U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-255/OL-84-01

Docket No. 50-255 License No. DPR-20

Licensee: Consumers Power Company
Rt. 2, Box 154
Covert, MI 49043

Facility Name: Palisades

Examination Administered At: Palisades Generating Station and Midland Training Center

Examination Conducted: December 11, 1984 and January 8-12, 1985

R. L. Higgins
Examiner(s): R. L. Higgins 1/21/85
Date

R. L. Higgins for
A. Prichard 1/21/85
Date

R. L. Higgins for
R. Clark 1/21/85
Date

TM Burdick for
Approved By: J. I. McMillen 1/22/85
Date

Examination Summary

The written examination was administered on December 11, 1984 (Report No. 50-255/OL-84-01)
The similator examinations were administered on January 8-10, 1985.
The plant walk-through examinations were administered on January 11 and 12, 1985.
Written examinations were administered to 2 Reactor Operators, 7 Senior Reactor Operators, and 5 Instructors. 7 Senior Reactor Operators, 1 Reactor Operator, and 2 Instructors passed the written examination.
The utility and the NRC agreed that simulator and plant walk through examinations would only be administered to those who passed the written examination.
All individuals who passed the written examination also passed the simulator and plant walk-through examinations.

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REPORT DETAILS

1. Examiners

R. L. Higgins, Region III, Chief Examiner
A. Prichard, PNL
R. Clark, PNL

2. Examination Review Meeting

On December 11, 1984, at the conclusion of the written examination, the Chief Examiner met with the following facility personnel to review the SRO and RO examinations.

Ron Frigo - Shift Engineer
Dennis Willemin - Nuclear Instructor
Gene Dziedzic - Training Manager
Bob Heimsath - Nuclear Instructor
Don Lengschwager - Administrative Shift Supervisor
Walt Hunt - Nuclear Instructor
Gary Groff - Reactor Operator

No questions on either examination were requested to be deleted. Discussions concerning the answer keys for each exam resulted in minor changes to the answers for several questions.

The Chief Examiner made several observations concerning the training material provided by the facility:

1. The plant data book is extremely hard to read, increasing the probability of an operator making an error while reading it.

2. The plant lesson plans are difficult to read and in many instances incomplete. There is very little detailed explanation of systems or components.

3. The plant lesson notes are out of date. Extensive space is devoted to systems and components which are no longer used (reactor regulating system, cooling tower fan reverse mode of operation, Feed Only Good Generator system).

3. Exit Meeting

On January 10, 1985, at the conclusion of the simulator examinations, the chief examiner and one of the other examiners met with the following utility personnel to discuss generic observations made during the simulator examinations:

Jim Lang, Director of Nuclear Operations Training
Gene Dziedzic, Palisades Training Manager
Tracy Onnen, Midland Training Center Supervisor
Rick Buckner, Former Palisades Simulator Supervisor
Bob Heimsath, Current Palisades Simulator Supervisor
Leroy Schmiedeknect, Palisades Simulator Instructor

The following topics were discussed:

1. Completed Estimated Critical Position Forms corresponding to each startup initial condition need to be kept available in order to expedite start-up scenarios.

2. In order to shorten the length of time needed to conduct simulator examinations, novice simulator operators should not be used to operate the simulator during NRC examinations.

3. When personnel outside of the control room need to be notified, the examinees should actually notify the simulator operator. Many examinees simply informed their examiner that they intended to notify the personnel outside the control room rather than actually notifying the simulator operator.

At the conclusion of the plant walk-through examinations the chief examiner informed the Training Manager, Gene Dziedzic, of the results of the simulator and walk-through portions of the examination as well as any significant observations. Mr. Dziedzic was informed that all examinees who were administered simulator and plant walk-through examinations passed. The following observations were made by the examiners during the course of the examinations.

1. One examinee was unfamiliar with visitor escort procedures.

2. One examinee was unfamiliar with the location of the relays which are required to be bypassed by Off Normal Procedure 11.

3. One examiner was unduly delayed in obtaining escorted access across a step-off pad.

4. Many caution tags were discovered which had been in place for extensive periods of time. Several had been in place since 1976.

5. A very large number of component problem identification tags were noted in the Control Room. Though it is indicative of thoroughness that these deficiencies are identified, that such a large number have yet to be repaired is evidence that much effort is still needed.

6. One of the examiners stated that the examinees he observed during the simulator and plant walk-through examinations were the most impressive examinees he has ever seen.

Master

U.S. NUCLEAR REGULATORY COMMISSION

REACTOR OPERATOR LICENSE EXAMINATION

Facility: PALISADES (255)

Reactor Type: PWR-CE

Date Administered: 12-11-84

Examiner: Joe Upton, et al., PNL

Candidate: Answer Key

INSTRUCTIONS TO CANDIDATE:

Print your name on the line above marked "Candidate." Use separate paper for your answers and write on only one (1) side of the paper, unless a specific question instructs you otherwise. Staple this question package to your answer sheets. The grade points available for each question are indicated in parenthesis after each question. The passing grade is at-least-70% in each of the four (4) categories and is at-least-80% for the total grade. The examination questions and answers will be picked up six (6) hours after the examination was started. Read the statement at the bottom of this page; affirm the statement by signing your name.

Category Value	% of Total	Candidate's Score	% of Cat. Value	Category
25	25%		%	1. Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer and Fluid Flow
25	25%		%	2. Plant Design Including Safety and Emergency Systems
25	25%		%	3. Instruments and Controls
25	25%		%	4. Procedures: Normal, Abnormal, Emergency, and Radiological Control
100	100%			TOTALS

Total Grade ______ %

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

____

Candidate's Signature

1.0 PRINCIPALS OF NUCLEAR REACTOR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW

The following statements apply to Questions 1.01, 1.02, 1.03 and 1.04.

The Palisades power plant has been operating continuously at a steady 80% of full power for 10 days. All control rods are fully withdrawn from the nuclear-reactor core. The fuel-burnup status is that the core has reached 5000 MWD/MTU on cycle 6. Use any of the provided figures and tables. Show your work and your procedure for arriving at your answers.

Points Available

QUESTION 1.01

The power plant is operating as described above and all of its parameters are equal to their respective programmed values.

a. What are the values for T_c, T_{ave} and T_h at which the power plant is operating? (0.5)

b. In the process of determining a power calibration for the nuclear reactor while operating at these programmed values, it was recorded that the flowrate of coolant through the reactor core is 1.61×10^8 lbm/hr. It has also been previously recorded that for these conditions of the coolant the coefficient of specific heat of the coolant is 1.3 Btu/lbm-°F. Given this information, what is the rate of thermal-energy (heat) addition to the coolant? (0.5)

c. If, on your next shift, you observe that:

1. T_c is now higher than the previous (programmed) T_c by 2°F,
2. the reactor-coolant flowrate has not changed,
3. the turbine-generator output power has not changed,
4. the power plant is stable;

what change in °F has occurred to T_h? In addition, specify whether the steam flowrate, the steam temperature and the steam pressure have increased or decreased. (2.0)

- Section 1 continued on next page -

Points
Available

ANSWER 1.01

a. Using the curves on Figure 3.3,

T_c = 535°F

T_{ave} = 554°F (+0.5)

T_h = 573°F.

b. delta-T = 573 - 535 = 38°F (+0.2 - Use the student's values for T_h and T_c)

$\dot{Q} = \dot{m}_c c_p$ delta-T (+0.2)

$= (1.61 \times 10^8)(1.3)(38)$

$= 7.95 \times 10^9$ Btu/hr (+0.1 - The acceptable answer will vary depending upon the student's value for ΔT)

c. If the output power has not changed and the power plant is stable, then the heat generated in the primary coolant has not changed. If the primary coolant flowrate has not changed, then delta-T cannot have changed. Hence, T_h = 575°F. (+1.0)

If the output power is the same and if T_{ave} is higher, then the steam temperature is higher, the steam pressure is higher and the steam flowrate is lower. (+1.0)

Reference(s)

1. Millstone II: Technical Data Book, OP form 2204-1.

2. Generic: "Academic Program for Nuclear Power Plant Personnel," Volume III, pp. 2-138 through 2-139, Generic Physics Corporation.

3. St. Lucie 1&2: "Power Plant Thermodynamics," pp. 68, 74, 83.

- Section 1 continued on next page -

Points Available

ANSWER 1.01 (contd)

Reference(s) (contd)

4. Palisades: Technical Data Book, Figure 3.3, Rev. 0.

5. Palisades: Nuclear Operator Training Program, Volume VII-A, Phase I, Section 34, "Fluid Flow."

QUESTION 1.02

The power plant is operating as described above (80% power, ARO, etc.) and all of its parameters are equal to their respective programmed values.

a. What is the Pressurizer level in %? (0.5)

b. What is the temperature in the Pressurizer? (0.5)

c. What is the degree of subcooling in °F in the hot leg? (0.5)

d. Why is the Pressurizer level controlled to vary as the power level varies? (1.5)

ANSWER 1.02

a. Using T_{ave} = 554°F **(use the student's T_{ave} from 1.01)** on Figure 5-7, the Pressurizer level is at 54.5% **(+0.5 for 54 to 55%).**

b. Assuming that the pressure in the Pressurizer is at its programmed value of 2010 psia, then from the steam tables, the temperature (saturation temperatures for 2010 psia) in the Pressurizer is 637°F **(+0.5 for 633 to 640°F).**

- Section 1 continued on next page -

Points
Available

ANSWER 1.02 (contd)

c. The degree of subcooling in the hot leg is

$$T_{pzr}-T_h = 637 - 573 \quad \textbf{(+0.5)}$$

$$= 64°F. \quad \textbf{(+0.5)}$$

d. As the power level increases, T_{ave} and delta-T both increase for the primary coolant. Hence, when there is a power decrease from a high power level, there will be a larger temperature drop than if the power decrease occurred from a lower power level. This temperature drop results in shrinkage of the volume of reactor (primary) coolant; i.e., a drop in Pressurizer level. Hence, there is the potential for a greater Pressurizer level drop for higher power levels **(+0.5)**. For proper control of the Pressurizer, it is desired to keep the Pressurizer heaters covered **(+0.5)**. Pushing the design in the direction of smaller-sized Pressurizers are the goals to minimize the size of the Pressurizer both for cost and for limiting the energy release in a LOCA **(+0.5)**. A design of a certain-sized Pressurizer with a programmed-level curve as shown in the figure is the result.

Reference(s)

1. Millstone II: Technical Data Book, OP FORM 2204-3, Rev. 2.

2. Generic: C-E Simulator Training Manual, RCS, pp. 37.

3. Generic: "Academic Program for Nuclear Power Plant Personnel," Volume III, pp. 2-45 through 2-56, General Physics Corporation.

