ENCLOSURE 1

EXAMINATION REPORT

Facility Licensee: Mississippi Power and Light Company
P. O. Box 23054
Jackson, MS 39205

Facility Name: Grand Gulf Nuclear Station

Facility Docket No. 50-416

Written, oral, and simulator examinations were administered at Grand Gulf Nuclear Station near Port Gibson, Mississippi.

Chief Examiner: John F. Munro 7/24/84
John F. Munro Date Signed

Approved by: Munro for BAW 7/24/84
Bruce A. Wilson, Section Chief Date Signed

Summary:

Examinations on June 5-7, 1984

Written, oral, and/or simulator replacement examinations were administered to four ROs; three ROs passed these examinations.

Oral and/or simulator recertification examinations were administered to two SROs and two ROs; all individuals passed these examinations.

8410090473 840731
PDR ADOCK 05000416
Q PDR

REPORT DETAILS

1. Persons Examined

RO Candidates:

S. F. Franco
M. J. Ellis
P. W. Brewer (O/S)
B. R. Cupit (O/S)

SRO Recertification:

G. Lhamon (O/S)
C. Bottemiller (O/S)

RO Recertification:

R. Jacobson (S)
M. Dorsett (S)

Other Facility Employees Contacted:

*J. Yelverton, Manager of Plant Support
*G. Lhamon, Operations Training Supervisor
*R. Fron, Technical Assistant to Manager of Plant Support
*M. Wright, Technical Assistant to Plant Manager for Operations
T. Mayfield, Training Instructor
B. Bryant, Training Instructor
G. Adkins, Training Instructor
V. Stairs, Training Instructor

*Attended Exit Meeting

2. Examiners:

J. Munro, NRC, Chief Examiner
S. Guenther, NRC
C. Dodd, EG&G

3. Examination Review Meeting

At the conclusion of the written examination, the examiners met with facility representatives, G. Lhamon, G. Adkins, and B. Bryant to review the written examination and answer key. The following comments were made by the facility reviewers:

*a. Question 2.02a - The Reactor Mode Switch only gives an alarm and thus would not be a permissive or interlock for automatic initiation.

Resolution - Facility Drawings Nos. E-1220-01 & 02, Rev. 12 & 13, Control Copy #11, confirm the accuracy of the above statement. The answer key has been changed accordingly.

b. Question 2.02a - LOCA setpoint on Suppression Pool Make-Up; if setpoints given, it may cause confusion to grader since one path (LOCA Signal) uses HPCS logic.

Resolution - Facility Drawing No. E-1220-01, Rev. 12, Control Copy #11, confirms that the two LOCA signals referred to originate from two distinct system logics, E12 and M71, and have different numerical setpoints. The question required only the name of the initiating signals, e.g., LOCA; however, if erroneous setpoints are provided, appropriate partial credit will be assigned.

*c. Question 2.07b - +55 inches is not a trip on RCIC.

Resolution - Facility Drawings Nos. E-C1185-06 (Rev. A), E-1185-06 (Rev. 6), E-1185-34 (Rev. 5), E-1185-35 (Rev. 3A), and E-1185-42 (Rev. 5c), all Control Copy #11, confirm the accuracy of the above statement. The question has been deleted from the examination.

*d. Question 2.09 - SLC "B" initiation isolates RWCU valves F001 and F251 only; delete valve F253 as an answer.

Resolution - Facility Procedure 05-1-02-III-5 (Rev. 15) supports the accuracy of the above statement. The answer key has been changed accordingly.

e. Question 3.03e - Facility reviewers questioned whether isolation of "B" & "D" Main Steam Lines would produce a half scram.

Resolution - The facility reviewer designated to provide supporting documentation for comments, verbally indicated that the answer key was correct. No change to the answer key is necessary.

f. Question 3.04b - RHR Pumps will start on initiation signal with RSD switch in emergency.

Resolution - Facility Drawing No. E-1181-43 (Rev. 6), Control Copy #11, indicates that there is no specific transfer switch for RHR Pump 2A (on RSD Panel). Thus, if operator pump control was shifted to the RSD Panel, the RHR Pump would start in the LPCI mode. The answer key has been changed accordingly.

g. Question 4.06 - Drywell Pressure setpoint has been revised to 1.23 psig.

