UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

ENCLOSURE 1

EXAMINATION REPORT NO. 50-335/OL-84-01

Facility Licensee: Florida Power and Light Company
P. O. Box 128
Ft. Pierce, FL 33454

Facility Name: St. Lucie 1 & 2

Facility Docket Nos.: 50-335 and 50-389

Written and oral examinations were administered at St. Lucie near Ft. Pierce, Florida.

Chief Examiner: Sandy Lawyer 11/9/84
for A. H. Johnson Date Signed

Approved by: Sandy Lawyer 11/9/84
for Bruce A. Wilson, Section Chief Date Signed

Summary:

Examinations on June 18-22, 1984

Written and oral examinations were administered to 12 candidates 7 of whom passed. For the purpose of dual licensing on St. Lucie 1 & 2, plant oral difference examinations were administered to 15 candidates, all of whom passed.

REPORT DETAILS

1. Persons Examined

SRO Candidates:

G. Kaasa
R. Kress
S. Nye
W. Taylor
S. Valdez
C. Couture - (Differences)
M. Gilmore - (Differences)
W. Hagar - (Differences)
J. Imbriale - (Differences)
H. Johnson - (Differences)
C. Wood - (Differences)

RO Candidates:

E. Holland
J. Holt
D. Hurlbut
M. Langnes
M. Serruto
C. Simpkins
M. Wright
M. Ames - (Differences)
D. Davidson - (Differences)
L. Heffelfinger - (Differences)
R. McElroy - (Differences)
S. Patterson - (Differences)
B. Poole - (Differences)
J. Sandy - (Differences)
J. Talley - (Differences)
C. Ward - (Differences)

Other Facility Employees Contacted:

*J. Barrow, Operations Superintendent
*D. A. Sager, Operations Supervisor
*P. L. Fincher, Nuclear Training Supervisor
*J. Spodick, Nuclear Training
*C. D. Marple, Nuclear Training

*Attended Exit Meeting

2. Examiners:

**A. H. Johnson
L. Lawyer
J. D. Smith
R. Clark
J. Upton
A. Prichard

**Chief Examiner

3. Examination Review Meeting

At the conclusion of the written examinations, the examiners met with J. Spodick, C. Marple, P. Fincher, and R. Weller, to review the written examinations and answer keys. The following is a list of the comments made by the facility staff and the NRC resolution of each of the comments:

a. RO Exam

(1) Question 1.4.a.

Facility Comment: Raising Tave by dilution should also be an acceptable answer.

NRC Resolution: Facility procedures address this plant evolution. Additional answer was accepted and added to answer key.

(2) Question 1.8.b.

Facility Comment: At EOL, reactivity added to establish a 1 DPM SUR is different than @ BOL due to change in beta.

NRC Resolution: Answer is acceptable on the basis of listed references.

(3) Question 2.2

Facility Comment: Relief valves at charging pump suction are designed for thermal expansion relief instead of overpressure protection.

NRC Resolution: Answer accepted. See FSAR, pg. 9.3-29 attached to Master Examination.

(4) Question 2.3.a.

Facility Comment: Concern over wording of question. Should candidates assume sequence starts after dryout of the SG and should operator action be assumed?

NRC Resolution: Agree that more specific wording is desireable. Grading will take into account both concerns.

(5) Question 2.3.b.

Facility Comment: Question does not specify number of indications required for full credit.

NRC Resolution: Since answer key lists six indications, we believe it is reasonable to expect half that number for full credit.

(6) Question 2.4

Facility Comment: Drawing in answer key has been updated. Facility will supply more current drawing.

NRC Resolution: Updated drawing accepted.

(7) Question 2.5.b.

Facility Comment: Answer key is redundant with regard to non-essential header being isolated.

NRC Resolution: Agree. Clarified answer is listed on Master Examination.

(8) Question 2.5.d.

Facility Comment: Interlock on CCW Surge Tank isolation valve is different between Units 1 and 2.

NRC Resolution: Answer key changed. LP+SD #9 supports this answer.

(9) Question 2.8.a.

Facility Comment: Answer key incorrect. There is an alarm but no pressure indication.

NRC Resolution: Answer key changed. LP+SD #2 supports this answer.

(10) Question 2.8.b

Facility Comment: Additional acceptable answer should be performance of an RCS leakage calculation.

NRC Resolution: Answer accepted based on routine facility procedures.

(11) Question 2.9.a.

Facility Comment: Answer is incorrect. It should be (2) and (4) for OPC and (1), (2), (3), and (4) for backup OPC.

NRC Resolution: Answer is accepted based on Westinghouse turbine system description.

(12) Question 3.1

Facility Comment: Answer key is incorrect and/or confusing for all four parts.

NRC Resolution: Changes and additions made to answer key based on:

(a) Inadequate proofreading of answer key following typing.

(b) Obtaining answer from NRC Question Bank which was not sufficiently detailed.

(c) Lesson Plan and System Description provided by facility.

(13) Question 3.2

Facility Comment: Answer key in (a) and (c) is incorrect.

NRC Resolution: Typographical errors and clarification made based on LP+SD.

(14) Question 3.6

Facility Comment: Unit 2 MSIS setpoint is 600 psia.

NRC Resolution: Answer key changed based on Unit 2 Tech. Specs.

(15) Question 3.7

Facility Comment: Only Unit 1 has three HPSI pumps. Answers (a) and (c) are incorrect.

NRC Resolution: Answer (a) changed based on pumps characteristic curve attached to Master Examination.

Facility provided sketches are acceptable answers for Part (c) - based on LP+SD #4.

(16) Question 3.8

Facility Comment: Answer for Spray Valve under condition 2 is different if Unit 2 is assumed.

NRC Resolution: Off-Normal procedure 2-0120035 provides acceptable answer for Unit 2.

(17) Question 4.3

Facility Comment: Allocation of points on answer key do not agree with exam question.

NRC Resolution: Answer key changed. Inadequate proofreading following typing of answer key.

(18) Question 4.7

Facility Comment: Same comment as 4.3.

NRC Resolution: Same resolution as 4.3.

(19) Question 4.12

Facility Comment: Unit not specified in Question. Answer will be the same except for setpoints.

NRC Resolution: If candidates provide setpoints, they will be graded according to unit assumed.

(20) Question 4.13

Facility Comment: Answer should include "and/or level-pressure override."

NRC Resolution: Answer accepted. Relay logic diagram is attached to Master Examination.

(21) Question 4.15.c.

Facility Comment: Exposure limits are incorrect.

NRC Resolution: Correct limits were provided by facility and are attached to Master Examination.

(22) Question 4.16

Facility Comment: Question is vague and may elicit several different answers.

NRC Resolution: Question deleted following grading process based on obvious confusion shown by candidates.

b. SRO Exam

(1) Question 5.11

Facility Comment: Same as RO Exam, Question 3.6.

NRC Resolution: Same response.

(2) Question 5.13

Facility Comment: Possible confusion over the term "wet steam."

NRC Resolution: "Wet steam" was defined as having a quality between 5-95%.

(3) Question 6.1.a.

Facility Comment: Detail on answer key may not be provided by candiates. Point value on answer key does not match question.

NRC Resolution: Answer key changed to provide a number of acceptable responses. This is an essay type response that is difficult to assign specific point values to different parts.

(4) Question 6.1.b.

Facility Comment: Question is vague and may elicit a variety of acceptable responses.

NRC Resolution: Agree. Answer key has alternate response added based on Rod Misalignment procedure.

(5) Question 6.1.c.

Facility Comment: Answer key incorrect.

NRC Resolution: Answer key changed based on off-normal procedure instead of system description.

(6) Question 6.3

Facility Comment: Same as RO, Question 2.2.

NRC Resolution: Same response.

(7) Question 6.4

Facility Comment: Same as RO, Question 2.4.

NRC Resolution: Same response.

(8) Question 6.6

Facility Comment: Same as RO, Question 3.1.

NRC Resolution: Same response.

(9) Question 6.7

Facility Comment: Same as RO, Question 3.8.

NRC Resolution: Same response.

(10) Questions 7.1 and 7.2

Facility Comment: Answers are different depending upon which units are assumed in answer.

NRC Resolution: Either unit will be acceptable since question did not specify.

(11) Question 7.3.b.

Facility Comment: High pressurizer pressure should also be accepted.

NRC Resolution: Answer was accepted if proper assumptions were stated.

(12) Question 8.1.c.

Facility Comment: If Unit 2 is assumed, there is no correct answer since LCO is 15 kw/ft. per Unit 2 Tech Specs.

NRC Resolution: Will accept "No action required" if Unit 2 is specified.

(13) Question 8.3

Facility Comment: Concern was voiced over level of difficulty of question.

NRC Resolution: All relevant E-Plan information to answer question was provided with examination. No change to answer is required.

(14) Question 8.7.b.

Facility Comment: Answer key should state "seven days or until completion of job."

NRC Resolution: Answer accepted. HP-2 supports this addition.

(15) Question 8.10.c.

Facility Comment: Answer is incorrect or confusing.

NRC Resolution: Inadequate proofreading following typing. Answer changed to conform with T.S. 3.3.1.

4. Exit Meeting

At the conclusion of the site visit, the examiners met with representatives of the plant staff to discuss the results of the examination. Those individuals who clearly passed the oral examination were identified. There was no generic weakness. The cooperation given to the examiners and the effort to ensure an atmosphere in the control room conducive to oral examinations was also noted and appreciated.

ENCLOSURE 3
(1 of 2)

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

Facility: St Lucie 1 & 2

Reactor Type: CE

Date Administered: 19 June 84

Examiner: Upton Smith

Candidate: Answer key

INSTRUCTIONS TO CANDIDATE:

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

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

Final Grade ________ %

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

____
Candidate's Signature

Points Available

5.0 THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS AND THERMODYNAMICS (25)

The following statements apply to questions 5.1, 5.2, 5.3, 5.4 and 5.5. St. Lucie Unit 2 has been operating at a steady 100% of full power for 7 days with all of the control rods fully withdrawn from the nuclear reactor core. The burnup is 5000 EFPH on cycle 1. Use any of the provided figures and tables. Show your work and/or how you arrived at your answer.

5.1 What is the value of boron worth in (% Δk/k)/ppm? (1.5)

5.2 Explain why the critical Soluble Boron Concentration vs. burnup (Figure C.2) is somewhat constant at approximately 430 ppm between 500 and 3000 EFPH. (1.5)

5.3 The nuclear reactor power level is now reduced to 70% of full rated power at a rate of 1% per minute by inserting rods (programmed bank motion). What is the rod position when the power change is complete? (2.0)

5.4 Sketch a graph of Xenon worth for a period of 0 to 100 hours after the reduction to 70% of full power. Indicate values of pcm. and hours. Explain your work. (2.0)

5.5 You are to maintain this 70% of full power for the first 20 hours after the power-level change. You do not wish to move the control rods. Explain what, if any, change will be required to the boron concentration. Give numerical values, if possible. Neglect any effect due to Samarium poisoning. (2.0)

Answer(s)

5.1 Using Figure C.1, boron worth = 12.65 pcm/ppm. (+1.0 for 12.6 to 12.7)

% k/k = 10^{-3} pcm/ppm
= .01265 % Δk/k/ppm. (+0.5)

5.2 For the first (initial) fuel load, burnable poison rods [that are not part of a CEA (i.e., not moveable)] are loaded into the nuclear-reactor core. The material is usually a boron-10 based material which is inserted in place of thimble plugs. The result is that the poison burns out at a rate that when combined with the fuel burnup rate results in a flat reactivity change as a function of EFPH. When the burnable poison is consumed, the reactivity change vs. EFPH is essentially linear. (The burnable poison is used to insure a negative moderator temperature coefficient for a reactor core at initial BOC.) (+1.5)

Burnable poisons are depleting (+1.0) at the same rate as fuel is depleting and FP are built in (+0.5)

5.3 30%/(1%/min) = 30 min.
30 min. is fast with respect to Xenon changes

using Figure A.1
reactivity = -1137 - (-780)
= -357 pcm (+1.0 for -367 to -347 pcm)

using Figure A.6 (HZP is the only curve provided)
4532 - 357 = 4175 pcm
4175 pcm = 106 inches on Bank 5 (+1.0 for 82 to 109)

5.4 using Figure A.3

equal Xe 100% power = -2648 pcm
equal Xe 70% power = -~~2079~~ 2479 pcm (+1.0 for 2300 to 2450)

using Figure A.4

see estimate of curve (+1.0)

5.5 Xenon concentration increases for the first 7.5 hours from -2648 pcm to about -3500 pcm, a difference of about -852 pcm (+.33) So dilute (~~+0.8~~) (+0.5).

852 pcm/12.65 pcm/ppm = 67.4 ppm (+0.8 for 48 to 127 ppm) (+.33)

Xenon then decreases back to the original pcm in 20 hours. So borate (~~+0.8~~ +0.5) back 67.4 ppm (+0.4 based on curve at 20 hrs.). (+.33)

(~~+2.0 max~~) .33 for each of 3 numbers +.5 for dilute
+.5 for borate

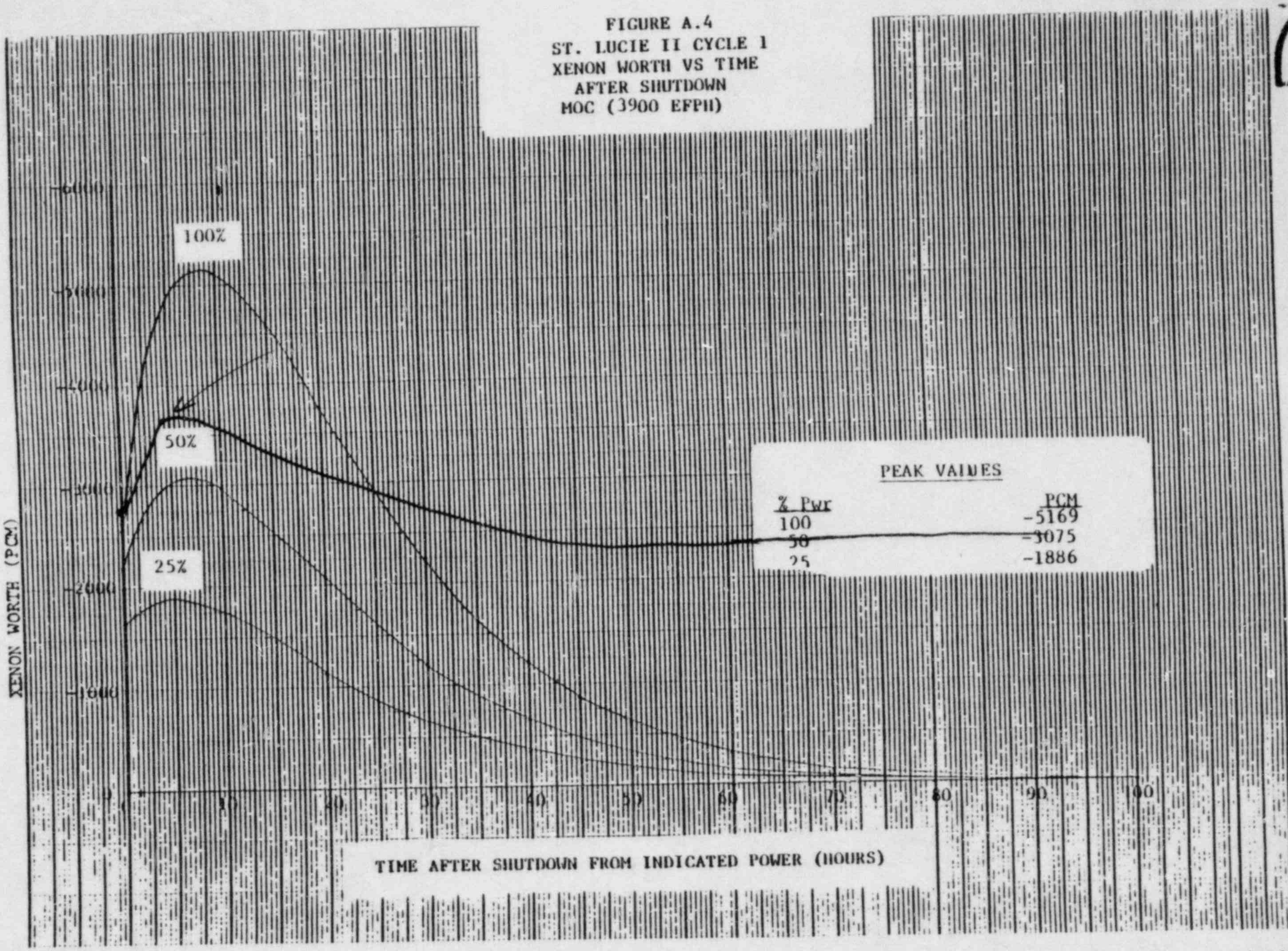
Reference(s)

1. Unit 2 Plant Physics Curves, SL 2.

2. C-E PWR Simulator Training Facility, "Reactor Theory," SL 1&2.

461510

FIGURE A.4
ST. LUCIE II CYCLE 1
XENON WORTH VS TIME
AFTER SHUTDOWN
MOC (3900 EFPH)



2/24/84

Points Available

5.6 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 about 50% of the energy released per fission event.

(b.) Fissioning of the isotope, U^{238}, provides about 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 energy is deposited as heat in the coolant.

Answer(s)

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

Reference(s)

1. C-E PWR Simulator Training Facility, "Reactor Theory," Florida Power and Light Company, SL 1&2.

pp 500 (82 H8) / ds - 42, 43

Points Available

5.7 The nuclear reactor has been operating at a constant 60% of full power for 4 days with all rods out. A control rod drops into the core.

a. Explain how and why the dropped rod will affect the maximum linear power density (the peak value for the generated thermal power per foot for the fuel pins) in the core of the nuclear reactor. (1.5)

b. If the dropped rod is not recovered for a time of one hour, explain how and why the maximum linear power density will further change. (1.5)

Answer(s)

5.7 a. . When a control rod is dropped into the nuclear-reactor core, the neutron flux will be depressed in the vicinity of the dropped control rod (+0.5).

. With the negative feedback via the reactivity coefficients, the total thermal power being generated in the nuclear-reactor core will not appreciably change (+0.2).

. Hence, the neutron flux will be greater than before in other parts of the core (+0.3).

. This means that the peak value for kW/ft will be greater than it was originally--occurring where the thermal neutron flux is the greatest (+0.5).

5.7 b. . After the control rod is in the core for one hour, the Xe^{135} concentration will build up in the vicinity of the dropped rod and will decrease in the regions of high thermal neutron flux (+0.5).

. This will further decrease the neutron flux in the vicinity of the dropped control rod and further increase the neutron flux where it had previously peaked (+0.5).

. The result is to further increase the maximum linear power density (+0.5).

Reference(s)

1. Nuclear Energy Training, Module 3, Reactor Operation, Sections 4 and 10, NUS Training Corporation.

2. C-E PWR Simulator Training Facility, "Reactor Theory," Florida Power and Light Company, SL 1&2.

ep 500(82 J8)/ds-30 Figure 66, ds-41 Figure 85

Points Available

5.8 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?

a. The power plant is operating at 100% of full power with the pressurizer pressure at 2250 psig. (1.0)

b. The power plant is in a cooldown mode and the RCS temperature is 480°F and the pressure is 900 psig. (1.0)

Use attached steam tables, if necessary.

Answer(s)

5.8 a. 228°F (+1.0)

5.8 b. 310°F (+1.0)

Reference(s)

1. "Academic Program for Nuclear Power Plant Personnel," Volume III, General Physics Corporation.

2. "Power Plant Thermodynamics," ~~N~~SL 1&2.

pp 36, 37, 49, 5ϕ, 52

Points Available

5.9 For a nuclear reactor at a certain operating condition, $T_C = 545°F$ and $T_H = 595°F$. The reactor coolant flowrate is 7×10^7 lbm/hr and the specific heat of water under these conditions is 1.3 Btu/lbm-°F.

What is the rate of thermal energy (heat) addition to the reactor coolant? Show your calculations. (2.0)

Answer(s)

5.9 $\dot{Q} = \dot{m}\, c_p\, \Delta T$ (+1.0)

$= (7 \times 10^7)(1.3)(595-545)$

$= 4.55 \times 10^9$ Btu/hr (+1.0)

Reference(s)

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

2. "Power Plant Thermodynamics," SL 1&2.
pp 60, 74, 93

Points Available

5.10 A cylindrical tank is 10 ft. in diameter and 20 ft. high. It is one-half filled with water. Air is the cover gas which is at atmospheric pressure (14.7 psia). The tank is drained with the vent closed until the volume of liquid is one-half of what it was before. (The tank is 1/4 filled now.) What is the pressure in the tank? (2.0)

Answer(s)

5.10

$V_T = (3.14)(5\ \text{ft})^2(20\ \text{ft})$	(+0.25)
$= 1570\ \text{ft}^3$	(+0.25)
$V_1 = (1570)(.5)$	(+0.25)
$= 785\ \text{ft}^3$	(+0.25)
$V_2 = (1570)(.75)$	(+0.25)
$= 1178\ \text{ft}^3$	(+0.25)
$P_1V_1 = P_2V_2$	(+0.25)
$(14.7)(785) = P_2(1178)$	
$P_2 = 9.80$ psia	(+0.25)

Reference(s)

1. "Academic Program for Nuclear Power Plant Personnel," Volume II, General Physics Corporation. Chapter 2, Section A
2. CE PWR Simulator Training Facility, "Power Plant Thermodynamics," pp. 16, 17, Florida Power & Light Company, SL 1&2.

Points Available

5.11 What is the difference between the MSIS actuation signals at Unit 1 and 2? *Include setpoints.* (1.5)

Answer(s)

5.11 For Unit 1 - MSIS is actuated at 600 psia

* For Unit 2 - MSIS is actuated at ~~500~~ 600 psia or 5 psig containment pressure

Reference(s)

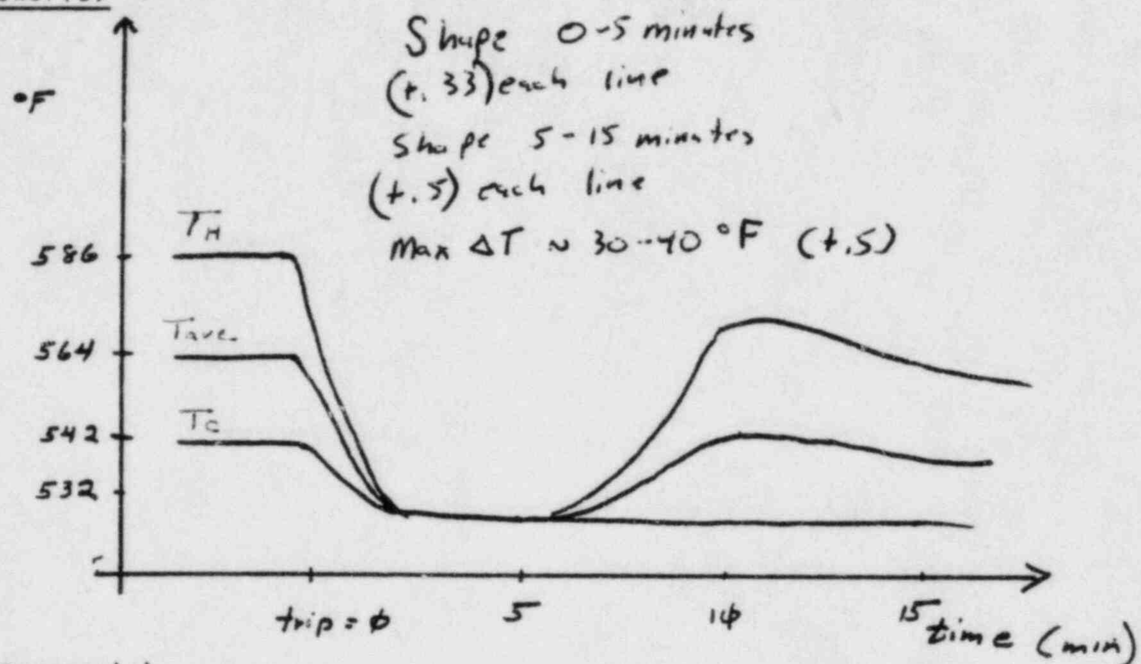
1. St. Lucie Exam #3, Question 1, SL 1&2.

* 2. *Tech-Specs, SL 2.*

Points Available

5.12 Unit 2 is operating at a steady 100% of full power. The reactor trips. Then, 5 minutes after the trip, the RCPs are stopped. Sketch a graph of T_H, T_C and T_{ave} as a function of time, covering a time period that starts before the trip to 15 minutes after the trip. (3.0)

Answer(s)



Reference(s)

1. NRC Exam Bank, Section 7, Question 6, SL 1&2.

Points Available

5.13 Describe the thermal condition of the secondary fluid at the following plant locations. For each location choose one of the following conditions: sub-cooled liquid, saturated liquid, wet steam, saturated vapor, superheated vapor. (1.5)
(5-95% quality)

* (a.) steam generator outlet

(b.) HP turbine exhaust

(c.) MSR outlet, inlet to LP turbine

(d.) condensate pump suction

(e.) feedwater inlet to the steam generator.

Answer(s)

(a.) saturated vapor

(b.) wet steam

(c.) superheated vapor

(d.) subcooled liquid

(e.) subcooled liquid

Reference(s)

1. NRC Exam Bank, Section 7, Question 2, SL 1&2.

- End of Section 5.0 -

Points Available

6.0 PLANT SYSTEM DESIGN, CONTROL AND INSTRUMENTATION (25)

6.1 The St. Lucie Unit I power plant is at 25% of full rated power and has been in this condition steadily for 4 days. The CEA position is 90 inches withdrawn on Group 5. Control mode is Manual Sequential. Then, load is increased on the turbine.

a. In Manual Sequential, the operator is going to adjust the rod positions in order to keep $T_{ave} - T_{ref}$ equal to zero. Explain the sequence of physical phenomena that will occur after the load on the turbine is given a step increase. (2.0)

b. While increasing the electrical power generated by the generator, a "GROUP DEVIATION" alarm is received. What would you anticipate to be the cause of this alarm? (0.5)

c. And what action(s) should you take? (1.5)

Answer(s)

6.1 a. . ~~HP governing valve will open further~~

. the steam flowrate will increase

1 answer { . the first-stage turbine pressure will increase (again)

. T_{ref} is calculated by the RRS to be higher

. Tave will drop

. the control rods are raised to make Tave - Tref

. ~~the neutron-flux level will increase~~

. ~~T_{ave} will increase~~

. ~~T_{ave} will level off due to the negative temperature~~ coefficient

(+0.~~4~~ each, +2.~~5~~0 max)

6.1 b. The cause of the GROUP DEVIATION alarm is usually the malfunction of an operating coil for one CEA (+0.5).

6.1 c. The operator should transfer from the MS to the MI mode (0.5). In the MI mode the operator should return the out-of-alignment rod to the group position (+0.5).

Go to

* Alternate respone is that the deviation is greater than the deviation setpoint (of 4.0 inches)

G.1 c

Go to off (+.5)

Match Tred - Tave (+.5)

~~[illegible]~~ to realign rods (+.5)

attempt

I&C maintenance should be notified, the occurrence logged as required and the control mode returned to either MS or AS (+0.5).

(+1.5 max)

Reference(s)

1. Lesson Plans and System Descriptions #29, "CEDMCS and Analog Display System," SL 1&2.

2. Nuclear Energy Training, Module 3, Reactor Operation, NUS Training Corporation.

Points Available

6.2 Which one of the statements below most accurately describes the design and operation of the quench tank, pressurizer and its control system? (1.0)

(a.) The steam in the pressurizer is maintained in a superheated condition so that the volume of the vapor will not shrink to zero.

(b.) The control system will prevent uncovering of the heaters following a 10% step decrease in the thermal power generated in the nuclear reactor.

(c.) The spray nozzle at the top of the pressurizer is connected through two air-operated spray control valves to the loop-1 hot leg.

(d.) The quench tank is sized to receive and condense steam from the discharges of the pressurizer safety valves; a total loss-of-load event would be handled by the opening of the rupture disc of the quench tank.

Answer(s)

6.2 The answer is (b.). (+1.0)

Reference(s)

1. Lesson Plans and System Descriptions #1, "Reactor Coolant System - RCPs - Pressurizer - Quench Tank," SL 1&2.
Introductory Summary Sheet of Lesson Plan, Purpose and Design Features of Pressurizer, pp 11, 12

Points Available

6.3 Sketch a diagram of the letdown and charging systems for Unit 2 in order to show the pressure relief protection.

a. Sketch on the included figure the letdown and charging system and indicate the locations to which the discharges of the relief valves are sent. (2.0)

b. For each relief valve indicate their setpoints by labeling the figure one with the following choices. (1.0)

(1) 200 psig

(2) RC pressure + 10%

(3) 75 psig

(4) 650 psig

(5) 75 - 100 psig .166 [illegible]

~~(6) 1500 psig~~ ~~15~~ [illegible]

(7) 14.7 psia 1 max

Answer(s)

See the attached figure. (+3.0)

Reference(s)

1. Lesson Plans and System Descriptions #3, "CVCS and Boron Concentration Control System," SL 1&2.

Figure "St. Lucie Plant - Unit 2 Chemical Volume and Control System at the end of the Lesson Plan

* The attached figure is correct, but the relief valves at the suctions of the charging pumps are designed for thermal-expansion relief. Thus, instead of "(1) 200 psig" for the answer, an answer referring to the thermal-expansion or design pressure of the charging pumps suction piping is acceptable. See the included documentation.

