UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

ENCLOSURE 1

EXAMINATION REPORT

Facility Licensee: Georgia Power Company
P. O. Box 4545
Atlanta, GA 30302

Facility Docket Nos.: 50-321 and 50-366

Facility License Nos.: DPR-57 and NPF-5

Examinations administered at Edwin I. Hatch Nuclear Plant, Baxley, Georgia

Chief Examiner: John F. Munro (signed) — 8/29/84
John F. Munro — Date Signed

Approved by: Bruce A. Wilson (signed) — 8/31/84
Bruce A. Wilson, Section Chief — Date Signed

Summary:

Examinations on July 10-13, 1984

Written, oral, and/or simulator replacement examinations were administered to two SROs (including one instructor certification) and six ROs; both SROs and four ROs passed these examinations.

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PDR ADOCK 05000321
Q PDR

REPORT DETAILS

1. Persons Examined

SRO Candidates:

J. G. Rogers
C. B. Smith (IC)

RO Candidates:

R. C. Bartles
M. R. Davis
F. C. Godfrey, III
R. E. Miller
L. W. Swinson, Jr.
A. D. Yawn

NOTE: "IC" indicates an Instructor Certification.

Other Facility Employees Contacted:

T. Greene, Deputy General Manager (E)
S. Baxley, Superintendent of Operations (E)
D. F. Moore, Training Manager, Hatch (E)
R. S. Grantham, Supervisor Operations Training (R/E)
C. E. Brantley, Senior Simulator Instructor (R/E)
G. W. Neeley, Simulator Engineer (R/E)
L. S. Gooden, Senior Simulator Instructor (R/E)
D. Giddens, Senior Simulator Instructor (R/E)

NOTE: (1) "R" indicates present at examination review.
(2) "E" indicates present at exit meeting.

2. Examiners:

J. Munro, NRC, Chief Examiner
K. Brockman, NRC
R. Persons, EG&G
D. Hill, EG&G

3. Examination Review Meeting

At the conclusion of the written examinations, the license examiners met with facility representatives (identified in (1) above) to review the written examinations and answer keys. Specific facility comments and associated NRC resolution of those comments are as follows:

*a. Question 1.01a - Indicated level is higher than actual level; however, the correct differential is 10" vice 7" as stated in the answer key.

Resolution - E. I. Hatch Nuclear Training, Vol. 5, page 2.3-5 states that the level differential between the inside and outside of the dryer skirt is about 7" at high power conditions, however, page 2.2-8 states that the differential is 10" of water. The facility provided additional references (i.e., GE documents 257-HA-771 and 383-HA-428) which confirm the level differential and the correct answer to be 10". Partial credit will be granted for an answer of 7", which is correct for STP conditions.

*b. Question 2.05a - The Standby Gas Treatment System is actually initiated by refueling floor vent exhaust high radiation signal and not a refueling floor high radiation signal as stated in the answer key and E.I. Hatch Nuclear Training, Vol. 5.

Resolution - Hatch procedures HNP-2-2064 and HNP-2-1903 confirm that the ventilation exhaust high radiation signal does initiate the SBGTS. The answer key has been changed accordingly.

c. Question 2.08a - The Unit 1/Unit 2 Differences Lesson Plan and the Unit 2 and Tech. Spec. bases state that the low low set logic is intended to limit the loads on the containment/torus and the SRV discharge lines.

Resolution - The comment is correct, however, it does not address the possible effect on an SRV discharge pipe if its breaker sticks shut during repeated actuation (opening) of the SRV; i.e., it does not answer the question. No change to the answer key is warranted.

d. Question 3.04 - The question did not specify Unit 1 or Unit 2 so the setpoints for either unit should be accepted for full credit.

Resolution - The comment is accepted as valid and either the Unit 1 or Unit 2 setpoints will be acceptable for full credit. The answer key has been changed to incorporate both Units' setpoints.

e. Question 3.06b - The low low set initiation setpoint per Technical Specification 3.4.2.2/4.4.2.2 is 1054 psig.

Resolution - The specific LSS initiation setpoint is not requested in the question and is not required for full credit. However, either setpoint (i.e., 1044 or 1054 psig) if offered by the candidate, will be acceptable and will not result in a loss of credit.

f. Question 5.05a - This question is the same as question 1-46.a in the E. I. Hatch Question and Answer Bank (Rev. 1/7-84). The wording of the answer to question 1-46.a should be acceptable as an answer to this exam question.

Resolution - Both answers (i.e., the exam answer key for question 5.05a and Hatch Q&A Bank question 1-46.a) describe the same concept using slightly different phraseology. A candidate whose answer demonstrates an adequate understanding of the concept will receive full credit regardless of the phraseology he uses. No change to the answer key is necessary.

* The LPs should be corrected to reflect actual plant conditions.

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 examinations. Those individuals who clearly passed the oral and/or simulator examinations were identified.

The examiners noted that use of procedures by the candidates during simulator examinations was good and thus maintained the improvement established on the last examination at E. I. Hatch Nuclear Plant.

The following generic weaknesses were noted by the examiners during the oral and simulator examinations:

- Some candidates demonstrated an inability to explain the basic theories of Nuclear plant operation e.g. the fission process and the ionization process.

- Some candidates demonstrated an inability to explain "actual" plant conditions e.g. EHC Pressure set at 990 psig per meter reading.

- Some RO candidates did not effectively communicate plant conditions to the SRO during the simulator examinations.

The cooperation give to the examiners was appreciated and noted.

ENCLOSURE 3
(2 of 2)

MASTER COPY

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: HATCH 1&2

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 84/02/10

EXAMINER: PERSONS, R.

APPLICANT: ______________________

INSTRUCTIONS TO APPLICANT:

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

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
25.50	25.50	________	________	1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
24.50	24.50	________	________	2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
25.00	25.00	________	________	3. INSTRUMENTS AND CONTROLS
25.00	25.00	________	________	4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
100.00	100.00	________	________	TOTALS

FINAL GRADE ________________%

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

____
APPLICANT'S SIGNATURE

QUESTION 1.01 (2.50)

Indicated reactor water level at 100% power differs from the actual water level directly above the core.

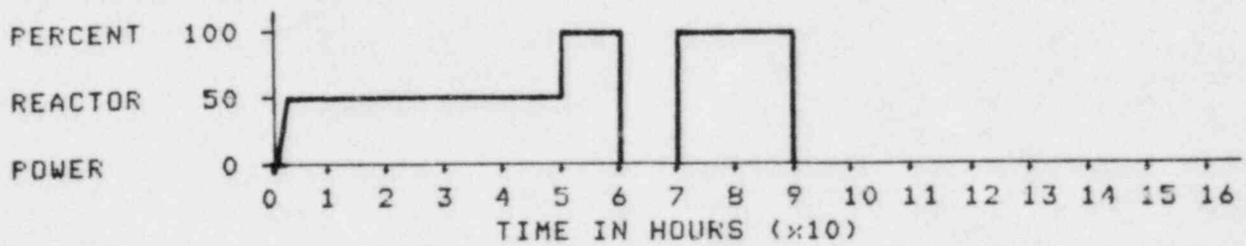
a. WHICH level (actual or indicated) is higher and by HOW MANY inches? (1.0)

b. EXPLAIN WHY the above difference occurs. (1.5)

QUESTION 1.02 (3.00)

For the power history below, SKETCH a curve of core xenon concentration versus time.

NOTE: Time is in increments of TEN (10) HOURS and the core is XENON FREE at time zero.



QUESTION 1.03 (1.50)

A centrifugal pump is operating at 3600 RPM with a pump head of 160 ft. Pump speed is then reduced so that pump head is 100 ft. WHAT is the new pump speed? SHOW ALL WORK.

QUESTION 1.04 (3.00)

With Unit 1 at rated conditions the EHC pressure setpoint (on the controlling pressure regulator) is lowered to its minimum value with the DECREASE pushbutton on the 9-7 Panel. Assuming NO further operator action, answer the following using attached FIGURE 1:

a. WHY does APRM power gradually decrease in AREA 1? (0.5)

b. WHAT is causing total steam flow to be >100% rated flow at POINT 2? (0.5)

c. WHY did total feed flow increase to full scale at POINT 3? (0.5)

d. WHAT caused total feed flow to go to zero at POINT 4? (0.5)

e. WHAT is indicated by the oscillations in the wide range reactor pressure trace (AREA 5)? (0.5)

f. WHY do the peaks in the pressure oscillations occurring in AREA 5 become farther apart with time? (0.5)

QUESTION 1.05 (3.00)

Assume Unit 2 is operating at 100% power and ONE reactor recirc pump trips. HOW will each of the parameters listed below INITIALLY change (increase or decrease)? Briefly STATE ONE (1) REASON WHY the change occurs.

a. Reactor power (1.0)

b. Reactor water level (1.0)

c. Feedwater flow (1.0)

QUESTION 1.06 (2.00)

Regarding delayed neutrons:

a. HOW are they produced? (0.5)

b. As the core ages WHAT happens to the magnitude of the effective delayed neutron fraction (Beta effective) and WHY? (1.0)

c. HOW does the change in Part (b) above affect the core's response time follwing a reactivity change? (0.5)

QUESTION 1.07 (1.50)

The following statements are concerned with subcritical multiplication. CHOOSE ONE of the capitalized words to make each statement true.

a. As K-effective (K-eff) approaches unity, a (LARGER/SMALLER) change in neutron level occurs for a given change in K-eff. (0.5)

b. As K-eff approaches unity, a (SHORTER/LONGER) period of time is required to reach the equilibrium neutron level for a given change in K-eff. (0.5)

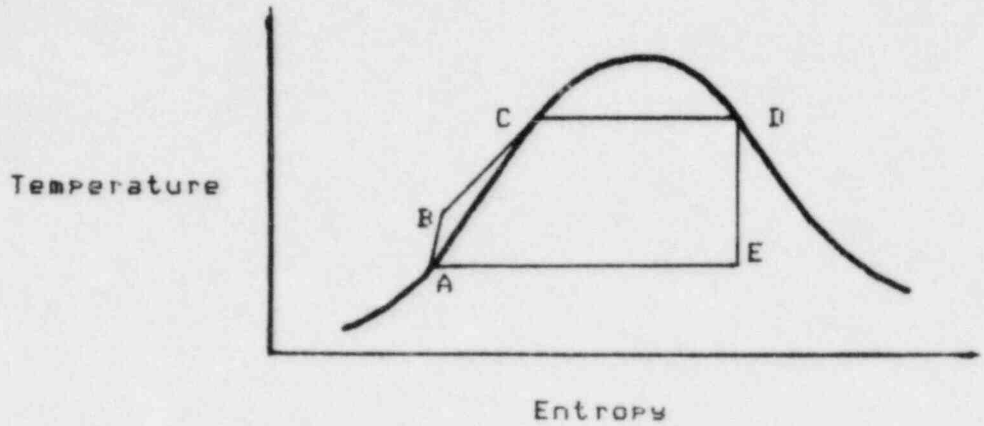
c. As K-eff approaches unity, the count rate doubling technique becomes (MORE/LESS) accurate. (0.5)

QUESTION 1.08 (2.00)

During a reactor startup, while at low power, a control rod is notched out resulting in a stable period of 100 seconds. Neglecting any effects due to heating of the moderator, CALCULATE the time required for reactor power to increase by a factor of TEN (10).

QUESTION 1.09 (2.00)

The BWR system is designed to emulate the Carnot cycle by use of the Rankine vapor cycle sketched below. MATCH the thermodynamic process in COLUMN 1 with the correct curve segment from the sketch listed in COLUMN 2.



COLUMN 1	COLUMN 2
a. Vaporization process	AB
b. Condensate/FW pump pressure increase	BC
c. Condensing process	CD
d. Turbine Expansion	DE
	EA

QUESTION 1.10 (2.00)

EXPLAIN WHAT would happen to core flow DISTRIBUTION on a power increase with NO CHANGE IN MEASURED CORE FLOW if the core fuel bundles were NOT ORIFICED.

QUESTION 1.11 (3.00)

Following a norm[illegible] reduction in power from 90% to 70% with recirculation flow, HOW will the following change (increase, decrease, or remain the same) AND WHY:

a. The pressure difference between the reactor and the turbine steam chest. (1.0)

b. Condensate depression at the exit of the condenser. (1.0)

c. Final Feedwater temperature. (1.0)

QUESTION 2.01 (2.00)

Answer the following for Unit 1 and Unit 2:

a. WHICH main steam line supplies steam to HPCI? (1.0)

b. WHAT is the cooling water supply to the recirc MG set air coolers? (1.0)

QUESTION 2.02 (3.00)

With regard to the Remote Shutdown Panel (RSP):

a. LIST FIVE (5) different systems/components that may be operated from the RSP. Be specific (i.e., 'B RHR loop', 5 OTHERS required). (2.5)

b. YES or NO. WILL 'B' RHR pump start on a valid LOCA initiation signal if it is being controlled from the RSP? (0.5)

QUESTION 2.03 (3.00)

Concerning the Standby Liquid Control System:

a. WHY is it necessary for the system to be capable of injecting the contents of the SLC tank in a MAXIMUM time of 125 minutes? (1.0)

b. WHY is the SLC pump suction piping heat traced? (1.0)

c. WHAT are THREE (3) uses of the SLC injection sparger, OTHER THAN poison injection? (1.0)

QUESTION 2.04 (2.00)

Briefly, EXPLAIN HOW the HPCI System will respond to a valid auto initiation signal if the HPCI DC condensate pump trips on overload one minute after the initiation signal is received? Discuss WHAT specifically happens in the HPCI System assuming NO OPERATOR ACTION and state whether the system will perform its intended function.

QUESTION 2.05 (3.00)

Regarding the Standby Gas Treatment System (SGTS):

a. WHAT are THREE (3) of the four conditions which will auto initiate the system? Setpoints NOT required. (1.5)

b. WHAT will cause the system's deluge fire sprinklers to auto initiate? Setpoint NOT required. (0.5)

c. If the deluge fire sprinklers initiate in an operating SGTS train, WHAT TWO (2) automatic actions should occur in the affected train. (1.0)

QUESTION 2.06 (3.50)

Concerning the Control Rod Drive (CRD) Hydraulic System:

a. Upon completion of a reactor scram with all CRDs fully inserted, WHAT are the TWO (2) sources of water continuing to fill the scram discharge volume until the scram is reset? (1.0)

b. WHAT are TWO (2) possible indications/events resulting from a leaking scram outlet valve? (1.0)

c. WHAT major difference exists between the Unit 1 and Unit 2 CRD pump controls? Briefly, EXPLAIN its function. (1.5)

QUESTION 2.07 (3.00)

Regarding the Residual Heat Removal (RHR) System:

a. With an RHR System aligned for Shutdown Cooling Mode, WHY is it necessary to prevent the RHR pumps' minimum flow valve from opening? (1.0)

b. With the system operating in Shutdown Cooling Mode, HOW will the system be effected if reactor pressure exceeds 135 psig? (1.0)

c. WHAT TWO (2) automatic actions will NOT occur in an RHR loop if the RHR logic is in full test (i. e., both logic circuits in test) when a LPCI initiation signal occurs? (1.0)

QUESTION 2.08 (2.50)

With regard to the Main Steam Safety Relief Valves (SRVs):

a. EXPLAIN HOW/WHY an SRV discharge pipe (tail pipe) could be damaged due to its vacuum breaker STICKING SHUT during repeated actuation (lifting) of the SRV? (1.5)

b. How (INCREASE, DECREASE, REMAINS THE SAME) would Drywell Pressure be expected to respond to an SRV discharge line vacuum breaker STICKING OPEN during actuation of the SRV? Briefly, JUSTIFY your answer. (1.0)

QUESTION 2.09 (2.50)

Assume the plant is operating at 75% power with 'A' & 'B' Condensate pumps and 'A' & 'B' Condensate Booster pumps running. The 'C' Condensate and 'C' Condensate Booster pumps are in standby AUTO.

For each of the conditions below indicate HOW the above configuration would automatically change.

a. A 'Condensate Booster Pumps Suction Low Pressure' alarm is received at 43 psig and suction pressure continues to decrease to 38 psig where the configuration changes with no booster pump trips. (0.5)

b. 'B' Booster pump auxiliary lube oil pump fails to start and the booster pump's lube oil pressure decreases to zero psig. (1.0)

c. A 'Condensate Pump Low Level' alarm is received. (1.0)

QUESTION 3.01 (1.00)

Regarding the Reactor Manual Control System (refer to attached Figure 9.2.1(1)) WHAT is the alignment of the four (4) directional control valves during the 'settle mode?'