Resolution - Temporary Change Notice #1, dated April 18, 1984, to procedure 05-S-01-EP-3 confirms the accuracy of the above statement. The answer key has been changed accordingly.

*These answer key responses were derived from facility supplied System Descriptions (SD's). These SD's should be corrected to reflect the existing plant conditions.

ENCLOSURE 3

Master

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: _GRAND_GULF_1_____________

REACTOR TYPE: _BWR_____________________

DATE ADMINISTERED: _82/06/05_______________

EXAMINER: _DODD,_C.________________

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
25.00	24.94	__________	________	1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
24.75	24.69	__________	________	2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
25.00	24.94	__________	________	3. INSTRUMENTS AND CONTROLS
25.50	25.44	__________	________	4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
100.25	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)

Given the following formula:

Rod Worth is proportional to (Lf + Ls)(Orod/Oavg.)squared

a. WHICH of the following situations is correct in MOST cases? EXPLAIN your choice. (1.5)

1. A control rod's worth is greatest when it is fully withdrawn and all other rods remain inserted.

2. A control rod's worth is greatest when it is fully inserted with all others withdrawn.

b. EXPLAIN HOW an increase in the void concentration effects rod worth. WHY? (1.0)

QUESTION 1.02 (3.50)

For the power history below, SKETCH a curve of core xenon concentration versus time. NOTE: TIME IS IN INCREMENTS OF 10 HOURS.
Time zero is XENON FREE.

```
PERCENT  100 -

REACTOR   50 -

POWER      0 -  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .
                0  1  2  3  4  5  6  7  8  9  10 11 12 13 14 15 16
                          TIME IN HOURS (x10)
```

QUESTION 1.03 (3.00)

A. After making a rod notch with the reactor critical, you notice a 100 second period. HOW MUCH reactivity was added by the rod notch? (ASSUME BOL, SHOW ALL CALCULATIONS) (1.0)

B. After a reactor scram from power the shortest STABLE period possible is -80 seconds. EXPLAIN this statement. (1.0)

C. Is the INITIAL period IMMEDIATELY following the scram shorter than -80 seconds? EXPLAIN. (1.0)

QUESTION 1.04 (3.00)

MATCH the Failure Mechanism from column (1) AND the Limiting Condition from column (2) WITH the associated Power Distribution Limits (a-c) below.

a. Linear Heat Generation Rate (LHGR)
b. Average Planar Linear Heat Generation Rate (APLHGR)
c. Minimum Critical Power Ratio (MCPR)

1 - FAILURE MECHANISM	2 - LIMITING CONDITION
1. FUEL CLAD CRACKING DUE TO LACK OF COOLING CAUSED BY OTB	1. 1% PLASTIC STRAIN
2. FUEL CLAD CRACKING DUE TO HIGH STRESS FROM PELLET EXPANSION	2. PREVENT TRANSITION BOILING
3. GROSS CLAD FAILURE DUE TO DECAY HEAT & STORED HEAT FOLLOWING A LOCA	3. LIMIT CLAD TEMP TO 2200 F

(3.00)

QUESTION 1.05 (1.00)

The following statements are concerned with subcritical multiplication. CHOOSE the one CAPITALIZED word that will make the sentence correct.

a. As Keff approaches unity, a LARGER/SMALLER change in neutron level occurs for a given change in Keff. (0.5)

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

QUESTION 1.06 (2.00)

Consider a turbine trip from approximately 29% reactor power. DESCRIBE what would be expected to occur in the plant during the next 15 minutes with no operator action. INCLUDE significant parameters such as, temperature(s), pressure, power, etc. Provide explanation(s) for plant & parameter conditions. (2.0)

QUESTION 1.07 (2.50)

a. Approximately WHAT percentage of neutrons from U-235 are born delayed? (0.5)
b. HOW does the percentage of delayed neutrons produced in the CORE vary over core life and WHY? (1.0)
c. HOW do delayed neutrons contribute to the control capability of a commercial reactor? (1.0)

QUESTION 1.08 (1.50)

Using the Steam Tables provided, CALCULATE a reactor cooldown rate assuming an initial reactor pressure of 985 psia and a reactor pressure of 385 psia one hour later. SHOW ALL WORK.