LETDOWN & CHARGING PRESSURE RELIEF DIAGRAM

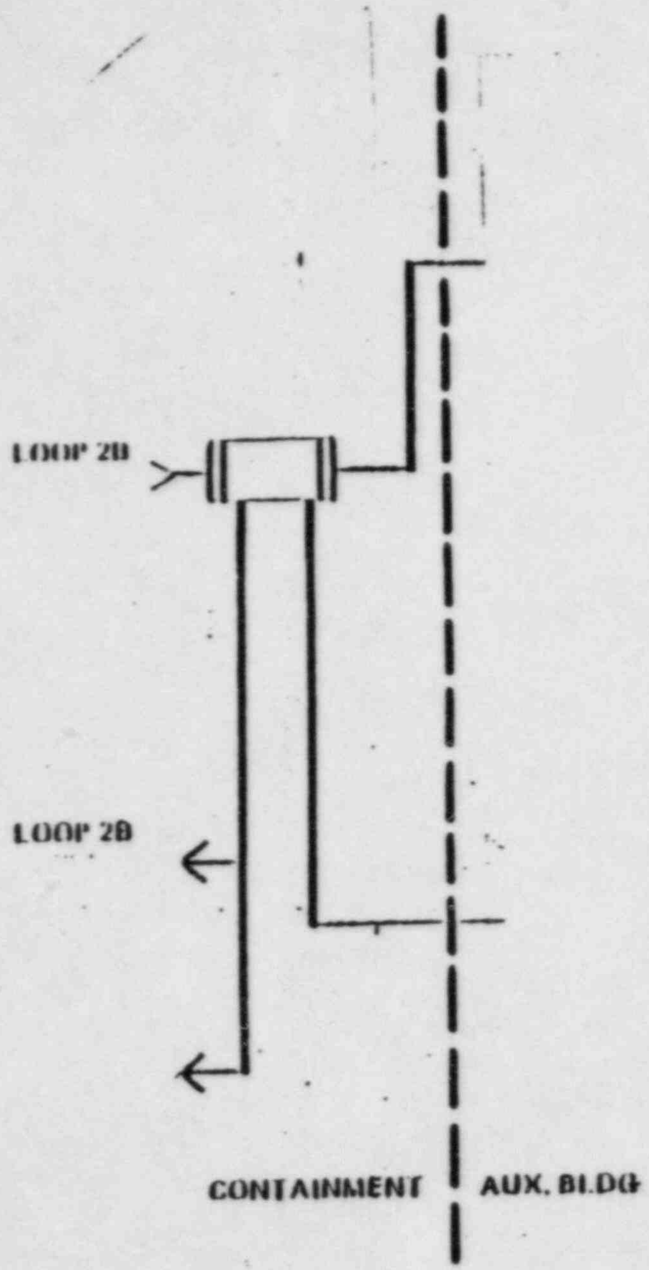


FIGURE 6.3

LETDOWN & CHARGING PRESSURE RELIEF DIAGRAM (+0.4)

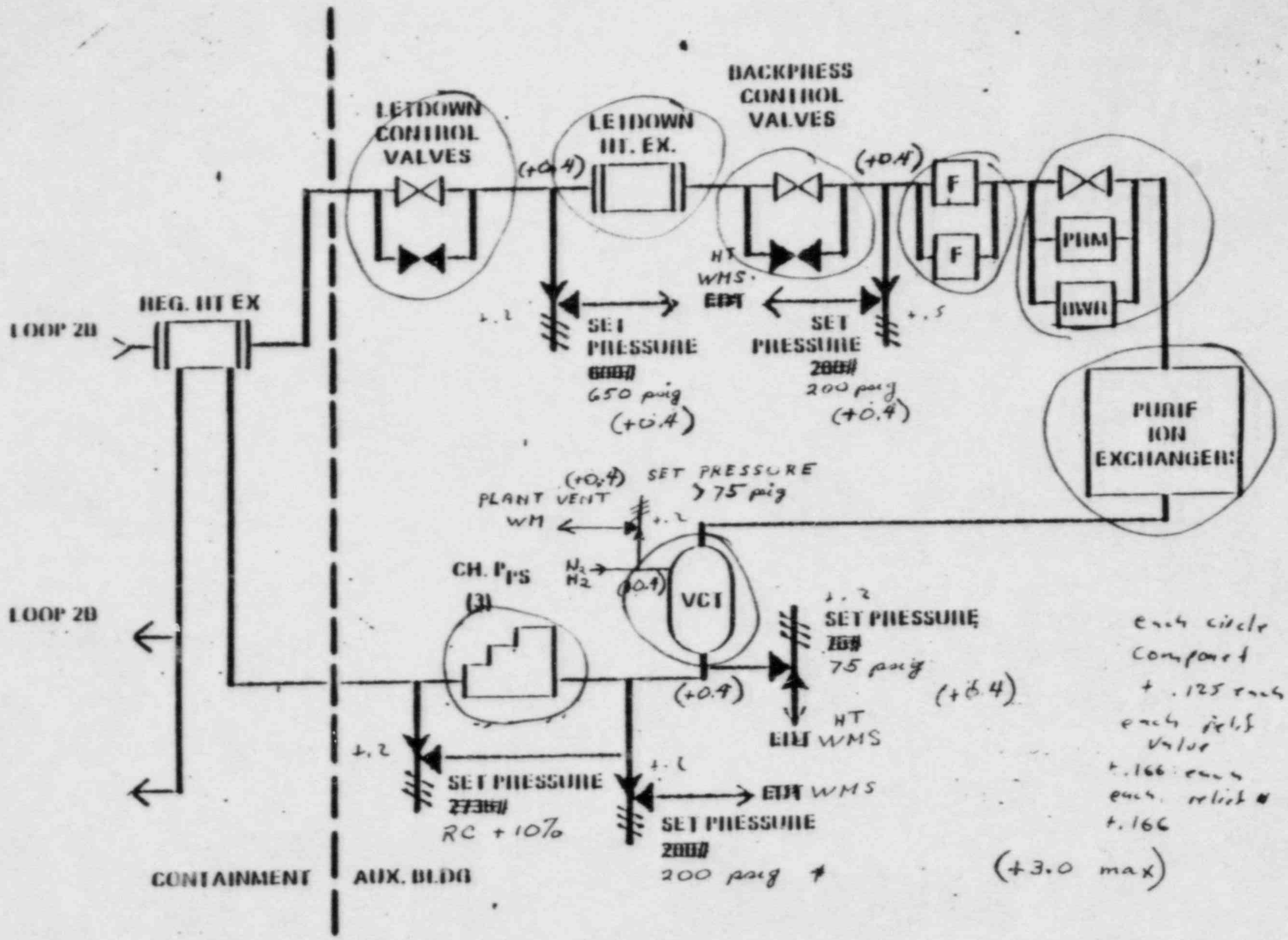


FIGURE 6.3 answer

The relief valve set pressure is equal to the design pressure (600 psig) of the intermediate pressure letdown piping and letdown heat exchanger.

2) <u>Low-pressure letdown relief valve, V-2354</u>

The relief valve downstream of the letdown backpressure control valves protects the low pressure piping, purification filters, ion exchangers and letdown strainer from overpressure. The valve capacity is equal to capacity of intermediate pressure letdown relief valve V-2345. The set pressure is equal to the design pressure (200 psig) of the low pressure piping and components.

3) <u>Charging pump discharge relief valves, V-2324, V-2325, V-2326</u>

The relief valves on the discharge side of the charging pumps are sized to pass the maximum rated flow of the associated pump with maximum backpressure without exceeding the maximum rated total head for the pump assembly. The valves are set to open when the discharge pressure exceeds the reactor coolant system design pressure (2485) by 10 percent.

4) <u>Charging pump suction relief valves, V-2315, V-2318, V-2321</u>

The relief valves on the suction side of the charging pumps are sized to pass the maximum fluid thermal expansion rate that would occur if the pump were operated with the suction and discharge isolation valves closed. The set pressure is less than design pressure of charging pump suction piping.

This is equivalent [illegible]

5) <u>Charging line thermal relief valve, V-2435</u>

The relief valve on the charging line downstream of the regenerative heat exchanger is sized to relieve the maximum fluid thermal expansion rate that would occur if hot letdown flow continued after charging flow was stopped by closing the charging line distribution valves. The valve is a spring-loaded check valve.

6) <u>Volume control tank relief valve, V-2115</u>

The relief valve on the volume control tank is sized to pass a liquid flow rate equal to the sum of the following flow rates: the maximum operating flow rate from the reactor coolant pump controlled bleedoff line; the maximum letdown flow rate possible without actuating the high flow alarm on the letdown flow indicator; the design purge flow rate of the sampling system; and the maximum flow rate that the boric acid makeup system can produce with relief pressure in the volume control tank. The set pressure is equal to the design pressure of the volume control tank.

Question 6.3

Points Available

6.4 Draw a one-line diagram of the 125 VDC Emergency and 120 VAC Vital and Instrument System, including the batteries, chargers, inverters, switches, breakers and backups. Use the attached figure, Figure 2.4. It is not necessary to draw the connections to and from the MCC 1B5, the 125 V DC BUS 1B and the 120 V AC Instrument BUS 1MD. (3.0)

Answer(s)

6.4 See diagram. (+3.0 max) Updated diagrams are included and are used for the grading.

Reference(s)

1. System Descriptions #6, "Class 1E Electrical System," SONGS 2&3.

2. Lesson Plans and System Descriptions #21, "120 Volt Instrument AC; 120 Volt Vital AC" SL 1&2. Lesson Plan # 33, Figure 4

3. Lesson Plans and System Descriptions #22, "125 Volt DC System," SL 1&2. Figure 4

125 VOLT DC / 120 VOLT AC
ELECTRICAL SYSTEM

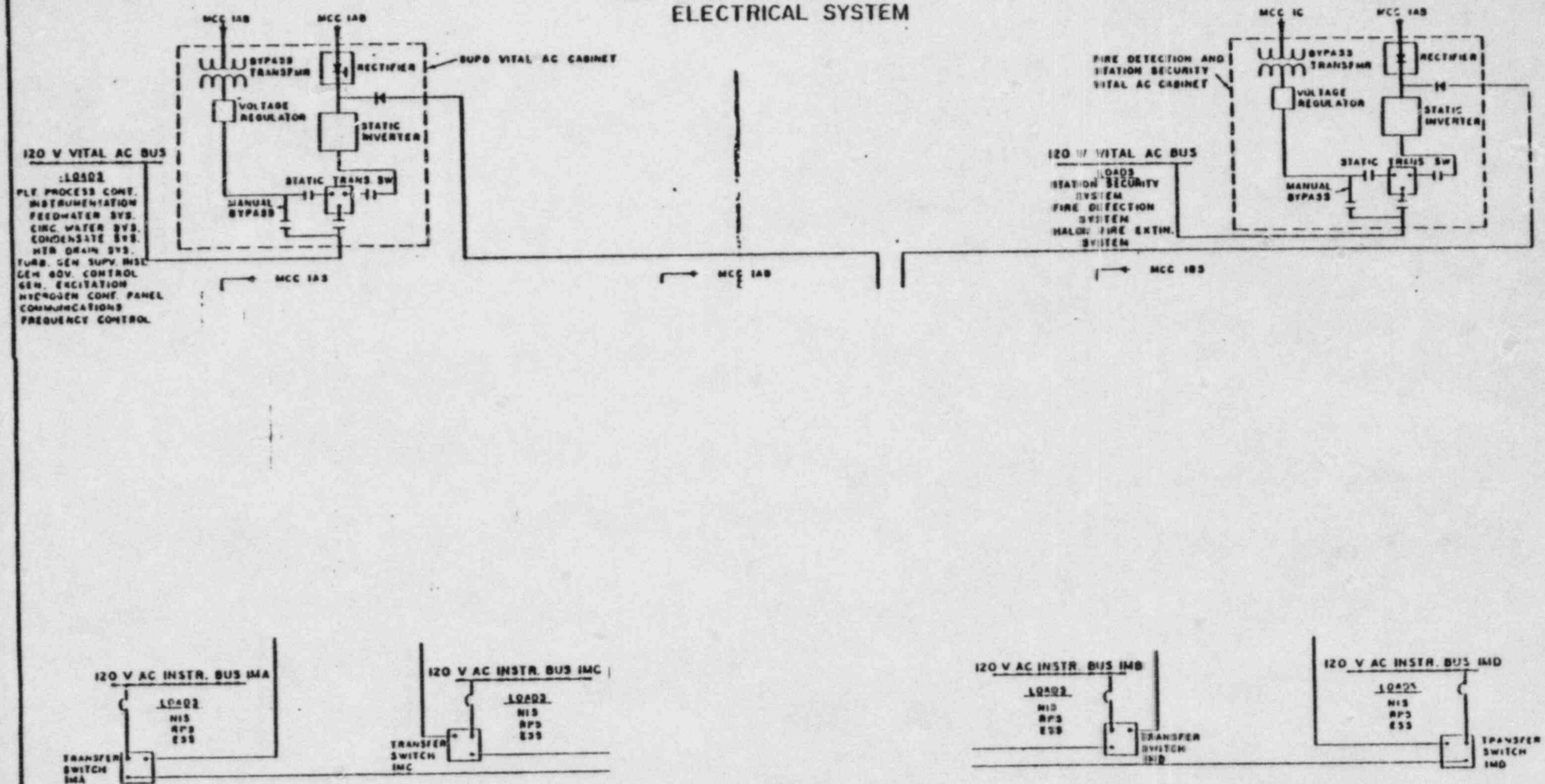


FIGURE 6A

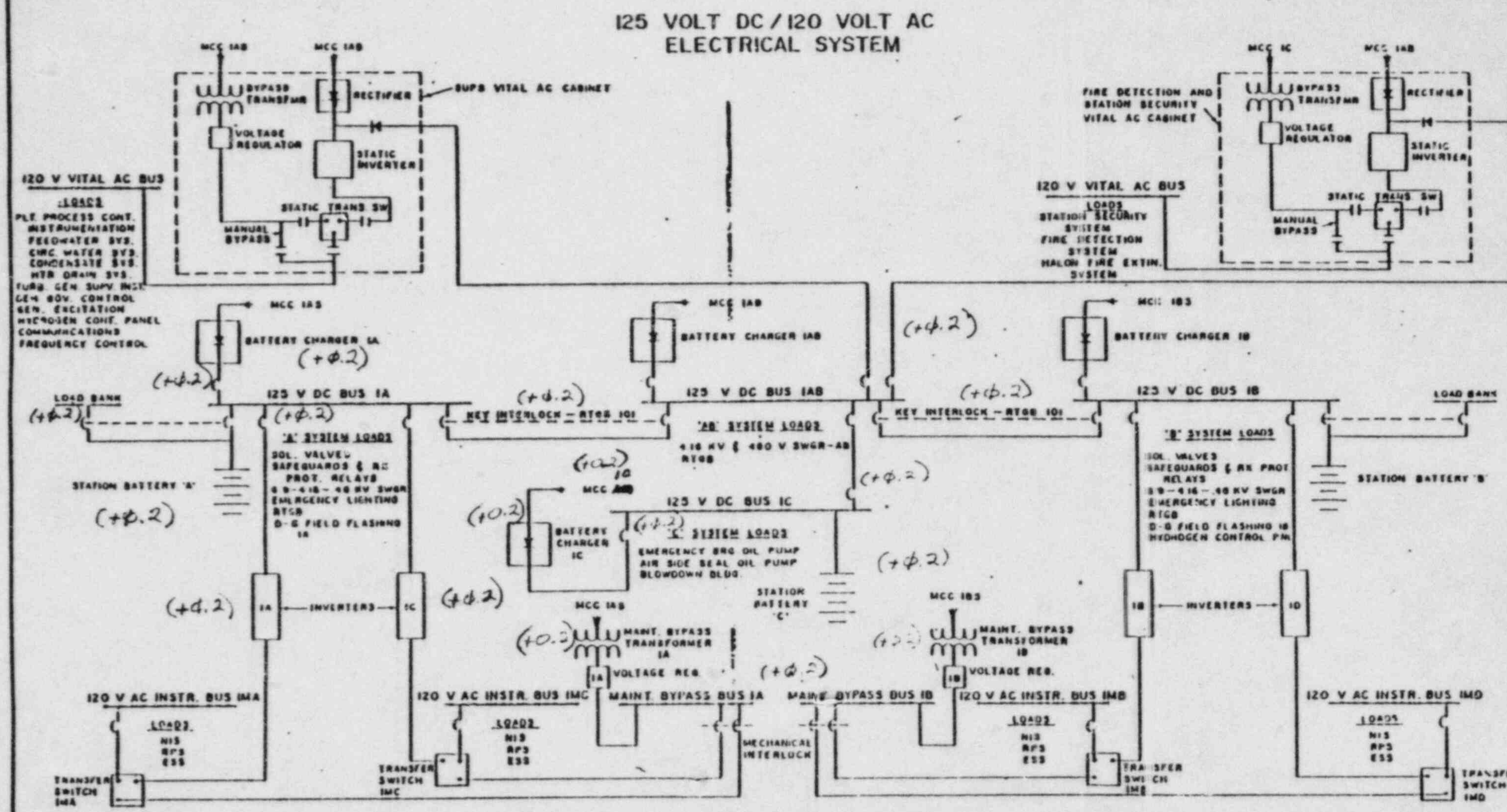


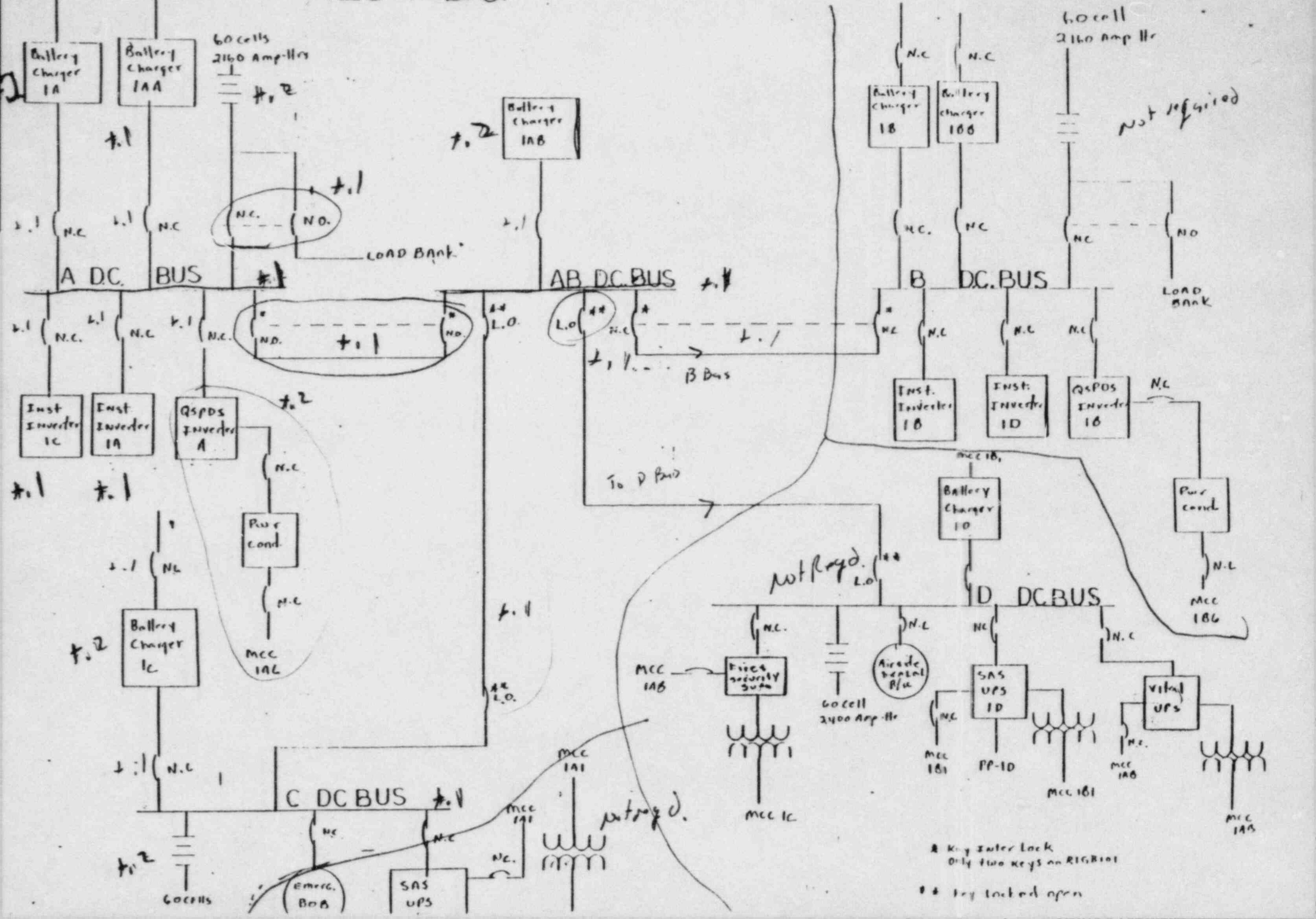
FIGURE 4 Answer

(+3.Φ max)

ST. LUCIE UNIT 1
125 V D.C.

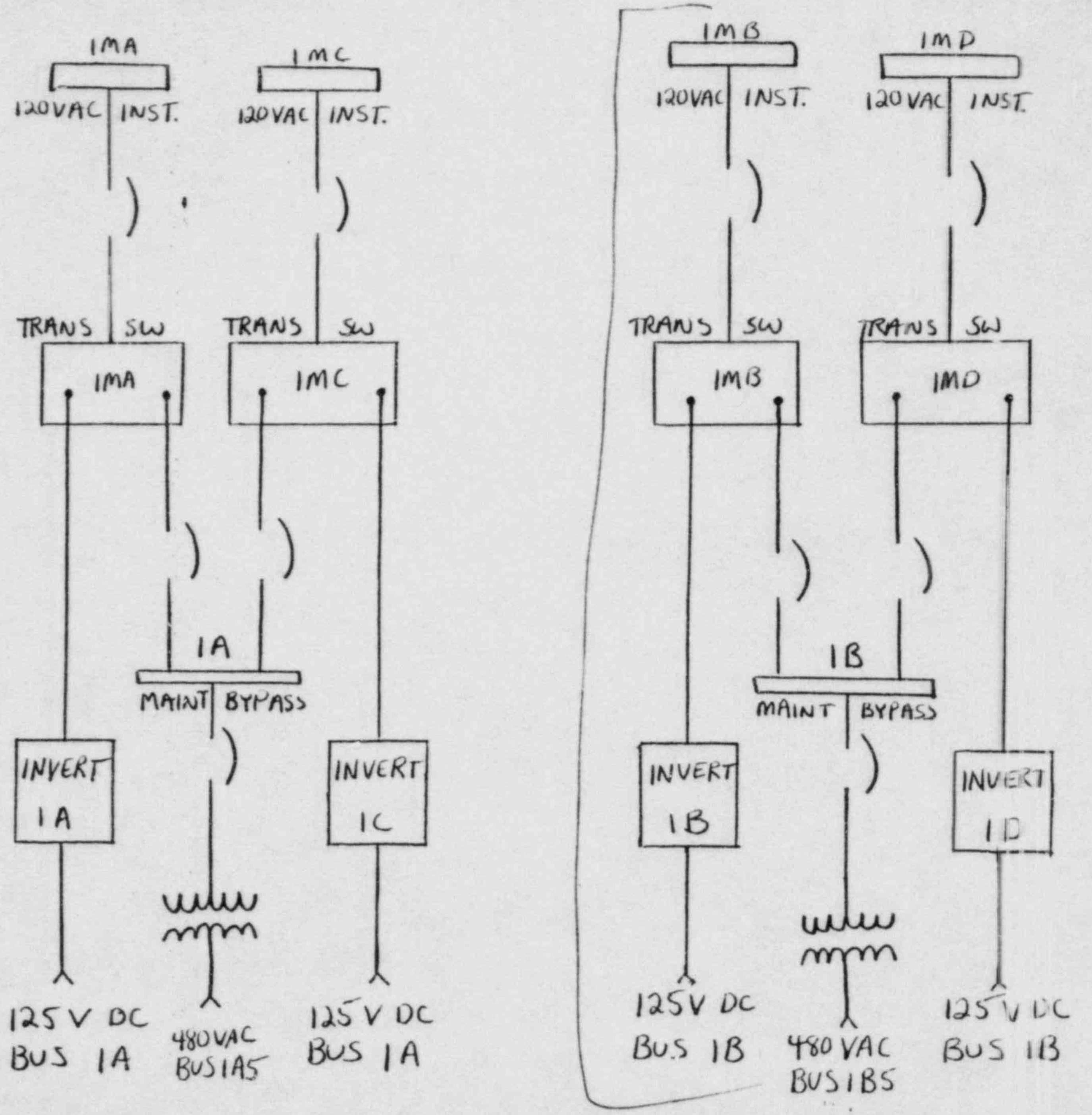
Quest. 6.4

(J)



Ques. 6.4

FIG. 4 UNIT #1 120 VAC INSTRAMENT DISTRIBUTION



Points Available

6.5 Consider the venting system for the reactor vessel and the pressurizer.

a. What is the basic purpose of the venting system and under what conditions (normal and emergency) is it anticipated that the system would be used? (1.5)

b. Draw a one-line diagram showing this venting system. It is important to indicate to where the gases were vented. (2.0)

Answer(s)

6.5 a. The basic purpose of the venting system is to permit the operator to vent the pressurizer steam space or the reactor vessel from the control room (~~+0.5~~) +0.75. It is used to vent gases during filling and draining operations or after an accident (~~+0.5~~) +0.75.

* 6.5 b. See the attached diagram. (+2.0 max)

Reference(s)

1. System Description #39, "Pressurizer and Pressurizer Control System," SONGS 2&3.

2. Reactor Coolant System P&ID Diagram, Figure 5.1-3, SL 1.

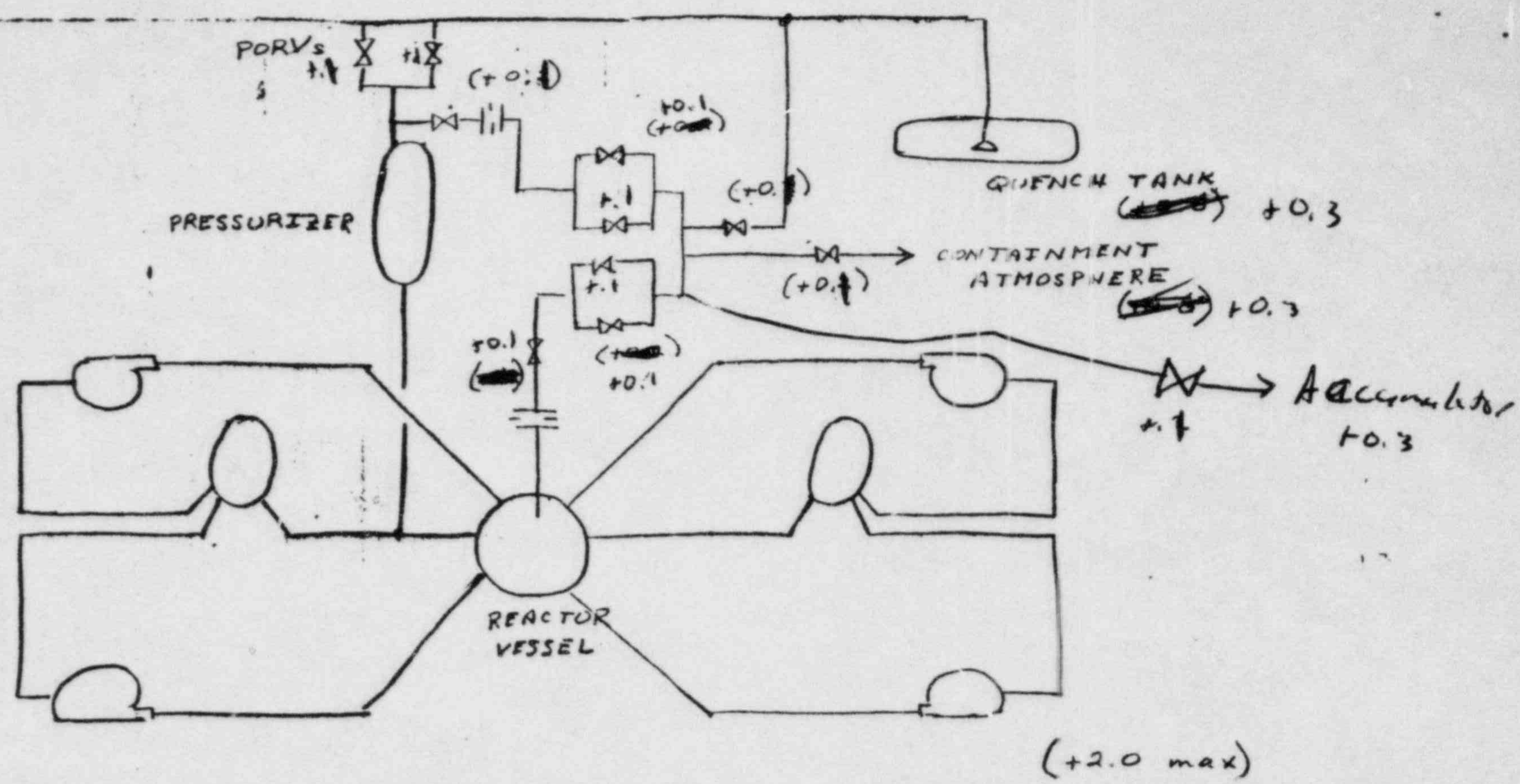


Figure 2.7 Answer

This was used to Grade

Quest. 2.7

REACTOR COOLANT GAS VENT SYSTEM
UNIT 1 (UNIT 2 SIMILAR)

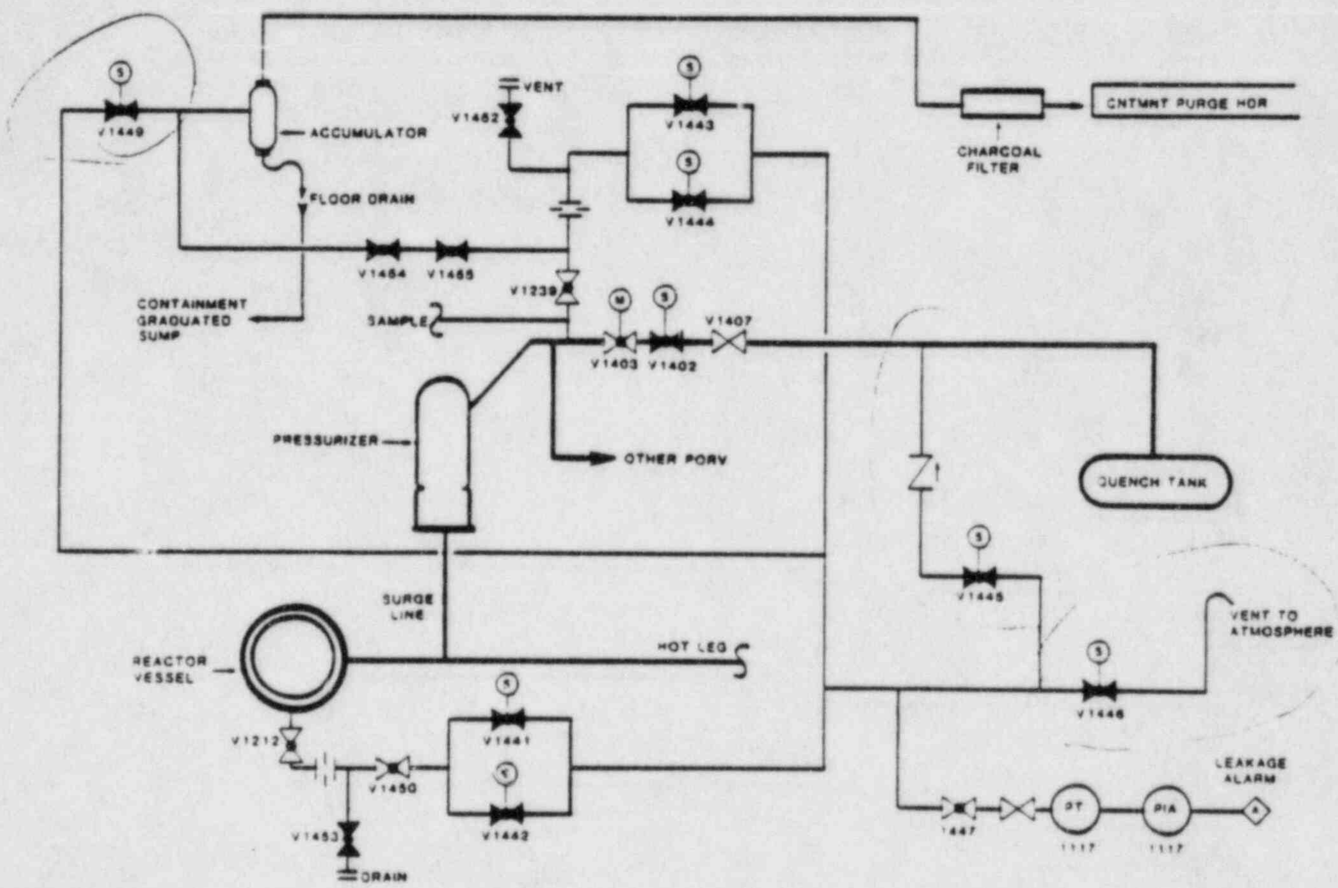


FIGURE 14

Points Available

6.6 Consider the excore neutron detectors and their instrumentation for Unit 1.

* a. What are ~~the~~ two indications of a failed section of a dual-section UIC safety channel? (1.0)

b. Describe three control functions of the wide-range channels. (1.0)

* c. What ~~is the~~ are two function(s) (inputs to other systems) of the control channels. (1.0)

d. What is Q power? (1.0)

Answer(s)

6.6 a. Subchannel deviation alarm, hi-power channel trip, etc. TM/LP Alarm, DNBR Alarm

6.6 b. SUR trip to the RPS, ~~[illegible]~~, SUR enable 10^{-4} % Zero Mode bypass .1% power

* 6.6 c. Input to RRS for ~~RX no~~ reactor vs. turbine power, input to ~~ASI~~ CPCs for calculation of ASI, ~~power ratio,~~ input to core-barrel movement monitor

* 6.6 d. Max of ϕ or B(ΔT) power

Reference(s)

1. NRC Question Bank, Section 4, Number 18, SL 1&2.

Points Available

6.7 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 1% of full power, CVCS controllers are in automatic and charging pump #2 and #3 are the first and second backup pumps respectively.