QUESTION 3.02 (2.00)

With the plant operating at 100% power, Recirc in 'MASTER MANUAL', an electrical fault causes the load selector input to the EHC system to decrease to 90%. WHAT will be the RESPONSE of the FOLLOWING to this occurence, and WHY WILL THAT RESPONSE OCCUR?

NOTES: (1) Continue your discussion to a stable condition (~ 1 minute after fault).
(2) Assume NO OPERATOR ACTION.
(3) Provide the actual EHC LOGIC COMPONENT which develops the positioning signal to the valves.
(4) EHC Logic diagram, Figure 9.4(7), attached for reference.

a. Turbine CONTROL Valve Position (1.0)

b. Turbine BYPASS Valve Position (1.0)

QUESTION 3.03 (2.00)

Briefly DESCRIBE each of the FOUR different ranges of Reactor Vessel Level Indication in terms of the following:

(1) The NAME of the indicating range,
(2) Its SPAN,
(3) Its ZERO REFERENCE,
(4) Its CALIBRATION TEMPERATURE (Hot or Cold).

QUESTION 3.04 (3.00)

If the instrument volume on the Scram Discharge Volume were to progressively fill during plant operations (Condition 1), certain indications should be received in the control room.

PROVIDE ALL automatic ACTIONS OR CONTROL ROOM INDICATIONS initiated by the instrument volume and the SETPOINTS for each.

QUESTION 3.05 (3.00)

Refer to attached Figure 4.1(8), Recirculation Flow Control for the following:

a. The plant is operating a 26% power and both recirc pump M/A transfer stations are in MANUAL and set for 28% speed. The recirc flow 'A' limit annunciator is clear. For each of the following instances, indicate HOW the speed of Recirc Pump 'A' would change (increase, decrease, or remain the same) AND WHICH component(s) of the control system is (are) limiting.

1. Recirc Pump 'A' M/A transfer station placed in AUTO. (1.0)

2. The generator speed tachometer output feedback signal fails low due to a loss of continuity through the field breaker contacts. (1.0)

b. Following a 'runback' of the recirc system from 100% power due to the trip of one feedpump, WHAT action must be taken by the control room operator prior to resetting the 'runback'? WHY? (Assume RFP is restarted prior to reset.) (1.0)

QUESTION 3.06 (3.00)

Regarding the Unit 2 SRVs and associated Low Low Set (LLS) Logic System:

a. There are three lights associated with each SRV - RED, GREEN, and AMBER. EXPLAIN WHAT each of the different colored lights indicate AND WHETHER it would be energized or de-energized during the time its SRV was open as a result of reactor pressure reaching the SRV's relief setpoint. (2.0)

b. LIST the TWO (2) conditions (signals) needed to arm the LLS logic. (1.0)

QUESTION 3.07 (2.00)

For the Off-gas Radiation Monitoring System:

a. WHAT THREE (3) combinations of radiation instrument trip signals will cause an Off-gas System auto-isolation? (1.0)

b. WHICH Off-gas System valves close on an auto-isolation? (1.0)

QUESTION 3.08 (3.00)

WHAT are SIX (6) of the seven automatic actions which should occur, OTHER THAN a Group 1 isolation, if Main Steam Line Radiation Monitors 'A' and 'B' reach their High-High trip setpoint?

QUESTION 3.09 (1.00)

CHOOSE the correct CAPITALIZED WORD for each of the lettered blanks below to describe the response of the Reactor Water Level Control System. The system is operating in DIFFERENTIAL PRESSURE CONTROL mode during a plant startup when an increase in steaming rate occurs.

Reactor water level will decrease causing the startup level control valve to ___(a)___ [OPEN/CLOSE]. This causes the differential pressure across the startup level control bypass valve to ___(b)___ [INCREASE/DECREASE]. The reactor feed pump speed controller senses this change in differential pressure and ___(c)___ [INCREASES/DECREASES] the reactor feed pump speed.

QUESTION 3.10 (3.00)

Concerning the Neutron Monitoring System:

a. WHY is it necessary to gamma compensate the Source and Intermediate Range Monitor signals? (1.0)

b. WHAT are the THREE (3) conditions which result in an SRM inoperative trip? (1.5)

c. At WHAT percent power should the APRM flow biased scram occur with 50% recirc loop flow? (0.5)

QUESTION 3.11 (2.00)

WHAT are FOUR (4) of the six conditions which will cause a PCIS Group 5 (Reactor Water Cleanup System) isolation? Setpoints NOT required.

QUESTION 4.01 (2.50)

According to the 'Power Changes' procedure, HNP-2-1005:

a. WHAT is Unit 2's licensed maximum thermal power for steady state operation? (0.5)

b. In WHAT instance may this maximum power be exceeded? HOW is it verified? (1.0)

c. Is increasing power at a rate of 600 MWE per hour acceptable? (YES or NO) (0.5)

d. TRUE or FALSE. Limit Generator Load to 55% of rated with only one Reactor Feedwater Pump in service. (0.5)

QUESTION 4.02 (1.00)

During a plant startup per HNP-2-1001, 'Normal Startup':

a. As it becomes apparent that criticality is impending, WHAT control rod withdrawal scheme is to be employed? (0.5)

b. Generally, WHICH rods in each Rod Worth Minimizer group are of the highest INTEGRAL worth? (0.5)

QUESTION 4.03 (2.00)

With Unit 2 operating at 50% power a loss of the 125/250 VDC SWITCHGEAR 2A (2R22-S016) occurs. Answer the following concerned with HNP-2-1913, 'Loss of DC Busses':

a. WHAT THREE (3) automatic actions should occur? (1.5)

b. HOW is reactor water level to be controlled following loss of the swicthgear? (0.5)

QUESTION 4.04 (3.00)

WHAT are THREE (3) of the five entry conditions for 'Inability to Shutdown with Control Rods,' HNP-2-1909?

QUESTION 4.05 (2.00)

During a 'Fast Reactor Shutdown with MSIV's Closed,' HNP-2-1025:

a. WHAT is the maximum reactor vessel cooldown rate allowed during this shutdown? (0.5)

b. WHAT method is to be used to accomplish the cooldown? (0.5)

c. Reactor vessel level should be maintained between ___(1)___ inches and ___(2)___ inches by utilizing the ___(3)___ system in conjunction with the ___(4)___ system. FILL IN THE NUMBERED BLANKS. (1.0)

QUESTION 4.06 (1.50)

In accordance with HNP-8002, 'Radiation Exposure Limits':

a. WHAT is the maximum whole body exposure you may receive in any week? (0.5)

b. Approval to receive in excess of 1,250 mRem/quarter will require written approval from WHICH TWO (2) individuals? (1.0)

QUESTION 4.07 (3.00)

NOTE: An ACTION STEP may have MULTIPLE actions and/or checks.

Following a reactor scram with the MSIV's open, WHAT are SIX (6) of the 8 immediate action steps to be performed from memory?

QUESTION 4.08 (3.00)

NOTE: An ACTION STEP may have MULTIPLE actions and/or checks.

For a reactor shutdown from outside the control room, HNP-2-1908:

a. WHAT TWO (2) methods are recommended in the procedure to scram the reactor from outside the control room? (2.0)

b. WHAT are TWO (2) of the three Operator Action Steps that should be performed, if possible, prior to evacuating the control room? (1.0)

QUESTION 4.09 (2.00)

WHAT are FOUR (4) of the six conditions listed in HNP-2-1902 which may indicate a SMALL BORE pipe break inside primary containment?

QUESTION 4.10 (2.00)

Regarding HNP-2-1933, 'Inability to Move a Control Rod', briefly DESCRIBE the double clutching method from the '00' position.

QUESTION 4.11 (2.00)

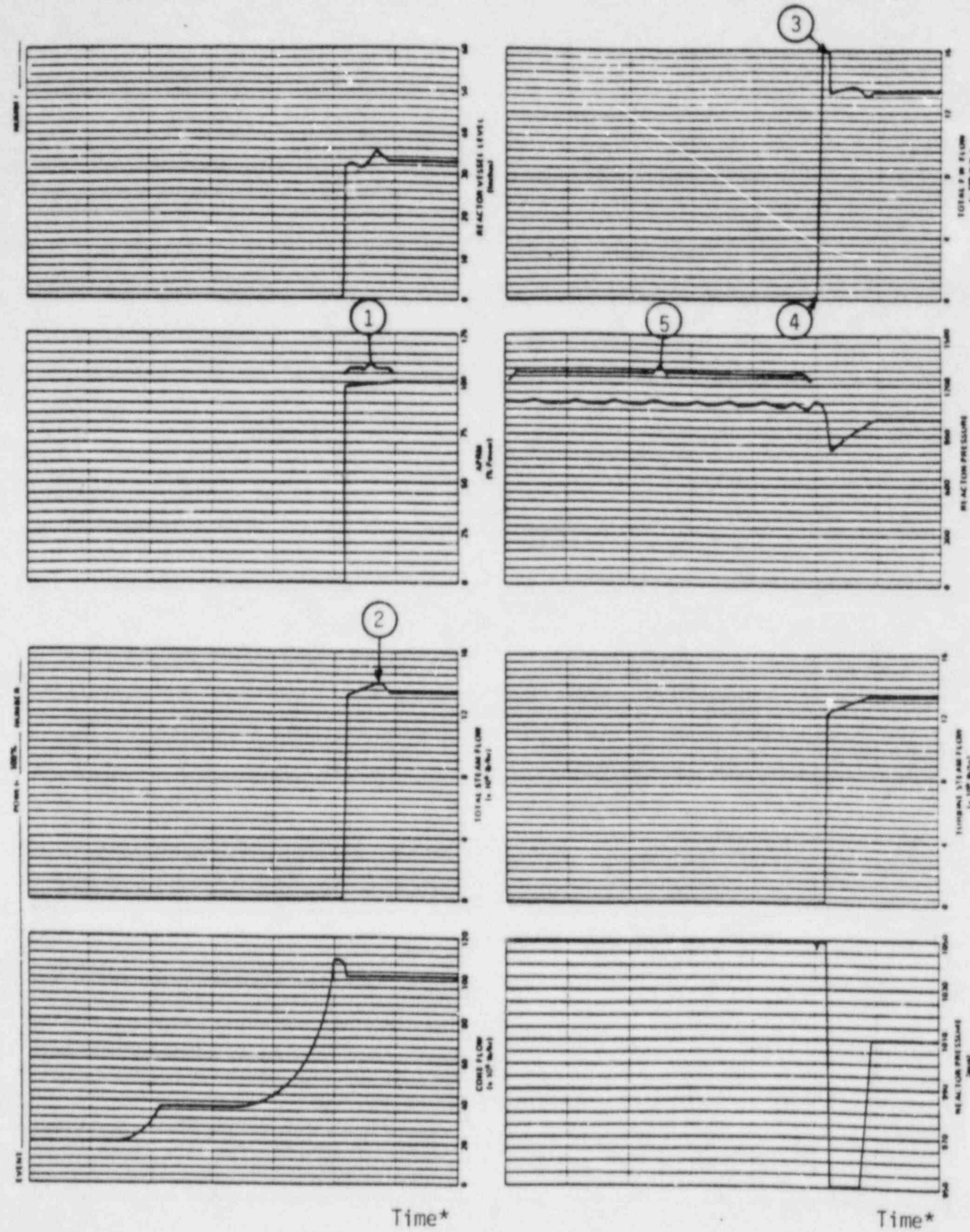
NOTE: An ACTION STEP may have MULTIPLE actions and/or checks.

If an auto turbine trip is annunciated, WHAT are FOUR (4) of the five Immediate Operator Action Steps per HNP-2-2001, 'Annunciator Response Procedures?'

QUESTION 4.12 (1.00)

With Unit 2 operating at 90% power, a Safety/Relief Valve (SRV) inadvertently opens. According to HNP-2-1907, 'Failure of Safety/Relief Valves to Operate,' WHAT are TWO (2) IMMEDIATE symptoms (NOT annunciator alarms) that might indicate the SRV is open?