4. Palisades: System Lesson Notes, "No. 5 - Pressurizer Pressure and Level Control," Rev. 1, pp. 3-6, Figure 5-7.

5. Palisades: Nuclear Operator Training Program, Volume VII-A, Phase I, Section 31, "Thermodynamics."

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Points Available

QUESTION 1.03

The nuclear-reactor power plant is operating as described above and is operating with all of its parameters at their respective programmed values. The results of the physics tests show that the differential boron worth is 0.011 %-reactivity per ppm.

a. What is the value for the boron concentration? Give the HFP value. (0.5)

b. If the control rods are to be inserted while in the manual-sequential (MS) mode until 0.3 %-reactivity worth of the rods have been inserted and if the boron concentration is to be adjusted to maintain the present power level, what is the new concentration of boron required for steady operation? Neglect any effect from changes in the Xenon or Samarium concentrations. (1.5)

c. Describe the impact this plant change in "b." would have on the distribution of the neutron flux. (1.0)

d. What is the new control rod position in inches withdrawn? (1.5)

ANSWER 1.03

a. The answer from the curve on Figure 6.1 is 500 ppm **(+0.5)** for the boron concentration.

b. %-reactivity = 0.3%

diff-boron-worth = 0.011 %-reactivity/ppm,

hence,

$$\text{delta-boron-concentration} = \frac{0.3}{0.011} \quad (+0.5)$$

$$= 27.3 \text{ ppm} \quad (+0.5)$$

- Section 1 continued on next page -

Points
Available

ANSWER 1.03 (contd)

and then the

new-boron-concentration = 500 - 27

= 473 ppm

(+0.5 for 468 to 478 ppm, -0.5 for 522 to 532 ppm, +1.5 max, +0.0 min)

c. The insertion of rods part way into the reactor core will result in a redistribution of the neutron flux, reducing the neutron flux in the vicinity of the inserted rods and increasing the neutron flux elsewhere **(+1.0)**. AO would increase.

d. 75 inches withdrawn on Bank D (1.5)

Reference(s)

1. Millstone II: Technical Data Book, "Inverse Boron Worth vs. Core Average Burnup," OP FORM 2208-5, Rev. 6.

2. Ibid., OP FORM 21018-2, Rev. 5.

3. Ibid., OP FORM 2208-7, Rev. 6.

4. Millstone II: Hot License School, "Reactivity Theory and Operating Characteristics," Lesson #7/8 - Enabling Objectives.

5. Palisades: Technical Data Book, "Predicted Boron Concentration vs. Burnup (HFP, ARO, EQUAL, Xenon)," Figure 6.1, Rev. 1.

6. Palisades: Nuclear Operator Training Program, Volume VII-A, Phase I, Section 17, "Reactor Control," pp. 7.1-1 through 7.3-7.

7. Palisades: Technical Data Book, Figure 1.5.

- Section 1 continued on next page -

Points Available

QUESTION 1.04

The nuclear-reactor power plant is operating as described above (80%, ARO, etc.) and all parameters are at their respective programmed values. The boron concentration is 500 ppm. The power plant is to be taken to 100% of full power.

a. What would be the magnitude of the change in reactivity due to the change in the temperatures of the system? Specify whether this change (due to changing to 100% power level) would add or take away reactivity to the core. (1.0)

b. Explain the doppler reactivity effect; that is, how is this reactivity effect produced in the nuclear-reactor core? (2.0)

c. If the nuclear-reactor power level is taken to 100% of full power in 20 minutes, make a graph of the Xenon worth as a function of time. Show time from the point of increasing power and for the next 50 hours. Indicate on the graph numeric values for Xenon worth. (1.5)

ANSWER 1.04

a. From Figure 3.1,

power defect at 80% of full power = 1.02
power defect at 100% of full power = 1.25.

Hence the reactivity effect due to doppler in going from 80% to 100% is 0.23% **+(0.5)**. This is a negative-feedback effect, so increasing power results in introducing a negative reactivity, i.e., doppler will "take away" reactivity **(+0.5)**.

- Section 1 continued on next page -

Points
Available

ANSWER 1.04 (contd)

b. The doppler effect is the name given to the reactivity effect produced by changing the temperature of the fuel **(+0.5)**. An increase in the fuel temperature means that the atoms of the fuel rod are vibrating faster. The microscopic cross sections for neutron absorption in U-238 (and similarly in other elements and isotopes) as a function of neutron energy has several large peaks. With the increased fuel temperature, with increased vibration of the U-238 atoms, the peaks essentially broaden **(+1.0)**. This results in an increase in the absorption of neutrons by U-238 atoms as the neutrons are slowing down (decreasing speed, energy) **(+0.5)**. As absorption in U-238 generally does not result in a fissioning event, the increased absorption will reduce k_{eff}. U-238 is referred to because, in power reactors with small enrichments, it is U-238 that contributes most of the doppler effect. Pu-240 is the next most important isotope with respect to the doppler effect.

c. The sketch is shown below.

(+0.5 for the shape of the curve showing a 0.25% valley)
(+0.5 for 2.5 equil Xe-worth at 80%)
(+0.5 for 2.63 equil Xe-worth at 100%)

- Section 1 continued on next page -

Points
Available

ANSWER 1.04 (contd)

References

1. Generic: "Academic Program for Nuclear Power Plant Personnel," Volume II, pp. 4-131 through 4-143, General Physics Corporation.

2. Millstone II: Technical Data Book, OP FORM 2208-2.

3. Millstone II: Hot License School, "Reactor Theory and Operating Characteristics," Lesson #5, #7/8, #9 - Enabling Objectives.

4. Palisades: Technical Data Book, "Power Defect vs. Power," Figure 3.1, Rev. 1.

5. Palisades: Technical Data Book, "Equilibrium Xenon vs. Reactor Power," Figure 2.1, Rev. 1.

6. Palisades: Technical Data Book, "Xenon Worth vs. Time Step Increases to 100% From Various Power Levels," Figure 2.4, Rev. 1.

7. Palisades: Nuclear Operator Training Program, Volume VII-A, Section 18, "Reactor Operation," pp. 8.2-1 through 8.2-5.

QUESTION 1.05

If a small leak develops through a pressurizer safety relief valve and if the quench-tank pressure is 20 psia, what temperature would you expect to measure downstream of the valve for the following two (2) conditions?

a. The power plant is operating at 100% of full power with the pressurizer pressure at 2010 psig. (0.5)

b. The power plant is in a cooldown mode and the RCS (PCS) temperature is 480°F and the pressure is 900 psia. (0.5)

Use the attached steam tables, if necessary.

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Points
Available

ANSWER 1.05

a. 228°F (~~+1.0~~ 0.5)

b. 310°F (~~+1.0~~ 0.5)

Reference(s)

1. Generic: "Academic Program for Nuclear Power Plant Personnel," Volume III, pp. 2-45 through 2-50, General Physics Corporation.

2. St. Lucie 1&2: "Power Plant Thermodynamics," pp. 36, 37, 49, 50, 52.

3. Palisades: Nuclear Operator Training Program, Volume VII-A, Phase I, Section 31, "Thermodynamics."

QUESTION 1.06

Which one of the statements below most correctly describes the transformation of the energy from fission events into heat energy? (1.0)

(a.) The energy released as kinetic energy of fission fragments provides less than 50% of the energy released per fission event.

(b.) Fissioning of the isotope U-238 provides more than 50% of the thermal energy generated in the core.

(c.) About 200 Mev of energy is released per fission event (neglecting neutrinos) of which about 15 Mev is released after a delay time.

(d.) All of the fission-event kinetic energy is absorbed in the coolant.

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Points Available

ANSWER 1.06

The answer is (c.) **(+1.0)**

Reference(s)

1. Millstone II: Hot License School, "Reactor Theory and Operating Characteristics," Lesson #2 - Enabling Objectives.

2. St. Lucie 1&2: CE PWR Simulator Training Facility, "Reactor Theory," Florida Power and Light Company, pp. 500 (82H8)/ds-42, 43.

3. Palisades: Nuclear Operator Training Program, Volume VII-B, Phase I, Section 8, "Basic Nuclear Concepts, "pp. 12.4-1 through A-3.

QUESTION 1.07

answer the following True/False questions
~~Which three (3) of the statements below are correct~~ for a nuclear reactor of the Palisades type.

(a.) A specific neutron reaction rate is dependent on the magnitude of the cross section of the material in question and on the magnitude of the neutron flux (0.5)

(b.) If the thermal-neutron flux is doubled, the thermal power produced in the nuclear reactor is doubled. (0.5)

(c.) The neutron microscopic cross section (σ) for a certain element varies with neutron energy and is dependent on the isotope of the element. (0.5)

(d.) The thermal-neutron microscopic fission cross section for U-238 is larger than that for U-235. (0.5)

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Points Available

ANSWER 1.07 (a) True (b) True (c) True (d) False

~~The answer is (a.), (b.) and (c.) (+0.33 each)~~ (0.5 each)

Reference(s)

1. Millstone II: Hot License School, "Reactor Theory and Operating Characteristics," Lesson #3 and #4 - Enabling Objectives.

2. St. Lucie 1&2: CE PWR Simulator Training Facility, "Reactor Theory," Florida Power and Light Company, p. 500 (82H8)/ds-30, 65, 66, 73, 74, Figures 18, 19.

3. Palisades: Palisades Nuclear Operator Training Program, Volume VII-B, Phase I, Sections 9-11, "Basic Nuclear Concepts," pp. 13.1-1 through 15.5-5.

QUESTION 1.08

Select the letters of the correct responses from those listed below. The responses are in answer to the statement, "Water hammer can be caused by ... ." (1.5)

(a.) ... operating a pump with too high of a net positive section head.

(b.) ... operating a pump near the peak of its efficiency curve.

(c.) ... suddenly closing a valve in a pope containing flowing liquid or gas.

(d.) ... starting a feed pump when the feed ring is filled with steam.

- Section 1 continued on next page -

Points
Available

ANSWER 1.08

The answer is both (c.) and (d.).

[+1.5 for both, +0.5 for (c.) only, +0.5 for (d.) only]

Reference(s)

1. Millstone II: NRC Reference Data, Book 16, Exam Bank.

2. Palisades: Nuclear Operator Training Program, Volume VII-A, Phase I, Section 34, "Fluid Flow."

3. Generic: Giles, R. V., "Theory and Problems of Fluid Mechanics and Hydraulics," Schaum Publishing Col, New York, 1962, pp. 195-196.

QUESTION 1.09

Assume that with a normal flowrate of the primary coolant enough heat will be generated to reach the DNB point when the local power density reaches 27 kW/ft.

a. If the actual maximum local power density is 9 kW/ft, calculate the DNBR. (0.5)

b. If the nuclear-reactor power level is increased from that in part "a.", will the DNBR increase, remain the same, or decrease? (0.5)

c. If the flowrate through the core drops 5% below the normal flowrate and if the maximum local power density is the same as in part "a.", will DNBR increase, remain the same, or decrease? (0.5)

- Section 1 continued on next page -

Points
Available

ANSWER 1.09

a. DNBR = critical local heat density/actual maximum local heat density

= 27/9 = 3 (+0.5)

b. The DNBR will decrease (+0.5) because the maximum local heat density will increase.

c. If the flowrate is reduced, the critical local heat flux will be less than 27 kW/ft. Hence, DNBR will be less than 2.7. (+0.5)

Reference(s)

1. Generic: "Nuclear Energy Training," Module 4, Plant Performance, pp. 8.2-1 through 8.2-4, NUS Training Corporation.

QUESTION 1.10

The Palisades power plant is operating steadily at 100% of full power with the xenon concentration equal to its 100%-power equilibrium value and with all rods out of the core. The burnup is 5000 MWD/MTU and the boron concentration is 500 ppm. Estimate the shutdown margin. Show your work and state all of your assumptions. (2.0)

- Section 1 continued on next page -

Points Available

ANSWER 1.10

Reactivity for ARO	-6.55%	**(+0.5 for 6 to 7% magnitude)**
Reactivity for highest-worth rod	+1.88%	**(+0.5 for 1.5 to 2% magnitude)**
Reactivity for power defect	+1.25%	**(+0.5 for 1.1 to 1.4% magnitude)**
Reactivity for boron	small	~~**(+0.1 for magnitude)**~~
Shutdown margin	-3.42%	**(+0.5 for correct magnitude, correct use of signs)**

~~**(+2.0 max)**~~

Reference(s)

1. Millstone II: Safety Technical Specifications, "Reactivity Control Systems - Boration Control," Section 3/4.1.1, pp. 3/4 1-1 to 3/4 1-2.