Condition #1 - pressure 2270 psig and decreasing
level - 4.0% and increasing

Condition #2 - pressure 2290 psig and increasing
level + 4.5% and increasing

Condition	Spray Valve(s)	Prop Htrs	Backup Htrs	Letdown Valve(s)	CP #1	CP #2	CP #3	
#1	______	____	____	______	____	____	____	(1.5)
#2	______	____	____	______	____	____	____	(1.5)

Answer(s)

6.7

Condition	Spray Valve(s)	Prop Htrs	Backup Htrs	Letdown Valve(s)	CP #1	CP #2	CP #3
#1	Closed (0.3)	On (0.3)	Off (0.1)	Min (0.1)	On (0.1)	On (0.3)	On (0.3)
#2	~~Closed~~ Open (0.3)	On (0.3)	On (0.3)	Open (0.3)	On (0.1)	Off (0.1)	Off (0.1)

Reference(s)

1. System Description #39, "Pressurizer and Pressurizer Control System," SONGS 2&3.
2. Lesson Plans and System Descriptions #27, "Pressurizer Level and Pressure Control," Lesson Plan #91, pp. 5, Figure 4, SL 1&2.
3. Off-Normal Operating Procedure 2-0120035, Revision 5, "Pressurizer Pressure and Level Off-Normal Operation", pp. 2, SL 2. (attached)

Ques. 6.7, 3.8

2

ST. LUCIE UNIT NO. 2
OFF-NORMAL OPERATING PROCEDURE NUMBER 2-0120035, REVISION 5
PRESSURIZER PRESSURE AND LEVEL-OFF-NORMAL OPERATION

5.0 INSTRUCTIONS:

5.1 Immediate Automatic Actions:

1. Abnormal Pressurizer Pressure Condition.

   A. Pressurizer safety valves open at 2500 psia.

   B. High pressure reactor trip and power operated relief valves open at 2370 psia.

   C. High pressure alarm actuates at 2340 psia and a back-up signal will de-energize all pressurizer heaters.

   D. Proportional heaters cycle from minimum output at 25 psi above setpoint to maximum output at 25 psi below setpoint.

   E. **Spray valves cycle from full closed at 25 psi above setpoint to full open 75 psi above setpoint.** Set point = 2250

   F. Back-up heaters energize at <2200 psia and de-energize at >2220 psia.

   G. Low pressure alarm actuates at 2100 psia.

   H. TM/LP reactor trip initiates at 1887 psia minimum pressure.

   I. SIAS initiates at 1736 psia. /R5

2. Abnormal Pressurizer Level Condition.

   A. All Pressurizer heaters de-energize at 27% indicated level, and respective Pressurizer Heater Transformer feeder breaker opens.

   B. Low level alarm actuates and a backup signal to start the back-up Charging Pump is received at 5% below RRS setpoint.

NOTE
Only one back-up Charging Pump is in the level control system.

   C. The back-up Charging Pump receives a signal to start at 3% below RRS setpoint, decreasing.

   D. The back-up Charging Pump receives a signal to stop at 1% below RRS setpoint, increasing, and letdown flow decreases to minimum (29 gpm).

   E. All back-up heaters energize and a back-up stop signal to the back-up Charging Pump is received at 4% above RRS setpoint.

   F. Maximum letdown is 128 gpm at 9% above RRS setpoint.

   G. High level alarm actuates at 10% above RRS setpoint.

Points Available

6.8 Concerning the safety injection tanks of Unit 2.

a. How many SITs must empty into the nuclear reactor vessel following a LOCA to be consistent with the Safety Analysis assumptions? (0.5)

b. How are the tank components operated and designed to assure compliance with the single-failure criteria? (1.0)

c. Discuss the SIT discharge-valve interlocks. (1.0)

Answer(s)

6.8 a. 3

6.8 b. Operation - discharge valves open and r~~o~~acked out at elevated pressures

Design - parallel vent valves to ensure venting capability

6.8 c.

<275 psia	close permissive
>500 psia	auto open
SIAS	open signal

Reference(s)

1. NRC Exam Bank, Section 8, Question 6, SL 1&2.

Points Available

6.9 The plant is at 100% of full power. All controllers of the Steam Bypass Control System are in manual. Is the quick-open signal disabled? Yes or No. (1.0)

Answer(s)

6.9 Yes

Reference(s)

1. NRC Exam Bank, Section 8, Question 3, SL 1&2.

- End of Section 6.0 -

Points Available

7.0 **PROCEDURES: NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL** (25)

7.1 While handling fuel in unit 2 in the fuel handling building, a spent fuel assembly is dropped and the cladding integrity is breached.

a. What automatic actions take place? (1.5)

b. What are the immediate operator actions for this event? (1.5)

Answer(s)

7.1 a. 5.1.4

7.1 b. 5.2.3, 5.2.4

OP 1 or 02 2 accepted

high rad signal ...

Reference(s)

1. OP 2-1600030, Rev. 1, Attached

2

5.0 INSTRUCTIONS:

5.1 Immediate Automatic Actions:

5.1.1 For damage to new fuel - none.

5.1.2 If damage to spent fuel occurred inside Containment, then a Containment Isolation Actuation Signal (CIAS) may occur.

5.1.3 Containment evacuation alarm may actuate.

5.1.4 If damage to spent fuel occurred in the Fuel Handling Building (FHB), then a high radiation signal may alarm in the Control Room which will isolate the FHB and activate the Shield Building Ventilation System (SBVS). /R1

5.2 Immediate Operator Action:

5.2.1 Inform Control Room personnel of the accident.

5.2.2 If the accident occurred inside the Containment:

1. Sound the Containment Evacuation Alarm.
2. Stop 2-HVE-8A and 2-HVE-8B (Containment Purge Fans) if running. /R1
3. Verify I-FCV-25-1 through I-FCV-25-6 (Containment Purge valves) have closed. /R1
4. Evacuate the Containment and check for personnel contamination.
5. Notify Health Physics Department to perform a complete radiological evaluation.
6. Monitor applicable Process and Area Radiation Monitor channels for an increase in radiation levels inside Containment.

5.2.3 If the accident occurred in the Fuel Pool Area:

1. Evacuate the Fuel Pool Area and remain on the landing outside the north door until monitored for contamination.
2. Notify Control Room personnel.
3. Monitor applicable Area Radiation and Process Monitor channels for an increase in radiation levels inside the Fuel Pool Area.

5.2.4 Notify the Duty Call Supervisor and the Health Physics Supervisor.

Points Available

7.2 While operating at 100% equilibrium power, a condition develops such that the control room becomes uninhabitable and must be evacuated.

a. **What** immediate operator actions are required? (2.0)

b. If control room accessibility is not possible, the plant must be placed in cold shutdown and borated to 1900 ppm.

Why is boration required? (1.0)

Answer(s)

7.2 a. 4.0 Attached

7.2 b. >5% shutdown margin at 201°F any time in ~~xx~~ core life with most reactive CEA stuck out.

Reference

EOP 2-0030141, Rev. 9, 5.9 Attached.

what unit ?
EOP 1 or 2 used

2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE UNIT 2
EMERGENCY PROCEDURE 2-0030141, REVISION 9
CONTROL ROOM INACCESSIBILITY

1.0 SCOPE:

This procedure provides instructions for placing the plant in a safe condition when operations cannot safely be conducted from the Control Room.

2.0 SYMPTOMS:

Conditions exist such that the Control Room becomes uninhabitable and must be evacuated.

3.0 AUTOMATIC ACTIONS:

None

4.0 IMMEDIATE OPERATOR ACTION:

ACTION	NOTES
4.1 Manually trip the reactor and Turbine prior to leaving the Control Room, if possible.	4.1 Push buttons on RTGB-201 and 204.
4.2 Announce evacuation of the Control Room over the P.A. system.	
4.3 Implement the Emergency Plan, as necessary, in accordance with EPIP 3100021E, "Duties and Responsibilities of the Emergency Coordinator".	
4.4 Obtain the Remote Shutdown Room Keybox Master Key from the Control Room Key Locker.	4.4 Key Number 2. /R9
4.5 Evacuate all personnel from the Control Room.	

2

5.0 SUBSEQUENT ACTIONS: (continued)

CHECK

5.8 Periodically check the habitability of the Control Room and when conditions permit, reoccupy the Control Room. Return isolation switches to NORMAL for switches and controls that are operational and maintain the Unit at Hot Standby until a complete evaluation has been made. ____

5.9 If Control Room accessibility is not possible, place the unit in a Cold Shutdown condition as follows: ____

1. Commence boration to Cold Shutdown conditions by manual valve lineup. Borate to 1900 ppm as shown below. Carry out boron sampling as required during cooldown. ____

b

NOTE

This concentration will ensure >5% Shutdown Margin at 201°F at any time in core life, assuming the most reactive CEA stuck full out. Use both the local boron flow meters vs. time and BAM Tank level change to determine how many gallons of boron have been added. Do not interpolate values shown, always round critical boron concentracion DOWN to next lower value on table.

NOTE

If plant curves are available, they may be used to determine shutdown boron concentration requirements instead of this Table.

Boron Concentration Prior to Control Room Inaccessibility	No. of Gallons of Boron needed to reach 1900 ppm	BAM Tank level change (1 BAM Tank)
50 PPM	7223 Gallons	75%
100 PPM	7039 Gallons	73%
200 PPM	6668 Gallons	70%
300 PPM	6295 Gallons	66%
400 PPM	5919 Gallons	62%
500 PPM	5442 Gallons	57%
600 PPM	5162 Gallons	54%
700 PPM	4779 Gallons	50%
800 PPM	4395 Gallons	46%
900 PPM	4008 Gallons	42%
1000 PPM	3618 Gallons	38%
1100 PPM	3226 Gallons	34%
1200 PPM	2832 Gallons	30%
1300 PPM	2435 Gallons	25%
1400 PPM	2036 Gallons	21%
1500 PPM	1634 Gallons	17%

Points Available

7.3 During operation at 100% power, a loss of offsite power occurs generating a reactor trip signal but not accompanied by CEA insertion as indicated by the APS, core mimic, digital position readout, backup readout and core power.

a. List the immediate operator actions for the event. (2.0)

b. What parameter should have tripped the reactor? (0.5)

Answer(s)

Reference(s)

a. EOP 2-0030132, Rev. 4, pp. 4, Attached

b. 6.1 EOP 2-0030140, Rev. 6, low RCS Flowrate

ok high pressurizer pressure

they were given instructions to state assumptions

8.5

ST. LUCIE UNIT NO. 2
EMERGENCY PROCEDURE NUMBER 2-0030132, REVISION 4
ANTICIPATED TRANSIENT WITHOUT SCRAM

2

3.0 AUTOMATIC ACTIONS: (Cont.)

AUTOMATIC ACTION	INITIATING EVENT
3.7 SIAS and CIAS.	3.7 RCS pressure 1736 psia Containment pressure 5 psig
3.8 CSAS.	3.8 Containment pressure 9.3 psig

4.0 IMMEDIATE OPERATOR ACTIONS

	LOCATION
4.1 Verify required Auto Actions have occurred or manually initiate.	
4.2 Trip Turbine	4.2 RTGB-201
4.3 Ensure AFW flow.	4.3 RTGB-202
4.4 Trip reactor.	4.4 RTGB-201 or 204
4.5 Emergency borate.	4.5 RTGB-205

AND IF CEA'S DON'T DROP

4.6 Open reactor trip breakers locally.	4.6 Cable spreading room
4.7 Stop both M-G sets at the M-G sets or by opening breakers 2-40212 and 2-40511.	4.7 RAB 19.5' Elev. (M-G sets) RAB 43' Elev. L.C. 2A2/2B2 (Breakers to M-G sets)

AND IF CEA'S DON'T DROP

4.8 Re-energize CEA bus with either M-G set and attempt to insert CEA's.

4.9 Return to 2-0030130, "Reactor Trip/Turbine Trip", Immediate Operator Actions to determine type of transient and further action required.

Points Available

7.4 A main steam line break has occurred inside containment at S.L. #2 at 100% power. List the six (6) automatic actions caused by RPS and EFSAS signals and their setpoints. (2.0)

Answer

+.33 2.0

7.4 3.1 - 3.9, pp. 6, (~~0.25~~ ea) (~~2.0~~ max)

Reference

EP 2-0810040, Rev. 2, Attached

2

3.0 IMMEDIATE AUTOMATIC ACTION:

AUTOMATIC ACTION	INITIATING EVENT
3.1 Reactor trip.	3.1 S/G pressure <626 psia
3.2 Turbine trip.	3.2 Reactor trip bus low voltage
3.3 Generator lock-out.	3.3 Turbine trip
3.4 Transfer from Auxiliary to Start-up transformers.	3.4 Generator lock-out
3.5 MSIS.	3.5 S/G pressure <600 psia or Containment pressure >5 psig.
3.6 SIAS.	3.6 RCS pressure <1736 psia or Containment pressure >5 psig.
3.7 CIAS.	3.7 Containment pressure >5 psig or Containment radiation $\geq$10 R/HR or from SIAS actuation.
3.8 CSAS.	3.8 Containment pressure >9.3 psig concurrent with SIAS.
3.9 AFAS (feeds only the non-faulted S/G).	3.9 S/G level <20.6%

/R2

Points Available

7.5 What are the three (3) conditions that require emergency boration and what are the indications as stated in OP 2-0250030, Emergency Boration? (1.5)

Answer

Reference

pp. 1&2, OP 2-0250030, Rev. 2 Attached

2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE UNIT 2
OFF-NORMAL OPERATING PROCEDURE NO. 2-0250030
REVISION 2

1.0 TITLE:

EMERGENCY BORATION

2.0 REVIEW AND APPROVAL:

Reviewed by Facility Review Group February 9 1982

Approved by C. M. Wethy Plant Manager February 15 1982

Revision 2 Reviewed by FRG 8-24 1983

Approved by C M Wethy 10-20- 1983

3.0 PURPOSE AND DISCUSSION:

This procedure provides instructions for the injection of concentrated boric acid solution into the Reactor Coolant System (RCS) via the Charging Pumps.

In the event that normal charging flow is unavailable, flow can be directed to the Auxiliary HPSI header from the discharge of the Charging Pumps.

The Boron Concentration Control System is lined up to automatically emergency borate the RCS on a Safety Injection Actuation Signal (SIAS). When shutdown margin has been confirmed or the SIAS signal reset, it is desirable to restore the Boron Concentration Control System to the automatic make-up mode, or the Refueling Water Tank (RWT) to the suction of the Charging Pumps to prevent overborating.

4.0 SYMPTOMS:

Any one of the following conditions requires emergency boration:

4.1 Unanticipated or uncontrolled RCS cooldown following a reactor trip as indicated by:

1. Reactor Low Tave-Tref alarm

2. Decreasing reactor coolant wide range temperature indication

3. Uncontrolled decrease of Pressurizer level or pressure

4. Uncontrolled decrease in steam pressure

2

4.0 SYMPTOMS: (Cont.)

4.2 Unexplained or uncontrolled reactivity increase as indicated by:

1. Abnormal Control Element Assembly insertion

2. Abnormal increase in reactor coolant temperature, Tave or reactor power

3. Abnormal increase in reactor power or count rate when shut down

4.3 Loss of Shutdown Margin due to excessive Control Element Assembly insertion as indicated by:

1. Power dependent insertion (data processor) alarm

2. Power dependent insertion (ADS) alarm

Points Available

7.6 The fill and drain method of cooling the RCs is an alternative to natural circulation in the event of loss of condensate capacity. Explain how this method is accomplished as stated in OP 2-0120040, Natural Circulation/Cooldown. (2.5)

Answer

7.6 ~~ANS~~ Appendix ~~B~~ D, Attached.

Reference

EOP 2-0120040, Rev. 4.

11/5

2

APPENDIX D

RCS FILL AND DRAIN METHOD OF COOLING

REACTOR VESSEL HEAD REGION

> NOTE
> This method of RCS cooldown should only be employed in the event that rapid de-pressurization of the RCS is required, or Condensate Storage Tank level decreases below minimum required by Tech Specs.

> CAUTION
> DURING THIS EVOLUTION, PRESSURIZER LEVEL IS NOT A VALID INDICATOR OF RCS INVENTORY DURING TRANSIENT CONDITIONS. CARE SHOULD BE EXERCISED TO OBSERVE OTHER PARAMETERS WHICH WOULD INDICATE ANY LOSS OF RCS INVENTORY.

1. Take manual control of the charging and letdown system.

2. Lower RCS pressure by using auxiliary sprays into the Pressurizer.

3. As voiding occurs in the upper reactor vessel head, a surge of water from the RCS will cause Pressurizer level to increase rapidly. Terminate auxiliary spray prior to Pressurizer level increasing to 70% indicated level.

4. Cool the upper reactor vessel head region by charging with a Charging Pump to the RCS loop(s). Continue charging until either of the following conditions occur:

   4.1 Pressurizer level decreases to 30% indicated level

   OR

   4.2 The upper reactor head is charged solid.

> NOTE
> A solid upper head condition will be evident by an increasing Pressurizer level as charging to the loops is continued.

5. Repeat steps 1 through 4 above until SDC entry conditions are established.

> NOTE
> If the above were to prove unsuccessful, Pressurizer heaters may be used (if sufficient volume is available) to heat up the pressurizer and remove a vessel head void. This strategy should be used only as a last resort and will take an hour or more to be successful.

Points Available

7.7 Define the following from HP-2 procedure:

a. Radioactive Area (1.0)

b. High Radiation Area (1.0)

c. Contaminated Area (1.0)

Answer

7.7 a. 5.1.5

7.7 b. 5.1.6

7.7 c. 5.1.9

Reference

HP-2, Rev. 0, Attached

Unescorted access to the Radiation Controlled Area is limited to those individuals who have completed the Radiation Protection Training Program (see section 3.3.3) and are authorized by the Plant Manager, or Operations Superintendent. Individuals not receiving radiation protection training may enter the Radiation Controlled Area when escorted by an authorized employee.

5.1.4 Hot Spot Areas are areas on pipes and/or equipment, located in accessible areas that are reading more than ten (10) times the general area radiation level (i.e., 18 inches from contact), but not less than 100 mr/hr.

5.1.5 Radiation Area is any area, accessible to personnel, in which there exists radiation at such levels that a major portion of the body could receive in any one hour a dose in excess of 5 millirem, or in any 5 consecutive days a dose in excess of 100 millirem.

5.1.6 High Radiation Area is any area, accessible to personnel, in which there exists radiation at such levels that a major portion of the body could receive in any one hour a dose in excess of 100 millirem.

5.1.7 Airborne Radioactivity Area is any area in which airborne radioactive materials exist in concentrations in excess of the limits for restricted areas specified in 10 CFR 20, Appendix B, Table I, Column I; or any area in which concentrations exist which averaged over the number of hours in any week during which individuals are in the area, exceed 25 percent of the amounts specified in 10 CFR 20, Appendix B, Table I, Column I.

5.1.8 Radioactive Material Area is any area which contains radioactive material in excess of ten times the quantities of material specified in 10 CFR 20, Appendix "C".

5.1.9 Contaminated Area is any area which contains transferable surface radioactive contamination in excess of 1000 dpm/100 cm^2 β-γ averaged over a major portion of the area.

5.1.10 Locked High Radiation Area is any area accessible to personnel in which there exists radiation at such levels that a major portion of the body could receive in any one hour a dose in excess of 1000 millirem.

Points Available

7.8 a. Who has the authority to release a clearance if it is impossible to contact the individual that holds the clearance? (1.0)

b. List all individuals by title who can authorize clearances. (1.0)

Answer

7.8 a. The available supervisor having jurisdiction over that circuit or piece of equipment.

7.8 b. NPS, NPA, ~~or~~ NWE.

Reference

a. 5.11 OP 0010122, Rev. 21.

b. 8.1 OP 0010122, Rev. 21.

,75

Points Available

7.9 Answer True or False. (From OP-0010122, In Plant Clearance Orders) The operator need not have the equipment clearance order with him when executing or releasing a clearance. (0.5)

Answer

7.9 False

Reference

4.10 OP-0010122, Rev. 21.

Points Available

7.10 Answer True or False. (From OP-0010122 In Plant Clearance Orders) An air operated valve that fails open shall not be considered closed unless it is jacked closed with an installed jacking device. (0.5)

Answer

7.10 True

Reference

4.6 OP-0010122, Rev. 21.

Points Available

7.11 Operating procedure 2-0030124 has a statement that says "operation of the turbine at low frequencies is to be avoided. Explain why this precaution is imposed and what the consequences are of operating with degraded turbine frequency. (1.0)

Answer

7.11 Blade resonance (0.5)
Accumulative lifetime limit (0.5)

Reference

pp. 2, OP 2-0030124

Turb startup zero to full load

Points Available

7.12 Operating Procedure 2-1010020 has a limitation that states "instrument air pressure should be maintained above 85 psig. What component is responsible for that limit? (1.5)

The operability of

Answer

7.12 ~~[illegible]~~ MSIVs

Reference

OP 2-1010020, Rev. 3

Inst. Air. Syst.

Points Available

7.13 Operating Procedure 2-070002 (Condensate and Feedwater Operation) has a precaution that states "do not operate two condensate pumps in parallel under low or zero flow conditions. Explain why this is imposed. (1.0)

Answer

7.13 4.4 attached

Reference

OP 2-070020, Rev. 2

2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE UNIT NO. 2
OPERATING PROCEDURE 2-0700020
REVISION 2

1.0 TITLE:

CONDENSATE AND FEEDWATER SYSTEM OPERATION - NORMAL OPERATION

2.0 REVIEW AND APPROVAL:

Reviewed by Facility Review Group ______________ October 15, 1982

Approved by C. M. Wethy ______ Plant Manager October 15, 1982

Revision 2 Reviewed by FRG ______________ 1-26 1984

Approved by [signature] ______ Plant Manager 2-13-1984

3.0 PURPOSE:

This procedure provides instructions for valve operation required to supply heated feedwater in ample quantity and required quality to the Steam Generators for power production.

4.0 LIMITS AND PRECAUTIONS:

4.1 The respective 2A or 2B Condensate Pump motor breaker must be open before operating the 2C Condensate Pump transfer switch.

4.2 The Condensate Pump discharge valve must be closed or less than ten handwheel turns open when a Condensate Pump is started and the system is not pressurized.

4.3 The minimum recirculation flow for a Condensate Pump is 2500 GPM. To avoid damage to the pump, do NOT operate a Condensate Pump at minimum flow for longer than two hours.

4.4 Do not operate two Condensate Pumps in parallel under low or zero flow conditions. Operation of both pumps under these conditions will cause one pump not to meet its minimum flow requirements.

4.5 Although seal water is normally supplied from an orificed line from the pump discharge, Condensate Storage Tank supplied seal water should be lined up as a backup supply.

4.6 Do NOT start Condensate Pump motors more than three (3) times successively from ambient temperature. For subsequent starts at rated temperature, allow 20 minutes of running time or 40 minutes at stop.

Points Available

7.14 As per OP 2-0210020, what precaution must be taken when placing the standby ion exchanger in service and why is this necessary? (1.0)

Answer

7.14 4.6 Attached

Reference

OP 2-0210020, Rev. 7

- End of Section 7 -

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE UNIT 2
OPERATING PROCEDURE NUMBER 2-0210020
REVISION 7

2

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This document is not controlled. Before use, verify information with a controlled document.

1.0 TITLE:

CHARGING AND LETDOWN - NORMAL OPERATION

2.0 REVIEW AND APPROVAL:

Reviewed by Facility Review Group March 30 1982

Approved by C. M. Wethy Plant Manager March 30 1982

Revision 7 Reviewed by FRG 1-26 1984

Approved by [signature] Plant Manager 2-13-1984

3.0 PURPOSE:

This procedure provides instructions for the operation of the Charging and Letdown System (CVCS) including the purification section.

4.0 LIMITS AND PRECAUTIONS:

4.1 Explosive mixtures of hydrogen and air in the Volume Control Tank (VCT) shall be avoided at all times. The oxygen concentration shall be maintained less than 2 percent by volume.

4.2 The temperature of the reactor coolant downstream of the Letdown Heat Exchanger should be maintained less than 140°F.

4.3 To avoid operation of RV-2115 (VCT relief) due to the accumulation of non-condensable gases, the VCT should be vented before the pressure approaches 65 psig (high pressure alarm setpoint).

4.4 The charging and letdown systems should be started and stopped simultaneously to minimize pressure and temperature transients in the Charging and Letdown System.

4.5 Letdown flow should be maintained below 135 gpm (high flow alarm setpoint).

4.6 When placing the standby Ion Exchanger (IX) in service, care should be taken to ensure the resin bed has been borated to closely match Reactor Coolant System (RCS) boron concentration to prevent an inadvertent positive reactivity insertion. A new resin bed will remove boric acid from the coolant water as the anion resin changes from the hydorxyl form to the borate form.

4.7 When in Modes 1, 2, 3 and 4, two Charging Pumps shall be operable.

4.8 Minimum NPSH of the Charging Pumps is 9 psia. Low pump suction pressure trip is at 10 psia.

Points Available

8.0 ADMINISTRATIVE PROCEDURES, CONDITIONS AND LIMITATIONS (25)

8.1 Match the event in Column 1 with the proper action statement in Column 2. Assume Mode 1 operation at above 50% of rated thermal power. The action statements in Column 2 may be used more than once. Consider the events in Column 1 to be independent of each other. (3.5)

(a) The Moderator Temperature coefficient is shown to be $+1.0 \times 10^{-4}$ delta k/k/°F during 90% operation

(b) Tavg is 510°F.

Unit 2 LCO is 15 kW/ft Hence no action for "(c)" for Unit 2 is an acceptable answer.

* (c) Calculated Peak Linear Heat Density is 13.5 Kw/ft

(d) Containment inspection reveals a non-isolatable leak from the letdown line piping.

(e) The Refueling Water Storage Tank is at 45°F.

(f) The containment pressure is at 1 psig.

(g) Diesel generator fuel transfer pump has broken.

(1) Immediately initiate Boration at greater than 40 gpm of a solution greater than 1720 ppm Boron

(2) Place the nuclear reactor power plant in Hot Standby in 6 hours.

(3) Restore the nuclear reactor power plant to within limits in 15 minutes or be in Hot Standby in the next 15 minutes.

(4) Restore the nuclear reactor power plant to within limits within 1 hour or be in Hot Standby in 6 hours.

(5) Restore the required system to within limits within 72 hours or be in Hot Standby in the next 6 hours.