FIGURE 1 for Question 1.04



EHC PRESSURE SETPOINT DECREASE

*Each time increment is one (1) minute

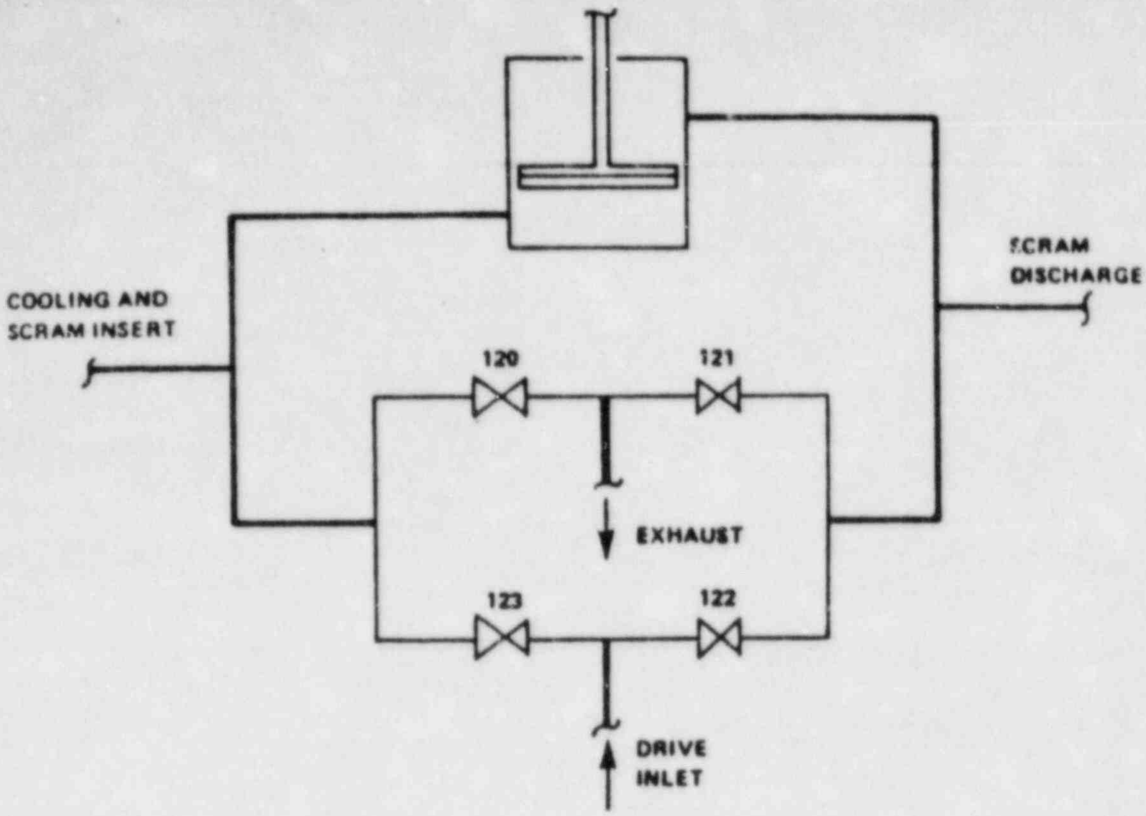


Figure 9.2.1 (1) Control Rod Sequence Timer

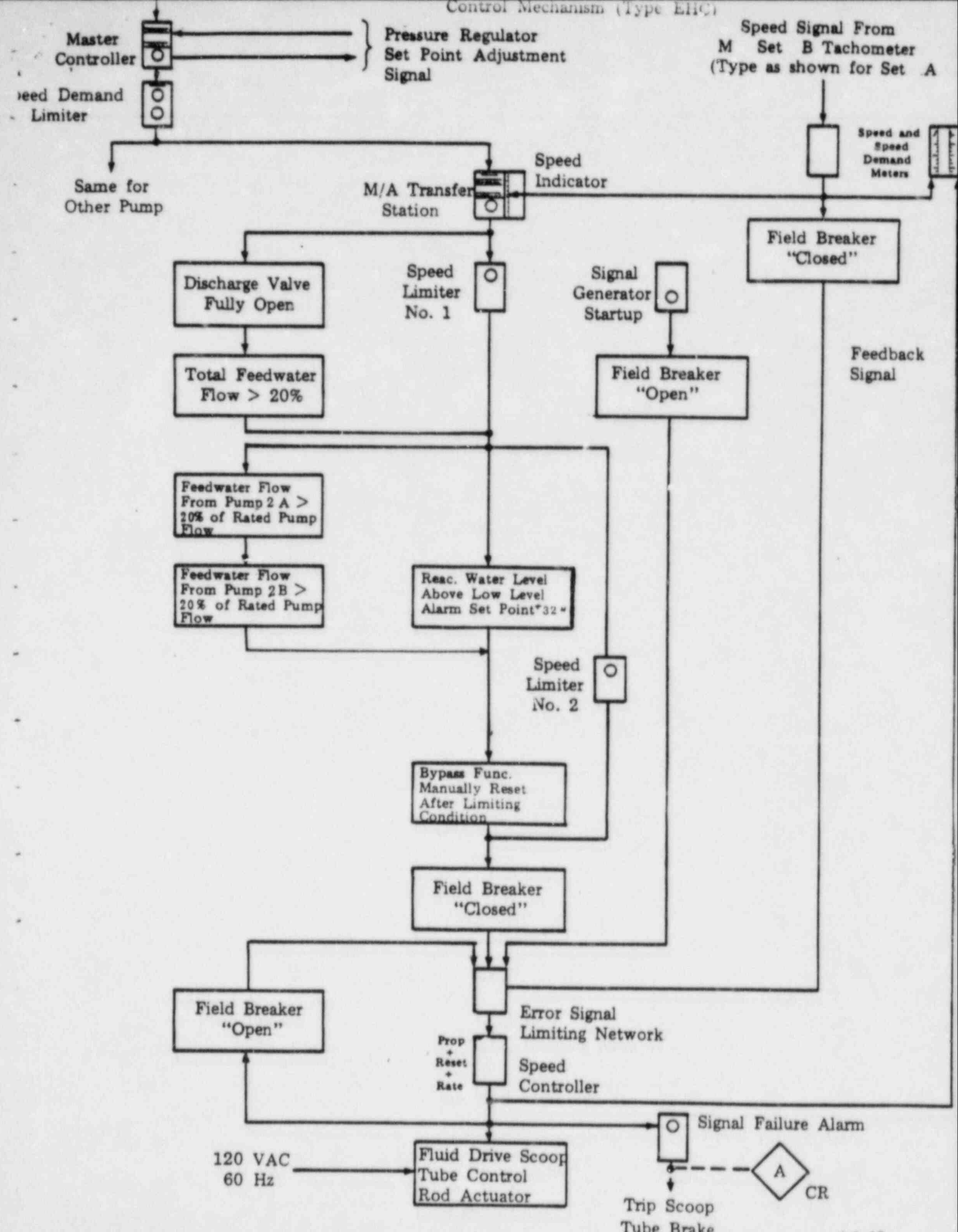


Figure 4.1 (8) Recirculation Flow Control

EQUATION SHEET

$f = ma$

$v = s/t$

Cycle efficiency = (Net work out)/(Energy in)

$w = mg$

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

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

$W = v\ \Delta P$

$A = \frac{\pi D^2}{4}$

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

$\Delta E = 931\ \Delta m$

$\dot{m} = V_{av} A \rho$

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

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

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

$Pwr = W_f \Delta h$

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

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

$TVL = 1.3/\mu$

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

$HVL = -0.693/\mu$

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

$SUR = 26.06/T$

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

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

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

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

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

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

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

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

$T = (\beta - \rho)/(\bar{\lambda}\rho)$

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

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

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

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

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

$I_1 d_1 = I_2 d_2$

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

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

$\Sigma = \sigma N$

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

$R/hr = 6\ CE/d^2\ (feet)$

<u>Water Parameters</u>

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.4C gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.
1 ft. H_2O = 0.4335 lbf/in.

<u>Miscellaneous Conversions</u>

1 curie = 3.7×10^{10} dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1 in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)
1 BTU = 778 ft-lbf
e = 2.718

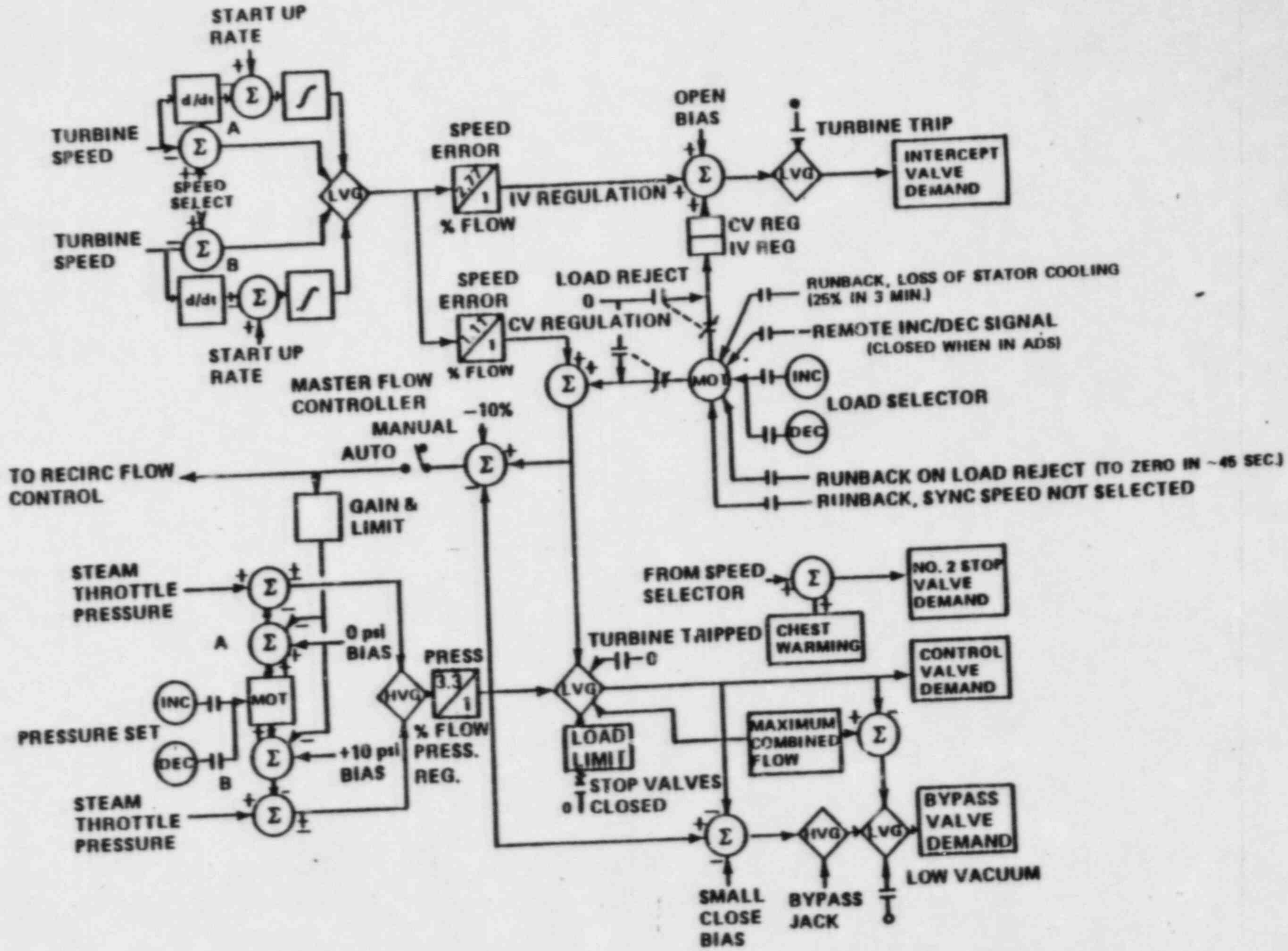


Figure 9.4(7) EHC Logic

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ANSWER 1.01 (2.50)

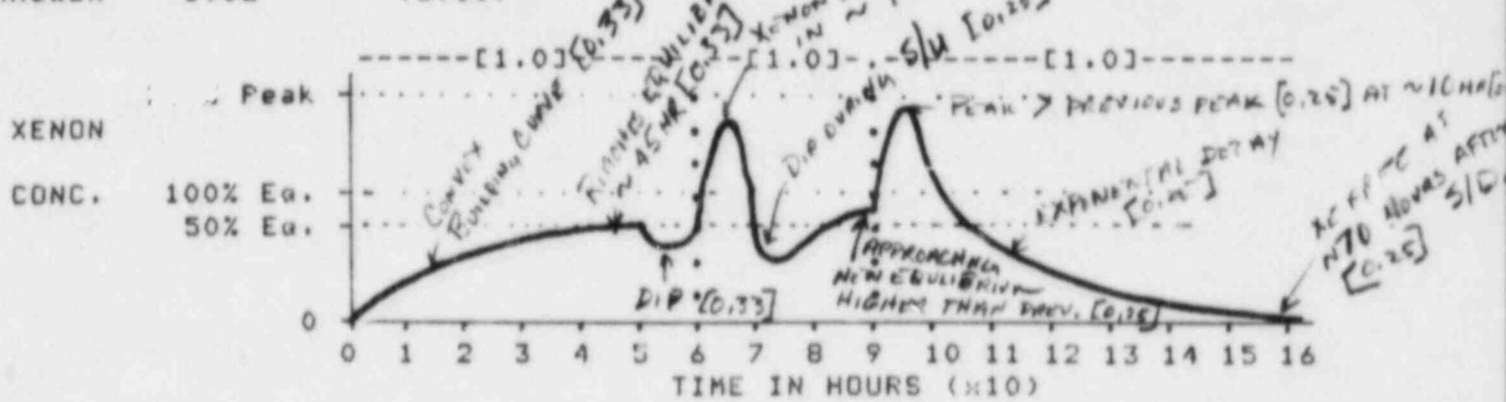
a. Indicated level is higher [0.5] by approximately ~~8"~~ 10" [0.5]. (1.0)
→ PARTIAL CREDIT [0.35] FOR 7" WHICH IS CORRECT FOR STP,

b. Indicated water level is sensed outside the dryer skirt [0.75]. —OR— IF DISCUSSED ANNULUS
Steam flow [0.25] at 100% power causes a pressure drop across the dryers ~~[0.75]~~ [0.5] (1.5)

REFERENCE

EIH Nuclear Training, Vol. 5, p. 2.3-5, AND
GE Documents 257-HA-771 & 383-HA-428.

ANSWER 1.02 (3.00)



REFERENCE

EIH Nuclear Training, Vol. 7, pp. 10.1-83 through 86.

ANSWER 1.03 (1.50)

According to centrifugal pump laws:

$$\text{Head} \sim (\text{Speed})^2 \quad [0.5]$$

Therefore,

$$\frac{100 \text{ ft}}{160 \text{ ft}} = \left[\frac{x}{3600 \text{ RPM}}\right]^2 \quad [0.5]$$

$$[x]^2 = 12.96 \times 10\text{E}6 \text{ RPM}^2 \ (0.625)$$

$$x = 2846 \text{ RPM} \quad [0.5]$$

REFERENCE

EIH Thermodynamics Lesson Plan, pp. 85 & 86.

ANSWER 1.04 (3.00)

a. The decreasing reactor pressure is causing an increase in core voids. (0.5)

b. Steam flow through the turbine bypass valves. (0.5)

c. The FWCS responding to the rapid decrease in reactor water level. (0.5)
(IF SAID RESPONDING TO HIGHER STM FLOW BUT DIDN'T MENTION WATER LEVEL -0.25)

d. The RFPs ran out of steam following the MSIV closure. (0.5)

e. SRVs lifting to control reactor pressure. (0.5)

f. Less core decay heat. (0.5)

REFERENCE

EIH Nuclear Training, Vol. 7, 10.4-10, and

BWR Transients, HXY-12.

ANSWER 1.05 (3.00)

a. Decreases [0.5]. Due to increased void content in the core as flow decreases [0.5]. (1.0)

b. Increases [0.5]. Due to increased voiding in the core or loss of tripped recirc pump suction from annulus [0.5]. (1.0)

c. Decreases [0.5]. Reactor Level Control System response to decreased steam flow or level increase [0.5]. (1.0)

REFERENCE

BWR-4 Transients, BXY-1.

ANSWER 1.06 (2.00)

a. The decay of delayed neutron precursors. (0.5)

b. As the core ages the Beta-Effective decreases [0.5] due to the burnout of U-235 (0.25) and the production of Pu-239 (0.15) and Pu-241 (0.1) [0.5]. (Both Pu-239 and Pu-241 have lower individual delayed neutron fractions than U-235.) (1.0)

c. The core's response time becomes slightly shorter (FASTER). (0.5)

REFERENCE

EIH Nuclear Training, Vol. 7, pp. 10.1-19, 48, & 49.

ANSWER 1.07 (1.50)

a. Larger. (0.5)

b. Longer. (0.5)

c. More. (0.5)

REFERENCE

EIH Nuclear Training, Vol. 7, pp. 10.1-41, 42, & 44.

ANSWER 1.08 (2.00)

o Final Power = (Initial Power)$e^{t/T}$,

o Final Power/ Initial Power = $e^{t/T}$, (0.5)

o $10 = e^{t/100s}$, (0.5)

o ln(10) = t/100s, (0.5)

o 2.3(100) = t = 230 seconds. (0.5)

REFERENCE

EIH Nuclear Training, Vol. 7, p. 10.1-62, and

ANSWERS -- HATCH 1&2 -84/07/10-PERSONS, R.

EIH Question Bank, Category 1, No. 20.

ANSWER 1.09 (2.00)

a. CD (0.5)

b. AB (0.5)

c. EA (0.5)

d. DE (0.5)

REFERENCE

EIH Thermodynamic Lesson Plan, pp. 52, 55, & 56, and

EIH Question Bank, Category 2, No. 15.

ANSWER 1.10 (2.00)

As power increases the amount of boiling (two-phase flow) increases [0.5]. The boiling will be the greatest in the core center due to it being the region of highest power [0.5]. Two-phase flow restricts cooling water flow due to the boiling action [0.5]. This will cause the higher powered bundles to receive less cooling water since their higher resistance to flow will divert flow to lower power fuel bundles [0.5], starving the higher power bundles.

REFERENCE

EIH Nuclear Training, Vol. 5, p. 2.2-6.

ANSWER 1.11 (3.00)

a. Decreases [0.25]. There is less steam flow, therefore less pressure drop through the main steam lines [0.75]. (1.0)

b. Increases [0.25]. With the same amount of cooling water through the condenser [0.25] and less of a heat load, [0.5] (condensate depression will increase) [0.75]. (1.0)

c. Decreases [0.25]. Less extraction steam from the turbine to heat the feedwater [0.75]. (1.0)

REFERENCE

EIH Heat Transfer Lesson Plan, pp. 75 & 78, and

EIH Nuclear Training, p. 10.4-11.

ANSWER 2.01 (2.00)

a. Unit 1 - 'B' [0.5], Unit 2 - 'C' [0.5]. (1.0)

b. Unit 1 - PSW [0.5], Unit 2 - RBCCW [0.5]. (1.0)

REFERENCE

EIH System Differences Lesson Plan, Rev. 1, pp. 6 & 10.

ANSWER 2.02 (3.00)

a. o SRV's D, F & G
   o RCIC
   o RHRSW B LOOP OR 'B' ~~and~~ 'D' PUMP ALSO ACCEPTABLE.
   o PSW B LOOP OR 'B' PUMP ALSO ACCEPTABLE.
   o B CRD PUMP (5 of 6 at 0.5 each) (2.5)

b. No. (0.5)

REFERENCE

EIH Nuclear Training, Vol. 7, 10.4-28. CAF (Verified by C. Dodd on 5/24/83.)

ANSWER 2.03 (3.00)

a. Poison injection must be fast enough to overcome reactivity due to cooldown. (1.0)

b. To ensure that the poison solution does not solidify in the lines and make the system inoperable., (1.0)
   0.25

c. o Core plate dP.
   o Core Spray System line break detection.
   o Jet pump dP.

   (3 at 0.33 each) (1.0)

REFERENCE

EIH Nuclear Training, Vol. 5, 4.6-4, 7, & 8.