2. Palisades: Technical Data Book, Figures 1.1, 3.1 and 3.2.

QUESTION 1.11

The nuclear reactor is shut down. The reactivity of the core -7% Δk/k and the source-range detector is reading 20 cpm. If 2% Δk/k of reactivity is added to the core, what would be the countrate on this source-range detector? (1.0)

- Section 1 continued on next page -

Points
Available

ANSWER 1.11

$$\frac{CR_2}{CR_1} = \frac{1-k_1}{1-k_2}$$

$$= \frac{\text{reactivity}_1}{\text{reactivity}_2} \qquad \textbf{(+0.5)}$$

$$= \frac{7}{5}\,(20)$$

$$= 28 \text{ cpm} \qquad \textbf{(+0.5)}$$

Reference(s)

1. Palisades: Nuclear Operator Training Program, Volume VII-A, Phase I, Section 22, "Fuel Loading and Startup," pp. 12.1-1 through 12.1-5.

2. Generic: "Academic Program for Nuclear Power Plant Personnel," Volume II, pp. 5-6 through 5-13, General Physics Corporation.

3. Millstone II: Hot License School, "Reactivity Theory and Operating Characteristics," Lesson #6 - Enabling Objectives.

- End of Section 1 -

2.0 PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS

Points Available

QUESTION 2.01

a. Why are there (2) seal-oil pumps associated with the Gland Seal System of the main generator? (1.0)

b. What backup is provided to keep seal oil flowing if one of the seal-oil pumps fails? (1.0)

c. What is the purpose of the defoaming tanks in the seal-oil system? (1.0)

ANSWER 2.01

a. One pump recirculates oil that is exposed to air and moisture **(+0.5)**, while the other pump recirculates oil that is exposed to hydrogen inside the generator **(+0.5)**.
a third pump is the air side seal oil backup pump (0.5)

b. The seal oil backup from the main-bearing oil-feed system is normally closed. If the motor-driven air-side seal-oil pump should stop, or if the seal-oil pressure at the seals should decrease to 8 psi above the hydrogen pressure, the backup regulator valve will open automatically and provide oil pressure for the seals **(+1.0)**.

c. Dissolved hydrogen must be removed from the oil to avoid producing an oil vapor that would circulate through the generator, reducing its efficiency **(+1.0)**. Incorrect!

Reference(s)

1. Palisades: PNTC Handout, Module 21d, Description of the (Main Generator) Seal Oil System, 7/16/84.
Lesson Notes No. 31

- Section 2 continued on next page -

Points
Available

QUESTION 2.02

While operating at a steady 100% of full power, the reactor operator causes an inadvertent boration. Tc drops 10°F before he/she recognizes the situation. Assuming the electrical output stays constant, explain how and why the Steam Generator and Feedwater Regulating System will respond to the inadvertent emergency boration. Assume no operator intervention. (Include the responses of the Steam Generator temperature, pressure and level and the main feedwater flowrate). (2.0)

ANSWER 2.02

- The steam temperature and pressure will decrease. **(+0.5)**
- Due to the drop in the Steam-Generator pressure, the Steam-Generator water level will rise. **(+0.5)**
- Due to the increase in the water level, the feedwater flowrate will decrease. **(+0.5)**
- Due to the drop in the Steam-Generator pressure, the steam flowrate will have to have increased. **(+0.5)**

Reference(s)

1. St. Lucie 1&2: NRC Question Bank, Section 4, Number 17.
2. Palisades: PNTC Handout, Module 20b, "Steam Generator Water Level Control," 8/1/84.
3. Millstone II: Book 10, System 4, "Feedwater System."
4. Generic: Nuclear Training, Volume 4, Plant Performance, Chapters 3.4 and 3.5, NVS Corporation.

- Section 2 continued on next page -

Points
Available

QUESTION 2.03

Following a SGTR event, all PCPs have been secured for about 2 hours.

a. What three (3) criteria must be satisfied before the PCPs may be restarted? (1.0)

b. According to SOP-1, what interfacing system must be in operation prior to restarting the PCPs? (1.0)

ANSWER 2.03

(a S/G is)

a. 1. Verify that the S/Gs are removing heat from the PCS. **(+0.33)**

2. Primary coolant inventory is established, e.g., PZR level at midscale, PCS pressure is 100 psi > S/G pressure. **(+0.33)**

3. PSC is $\geq$ 50°F subcooled. **(+0.33)**

b. Component Cooling Water System **(+1.0)**

Reference(s)

1. Palisades: PNP System Lesson Notes, No. 4, "PCP," 9/16/80.

2. Palisades: System Operating Procedures, PNP SOP-1, Para. 7.26, Rev. 13.

QUESTION 2.04

a. Give two indications of hydrogen buildup in containment. (1.0)

b. List three (3) subsystems of the post-accident hydrogen control system. (1.0)

- Section 2 continued on next page -

Points
Available

ANSWER 2.04

a. • Gradual drift upward in the containment pressure (which is not accompanied by an increase in the containment temperature or humidity) (+0.5).

• Analysis of gas in containment by chemistry (+0.5).

• Gradual reduction in the containment pressure after the operation of the hydrogen recombiners (+0.5).

(+1.0 max)

b. • Hydrogen recombiners (.5) - Recombination unit, power supply panel, and control panel

• Post-accident hydrogen sampling system (.25)

• Containment ventilation (hydrogen purge) system (.25)

~~(+0.33 each)~~

Reference(s)

1. Palisades: PNTC Handout, Module 27e, Hydrogen Recombiners, no date.

2. Palisades: PID M-224 Sh.2, Containment Hydrogen Monitoring System, 4/12/84.

3. Palisades: PID M-218 Sh.2, HVAC Containment Bldg., 7/18/84.

4. Millstone II: OP 2313C.

5. Millstone II: OP 2540E.

- Section 2 continued on next page -

Points Available

QUESTION 2.05

Consider the Pressurizer Pressure and Level Control System. For each of the two conditions indicated below, fill in the table as appropriate with open, closed, on, off or min. Assume that the nuclear reactor is at 100% of full power, CVCS controllers are in automatic and charging pumps #2 and #3 are the first and second backup pumps, respectively.

Condition #1 - pressure 2000 psig and decreasing, level -4.1% below setpoint and increasing.

Condition #2 - pressure 2130 psig and increasing, level 4.6% above setpoint and increasing.

Condition	Spray Valves	Prop. Heaters	Backup Heaters	Letdown Valves	CP #1	CP #2	CP #3	
#1	______	______	______	______	____	____	____	(1.5)
#2	______	______	______	______	____	____	____	(1.5)

ANSWER 2.05

#1	Closed (+0.3)	On (+0.3)	Off in manual (+0.3)	Min (+0.3)	On (+0.1)	On (+0.1)	On (+0.1)
#2	Open (+0.3)	Min (+0.3)	Off in manual (+0.3)	Open (+0.3)	On (+0.1)	Off (+0.1)	Off (+0.1)

Reference(s)

1. Palisades: PNTC Handout, Modules 25d&e, "Pressurizer Pressure and Level Control," 4/30/84 & 5/6/84.

- Section 2 continued on next page -

Points Available

QUESTION 2.06

a. List three (3) ways to trip the engine of the Emergency Diesel Generator. Give either manual-trip capabilities or conditions that would cause an automatic trip of the engine. (1.5)

b. List three (3) Emergency Diesel Generator breaker trips. (1.5)

ANSWER 2.06

a. Engine Trips

- Manual

  - Control room
  - Local control panel
  - Manually tripping overspeed trip device

- Auto trips

  - Generator differential relay
  - Engine overspeed
  - Overcrank
  - Low bearing oil pressure

(+0.5 each, +1.5 max)

b. Diesel Generator Breaker Trips

  - Manual trip at switchgear
  - Manual trip at C04
  - Engine trip
  - Loss of excitation
  - Overcurrent > 186-107 relay
  - Bus transfer (trip signal remains for 1.5 sec)

(+0.5 each, +1.5 max)

- Section 2 continued on next page -

Points
Available

ANSWER 2.06 (contd)

Reference(s)

1. Palisades: PNTC Handout, Module 22b, (Emergency) Diesel Generators, 7/25/84.

QUESTION 2.07

a. What is the shutoff head and design flowrate for a HPSI pump? (1.0)

b. What three (3) SIRWT parameters have Tech-Spec limits? (1.5)

c. Using Figure 2.07 draw a one-line diagram showing the water flow paths from the water supplies to the RCS through the HPSI pumps. Show only the valves that operate on a SIAS. (2.0)

ANSWER 2.07

a. 2750 ft or 1200 psig **(+0.5 for 2500 to 3000 ft or for 1100 to 1300 psig)** and 315 gpm **(+0.5 for 280 to 350 gpm)**

b. Volume, boron concentration, temperature **(+0.5 each)**

c. See attached sketch. Correct flow path **(+1.0)** Correct components (pumps, valves, tank, sump) **(+1.0)**

Reference(s)

1. Palisades: PNTC Handout, Module 27a, Safety Injection System, 5/21/84.

2. Millstone II: Book 8, System Prescription #5, HPSI Table A, Figure I.

3. Millstone II: Safety Technical Specifications, Section 3.1.2.8.

- Section 2 continued on next page -

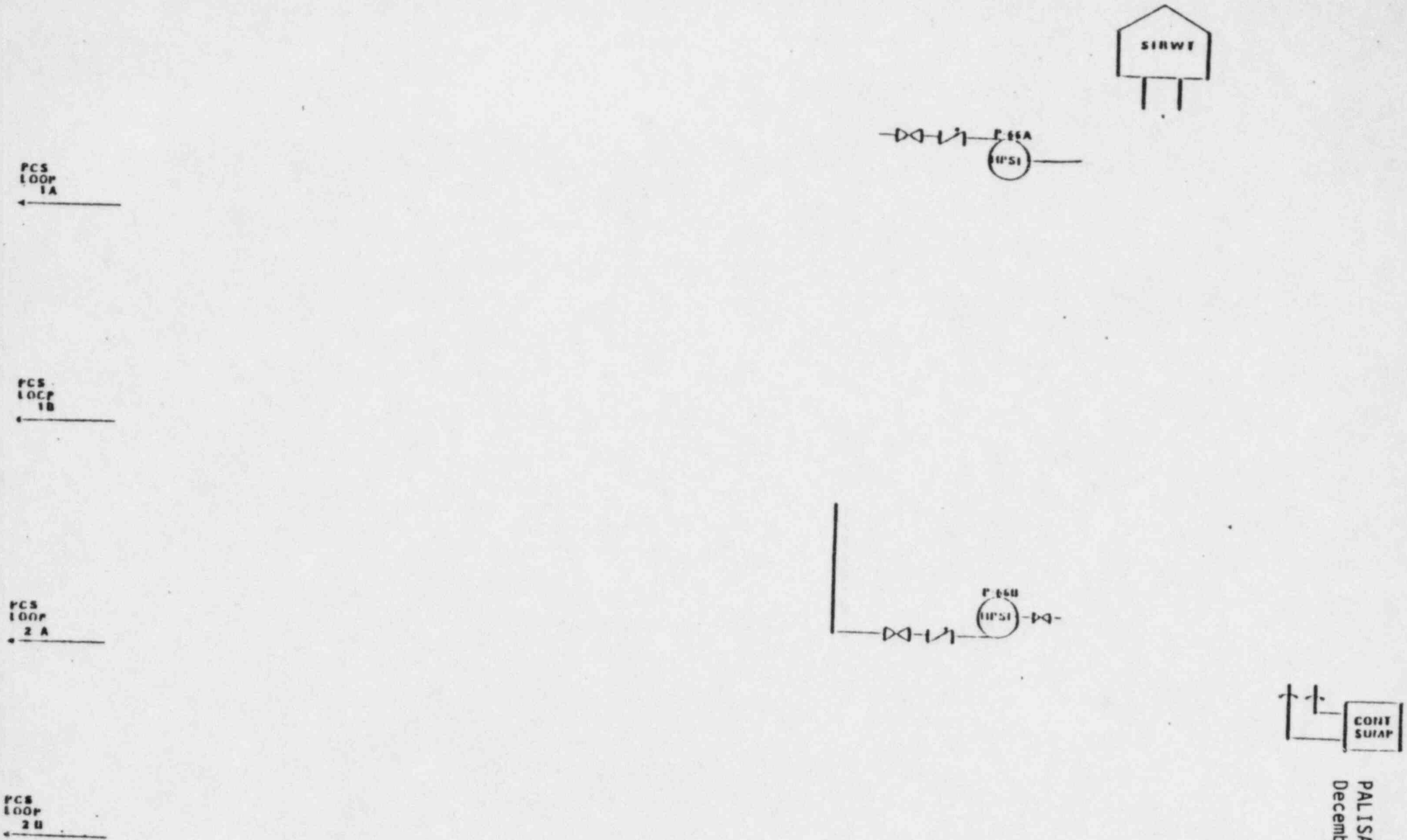

FIGURE 2.07. HPSI System

(Question)