Answer(s)

8.1 The answer for (a) is (2); (+0.5)
The answer for (b) is (3); (+0.5)
The answer for (c) is (4); (+0.5) ← Needs to indicate Unit 2 - if no Action required
The answer for (d) is (2); (+0.5)
The answer for (e) is (4); (+0.5)
The answer for (f) is (4); (+0.5)
The answer for (g) is (5); (+0.5)

Reference(s)

1. Tech-Specs 3.1.1.1, 3.1.1.5, 3.1.2.8, 3.2.1, 3.4.6.2 and 3.5.4; SL 1.

Points Available

8.2 Following a reactor trip, what are three administrative requirements that must be met prior to startup, in accordance with procedure AP-0010520? (1.5)

Answer(s)

8.2 Post-trip analysis (0.5), management review of causes of trip (0.5), satisfactory understanding of cause of the trip (0.5).

Reference(s)

1. Administrative Procedures 0010520, Facility Review Group, SL 1&2.

Points Available

8.3 The following sequential events occur at St. Lucie Unit 1. Using the attached portions of EPIP 3100022E Rev. 14, determine when you would declare an unusual event and when you would upgrade the situation to appropriate higher level events. Include the category of events that you would use to upgrade the event. Assume that you are the site emergency coordinator throughout this event. Do not include any tech spec action statements. (4.0)

(a) containment high range radiation monitor fails to zero output.

(b) The condenser vacuum drops to 10 psi vacuum.

(c) Electrical buses 2A1, 2A2, 2B1, 2B2 lose voltage.

(d) The blowdown monitors indicate a higher than normal reading.

(e) Subcooled margin monitor indicate 17°F subcooled.

(f) Pressurizer level starts to drop rapidly.

(g) Steam generator A level starts to increase.

(h) A fire is reported in the switch yard.

(i) Post 1 radiation monitor indicates 1600 mrem/hr.

(j) Grab sample if containment indicates levels 1200 times normal.

Answer(s)

8.3 (c) (+0.5) Category 7, unusual event (+0.5)
* (f) or (g) (+0.75) Category 1, site area emergency (+0.75)
(i) (+0.5) Category 2, general emergency (+0.5)
(-0.5) for identifying an alert

(+0.0 min, +4.0 max)

Reference(s)

1. EPIP 3100022E, SL 1&2.

ST. LUCIE PLANT
EMERGENCY PLANT IMPLEMENTING PROCEDURE 3100022E, REVISION 14
CLASSIFICATION OF EMERGENCIES

PRIMARY DEPRESSURIZATION
Page 1 of 4

EVENT / CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
NON SPECIFIC	A. Unplanned initiation of ECCS (Emergency Core Cooling System) 1. ECCS pumps running as indicated by motor amps, AND 2. ECCS header isolation valves open as indicated by valve position indication lights, OR B. Safety or relief valve fails to close 1. Reactor Coolant System (RCS) a. Indication of flow through pressurizer relief valves as indicated on the acoustic valve flow monitor, AND b. RCS pressure drops to < 1600 psia c. Indication of leaking/partially open relief valve: refer to ABNORMAL PRIMARY LEAK RATE 2. Main Steam System a. Unusual decrease in pressurizer pressure and level with decreasing Tave, AND b. Abnormal drop in steam generator pressure to less than 500 psia, AND c. Visual or audible verification of steam relief lifting			
ACTION	1. Complete actions listed on the Plant/Radiological checklist for Unusual Event			

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EMERGENCY PLANT IMPLEMENTING PROCEDURE 3100022E, REVISION 14
CLASSIFICATION OF EMERGENCIES

PRIMARY DEPRESSURIZATION
Sheet 2 of 4

CLASS / EVENT	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ABNORMAL PRIMARY LEAK RATE INDICATION OF LEAKING/PARTIALLY OPEN RELIEF VALVES	RCS leakage greater than allowed by Technical Specifications A. RCS water inventory balance indicates 1. greater than 1 gpm unidentified leakage, OR 2. greater than 10 gpm identified leakage, OR B. Inspection reveals any RCS pressure boundary leakage	RCS leak greater than 50 gpm - radioactive material release primary to containment atmosphere 1. Charging/letdown mismatch backed up by RCS water inventory balance indicating > 50 gpm and < 132 gpm leakage, AND 2. Containment or plant vent radiation process monitor reading above normal	LOCA greater than capacity of charging pumps 1.a Unusual decrease in pressurizer level and pressure with constant Tave, OR b Loss of RCS pressure causing loss of subcooling margin, AND 2.a Makeup rate greater than capacity of 3 charging pumps (132 gpm), OR 2.b Containment pressure >2 psig or containment radiation monitors indicate above normal values OR 3. Containment high range radiation monitors indicate $\geq 7.3 \times 10^3$ R/hr (post LOCA monitors between 100 and 1000 mR/hr if CHRRM inoperable)	A release has occurred or is in progress resulting in: Off-site Dose Calculation EPIP 3100033E worksheet values in excess of 1 R/hr (whole body) or an integrated dose of 5R (thyroid) or containment high range radiation monitor greater than or equal to 1.47×10^5 R/hr (post LOCA monitors greater than 1000 mR/hr if CHRRM inoperable)
ACTION	1. Complete actions listed on the Plant/Radiological checklist for Unusual Event	1. Complete actions listed on the Plant/Radiological checklist for Alert	1. Complete actions listed on the Plant/Radiological checklist for Site Area Emergency	1. Complete actions listed on the Plant/Radiological checklist for General Emergency

ST. LUCIE PLANT
EMERGENCY PLANT IMPLEMENTING PROCEDURE 3100022E, REVISION 14
CLASSIFICATION OF EMERGENCIES

PRIMARY DEPRESSURIZATION
Sheet 3 of 4

CLASS / EVENT	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ABNORMAL PRIMARY TO SECONDARY LEAK RATE	RCS PRI/SEC leakage greater than 1 gpm 1. RCS water inventory balance indicates: A. Greater than 1 gpm unidentified leakage, AND B. Blowdown process monitors or condensate air ejector process monitor reading (*) above normal or increasing	Rapid gross failure of one steam generator tube (w/i charging pp capacity) with loss of off-site power 1. Unusual decrease in pressurizer pressure and level with constant Tave followed by: 2. Above normal or increasing steam generator blowdown or condensate air ejector radiation process monitor readings(*), AND 3. Loss of the 6.9 KV and 4.16 KV buses (1A1, 1A2, 1B1, 1B2) or (2A1, 2A2, 2B1, 2B2), OR Rapid failure of steam generator tubes (> charging pp capacity) 1. Unusual decrease in pressurizer pressure and level with constant Tave, AND 2. Simultaneous unusual increase in one steam generator's pressure and level followed by 3. Above normal steam generator blowdown or air ejector radiation process monitor readings(*)	Rapid failure of steam generator tubes (> charging pp capacity) with a loss of off-site power 1. Unusual decrease in pressurizer pressure and level with constant Tave, AND 2. Above normal readings on radiation process monitors for steam generator blowdown or condensate air ejector(*), AND 3. Loss of the 6.9 KV and 4.16 KV buses (1A1, 1A2, 1B1, 1B2) or (2A1, 2A2, 2B1, 2B2) AND 4. Simultaneous unusual increase in one steam generator's pressure and level	
ACTION	1. Complete actions listed on the Plant/Radiological checklist for Unusual Event	1. Complete actions listed on the Plant/Radiological checklist for Alert	1. Complete actions listed on the Plant/Radiological checklist for Site Area Emergency	

(*) If operating

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EMERGENCY PLANT IMPLEMENTING PROCEDURE 3100022E, REVISION 14
CLASSIFICATION OF EMERGENCIES

PRIMARY DEPRESSURIZATION
Sheet 4 of 4

EVENT / CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
LOSS OF SECONDARY COOLANT	Rapid depressurization of secondary side with no primary to secondary leakage 1. Unusual decrease in pressurizer pressure and level with decreasing Tave, AND 2. Simultaneous abnormal drop in main steam or steam generator pressure to less than 500 psia	Major steam leak with greater than 10 gpm primary/secondary leak rate 1. Unusual decrease in pressurizer pressure and level with decreasing Tave, AND 2. Abnormal drop in main steam or steam generator pressure to < 500 psia, AND 3. Steam generator blowdown or condensate air ejector radiation process monitors(*) indicate above normal (OR known pri-sec leak of > 10 gpm)	Major steam leak with greater than 50 gpm primary/secondary leak rate and fuel damage indicated 1. Unusual decrease in pressurizer pressure and level with decreasing Tave, AND 2. Abnormal drop in main steam or steam generator pressure to < 500 psia, AND 3. Steam generator blowdown or condensate air ejector radiation process monitors(*) indicate above normal (OR known pri-sec leak of > 50 gpm) 4. Fuel damage as indicated by last known primary sample	
ACTION	1. Complete actions listed on the Plant/Radiological checklist for Unusual Event	1. Complete actions listed on the Plant/Radiological checklist for Alert	1. Complete actions listed on the Plant/Radiological checklist for Site Area Emergency	

(*) If operating

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EMERGENCY PLANT IMPLEMENTING PROCEDURE 3100022E, REVISION 14
CLASSIFICATION OF EMERGENCIES

ABNORMAL RADIATION, CONTAMINATION OR EFFLUENT RELEASE VALUES

CLASS / EVENT	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
UNCONTROLLED EFFLUENT RELEASE	Radiological effluent Tech Specs limits exceeded 1. Plant effluent monitor(s) exceed alarm setpoint(s) followed by 2. Confirmed analysis results for gaseous or liquid release which exceeds Technical Specification limits NOTE: If analysis not available within 1 hour, classify as UNUSUAL EVENT.	A release has occurred or is in progress that is 10 times the Tech Spec limit (as shown by sample/survey) NOTE: If analysis not available within 1 hour, classify as ALERT.	A release has occurred or is in progress resulting in: Off-site Dose Calculation EPIP 3100033E worksheet values in excess of 50 mR/hr (whole body) 250 mR/hr (thyroid) for 1/2 hour OR 500 mR/hr (whole body) 2500 mR/hr (thyroid) for two minutes at the site boundry or Containment High Range Radiation Monitor > 7.3×10^3 R/hr	A release has occurred or is in progress resulting in: Off-site Dose Calculation EPIP 3100033E worksheet values in excess of 1 R/hr (whole body) or an integrated dose of 5 R (thyroid) or containment high range radiation monitor $\geq 1.47 \times 10^5$ R/hr (post 1 or 2 monitor > 1000 mR/hr if CHRRM inoperable
HIGH RADIATION LEVELS IN PLANT		High radiation levels or high airborne contamination which indicates a severe degradation in the control of radioactive materials 1. Installed radiation monitoring stations indicate abnormally high radiation levels, OR 2. Installed airborne particulate or iodine activity monitors indicate abnormally high levels, AND 3. Levels >1000 times normal are confirmed by area surveys and/or analysis of grab samples (if available) NOTE: If analysis not available within 30 minutes, classify as ALERT.		
ACTION	1. Complete actions listed on the Plant/Radiological checklist for Unusual Event	1. Complete actions listed on the Plant/Radiological checklist for Alert	1. Complete actions listed on the Plant/Radiological checklist for Site Area Emergency	1. Complete actions listed on the Plant/Radiological checklist for General Emergency

ST. LUCIE PLANT
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CLASSIFICATION OF EMERGENCIES

MISCELLANEOUS EVENTS
Sheet 1 of 4

EVENT / CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ABNORMAL TEMPERATURE/ PRESSURE	Abnormal coolant temperature and/or pressure, abnormal fuel temperature 1. Subcooling margin monitor indicates less than 20°F subcooling, OR 2. Highest hot leg temperature is less than 20°F below the saturation temperature, OR 3. Plant incore thermocouples indicate abnormal fuel temperatures, OR greater than saturation at the prevailing pressure 4. RCS pressure exceeds 2500 psia for longer than 30 seconds			
REACTOR COOLANT PUMP(S) FAILURE	One or more reactor coolant pumps fail. Refer to resultant problems, i.e.: Fuel failure, OR Abnormal Temperature/Pressure, OR Other conditions requiring increased awareness			
ACTION	1. Complete actions listed on the Plant/Radiological checklist for Unusual Event			

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CLASSIFICATION OF EMERGENCIES

MISCELLANEOUS EVENTS
Sheet 2 of 4

CLASS / EVENT	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
OTHER CONDITIONS REQUIRING INCREASED AWARENESS	1. A plant shutdown is required by Technical Specifications OR 2. The plant is shut down under abnormal conditions (e.g., exceeding cooldown rates or primary system pipe cracks are found during operation) 3. Other conditions exist which, in the opinion of the Nuclear Plant Supervisor, warrant the declaration of an Unusual Event	1. The Technical Support Center AND/OR Near Site Emergency Operations Facility are activated for other than drill purposes 2. Other conditions exist which, in the opinion of the Nuclear Plant Supervisor, warrant the declaration of an Alert	1. The Emergency Centers are activated, AND 2. Monitoring Teams are mobilized, AND 3. A precautionary public notification is made concerning an abnormal plant condition for other than drill purposes 4. Other conditions exist which, in the opinion of the Nuclear Plant Supervisor, warrant the declaration of a Site Area Emergency	An event resulting in escalation of the Emergency Classification to General Emergency with Imminent Substantial Core Damage and potential for release of large amounts of radioactivity in a short period of time, such as: 1. LOCA with failure of ECCS 2. Loss of Secondary Heat Sink 3. Sustained station blackout with loss of secondary heat sink 4. Failure of containment heat removal systems in the later stages of an accident resulting in loss of containment 5. Other conditions exist which, in the opinion of the Nuclear Plant Supervisor, warrant the declaration of General Emergency
ACTION	1. Complete actions listed on the Plant/Radiological checklist for Unusual Event	1. Complete actions listed on the Plant/Radiological checklist for Alert	1. Complete actions listed on the Plant/Radiological checklist for Site Area Emergency	1. Complete actions listed on the Plant/Radiological checklist for General Emergency

MISCELLANEOUS EVENTS
Sheet 3 of 4

EVENT / CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
POTENTIAL CORE MELT FROM ANY CAUSE				Other plant conditions exist, in the opinion of the Emergency Coordinator, that make release of large amounts of radioactivity in a short time period appear possible or likely. One example is any potential or in progress core melt situation. a. Small or large LOCA's with failure of ECCS to perform leading to severe core degradation or melt, OR b. Transient initiated by loss of feedwater and condensate systems (heat removal system) followed by failure of emergency feedwater system for extended period, OR c. Transient requiring operation of shutdown systems with failure to scram which results in core damage or additional failure of core cooling and makeup systems, OR d. Failure of off-site and on-site power along with total loss of emergency feedwater makeup capability for several hours, OR e. Small LOCA and initially successful ECCS with subsequent failure of containment heat removal systems, OR
ACTION				1. Complete actions listed on the Plant/Radiological checklist for General Emergency

CATEGORY 6

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CLASSIFICATION OF EMERGENCIES

MISCELLANEOUS EVENTS
Sheet 4 of 4

EVENT / CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
POTENTIAL CORE MELT FROM ANY CAUSE				f. Any major internal or external event (e.g., fire, earthquake or tornado substantially beyond design basis) which has/could in EC's opinion cause massive common damage to plant systems resulting in any of the above NOTES: 1. Most likely containment failure mode is melt-through with release of gases only. Quicker releases are expected for failure of containment isolation system. 2. A General Emergency must be declared for these events. The likelihood of corrective action (repair of AFW pump etc.) should be considered (only) in considering precautionary Protective Action Recommendations.
ACTION				1. Complete actions listed on the Plant/Radiological checklist for General Emergency

CATEGORY 6

ST. LUCIE PLANT
EMERGENCY PLANT IMPLEMENTING PROCEDURE 3100022E, REVISION 14
CLASSIFICATION OF EMERGENCIES

ELECTRICAL MALFUNCTION

CLASS / EVENT	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
LOSS OF POWER	Loss of off-site power or loss of on-site AC power capability 1. Turbine generator trips with plant startup transformers non-functional, OR 2. Loss of voltage on both A3 and B3 4.16 KV buses for more than 15 seconds.	Loss of off-site power and loss of all on-site AC power 1. Turbine generator trip with plant startup transformers non-functional, AND 2. Failure of both emergency diesel generators to start or synchronize, OR Loss of all on-site DC power Drop in A and B DC bus voltages to < 70 volts	Loss of off-site power and loss of on-site AC power > 15 minutes 1. Turbine generator trips with plant startup transformers unavailable for service, AND 2. Sustained failure of both emergency diesel generators to start or synchronize for > 15 minutes, OR Loss of all vital on-site DC power for > 15 minutes Sustained drop in A and B DC bus voltages to 70V DC for > 15 minutes.	
ACTION	1. Complete actions listed on the Plant/Radiological checklist for Unusual Event	1. Complete actions listed on the Plant/Radiological checklist for Alert	1. Complete actions listed on the Plant/Radiological checklist for Site Area Emergency	

ST. LUCIE PLANT
EMERGENCY PLANT IMPLEMENTING PROCEDURE 3100022E, REVISION 14
CLASSIFICATION OF EMERGENCIES

DEGRADATION OF CONTROL CAPABILITIES
Sheet 1 of 2

CLASS / EVENT	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ENGINEERED SAFETY FEATURES/REACTOR PROTECTION SYSTEM SYSTEMS TO PLACE PLANT IN COLD SHUTDOWN/ CONTROL ROOM OPERATION AND FIRE PROTECTION SYSTEM	1. A safety features actuation/ reactor protection system functional unit shown in Technical Specification Table 33-1, 33-3 becomes inoperable per Technical Specifications and requires plant shutdown, OR 2. The fire suppression system or a portion thereof becomes inoperable per Technical Specification 3.7.11.1 and requires plant shutdown	1. Loss of functions needed for cold shutdown, OR 2. Failure of the Reactor Protection System to bring the reactor subcritical when needed, OR 3. Evacuation of Control Room (for other than drill purposes) with shutdown control established locally at the Hot Shutdown Control Panel	Loss of any function or system, in the opinion of the Emergency Coordinator, which precludes placing the plant in Hot Shutdown, OR Control Room is evacuated (for other than drill purposes) and shutdown control cannot be established locally at the Hot Shutdown Control Panel within 15 minutes, OR Failure of the Reactor Protection System to bring the reactor subcritical when needed, AND manual trip fails	
LOSS OF ALARMS	Significant loss of effluent monitoring capability, meterological monitoring instrumentation communications, indication and alarm panels, etc., which impairs ability to perform accident or emergency assessment	All annunciators lost	All annunciator alarms lost > 15 minutes with plant not in cold shutdown, OR Plant transient occurs with all alarms lost.	
ACTION	1. Complete actions listed on the Plant/Radiological checklist for Unusual Event	1. Complete actions listed on the Plant/Radiological checklist for Alert	1. Complete actions listed on the Plant/Radiological checklist for Site Area Emergency	

DEGRADATION OF CONTROL CAPABILITIES
Sheet 2 of 2

EVENT \ CLASS	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
LOSS OF CONTAINMENT INTEGRITY LOSS OF FISSION PRODUCT BARRIER	Loss of containment integrity requiring shutdown by Tech Specs A. Penetrations required to be closed during accident conditions are NOT: 1. Capable of being closed by the ESFAS, OR 2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions except as provided in Table 3.6-2 of Technical Specification 3.6.3, OR B. An equipment hatch is not closed and sealed, OR C. An airlock is not operable per Technical Specification 3.6.1.3, OR D. A sealing mechanism associated with a penetration (e.g., welds, bellows, or O-rings) becomes inoperable			Loss of 2 of 3 fission product barrier with potential for loss of the third (Any two of the following conditions exist and the third is imminent) 1. Severe fuel clad damage (refer to "FUEL ELEMENT FAILURE") 2. LOCA or TUBE RUPTURE on unisolable steam generator 3. Containment integrity breached NOTE: Refer to "Potential Core Melt From Any Cause"
ACTION	1. Complete actions listed on the Plant/Radiological checklist for Unusual Event			1. Complete actions listed on the Plant/Radiological checklist for General Emergency

Points Available

8.4 a. After the gaseous radwaste tank samples have been taken for a gaseous radwaste release, what are the operations administrative responsibilities before initiating the release? (1.0)

b. What actions are required by operations personnel if, during a routine gaseous radwaste release, the expected activity is significantly exceeded? (1.0)

Answer(s)

8.4 a. The operator should verify that the radwaste form was complete (+0.5) and that the release in question would not exceed limits (+0.5).

8.4 b. Verify that the release has been terminated (+0.5) and have radwaste resample the tank to determine if the indicated activity is real (+0.5).

Reference(s)

1. OP-530021, Rev. 2, SL 1&2.

Points Available

8.5 During shift turnover, you notice from the shift supervisor's log that one charging pump is out of service and maintenance is being performed on another charging pump. What control room indications, meters, personnel, logs, tags or forms would you check to insure that the proper procedures were being followed and the plant was being maintained in a safe condition? (2.0)

Answer(s)

8.5 Check to see that the appropriate charging pumps have been tagged out (+0.5). Check to see that the plant work order for the maintenance being performed has the proper sign-off (+0.5). Check the LCO log or status board (T.S. 3.5.2) for Beginning time of CCO (+0.5), verify that the oncoming crew is aware of the LCO and that proper actions are being taken to upgrade the status of one additional charging pump within the 72 hours allowed for this condition (+0.5).

Reference(s)

1. Tech-Specs 3.5.2, 3.1.2.4, Procedure 0010120; SL 1&2.

8.6 a. List 5 of the 8 general categories of information that the Unit 2 Nuclear Operator (NO) would receive on his Nuclear Operator Turnover Checklist. (i.e., The Unit 1 NO would receive information about the Unique tank status). (2.0)

b. Who should review the Nuclear Operator Turnover Checklist after relieving the shift? (1.0)

Answer(s)

8.6 a. See the attached list. (+0.4) for each item (heading) on the list, (+2.0 max).

b. NPS, ANPS, NWE (+0.33 each)

Reference(s)

1. Procedure 00101200, Appendix A, SL 1&2.

APPENDIX A

NUCLEAR OPERATOR TURNOVER CHECKLIST

Unit No. ______________

Date ______________

Off-going Shift: Day Peak Mid
(circle one)

1) Emergency Diesels available for operation: Yes No
(Including check of oil levels, fuel inventory, (circle one)
local alarms). If no, explain below:

2) Surveillances/Tests performed or in progress:

Name	Reason	Completion Time
1.		
2.		
3.		

3) Abnormal lineups or unique conditions:

Name	Reason
1.	
2.	
3.	

4) Equipment or systems out of service:

Name	Reason
1.	
2.	
3.	
4.	

APPENDIX A

NUCLEAR OPERATOR TURNOVER CHECKLIST
(continued)

5) Unit 1 or 2 Tank Status:

TANK	LEVEL	STATUS	TANK	LEVEL	STATUS
Chemical Drain			Primary Water		
Equipment Drain A			Holdup A		
Aerated Waste Storage			Holdup B		
Waste Condensate A			Holdup C		
Waste Condensate B			Holdup D		
Laundry Drain A			Boric Acid Makeup A		
Laundry Drain B			Boric Acid Makeup B		

6) Unique Tank Status: (Unit 1 NO only)

Monitor Storage A			Waste Monitor A		
Monitor Storage B			Waste Monitor B		
Monitor Storage C			Demineralized Water		

7)

GAS DECAY TANK	PRESSURE	STATUS		IN SERV.	OUT SERV.	RECIRC
A			B.A. CONC. A			
B			B.A. CONC. B			
C						

8) Additional Information:

9) Signatures:

Off-going Shift ____________________

On-coming Shift ____________________

NOTE: Route this sheet to NWE following turnover.

Points Available

8.7 a. What are 3 of the 6 conditions when an RWP is required? (1.0)

b. How long is a RWP good for? (1.0)

c. What are the Florida Power and Light Companyu guidelines for whole body exposure in 1 quarter? (+0.5)

d. Name 1 condition that the ANPS would refuse to sign a correctly filled out RWP. (1.0)

Answer(s)

8.7 a. Only the underlined portions of Sections 6.3.1.1, 6.3.1.2, 6.3.1.3, 6.3.1.4(1), (2), (3), (+0.33 each, +1.0 max).

* 8.7 b. 7 days (+1.0) or completion of the job

* 8.7 c. ~~500~~ 800 mrem/qtr (+0.5)

8.7 d. Actual or anticipate plant transients could change the radiation exposure in the area where the RWP was to be used. (+1.0)

Reference(s)

1. HP-2, Sections 6.3, 6.3.1 (attached), 6.1.1.2, 6.1.1.3, SL 1&2.

worn.

6.2.3.3 Personnel shall wear a minimum of coveralls, cotton gloves (for dry contamination) or rubber gloves (for wet contamination) and shoe covers for any maintenance work on contaminated systems.

6.2.3.4 For jobs requiring a Radiation Work Permit (RWP), the protective clothing requirements for the job shall be specified on the RWP. Personnel entering a RWP area to perform observation and inspection activities only, may wear less than the RWP clothing requirements if so directed by Health Physics.

6.3 Radiation Work Permits

The primary purpose of a Radiation Work Permit (RWP) is to provide Health Physics with a vehicle whereby they can evaluate and plan jobs in order to maintain radiation exposure ALARA. The Florida Power & Light Company RWP philosophy is based on the fact that control of radiation and contamination is accomplished primarily by training, Health Physics job surveillance, pre-job planning, post-job evaluation, and special instructions. A RWP normally describes the radiological conditions of a job, the protective clothing, monitoring to be performed, and any other special instructions.

6.3.1 RWP Requirements

An RWP shall be required for the following conditions.

6.3.1.1 Entry into high radiation areas, airborne radioactivity areas, areas contaminated to levels in excess of 10,000 dpm/100 cm^2, or into any area posted as "RWP REQUIRED FOR ENTRY."

6.3.1.2 Entry into the reactor containment at any time during and subsequent to initial reactor startup.

6.3.1.3 Maintenance or inspection of equipment contaminated in excess of 10,000 dpm/100 cm^2.

6.3.1.4 Work assignments involving changes (withdrawing, uncovering, opening, valving, disassembling, moving) that have the following potential as the work progresses:

1. Exposure of a major portion of the body to a radiation dose in excess of 100 mrem in any one hour.

2. Increasing surface contamination levels to exceed 10,000 dpm/100 cm^2.

3. Increasing airborne radioactivity to values exceeding 25% of those listed or referred to in

Points Available

8.8 With respect to Caution Tags:

a. Who authorizes the placing or removal of caution tags? (0.5)

b. Who authorizes the placing or removal of all caution tags identifying electrical grounds? (0.5)

c. What is the maximum length of time for which a caution tag may remain in force? Explain your actions in the event a caution tag has been in place for this duration. (1.0)

Answer(s)

8.8 a. ANPS or NPS

8.8 b. Electrical department foremen or supervisor

8.8 c. 1 month (+0.5), reviewed by ANPS/NPS and their current status reported to the OPS Supervisor (+0.5)

Reference(s)

1. Administrative Procedure 0010135, Revision 1, Section 5.1 and 5.2; SL 1&2.

2. Administrative Procedure 0010135, Revision 1, Section 7.3; SL 1&2.

Points Available

8.9 What are the three restrictions regarding temporary changes to procedures? (1.5)

Answer(s)

8.9 The intent of the procedure is not altered. (+0.5)

The changes are approved by 2 members of plant management, one holding a SRO license on the affected unit. (+0.5)

The change is documented, reviewed by FRG, and approved by the Plant Manager within 14 days. (+0.5)

Reference(s)

1. NRC Question Bank, Section 10, Question 28, SL 1&2.

Points Available

8.10 During plant operations an operator observes that the RPS channels A and B upper and lower safety channel nuclear instrumentation power indications are reading 0% of full power and RPS channels C and D upper and lower NI power indications are reading 100% of full power.

a. Identify 3 control room indications the operator could use to determine which channels had failed. (1.0)

b. What RPS functional units in the faulted channel should be removed from service? (1.0)

c. How should the RPS functional units in the faulted channels be removed from service? (1.0)

Answer(s)

* 8.10 a. Perform primary calorimetry (core flow and ΔT)
Perform secondary calorimetry (turbine, power or steam bypass demand)
Check S/G steam flow
Check incore detector indications
(+0.33 each, +1.0 max)

b. Variable power level, LPP, TM/LP, rate of change of power (+0.25 each)

* c. Bypass one of the functional audits and trip the other functional units in each of the affected RPS channels. Of the 2 faulted channels, bypass one and trip the other.

Reference(s)

TS.3.3.1

- Last Page of Section 8.0 -

- LAST PAGE OF EXAMINATION -

ENCLOSURE 3
(2 of 2)

U. S. NUCLEAR REGULATORY COMMISSION

REACTOR OPERATOR LICENSE EXAMINATION

Facility: St Lucie 1 & 2

Reactor Type: CE

Date Administered: 19 June 1984

Examiner: Upton Smith

Candidate: Answer Key

INSTRUCTIONS TO CANDIDATE:

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

Category Value	% of Total	Candidate's Score	% of Cat. Value	Category
25	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~~ 24	25			4. Procedures: Normal, Abnormal, Emergency, and Radiological Control
100				TOTALS

Final Grade ________ %

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

Candidate's Signature

Points Available

1.0 PRINCIPLES OF NUCLEAR REACTOR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW (25)

1.1 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 about 50% of the energy released per fission event.

(b.) Fissioning of the isotope, U^{238}, provides about 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 energy is deposited as heat in the coolant.

Answer(s)

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

Reference(s)

1. C-E PWR Simulator Training Facility, "Reactor Theory," Florida Power & Light Company, SL 1&2.
~~pp 500(82H8)/ds-30, ds-65, ds-66, Figures 18&19, ds-73, ds-74~~
pp 500(82H8)/ds - 42, 43

Points Available

1.2 Which one of the statements below is not correct for a nuclear reactor of the St. Lucie type? (1.0)

(a.) The product of the macroscopic cross section (Σ) and the neutron flux (ϕ) gives the neutron reaction rate (interactions per cm^3 per sec.).

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

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

(d.) The thermal-neutron microscopic fission cross section for Uranium-238 is very large (at least 500 barns).