ANSWER 2.04 (2.00)

The gland seal condenser will gradually fill with condensate [0.5] decreasing its ability to condense gland seal steam [0.5]. Eventually turbine gland seal will be lost and steam will leak from the turbine through the seals [0.5]. The system will, however, perform its intended function [0.5].

REFERENCE

EIH Nuclear Training, Vol. 6, Section II.B.

ANSWER 2.05 (3.00)

a. o Low reactor water level,

o High drywell pressure,

o High radiation in reactor building (vent exhaust,) ACCEPTED WITHOUT.

o High radiation on the refueling floor. Vent exhaust

(3 of 4 at 0.5 each) (1.5)

b. Charcoal bed high temperature (225 degrees F). (0.5)

c. Trips the fan [0.5] and the heaters [0.5]. (1.0)

REFERENCE

EIH Nuclear Training, Vol. 5, pp. 3.3-7 & 9.

ANSWER 2.06 (3.50)

a. Reactor water (leaking past the CRD seals) [0.5] and charging water from the CRDH System [0.5]. (1.0)

b. The affected control rod drifting in [0.5] or CRD high temperature [0.5]. ROD BLOCK, ALSO ACCEPTED SDV HI LEVEL ALARM, OR SCRAM AND HOT OR WARMER PIPING DOWNSTREAM OF SCRAM OUTLET VALVE, (1.0)

c. Unit 2 has a bypass feature (pushbuttons) [0.5] to provide the ability to override OR RESET the LOCA load shed [0.5] and restart the CRD pumps [0.5]. (1.5)

REFERENCE

EIH Nuclear Training, Vol. 5, Chapter 4.2, and

EIH System Differences Lesson Plan, Rev. 1, p. 6, and

NUREG/BR-005/Vol. 5, No. 4, Power Reactor Events, January 1984, p. 5, Event Summary No. 1.2 (event at Hatch Unit 2 on August 25, 1982).

ANSWER 2.07 (3.00)

a. To prevent rapid loss of reactor vessel inventory to the Torus. (1.0)

b. The SDC PCIS Valves (F008 & F009) will auto close [0.5] and the running RHR pump will trip [0.5]. OR HEAD SPRAY VALVE CLOSES (1.0)

c. The RHR inboard injection valves will not auto open [0.5] and the recirc discharge isolation valves will not auto close [0.5]. (1.0)

REFERENCE

EIH Nuclear Training, Vol. 6, Chapter 8.3, Section C.3.

HNP-2-1114, Rev. 15, p. 6.

ANSWER 2.08 (2.50)

a. Following the SRV's first actuation, the steam in its discharge line would condense causing a vacuum in the line [0.5]. This would result in suppression pool water being drawn up into the line [0.5] which could cause overpressurization of the line on the next actuation [0.5]. (SEE SRO KEY, ANSWER 6.09 a FOR OTHER ACCEPTABLE ANSWERS.) (1.5)

b. Increases [0.5]. The open vacuum breaker provides a direct path to the drywell [0.5]. (1.0)

REFERENCE

NUREG/BR-005/Vol.5, No. 4, Power Reactor Events, January 1984, p. 5, Event Summary No. 1.2 (event at Hatch Unit 2 on August 25, 1982).

ANSWER 2.09 (2.50)

a. 'C' Condensate pump will start (as booster suction pressure reaches 38 psig). (0.5)

b. 'B' Booster pump will trip (at 5 psig) and 'C' Booster pump will start when 'B' trips. (1.0)

c. Both running condensate pumps will trip (at 39") and both running boosters will trip (as their suction pressure decreases to 34 psig). (1.0)

REFERENCE

EIH Nuclear Training, Vol. 6, Chapter 5.4, Section III.B.2, & III.C.

ANSWER 3.01 (1.00)

120 open, 121, 122, and 123 shut.

REFERENCE

EIH Nuclear Training, Vol. 7, pp. 9.2.1-2.

ANSWER 3.02 (2.00)

a. The TCVs will close 10% [0.5] due to the load selector signal decreasing by 10% and being less than the pressure signal [0.5]. (1.0)

b. The BPVs will open by 10% [0.5] due to the change in the control valve demand signal to the BPV summer [0.5]. (1.0)

REFERENCE

EIH Nuclear Training, Vol. 7, pp. 9.4-12 through 16.

ANSWER 3.03 (2.00)

o NORMAL CONTROL RANGE: [0.1]
Span: 0" to +60", [0.2]
Referenced to INSTRUMENT ZERO (517"), [0.1]
Calibrated HOT. [0.1] (0.5)

o EMERGENCY SYSTEM RANGE:
Span: -150" to +60",
Referenced to INSTRUMENT ZERO (517"),
Calibrated HOT. (0.5)

o SHUTDOWN VESSEL FLOODING RANGE: OR VESSEL FLOODUP
Span: -17" to +383",
Referenced to INSTRUMENT ZERO (517"),
Calibrated COLD. (0.5)

o POST ACCIDENT FLOODING RANGE:
Span: -317" to -17",
Referenced to INSTRUMENT ZERO (517"),
Calibrated COLD. (0.5)

REFERENCE

EIH Training Manual, Vol. 5, pp. 2.3-2 through 4.

ANSWER 3.04 (3.00)

UNIT 1 SETPOINTS IN PARENTHESIS

- 'Scram Discharge Volume Not Drained' alarm [0.5] at 1.5 gallons [0.5]. (3 GALLONS)
- Control Rod Block alarm [0.5] at 36.2 gallons [0.5]. (18 GALLONS)
- Scram and alarm on SDV High Level Trip [0.5] at 57.15 gallons [0.5]. (71 GALLONS)

REFERENCE

EIH Nuclear Training, Vol. 7, p. 9.3-19, and

EIH Question Bank, Category 4, No. 33.

ANSWER 3.05 (3.00)

a. 1. Increase [0.5]. Master limiter, low speed limit [0.5].

2. Increase [0.5]. Scoop tube positioning unit [0.5]. (2.0)

b. The setpoint must be manually runback on each pump (If M/A transferred to MANUAL) prior to resetting the runback [0.5]. Otherwise the recirc pumps will ramp up to the previous setting causing a possible scram [0.5]. (1.0)

REFERENCE

EIH Nuclear Training, Vol. 5, pp. 4.1-22 through 24.

ANSWER 3.06 (3.00)

a. RED - Solenoid control valve has energized [0.33]; de-energized [0.33].

GREEN - Power available to the solenoid control valve [0.33]; energized [0.33].

AMBER - Pressure in tailpipe (>85 psig) [0.33]; energized [0.33]. (2.0)

b. (1) Any SRV has opened [0.5] and,

(2) reactor pressure has exceeded the high pressure scram setpoint (1044 psig) [0.5]. (1.0)

REFERENCE

EIH Nuclear Training, Vol. 6, Chapter 5.1, Section II.B.2.

ANSWER 3.07 (2.00)

a. o 2 upscale Hi-Hi-Hi radiation trips, or

o 1 upscale Hi-Hi-Hi radiation trip and 1 downscale trip, or

o 2 downscale trips, 1 from each channel. [0.33 each] (1.0)

b. Off-gas system outlet and drain valves -OR-

Discharge valve to stack, cooler condenser and moisture separator drain valves, and holdup line drain valve (F085). (1.0)

REFERENCE

EIH Nuclear Training, Vol. 7, p. 9.7.1-7, and

Vol. 6, p. 6.8-21, and

HNP-2-2067.

ANSWER 3.08 (3.00)

o Reactor Scram.

o Mechanical vacuum pump trip.

o Mechanical vacuum pump discharge valves isolate. (F010)

o Mechanical vacuum pump suction valves isolate. (F007)

o Gland steam seal exhauster trip.

o Gland steam seal exhauster isolation.

o Control room ventilation swaps to Mode II (pressurization mode).

(6 of 7 at 0.5 each)

REFERENCE

EIH Nuclear Training, Vol. 7, pp. 9.7.1-3 & 4.

HNP-2-1901, Rev. 8, p 1.

ANSWER 3.09 (1.00)

a. Open.

b. Decrease.

c. Increases. (0.33 each)

REFERENCE

EIH Nuclear Training, Vol. 6, Chapter 5.3, Section III.D, and Vol. 7, p. 9.5-12.

ANSWER 3.10 (3.00)

a. Because at low power levels, the signal produced by the decay or background gamma overshadows the signals produced by the neutrons and fission gammas. (SEE SRO KEY, ANSWER 6.08 a FOR PARTIAL CREDIT BASIS.) (1.0)

b. o Selector switch out of operate.

o High voltage-low.

o Any module unplugged. (0.5 each) (1.5)

c. 0.66W + 51, where W=% recirc loop flow

0.66(50) + 51 = 33 + 51 = 84% (0.5)

REFERENCE

EIH Nuclear Training, Vol. 7, pp. 9.1.1-8 & 12 and 9.1.3-9.

ANSWER 3.11 (2.00)

- o Low reactor water level.
- o High RWCUS equipment room temperature.
- o High RWCUS equipment room vent differential temperature.
- o High RWCUS differential flow.
- o High temperature after the Non-regen. Heat Exchanger.
- o SBLC actuation.

(4 of 6 at 0.5 each)

REFERENCE

EIH Nuclear Training, Vol. 5, p. 3.1-31, and

EIH Question Bank, Category 3, No. 3.

ANSWER 4.01 (2.50)

a. 2436 MWt. (0.5)

b. Thermal spike [0.5].
By using OD-3 printouts [0.5]. (1.0)

c. IS. YES. (600/60 = 10 MWE/min) (0.5)

d. TRUE. (0.5)

REFERENCE

HNP-2-1005, Rev. 12, pp. 1 & 2.

ANSWER 4.02 (1.00)

a. Notch and wait. (0.5)

b. The first rods. (0.5)

REFERENCE

HNP-2-1001, Rev. 21, p. 16.

ANSWER 4.03 (2.00)

a. o Turbine trip

o Reactor scram

o Recirc pumps trip

(3 at 0.5 each) (1.5)

b. With HPCI. (0.5)

REFERENCE

HNP-2-1913, Rev. 7, p. 2.

ANSWER 4.04 (3.00)

o Lack of neutron flux decrease indication on neutron monitors.

o Lack of FULL IN indication lamps for individual control rods.

o Improper digital position indication of selected control rod.

o Reactor power starting to increase, as indicated by nuclear instrumentation and steam production.

o Shutdown occured, but calculations indicate criticality will occur within the next hour.

(3 of 5 at 1.0 each)

REFERENCE

HNP-2-1909, Rev. 8, p. 1.

EIH Question and Answer Bank, Category 9, No. 10.

ANSWER 4.05 (2.00)

a. 100 degrees F per hour. (0.5)

b. Steam condensing mode of RHR. (0.5)

c. (1) +32

(2) +42

(3) RCIC

(4) RWCU [0.25 each] (1.0)

REFERENCE

HNP-2-1025, Rev. 2, p. 1.

ANSWER 4.06 (1.50)

a. 300 mRem. (0.5)

b. Immediate supervisor [0.5] and a Lab Supervisor [0.5]. (1.0)
OR HEALTH PHYSICS SUPERVISOR

REFERENCE

HNP-8002, Rev. 14, p. 3.

ANSWER 4.07 (3.00)

- o Place mode switch to SHUTDOWN.
- o Depress all four manual scram pushbuttons.
- o Verify flux is decreasing.
- o Check that green FULL IN lights are lit for operable rods. Manually insert any rods not FULL IN.
- o Depress the main turbine trip button and check that generator PCBS and exciter field ACB trips after driving steam is depleted.
- o After initial level transient is over observe that reactor level stabilizes between +32" and +42", by use of multiple indications.
- o Maintain reactor level between +32" and +42" utilizing startup FW configuration. If required, trip one RFPT. If required, transfer RFPT speed controller(s) to MANUAL.
- O Maintain reactor water level below steam lines.

(6 of 8 at 0.5 each)

REFERENCE

HNP-2001, Rev. 19, p. 1.

ANSWER 4.08 (3.00)

a. o Open the feeder breakers to the APRM circuitry at the RPS distribution panel [1.0].

o Simultaneously trip the mercoid switches on the SDV Hi-Hi level switches [1.0]. (2.0)

b. o Mode switch to SHUTDOWN.

o Trip main turbine.

o Place level control in single element.

(2 of 3 at 0.5 each) (1.0)

REFERENCE

HNP-2-1908, Rev. 11, pp. 1 & 2.

ANSWER 4.09 (2.00)

(ACCEPTED EACH INDIVIDUALLY AS 1 OF 4 REQ'D)

o Pressure and/or temperature increase in drywell.

o Reactor scram by high drywell pressure.

o Increasing airborne activity in drywell.
(ACCEPTED FISSION PRODUCT ACTIVITY OR INCREASE OF FISSION PRODUCT MONITOR FOR ADDED REFERENCE.)

o Generator load decrease.

o High level in the drywell floor drain sump.

o Drywell floor drain sump pump operating frequency increases.

(4 of 6 at 0.5 each)

REFERENCE

HNP-2-1902, Rev. 18, p. 1.

EIH NUCLEAR TRAINING, VOL. 7, PP 9.7.3-1 & 2.

ANSWER 4.10 (2.00)

1. Turn the EMERGENCY IN/NOTCH OVERRIDE switch to the EMERG ROD in position and hold for several seconds. Confirm the green full-in light is illuminated. (1.0)

2. Simultaneously turn the EMERGENCY IN/NOTCH OVERRIDE switch to NOTCH OVERRIDE and the ROD MOVEMENT CONTROL switch to ROD OUT NOTCH position. (1.0)

REFERENCE

HNP-2-1933, Rev. 7, pp. 1 & 2.

ANSWER 4.11 (2.00)

- o Check reactor scram if greater than 30% power and respond to same.
- o Depress the main turbine trip button.
- o Check that stop valves and CIVS close.
- o Check that generator PCBS and exciter field ACB trips after driving steam is depleted.
- o Check that extraction check valves close and extraction drains open.

(4 of 5 at 0.5 each)

REFERENCE

HNP-2-2001, Rev. 19, p. 4.

ANSWER 4.12 (1.00)

- o Decrease in main steam flow indication.
- o Increase in steam flow feedwater flow mismatch.
- o Amber light lit on open valve.
- o Decrease in main generator output and CV position.
- o Relief valve discharge temperature recorder upscale.

(2 at 0.5 each)

REFERENCE

HNP-2-1907, Rev. 16, p. 1, and

EIH Question Bank, Category 9, No. 1.

ENCLOSURE 3
(1 OF 2)

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

FACILITY: _HATCH_1&2_______________

REACTOR TYPE: _BWR-GE4_______________

DATE ADMINISTERED: _84/07/10_______________

EXAMINER: _PERSONS,_B._______________

APPLICANT: _______________________

INSTRUCTIONS_TO_APPLICANT:

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

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
24.00	24.24	__________	________	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.50	25.76	__________	________	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.50	25.76	__________	________	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
24.00	24.24	__________	________	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
99.00	100.00	__________	________	TOTALS

FINAL GRADE _______________%

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

____
APPLICANT'S SIGNATURE

QUESTION 5.01 (.50)

The reactor is operating at a steam dome temperature of 536 degrees F when reactor power is increased so that steam dome temperature increases to 544 degrees F. WHICH of the following statements is most correct?

a. Steam pressure increased, steam enthalpy increased.

b. Steam pressure remained constant, steam enthalpy decreased.

c. Steam pressure increased, steam enthalpy decreased.

d. Steam pressure remained constant, steam enthalpy increased.