QUESTION 1.09 (2.50)

WHAT design feature in the reactor vessel ensures proper flow distribution through the core fuel bundles? EXPLAIN what would happen, AND WHY it would happen, on a power increase with 'NO CHANGE IN RECIRC FLOW' if this feature were eliminated. (2.5)

QUESTION 1.10 (1.50)

EXPLAIN HOW nucleate boiling improves the heat transfer characteristics of the reactor core. (1.5)

QUESTION 1.11 (2.00)

a. DEFINE 'Condensate Depression' (0.5)

b. EXPLAIN WHY is it necessary for plants to operate with Condensate Depression? (1.0)

c. HOW would CYCLE EFFICIENCY be effected if the amount of condensate depression is REDUCED? (0.5)

QUESTION 2.01 (3.00)

a. Upon completion of a reactor scram, with all CRD's fully inserted, WHAT are the two sources of water continuing to fill the scram discharge volume until the scram has been reset? (1.0)

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

c. DESCRIBE in detail (OR TRACE on attached Fig. 2.01) the flow path of exhaust water from the CRD mechanism following a normal rod insertion. INCLUDE the specific component(s) or system section(s) the water travels through until it is no longer in the CRD system. (1.0)

QUESTION 2.02 (3.50)

a. WHAT signals will cause an AUTOMATIC initiation of the suppression pool makeup system? (INCLUDE ALL permissives and/or interlocks) (SETPOINTS NOT REQUIRED.) (1.5)

b. WHY is inadvertent dumping of the upper pool to the suppression pool undesirable? (TWO REASONS REQUIRED.) (2.0)

QUESTION 2.03 (2.50)

a. During a TURBINE STARTUP, turbine speed is limited by WHAT TWO components, devices or signals and WHAT determines which one is controlling? (1.5)

b. After the Load Reference Controller is in control of the turbine, WHAT TWO conditions/events will cause the SPEED CONTROLLER to resume control? (1.0)

QUESTION 2.04 (2.50)

a. The HPCS Service Water loop supplies WHAT loads? (TWO required) (1.0)

b. WHAT FOUR signals will cause the HPCS Service Water loop to start automatically? (setpoints NOT required) (1.0)

c. WHAT must be done before the HPCS Service Water Pump can be manually tripped following an automatic start? (0.5)

QUESTION 2.05 (1.50)

WHAT are THREE systems monitored by the Process Liquid Radiation Monitoring subsystem? (1.5)

QUESTION 2.06 (3.50)

With regard to the Low Pressure Core Spray (LPCS) System:

a. INDICATE whether the following statements are True or False:
   1. When a manual override is performed on a pump or valve, automatic pump restart or valve opening is disabled until the indications are reset. (0.5)

   2. Manually starting the LPCS pump automatically establishes an adequate Standby Service Water flow path. (0.5)

b. LIST 4 of the 5 actions that occur in the LPCS system on an AUTO initiation. (1.0)

c. FILL IN the blanks on the attached logic diagram (figure 2.06) for a LPCS pump auto start WITH the appropriate signal names AND setpoints. (1.5)

QUESTION 2.07 (2.25)

a. For the following conditions WILL the RCIC system INJECT if the condition was present at the time that an automatic initiation signal was received? (JUSTIFY YOUR ANSWER.)

   1. Turbine steam exhaust valve (F068) is NOT FULLY OPEN. (0.75)

   2. RCIC discharge flow element (FE-N001) has its EQUALIZING VALVE OPEN. (0.75)

b. Assume the RCIC turbine trip & throttle valve trips for each of the following conditions. CAN the RCIC turbine be restarted FROM THE CONTROL ROOM? EXPLAIN HOW reset is accomplished regardless of where.

   1. Overspeed trip of 110%. (0.75)

QUESTION 2.08 (3.00)

Concerning Grand Gulf's Fire Protection System:

a. WHAT will automatically start each of the THREE (3) fire protection pumps? BE SPECIFIC - SETPOINTS REQUIRED (2.0)

b. Explain HOW an Automatic Pre-Action Deluge System operates. (1.0)

QUESTION 2.09 (3.00)

With regard to the RWCU System:

a. On the two attached RWCU line diagrams, trace the flow pa ns for the 'pre-pump' AND 'post-pump' modes of operation. Trac one flow path on each diagram AND ensure appropriate labeling. A UME there is no blowdown flow and ALL water is returned to the reactor.(2.0)

b. On one of the attached figures circle ALL the RWCU isolation valves which will AUTO close as a result of a SBLC system 'B' initiation. (1.0)