Answer(s)

1.2 The answer is (d.). (+1.0)

Reference(s)

1. C-E PWR Simulator Training Facility, "Reactor Theory," Florida Power & Light Company, SL 1&2.

PP 500(82 H8)/ds - 30, 65, 66, 73, 74, Figures 18, 19

Points Available

1.3 St. Lucie Unit 1 has been operating at a constant 100% of full power for 7 days.

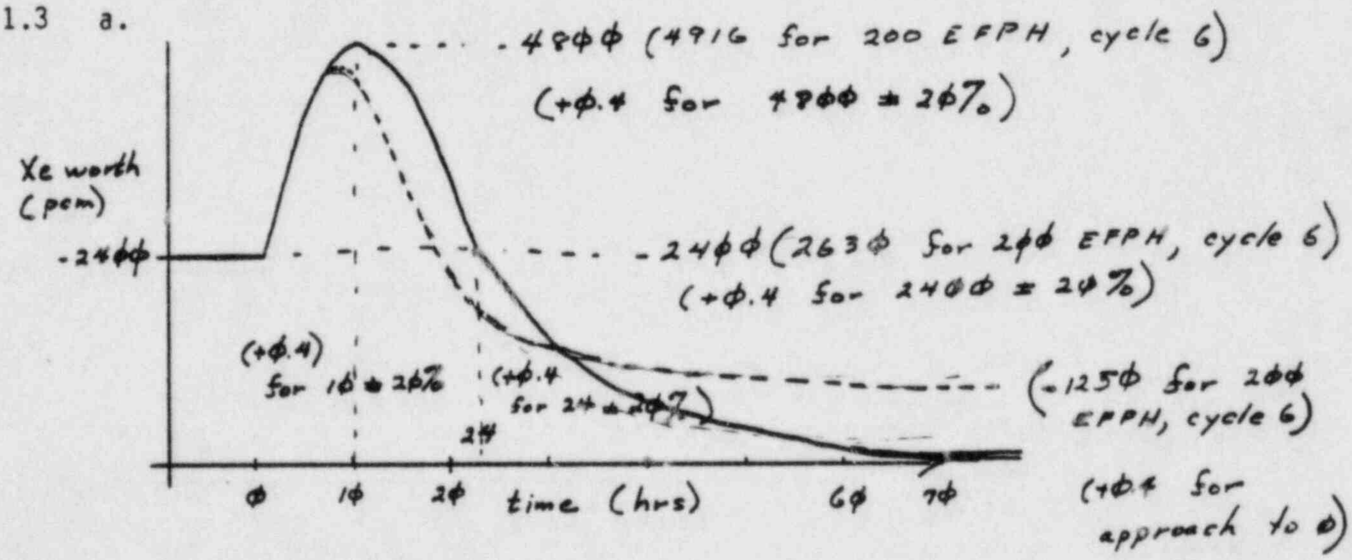
a. The reactor is tripped and not re-started. Sketch the trace of Xenon worth verses time. Start with the time of the trip and show 72 hours on the graph. Indicate the initial worth, the final worth and the worth at other significant points on the sketch. (2.0)

b. If instead, six hours after the trip, the reactor is brought critical and a constant power level maintained at 10^{-4}% of full power; show on the sketch of Part "a" the affect of this operation. (1.0)

c. Alternately, if six hours after the trip, the reactor is brought critical and a constant power level maintained at 20% of full power; show on the sketch of Part "a" the affect of this maneuver. (1.0)

Answer(s)

1.3 a.



b. There will be no affect. (+1.0)

c. See the curve in Part "a". (+1.0 for shape)

Reference(s)

1. "Florida Power and Light (St. Lucie) NRC Question Bank," Section 1, Question 3, SL 1&2.

2. Unit 1 Plant Curves, Figures A.1 - A.4, SL 1.

Points Available

1.4 During operation of the power plant at 100% of full power, at the end-of-cycle (EOC) portion of the fuel cycle, the boron concentration is at 50 ppm in the RCS. Due to coastdown, T_{ave} is 10°F less than T_{ref}. All rods are out, and the reactor regulating system is in manual.

a. Explain how you would raise T_{ave} back toward T_{ref} while maintaining the rods at their present positions. (1.0)

b. If the steam flowrate from the steam generator is given a step decrease, what is the sequence of physical phenomena that would cause the nuclear reactor power to decrease? For example, specify the changes (or lack of change) in the primary coolant, in the secondary loop, to the core reactivity and power level. (2.0)

Answer(s)

1.4 a. T_{ave} can be increased by reducing the load on the main turbine (+1.0). Alternately, the operator can ~~remove~~ reduce the boron concentration by dilution or by the additional use of the de-borating ion exchanger (+1.0). (+1.0 max)

1.4 b. . The steam flowrate decreases.

. The temperature and pressure in the steam generator increase (+0.5).

. T_c increases due to the lower steam flowrate (+0.2).

. Due to the increase in T_c, T_{ave} and T_H increase (+0.5).

. Due to α_M, the reactivity decreases and the nuclear reactor neutron flux (power) decreases (~~+0.5~~ +1.0).

. Due to the neutron flux decrease, the reactivity increases to bring the reactivity to zero (+0.5).

(+2.0 max).

Reference(s)

1. Nuclear Energy Training, Module 4, Plant Performance, Section 7.1, NUS Training Corporation.

2. Nuclear Energy Training, Module 3, Reactor Operation, Section 7.2, NUS Training Corporation.

Points Available

1.5 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?

a. The power plant is operating at 100% of full power with the pressurizer pressure at 2250 psia. (1.0)

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

Use the attached steam tables, if necessry. (1.0)

Answer(s)

1.5 a. 228°F (+1.0)

1.5 b. 310°F (+1.0)

Reference(s)

1. "Academic Program for Nuclear Power Plant Personnel," Volume III, General Physics Corporation.

2. "Power Plant Thermodynamics," SL 1&2.

pp 36, 37, 49, 5φ, 52

Points Available

1.6 For a nuclear reactor at a certain operating condition, $T_c = 545°F$ and $T_H = 595°F$. The reactor coolant flowrate is 7×10^7 lbm/hr and the specific heat of water under these conditions is 1.3 Btu/lbm-°F.

What is the rate of thermal energy (heat) addition to the reactor coolant? Show your calculations. (2.0)

Answer(s)

1.6 $\dot{Q} = \dot{m}\, c_p\, \Delta T$ (+1.0)

$= (7 \times 10^7)(1.3)(595-545)$

$= 4.55 \times 10^9$ Btu/hr (+1.0)

Reference(s)

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

2. "Power Plant Thermodynamics," SL 1&2.
pp 68, 74, 83

1.7 The power plant has been operating steadily at 100% of full power for 5 days. The power level must be decreased (for a surveillance) to 80% of full power.

a. The power level is reduced by inserting control rods. Explain what effect this would have on the axial-shape-index (ASI). (1.0)

b. If the operator reduces power level by borating and maintains T_{ave} on the programmed values, what effect would this have on the ASI? Explain. (1.0)

Answer(s)

1.7 a. Adding negative reactivity to the top of the nuclear reactor core (+0.5) causes the ASI increase (become more positive) (+0.5), because the neutron flux level decrease is higher in the top half of the core than in the bottom half.

1.7 b. As T_{ave} is reduced, T_H is reduced more than T_C is reduced. This, via temperature feedback, will cause the neutron flux level to increase more in the top half of the core than in the bottom half (+0.5). Hence, ASI will decrease (+0.5).

Reference(s)

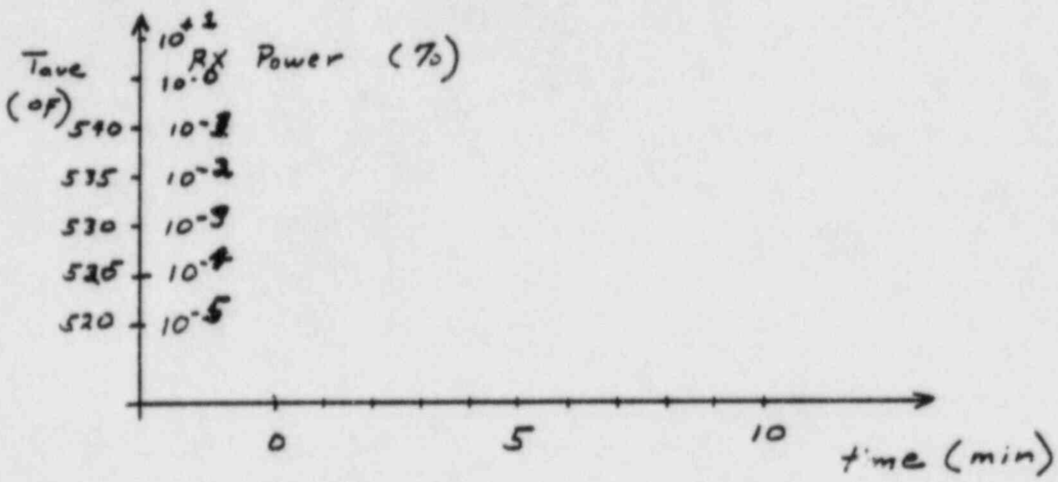
1. Nuclear Energy Training, Module 3, Plant Performance, Section 7.2, NUS Training Corporation.

2. C-E Pwr Simulator Training Facility, "Reactor Theory," Florida Power & Light Company, SL 1&2.

Points Available

1.8 The reactor is critical and at the end-of-cycle (EOC) portion of the fuel cycle. The boron concentration is 50 ppm. The power level is steady at 5×10^{-4}% of full power. T_{ave} has been controlled at 532°F by the steam bypass control system. An operator pulls rods, establishes a startup rate of 1 DPM (decade per minute) and leaves the rods in these positions.

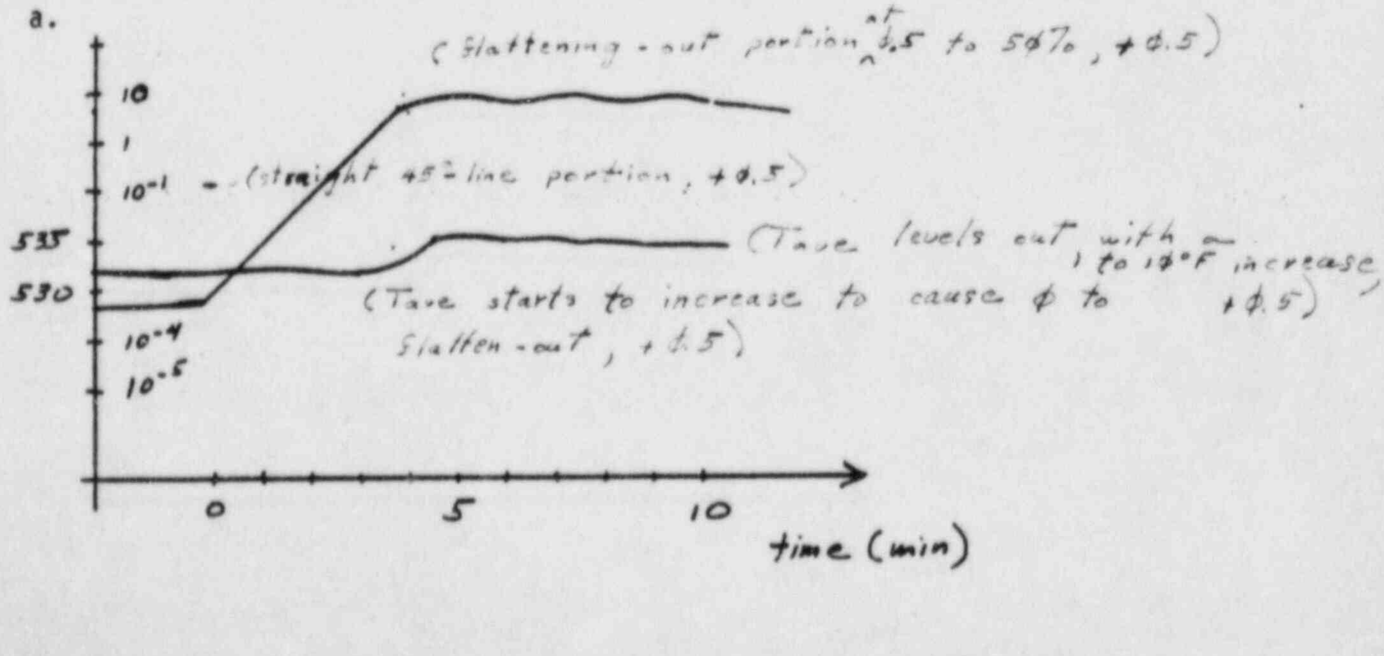
a. Sketch nuclear reactor power and T_{ave} as a function of time. Sketch your graphs on the axes given below. (2.0)



b. Would you expect the nuclear reactor power to be higher or lower for the same transient at the beginning-of-cycle with the boron concentration at 800 ppm? Explain. (1.0)

Answer(s)

1.8 a.



1.8 b. The nuclear reactor power would be higher (+0.5). At BOC the moderator temperature coefficient is almost zero and hence the overall power coefficient is less in magnitude coming largely from the doppler coefficient. Hence the negative reactivity from a temperature change will be smaller in magnitude (+0.5).

Reference(s)

1. Nuclear Energy Training, Module 3, Plant Performance, Section 7.2, NUS Training Corporation.

2. C-E PWR Simulator Training Facility, "Reactor Theory," Florida Power & Light Company, SL 1&2.

An alternately correct ~~[illegible]~~ reasoning is to note that at BOC the operator would have to add a greater amount of reactivity to the nuclear reactor core to attain the 1 DPM ramp rate.

$$\rho_{\$} = \frac{\rho_{\Delta k/k}}{\bar{\beta}_{eff}}$$

$$\rho_{\Delta k/k} = \bar{\beta}_{eff}\, \rho_{\$}$$

ρ as measured in \$ is independent of β and a given $\rho_{\$}$ produces a given reactor period. Hence, if $\bar{\beta}_{eff}$ is higher at BOC, then ρ as measured in $\Delta k/k$ must be higher. Hence, even if the temperature coefficients are the same, it will take a larger ΔT to turn the power ramp around.

Points Available

1.9 A cylindrical tank is 10 ft. in diameter and 20 ft. high. It is one-half filled with water. Air is the cover gas which is at atmospheric pressure (14.7 psia). The tank is drained with the vent closed until the volume of liquid is one-half of what it was before. (The tank is 1/4 filled now.) What is the pressure in the tank? (2.0)

Answer(s)

1.9	$V = (3.14)(5\ \text{ft})^2\ (20\ \text{ft})$	(+0.25)
	$= 1570\ \text{ft}^3$	(+0.25)
	$V\ \text{gas initial} = (1570)(.5)$	(+0.25)
	$= 785\ \text{ft}^3$	(+0.25)
	$V\ \text{gas final} = (1570)(.75)$	(+0.25)
	$= 1178\ \text{ft}^3$	(+0.25)
	$P_1\ V_1 = P_2\ V_2$	(+0.25)
	$(14.7)(785) = P_2\ (1178)$	~~(+0.25)~~
	$P_2 = 9.80\ \text{psia}$	(+0.25)

Reference(s)

1. "Academic Program for Nuclear Power Plant Personnel," Volume III, General Physics Corporation. Chapter 2, Section A

2. C-E PWR Simulator Training Facility, "Power Plant Thermodynamics", pp. 16, 17, Florida Power & Light Company, SL 182.

Points Available

The following statements apply to questions 1.10, 1.11 and 1.12. St. Lucie Unit 2 has been operating at a steady 100% of full power for 7 days with all of the control rods fully withdrawn from the nuclear reactor core. The burnup is 5000 EFPH on cycle 1. Use any of the provided figures and tables. Show your work and/or how you arrived at your answer.

1.10 What is the value of boron worth in (%Δk/k)/ppm? (1.5)

1.11 Explain why the critical Soluble Boron Concentration vs. burnup (Figure C.2) is somewhat constant at approximately 430 ppm between 500 and 3000 EFPH. (1.5)

1.12 The nuclear reactor power level is now reduced at 70% of full rated power at a rate of 1% per minute by inserting rods (programmed bank motion). What is the rod position when the power change is complete? (2.0)

Answer(s)

1.10 Using Figure C.1, boron worth = 12.65 pcm/ppm. (+1.0 for 12.6 to 12.7)

$\%$ k/k = 10^{-3} pcm/ppm
= .01265 %Δk/k/ppm. (+0.5)

1.11 For the first (initial) fuel load, burnable poison rods [that are not part of a CEA (i.e., not moveable)] are loaded into the nuclear-reactor core. The material is usually a boron-10 based material which is inserted in place of thimble plugs. The result is that the poison burns out at a rate that when combined with the fuel burnup rate results in a flat reactivity change as a function of EFPH. When the burnable poison is consumed, the reactivity change vs. EFPH is essentially linear. (The burnable poison is used to insure a negative moderator temperature coefficient for a reactor core at initial BOC.) (+1.5)

1.12 30%/(1%/min) = 30 min.
    30 min. is fast with respect to Xenon changes

using Figure A.1
    reactivity = -1137 - (-780)
        = -357 pcm (+1.0 for -367 to -347 pcm)

using Figure A.6,
    4532 - 357 = 4175 pcm
    4175 pcm = 106 inches on Bank 5 (+1.0 for 82 to 109)

<u>Reference(s)</u>

1. Unit 2 Plant Physics Curves, SL 2.

2. C-E PWR Simulator Training Facility, "Reactor Theory," SL 1&2.

Points Available

2.0 PLANT DESIGN, INCLUDING SAFETY AND EMERGENCY SYSTEMS (25)

2.1 Which one of the statements below most accurately describes ~~most correctly~~ the design and operation of the quench tank, pressurizer and its control system? (1.0)

(a.) The steam in the pressurizer is maintained in a superheated condition so that the volume of the vapor will not shrink to zero.

(b.) The control system will prevent uncovering of the heaters following a 10% step decrease in the thermal power generated in the nuclear reactor.

(c.) The spray nozzle at the top of the pressurizer is connected through two air-operated spray control valves to the loop-1 hot leg.

(d.) The quench tank is sized to receive and condense steam from the discharges of the pressurizer safety valves; a total loss-of-load event would be handled by the opening of the rupture disc of the quench tank.

Answer(s)

2.1 The answer is (b.). (+1.0)

Reference(s)

1. Lesson Plans and System Descriptions #1, "Reactor Coolant System - RCPs - Pressurizer - Quench Tank," SL 1&2.

<u>Points Available</u>

2.2 Sketch a diagram of the letdown and charging systems for Unit 2 in order to show the pressure relief protection.

a. Sketch on the included figure the letdown and charging system and indicate the locations to which the discharges of the relief valves are sent. (2.0)

b. For each relief valve indicate their setpoints by labeling the figure ~~one~~ with the following choices. (1.0)

(1) 200 psia

(2) RC pressure + 10%

(3) 75 psig

(4) 650 psig

(5) 75 - 100 psig

(6) 1500 psig

(7) 14.7 psia

Answer(s)

See the attached figure. (+3.0)

Reference(s)

1. Lesson Plans and System Descriptions #3, "CVCS and Boron Concentration Control System," SL 1&2. {Figure "St. Lucie Plant - Unit 2 Chemical Volume and Control System" at the end of the Lesson Plan}

The attached figure is correct, but the relief valves at the suctions of the charging pumps are designed for thermal expansion relief. Thus, instead of "(1) 2[illegible] psig" for an answer, an answer referring to thermal expansion or design pressure of the charging pump suction piping is acceptable. See the included documentation.

LETDOWN & CHARGING PRESSURE RELIEF DIAGRAM

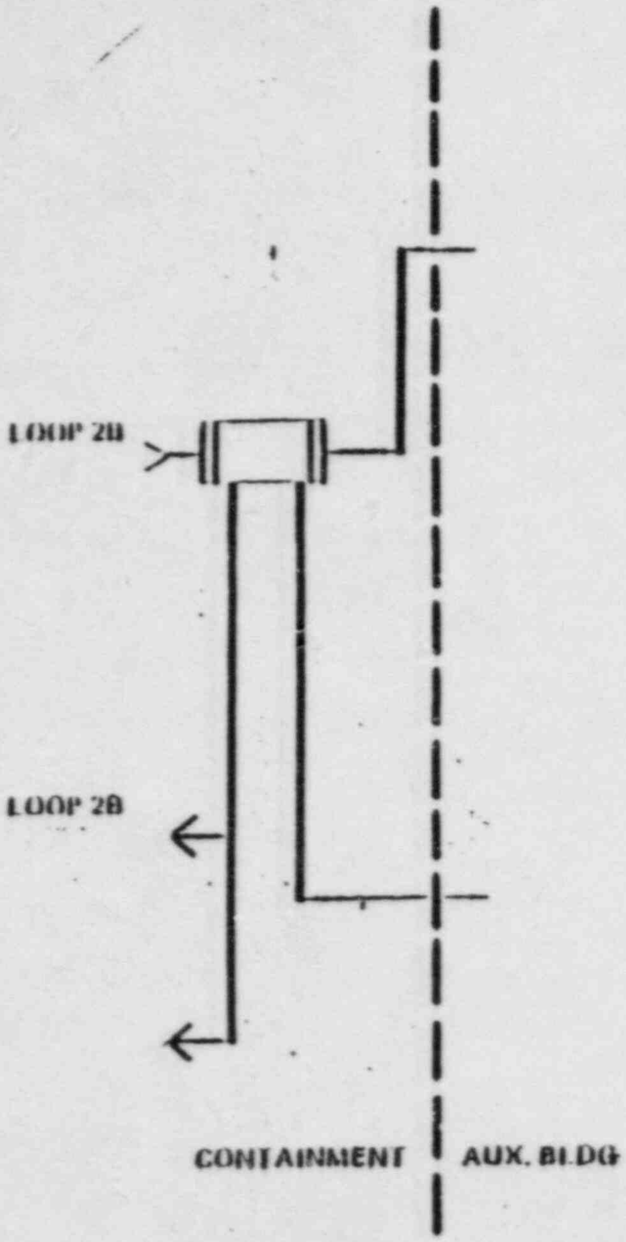


FIGURE 2.2

LETDOWN & CHARGING PRESSURE RELIEF DIAGRAM

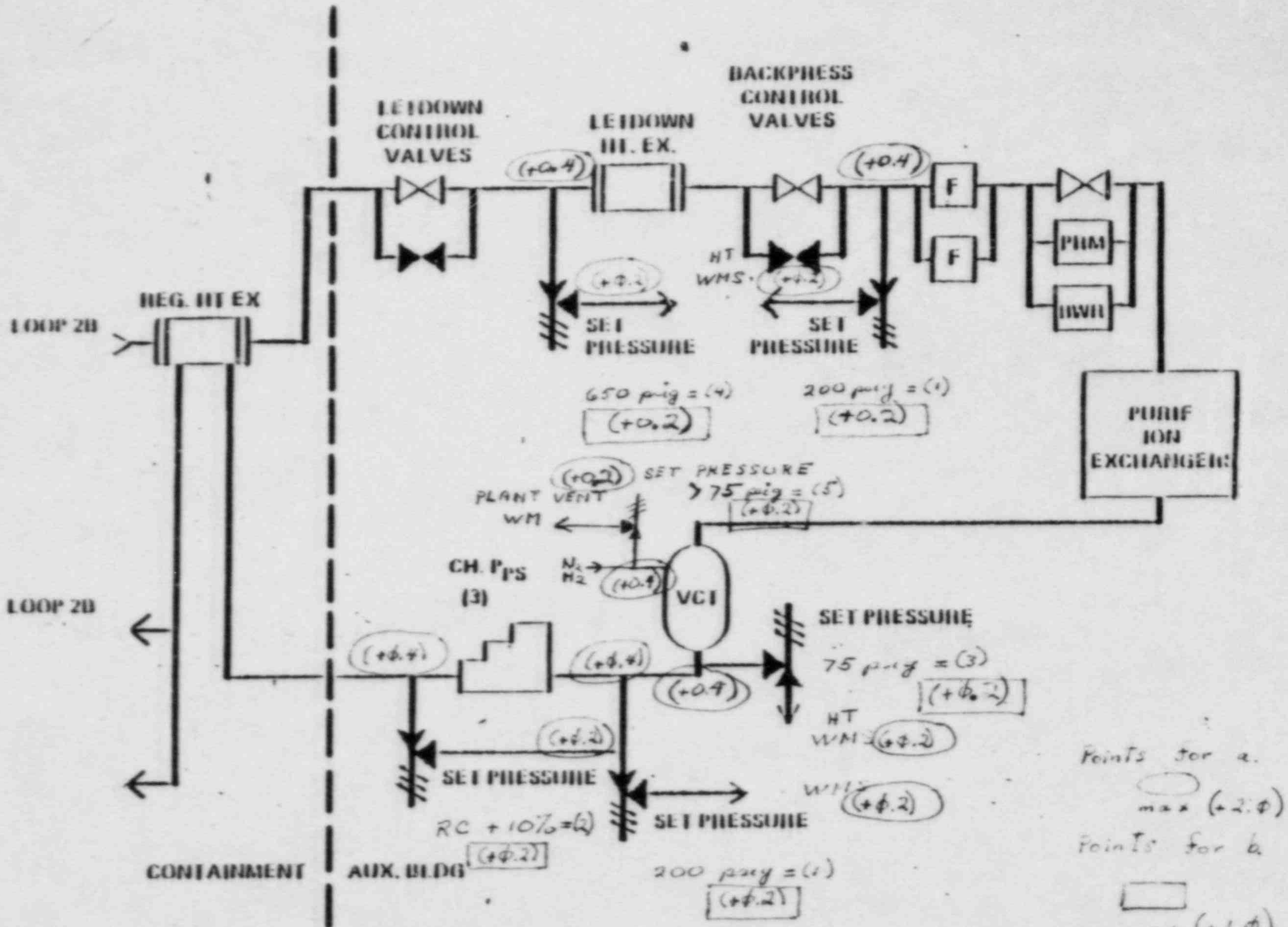


FIGURE 2.2 answer

The relief valve set pressure is equal to the design pressure (600 psig) of the intermediate pressure letdown piping and letdown heat exchanger.

2) Low-pressure letdown relief valve, V-2354

The relief valve downstream of the letdown backpressure control valves protects the low pressure piping, purification filters, ion exchangers and letdown strainer from overpressure. The valve capacity is equal to capacity of intermediate pressure letdown relief valve V-2345. The set pressure is equal to the design pressure (200 psig) of the low pressure piping and components.

3) Charging pump discharge relief valves, V-2324, V-2325, V-2326

The relief valves on the discharge side of the charging pumps are sized to pass the maximum rated flow of the associated pump with maximum backpressure without exceeding the maximum rated total head for the pump assembly. The valves are set to open when the discharge pressure exceeds the reactor coolant system design pressure (2485) by 10 percent.

4) Charging pump suction relief valves, V-2315, V-2318, V-2321

The relief valves on the suction side of the charging pumps are sized to pass the maximum fluid thermal expansion rate that would occur if the pump were operated with the suction and discharge isolation valves closed. The set pressure is less than design pressure of charging pump suction piping.

This is equivalent to a setpoint # of ~ 100 JAS.

5) Charging line thermal relief valve, V-2435

The relief valve on the charging line downstream of the regenerative heat exchanger is sized to relieve the maximum fluid thermal expansion rate that would occur if hot letdown flow continued after charging flow was stopped by closing the charging line distribution valves. The valve is a spring-loaded check valve.

6) Volume control tank relief valve, V-2115

The relief valve on the volume control tank is sized to pass a liquid flow rate equal to the sum of the following flow rates: the maximum operating flow rate from the reactor coolant pump controlled bleedoff line; the maximum letdown flow rate possible without actuating the high flow alarm on the letdown flow indicator; the design purge flow rate of the sampling system; and the maximum flow rate that the boric acid makeup system can produce with relief pressure in the volume control tank. The set pressure is equal to the design pressure of the volume control tank.

Points Available

2.3 A steamline break has occurred inside containment in the St. Lucie Unit 2 power plant. The event will dry-out the affected steam generator.

a. Starting with the fact that the flowrate of the steam from the steam generator has increased, describe the sequence of physical events that could lead to excessive RCS pressure. (2.0)

b. List the significant indications in the Unit 2 control room that would indicate that a steamline break had occurred inside containment and not a LOCA inside containment. (2.0)

c. Describe the actions that an operator could take to prevent the pressure of the nuclear reactor coolant from reaching excessive repressurization. (1.0)

Answer(s) *The grading will take into account operator actions to prevent repressurization.*

2.3 a.

- The steam flowrate from the steam generator will increase.
- The increased steam flowrate will reduce the pressure and temperature in the steam generator and cause a MSIS (+0.1).
- The low steam generator pressure, high steam generator flowrate and/or high containment pressure will cause a reactor trip (+0.1).
- The increased steam flowrate will reduce T_c and T_{ave} of the nuclear reactor coolant (+0.3).
- The reduced T_{ave} will reduce the pressurizer level (+0.3).
- *Charging vs. letdown will go to max flowrate* ([illegible])
- The drop in pressurizer level will be large enough to keep the heaters off (+0.3).
- The reduced T_{ave} and with no pressurizer heaters on will cause the pressurizer pressure to drop (+0.3).
- The reduced pressure allows the HPSI system to inject cold water (+0.3).

- Question 2.3 Continued on the Next Page -

. When the steam generator boils dry, the heat extraction rate from the primary coolant to the secondary fluid is reduced so that the heat input rate exceeds the heat extraction rate (+~~0.3~~).

. The heat balance results in T_{ave} increasing (+~~0.3~~).

. The increase in T_{ave} can increase the pressurizer level and pressure (+~~0.3~~).

(+2.0 max)

2.3 b. . T_{ave} decreasing, nuclear reactor power increasing with constant or decreasing turbine power.

. Increasing containment pressure, temperature and humidity.

. NO increase in the radiation level in containment.

. Abnormally low level and pressure in one steam generator.

. Increasing containment sump level.

. NO H_2 production indication.

(+0.5 each, +1.5 max)

2.3 c. . Terminate safety injection when the necessary criteria are met.

. Remove heat through the good steam generator.

. Adjust (secure) charging and letdown.

. Initiate pressurizer sprays.

• Maintain subcooled margin.

(~~[illegible]~~, +1.0 max)
0.25 each

Reference(s)

1. "Academic Program for Nuclear Power Plant Personnel," Volume III, General Physics Corporation.