PALISADES
December 11, 1984

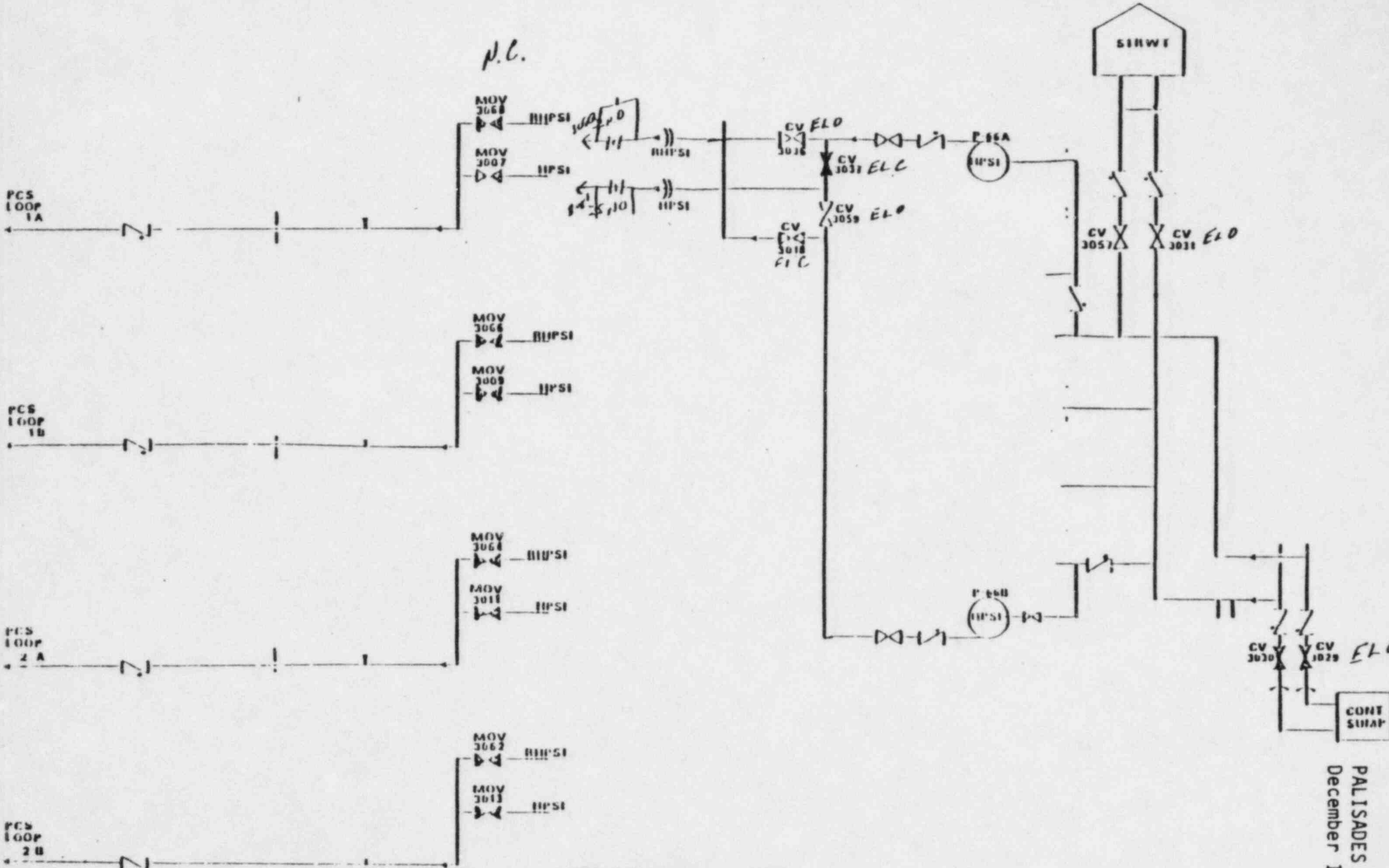

FIGURE 2.07. HPSI System

(Answer)

Points Available

QUESTION 2.08

a. List the six (6) actuation signals that are generated for the CCW System upon receipt of an SIAS. (2.0)

b. List six (6) of the systems serviced by the Component Cooling Water System. (2.0)

c. How does the CCWS respond to a loss of offsite power under the following conditions:

Rx shutdown
RCS temperature between 350°F and 130°F? (1.0)

ANSWER 2.08

a. 1. ESF room cooling coil inlet and outlet valves open
2. Containment inlet and outlet isolation valves shut
3. SFP HX inlet valve shut
4. BA evaporation inlet and outlet valves shut
5. CCW spare pumps start
6. SDC HX inlet valves open

(+0.33 each, +2.0 max)

b. 1. Shutdown-cooling heat exchanger
2. ESF pumps
3. SFP heat exchangers
4. Evaporators
5. RCP's seals and bearing cooling
6. Letdown heat exchanger
7. Reactor shield cooling
8. CRDM cooling
9. Charging pump cooling
10. Primary system sample cooler
11. Waste gas compressors aftercoolers
12. Vacuum degasifier pump seal water cooler

(+0.33 each, +2.0 max)

- Section 2 continued on next page -

Points
Available

ANSWER 2.08 (contd)

c. The CCW pump that is operating will trip and the ESF load sequencer will begin. Pumps P-52A,B and C will not start automatically ~~start after 23 seconds~~; ~~P-52C is placed in standby after 40 seconds, then starts if the pressure is less than 80 psig.~~ (+1.0)

Reference(s)

1. Palisades: PNTC Handout, Module 15b, Component Cooling Water System, Rev. 0.

QUESTION 2.09

During equilibrium, 100% power operations, the operator notices that the instrument air pressure is dropping. If the drop is rapid, what 2 components respond automatically? (0.5)

ANSWER 2.09

a. The two (2) backup air compressors start. (+0.5)
Service air supply isolation valve closes

Reference(s)

1. Palisades: Plant System Functional Description, SFD 6.2.1, 2, 3/9/84.

2. Palisades: Emergency Operating Procedure, EOP-5, Rev. 12.

- End of Section 2 -

3.0 INSTRUMENTS AND CONTROLS

Points Available

QUESTION 3.01

Using the attached Figure 3.01 of the CVCS, what are the nominal readings on the five (5) instruments indicated A through E. The power plant is operating at 100% of full power and one charging pump is running. (2.5)

ANSWER 3.01

A. Charging flow through Regen HX, <140 gpm (CAF)

B. Temperature out of Letdown HX, ≤120°F.

C. Pressure downstream of backpressure valves, ≤190 psi

D. Letdown flow, ≤~~100~~ 40 gpm

E. Pressure in VCT, <100 psi

(+0.5 each)

Reference(s)

1. Palisades: PNTC Handout, Module 25a, CVCS 4/17/84.
2. Millstone II: OP 2340A.

- Section 3 continued on next page -

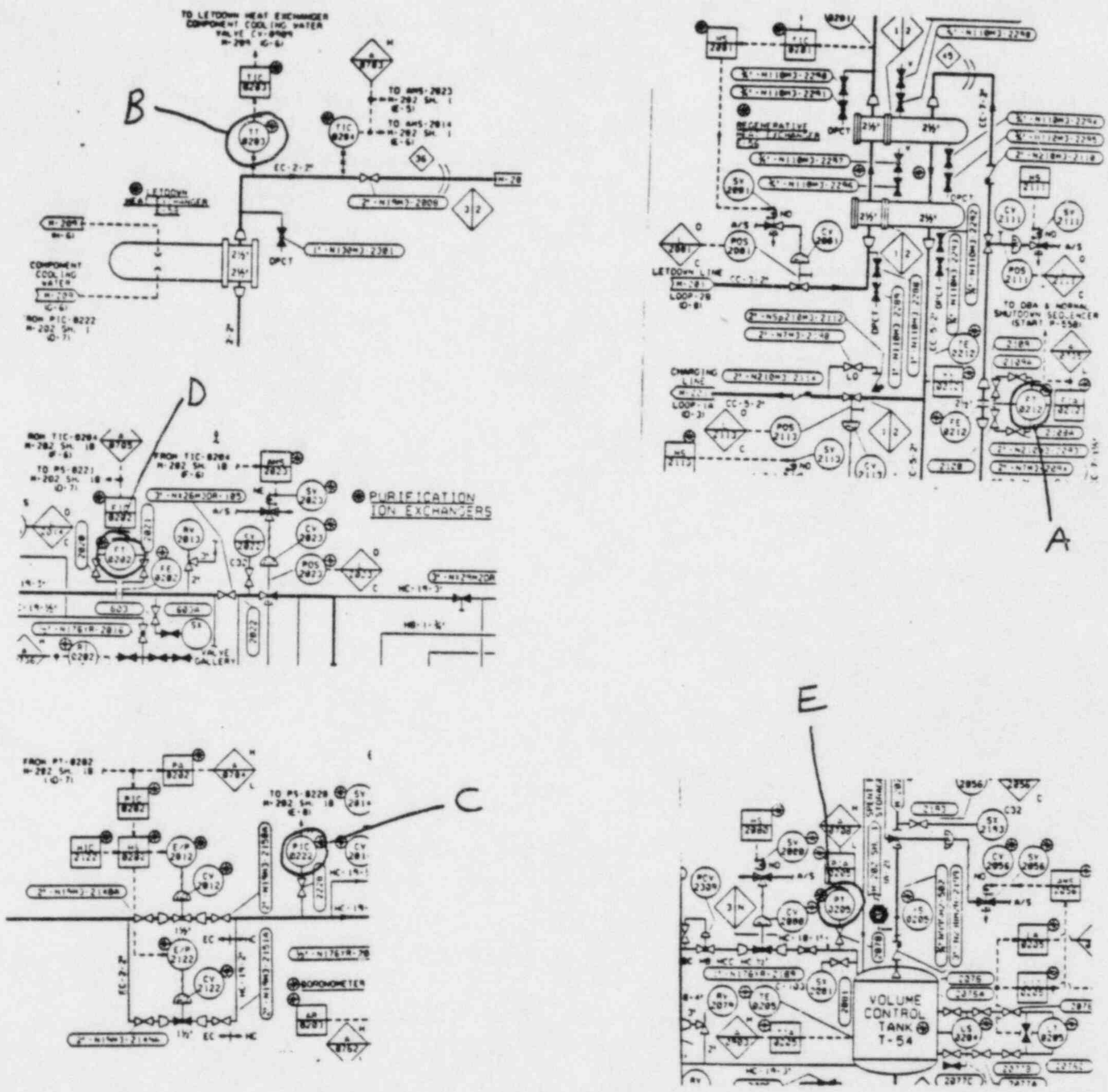

FIGURE 3.01. Portions of CVCS

- Section 3 continued on next page -

Points
Available

QUESTION 3.02

The power plant is operating at 85% of full power with all-rods-out. Explain how and why changes in the following parameters will affect the TM/LP pressure set point (increase pressure setpoint, decrease, or stay the same). Assume the operators take action to maintain a constant 85% electrical output and the operators maintain programmed operating limits. Consider each item separately.

a. PCS pressure decrease. (1.0)

b. Hot-leg PCS temperature, T_h, increase. (1.0)

c. Cold-leg PCS temperature, T_c, decrease. (1.0)

d. PCS flowrate increase. (1.0)

ANSWER 3.02

a. There is no effect (+0.5) because the PCS pressure is not an input to the set-point determination (+0.5).

b. The set-point pressure will increase (+0.5). An increase in T_h is an increase in ΔT across the core and hence in the core power. An increase in the core power means that the core conditions are closer to a DNBR of 1.3. (+0.5)

c. The set-point pressure will increase (+0.5). T_c is used as indicated in "b." as well as in a worse-channel calculation. (+0.5)

d. There is no effect (+0.5) because flowrate is not an input to the TM/LP calculation (+0.5). It is assumed that the flowrate is at least some 91% of full flow.