QUESTION 3.01 (3.00)

Assume the FEEDWATER LEVEL CONTROL SYSTEM is being operated in 3-ELEMENT control using reactor LEVEL DETECTOR CHANNEL 'A'. Reactor power is at 85%, STEADY STATE.
For each of the instrument or control signal failures listed below, STATE HOW REACTOR LEVEL WILL INITIALLY RESPOND (increase, decrease, or remains constant) and BRIEFLY EXPLAIN WHY in terms of WHAT is happening in the Feedwater Control System IMMEDIATELY AFTER THE FAILURE.

(FOR EXAMPLE, your answers should include the following detail, 'Causes reactor level to decrease due to a steam flow/feed flow error signal, steam flow < feed flow, resulting in a signal to increase the speed of the reactor feedpump(s),' IF APPLICABLE.)

a. B FEEDWATER line FLOW signal fails HIGH (1.0)

b. Channel A REACTOR LEVEL detector signal fails LOW (1.0)

c. LOSS OF CONTROL SIGNAL to B Reactor Feed Pump Speed Controller (1.0)

QUESTION 3.02 (4.00)

a. What actuates the red/green SRV indicating lights on 1H13-P601? (satpoint not required) (1.0)

b. If the switch for an SRV at the Remote Shutdown Panel is in the OFF position:

   1. In what AUTOMATIC modes, if any, will the SRV function (safety, relief, ADS)? (0.5)
   2. If the switch in the control room is taken to the OPEN position, will the valve open? (0.5)

c. When manually initiating ADS using the armed pushbuttons, what initiation logic requirements are BYPASSED? (1.0)

d. WHAT is/are the power supply(s) to the ADS logic AND solenoids? e.g. for example: 480VAC, RPS 120VAC, 125 VDC, etc.

   - Logic --------- (0.5)

   - Solenoids --------- (0.5)

QUESTION 3.03 (2.50)

For each of the following, state whether a ROD BLOCK, HALF-SCRAM, FULL SCRAM, or NO PROTECTIVE ACTION is generated for that condition. NOTE: IF two or more actions are generated, i.e. rod block and a half-scram, state the most severe, i.e. half-scram.

a. APRM B Downscale, Mode Switch in RUN (0.5)
b. 12 LPRM inputs to APRM C, Mode Switch in STARTUP (0.5)
c. Flow Units A and B Upscale (>108% flow), Mode Switch in RUN (0.5)
d. Reactor water level 38", Reactor power 18%, Mode Switch in RUN (0.5)
e. Main Steam Lines B and D ISOLATED, Mode Switch in RUN (0.5)

QUESTION 3.04 (2.50)

With regard to the RHR/LPCI System:

a. A break occurs in a recirculation loop. RHR initiates in the LPCI mode. After a period of time, the LPCI injection valve (F042A) starts to go shut. WHAT is happening? EXPLAIN the system functioning (INCLUDING signals and setpoints) that causes this condition. Assume no operator action and all systems are functioning as intended. (2.0)

b. If RHR pump control has been transferred to the Remote Shutdown Panel and a LPCI initiation signal is received, will the transferred pumps start in the LPCI mode? (answer yes or no) (0.5)

QUESTION 3.05 (3.50)

With regard to Nuclear Instrumentation System:

a. Consider SRM detector location
   1. On the attached figure 3.05A, indicate the location of the SRM detectors by identifying the alphabetical designator in the circle. (0.6)
   2. What is the AXIAL position, with respect to the core, of these detectors during a reactor startup? (Indicate ~ distance) (0.4)

b. For the IRM range that follows INDICATE the expected level AND any automatic action(s) that will take place. Switching from Range 5, reading 25, up to Range 7.
   NOTE: Figure 3.05B (Range Scale) provided (1.0)

c. WHICH AGAF value (P1 printout) is more conservative? (0.5)
   1. - .99
   2. - 1.01

d. The Channel Calibration has just commenced on Rx Recirc Flow Channel 'A'. WHAT, if any, alarm(s) and/or automatic action(s) will occur as the Flow Unit Mode Switch is placed in 'External Test' (1.0)

QUESTION 3.06 (4.50)

With regards to Recirculation & Recirculation Flow Control:

a. WHAT action(s) result in the recirc system if the 'System A/B Drywell High Pressure Interlock' is actuated during operation at high power in 'Flux Auto'? (1.0)

b. WHAT are the THREE (3) inputs to the 'Reactor Vessel Thermal Shock Interlock'? (Include SETPOINTS required to satisfy this interlock.)(1.5)

c. WHAT condition(s) actuate(s) the End of Cycle RPT trip and WHY is this trip necessary? (2.0)

QUESTION 3.07 (3.00)

a. BRIEFLY EXPLAIN what condition will generate EACH of the following indications on the Operator Control Module.