QUESTION 5.02 (3.00)

Following a normal reduction in power from 90% to 70% with recirculation flow, HOW will the following change (increase, decrease, or remain the same) AND WHY:

a. The pressure difference between the reactor and the turbine steam chest. (1.0)

b. Condensate depression at the exit of the condenser. (1.0)

c. Final Feedwater temperature. (1.0)

QUESTION 5.03 (1.00)

Regarding the core thermal limits:

a. The process computer output, CMFLPD, is used to monitor WHICH core thermal limit? (0.5)

b. WHICH core thermal limit ensures peak cladding temperature will not exceed 2200 degrees F following a LOCA? (0.5)

QUESTION 5.04 (2.00)

Briefly EXPLAIN WHAT happens to the magnitude of the moderator temperature coefficient of reactivity (INCREASES or DECREASES) and WHY considering the following changes:

a. An increase in moderator temperature. (1.0)

b. A decrease in control rod density. (1.0)

QUESTION 5.05 (2.50)

Concerning heat transfer in the reactor:

a. Briefly EXLAIN WHY nucleate boiling improves the heat transfer characteristics of the core over simple forced convection? (1.5)

b. Considering the heat transfer mechanism following fuel dryout after a large break LOCA from a high reactor power, WHY are the central fuel rods in a fuel bundle more likely to exceed the 2200 degree F limit for peak clad temperature than the edge or corner rods which have higher local peaking factors? (1.0)

QUESTION 5.06 (1.50)

Attached FIGURE 1 shows a basic closed loop fluid system with its head vs. flow plot. The two pumps are identical, single speed, radial, centrifugal pumps. Initially, assume Pump 1 is operating to supply flow to Component 1, as shown.

a. WHAT is Point X on the System Head vs. Flow Plot? (0.5)

b. WHICH pump curve, A or B, most accurately shows BOTH PUMPS operating to supply system flow? (0.5)

c. WHICH WAY, to the LEFT or to the RIGHT, would the System Curve shift if Component 2 was valved into the system, in addition to Component 1? (0.5)

QUESTION 5.07 (2.00)

A reactor is operating at 2000 MW (100% of rated thermal power) with the APRM flow biased scram setpoint at 117% of rated. The total scram delay time is 10 seconds, measured from the time the scram setpoint is exceeded at 117% until sufficient negative reactivity has been inserted by the scram to turn power.

If a sudden insertion of positive reactivity results in a 10 second period, WHAT will be the peak power IN MEGAWATTS for the excursion? SHOW ALL WORK.

NOTE: The instantaneous APRM scram setpoint should NOT be considered in answering this question.

QUESTION 5.08 (.50)

Using the rule of thumb for the time required to reach peak xenon, APPROXIMATE the TIME to reach peak xenon following a scram from 50% power.

QUESTION 5.09 (3.00)

Regarding the xenon transient following a significant DECREASE in reactor power from high power operation:

a. Briefly, EXPLAIN WHY the xenon concentration will peak following the manuever. (1.0)

b. HOW will peripheral control rod worth be affected (INCREASE, DECREASE, REMAIN THE SAME) during the xenon peak? BRIEFLY EXPLAIN your answer. (1.5)

c. If the decrease in reactor power was from 100% to 50%, would the new (50% power) equilibrium xenon reactivity be MORE THAN, LESS THAN or EQUAL TO one half the 100% equilibrium value? (0.5)

QUESTION 5.10 (3.00)

The reactor is operating at 70% power with all systems functioning normally when Recirc Flow Controller 'A' fails HIGH. Using attached Figure 2, IDENTIFY the CAUSE of the recorder indication changes at EACH of the NUMBERED POINTS described below.

NOTES:
- o Time intervals on the graph are in 1 minute increments.
- o The transient begins ~1 minute, 15 seconds from the beginning of each graph.

(1) The decrease in reactor water level. (0.5)

(2) The increase in reactor power. (0.5)

(3) The decrease in core flow. (0.5)

(4) The increase in reactor pressure. (0.5)

(5) The increase in total feedwater flow. (0.5)

(6) The slight increase in total steam flow. (0.5)

QUESTION 5.11 (3.00)

Attached Figure 3 shows selected plant parameter responses for a TURBINE TRIP transient initiated from rated conditions with NO OPERATOR ACTION.

NOTES:
(1) Time intervals on graphs are 1 minute each.
(2) Use of graphs not directly referred to in question MAY be required to correctly answer all parts.
(3) Malfunction(s) other than the initiating one MAY be involved.

ANSWER the following:

a. Why does core flow decrease [Point 1] and why doesn't it decrease to zero [Point 2]? (1.0)

b. Why does reactor pressure increase [Point 3] and remain high [Point 4]? (1.0)

c. Why does reactor level decrease initially [Point 5] and what is causing the peaks in level later [Point 6]? (1.0)

QUESTION 5.12 (2.00)

Consider the attached process computer P-1 print-out, Figure B, part of an ACTUAL P-1.

a. IS the output signal from APRM 1 MORE or LESS conservative than the output signal from APRM 3? (0.5)

b. If all the fuel has a design LHGR limit of 13.4 KW/ft., WHAT is the MAXIMUM actual LHGR in the core? (1.0)

c. IS the axial power distribution bottom or top core peaked? (0.5)

QUESTION 6.01 (3.50)

Concerning the Control Rod Drive (CRD) Hydraulic System:

a. Upon completion of a reactor scram with all CRDs fully inserted, WHAT are the TWO (2) sources of water continuing to fill the scram discharge volume until the scram is reset? (1.0)

b. WHAT are TWO (2) possible indications/events resulting from a leaking scram outlet valve? (1.0)

c. WHAT major difference exists between the Unit 1 and Unit 2 CRD pump controls? Briefly, EXPLAIN its function. (1.5)

QUESTION 6.02 (3.00)

Regarding the Residual Heat Removal (RHR) System:

a. With an RHR System aligned for Shutdown Cooling Mode, WHY is it necessary to prevent the RHR pumps' minimum flow valve from opening? (1.0)

b. With the system operating in Shutdown Cooling Mode, HOW will the system be effected if reactor pressure exceeds 135 psig? (1.0)

c. WHAT TWO (2) automatic actions will NOT occur in an RHR loop if the RHR logic is in full test (i. e., both logic circuits in test) when a LPCI initiation signal occurs? (1.0)

QUESTION 6.03 (2.50)

Assume the plant is operating at 75% power with 'A' & 'B' Condensate pumps and 'A' & 'B' Condensate Booster pumps running. The 'C' Condensate and 'C' Condensate Booster pumps are in standby AUTO.

For each of the conditions below indicate HOW the above configuration would automatically change.

a. A 'Condensate Booster Pumps Suction Low Pressure' alarm is received at 43 psig and suction pressure continues to decrease to 38 psig where the configuration changes with no booster pump trips. (0.5)

b. 'B' Booster pump auxiliary lube oil pump fails to start and the booster pump's lube oil pressure decreases to zero psig. (1.0)

c. A 'Condensate Pump Low Level' alarm is received. (1.0)

QUESTION 6.04 (2.00)

Answer the following for Unit 1 and Unit 2:

a. WHICH main steam line supplies steam to RCIC? (1.0)

b. WHAT is the cooling water supply to the recirc MG set air coolers? (1.0)

QUESTION 6.05 (3.00)

WHAT are SIX (6) of the seven automatic actions which should occur, OTHER THAN a Group 1 isolation, if Main Steam Line Radiation Monitors 'A' and 'B' reach their High-High trip setpoint?

QUESTION 6.06 (2.50)

Regarding the Reactor Manual Control System:

a. Referring to attached Figure 9.2.1 (1), WHAT is the alignment of the four (4) directional control valves during the 'settle mode?' (1.0)

b. With the Refuel Platform over the core and the Reactor Mode Switch in REFUEL, WHAT are THREE (3) of the four refueling interlocks which will result in a control rod withdrawal block? (1.5)

QUESTION 6.07 (1.00)

CHOOSE the correct CAPITALIZED WORD for each of the lettered blanks below to describe the response of the Reactor Water Level Control System. The system is operating in DIFFERENTIAL PRESSURE CONTROL mode during a plant startup when an increase in steaming rate occurs.

Reactor water level will decrease causing the startup level control valve to ___(a)___ [OPEN/CLOSE]. This causes the differential pressure across the startup level control bypass valve to ___(b)___ [INCREASE/DECREASE]. The reactor feed pump speed controller senses this change in differential pressure and ___(c)___ [INCREASES/DECREASES] the reactor feed pump speed.

QUESTION 6.08 (3.00)

Concerning the Neutron Monitoring System:

a. WHY is it necessary to gamma compensate the Source and Intermediate Range Monitor signals? (1.0)

b. WHAT are the THREE (3) conditions which result in an SRM inoperative trip? (1.5)

c. At WHAT percent power should the APRM flow biased scram occur with 50% recirc loop flow? (0.5)

QUESTION 6.09 (2.50)

With regard to the Main Steam Safety Relief Valves (SRVs):

a. EXPLAIN HOW/WHY an SRV discharge pipe (tail pipe) could be damaged due to its vacuum breaker STICKING SHUT during repeated actuation (lifting) of the SRV? (1.5)

b. How (INCREASE, DECREASE, REMAINS THE SAME) would Drywell Pressure be expected to respond to an SRV discharge line vacuum breaker STICKING OPEN during actuation of the SRV? Briefly, JUSTIFY your answer. (1.0)

QUESTION 6.10 (2.50)

With the plant operating at 100% power (Unit 1), recirc in Master Manual, an operator inadvertently DECREASES the 'Pressure Set' by 5 psi. WHAT will be the INITIAL response and FINAL status of the following parameters due to this action? Briefly EXPLAIN. Assume NO operator action. See attached Figure 9.4(7). ANSWER on the attached handout page.

a. TCV position

b. BPV position

c. Power

d. Pressure

QUESTION 7.01 (2.50)

According to the 'Power Changes' procedure, HNP-2-1005:

a. WHAT is Unit 2's licensed maximum thermal power for steady state operation? (0.5)

b. In WHAT instance may this maximum power be exceeded? HOW is it verified? (1.0)

c. Is increasing power at a rate of 600 MWE per hour acceptable? (YES or NO) (0.5)

d. TRUE or FALSE. Limit Generator Load to 55% of rated with only one Reactor Feedwater Pump in service. (0.5)

QUESTION 7.02 (1.00)

During a plant startup per HNP-2-1001, 'Normal Startup':

a. As it becomes apparent that criticality is impending, WHAT control rod withdrawal scheme is to be employed? (0.5)

b. Generally, WHICH rods in each Rod Worth Minimizer group are of the highest INTEGRAL worth? (0.5)

QUESTION 7.03 (3.00)

With the reactor being maintained in 'Hot Standby' per HNP-2-1015:

a. With the MSIVs SHUT, WHAT are TWO (2) methods of decay heat removal, OTHER THAN SRVs, which may be substituted for RHR steam condensing mode? (1.0)

b. With the MSIVs OPEN, WHAT FOUR (4) steps should be taken to reduce thermal duty on the feedwater nozzles? (2.0)

QUESTION 7.04 (2.00)

With Unit 2 operating at 50% power a loss of the 125/250 VDC SWITCHGEAR 2A (2R22-S016) occurs. Answer the following concerned with HNP-2-1913, 'Loss of DC Busses':

a. WHAT THREE (3) automatic actions should occur? (1.5)

b. HOW is reactor water level to be controlled following loss of the swicthgear? (0.5)

QUESTION 7.05 (2.00)

For Post LOCA Hydrogen Recombiner Operation per HNP-2-1235:

a. WHY must RHR pressure be greater then or equal to 100 psig prior to startup of the recombiner? BE SPECIFIC. (1.0)

b. If containment pressure exceeds 20 psig during operation of a recombiner, WHAT TWO (2) automatic action(s) will occur? (1.0)

QUESTION 7.06 (3.00)

WHAT are THREE (3) of the five entry conditions for 'Inability to Shutdown with Control Rods,' HNP-2-1909?

QUESTION 7.07 (2.00)

Regarding HNP-2-1933, 'Inability to Move a Control Rod', briefly DESCRIBE the double clutching method from the '00' position.

QUESTION 7.08 (3.00)

NOTE: An ACTION STEP may have MULTIPLE actions and/or checks.

Consider a Pipe Break Inside Primary Containment, per HNP-2-1902:

a. WHAT are the FOUR (4) Immediate Operator Action Steps OTHER THAN the step for assuring the reactor is shutdown? (2.0)

b. With a large break in the Drywell ERRATIC level indication may occur. WHAT causes this? (1.0)

QUESTION 7.09 (1.50)

If a radiological event occurs the Shift Supervisor is required to contact the Control Room and request information about the event as the first action step of Emergency Procedure HNP-4323. WHAT THREE (3) pieces of information should the Shift Supervisor request?

QUESTION 7.10 (2.00)

During a 'Fast Reactor Shutdown with MSIV's Closed,' HNP-2-1025:

a. WHAT is the maximum reactor vessel cooldown rate allowed during this shutdown? (0.5)

b. WHAT method is to be used to accomplish the cooldown? (0.5)

c. Reactor vessel level should be maintained between ___(1)___ inches and ___(2)___ inches by utilizing the ___(3)___ system in conjunction with the ___(4)___ system. FILL IN THE NUMBERED BLANKS. (1.0)

QUESTION 7.11 (2.00)

NOTE: An ACTION STEP may have MULTIPLE actions and/or checks.

If an auto turbine trip is annunciated, WHAT are FOUR (4) of the five Immediate Operator Action Steps per HNP-2-2001, 'Annunciator Response Procedures?'

QUESTION 7.12 (1.50)

With Unit 2 operating at 90% power, a Safety/Relief Valve (SRV) inadvertently opens. According to HNP-2-1907, 'Failure of Safety/Relief Valves to Operate,' WHAT are THREE (3) IMMEDIATE symptoms (NOT annunciator alarms) that might indicate the SRV is open?

QUESTION 8.01 (3.00)

WHAT are THREE (3) conditions requiring a control rod to be considered inoperable per the Unit 1 Technical Specifications?

QUESTION 8.02 (2.00)

With the reactor head installed and irradiated fuel in the reactor vessel, according to the Unit 1 Technical Specifications, WHAT are the Limiting Safety System Settings for the following:

a. The Reactor High Pressure Scram. (0.5)

b. The Nuclear Steam System relief valve settings. (1.5)

QUESTION 3.03 (2.00)

During preparation for a Unit 2 plant startup from operational Condition 3 the Shift Supervisor (SS) is informed that one of the LPCI pumps has failed its surveillance test, but it is probable that the pump can be repaired within 36 hours. Based on this information the SS decides that even if repairs take longer he still has 7 days before he exceeds the time requirement for the action statement, and therefore orders the reactor plant startup to commence. IS THIS DECISION CORRECT? EXPLAIN WHY you agree or disagree with his decision.

QUESTION 8.04 (3.00)

According to the Administrative Procedure for Equipment Clearance and Tagging, HNP-501, if a supervisor HAS NOT released his subclearance:

a. WHAT conditions must exist prior to initiating a release of the subclearance by another authorized person? (2.0)

b. Providing the above conditions exist, either of two people can approve release of the subclearance. WHO ARE THEY? (1.0)

QUESTION 8.05 (3.00)

a. Per HNP-514, 'Control of Locked Valves,' SPECIFICALLY, HOW is Independent Verification of a LOCKED CLOSED valve to be performed? (1.0)

b. WHAT are THREE (3) things the operator should check to verify a valve's proper position and functioning? (2.0)

QUESTION 8.06 (1.00)

According the plant Standing Orders, if a double notch occurs while withdrawing a control rod, WHAT corrective action should be taken?

QUESTION 8.07 (2.50)

For the suppression chamber water temperatures listed below, WHAT ACTION(S) is(are) required by the Unit 2 Technical Specifications with Unit 2 in OPERATIONAL CONDITION 1 or 2?

a. 97 Degrees F. (0.5)

b. 108 Degrees F during HPCI testing. (1.0)

c. 121 Degrees F following a scram with the MSIV's SHUT. (1.0)

QUESTION 8.08 (3.00)

Unit 2 is at 100% power with only one outstanding LCO:

The 'A' core spray pump is INOP due to an in-progress (1 day) repair. It is anticipated that the pump will be returned to service in 5 days (within 7 day action statement limit).

The Shift Supervisor has just given approval to commence a 2 day turbo charger repair on Diesel Generator 2A. This decision was based on the following:

- o Satisfactory operability of all other normal and emergency power supplies.
- o Satisfactory completion (and scheduling) of the diesel generator and offsite/onsite circuit surveillances.
- o The fact that a 2 day repair is within the 72 hour action statement time limit.

With Technical Specification Sections 3.5.3.1 and 3.8.1.1 attached for reference, DETERMINE if the Shift Supervisor's decision was CORRECT or INCORRECT and briefly EXPLAIN WHY.