- Section 3 continued on next page -

Points
Available

ANSWER 3.02 (contd)

Reference(s)

1. Palisades: Safety Technical Specifications, 2.3, Amend. 82, 6/7/84.

2. Millstone II: Book 7, System Description #3, "RPS-TM/LP."

QUESTION 3.03

During equilibrium, 100% power operation a reactor trip occurs. The four (4) Steam Dump/Turbine Bypass valves have opened excessively.

a. Identify and explain two (2) indications in the control room that the 4 Steam Dump/Turbine Bypass valves have opened excessively. (2.0)

b. Explain the actions of the operators that are necessary to control the Steam Dump/Turbine Bypass Control System and to maintain the power plant in a stable and safe condition. Specify the parameters and values to which they should be controlled to correct the situation. (1.0)

ANSWER 3.03

a. Excessive cooldown would reduce T_{ave} and could reduce T_{ave} below the no-load value. The Steam-Generator pressure would be low due to the excessive cooldown. And the lowering of T_{ave} would produce a low pressure in the Pressurizer. **(+1.0 each, +2.0 max)**

b. Take manual control and close the atmospheric dumps and close the steam-dump valves to the condenser. **(+0.5)** Manually control the turbine-bypass valves to maintain 920 psia in the S/G and 532°F T_{ave}. **(+0.5)**

- Section 3 continued on next page -

Points Available

ANSWER 3.03 (contd)

Reference(s)

1. Palisades: PNTC Handout, Module 18b, "Main Steam System," no date.

2. Songs 2&3: System Description #48, "SBCS."

3. Millstone II: Book 9, System Description #5, "RRS," pp. 11-23, System 6.

QUESTION 3.04

For the conditions listed below, indicate the ESFS channels (actuation signals) that should actuate. Consider the conditions in each part of this question separate from the conditions in the other parts.

a. S/G #1 pressure 800 psia | S/G #2 pressure 600 psia
S/G #1 level 68%, narrow range | S/G #2 level 45%, narrow range
Pressurizer pressure 1800 psia | Pressurizer level 20%, narrow range
Containment pressure 3 psig (1.0)

b. S/G #1 pressure 900 psia | S/G #2 pressure 900 psia
S/G #1 level 68%, narrow range | S/G #2 level 68%, narrow range
Pressurizer pressure 1000 psia | Pressurizer level 33%, narrow range
~~RWST~~ SIRW level 8% | Containment pressure 30 psig

Containment high-radiation sensor channels (all) are reading 500 mr/hr. (2.0)

c. S/G #1 pressure 900 psia | S/G #2 pressure 400 psia
S/G #1 level 60%, narrow range | S/G #2 level 5%, narrow range
Pressurizer pressure 1750 psia | Pressurizer level 15%, narrow range

Containment high-radiation sensor channels (all) are reading 30 R/hr. (1.0)

- Section 3 continued on next page -

Points
Available

ANSWER 3.04

a. None (+1.0)

b. safety injection, containment spray, containment isolation, containment air cooler DBA mode **(+0.5 each, +2.0 max)**

c. steam-line isolation, containment isolation **(+0.5 each)**

Reference(s)

1. Palisades: PNTC Safety Technical Specifications, Section 3.16, Table 3.16.1, Amend. 80, 4/10/84 (corrected 4/29/84).

QUESTION 3.05

During equilibrium full power operation, describe how the RPS will respond to the following Channel A input failures. Include meter response, alarms, channel pretrips and trips. Also, indicate the RPS channels that should be bypassed by the operator for each failure.

a. Channel A upper NI fails high. (2.5)

b. Channel A Steam Generator pressure fails high. (1.0)

c. Channel A Pressurizer pressure input fails to zero. (1.0)

ANSWER 3.05

a. Power-range subchannel deviation alarm **(+1.5)**
High-power trip - bypass **(+0.5)**
High-power reading for upper NI **(+0.5)**

b. No alarms **(+1.0)** Bypass low S/G press trip.

c. TM/LP trip and pretrip - bypass **(+0.5)**
Bypass high Pressurizer pressure **(+0.5)**

- Section 3 continued on next page -

Points
Available

ANSWER 3.05 (contd)

Reference(s)

1. Palisades: PNTC Handout, Module 28b, Reactor Protective System, 5/11/84.

2. Palisades: PNP System Lesson Notes No. 14, RPS, 9/29/80.

3. Millstone II: Book 9, System Description #3, "RPS."

QUESTION 3.06

The power plant is operating at 100% of full power. The Pressurizer level control is selected to Loop-1 T_{ave}. The Loop-hot-leg temperature signal fails and reads 150°F low.

a. Explain how and why the charging pumps and letdown valves should respond to the failed temperature element. (2.0)

b. Describe two (2) ways an operator could determine that the Loop-1 T_{ave} had failed. (1.0)

c. What actions should the operator take to deal with the failed temperature indicator? (1.0)

- Section 3 continued on next page -

Points Available

ANSWER 3.06

a. The controlling T_{ave} will drop (+0.25). That will cause the RRS to produce a minimum Pressurizer level setpoint (+0.75). That will cause minimum charging (+0.5) and maximum letdown (+0.5).

b. The operator can tell that the Loop-1 T_{ave} had failed by noting the T_{ave} deviation alarm, the pen recorder, the T_{ave}-T_{ref} signal, the gross-deviation alarm, and by observing that there had been no sudden drop in the pressure of the Steam Generators (+0.5 each, +1.0 max).

c. Change the Pressurizer level control to Loop-2 T_{ave} (+0.75) and verify that AWP is selected to Loop-2 (+0.25).

Reference(s)

1. Palisades: PNTC Handout, Module 26b, "Reactor Regulating and Rod Control," 4/25/84.

2. Palisades: PNP System Lesson Notes No. 13, Reactor Regulating System, 10/7/80.

QUESTION 3.07

During normal 100% power operation the reactor is tripped due to a S/G tube rupture and the Steam-Generator level drops below 25%.

a. How and why should the auxiliary feed water system be aligned after the accident has progressed for 30 minutes, if the use of auxiliary feed water is required? (2.0)

b. What are the main and alternate sources of auxiliary feed water? (1.0)

- Section 3 continued on next page -

Points
Available

ANSWER 3.07

a. The steam to the steam-driven feed pump should be isolated **(+0.5)** to prevent an unmonitored radiation release to the atmosphere **(+0.5)**. All auxiliary feed should be isolated from the affected S/G or most affected S/G **(+0.5)** to prevent overfilling of the S/G or prevent unborated water from entering the primary system **(+0.5)**.

b. Condensate Storage Tank and Fire Fighting System supply header, respectively. **(+1.0)**

Reference(s)

1. Palisades: PNP EOP-8.2, SGTR, 12/20/82.

2. Palisades: PNP System Lesson Notes No. 21, Feedwater and Condensate, 10/14/80.

3. Millstone II: EOP 2525, Step 3.16.

4. Millstone II: EOP 2534, Steps 3.10-3.12.

5. Millstone II: Book 10, System Description #5, "Auxiliary Feedwater."

- End of Section 3 -

4.0 PROCEDURES: NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL

Points Available

QUESTION 4.01

List the seven (7) IMMEDIATE ACTIONS required of an operator in response to a reactor trip which is initiated from one or more normal inputs to the Reactor Protective System and during which standby power is available. (3.5)

ANSWER 4.01

- Insure that the Full Length Control Rods are indicating fully inserted and that the reactor power is decreasing.
- Verify turbine trip and generator breakers opened; manually trip the turbine, then generator, if necessary.
- Verify both Emergency Diesel Generators have started.
- Trip one Main Feed Pump if both are running.
- Trip the other Main Feed Pump as T_{ave} nears 525°F.
- If safety injection has been initiated, trip all Primary Coolant Pumps after insuring that the Reactor has been tripped $\geq$ 5 seconds. Follow up with this procedure and Natural Circulation Procedure, ONP 21.
- Insure and/or establish Auxiliary Feedwater flow to restore normal level in the Steam Generators.

(+0.5 each)

Reference(s)

1. Palisades: Emergency Operating Procedure, "Reactor Trip," EOP 1, Rev. 14, Section 3.0, pp. 1.

- Section 4 continued on next page -

Points Available

QUESTION 4.02

A new step has been recently added to EOP 8, "Loss of Coolant Accident." This step states, "Place handswitches for ES Rooms Sump Pumps **P-72A, **P-72B, **P-73A and **P-73B to OFF." What is the reason for the addition of this step? (2.0)

ANSWER 4.02

The reason for this addition is to prevent operation of Engineering-Safeguards Sump Pumps during accidents where highly radioactive water may be present in the sumps. **(+2.0)**

Reference(s)

1. Palisades: Emergency Operating Procedure, "Loss of Coolant Accident," EOP 8.1, Rev. 16, Temporary Change to Procedure 0-84-213, 9-29-84, pp. 1.

QUESTION 4.03

Define the following terms.

a. Radiation Area (1.0)

b. Rad (1.0)

c. Dose (1.0)

- Section 4 continued on next page -

Points
Available

ANSWER 4.03

a. Radiation Area is any area, accessible to personnel, in which there exists radiation at such levels that a major portion of the body can receive in any hour a dose in excess of 5 ~~rem~~. mrem
(+1.0)

b. A Rad is that amount of ionizing radiation that corresponds to the absorption of 100 ergs per gram of body tissue. **(+1.0)**

c. Dose is that quantity of radiation absorbed, per unit of mass, by the body or any portion of the body. **(+1.0)**

Reference(s)

1. Generic: Code of Federal Regulations, 10 CFR 20.202, pp. 210.

2. Generic: Code of Federal Regulations, 10 CFR 20.4.

3. Millstone II: Station Health Physics Procedure, "Posting of Radiological Controlled Areas," SHP 4906, Rev. 2, Sections 3.4, 3.5, 3.9, pp. 2-3.

