 70% power when a decrease of 5% or greater in power is detected in any power range N.I. channel. (0.5)

Reference: Study Guide 12, pages 7, and 8.

End of Section 3

SECTION FOUR

REACTOR OPERATOR EXAMINATION

4.01 (2.5)

(a) Under nonemergency conditions, give three general types of activities for which procedures are required to be used. (An example would be an evolution affecting plant reliability). (1.5)

(b) It is desired to perform minor troubleshooting of a pressure controller. One valve will be shut to perform the troubleshooting, and then it will be reopened. No in-place procedure is required for this activity. May this activity be performed without equipment control? Explain. (1.0)

(a) Any three of the following (0.5 each, max. of 1.5): manipulation of all safety related equipment, major (complex) evolutions, infrequently performed evolutions, manipulations of major (nonsafety) equipment, or evolutions affecting plant plant safety.

(b) In general, no, since equipment control is required for most evolutions not controlled by procedures. (0.5) However, the Shift Superintendent may approve a minor maintenance item, not important to safety, without equipment control. (0.5)

Ref: SOI-14 -42, SOI-14-12 pp. 8-10

4.02 (3.6)

List the twelve immediate actions for a reactor trip with safety injection initiation. Valve numbers and positions, and setpoints are not required.

(1) Verify reactor trip
(2) Verify turbine trip
(3) Verify electrical busses energized
(4) Check if SI initiated
(5) Verify RCP's tripped
(6) Verify SI system valve alignment
(7) Verify SI system pumps running
(8) Verify Charging pump suction aligned to RWST
(9) Verify SI flow
(10) Verify containment isolation
(11) Verify Auto AFW Initiation
(12) Check Containment Pressure (0.3 pts. each)

Ref: SO1-1.0-10

4.03 (3.0)

(a) List the four highest priority critical safety functions. (2.0)

(b) Explain how red and orange conditions in critical safety function trees require different priority sequences. (1.0)

(a) Maintenance or control of subcriticality, core cooling, reactor coolant inventory, and heat sink (0.5 pts. each, correct order not required)

(b) Both require departure from any other EOI in use, but a red condition requires immediate implementation of the EOI (0.5), whereas for an orange condition it is required to complete the current pass through all of the status trees (to look for higher priority red conditions) (0.5).

Ref: SO1-1.0-1 pp. 11-14

4.04 (2.0)

According to the Emergency Operating Instructions, what events should the following symptoms correspond to?

(a) Reactor Trip and a Steam/Feed Mismatch Reactor Trip Alarm ON (0.5)

(b) Reactor Trip, Safety Injection, Electrical Power Available, Emergency Systems Operating as Required, and Steam Generator 'A' level is rising uncontrolled. (0.5)

(c) Six core exit thermocouples indicating 700 F (0.5)

(d) Same as (b) except steam generator levels are under control and containment radiation monitor R1255 is above its alarm setpoint. (0.5)

(a) Loss of Secondary Coolant

(b) Steam Generator Tube Rupture

(c) Potential Loss of Core Cooling

(d) Loss of Reactor Coolant

Ref: S01-1.0-30, S01-1.0-40, S01-1.2-2, S01-1.0-20

4.05 (2.0)

Abnormal Operating Instruction SO1-2.1-10, Loss of Component Cooling Water, requires that for a total loss of component cooling water which cannot be promptly restored, a natural circulation cooldown be performed using the test pump only, to control pressurizer level.

(a) Explain why natural circulation is required by the procedure. (1.0)

(b) Explain why a cooldown is required by the procedure. (0.5)

(c) Explain why the use of letdown to control pressurizer level is not permitted. (0.5)

(a) The total, unrestored loss of CCW requires a reactor trip (0.5) and trip of all RCPs (0.5), so only natural circulation is available to cool the core.

(b) A cooldown is implied by Tech. Specs. (3.0 and 3.1.2.E) with no RCPs operating.

(c) There will not be adequate CCW flow to the nonregenerative heat exchanger to cool letdown.

Ref: TS 3.0, 3.1.2.E, SO1-2.1-10

4.06 (2.5)

(a) Upon confirmed loss of which vital or utility bus (or buses) is an immediate reactor trip NOT required? (0.9)

(b) Which AC Distribution buses are required to be OPERABLE by the Technical Specifications, assuming the plant is in Mode 1, and NOT in any Action statements of the Specifications? (1.1)

(c) What equipment is required other than fuel tanks by the Technical Specifications for the emergency diesel generators. (0.5)

(a) Vital buses 3A, 5 and 6 (0.3 each)

(b) 4160 Buses 1C and 2C, 480 Buses 1, 2, and 3, and Vital Buses 1, 2, 3, 3A, 4, 5, and 6. (0.1 pts each)

(c) Fuel transfer pumps

Ref: SO1-2.6-3, TS 3.7

4.07 (3.0)

The plant is being started up from hot standby to minimum load.

(a) What rod positions are procedurally required for criticality unless otherwise directed by the Shift Superintendent? (1.0)

(b) When, according to the procedure, should criticality be anticipated? (1.0)

(c) What is the limit on Start Up Rate during the startup? (0.5)

(d) What is the limit on turbine backpressure prior to rolling the turbine? (0.5)

(a) Shutdown Banks 1 and 2 and Control Bank 1 at 318 steps, Control Bank 2 at 100 steps (0.25 pts each).

(b) At any time when control rods are being withdrawn, or when boron dilution is in progress (0.5 pts each)

(c) 1.0 dpm

(d) 5.5" Hg

Ref: SO1-3-2

4.08 (2.0)

(a) Define a high radiation area. (1.0)

(b) What are the SONGS quarterly Administrative Limits for radiation exposure? (1.0)

(a) A high radiation area is any area which is accessible (0.1) in which a major portion of a person's body (0.1) could receiv in excess of 100 mrem in one hour (0.8)

(b) 900 mrem whole body (0.34), 3750 mrem skin (0.33), and 4700 mrem extremities (0.33)

Ref: SONGS Radiation Training Handout

4.09 (2.5)

With respect to boration and dilution of the reactor coolant system:

(a) Why is boric acid injection pump discharge flow required to be limited to 7.5 gpm? (0.5)

(b) Why is it required to have an RHR or RCP operating while borating? (0.5)

(c) What is the required setting for the Boric Acid Blend System when not diluting or borating? (0.5)

(d) How is pressurizer boron concentration maintained within specified tolerances of the reactor coolant system boron concentration? (1.0)

(a) To prevent pump cavitation

(b) To provide adequate mixing of the boron

(c) AUTO MAKEUP

(d) Placing pressurizer heaters in MANUAL (0.5), causing automatic spray operation to increase the turnover rate of coolant in the pressurizer (0.5)

Ref: SO1-4-13

4.10 (1.9)

(a) Which is the lowest Emergency Action Level which is characterized by major failures of plant functions needed for protection of the public? (0.6)

(b) What are the immediate operator actions for a severe earthquake which occurs while operating at power? (1.3)

END OF SECTION FOUR

(a) Site Area Emergency

(b) (1) If any equipment failure is indicated, trip the reactor. (0.65)
(2) Trip the Circulating Water Pumps if the reactor is tripped. (0.65)

Ref: SO1-2.5-1, SONGS Emergency Plan