2. SO23-3-5.6, Attachments 1&2, SONGS 2&3.

Points Available

2.4 Draw a one-line diagram of the 125 VDC Emergency and 120 VAC Vital and Instrument System, including the batteries, chargers, inverters, switches, breakers and backups. Use the attached figure, Figure 2.4. It is not necessary to draw the connections to and from the MCC 1B5, the 125 V DC BUS 1B and the 120 V AC Instrument BUS 1MD. (3.0)

Answer(s)

2.4 See diagram. (+3.0 max)

Reference(s)

1. System Descriptions #6, "Class 1E Electrical System," SONGS 2&3.

2. Lesson Plans and System Descriptions #21, "120 Volt Instrument AC; 120 Volt Vital AC" SL 1&2. Lesson Plan #33, Figure 4

3. Lesson Plans and System Descriptions #22, "125 Volt DC System," SL 1&2. Figure 4

Updated diagrams are included and are used for the grading.

125 VOLT DC / 120 VOLT AC ELECTRICAL SYSTEM

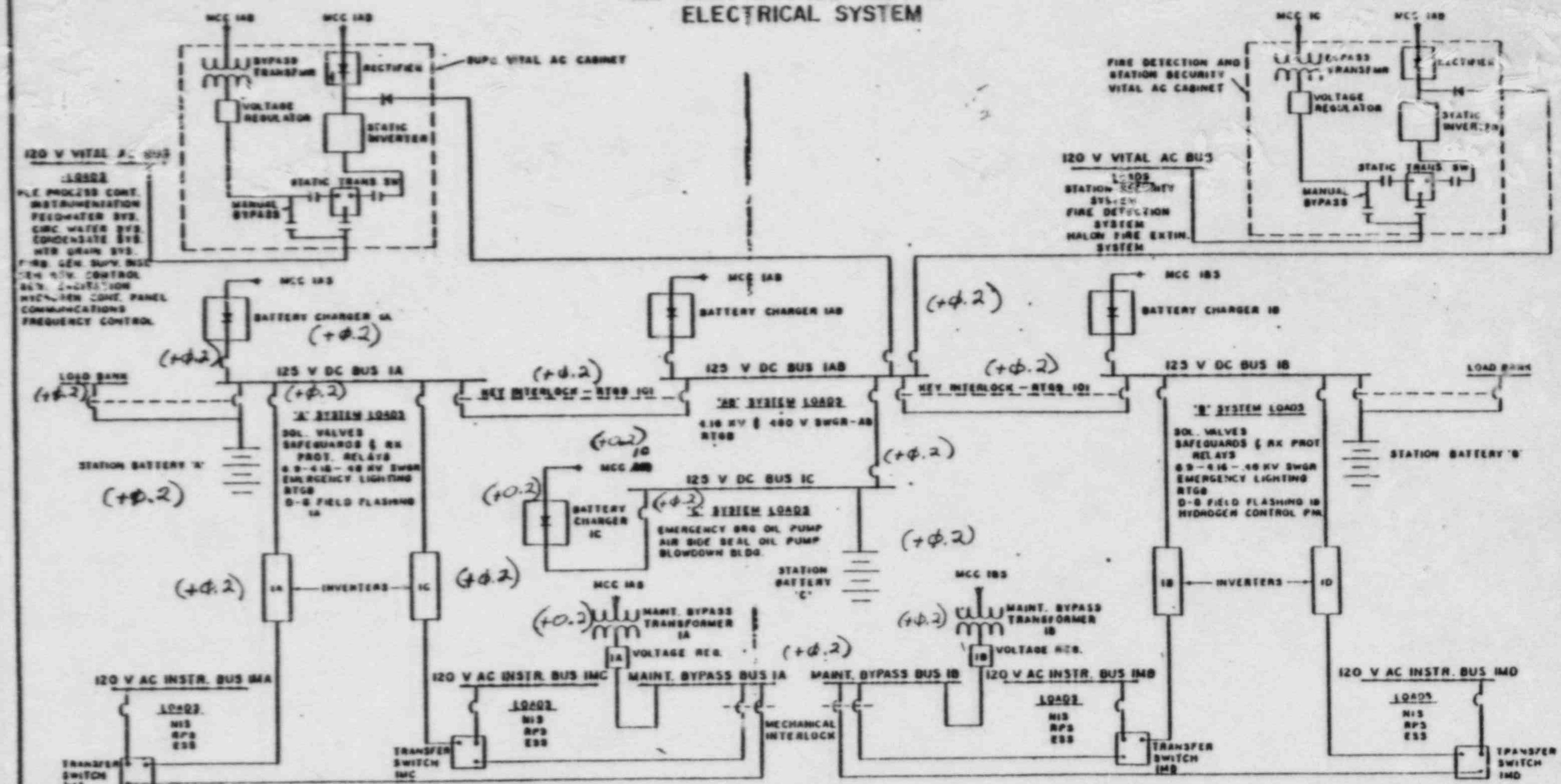


FIGURE 24 Answer

(+3.φ max)

UPDATED

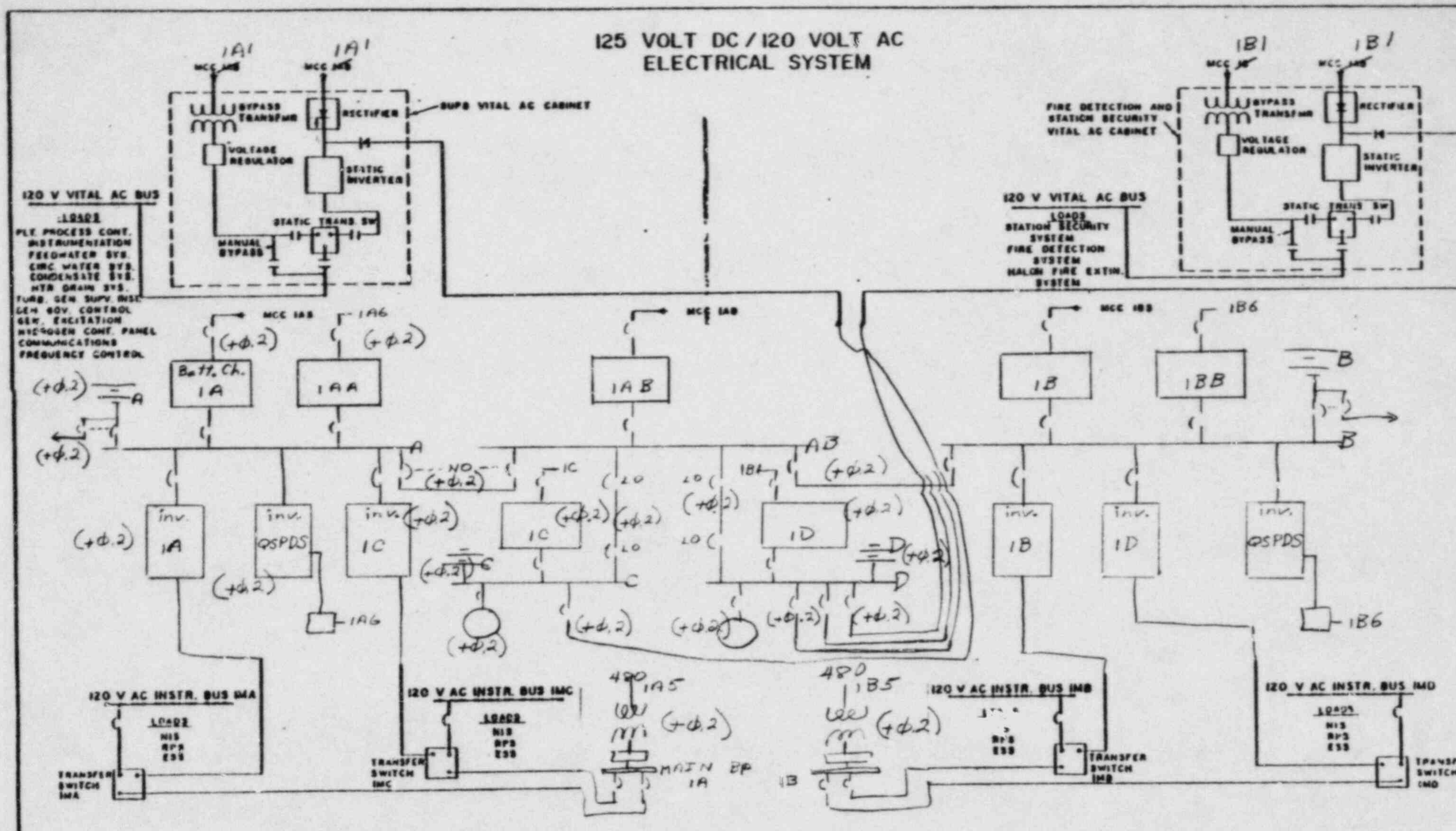
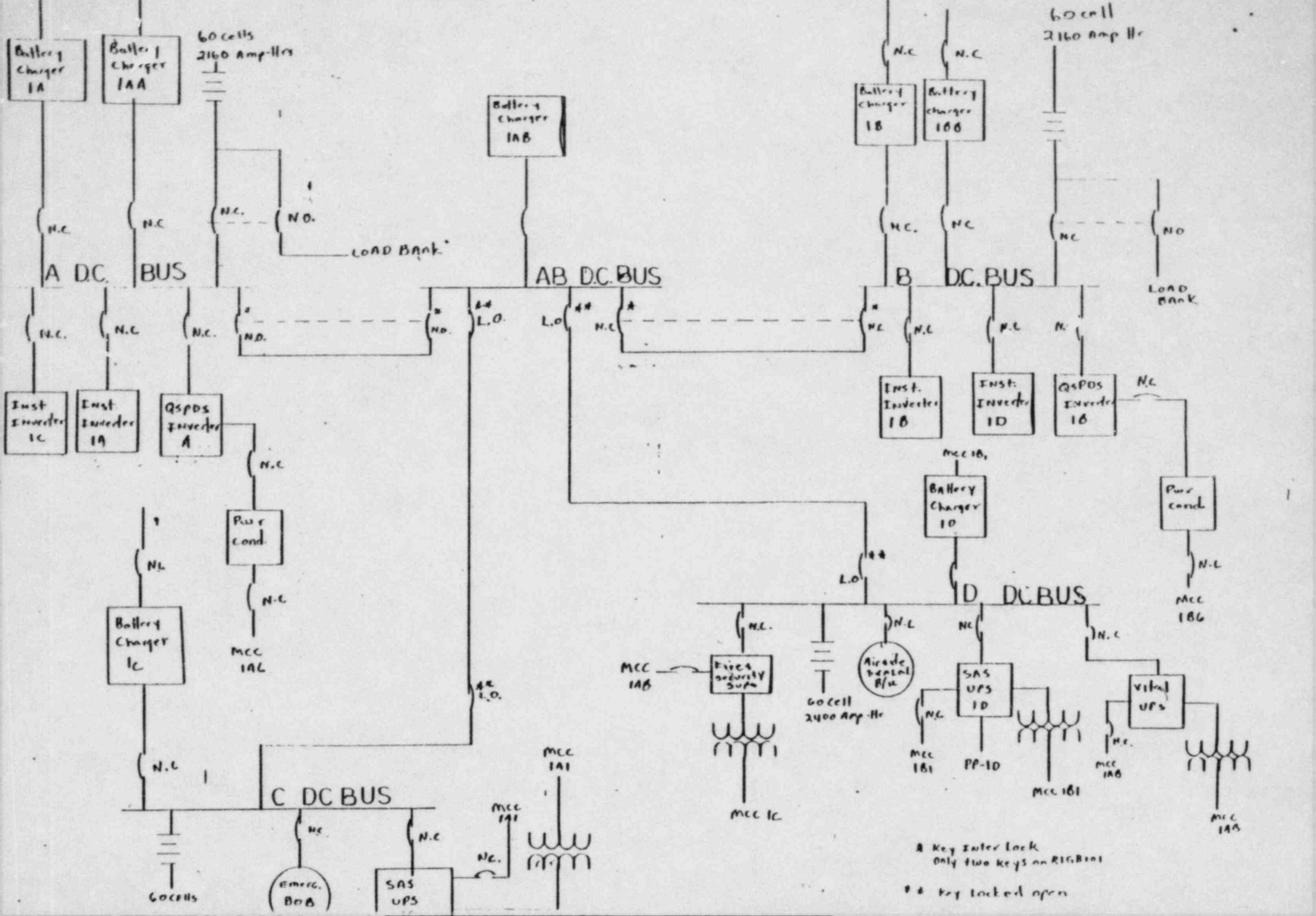


FIGURE 24

ST. LUCIE UNIT 1
125 V D.C.

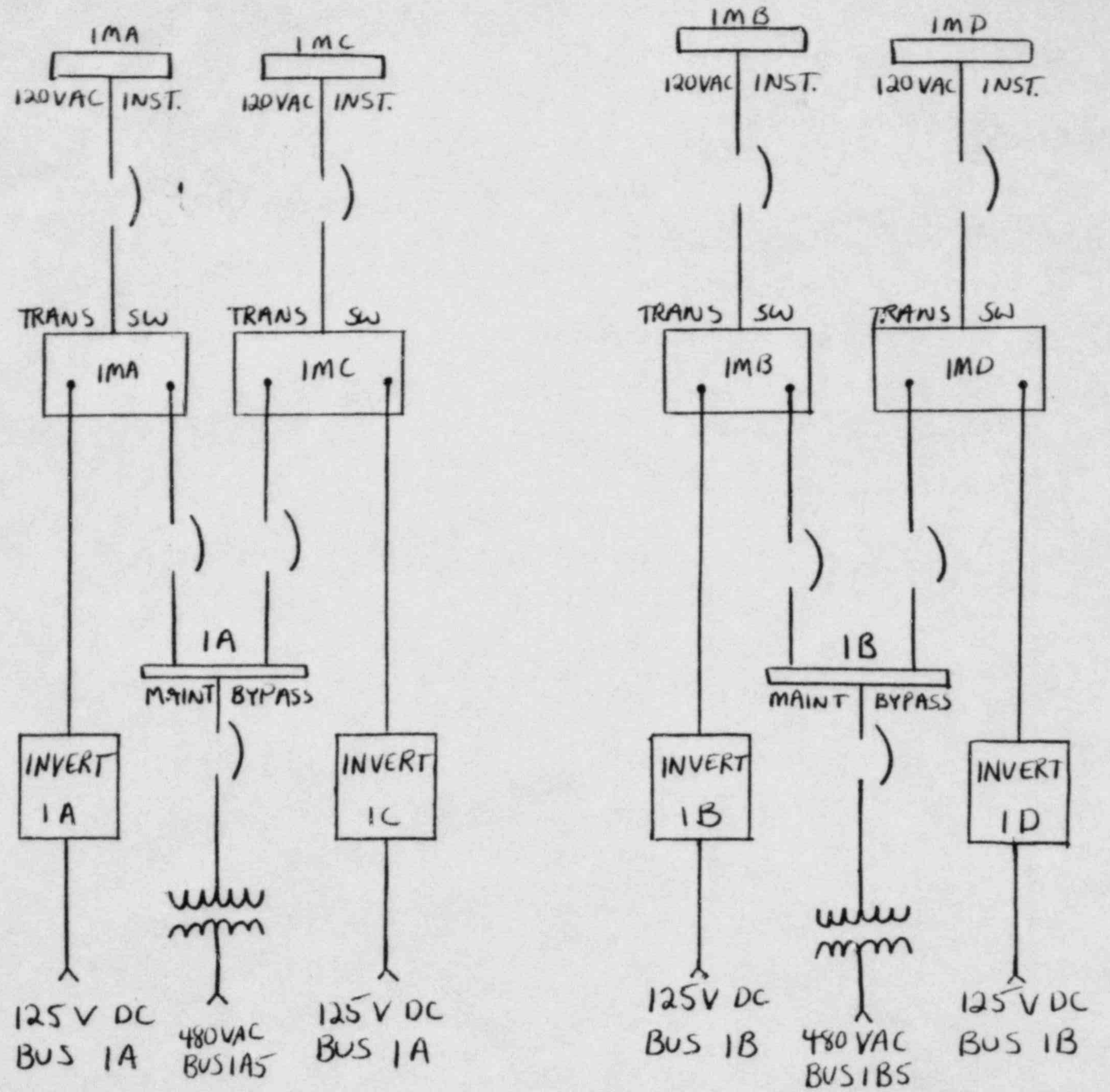
Quest. 2.4

(J)



Quest. 2.4

FIG. 4 UNIT #1 120 VAC INSTRAMENT DISTRIBUTION



9

Points Available

2.5 Consider the Component Cooling Water (CCW) System for Unit 1.

a. Which of the following is NOT supplied by the CCW System? (0.5)

1. Regenerative heat exchanger in the CVCS

2. RCP thermal barrier

3. RCP motor-oil cooler

4. Seal-water heat exchangers for the safety injection pumps

b. If a SIAS is generated, list all of the automatic actions which should take place in the CCW System. (1.0)

c. List 5 different components in each essential header (critical loop) which receive CCW during a LOCA. (1.0)

d. If there is a break in the CCW System piping such that the CCW surge tank level decreases, what automatic actions should take place on a lo-level and on a lo-lo-level signal? (1.0)

Answer(s)

2.5 a. The letdown regenerative heat exchanger is NOT serviced by the CCW. The answer is #1. (+0.5)

2.5 b. . The nonessential header (noncritical loops) of the CCW System is (are) isolated (from the CCW pumps)(+0.5). 8 valves may be listed here, along with their actions. That would be equivalent to the statement that the nonessential header is isolated.

. Both of the CCW pumps are placed into operation ~~and the essential headers separated~~ (+0.5).

2.5 c. . Shutdown heat exchanger

. 2 containment emergency air coolers

. CS pump

. CCW pump (not anticipated as an answer)

. HPSI pump

- Question 2.5 Continued on the Next Page -

. LPSI pump

(+0.2 each, + 1.0 max)

2.5 d. . On low level in the CCW surge tank, makeup water from the nuclear service water system will start (~~+0.5~~ +1.0).

. On low-low level in the CCW surge tank, the CCW surge tank isolation valve will close (+0.5). This will not occur on Unit 1.

(+1.0 max)

Reference(s)

1. System Description #17, "Component Cooling Water System," SONGS 2&3.

2. Lesson Plans and System Descriptions #9, "Component Cooling Water System," SL 1.

Points Available

2.6 Which statement describes most correctly the CEAs in either Unit 1 or 2. (1.0)

1. The withdrawal speed is fixed in automatic control but may be varied between 10 and 60 steps per minute in manual control.

2. The worth of a specific CEA is solely dependent upon the CEA-bank to which it is connected.

3. The bottom half of each rod is made of boron-carbide and the top half is made of an alloy of silver, indium and cadmium.

4. The manual sequential mode allows the operator to control the regulating banks of CEAs and to command their insertion or withdrawal at the high rate of speed.

Answer(s)

2.6 The answer is #4. (+1.0)

Reference(s)

1. System Description #13, "Control Element Drive Mechanism Control System," SONGS 2&3.

2. System Description #44, "Reactor Vessels and Internals," SONGS 2&3.

3. Lesson Plans and System Descriptions #2, "Reactor Vessel and Core Design," SL 1&2.

4. Lesson Plans and System Descriptions #29, "CEDMCS and Analog Display System," SL 1&2.

Points Available

2.7 Consider the venting system for the reactor vessel and the pressurizer.

a. What is the basic purpose of the venting system and under what conditions (normal and emergency) is it anticipated that the system would be used? (1.0)

b. Draw a one-line diagram showing this venting system. It is important to indicate to where the gases are vented. (2.0)

Answer(s)

2.7 a. The basic purpose of the venting system is to permit the operator to vent the pressurizer steam space or the reactor vessel from the control room (+0.5). It is used to vent gases during filling and draining operations or after an accident (+0.5).

* 2.7 b. See the attached diagram. (+2.0 max)

Reference(s)

1. System Description #39, "Pressurizer and Pressurizer Control System," SONGS 2&3.

2. Reactor Coolant System P&ID Diagram, Figure 5.1-3, SL 1.

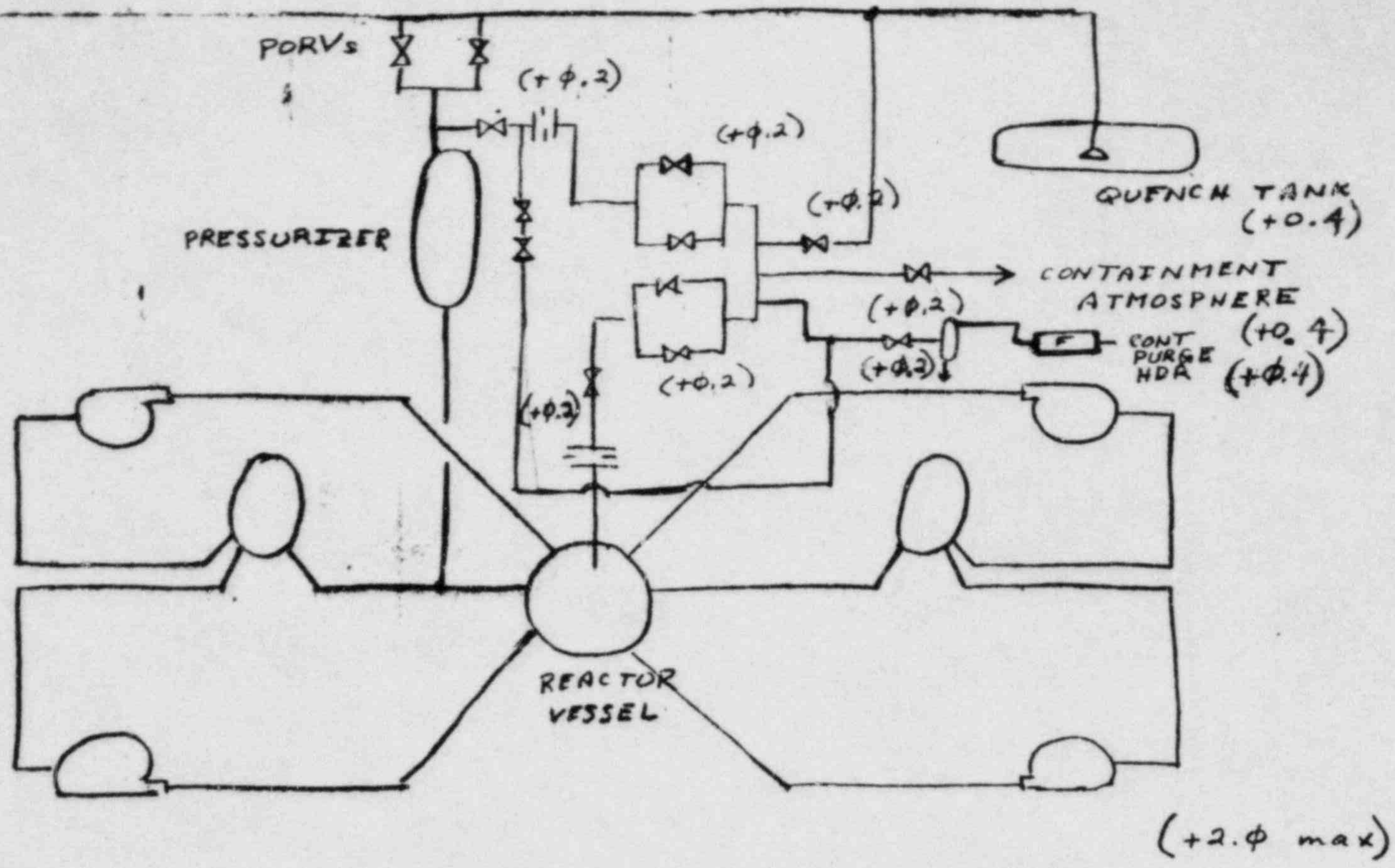


Figure 2.7 Answer

Quest. 27

REACTOR COOLANT GAS VENT SYSTEM
UNIT 1 (UNIT 2 SIMILAR)

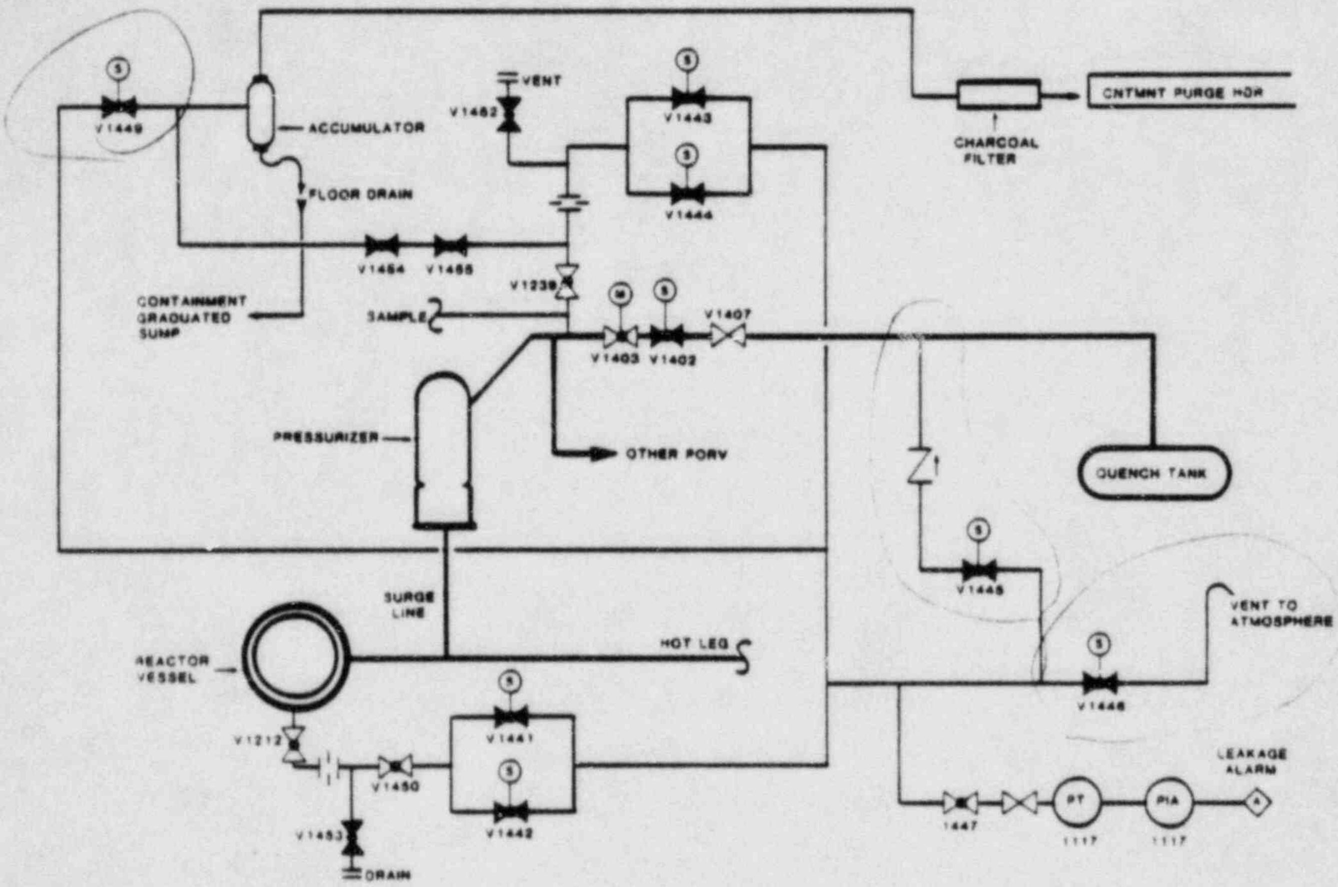


FIGURE 14

Points Available

2.8 Sealing between the nuclear reactor vessel head flanges is provided in the system design.

a. What control room instrumentation can be used to alert the operator of a failure in the sealing? Consider the failure of the inner seal and also the failure of both seals. (1.0)

b. How is the amount of leakage from the RCS from between the flanges determined? (1.0)

c. If there is a leakage from between the seals, the leakage is of what type?

(1.) unidentified leakage

(2.) identified leakage

(3.) controlled leakage

(4.) pressure boundary leakage (0.5)

Answer(s)

2.8 a. A line from the annulus is equipped with a pressure transducer and transmitter which provides ~~pressure indication and~~ an alarm (+0.7). The containment-air-monitor can be used to watch for failure of both O-ring seals (+0.3).

2.8 b. The 3/4-inch line from the annulus can be valved to the RCDT and the volume, or level, in the drain tank observed (+1.0). Alternatively, the valve can be opened to reduce the pressure to that of the RCDT and then closed. The length of time for the pressure to reach ~~a~~ the alarm ~~higher~~ valve can be observed (+1.0). (+1.0 max)

An acceptable answer is to perform the RCS leakage calculation (+1.0).