QUESTION 4.04

a. What is the reason that any operations which involve the complete (or partial) draining of the Component Cooling Water (CCW) System or involve cross-connecting the CCW System to another system should be conducted with caution? (1.0)

b. SOP 16 states, "Control valve **CV-0951 must not be opened whenever either control valve **CV-0950 or control valve **CV-0913 is open. These valves are operated from control consoles **EC-3 and **EC-33." Why is the statement in the SOP; i.e., what would happen if the statement was not adhered to? (1.0)

- Section 4 continued on next page -

Points
Available

ANSWER 4.04

a. The water of the CCW System may be radioactively contaminated (+1.0). The chromate additive is often the element that has become radioactive.

b. If this statement is not adhered to, then the contents of the CCW System will be pumped directly to the lake. (+1.0)

Reference(s)

1. Palisades: System Operating Procedure, "Component Cooling Water System," SOP 16, Rev. 3, Section 5.0, pp. 2.

QUESTION 4.05

List five (5) trouble conditions (System Malfunctions) associated with the Emergency Diesel Generator that will cause an alarm in the control room. (2.5)

ANSWER 4.05

1. Diesel Generator Breaker
   **152--107 Trip

2. Diesel Generator Breaker
   **152--213

3. Diesel Generator Start
   Air High-Low Pressure
   1-1
   1-2

4. Diesel Generator Fail to Start
   1-1
   1-2

low lube oil level
prelube oil pump ~~fail~~ failure
low lube oil pressure
hi jacket water temperature
Overcrank
low air pressure
low jacket water level
low lube oil temp
high lube oil temp
Overspeed/underspeed
low raw water press
hi lube oil diff press
engine trouble
fuel level hi/low

- Section 4 continued on next page -

Points
Available

ANSWER 4.05 (contd)

5. Diesel Generator Day Tank
   High-Low Level
   1-1 (**T-25A)
   1-2 (**T-25B)

6. Diesel Oil Storage Tank
   **T-10 Low Level

   (any 5, + 0.5 each)

   Reference(s)

   1. Palisades: System Operating Procedure, "Emergency Diesel Generator," SOP 22, Rev. 4, Attachment 1, pp. 1.

QUESTION 4.06

After it has been observed that there exists the potential for an emergency event,

a. who is responsible for activation of the Emergency Implementation Procedure (EI-1)? (0.5)

b. Upon the activation of the Emergency Implementation Procedure, who will assume the title and responsibilities of the Site Emergency Director (SED)? (0.5)

c. Who should relieve the initial Site Emergency Director, i.e., who is first on the list of "normal line of succession for the SED" (EI-1)? Provide an answer for "days" and for "nights." (1.0)

- Section 4 continued on next page -

Points
Available

ANSWER 4.06

a. the Shift Engineer **(+0.5)**

b. the Shift Engineer **(0.5)**

c days - Plant Manager **(+0.5)**
nights - Duty and Call Superintendent **(+0.5)**

Reference(s)

1. Palisades: Emergency Implementation Procedure, "Activation of the site Emergency Plan/Emergency Classification, EI-1, Rev. 9, pp. 1.

QUESTION 4.07

During the startup and increase-in-load of the turbine generator, the moisture separator reheaters (MSRs) are placed on-line. This is accomplished by opening the reheater-steam shutoff valves and placing into operation the reheater control system.

a. At what load (give approximate answer in %) in the turbine generator are the MSRs placed on-line. (0.5)

b. The reheater control valves are automatically timed to reach the fully-opened position with some time delay. What is this length of time (give approximate value including units)? (0.5)

ANSWER 4.07

a. 35% **(+0.5 for 30 to 40%)**

b. 2 hours **(+0.5 for 1 to 3 hours)**

- Section 4 continued on next page -

Points
Available

ANSWER 4.07 (contd)

Reference(s)

1. Palisades: System Operating Procedure, "Main Turbine and Generating Systems," SOP 8, Rev. 15, pp. 10.

QUESTION 4.08

At the Palisades Nuclear Plant, there is a Steam-Generator tube rupture which results in a leak greater than the capability of the CVCS.

a. List four (4) Automatic Actions that should occur. (2.0)

b. List two (2) indications and/or information that you could use to determine which Steam Generator has had the rupture. (1.0)

ANSWER 4.08

a. Automatic Actions

- Standby charging Pumps start.
- Reactor trip.
- Safety injection.
- Steam Generator blowdown valves shut.

(+0.5 each)

- Section 4 continued on next page -

Points
Available

ANSWER 4.08 (contd)

b. • Steam Generator sample.

• Radiation survey of steam pipe.

• Steam Generator level.

• Feedwater system response, i.e., valve position, level recorder trace, etc.

(+0.5 each, +1.0 max)

Reference(s)

1. Palisades: Emergency Operating Procedure, "Steam Generator Tube Rupture," EOP 8.2, Revision 14, pp. 1.

QUESTION 4.09

A reactor trip has occurred and a natural-circulation cooldown will be required.

List four (4) of the five (5) conditions that would indicate natural circulation has been established in at least one loop (about 10 min after RCPs were tripped). (2.0)

- Section 4 continued on next page -

Points
Available

ANSWER 4.09

- Loop ΔT (T_h - T_c) less than normal full power ΔT (47°F).
- Cold-leg temperatures constant or decreasing.
- Hot-leg temperatures stable (i.e., not steadily increasing).
- No abnormal differences between hot leg RTDs and core exit thermocouples. (Do not depend on a single indication. Use several thermocouples and all RTDs)
- Steam Generator levels are ≥ -84% (on wide range Steam Generator level instruments).

(+0.5 each, +2.0 max)

Reference(s)

1. Palisades: Off-Normal Operating Procedures, "Natural Circulation," ONP 21, Rev. 6, pp. 2.

QUESTION 4.10

The control room has just become uninnabitable due to a toxic gas problem. Before evacuation of the control room, the reactor is tripped at the console.

a. Now that the control room is no longer in operation, what action should be taken to ensure that the reactor has been tripped? (1.0)

b. If there had been no verification of a turbine trip before evacuation of the control room, what action should be taken? (1.0)

- Section 4 continued on next page -

Points
Available

ANSWER 4.10

a. "Ensure reactor trip by opening RPS 42-1 and 42-2 at CRDM clutch power supply transformers in cable spreading area, next to Battery Room (Room 224)." **(+1.0)**

b. "Manually trip the turbine locally at the turbine governor pedestal." **(+1.0)**

QUESTION 4.11

If a certain individual (male, over 18 years) has a completed NRC Form 4, itemize his quarterly limits for external radiation exposure by filling in the following table. (2.0)

	Palisades Control Level	NRC 10 CFR 20 Limit
Whole Body	-------	---------
Extremities	-------	---------
Skin	-------	---------

- Section 4 continued on next page -

Points
Available

ANSWER 4.11

In the following table, the answers are on the first line, the acceptable range is on the second and the point count on the third. All numbers are in mrem!

	Palisades Control Level	NRC 10 CFR 20 Limit
Whole Body	2500 **2500** **(+0.5)**	3000 **3000** **(+0.5)**
Extremities	15000 **12000-18000** **(+0.25)**	18750 **15250-21750** **(+0.25)**
Skin	4000 **3000-5000** **(+0.25)**	7500 **6500-8500** **(+0.25)**

Reference(s)

1. Palisades: Administrative Procedure, "Radiation Dosimetry," 7.04, Rev. 1, pp. 4.

- End of section 4 -

- End of Examination -

Master

U.S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: Palisades
REACTOR TYPE: CE PWR
DATE ADMINISTERED: December 11, 1984
EXAMINER: R. L. Higgins
APPLICANT: ____________________

INSTRUCTIONS TO APPLICANT:

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%.

Category Value	% Of Total	Applicant's Score	% Of Category Value	Category
25	25	_____	_____	5. Theory of Nuclear Power Plant Operation, Fluids, and Thermodynamics
25	25	_____	_____	6. Plant Systems Design, Control, and Instrumentation
25	25	_____	_____	7. Procedures - Norman, Abnormal, Emergency, and Radiological Control
25	25	_____	_____	8. Administrative Procedures, Conditions and Limitations
100	100	_____	_____	TOTALS

Final Grade ________%

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

____
Applicant's Signature

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

5.1 a. How are the terms "condenser vacuum" and "condenser back pressure" related? (1.0)

b. What backpressure limit has been established? (0.5)

c. Whose permission is required to exceed this limit? (0.5)

5.2 What is a divergent azimuthal xenon oscillation? (2.0)

5.3 What two factors cause β to be different than βeff? (2.0)

5.4 How is control rod worth related to moderator temperature? (2.0)

5.5 It has been suggested that the Sb-123 part of the sustainer source be neutron-irradiated for a couple of years to produce a longer lasting source of neutrons. Explain why you agree or disagree with this statement. (2.0)

5.6 Why, for the same boron concentration, is the differential boron worth greater at BOL than it is at EOL? (1.0)

5.7 a. What burnable poison, in addition to boron, is loaded into the Palisades reactor? (0.5)

b. Give two reasons for using this burnable poison instead of boron. (1.5)

5.8 What five indications are used to confirm the establishment of natural circulation flow? (2.0)

5.9 a. What is the purpose of establishing AO limits? (0.75)

b. What is meant by the term "quadrant power tilt" (Tq)? (0.75)

c. What action must be taken if Tq exceeds 15% when power is above 50%? (0.5)

5.10 Explain how the buildup of Pu-240 affects the magnitude of the fuel temperature coefficient. (2.0)

5.11 a. What is meant by the term "condensate depression"? (0.75)

b. Why is condensate depression desirable? (0.75)

5.12 The safety injection tanks are required by Tech Specs to have a level between 186 inches (55.5%) and 198 inches (59%). Why is a level greater than 198 inches (59%) undesirable? (1.5)

Assume

5.13 Answer the following multiple-choice questions concerning natural circulation. ($\dot{Q} = \dot{m}\ Cp\ \Delta T$)

a. If the heat generation rate decreases by a factor of two, the mass flow rate through the reactor will decrease by a factor of: (0.5)

1. two 2.00
2. 1.41
3. 2.83
4. 1.25

b. If the temperature difference between the hot leg and cold leg decreases by a factor of two, the mass flow rate will decrease by a factor of: (0.5)

1. two 2.00
2. 1.41
3. 2.83
4. 1.25

5.14 a. If the flow rate through a centrifugal pump increases, will the available NPSH increase or decrease? (0.5)

b. If the flow rate through a centrifugal pump increases, will the required NPSH increase or decrease? (0.5)

5.15 What is pump runout? (1.0)

6. Plant Systems Design, Control, and Instrumentation

6.1 a. What are the backup water supply sources for the Auxiliary Feedwater System? (1.0)

b. Explain how the FOGG system functions. (2.0)

6.2 What component will limit damage to a control rod should a "dry scram" (no water in the dashpot) occur? (0.5)

6.3 a. How is the primary coolant pump impeller fastened to the primary coolant pump shaft? (1.0)

b. Give two reasons for fastening the impeller to the shaft in the manner that it is. (1.0)

6.4 a. Other than the pressurizer safety valves and PORVs, what three other relief valves relieve to the Quench Tank? (1.5)

b. What two interlocks are associated with CV-0155, the air operated valve in the primary water supply line to the Quench Tank? (1.0)

6.5 a. State the type of detector used in the Wide Range Log Channel and briefly describe how the detector produces an electronic signal. (2.0)

b. Explain how the Wide Range Log Channel differentiates between neutron signal and gamma signal. (1.0)

6.6 What two parameters are used to determine the TMLP trip set point? (1.0)

6.7 Name the three override functions associated with the feedwater regulating valves and give the basis for each override function. (3.0)

6.8 a. Why do the reheater unit steam inlet valves trip shut on a turbine trip? (1.0)

b. What are the normal and backup air supplies to the MSIVs? (1.0)

6.9 a. What is the purpose of the reverse mode of cooling tower fan operation? (0.5)

b. What is the maximum recommended time interval for fan reverse operation? (0.5)

6.10 What is the purpose of adding concentrated sulfuric acid to the cooling tower basin? (1.0)

6.11 How does a DBA signal affect the containment air coolers? (2.0)

6.12 Explain how the SIRWT temperature is maintained above 40°F? (1.0)

6.13 Limitations are established to ensure proper minimum component cooling water pump discharge pressure for the number of pumps running and heat exchangers in service. What is the basis for these limitations? (1.0)

6.14 Explain how the breaker failure relay on the 345 KV switchyard breakers functions. (2.0)

7. Procedures - Normal, Abnormal, Emergency, and Radiological Control

7.1 What action must be taken if a plant transient occurs resulting in a low steam generator pressure safety actuation at a time when a main feedwater regulating valve is pinned open? (1.0)