1. Data Fault (0.5)

2. Scram Valves (0.5)

3. Channel Disagree (0.5)

4. Insert Required (0.5)

b. Above the HPSP, continuous withdrawal of a control rod is limited to ___________ . (0.5)
If a rod runs into this block and is then deselected and reselected, can it be withdrawn further or is its motion still blocked? (0.5)

QUESTION 3.08 (2.00)

The Area Radiation Monitors have installed check sources which, when activated or deactivated will provide an indication of an ARM's operability. BRIEFLY describe HOW operability is demonstrated with the check source in BOTH the activated & deactivated condition. (2.0)

QUESTION 4.01 (3.50)

Answer the following questions concerning Integrated Operating Instruction 03-1-01-1, 'Cold Shutdown to Generator Carrying Minimum Load'

a. WHEN is it recommended that continuous withdrawal mode of rod movement NOT be used? (INCLUDE all circumstances or conditions) (2.0)

b. LIST FOUR (4) conditions which must be met/satisfied to place the reactor mode switch in RUN. (INCLUDE SETPOINTS IF APPLICABLE) (1.5)

QUESTION 4.02 (3.50)

In accordance with ONEP 05-1-02-V-8 (Loss of Condenser Vacuum):

a. Other than a decreasing vacuum, WHAT are FOUR (4) possible symptoms or indications of a lowering main condenser vacuum? (EXCLUDE annunciators and alarms.) (2.0)

b. LIST THREE (3) automatic actions that occur DIRECTLY as a result of decreasing condenser vacuum. (Setpoints NOT required, BUT NOTE ORDER of OCCURRENCE) (1.5)

QUESTION 4.03 (1.50)

The plant is operating at 40% power. CRD Hydraulic pump B is out of service being repaired. CRD Hydraulic pump A trips and will not restart. HOW LONG may operation continue and HOW is the plant shutdown per the CRD Malfunction ONEP? (Assume CRD pumps are NOT returned to service.) (1.5)

QUESTION 4.04 (3.00)

The Shift Supervisor has determined that the control room should be evacuated.

a. WHAT actions or verifications should be performed by the Control Room Operator PRIOR to leaving the control room? (2.0)

b. WHAT are THREE (3) systems that can at least be partially operated from the Remote Shutdown Panel, OTHER THAN SRV's and RHR sys.? (1.0)

QUESTION 4.05 (2.50)

Concerning the Emergency Procedure for Cooldown, (EP-2):

a. LIST FOUR systems which may be used for RPV cooldown if the main condenser is NOT available? INCLUDE the operating mode of the system if applicable. (2.0)

b. WHAT is the preferred system for MAINTAINING RPV water level?(0.5)

QUESTION 4.06 (2.50)

Regarding the Containment Control Emergency Procedure (EP-3):

LIST 5 of the 6 entry conditions, WITH SETPOINTS, for this procedure. (2.5)

QUESTION 4.07 (4.00)

a. WHAT TWO components or systems will/may isolate on a complete loss of Component Cooling Water? (1.0)

b. WHAT are THREE immediate operator ACTION STEPS to be taken for a complete loss of CCW? (EXCLUDE investigations, verifications, and notifications). (2.0)

c. WHEN is a partial loss of CCW treated as a complete loss of CCW? (1.0)

NOTE: ACTION STEPS may have multiple actions and/or checks.