QUESTION 8.09 (2.50)

Regarding the Unit 2 Technical Specifications for refueling operations:

a. HOW MANY SRM's must be operable? (0.5)

b. WHERE must the operable SRM detectors be located? (1.0)

c. WHAT TWO (2) conditions require removal of the SRM scram shorting links? (1.0)

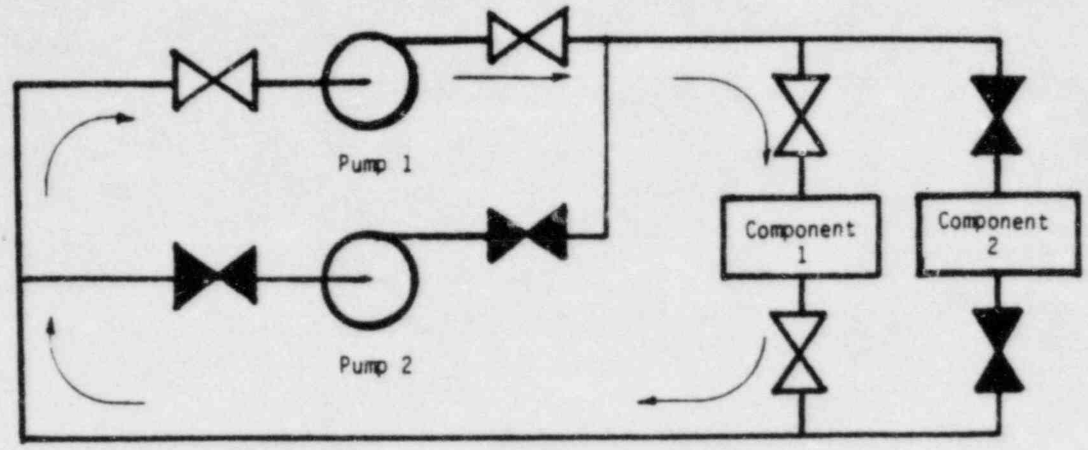
QUESTION 8.10 (2.00)

For an 'SRO Procedure Change' per HNP-9:

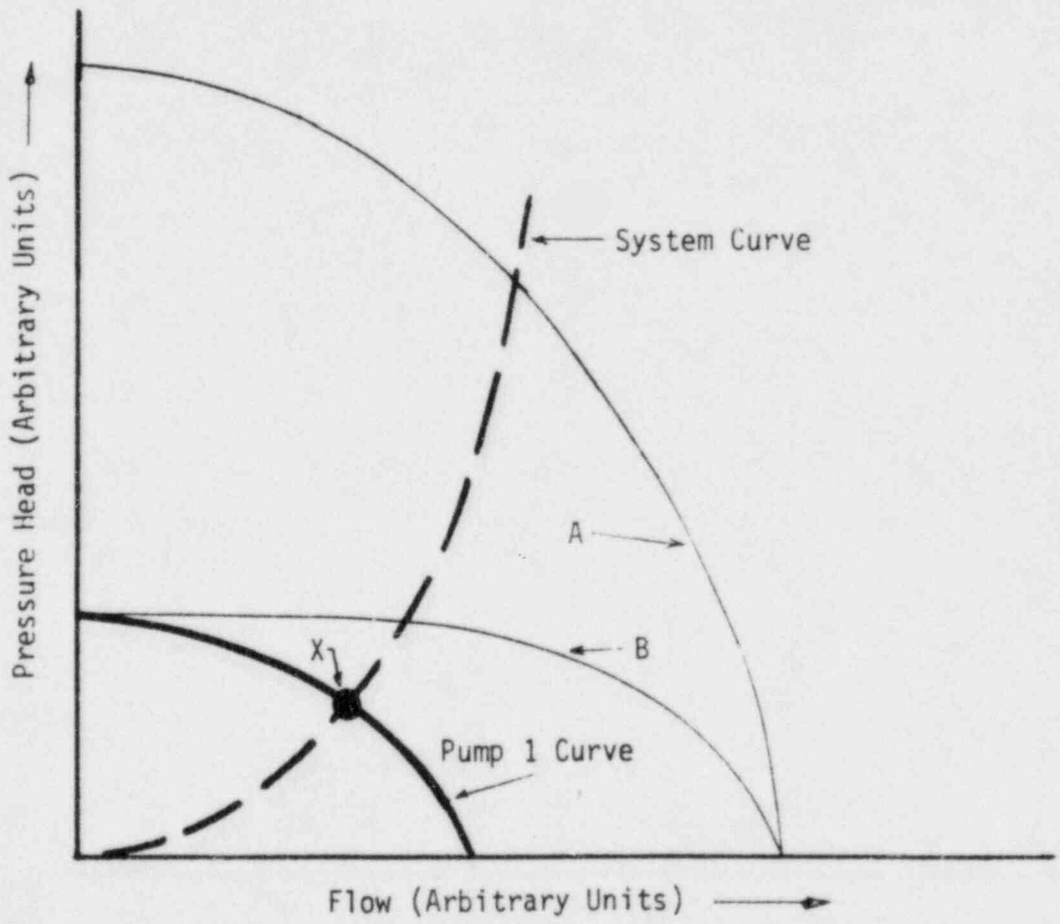
a. HOW MANY times can the procedure be used? (0.5)

b. In WHAT THREE (3) instances is a DEVIATION from the intent of a procedure permitted? (1.5)

FIGURE 1 for Question 5.06

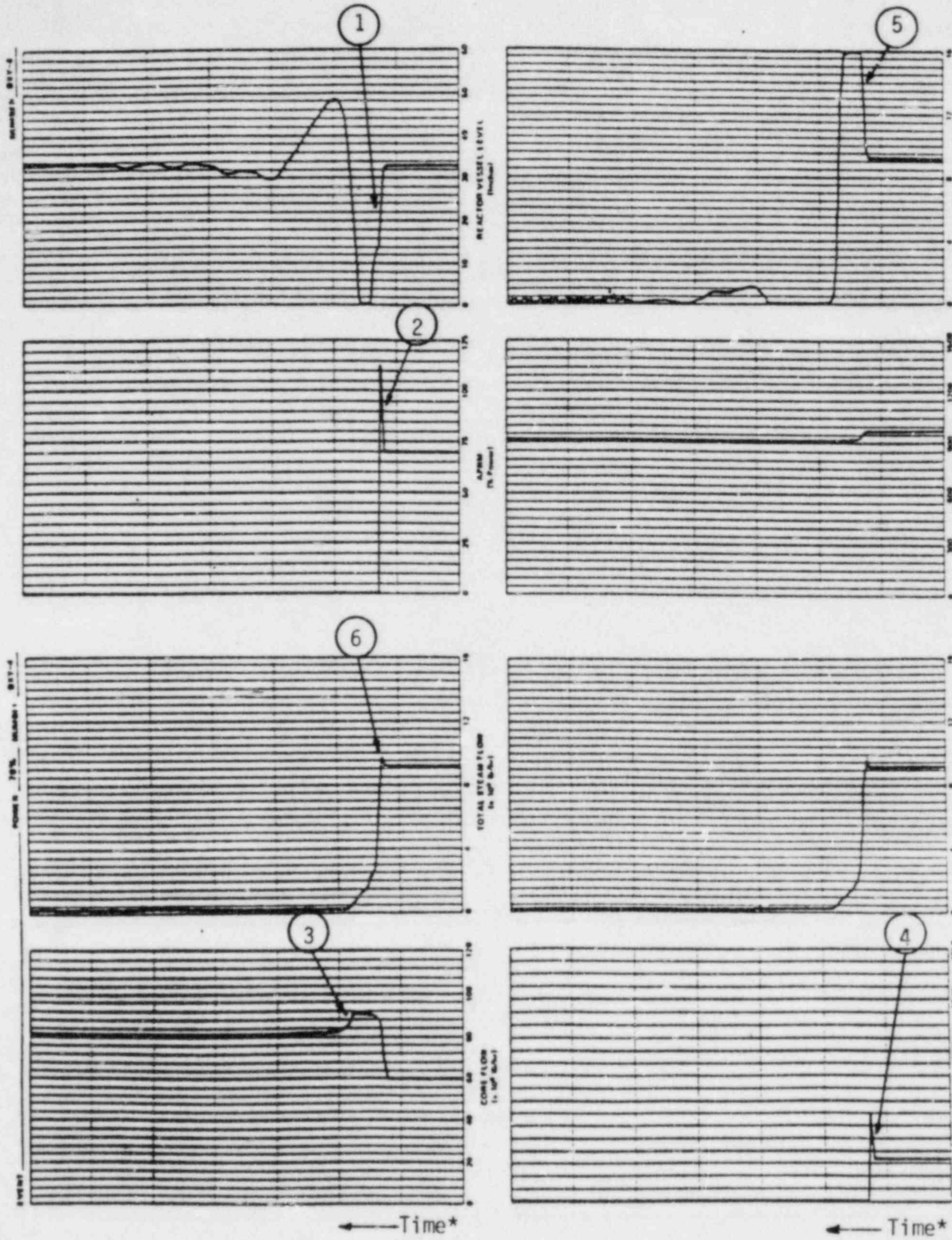


SYSTEM



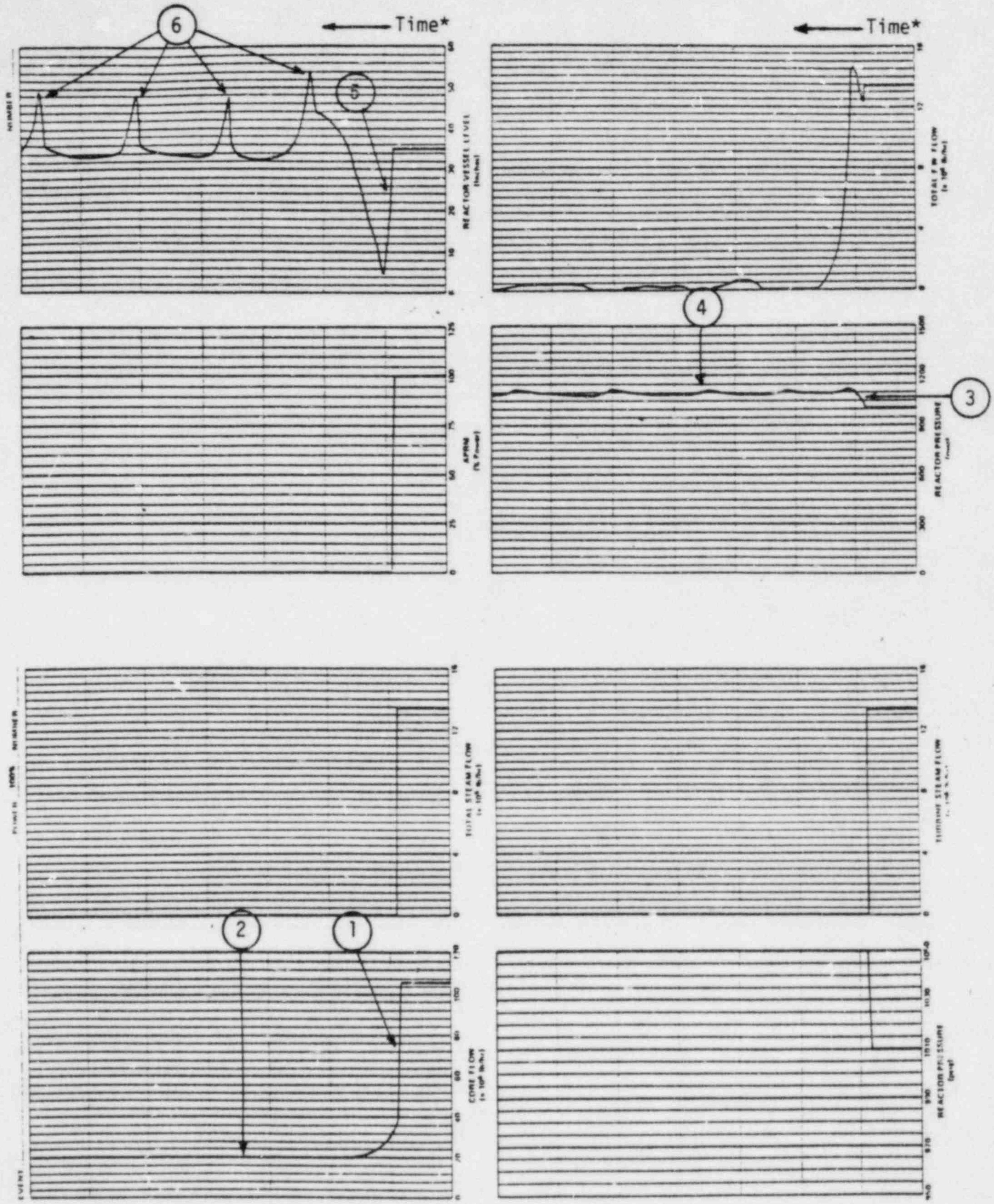
SYSTEM HEAD VS. FLOW PLOT

FIGURE 2 for Question 5.10



*Each time increment is one (1) minute

Figure 3 for Question 5.11



*Each time increment is one (1) minute

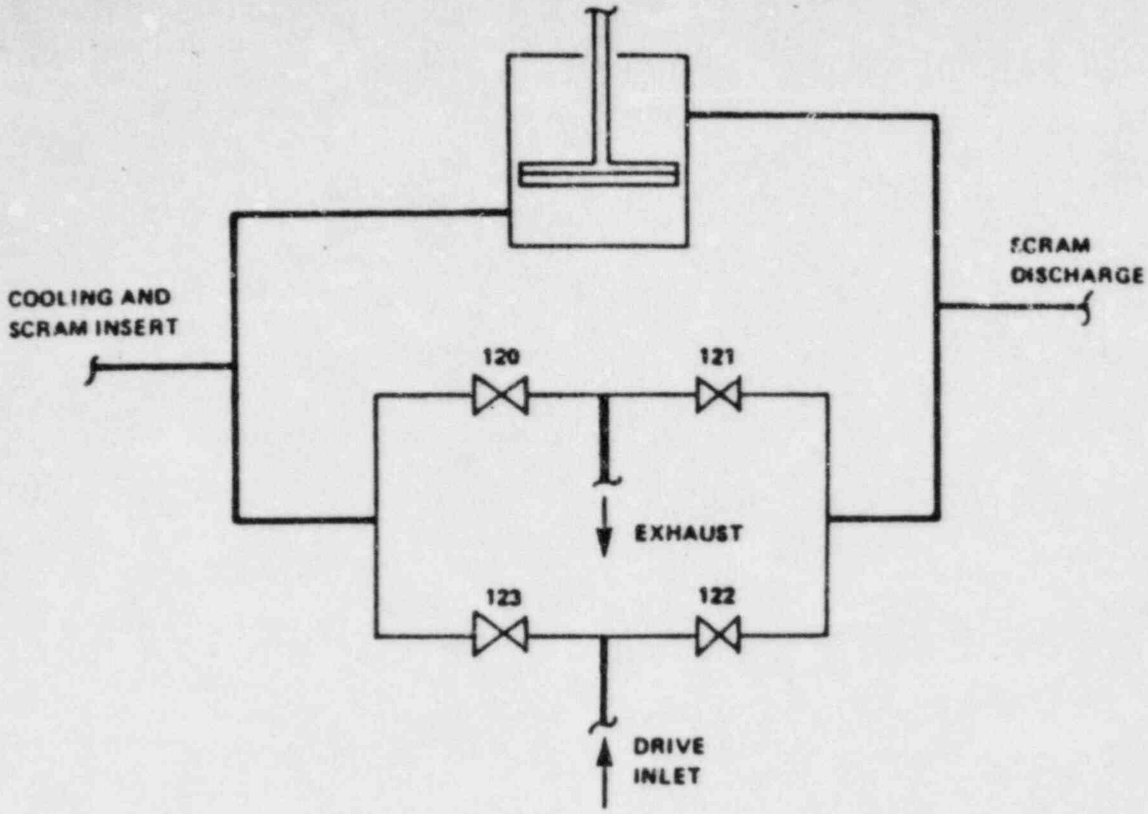


Figure 9.2.1 (1) Control Rod Sequence Timer

3/4.5.3 LOW PRESSURE CORE COOLING SYSTEMS

CORE SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.3.1 Two independent Core Spray System (CSS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE CSS pump, and

b. An OPERABLE flow path capable of taking suction from at least one of the following OPERABLE sources and transferring the water through the spray sparger to the reactor vessel;

1. In CONDITION 1, 2 or 3, from the suppression pool.

2. In CONDITION 4 or 5*;

a) From the suppression pool, or

b) When the suppression pool is being drained, from the condensate storage tank containing at least (150,000) gallons of water.