7.2 a. "By commitment to the NRC, no load greater than ________ shall be handled over the spend fuel pool when it contains inadiated fuel which has less than 12 months' decay time unless it is moved per an approved procedure." (0.5)

b. Prior to removing the spend fuel pool bulkhead gate, the boron concentration in the tilt pit and fuel pool shall be verified to be greater than ______." (0.5)

c. "All loose hand tools, while in use over the reactor or fuel pool, shall be captured." What does "captured" mean, and why is it necessary? (1.0)

7.3 A PCP can continue to be operated with one of the three main pressure seals failed.

a. List two of the three indications which would signify the failure of an additional main seal. (1.0)

b. What actions must be taken if a PCP experiences a multiple seal failure? (2.0)

7.4 Steam Generator dryout will be apparent from four different indications. Name three of them. (1.5)

7.5 When must low temperature overpressure protection be armed? (1.0)

7.6 When must an inverse multiplication plot be used to predict the time when criticality will be achieved? (1.0)

7.7 What action must be taken if PCS temperature drops below 525°F while the reactor is critical? (1.5)

7.8 a. Per SOP 1, heatup and cooldown rates for the PCS shall not exceed ______. (0.5)

b. Per SOP 1, heatup and cooldown rates for the pressurizer shall not exceed ________. (0.5)

7.9 Why can no more than two PCPs be operated when PCS temperature is less than 250°F? (1.0)

7.10 Why should the waste gas decay tanks be purged with nitrogen prior to being opened? (1.0)

7.11 What two immediate actions must be taken if all three service water pumps trip (no service water system piping break exists)? (1.0)

7.12 Why does the Loss of Component Cooling procedure require that the Variable Speed Charging Pump be stopped should a loss of Component Cooling occur but permit the manual operation of the constant speed charging pumps? (1.0)

7.13 Under what two circumstances does the Loss of Instrument Air procedure require the reactor to be tripped? (1.0)

7.14 One immediate action in the Loss of Coolant Accident procedure requires that an attempt be made to locate and isolate the break if possible. Name the five steps which the procedure requires the operator to take to try to locate and isolate the the break. (2.5)

7.15 What three immediate actions are required to be taken if a fire renders several major pieces of safety related equipment inoperable? (1.5)

7.16 List the four methods specified in the Steam Generator Tube Rupture procedure for determining which Steam Generator is leaking. (2.0)

7.17 Explain why a void forming in the Reactor Vessel will cause erratic indication of the startup neutron detectors. (2.0)

8. Administrative Procedures, Conditions, and Limitations

8.1 a. What plant operating limitation is imposed, if any, when no feedwater heaters are available? (0.5)

b. Why is this load restriction imposed? (1.0)

8.2 What three actions must be taken if the maximum ice limits for the Palisades switchyard lines are exceeded? (1.5)

8.3 a. How many Senior Operators must be on shift when the reactor is operating? (0.5)

b. For shift manning considerations, when is the reactor considered operating? (1.0)

8.4 a. What requirement is imposed on the PORVs when PCS temperature is above 325°F and LTOP is unarmed? (0.5)

b. What is the basis for the requirement of part "a"? (1.0)

8.5 What action must be taken if one of the Engineered Safeguards Room radiation monitors RIA-1810 or 1811 becomes inoperable? (1.0)

8.6 If an LCO or associated action requirement can not be satisfied because of circumstances in excess of those addressed in the specification, what actions must be taken? (1.0)

8.7 What requirement must be met prior to reducing primary coolant boron concentration below cold shutdown boron concentration? (1.0)

8.8 a. When must a vent path from the reactor vessel and pressurizer be available for use? (0.5)

b. What level must be established in both containment spray headers prior to raising PCS temperature above 325°F? (0.5)

c. How many AFW pumps are required to be operable prior to raising PCS temperature above 325°F? (0.5)

d. When are two independent Control Room Emergency Air Cleanup Systems required to be operable? (0.5)

8.9 A consideration in deciding if an occurrence is reportable is whether or not the occurrence involved a Basic Component. What is a Basic Component? (1.5)

8.10 a. What is the Tech Spec limit for specific activity of the secondary coolant in a steam generator? (0.5)

b. What is the basis for the limit in part "a"? (1.0)

c. What action must be taken if the limit in part "a" is exceeded? (1.0)

8.11 a. Which on-shift individual functions as Site Emergency Director upon initiation of the Site Emergency Plan until relieved by a higher ranking supervisor? (0.5)

b. What is the minimum number of fire brigade personnel required to be maintained onsite at all times? (0.5)

c. Which on-shift individual maintains the status board for equipment affecting limiting conditions of plant operation? (0.5)

d. Which on-shift individual functions as the Senior Management representative for site operation when Senior Management is not onsite? (0.5)

e. Which on-shift individual is responsible for notifying the NRC Operations Center via the Emergency Notification System of the declaration of any of the Emergency Classes specified in the Emergency Plan? (0.5)

8.12 With the plant less than 210°F, what component(s) must be operable prior to removing more than one preferred AC bus from service? (1.0)

8.13 When directed by a procedure to check the position of a valve or breaker and its position is found to be different from that specified, what action must be taken? (1.0)

8.14 What is Emergency Maintenance? (1.0)

8.15 Which two categories of equipment require shift supervisor approval prior to the performance maintenance? (1.0)

8.16 a. When shall a CPIT sticker be removed? (0.5)

b. Which personnel are permitted to remove CPIT stickers? (0.5)

8.17 What actions should personnel take, unless directed otherwise by Radiation Safety, if the red light illuminates on a Constant Air Monitor? (1.0)

8.18 Name the three decisions the Site Emergency Director is responsible for making which may not be delegated by him to another individual. (1.5)

Master

Answers - Section 5

5.1 a. Atmospheric pressure minus condenser backpressure equals condenser vacuum. (1.0)

b. 4.3 inches of mercury (0.5)

c. Specific written approval of the Plant Superintendent, Operations/Maintenance Superintendent, or Operations Supervisor. (0.5)

Ref: Standing Order No. 16

5.2 Cyclical increase and decrease in xenon concentration (and corresponding opposite variation in power) in an angular direction around the longitudinal axis of the core. (1.0) Divergent means that the cycles get larger in magnitude over time. (1.0)

Ref: Tech Specs p. 3-68

5.3 Since delayed neutrons are born at a lower energy, a smaller fraction of delayed neutrons will leak out of the core, causing βeff to be larger than β. (1.0)

Since delayed neutrons are born at an energy below the threshold of fast fission, they will not induce fast fission, causing Beff to be smaller than β. (1.0)

Ref: NUS Volume 3 p. 5.3-2

5.4 As the moderator temperature increases, the moderator density decreases. There is less competition for the absorption of thermal neutrons and the decreased moderation results in a higher epithermal flux impinging on the control rods. (1.0) Since the control rods see a higher epithermal flux and have less competition from the moderator, they are worth more. (1.0)

Ref: NUS Volume 3 p. A.9-6

5.5 Disagree. (0.5) Most of the Sb-123 would be transformed to Te-124. Therefore there would be very little Sb-123 left to absorb a neutron, emit a high energy gamma during beta decay, and induce neutron emission. (1.5)

Ref: NUS Volume 3 p. A.11-4

5.6 Due to the buildup of fission product poisons (which compete with the boron for neutrons). (1.0)

Ref: NUS Volume 3 p. 15-23

5.7 a. Gadolinium (0.5)

b. Gadolinium is a better neutron absorber (0.75)
Gadolinium burns out faster (0.75)

Ref: Nuclear Operator Training Program, Volume II.C, Module 24b, Student Handout p. 12

5.8 1. Loop ΔT (Th-Tc) less than normal full power ΔT (47°F). (0.4)

2. Cold leg temperature constant or decreasing. (0.4)

3. Hot leg temperatures stable (not steadily increasing). (0.4)

4. No abnormal differences between hot leg RTDs and core exit thermocouples. (0.4)

5. Steam Generator levels greater than -84%. (0.4)

Ref: ONP-21 p. 2

5.9 a. Ensure operation within allowable LHR limits. (0.75)

Ref: Tech Specs p. 3-66a

b. The difference between nuclear power in any core quadrant and the average in all quadrants. (0.75)

Ref: Tech Specs p. 1-2

c. Be in hot standby within 12 hours. (0.5)

Ref: Tech Specs p. 3-112

5.10 Pu-240 has a strong resonance low in the epithermal range. (1.0)
Its build up results in more resonance absorption, causing the fuel temperature coefficient to become more negative. (1.0)

Ref: NUS Volume 3 p. A.9-1

5.11 a. The process of cooling the bulk condensate below saturation temperature. (0.75)

b. Increases the NPSH of the condensate pumps. (0.75)

Ref: Westinghouse Thermal Hydraulic Principles and Applications to the Pressurized Water Reactor II, p. 9-22, 10-55

5.12 Less than the required amount of water would be injected during an accident (0.75) because less nitrogen gas would be available to push the water into the reactor. (0.75)

Ref: Tech Specs p. 3-31

5.13 a. 4 - 1.25 (0.5)

b. 2 - 1.41 (0.5)

Ref: Westinghouse Thermal Hydraulic Principles and Applications to the Pressurized Water Reactor II, p. 14-25

5.14 a. Decrease (0.5)

b. Increase (0.5)

Ref: Westinghouse Thermal Hydraulic Principles and Applications to the Pressurized Water Reactor II, P. 10-55, 57, 44

5.15 A condition of maximum flow in a pump occurring when the pump head equals zero. (1.0)

Ref: Westinghouse Thermal Hydraulic Principles and Applications to the Pressurized Water Reactor II, p. 10-81

Answers - Section 6

6.1 a. Fire protection system for all AFW pumps (0.5)
Service Water system for P-8C (0.5)

b. The Feed Only Good Generator (0.5) will shut motor operated valves in the AFW supply lines to the steam generator (0.5) which has a level less than 447 inches above the bottom of the steam generator support skirt (0.5) and concurrently has a pressure which is more than 150 psi lower than the pressure in the other steam generator. (0.5)

Ref: AFW Lesson Plan Outline

6.2 Energy absorbing device. (0.5)

Ref: Lesson Notes 2-6

6.3 a. They are bolted and pinned together. (1.0)

b. 1. Allow differential thermal expansion (0.5)
2. Facilitate removal of the impeller for maintenance (0.5)

Ref: Lesson Notes 3-14

6.4 a. Shutdown cooling relief (0.5)
SI tanks drain relief (0.5)
Letdown line relief (0.5)

Ref: M-201 sheet 3

b. CV-0155 can not be opened if a containment high pressure (0.5) or a containment high radiation signal is present (0.5).