QUESTION 4.08 (5.00)

Give the reason(s) for EACH of the following Procedural Precautions and/or Limitations:

a. ONEP 05-1-02-IV-1, 'Control Rod/Drive Malfunction', cautions not to exceed 350 psi drive water differential pressure. (1.0)

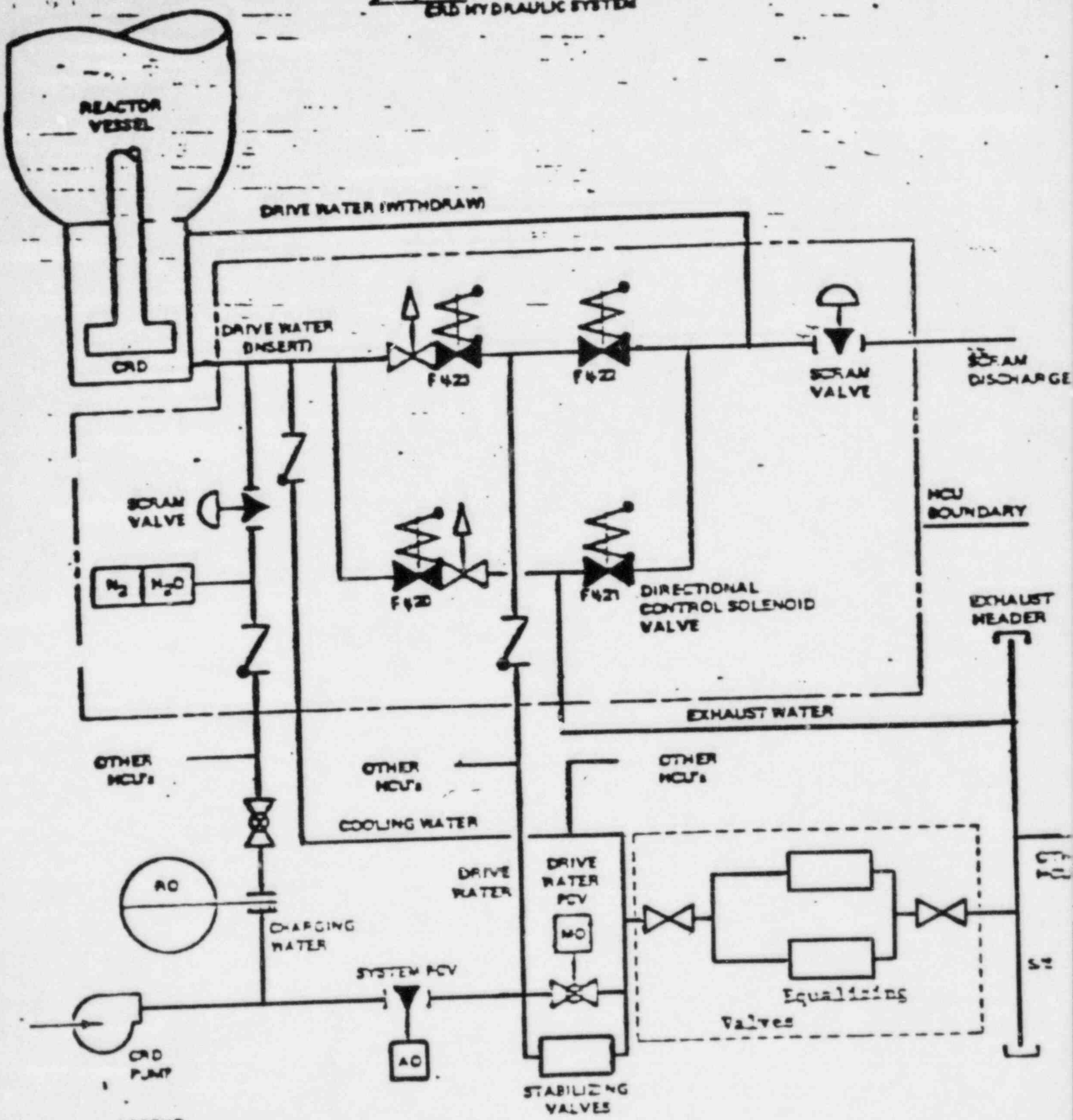
b. SOI 04-1-01-G33-1, 'Reactor Water Cleanup', cautions to always operate the RWCU system in 'pre-pump' mode whenever the reactor is at <25 psig pressure. (1.0)

c. SOI 04-1-01-E12-1, 'RHR System', cautions not to operate the RHR system in LPCI mode unless required during an emergency condition. (1.0)

d. SOI 04-1-01-P75-1, 'Standby Diesel Generator', cautions not to operate the Diesel Generator without air pressure. (1.0)

e. IOI 03-1-01-3, 'Plant Shutdown', cautions when shutdown cooling is in operation not to allow shutdown cooling flow to decrease below 1000 gpm. (1.0)