APPLICABILITY: CONDITIONS 1, 2, 3, 4, and 5*.

ACTION:

a. In CONDITION 1, 2 or 3;

1. With one CSS subsystem inoperable, POWER OPERATION may continue provided both LPCI subsystems are OPERABLE; restore the inoperable CSS subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both CSS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

3. In the event the CSS is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

* The core spray system and the suppression chamber are not required to be OPERABLE provided that the reactor vessel head is removed and the cavity is flooded, the spent fuel pool gates are removed, and the water level is maintained within the limits of Specifications 3.9.9 and 3.9.10

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

b. In CONDITION 4 or 5*;

   1. With one CSS subsystem inoperable, operation may continue provided that at least one LPCI subsystem is OPERABLE within 4 hours; otherwise, suspend all operations that have a potential for draining the reactor vessel.

   2. With both CSS subsystems inoperable, operation may continue provided that at least one LPCI subsystem is OPERABLE and both LPCI subsystems are OPERABLE within 4 hours. Otherwise, suspend all operations that have a potential for draining the reactor vessel and verify that at least one LPCI subsystem is OPERABLE within 4 hours.

   3. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.5.3.1 Each CSS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying the condensate storage tank minimum required volume when the condensate storage tank is required to be OPERABLE.

b. At least once per 31 days by:

   1. Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water, and

   2. Verifying that each valve (manual, power operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 92 days by:

   1. Verifying that each CSS pump can be started from the control room and develops a flow of at least 4625 gpm on recirculation flow against a system head corresponding to a reactor vessel pressure of $\geq$ 113 psig, and

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

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

1. A separate day tank containing a minimum of 900 gallons of fuel,
2. A separate fuel storage tank containing a minimum of 32,000 gallons of fuel, and
3. A separate fuel transfer pump.

APPLICABILITY: CONDITIONS 1, 2, and 3.

ACTION:

a. With either one offsite circuit or one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter;* restore at least two offsite circuits and three diesel generators** to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore at least two offsite cir-

*For the loss of the 2C diesel generator from 9:00 a.m. EST 12/16/81 thru 9:00 a.m. EST 1/3/82, perform Surveillance Requirement 4.8.1.1.1.a at least once per 8 hours, and perform Surveillance Requirement 4.8.1.1.2.a.4 at three day staggered intervals for diesel generators 2A and 1B. The provisions of Specification 3.0.4 do not apply for this change.

**For the loss of the 2C diesel generator from 12/16/81 thru 1/3/82, restore diesel generators 2A and 1B to Operable status. The provisions of Specification 3.0.4 do not apply for this change.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

cuits and three diesel generators** to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of three diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore three diesel generators** to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availablity, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring, manually and automatically, unit power supply from the normal circuit to the alternate circuit.

**For the loss of the 2C diesel generator from 12/16/81 thru 1/3/82, restore diesel generators 2A and 1B to Operable status. The provisions of Specification 3.0.4 do not apply for this change.

EQUATION SHEET

$f = ma$

$v = s/t$

Cycle efficiency = (Net work out)/(Energy in)

$w = mg$

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

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

$W = v\ \Delta P$

$A = \frac{\pi D^2}{4}$

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

$\Delta E = 931\ \Delta m$

$\dot{m} = V_{av} A \rho$

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

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

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

$Pwr = W_f \Delta h$

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

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

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

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

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

$SUR = 26.06/T$

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

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

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

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

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

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

$T = (\beta - \rho)/(\bar{\lambda}\rho)$

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

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

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

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

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

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

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

$I_1 d_1 = I_2 d_2$

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

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

$\Sigma = \sigma N$

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

$R/hr = 6\ CE/d^2$ (feet)

Water Parameters

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.48 gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.
1 ft. H_2O = 0.4335 lbf/in.

Miscellaneous Conversions

1 curie = 3.7×10^{10} dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1 in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)
1 BTU = 778 ft-lbf
e = 2.718

ANSWER SHEET for Question 6.10

INITIAL RESPONSE:

a. TCV position ____________

b. BPV position ____________

c. Power ____________

d. Pressure ____________

Reason: ____

____

____

FINAL STATUS:

a. TCV position ____________

b. BPV position ____________

c. Power ____________

d. Pressure ____________

Reason: ____

____

____

DATE 5-24-93 TIME 0800 HATCH - 1 SEQ. NO. 11

PERIODIC NSS CORE PERFORMANCE LOG

LOCATION	1	2	3	4	5	6	7	8	9	10	11	12
AXIAL REL PWR	0.57	1.08	1.21	1.17	1.14	1.13	1.13	1.14	1.07	1.00	0.82	0.54
REGION REL PWR	0.90	0.99	0.90	1.06	1.23	1.06	0.90	0.99	0.90			
RING REL PWR	1.12	1.16	1.25	1.16	1.12	1.07	0.67					
APRM GAF	0.99	1.00	1.01	1.00	1.00	1.00						

REGION	1	2	3	4	5	6	7	8	9
MFLCPR	0.824	0.779	0.824	0.817	0.898	0.817	0.824	0.778	0.824
LOC	11-18	31-12	41-18	13-22	33-34	39-22	11-36	31-42	41-36
FLOW	0.1200	0.1214	0.1200	0.1216	0.1156	0.1216	0.1200	0.1214	0.1200
PKF	1.31	1.24	1.31	1.26	1.42	1.26	1.31	1.24	1.31
MFLPD	0.860	0.858	0.860	0.812	0.866	0.812	0.860	0.858	0.860
LOC	17-10- 5	31-10- 5	35-10- 5	11-24- 5	21-22- 5	41-30- 5	17-44- 5	31-44- 5	35-44- 5
PKFL	2.08	2.08	2.08	1.57	2.10	1.57	2.08	2.08	2.08
MAPRAT	0.857	0.845	0.857	0.803	0.865	0.803	0.857	0.845	0.857
LOC	17-10- 5	31-10- 5	35-10- 5	11-24-15	21-22- 5	41-30-15	17-44- 5	31-44- 5	35-44- 5
PKFS	1.77	1.78	1.77	1.68	1.85	1.68	1.77	1.78	1.77

CMWT	2421.
PCT PWR	99.4
GMWE	795.6
CMFCP	0.898
CMFLPD	0.866
CMAPR	0.865
CMPF	2.095
CAEQ	0.133
CAQA	0.147
CAVF	0.386
CAPD	49.115
CRD	0.066
CRSYM	2.
PR	1008.45
DPC-M	19.92
DPC-C	24.07
RWL	36.45
DHS	23.30
WFW	9.65
WD	33.12
WTSUB	76.18
WTHB	-1.00
WT	75.52
PCTWTR	96.2
WTFLAG	2.0
ITER	1.0
IREC	0.0
IEQL	1.0
IXYFLG	0.0

FAILED SENSORS 2 4 7

FAILED LPRM LIST BASE CRIT CODE

2805,B,1 1213,D,1 4413,B,1 4413,C,1
2021,A,2 3621,A,2 3621,B,1 4421,C,2
4421,D,1 0437,D,1 1237,D,1

THE 12 MOST LIMITING BUNDLES

FOR MFLCPR				FOR MFLPD				FOR MAPRAT			
MFLCPR	LOC	MCPR	CPRLIM	MFLPD	LOC	MRPD	RPDLIM	MAPRAT	LOC	MAPLHGR	LIMLHGR
0.898	33-34	1.437	1.290	0.866	21-22- 5	11.60	13.40	0.865	21-22- 5	10.22	11.81
0.898	19-20	1.437	1.290	0.866	31-32- 5	11.60	13.40	0.865	31-32- 5	10.22	11.81
0.898	33-20	1.437	1.290	0.866	31-22- 5	11.60	13.40	0.865	31-22- 5	10.22	11.81
0.898	19-34	1.437	1.290	0.866	21-32- 5	11.60	13.40	0.865	21-32- 5	10.22	11.81
0.885	33-32	1.458	1.290	0.860	35-10- 5	11.52	13.40	0.857	35-10- 5	9.75	11.37
0.885	19-22	1.458	1.290	0.860	35-44- 5	11.52	13.40	0.857	35-44- 5	9.75	11.37
0.885	19-32	1.458	1.290	0.860	17-10- 5	11.52	13.40	0.857	17-10- 5	9.75	11.37
0.885	33-22	1.458	1.290	0.860	17-44- 5	11.52	13.40	0.857	17-44- 5	9.75	11.37
0.882	21-34	1.462	1.290	0.858	31-10- 5	11.50	13.40	0.846	33-20- 5	10.24	12.10
0.882	21-20	1.462	1.290	0.858	31-44- 5	11.50	13.40	0.846	33-34- 5	10.24	12.10
0.882	31-34	1.463	1.290	0.858	21-10- 5	11.50	13.40	0.846	19-20- 5	10.24	12.10
0.882	31-20	1.463	1.290	0.858	21-44- 5	11.50	13.40	0.846	19-34- 5	10.24	12.10

THE NUMBER OF BUNDLES WITH MFLCPR GREATER THAN 1.0 = 0
THE NUMBER OF BUNDLES WITH MFLPD GREATER THAN 1.0 = 0
THE NUMBER OF BUNDLES WITH MAPRAT GREATER THAN 1.0 = 0

Figure B

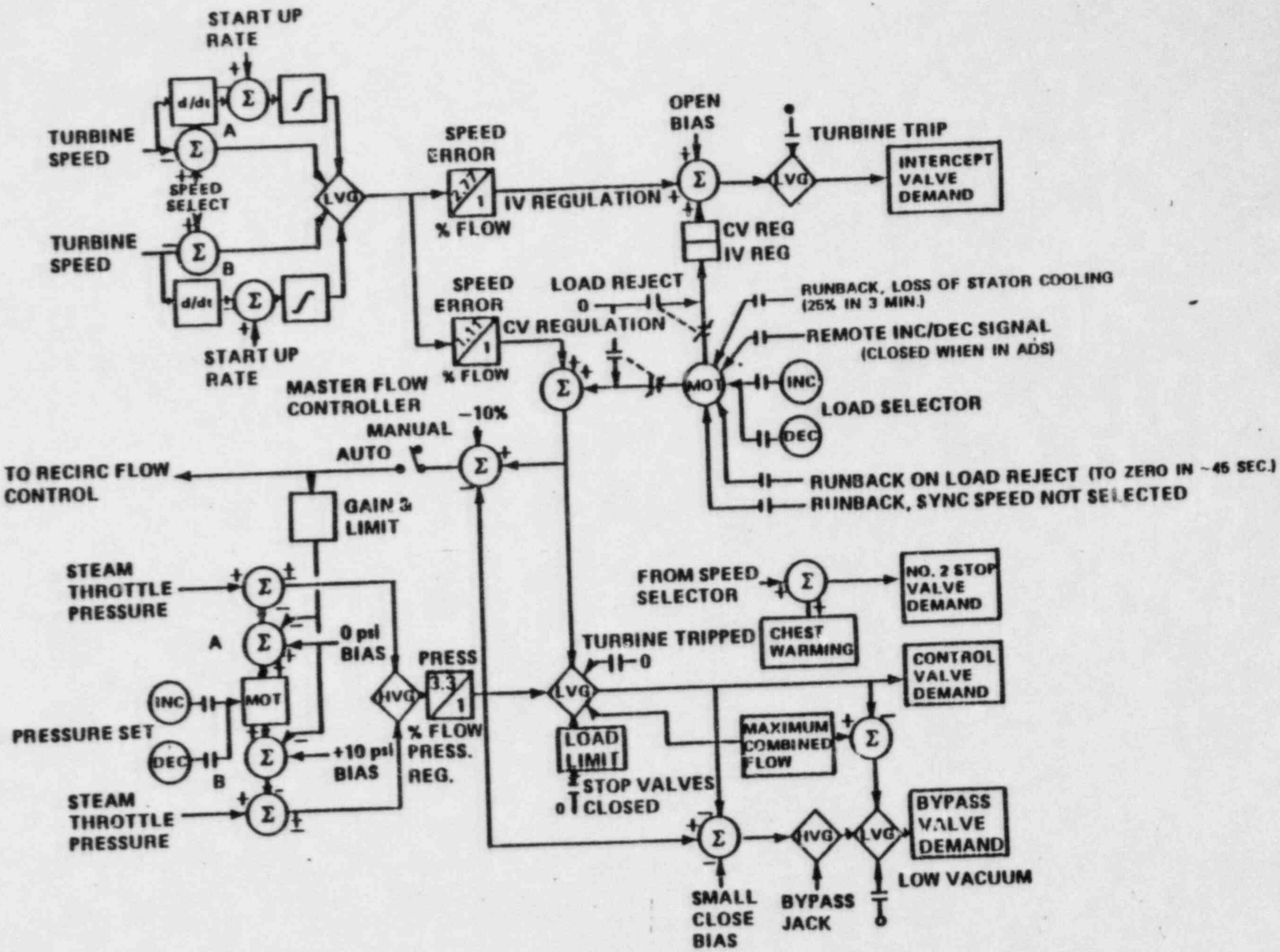


Figure 9.4(7) EHC Logic

ANSWERS -- HATCH 1&2 -84/07/10-PERSONS, R.

MASTER COPY

ANSWER 5.01 (.50)

C. Steam pressure increased, steam enthalpy decreased.

REFERENCE

Steam Tables and

EIH Question Bank, Category 7, No. 26.

ANSWER 5.02 (3.00)

a. Decreases [0.25]. There is less steam flow, therefore, less pressure drop through the main steam lines [0.75]. (1.0)

b. Increases [0.25]. With the same amount of cooling water through the condenser, and less of a heat load, condensate depression will increase [0.75]. (1.0)

c. Decreases [0.25]. Less extraction steam from the turbine to heat the feedwater [0.75]. (1.0)

REFERENCE

EIH Heat Transfer Lesson Plan, pp. 75 & 78, and

EIH Nuclear Training, p. 10.4-11.

ANSWER 5.03 (1.00)

c. LHGR. (0.5)

d. APLHGR. or MAPLHGR (0.5)

REFERENCE

EIH Nuclear Training, Vol. 7, pp. 10.2-9 & 14.

ANSWER 5.04 (2.00)

a. Increases [0.25]. Because each degree increase in moderator temperature results in a larger moderator density decrease [0.75]. (1.0)

b. Decreases [0.25]. Because as control rod density decreases, neutrons leaking from the volume near the fuel rods have less of a chance for non-fission absorption [0.75] (and a greater chance to cause fission). (1.0)

REFERENCE

EIH Nuclear Training, Vol. 7, p. 10.1-68.

ANSWER 5.05 (2.50)

a. The formation of bubbles serves to agitate and break-up [0.5] the relatively stagnant fluid boundary film [0.5]. As nucleate boiling progresses, relatively colder water replaces the gaps on the clad surface left by the bubbles as they detach and move into the coolant stream [0.5]. (1.5)

b. The edge and corner rods can dissipate heat by radiation away from the bundle [0.5] while the central rods radiate much of their heat to other central rods [0.5]. (1.0)

REFERENCE

EIH Nuclear Training, Vol. 7, pp. 10.2-2 & 15.

ANSWER 5.06 (1.50)

a. System operating point. (0.5)

b. Curve B. (0.5)

c. Right. (0.5)

REFERENCE

EIH Thermodynamics Lesson Plan, pp. 88 & 89.

ANSWER 5.07 (2.00)

Power at setpoint = (117%)(2000 MW) = 2340 MW (0.5)

Peak Power = $(2340\ MW)e^{t/T} = (2340)e^{10s/10s} = (2340)e$ (1.0)

Peak Power = 6361 MW. (0.5)

REFERENCE

EIH Nuclear Training, Vol. 7, p. 10.1-62.

ANSWER 5.08 (.50)

Time in hours = $(\%\ power)^{1/2} = (50)^{1/2} = \sim 7$ hr.

REFERENCE

EIH Nuclear Training, Vol. 7, p. 10.1-86.