Ref: Lesson Notes 8-8

6.5 a. The Wide Range Log Channel uses a fission chamber. (0.5) The fission chamber is coated with uranium-235, which fissions when absorbing neutron. (0.75) The fission will produce highly positively-charged fission products and 92 negatively charged electrons, generating a large electronic pulse. (0.75)

b. The wide range uses two methods: pulse height discrimination in the lower 5 decades of the detector's range (0.5) and Campbelling (square law detection) in the upper 5 dacades of the detector's range. (0.5)

Ref: Lesson Notes 11-8, 14-28

6.6 reactor inlet temperature (0.5)
reactor outlet temperature (0.5)

Ref: Tech Specs p. 2-8

6.7 1. Feedwater regulating valves fail shut on high level (0.5) to prevent overfeeding the steam generator thereby causing carryover and resultant turbine damage. (0.5)

2. Feedwater regulating valves fail shut on low steam generator pressure (0.5) to prevent feeding water to a steam generator which has suffered a main steam line break, so that the resultant PCS cooldown and containment pressurization will be less severe. (0.5)

3. Feedwater regulating valves fail "as is" when a turbine trip occurs (0.5) assuring a predictable feedwater flow to the steam generators as the speed of the feedwater pump ramps down after the trip (0.5)

Ref: Lesson Notes 17-3

or prevent overpressurizing the MSRs

6.8 a. Prevent unnecessarily drawing steam from the main steam line. (CAF) flooding reheater drain tanks and feedwater heaters (1.0)
E-6A/B, which could lead to water hammer and unnecessary steam release.

b. Normal supply - (125 psi) instrumentation air to the turbine bldg (0.5)
Backup - turbine building high pressure air (0.5)

Ref: Lesson Notes 18-21; Lesson Plan 18b p. 20

6.9 a. Melting ice on the cooling towers. (0.5)

b. (~~30 minutes~~) never used (0.5)

Ref: Lesson Notes 23-10

6.10 Preserve the redwood cooling towers. maintain pH to prevent scaling (1.0)

Ref: Lesson Notes 23-7

6.11 1. (All 8 fans trip) ~~(0.5)~~

2. After emergency diesel generator auxiliary power is available the 100 hp fans V-1A, V-2A, V-3A ~~start~~ and V-4A ~~is placed on Standby.~~ start (1.0)

3. The high capacity service water outlet valves from the containment air coolers open. 1.0 ~~(0.5)~~

Ref: Lesson Notes 24-9

6.12 Heating steam is supplied to SIRWT heat exchanger E-57. (0.5) SIRWT recirc pump recircs SIRWT water through the heat exchanger. (0.5)

Ref: Lesson Notes 10-8; M204 F-2

6.13 A discharge pressure less than the minimum CCW pump discharge pressure limits may lead to tube vibration in the CCW heat exchangers (0.5) due to the increase in CCW volume flow rate (0.5).

Ref: System Functional Description 1-11 p. 14

6.14 If a fault occurs, breaker failure relay timing is initiated by operation of any breaker trip signal. (1.0) If a breaker fails to open and clear the fault during the time delay (6 cycles) adjacent breakers will be tripped by the breaker failure auxiliary relays to clear the failed breaker from the fault. (1.0)

Ref: Lesson Notes 32-6

Answers - Section 7

7.1 Immediately trip the condensate pumps. (1.0)

Ref: Standing Order No. 55

7.2 a. The weight of a fuel bundle. (0.5)

b. 1750 ppm (.4 awarded for refueling boron conc, which is 1720 ppm) (0.5)

c. The tool shall be attached to one end of a rope, and the other end of the rope shall be fastened to a guard rail or other stationary, sturdy object outside the reactor or fuel pool. (0.5) In this way a tool which falls into the reactor or fuel pool can be easily retrieved. (0.5)

Ref: SOP 28 steps 4.4, 4.6, 4.16

7.3 a. Any two of the following:

1. seal leak off temperature above 155°F. (0.5)
2. leak off flow greater than 2 gpm. (0.5)
3. seal pressure greater than 1400 psi. (0.5)

b. 1. Conduct an orderly plant shutdown (in accordance with GOP 8) (0.5), keeping the affected pumps running (0.5).

2. Initiate plant cooldown (in accordance with GOP 9) (0.5). When the step (in GOP 9) is reached calling for the stopping of 2 PCPs - stop the affected pump and another in the opposite loop <u>not</u> connected to the pressurizer spray line. (0.5)

Ref: Standing Order Number 32

7.4 Any three of the following:

1. Wide range steam generator level less than -125% (0.5)
2. Decrease in steam generator pressure below Psat of the primary. (0.5)
3. Abnormal increase in primary system temperature and pressure. (0.5)
4. Increase in letdown flow. (0.5)

Ref: EOP 1 p. 3

7.5 Whenever PCS temperature is 300°F or less (0.5) or whenever MOV-3015 or 3016 are open. (0.5)

Ref: GOP 2 Step 5.1

7.6 Whenever only one source range instrument is operable. (1.0)

Ref: GOP 3 step 5.0

7.7 Immediately insert regulating rods (SOP 6) (0.5) until temperature is restored to 525°F or above (0.5) or until Groups 3 and 4 rods are fully inserted. (0.5)
half credit granted for "restore Tave to 525 or shutdown"

Ref: Temporary Change to GOP 3 Step 2.2

7.8 a. 60°F/hr (0.5)

b. 150°F/hr (0.5)

Ref: SOP 1 step 4.0.a

7.9 Because of the denser water at low temperatures, the mass flow rate through the core would be greater (0.5), resulting in excessive core uplift. (0.5)
half credit - reduce PCS heatup rate

Ref: SOP 1 step 4.0.o

7.10 Reduce the possibility of an explosion (0.5) by purging any hydrogen present out of the tank using a noncombustible gas. (0.5)
prevent inadvertent release of radioactive gas ← partial credit ~~1.0~~ (0.5)

Ref: SOP 18A step 5.0.a

7.11 1. Attempt to restart a service water pump. (0.5)
2. If noncritical service water can not be restored immediately, trip the reactor and turbine/generator. (0.5)

Ref: EOP 3 step 3.0

7.12 Component Cooling water is needed to cool the variable speed charging pump's fluid drive. The constant speed charging pumps do not have fluid drives. (1.0)

Ref: Lesson Notes No. 6, p. 28

7.13 Any two of the following:

1. Instrument air pressure is less than 50 psig. (0.5)
2 First indication of erratic equipment behavior. (0.5)
3. Instrument air pressure is dropping rapidly. (0.5)

Ref: EOP 5, p. 1

7.14 1. Isolate letdown line by closing CV-2009 (outside containment) or CV-2001 (inside containment). (0.5)

2. Verify PORVs and isolation MOVs closed. (0.5)

3. Isolate PCS sample lines (close CV-1910 and CV-1911) (0.5)

4. Check Acoustical Monitor Panel EC-51 and Pressurizer Relief Valve Discharge Temperature Indicators for symptoms of pressurizer relief valve leakage. (0.5)

5. Check Quench Tank level, pressure and temperature indications normal. (0.5)

Ref: EOP 8.1 step 3.5

7.15 1. Trip the reactor. (0.5)

2. Emergency borate to the cold shutdown concentration. (0.5)

3. Activate the Site Emergency Plan. (0.5)

Ref: EOP 10.1 Step 3.0

7.16 1. Steam Generator sample (0.5)

2. Radiation survey of steam pipe (0.5)
steam line radiation monitors (.3)

3. Steam Generator level (0.5)

4. Feedwater response (0.5)

Ref: EOP 8.2 Step 3.1

7.17 The Reactor Vessel level will fluctuate as the void forms (0.5). When the level in the Reactor Vessel is above the startup range detectors, fewer neutrons will reach the detectors, so the count rate will be low (0.75). When the level in the Reactor Vessel is below the startup range detectors, the amount of neutron attenuation will decrease, so more neutrons will reach the detectors, causing the count rate to rise (0.75).

Ref: ONP 21 step 4.0 (16)(f)

Answers - Section 8

8.1 a. Power Operation is not permitted. (0.5)

b. Minimum feedwater temperature of 250°F to the steam generators can not be obtained. (1.0)

Ref: Standing Order No. 14 Step 3.c

8.2 1. The Plant General Manager or Duty and Call Superintendent shall be notified of the existing condition. (0.5)

2. Power Control shall be informed of the existing condition. (0.5)

3. Commence plant shutdown to hot standby. (0.5)

Ref: Standing Order No. 22

8.3 a. 2 (0.5)

(greater than cold shutdown)

b. When it is in a mode other than cold shutdown or refueling. (1.0)

Ref: Standing Order No. 51

8.4 a. The PORV breakers (52-196 and 52-224) shall be normally open. (0.5)

b. Due to the possibility that a Cable Spreading Room fire or a Control Room fire could cause "hot shorts" simultaneously in control circuits for MO-1042A and PRV-1042B or in MO-1043A and PRV-1043B, (0.5) resulting in the equivalent of a LOCA during a fire. (0.5)

Ref: Standing Order No. 52

8.5 The ventilation dampers associated with that RIA shall be closed. (1.0)

Ref: Standing Order No. 53

8.6 The plant shall be placed in at least hot shutdown within the next six hours, (0.5) and in at least cold shutdown within the following 30 hours. (0.5)

Ref: Standing Order No. 54, 3.0.3

8.7 A steam bubble (0.5) and normal water level are established in the pressurizer. (0.5)

Ref: Standing Order No. 54, 3.1.3e

or

PCP or shutdown cooling pump shall be in operation
SOP 1 step 4.0 r

o: shutdown banks in fully withdrawn position (.5)
Ref: SOP 2A step 4.0.b

Containment integrity must be established. (1.0)

Ref: Standing Order No. 54, 3.6.1.c

8.8 a. At all times. (0.5)

b. A level greater than the 735 foot elevation. (0.5)

c. Three (0.5)

d. At all times. (0.5)

Ref: Standing Order No. 54, 3.1.9; 3.2.4; 3.5.1; 3.14.1

8.9 A component necessary to assure any one of the following for full credit

1. The integrity of the reactor coolant pressure boundary, or (0.5)
2. The capability to shutdown the reactor and maintain it in a safe shutdown condition, or (0.5)
3. The capability to prevent or mitigate the consequences of accidents which could result in potential exposures comparable to those referred to in 10 CFR 100.11. (0.5)

Ref: Admin. Procedure 3.03, Attachment 13, p. 2

8.10 a. .1 uCi/gram dose equivalent I-131 (0.5)

b. Limit offsite radiation dose to a small fraction of 10 CFR 100 limits (0.5) in the event of a steam line rupture. (0.5)

c. The reactor shall be placed in hot shutdown within 6 hours (0.5) and in cold shutdown within the following 30 hours. (0.5)

Ref: Tech Spec p. 3-20, 21

8.11 a. Shift Engineer (0.5)

Ref: Admin. Proc. 4.00 step 4.8.1.o

b. 5 (0.5)

Ref: Tech Specs, Step 6.2.2.f

c. Shift Supervisor/Operations (0.5)

Ref: Admin Proc 4.00 step 4.7.1.d

d. Shift Engineer (0.5)

Ref: Admin Proc. 4.00 step 4.7.1.n

e. Shift Engineer (0.5)

Ref: Admin Proc 4.01 step 5.4.2

8.12 Both Diesel Generators (partial credit - other 3 busses or bypass regulator) (1.0)

- Ref: Admin Proc 4.01 step 5.11.3

8.13 Notify the Shift Supervisor (0.5) who will authorize the positioning of the valve or breaker. (0.5)
note the position on the exceptions portion of the valve/bkrs checklist

Ref: Admin. Proc. 4.02 step 5.2

8.14 Maintenance or repair activities immediately necessary to allow safe shutdown of the plant or to reduce the imminent possibility of injury to personnel, significant damage to major equipment, or hazard to health and safety of the public. (1.0)

Ref: Admin. Proc. 4.03 step 5.0.b

8.15 1. Equipment which affects plant operation on any mode. (0.5)

2. Q and/or TS related equipment. (0.5)

Ref: Admin. Proc. 4.03 step 6.2

8.16 a. When the corresponding MO is declared operable. (0.5)

b. Only Operations Department personnel. (0.5)

Ref: Admin. Proc. 4.03 step 11.2

8.17 1. Evacuate the area and proceed to the Change Room at Access Control. (0.5)

2. Perform a whole body frisk (.25) and report to the Radiation Safety Office (.25).

Ref: Admin Proc 7.00 step 5.0.b

or

Evacuate the area (0.5) and notify the Control Room and Radiation Safety (0.5).

Ref: Admin Proc 7.03 Attachment 2, Step 19

8.18 1. Decision to recommend protective actions to offsite organizations. (0.5)

2. Decision to evacuate the site. (0.5)

3. Decision to authorize exposures that exceed the 10 CFR 20 regulatory limits for emergency workers. (0.5)

Ref: Site Emergency Plan, p. 5-6

4. classification of the emergency (0.5)

5. activation and directing of the TSC (0.5)