2.8 c. Identified leakage

Reference(s)

1. Tech-specs 3.4.5.2, SONGS 2&3.

2. System Description #44; Reactor Vessel and Internals System;" pp. 15, 18, 20; Figures 31, 32, 34, SONGS 2&3.

3. Lesson Plans and System Descriptions #2, "Reactor Vessel and Core Design," SL 1&2.

Points Available

2.9 For each of the functions listed in (a.), (b.) and (c.) below; match the affected valves, (1.), (2.), (3.) and/or (4.) of the main turbine. Assume that the valve position limit is at 100%. (3.0)

(a.) valves closed by the overspeed protection controller (OPC)

(b.) valves closed by the overspeed trip mechanism

(c.) valves opened when the turbine is latched

(1.) throttle valves

(2.) governor valves

(3.) reheat stop valves

(4.) interceptor valve

Answer(s)

2.9 a. (~~1~~ 2.), (4.) (+1.0)

b. (1.), (2.), (3.), (4.) (+1.0)

c. (2.), (3.), (4.) (+1.0)

Reference(s)

1. OP 2-0030124, SL 2.
2. April 1983 NRC Exam.

- End of Section 2.0 -

(1.), (2.), (3.) and (4.) is an acceptable answer provided that the answer includes a clarification to the effect that (1.) and (3.) are closed by the back-up overspeed protection

Points Available

3.0 INSTRUMENTS AND CONTROLS (25)

3.1 Consider the excore neutron detectors and their instrumentation for Unit 1.

a. What are the indications of a failed section of a dual-section UIC safety channel? (1.0)

b. Describe three control functions of the wide-range channels. (inputs to other systems) (1.0)

c. What ~~is~~ are the function[s] of the control channels. (1.0)

d. What is Q power? (1.0)

Answer(s)

3.1 a. Subchannel deviation alarm, hi-power channel trip, ~~etc.~~ TM/LP, LPD

3.1 b. SUR trip to the RPS, [illegible] bistable,

3.1 c. Input to RRS for ~~RX~~ ~~no~~ reactor vs turbine power, input to ~~ASI~~ CPCs for calculation of ASI, ~~power ratio,~~ input to core-barrel movement monitor

3.1 d. Max of ϕ or $B_{\ast}$ (ΔT) power

Reference(s)

1. NRC Question Bank, Section 4, Number 18, SL 1&2.

b ZPMP [illegible] below $10^{-?}$% of

TM/LP

LO flow

Level 1 bistable [illegible] bypass

[illegible] $10^{-?}$% [illegible]

[illegible] at 1000 cps [illegible]

from monitoring 4 detectors/channel

to 1 detector/channel

Points Available

3.2 Consider CEDS and the CEDMCS.

a. What features at both Unit 1 and Unit 2 can be by-passed? (1.0)

b. What interlocks can be by-passed at Unit 2 but not at Unit 1? (1.0)

c. Wht interlocks can be by-passed at Unit 1 but not at Unit 2? (1.0)

Answer(s)

* 3.2 a. ~~Function~~ Motion inhibits from the metroscope and ADS

3.2 b. CWP (regulating groups withdrawal prohibit)

* 3.2 c. ISH from the logic cabinet

Reference(s)

1. St. Lucie Exam #1, Question 21, SL 1&2

Points Available

3.3 Give at least 3 indications in the Unit 2 control room that can be used as a measure of RCP cavitation? (1.5)

Answer(s)

3.3 . Erratic or lower RCP motor amperage (current)

. Lower or erratic signal from the Δp across the pump

* . Increased RCP vibrational signal

. Lower subcooled margin indication

. Lower DNBR margin indication

. Fluctuating steam generator Δp

. Erratic or lower RCS flowrate, as presented by the plant computer

* (+0.33 each, +1.0 max.)

Reference(s)

1. Lesson Plans and System Descriptions #1, "Reactor Coolant System," SL 1&2.
2. System Description #41, "Reactor Coolant System," SONGS 2&3.
3. SO 23-3-5.6, SONGS 2&3.
4. SO 23-3-1.7, SONGS 2&3.
5. Nuclear Heat Transport, El Wakel, Chapter 11.

Points Available

3.4 For the following Process Effluent Monitors of Unit 2, give the type of monitor that is used. Choose from SSL, SSG, MSG, or PIG.

a. Component-cooling-water monitor (0.5)

b. Liquid-waste discharge monitor (0.5)

c. Gaseous-waste discharge monitor (0.5)

d. plant ventillation monitor (high-range ~~m~~noble gas) (0.5)

e. Fuel-handling building stack monitor (0.5)

f. Steam-generator blowdown monitor (0.5)

Answer(s)

3.4 a. SSL
3.4 b. SSL
3.4 c. SSG
3.4 d. MSG
3.4 e. PIG
3.4 f. SSL

Reference(s)

1. St. Lucie Exam #4, Question 40, SL 1&2

Points Available

3.5 Explain the principle of operation (detection) of a moveable in-core detectors. ~~Include setpoints~~ (1.0)

Answer(s)

3.5 The principle of detection of a fission chamber.

Reference(s)

1. System Descriptions #28, "Incore Flux Monitoring System," SONGS 2&3.

2. Lesson Plans and System Descriptions #35, "Incore Instrumentation System," SL 1&2.

Points Available

3.6 What is the difference between the MSIS actuation signals at Unit 1 and 2? Include setpoints. (1.5)

Answer(s)

3.6 For Unit 1 - MSIS is actuated at 600 psia

* For Unit 2 - MSIS is actuated at ~~5~~ 600 psia or 5 psig containment pressure

Reference(s)

1. St. Lucie Exam #3, Question 1, SL 1&2.

* 2. Tech-specs, SL 2.

Points Available

* 3.7 For ~~each of~~ Unit~~s~~ 1 ~~and 2~~, there are 3 HPSI pumps.

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

b. What is the primary water supply for the HPSI pumps and what is or what should be the available volume of water? (1.0)

c. Draw a one line diagram showing one HPSI pump and all valves and measurement transducers between the water supply(s) and the RCS. Circle the valve(s) that open (or close) on a SIAS. (3.0)

Answer(s)

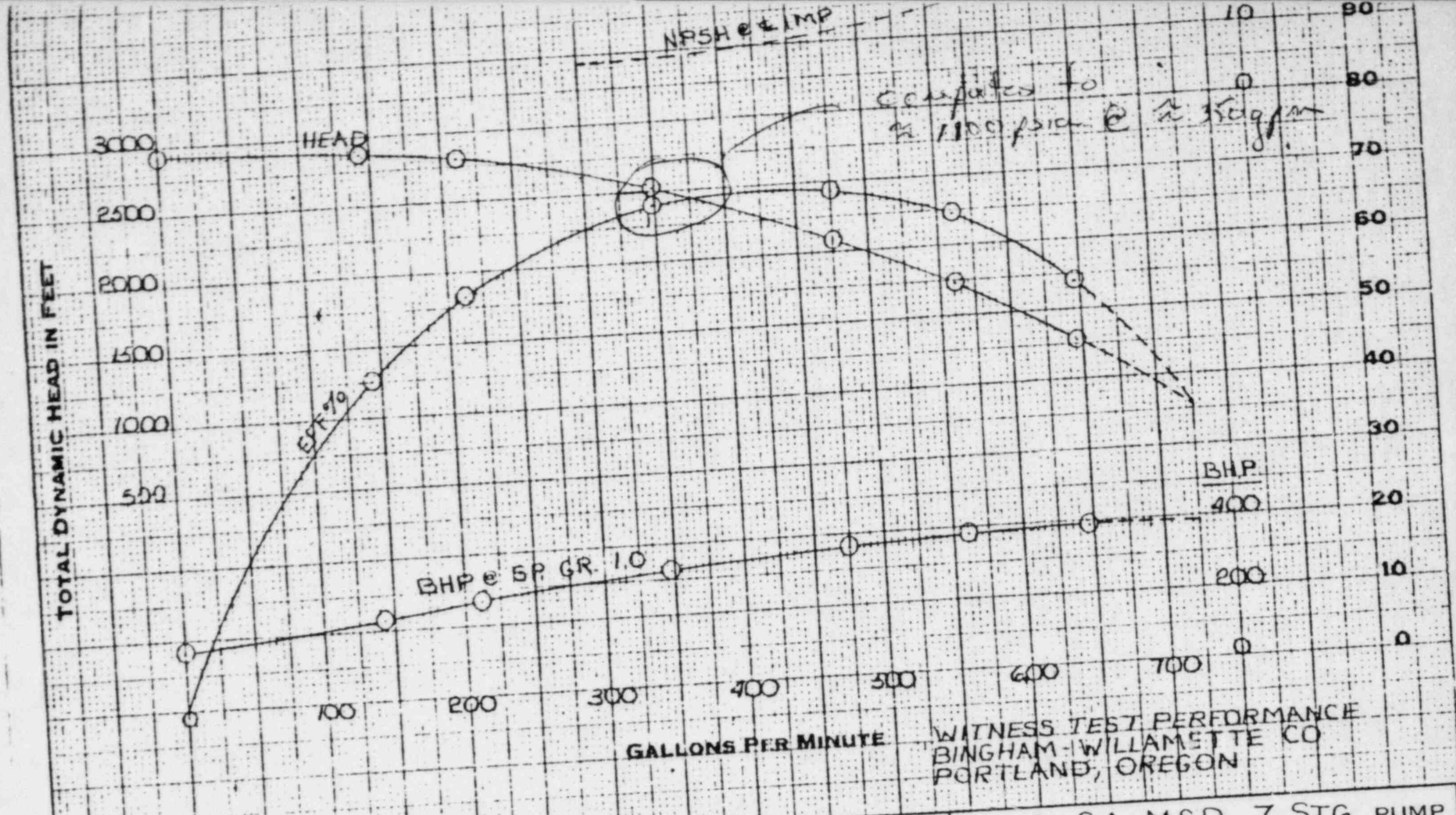
* 3.7 a. 1~~6~~100 psig (+0.5) and 3~~48~~50 gpm (+0.5). See the attached characteristic curve.

3.7 b. The refueling water tank, RWT, (+0.5) which has a 558,000 gal. volume (+0.5). Tech-Specs require 401,800 gal. (+0.5 instead of actual volume).

* 3.7 c. See the attached sketches for Units 1 and 2. The question refers to Unit 1.

Reference(s)

1. Lesson Plans and System Descriptions #4, "HPSI System," SL 1&2.



COMBUSTION ENG'G INC FLORIDA POWER & LIGHT CO CUST'S P.O. 9000650 PUMP No 200114	CHARACTERISTIC CURVE SHEET BINGHAM PUMP DIVISION BINGHAM-WILLAMETTE COMPANY PORTLAND OREGON & SHREVEPORT LA C.H 9-22-71	IMPELLER MAX DIA 9 3/4 MIN DIA EYE AREA 11.9 SQ IN	3X4X9A MSD 7 STG PUMP DIA IMPELLER 9 3/4 NPSH REQUIRED	IMPELLER PATT. 1) 313MSD-C 2-4) 313 CPB-112 5-7) 313 MSD-495 REFERENCE
				3586 R.P.M. CURVE No 29950

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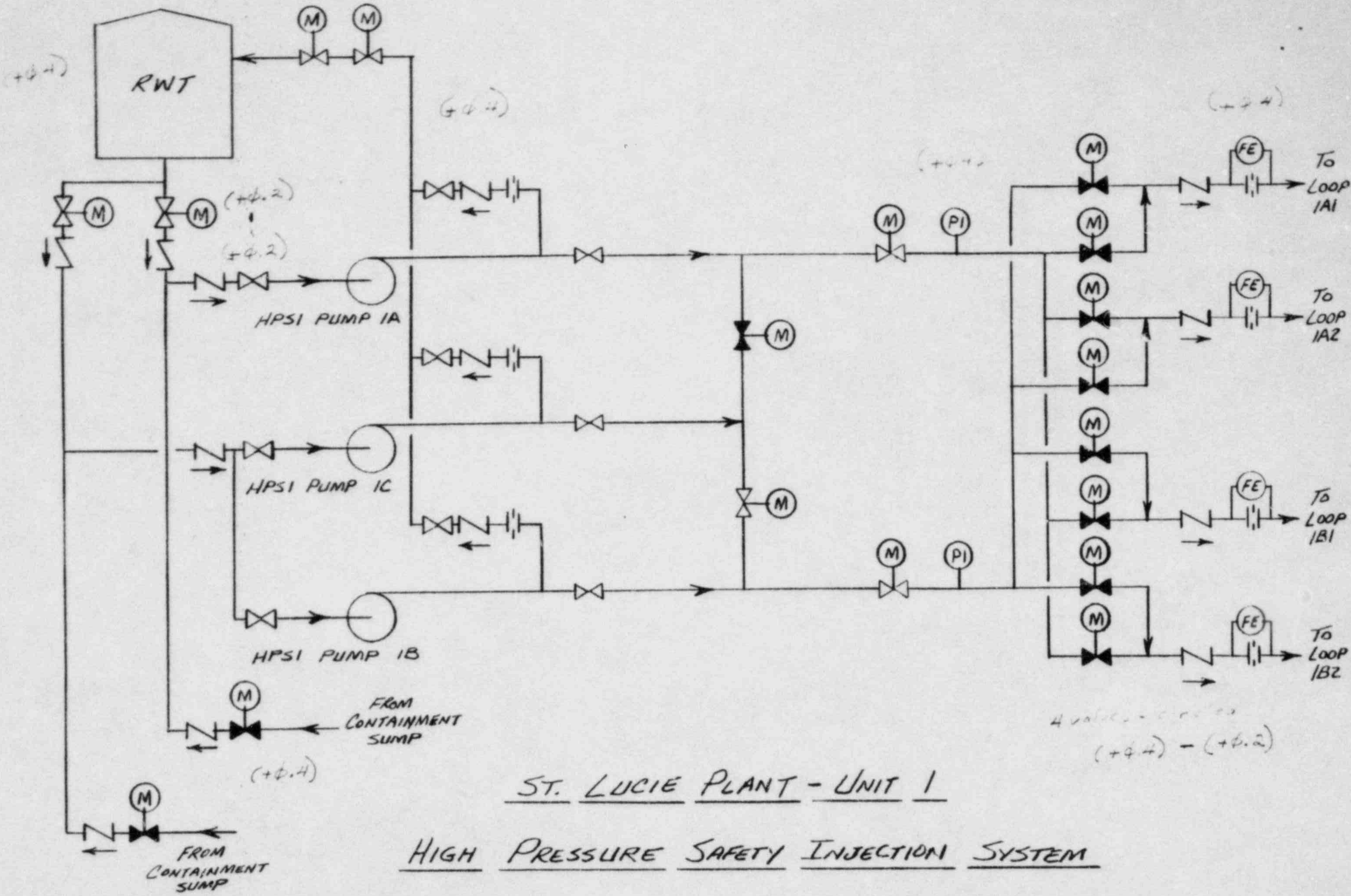
Quest. 3.7

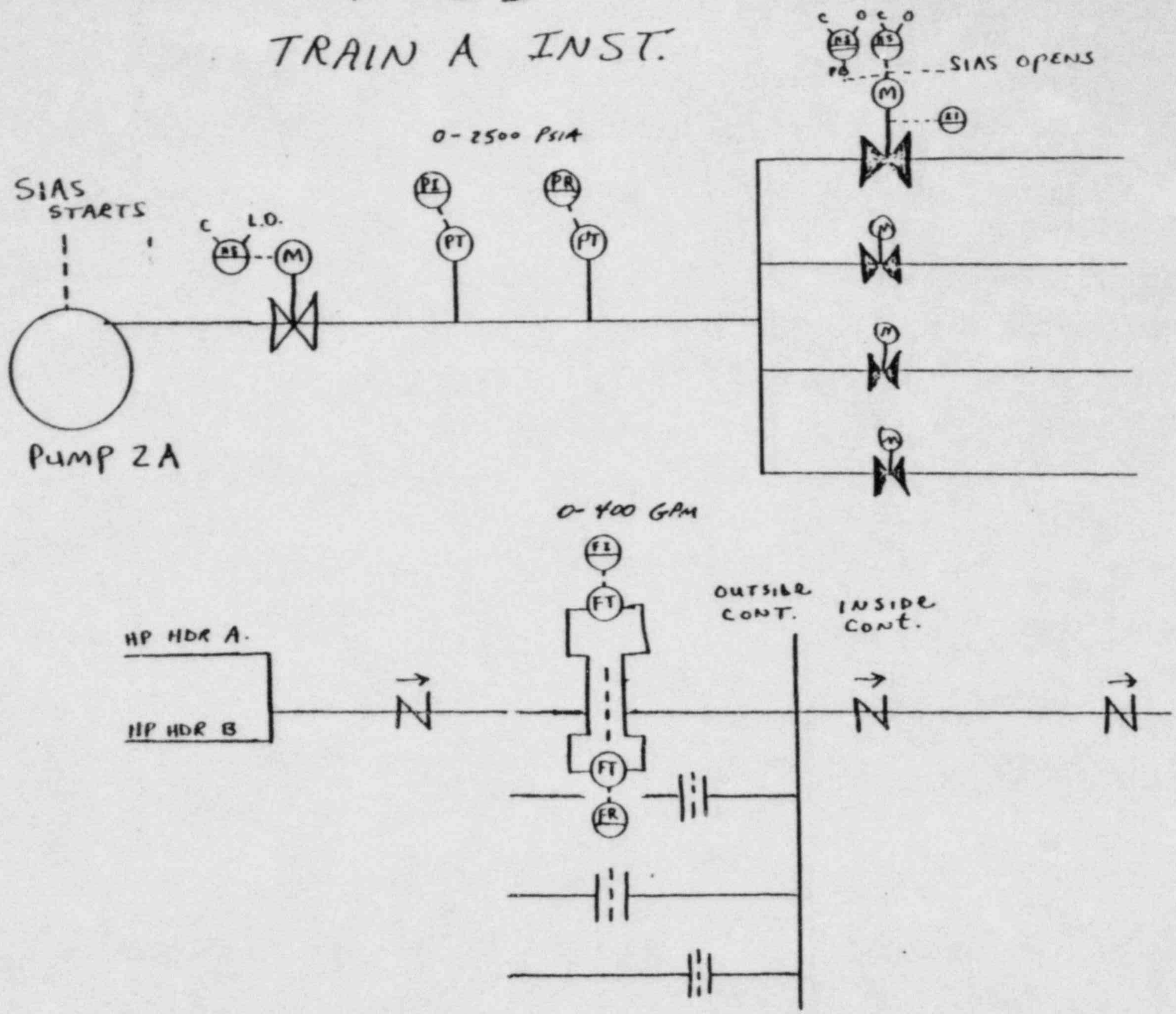
Quest. 3.7 *



ST. LUCIE PLANT - UNIT 1

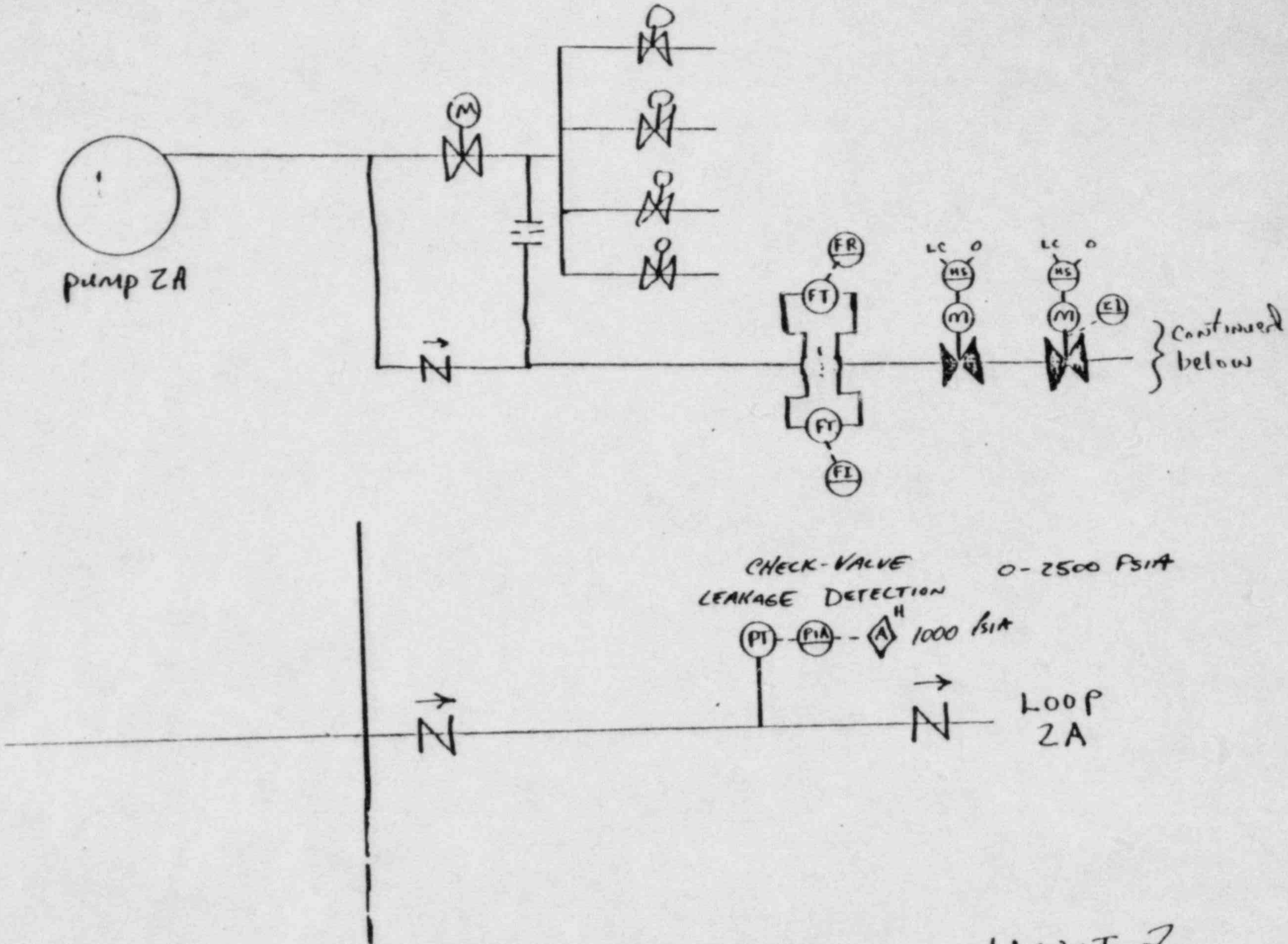
HIGH PRESSURE SAFETY INJECTION SYSTEM

HPSI
TRAIN A INST.



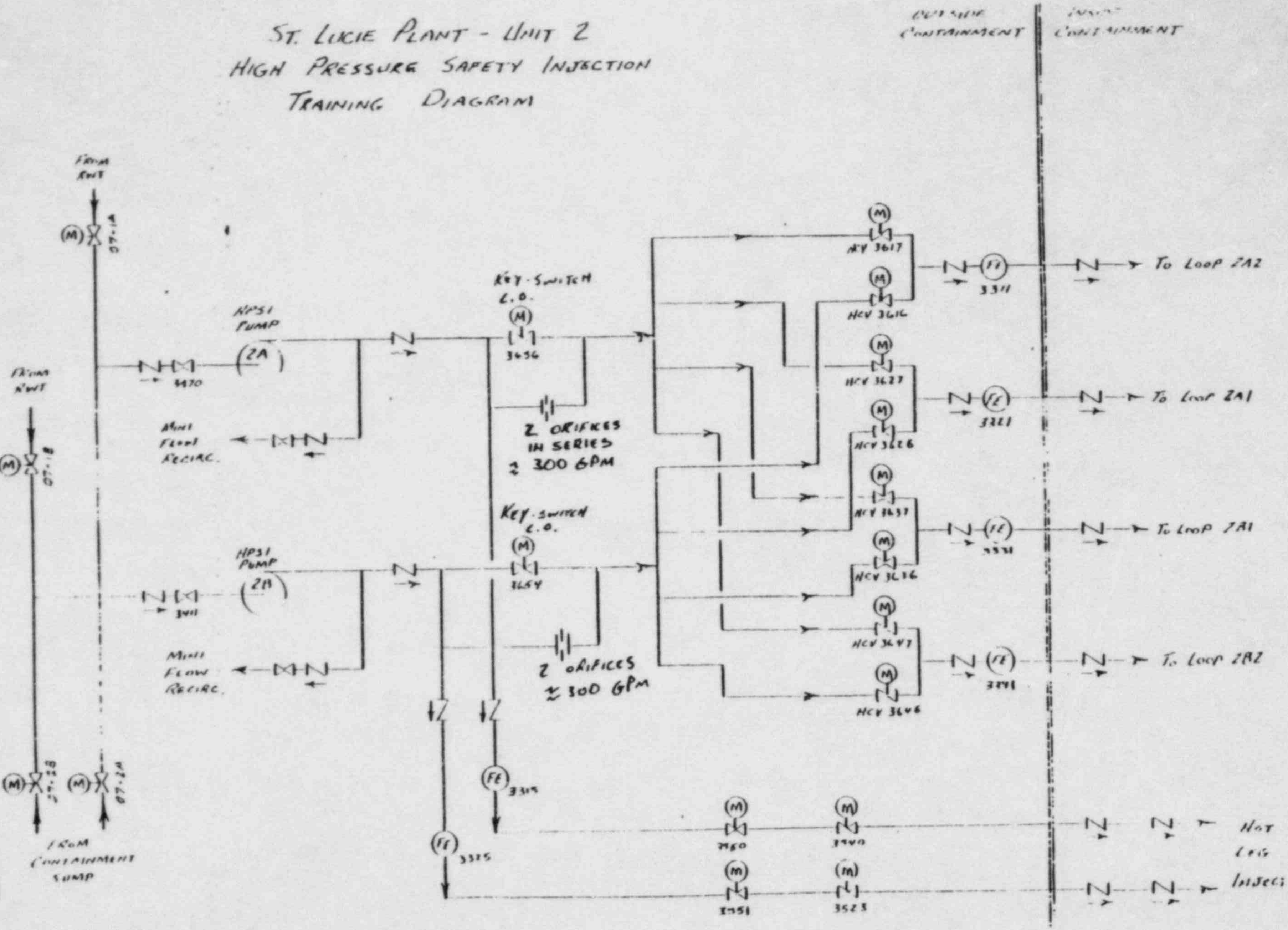
UNIT 2.

TRAIN A HOT LEG INJECTION. INSTRUMENTATION.



UNIT 2.

St. Lucie Plant - Unit 2
High Pressure Safety Injection
Training Diagram



Points Available

3.8 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 1% of full power, CVCS controllers are in automatic and charging pump #2 and #3 are the first and second backup pumps respectively.

Condition #1 - pressure 2270 psig and decreasing
level - 4.0% and increasing

Condition #2 - pressure 2290 psig and increasing
level + 4.5% and increasing

Condition	Spray Valve(s)	Prop Htrs	Backup Htrs	Letdown Valve(s)	CP #1	CP #2	CP #3	
#1	________	____	______	________	_____	_____	_____	(1.5)
#2	________	____	______	________	_____	_____	_____	(1.5)

Answer(s)

3.8

Condition	Spray Valve(s)	Prop Htrs	Backup Htrs	Letdown Valve(s)	CP #1	CP #2	CP #3
#1	Closed (0.3)	On (0.3)	Off (0.1)	Min (0.1)	On (0.1)	On (0.3)	On (0.3)
#2	Closed or Open ~~Closed~~ (0.3)	On (0.3)	On (0.3)	Open (0.3)	On (0.1)	Off (0.1)	Off (0.1)

either answer is acceptable

Reference(s)

1. System Description #39, "Pressurizer and Pressurizer Control System," SONGS 2&3.

2. Lesson Plans and System Descriptions #

3. Off-Normal Operating Procedure 2-0120035, Revision 5, "Pressurizer Pressure and Level Off-Normal Operation", pp. 2, SL 2. (Attached)

Ques. 6.7

2

ST. LUCIE UNIT NO. 2
OFF-NORMAL OPERATING PROCEDURE NUMBER 2-0120035, REVISION 5
PRESSURIZER PRESSURE AND LEVEL-OFF-NORMAL OPERATION

5.0 INSTRUCTIONS:

5.1 Immediate Automatic Actions:

1. Abnormal Pressurizer Pressure Condition.

   A. Pressurizer safety valves open at 2500 psia.

   B. High pressure reactor trip and power operated relief valves open at 2370 psia.

   C. High pressure alarm actuates at 2340 psia and a back-up signal will de-energize all pressurizer heaters.

   D. Proportional heaters cycle from minimum output at 25 psi above setpoint to maximum output at 25 psi below setpoint.

   E. Spray valves cycle from full closed at 25 psi above setpoint to full open 75 psi above setpoint. Set point = 2250

   F. Back-up heaters energize at <2200 psia and de-energize at >2220 psia.

   G. Low pressure alarm actuates at 2100 psia.

   H. TM/LP reactor trip initiates at 1887 psia minimum pressure.

   I. SIAS initiates at 1736 psia. /R5

2. Abnormal Pressurizer Level Condition.

   A. All Pressurizer heaters de-energize at 27% indicated level, and respective Pressurizer Heater Transformer feeder breaker opens.

   B. Low level alarm actuates and a backup signal to start the back-up Charging Pump is received at 5% below RRS setpoint.

NOTE
Only one back-up Charging Pump is in the level control system.

   C. The back-up Charging Pump receives a signal to start at 3% below RRS setpoint, decreasing.

   D. The back-up Charging Pump receives a signal to stop at 1% below RRS setpoint, increasing, and letdown flow decreases to minimum (29 gpm).

   E. All back-up heaters energize and a back-up stop signal to the back-up Charging Pump is received at 4% above RRS setpoint.

   F. Maximum letdown is 128 gpm at 9% above RRS setpoint.

   G. High level alarm actuates at 10% above RRS setpoint.

Points Available

3.9 While operating at a steady 100% of full power, the reactor operator causes an inadvertent emergency boration. T_c drops 10°F before he/she recognizes the situation. Explain the response of the steam generator and of the Feedwater Regulating System to the inadvertent emergency boration event. (3.0)

Answer(s)

3.9
- The heat transfer rate to the secondary fluid is reduced. (+1.0)
- 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 regulating valve will throttle down. (+0.5)
- Due to the drop in the steam-generator pressure, the steam flowrate will decrease.
- Due to the reduced steam flowrate the feedwater regulating valve will tend to throttle down. (+0.5)
- The flowrate mismatch will bring the steam-generator water level down. (+0.5)

(+3.0 max)

Reference(s)

1. NRC Question Bank, Section 4, Number 17, SL 1&2.

- End of Section 3.0 -

Points Available

4.0 PROCEDURES: NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL (25)

4.1 A main steam line break occurs while operating at 100% power operation. List the immediate Operator actions that take place. (2.0)

Answer(s)

4.1 ~~3.1 - 3.9~~ 4.1 - 4.6 of the given reference, p. ~~8~~ 7 ~~(±0.25 each) (+2.0 max)~~ .22

6 answers

(+2.0 max)

Reference(s)

1. EP 2-0810040, Revision 2, SL 2. (Attached)

2

3.0 IMMEDIATE AUTOMATIC ACTION:

AUTOMATIC ACTION	INITIATING EVENT
3.1 Reactor trip.	3.1 S/G pressure <626 psia
3.2 Turbine trip.	3.2 Reactor trip bus low voltage
3.3 Generator lock-out.	3.3 Turbine trip
3.4 Transfer from Auxiliary to Start-up transformers.	3.4 Generator lock-out
3.5 MSIS.	3.5 S/G pressure <600 psia or Containment pressure >5 psig.
3.6 SIAS.	3.6 RCS pressure <1736 psia or Containment pressure >5 psig.
3.7 CIAS.	3.7 Containment pressure >5 psig or Containment radiation ≥10 R/HR or from SIAS actuation.
3.8 CSAS.	3.8 Containment pressure >9.3 psig concurrent with SIAS.
3.9 AFAS (feeds only the non-faulted S/G).	3.9 S/G level <20.6%

/R2

4.0 IMMEDIATE OPERATOR ACTION:

ACTION	NOTES
4.1 Carry out immediate operator actions for a reactor trip per OP 2-0030130, "Reactor Trip/Turbine Trip."	
4.2 Ensure SIAS actuates if conditions require.	4.2 SIAS actuates on: RCS pressure <1736 psia Containment pressure >5 psig
4.3 If SIAS actuates on low RCS pressure, verify CEAs inserted > 5 sec., then stop all Reactor Coolant Pumps.	
4.4 Ensure MSIS actuates if conditions require.	4.4 MSIS actuates on: S/G pressure <600 psia Containment pressure >5 psig
4.5 Determine affected S/G.	4.5 Observe S/G pressures and levels
4.6 Ensure AFAS is establishing flow <u>only</u> to the non-effected S/G.	