ANSWER 5.09 (3.00)

a. The decrease in the burnout term [0.5] with the production of Xenon from Iodine still at the higher power rate dominates [0.5] causing the xenon concentration to increase. (1.0)

b. Peripheral rod worth will increase [0.5] because the highest xenon concentration will be in the center of the core where the highest flux existed previously [0.5]. This will suppress the flux in the center of the core and increase the flux in the area of the peripheral rods, thereby, increasing their worth [0.5]. (1.5)

c. More than one half the value at 100% power. (0.5)

REFERENCE

EIH Nuclear Training, Vol. 7, pp. 10.1-79 through 86.

ANSWER 5.10 (3.00)

(1) Due to void collapse caused by the high APRM scram or increased recirc suction flow from the annulus. (0.5)

(2) Due to the increase in recirc flow. (0.5)

(3) Due to the unaffected recirc pump runback to min. when feedflow decreases to <20%. (0.5)

(4) Due to increasing reactor power. (0.5)

(5) Due to FWCS response to decreasing reactor water level. (0.5)

(6) Due to TCVs opening to control reactor pressure. (0.5)

REFERENCE

BWR-4 Transients, BXY-4.

ANSWER 5.11 (3.00)

a. RPT on turbine trip [0.5]. Natural circulation from decay heat [0.5]. (1.0)

b. Turbine BPV's fail to open [0.5]. SRV's control pressure at higher value [0.5]. (1.0)

c. Void collapse due to pressure increase and the scram [0.5]. Level swell from SRV's lifting [0.5] (1.0)

REFERENCE

BWR-4 Transients, DXY-7.

ANSWER 5.12 (2.00)

a. More. (0.5)

b. Maximum LHGR = 0.866 x 13.4 KW/ft = 11.6 KW/ft. (1.0)

c. Bottom. (0.5)

REFERENCE

EIH Nuclear Training, Vol. 7, Chapter 10, and

GEI-92823-B.

ANSWER 6.01 (3.50)

a. Reactor water (leaking past the CRD seals)[0.5] and charging water from the CRDH System [0.5]. (1.0)

b. The affected control rod drifting in [0.5] or CRD high temperature [0.5]. ALSO ACCEPTED SDV HI LEVEL ALARM, ROD BLOCK, OR SCRAM AND HOT OR WARMER PIPING DOWNSTREAM OF SCRAM OUTLET VALVE, (1.0)

c. Unit 2 has a bypass feature (pushbuttons) [0.5] to provide the ability to override OR RESET the LOCA load shed [0.5] and restart the CRD pumps [0.5]. (1.5)

REFERENCE

EIH Nuclear Training, Vol. 5, Chapter 4.2, and

EIH System Differences Lesson Plan, Rev. 1, p. 6, and

NUREG/BR-005/Vol. 5, No. 4, Power Reactor Events, January 1984, p. 5, Event Summary No. 1.2 (event at Hatch Unit 2 on August 25, 1982).

ANSWER 6.02 (3.00)

a. To prevent rapid loss of reactor vessel inventory to the Torus. (1.0)

b. The SDC PCIS Valves (F008 & F009) will auto close [0.5] and the running RHR pump will trip [0.5]. OR HEAD SPRAY VALVE CLOSES. (1.0)

c. The RHR inboard injection valves will not auto open [0.5] and the recirc discharge isolation valves will not auto close [0.5]. (1.0)

REFERENCE

EIH Nuclear Training, Vol. 6, Chapter 8.3, Section C.3.

HNP-2-1114, Rev. 15, p. 6.

ANSWER 6.03 (2.50)

a. 'C' Condensate pump will start (as booster suction pressure reaches 38 psig). (0.5)

b. 'B' Booster pump will trip (at 5 psig) and 'C' Booster pump will start when 'B' trips. (1.0)

c. Both running condensate pumps will trip (at 39") and both running boosters will trip (as their suction pressure decreases to 34 psig). (1.0)

REFERENCE

EIH Nuclear Training, Vol. 6, Chapter 5.4, Section III.D.2, & III.C.

ANSWER 6.04 (2.00)

a. Unit 1 - 'D' [0.5], Unit 2 - 'A' [0.5]. (1.0)

b. Unit 1 - PSW [0.5], Unit 2 - RBCCW [0.5]. (1.0)

REFERENCE

EIH System Differences Lesson Plan, Rev. 1, pp. 6 & 11.

ANSWER 6.05 (3.00)

o Reactor Scram.

o Mechanical vacuum pump trip.

o Mechanical vacuum pump discharge valves isolate. (F010)

o Mechanical vacuum pump suction valves isolate. (F007)

o Gland steam seal exhauster trip.

o Gland steam seal exhauster isolation.

o Control room ventilation swaps to Mode II (pressurization mode).

(6 of 7 at 0.5 each)

REFERENCE

EIH Nuclear Training, Vol. 7, pp. 9.7.1-3 & 4.

HNP-2-1901, Rev. 8, p. 1.

ANSWER 6.06 (2.50)

a. 120 open, 121, 122, and 123 shut. (1.0)

b. o A load on any refueling platform hoist.

o The fuel grapple not fully up.

o The service platform hoist loaded.

o Selection of a second rod for movement with any other rod withdrawn from the fully inserted position.

(3 of 4 at 0.5 each) (1.5)

REFERENCE

EIH Nuclear Training, Vol. 7, pp. 9.2.1-2, 18, & 19, and

Vol. 6, pp. 6.9-8 & 9.

ANSWER 6.07 (1.00)

a. Open.

b. Decrease.

c. Increases. (0.33 each)

REFERENCE

EIH Nuclear Training, Vol. 6, Chapter 5.3, Section III.D, and

Vol. 7, p. 9.5-12.

ANSWER 6.08 (3.00)

a. Because at low power levels, the signal produced by the decay or background gamma overshadows the signals produced by the neutrons and fission gammas. * (SEE NOTE BELOW) (1.0)

b. o Selector switch out of operate.

o High voltage-low.

o Any module unplugged. (0.5 each) (1.5)

c. 0.66W + 51, where W=% recirc loop flow

0.66(50) + 51 = 33 + 51 = 84% (0.5)

REFERENCE

EIH Nuclear Training, Vol. 7, pp. 9.1.1-8 & 12 and 9.1.3-9.

ANSWER 6.09 (2.50)

a. Following the SRV's first actuation, the steam in its discharge line would condense causing a vacuum in the line [0.5]. This would result in suppression pool water being drawn up into the line [0.5] which could cause overpressurization of the line on the next actuation [0.5]. (1.5)
ALSO ACCEPT: WATER HAMMER OR EXCESSIVE JET FORCES (THRUST) ON T'QUENCHER

b. Increases [0.5]. The open vacuum breaker provides a direct path to the drywell [0.5]. (1.0)

REFERENCE

NUREG/BR-005/Vol.5, No. 4, Power Reactor Events, January 1984, p. 5, Event Summary No. 1.2 (event at Hatch Unit 2 on August 25, 1982).

* PARTIAL CREDIT BASIS:

0.25 - AT LOW POWER LEVELS -OR- AT POWER LEVELS OR RANGE WHERE SRM & IRM ARE USED.

0.25 - SIGNALS FROM DECAY OF BACKGROUND GAMMA

0.5 - OVERSHADOW NEUTRON & FISSION GAMMA -OR- ARE NOT PROPORTIONAL TO REACTOR POWER -OR- CAUSE INDICATED POWER TO BE INACCURATE MEASURE OF ACTUAL

ANSWER 6.10 (2.50)

INITIAL RESPONSE:

a. TCVs - Remain at 100% open (or open to 100%) [0.25].

b. BPVs - Open 16.5% [0.25].

c. Power - Decreases [0.25].

d. Pressure - Decreases [0.25].

REASON: Above caused by PCU calling for ~115% steam flow ((950-915) x 3.3) [0.25].

FINAL STATUS:

a. TCVs - At 100% position (or initial) [0.25].

b. BPVs - Shut [0.25].

c. Power - Slightly lower [0.25].

d. Pressure - Slightly lower [0.25].

REASON: Above caused by the decrease in pressure and power causing BPVs to shut -- PCU cycling to new equilibrium state ((945-915) x 3.3) [0.25].

REFERENCE

EIH Nuclear Training, Vol. 7, Chapter 9.4.

ANSWERS -- HATCH 1&2 -84/07/10-PERSONS, R.

ANSWER 7.01 (2.50)

a. 2436 MWt. (0.5)

b. Thermal spike [0.5].
By using OD-3 printouts [0.5]. (1.0)

c. IS YES (600/60 = 10 MWE/min) (0.5)

d. TRUE. (0.5)

REFERENCE

HNP-2-1005, Rev. 12, pp. 1 & 2.

ANSWER 7.02 (1.00)

a. Notch and wait. (0.5)

b. The first rods. (0.5)

REFERENCE

HNP-2-1001, Rev. 21, p. 16.

ANSWER 7.03 (3.00)

a. Running HPCI or RCIC CST to CST. (1.0)

b. o Minimize time in Hot Standby.

o Maintain full RWCU return flow to vessel.

o Hold Hot Standby at 400 psig or lower.

o Bypass a minimum amount of steam to the condenser.

(4 at 0.5 each) (2.0)

REFERENCE

HNP-2-1015, Rev. 2, p. 2, and

EIH Question Bank, Category 9, No. 30.

ANSWER 7.04 (2.00)

a. o Turbine trip

o Reactor scram

o Recirc pumps trip

(3 at 0.5 each) (1.5)

b. With HPCI. (0.5)

REFERENCE

HNP-2-1913, Rev. 7, p. 2.

ANSWER 7.05 (2.00)

a. It is a permissive for the heater power to come on. (1.0)

b. The recombiner blower and heater will shutdown. (1.0)

REFERENCE

HNP-2-1235, Rev. 5, pp. 2 & 4, and

EIH Question and Answer Bank, Category 9, No. 12.

ANSWER 7.06 (3.00)

- o Lack of neutron flux decrease indication on neutron monitors.
- o Lack of FULL IN indication lamps for individual control rods.
- o Improper digital position indication of selected control rod.
- o Reactor power starting to increase, as indicated by nuclear instrumentation and steam production.
- o Shutdown occured, but calculations indicate criticality will occur within the next hour.

(3 of 5 at 1.0 each)

REFERENCE

HNP-2-1909, Rev. 8, p. 1.

EIH Question and Answer Bank, Category 9, No. 10.

ANSWER 7.07 (2.00)

1. Turn the EMERGENCY IN/NOTCH OVERRIDE switch to the EMERG ROD in position and hold for several seconds. Confirm the green full-in light is illuminated. (1.0)
2. Simultaneously turn the EMERGENCY IN/NOTCH OVERRIDE switch to NOTCH OVERRIDE and the ROD MOVEMENT CONTROL switch to ROD OUT NOTCH position. (1.0)

REFERENCE

HNP-2-1933, Rev. 7, pp. 1 & 2.

ANSWER 7.08 (3.00)

a. o Manually initiate automatic actions that should have occurred but did not.

o Ensure primary and secondary containment isolation.

o Ensure diesel generators running.

o Trip main turbine.

(0.5 each) (2.0)

b. Sensing line flashing to steam. (1.0)

REFERENCE

HNP-2-1902, Rev. 18, pp. 2 & 5.

ANSWER 7.09 (1.50)

o The location of the incident,

o The nature (type) of incident, and

o The dose rate of areas involved.

(3 at 0.5 each)

REFERENCE

HNP-4323, p. 1.

ANSWER 7.10 (2.00)

a. 100 degrees F per hour. (0.5)

b. Steam condensing mode of RHR. (0.5)

c. (1) +32

(2) +42

(3) RCIC

(4) RWCU [0.25 each] (1.0)

REFERENCE

HNP-2-1025, Rev. 2, p. 1.

ANSWER 7.11 (2.00)

- o Check reactor scram if greater than 30% power and respond to same.
- o Depress the main turbine trip button.
- o Check that stop valves and CIVS close.
- o Check that generator PCBS and exciter field ACB trips after driving steam is depleted.
- o Check that extraction check valves close and extraction drains open.

(4 of 5 at 0.5 each)

REFERENCE

HNP-2-2001, Rev. 19, p. 4.

ANSWER 7.12 (1.50)

o Decrease in main steam flow indication.

o Increase in steam flow feedwater flow mismatch.

o Amber light lit on open valve.

o Decrease in main generator output and CV position.

o Relief valve discharge temperature recorder upscale.

(3 at 0.5 each)

REFERENCE

HNP-2-1907, Rev. 16, p. 1, and

EIH Question Bank, Category 9, No. 1.

ANSWER 8.01 (3.00)

- o The control rod drive cannot be moved with control rod drive pressure.
- o Control rods which exceed the maximum allowable scram insertion time.
- o A control rod with an inoperable accumulator.
- o Those whose position cannot be positively determined.

(3 of 4 at 1.0 each)

REFERENCE

EIH Unit 1 Tech Specs, pp. 3,3-1, 1a, & 2 (Ammend. 0).

ANSWER 8.02 (2.00)

a. Scram at less than or equal to 1045 psig. (0.5)

b. Nuclear steam system relief valves open as follows:
- o 4 valves at 1080 psig
- o 4 valves at 1090 psig
- o 3 valves at 1100 psig [3 at 0.5 each] (1.5)

NOTE: Number of valves not required for full credit.

REFERENCE

EIH Unit 1 Tech Specs, p. 1.2-1 (Ammend. 39).

ANSWER 8.03 (2.00)

NO [0.5]. Entry into an OPERABLE condition must be made with the full complement of required systems as specified in the LCOs being met [1.0] without regard for provisions contained in the ACTION statements statements [0.5].

REFERENCE

EIH Unit 2 Tech Specs, p. B 3/4 0-1 (Ammend. 8).

ANSWER 8.04 (3.00)

a. o The responsible supervisor cannot be located to release his subclearance personnally [0.66].

o The equipment or system is operable [0.66].

o No work activity is being performed on the equipment or system [0.66]. (2.0)

b. The General Manager [0.5] or Deputy General Manager [0.5]. (1.0)

REFERENCE

HNP-501, Rev. 15, p 1.

ANSWER 8.05 (3.00)

a. Independent verification requires that a second person perform a physical verification of the valve's position [0.5] after the first person has completed the data sheet and returned it to the Shift Supervisor [0.5]. (1.0)

b. Make a visual check of the valve stem position and attempt to turn the valve to the 'closed' position [0.66]. Verify the operability of the locking device by attempting to misposition the valve [0.66]. Check the valve hand wheel integrity [0.66]. (2.0)

REFERENCE

HNP-514, Rev. 1, p. 1.

ANSWER 8.06 (1.00)

The rod should be inserted one notch [0.5] using the EMERGENCY IN switch [0.5].

REFERENCE

EIH Standing Order No. 83-40.

ANSWER 8.07 (2.50)

a. Initiate suppression pool cooling. (0.5)

b. Stop HPCI testing and initiate suppression pool cooling. (1.0)

c. Depressurize the reactor pressure vessel to less than 200 psig at normal cooldown rates. (1.0)

REFERENCE

EIH Unit 2 Technical Specifications, pp. 3/4 6-11 & 12.

ANSWER 8.08 (3.00)

No. It is not correct [0.5], by the definition of operability, RHR pump 'A' is inop. upon loss of its emergency power supply (D/G 2A), a circumstance in excess of those addressed in the action statements of T. S. 3.5.3.1. Thus by T. S. 3.0.3, Unit 2 must be in Hot S/D within 6 hours and in Cold S/D within the following 30 hours [2.5].

(Partial credit of 2 points will be given for missing T. S. 3.0.3 and requiring Hot S/D in 12 hours and Cold S/D in 24 hours.)

REFERENCE

EIH Unit 2 Technical Secification Sections 3.5.3.1 and 3.8.1.1.

ANSWER 8.09 (2.50)

a. Two. (0.5)

b. One of the SRM detectors located in the quadrant where core alterations are being performed [0.5] and the other SRM detector located in an adjacent quadrant [0.5]. (1.0)

c. During core alterations and shutdown margin demonstrations. (1.0)

REFERENCE

EIH Unit 2 Technical Specifications, p. 3/4 9-3 (Ammend. 0).

ANSWER 8.10 (2.00)

a. Once. (0.5)

b. When it is required to:
   o preserve the integrity of reactor fuel [0.5],
   o prevent unnecessary equipment damage [0.5],
   o preserve lives [0.5]. (1.5)

REFERENCE

HNP-9, Rev. 25, pp. 46 & 47.