/R2

"Natural Circulation/Cooldown," **Points Available**

4.2 As stated in EOP 2-0120040, the fill and drain method of cooling the RCS is an alternative to natural circulation in the event of loss of condensate capacity. Explain how this method is accomplished. (2.0)

Answer(s)

4.2 Appendix ~~B~~ D attached. (+2.~~5~~0)

Reference(s)

1. EOP 2-0120040, "Natural Circulation/Cooldown," Revision 4, SL 2.

2

APPENDIX D

RCS FILL AND DRAIN METHOD OF COOLING

REACTOR VESSEL HEAD REGION

NOTE
This method of RCS cooldown should only be employed in the event that rapid de-pressurization of the RCS is required, or Condensate Storage Tank level decreases below minimum required by Tech Specs.

CAUTION
DURING THIS EVOLUTION, PRESSURIZER LEVEL IS NOT A VALID INDICATOR OF RCS INVENTORY DURING TRANSIENT CONDITIONS. CARE SHOULD BE EXERCISED TO OBSERVE OTHER PARAMETERS WHICH WOULD INDICATE ANY LOSS OF RCS INVENTORY.

1. Take manual control of the charging and letdown system.

2. Lower RCS pressure by using auxiliary sprays into the Pressurizer.

3. As voiding occurs in the upper reactor vessel head, a surge of water from the RCS will cause Pressurizer level to increase rapidly. Terminate auxiliary spray prior to Pressurizer level increasing to 70% indicated level.

4. Cool the upper reactor vessel head region by charging with a Charging Pump to the RCS loop(s). Continue charging until either of the following conditions occur:

   4.1 Pressurizer level decreases to 30% indicated level

   OR

   4.2 The upper reactor head is charged solid.

NOTE
A solid upper head condition will be evident by an increasing Pressurizer level as charging to the loops is continued.

5. Repeat steps 1 through 4 above until SDC entry conditions are established.

NOTE
If the above were to prove unsuccessful, Pressurizer heaters may be used (if sufficient volume is available) to heat up the pressurizer and remove a vessel head void. This strategy should be used only as a last resort and will take an hour or more to be successful.

Points Available

4.3 Define the following from HP-2 procedure:

a. Radiation Area (0.5)

b. High Radiation Area (0.5)

c. Contaminated Area (0.5)

Answer(s)

4.3 a. 5.1.5 (~~+1.0~~ +0.5)

4.3 b. 5.1.6 (~~+1.0~~ +0.5)

4.3 c. 5.1.9 (~~+1.0~~ +0.5)

Reference

1. HP-2, Revision 0, SL 2. (Attached)

Unescorted access to the Radiation Controlled Area is limited to those individuals who have completed the Radiation Protection Training Program (see section 3.3.3) and are authorized by the Plant Manager, or Operations Superintendent. Individuals not receiving radiation protection training may enter the Radiation Controlled Area when escorted by an authorized employee.

5.1.4 Hot Spot Areas are areas on pipes and/or equipment, located in accessible areas that are reading more than ten (10) times the general area radiation level (i.e., 18 inches from contact), but not less than 100 mr/hr.

5.1.5 Radiation Area is any area, accessible to personnel, in which there exists radiation at such levels that a major portion of the body could receive in any one hour a dose in excess of 5 millirem, or in any 5 consecutive days a dose in excess of 100 millirem.

5.1.6 High Radiation Area is any area, accessible to personnel, in which there exists radiation at such levels that a major portion of the body could receive in any one hour a dose in excess of 100 millirem.

5.1.7 Airborne Radioactivity Area is any area in which airborne radioactive materials exist in concentrations in excess of the limits for restricted areas specified in 10 CFR 20, Appendix B, Table I, Column I; or any area in which concentrations exist which averaged over the number of hours in any week during which individuals are in the area, exceed 25 percent of the amounts specified in 10 CFR 20, Appendix B, Table I, Column I.

5.1.8 Radioactive Material Area is any area which contains radioactive material in excess of ten times the quantities of material specified in 10 CFR 20, Appendix "C".

5.1.9 Contaminated Area is any area which contains transferable surface radioactive contamination in excess of 1000 dpm/100 cm^2 β-γ averaged over a major portion of the area.

5.1.10 Locked High Radiation Area is any area accessible to personnel in which there exists radiation at such levels that a major portion of the body could receive in any one hour a dose in excess of 1000 millirem.

In accordance with

Points Available

4.4 a. Who has the authority to release a clearance if it is impossible to contact the individual that holds the clearance? (1.0)

b. List all of the individuals by title <u>who</u> can authorize clearances? (1.0)

Answer(s)

4.4 a. The available supervisor having jurisdiction over that circuit or piece of equipment. (+1.0)

4.4 b. NPS, ANPS, or NWE. (+1.0)

Reference

1. OP-0010122, Revision 21, Section 5.11, SL 1&2.

2. OP-0010122, Revision 21, Section 8.1, SL 1&2.

Points Available

4.5 Answer **True** or **False**. (From OP-0010122) The operator need not have the equipment clearance order with him when executing or releasing a clearance. (0.5)

Answer(s)

4.5 False (+~~4~~0.5)

Reference(s)

1. OP-0010122, Revision 21, Section 4.10, SL 1&2.

Points Available

4.6 Answer True or False. (From OP-0010122) An air-operated valve that fails open shall not be considered closed unless it is jacked closed with an installed jacking device. (0.5)

Answer(s)

4.6 True (+0.5)

Reference(s)

1. OP-0010122, Revision 21, SL 1&2.

Points Available

4.7 Operating procedure 2-0030124 has a statement that says "Operation of the turbine at low frequencies is to be avoided." Explain why this precaution is imposed and what the consequences are of operating with degraded turbine frequency. (1.5)

Answer(s)

1.0 ~~0.75~~

4.7 Blade resonance is the problem (+0.5).

0.5 ~~0.75~~

Accumulative lifetime limit is affected (+0.5). turbine damage

Reference(s)

1. OP-2-0030124, "Turbine Startup, Zero to Full Load," p. 2, SL 2.

1.0

Points Available

4.8 Operating Procedure 2-1010020 has a limitation that states "instrument air pressure should be maintained above 85 psig. What component is responsible for that limit? (1.0)

The operability of

Answer(s)

4.8 MSIVs (+1.0).

Reference(s)

1. OP 2-1010020, Revision 3, "Instrument Air System," SL 2.

Points Available

4.9 Operating Procedure 2-070002ø(Condensate and Feedwater Operation System-Normal Operation) has a precaution that states "do not operate two condensate pumps in parallel under low or zero flow conditions." Explain why this is imposed. (1.5)

Answer(s)

4.9 See the attached reference. (~~1.0~~ +1.5)

Reference(s)

1. OP 2-0700020, Revision 2, "Condensate and Feedwater Operation," Section 4.4, SL 2. (Attached)

12 5

2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE UNIT NO. 2
OPERATING PROCEDURE 2-0700020
REVISION 2

1.0 TITLE:

CONDENSATE AND FEEDWATER SYSTEM OPERATION - NORMAL OPERATION

2.0 REVIEW AND APPROVAL:

Reviewed by Facility Review Group ______ October 15, 1982

Approved by C. M. Wethy Plant Manager October 15, 1982

Revision 2 Reviewed by FRG ______ 1-26 1984

Approved by [signature] Plant Manager 2-13-1984

3.0 PURPOSE:

This procedure provides instructions for valve operation required to supply heated feedwater in ample quantity and required quality to the Steam Generators for power production.

4.0 LIMITS AND PRECAUTIONS:

4.1 The respective 2A or 2B Condensate Pump motor breaker must be open before operating the 2C Condensate Pump transfer switch.

4.2 The Condensate Pump discharge valve must be closed or less than ten handwheel turns open when a Condensate Pump is started and the system is not pressurized.

4.3 The minimum recirculation flow for a Condensate Pump is 2500 GPM. To avoid damage to the pump, do NOT operate a Condensate Pump at minimum flow for longer than two hours.

4.4 Do not operate two Condensate Pumps in parallel under low or zero flow conditions. Operation of both pumps under these conditions will cause one pump not to meet its minimum flow requirements.

4.5 Although seal water is normally supplied from an orificed line from the pump discharge, Condensate Storage Tank supplied seal water should be lined up as a backup supply.

4.6 Do NOT start Condensate Pump motors more than three (3) times successively from ambient temperature. For subsequent starts at rated temperature, allow 20 minutes of running time or 40 minutes at stop.

"Charging and Letdown" **Points Available**

4.10 What precaution in OP 2-0210020 "Charging and Letdown" must be taken when placing the standby ion exchanger in service and why is this necessary? (1.0)

Answer(s)

4.10 See the attached reference. (+1.0)

Reference(s)

1. OP 2-0210020, "Charging and Letdown" Revision 7, Section 4.6, SL 2. (Attached)

paraphrasing is allowed
H

2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE UNIT 2
OPERATING PROCEDURE NUMBER 2-0210020
REVISION 7

1.0 TITLE:

CHARGING AND LETDOWN - NORMAL OPERATION

2.0 REVIEW AND APPROVAL:

Reviewed by Facility Review Group ____ March 30 1982

Approved by C. M. Wethy Plant Manager March 30 1982

Revision 7 Reviewed by FRG 1-26 1984

Approved by [signature] Plant Manager 2-13-1984

3.0 PURPOSE:

This procedure provides instructions for the operation of the Charging and Letdown System (CVCS) including the purification section.

4.0 LIMITS AND PRECAUTIONS:

4.1 Explosive mixtures of hydrogen and air in the Volume Control Tank (VCT) shall be avoided at all times. The oxygen concentration shall be maintained less than 2 percent by volume.

4.2 The temperature of the reactor coolant downstream of the Letdown Heat Exchanger should be maintained less than 140°F.

4.3 To avoid operation of RV-2115 (VCT relief) due to the accumulation of non-condensable gases, the VCT should be vented before the pressure approaches 65 psig (high pressure alarm setpoint).

4.4 The charging and letdown systems should be started and stopped simultaneously to minimize pressure and temperature transients in the Charging and Letdown System.

4.5 Letdown flow should be maintained below 135 gpm (high flow alarm setpoint).

4.6 When placing the standby Ion Exchanger (IX) in service, care should be taken to ensure the resin bed has been borated to closely match Reactor Coolant System (RCS) boron concentration to prevent an inadvertent positive reactivity insertion. A new resin bed will remove boric acid from the coolant water as the anion resin changes from the hydorxyl form to the borate form.

4.7 When in Modes 1, 2, 3 and 4, two Charging Pumps shall be operable.

4.8 Minimum NPSH of the Charging Pumps is 9 psia. Low pump suction pressure trip is at 10 psia.

Points Available

4.11 OP 200250030, (Emergency Boration) Requires three general conditions for emergency boration. List these conditions and provide one indication/condition for each. (1.5)

Answer(s)

4.11 See the attached reference. (+1.5)

Reference(s)

1. OP 2-0250030, Revision 2, pp. 1-2, SL 2. (Attached)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE UNIT 2
OFF-NORMAL OPERATING PROCEDURE NO. 2-0250030
REVISION 2

2

1.0 TITLE:

EMERGENCY BORATION

2.0 REVIEW AND APPROVAL:

Reviewed by Facility Review Group February 9 1982

Approved by C. M. Wethy Plant Manager February 15 1982

Revision 2 Reviewed by FRG 8-24 1983

Approved by C M Wethy 10-20- 1983

3.0 PURPOSE AND DISCUSSION:

This procedure provides instructions for the injection of concentrated boric acid solution into the Reactor Coolant System (RCS) via the Charging Pumps.

In the event that normal charging flow is unavailable, flow can be directed to the Auxiliary HPSI header from the discharge of the Charging Pumps.

The Boron Concentration Control System is lined up to automatically emergency borate the RCS on a Safety Injection Actuation Signal (SIAS). When shutdown margin has been confirmed or the SIAS signal reset, it is desirable to restore the Boron Concentration Control System to the automatic make-up mode, or the Refueling Water Tank (RWT) to the suction of the Charging Pumps to prevent overborating.

4.0 SYMPTOMS:

Any one of the following conditions requires emergency boration:

4.1 Unanticipated or uncontrolled RCS cooldown following a reactor trip as indicated by:

1. Reactor Low Tave-Tref alarm
2. Decreasing reactor coolant wide range temperature indication
3. Uncontrolled decrease of Pressurizer level or pressure
4. Uncontrolled decrease in steam pressure

FOR INFORMATION ONLY
This document is not controlled. Before use, verify information with a controlled document.

2

4.0 SYMPTOMS: (Cont.)

4.2 Unexplained or uncontrolled reactivity increase as indicated by:

1. Abnormal Control Element Assembly insertion

2. Abnormal increase in reactor coolant temperature, Tave or reactor power

3. Abnormal increase in reactor power or count rate when shut down

4.3 Loss of Shutdown Margin due to excessive Control Element Assembly insertion as indicated by:

1. Power dependent insertion (data processor) alarm

2. Power dependent insertion (ADS) alarm

Points Available

4.12 At 100% equilibrium power a S/G tube ruptures. List six (6) ~~to~~ (9) control room indications/alarms that will occur as a result of this event and explain what automatic actions should occur. (3.0)

0.5

Answer(s)

4.12 See the attached reference. (+0.5 each)

Reference(s)

1. EOP 2-0120041, Revision 6, pp. 2-4, SL 2. (Attached)

same for both units — SIAS setpoints are diff

ST. LUCIE UNIT 2
EMERGENCY PROCEDURE NUMBER 2-0120041, REVISION 6
STEAM GENERATOR TUBE RUPTURE

2

1.0 SCOPE:

This procedure provides operator instruction for two conditions:

1.1 S/G tube leak less than charging pump capacity (Reactor shutdown-SIAS not received).

1.2 S/G tube leak greater than charging pump capacity (SIAS received). The procedure leaves the RCS in a cold shutdown condition and the affected Steam Generator (S/G) isolated.

2.0 SYMPTOMS:

NOTE
These symptoms are alike for both a large and small leak.

2.1 Unique to this incident:

1. S/G Blowdown Monitor Alarm.
2. Condenser Air Ejector Alarm.
3. Main Steam Line Monitor Alarm. R6

2.1 Radiation monitoring system

NOTE
Any or all of the following may be evident due to a tube failure.

2.2 Decreasing Pressurizer level.

2.2 Indications
Backup Charging pump starts
Pressurizer heaters de-energize

Alarms
H-17, H-18, H-29
H-30

2.3 Decreasing Pressurizer pressure.

2.3 Indications
Backup Pressurizer heaters energize

Alarms
H-9, H-10, H-1, H-2, H-3, H-4

2.4 Initial increase in affected S/G level followed by return to programmed level.

2.4 Dependent on size of tube leak

Indications
LR-9011, 9021

2

2.0 SYMPTOMS: (continued)

2.5 Feed flow < steam flow on affected S/G.	2.5 Dependent on size of tube leak Indications FR-8011/9011 FR8021/9021
2.6 Decreasing letdown flow.	2.6 Caused by decreasing Pressurizer level Indications FIA-2202
2.7 Increasing charging flow.	2.7 Will cause VCT level to decrease Indications LIC-2226 FIA-2212 Alarms M-3

2

ST. LUCIE UNIT 2
EMERGENCY PROCEDURE NUMBER 2-0120041, REVISION 6
STEAM GENERATOR TUBE RUPTURE

3.0 AUTOMATIC ACTIONS:

3.1 Pressurizer level controls close letdown throttle valve to minimize letdown flow.	3.1 On both large leak and on small leak
3.2 S/G blowdown and sample valves close on high radiation	3.2 On both large leak and on small leak
3.3 Pressurizer backup heaters energize.	3.3 On small leak only
3.4 Reactor trip from TM/LP (variable).	
3.5 SIAS when RCS pressure decreases to 1736 psia.	
3.6 CIAS from initiation of SIAS.	
3.7 Turbine trip from reactor trip.	
3.8 Feedwater Regulating valves close and 15% bypass valves open to 5% flow position.	/R6
3.9 Pressurizer heaters de-energize on low-low level.	

Points Available

4.13 During blackout conditions with the diesel generator supplying power, how are the pressurizer backup heaters put in service as indicated in OP 2-0120035, "Pressurizer Pressure and Level-Off-Normal Operation"? (1.5)

Answer(s)

4.13 See the attached reference. (+1.5)

Reference(s)

1. OP 2-0120035, Revision 4, "Caution," SL 2. (Attached)
2. "Pressurizer Back-up Heater B1 & B4 - Circuit Description", Ebasco Services Incorporated, May 6, 1983, (Attached) SL 1&2.

2

5.0 INSTRUCTIONS: (Cont.)

5.1 (Cont.)

5.1.1 (Cont.)

5. All back-up heaters energize and a back-up stop signal to the back-up Charging Pump is received at 4% above RRS setpoint. /R4

6. Maximum letdown is 128 gpm at 9% above RRS setpoint.

7. High level alarm actuates at 10% above RRS setpoint.

5.2 Immediate Operator Actions:

5.2.1 Abnormal Pressurizer Pressure.

1. Ensure Pressurizer spray, and Proportional and Back-up heaters are operating properly in automatic. If not, shift spray valve controller to MANUAL and energize or de-energize heaters, whichever is applicable.

2. Ensure Power Operated Relief Valves are closed. If open, isolate by closing V-1476 and/or V-1477 (PORV block valves). Refer to OP 2-0120036, "Pressurizer Relief/Safety Valve-Off-Normal Operation". /R4

3. Ensure SE-02-03 and SE-02-04 (Auxiliary spray valves) are closed. If open, attempt to close using key switch. If still open, stop all Charging Pumps and isolate letdown. Refer to OP 2-0210030, "Charging and Letdown Off-Normal Operation".

4. Ensure pressure anomaly is not caused by a large rate of change of T_{ave}.

CAUTION

DURING BLACKOUT CONDITIONS WITH THE DIESEL GENERATOR SUPPLYING POWER, THE CONTROL BISTABLES FOR THE BACK-UP HEATERS ARE NOT ENERGIZED AND MUST BE BYPASSED.

THEREFORE, IN A BLACKOUT, THE CONTROL SWITCHES ON RTGB-203 MUST BE RESET AND THE KEY SWITCH SELECTED TO PRESSURE OVERRIDE TO REGAIN HEATER CONTROL. NOTE, HOWEVER, THIS WILL ONLY ENERGIZE B1 AND B4 BANKS OF BACK-UP HEATERS.

pressure override
AND
OR
level-pressure override

NOTE

If SIAS has occurred, Pressurizer heaters cannot be re-energized as above until SIAS has been reset.

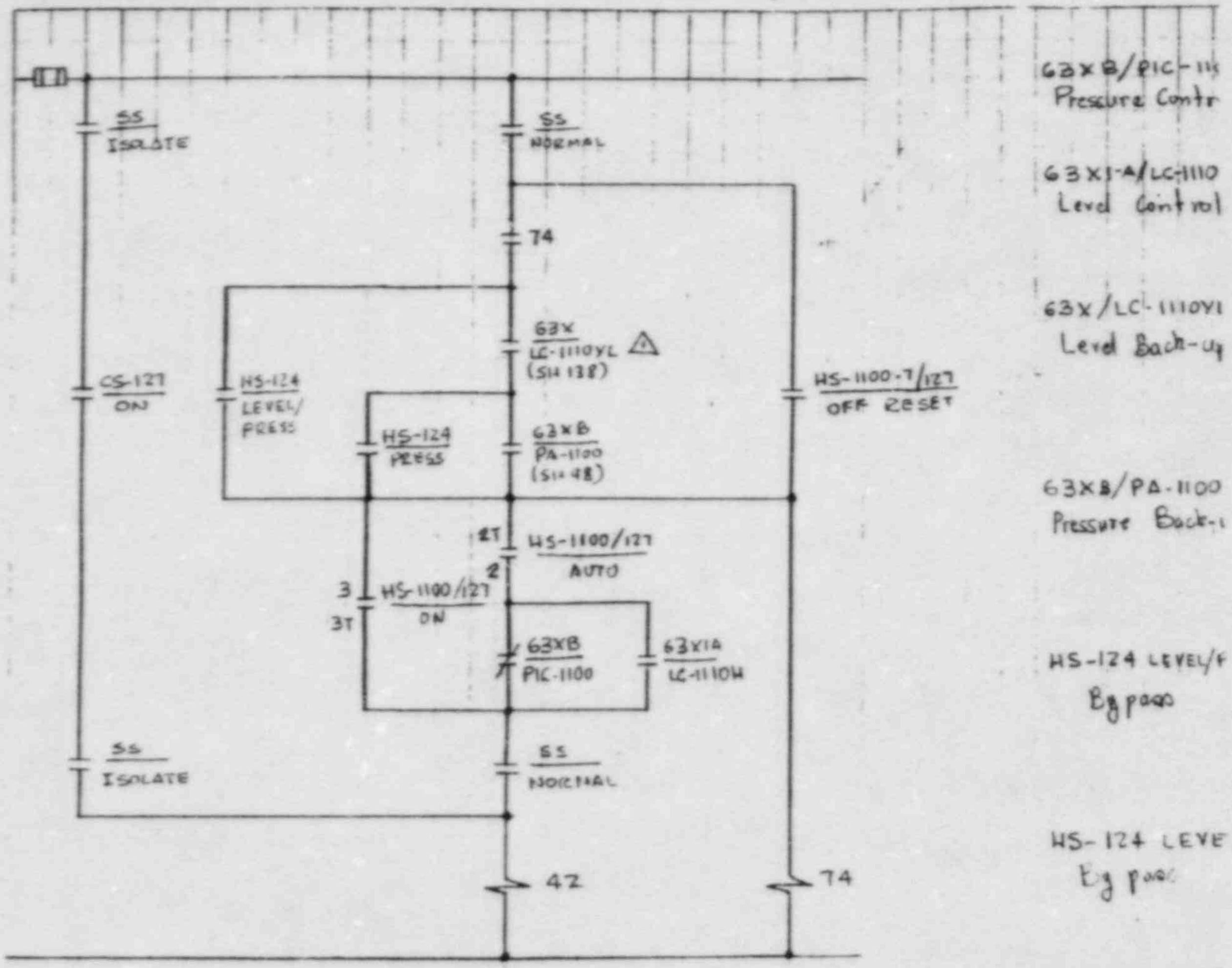
EBASCO SERVICES INCORPORA

Quest ~~3.5~~ 4.13

CLIENT FP&L

PROJECT St Lucie U1 & 2

SUBJECT Pressurizer Back-up heater B1 & B4 - Circuit Description -



HS-124/LEVEL ENERGIZES RELAY 63X/LC-1110YL BYPASSING INPUT FROM LC-1110XL (PRESSURIZER LEVEL CONTROL @ 27%)

Points Available

4.14 In the Off-Normal Procedure No. 2-0030130 (Reactor Trip/Turbine Trip) there is a caution statement that states: "Do not overfeed the S/Gs...accomplished by throttling the MFW 15% bypass or FW valves." Explain the reason why this caution is required. (1.0)

Answer(s)

4.14 This could cause T_{ave} to go below 532°F and apply severe thermal shock to the S/Gs. (+1.~~0~~0) (+0.5 each)

Reference(s)

1. OP 2-0030130, Revision 5, "Reactor Trip/Turbine Trip," p. 9, SL 2.

overcooling

20.5

Points Available

4.15 a. What is the primary purpose of the Radiation Work Permit (RWP) at the St. Lucie plant? (1.0)

b. Identify 3 conditions when a RWP shall be used. (1.0)

c. What are the Florida Power and Light Company guideline values for whole body exposure in 1 quarter? (Assume the individual has a NRC-4 form on file.) (1.5)

Answer(s)

~~4.15 a.~~

4.15 b. Only the underlined portions of Sections 6.3.1.1, 6.3.1.2, 6.3.1.3, 6.3.1.4(1), (2), (3), (+0.33 each, +1.0 max).

~~4.15 b. 7 days (1.0)~~

4.15 c. 500 mrem/~~qtr~~ wk (~~+0.5~~ +1.5) might get 300 mrem/~~qtr~~ wk outage
800 mrem/QTR

~~4.15 d. Actual or anticipate plant transients could change the radiation exposure in the area where the RWP was to be used. (+1.0)~~

Reference(s)

1. HP-2, Sections 6.3, 6.3.1 (attached), 6.1.1.2, 6.1.1.3, SL 1&2.

will provide

worn.

6.2.3.3 Personnel shall wear a minimum of coveralls, cotton gloves (for dry contamination) or rubber gloves (for wet contamination) and shoe covers for any maintenance work on contaminated systems.

6.2.3.4 For jobs requiring a Radiation Work Permit (RWP), the protective clothing requirements for the job shall be specified on the RWP. Personnel entering a RWP area to perform observation and inspection activities only, may wear less than the RWP clothing requirements if so directed by Health Physics.

6.3 Radiation Work Permits

4.15a The primary purpose of a Radiation Work Permit (RWP) is to provide Health Physics with a vehicle whereby they can evaluate and plan jobs in order to maintain radiation exposure ALARA. The Florida Power & Light Company RWP philosophy is based on the fact that control of radiation and contamination is accomplished primarily by training, Health Physics job surveillance, pre-job planning, post-job evaluation, and special instructions. A RWP normally describes the radiological conditions of a job, the protective clothing, monitoring to be performed, and any other special instructions.

6.3.1 RWP Requirements

An RWP shall be required for the following conditions.

4.15b

6.3.1.1 Entry into high radiation areas, airborne radioactivity areas, areas contaminated to levels in excess of 10,000 dpm/100 cm^2, or into any area posted as "RWP REQUIRED FOR ENTRY."

6.3.1.2 Entry into the reactor containment at any time during and subsequent to initial reactor startup.

6.3.1.3 Maintenance or inspection of equipment contaminated in excess of 10,000 dpm/100 cm^2.

6.3.1.4 Work assignments involving changes (withdrawing, uncovering, opening, valving, disassembling, moving) that have the following potential as the work progresses:

1. Exposure of a major portion of the body to a radiation dose in excess of 100 mrem in any one hour.

2. Increasing surface contamination levels to exceed 10,000 dpm/100 cm^2.

3. Increasing airborne radioactivity to values exceeding 25% of those listed or referred to in

6.3.1.5 Health Physics may enter an area without an approved RWP to conduct radiological surveys. Upon certain occasions, the presence of Health Physics may be substituted for an RWP, as specified by procedure HP-1.

6.3.2 Health Physics procedure HP-1, Radiation Work Permit, details requirements for issuing, using, and terminating a RWP.

6.4 General Work Guidelines

6.4.1 Eating, drinking, and smoking shall not be permitted in Contaminated Areas. Eating and drinking shall not be permitted in the Radiation Controlled Area except in designated areas when approved by the Health Physics Supervisor and under the control of special instructions. The special instructions shall be in the form of a written order or posted sign, which when followed, provides reasonable assurance that radioactive materials will not be ingested.

6.4.2 Appropriate personnel monitoring devices as specified by Health Physics shall be worn at all times. Pocket dosimeters should be read at periodic intervals.

6.4.3 Personnel should report any unsafe radiological condition to Health Physics.

6.4.4 Personnel working in a High Radiation Area shall be provided with an instrument capable of continuous indication of the radiation field or be accompanied by Health Physics.

6.4.5 Protective clothing as specified by the Health Physics Manual, RWP, or Health Physics shall be worn by all personnel in the area.

6.4.6 Personnel leaving the Radiation Controlled Area shall monitor themselves for contamination when leaving the area.

6.4.7 Individuals under 18 years of age shall not be admitted to the Radiation Controlled Area.

6.5 ALARA Program Guidelines

Florida Power and Light Company is committed to ensure that radiation exposure to personnel is kept as low as reasonably achievable (ALARA) by incorporating a formal program into company radiation protection policies. The ALARA concept is not a new idea but has been applied in the company's nuclear program since it's existence. Basically the ALARA concept means that anytime personnel exposure can be effectively reduced without excessive cost, it should be done. This refers to total person-rem exposure for the facility as well as individual exposure. This necessitates a general awareness by all

Ques 4.15

RADIATION ACCUMULATED DOSE LIMITS — FEDERAL (10 CFR 20)

Whole body, head and trunk, blood forming organs, lens of eyes, and gonads.	Without Dose Record	1.25 R/Qtr.
	With Dose Record	3.00 R/Qtr.
Extremities — Hands, forearms, feet and ankles		18.75 R/Qtr.
Skin of Whole Body		7.5 R/Qtr.
Fetal exposure		500 mR/per gestation period

* Shall not exceed lifetime whole body dose of 5 (N-18) Rem where N is age at last birthday.

PLANT GUIDE FOR EXPOSURE RATES

All personnel <u>with</u> a Dose Record	Outage	500 mR/wk.
	Non-outage	300 mR/wk.
All personnel, No Dose Record		250 mR/Qtr.
Females, child bearing age		250 mR/Qtr. (extendable to no greater then 500 mR/2 consecutive months)
All personnel, Incomplete Dose Records, (Extendable to 1100)		800 mR/Qtr.
All personnel with Complete Dose Records, (Extendable to 2750)		800 mR/Qtr.
Annual Dose		5 R/year

Points Available

4.16 ~~When the reactor is shutdown in Hot Standby, both groups ~~or~~ of shutdown CEAs are required to be above the exercise limit. Explain the reason for this requirement.~~ (1.0)

QUESTION WITHDRAWN

Answer(s)

4.16 To prevent erosion of the guide tubes. (+1.0)

Reference(s)

1. OP 2·0030128, p. 1, SL 2.

- End of Section 4.0 -

we will look at it. they would look at it