SIMPLIFIED SCHEMATIC OF
CRD HYDRAULIC SYSTEM



LEGEND

Figure 2.01

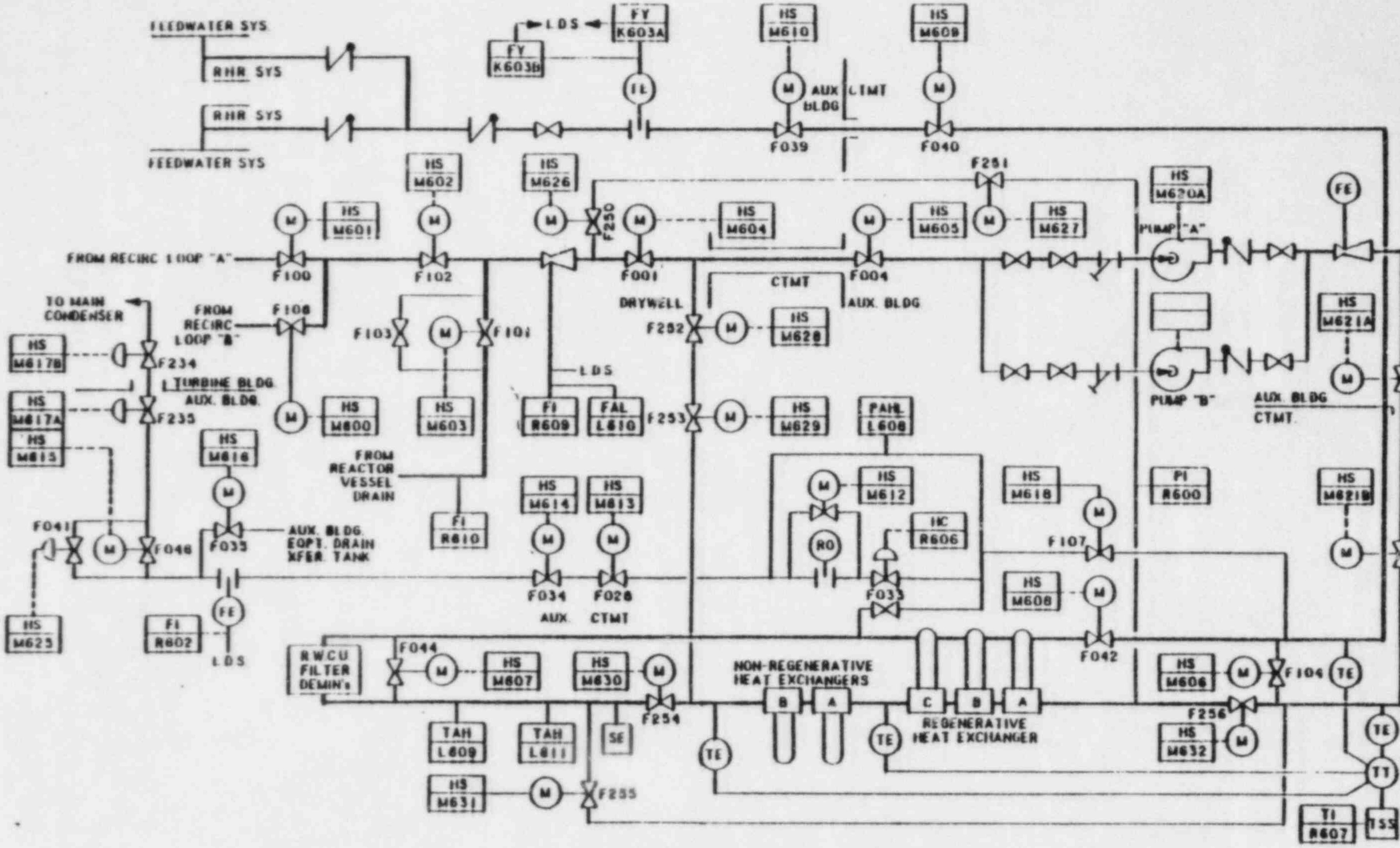


FIGURE I REACTOR WATER CLEANUP SYSTEM

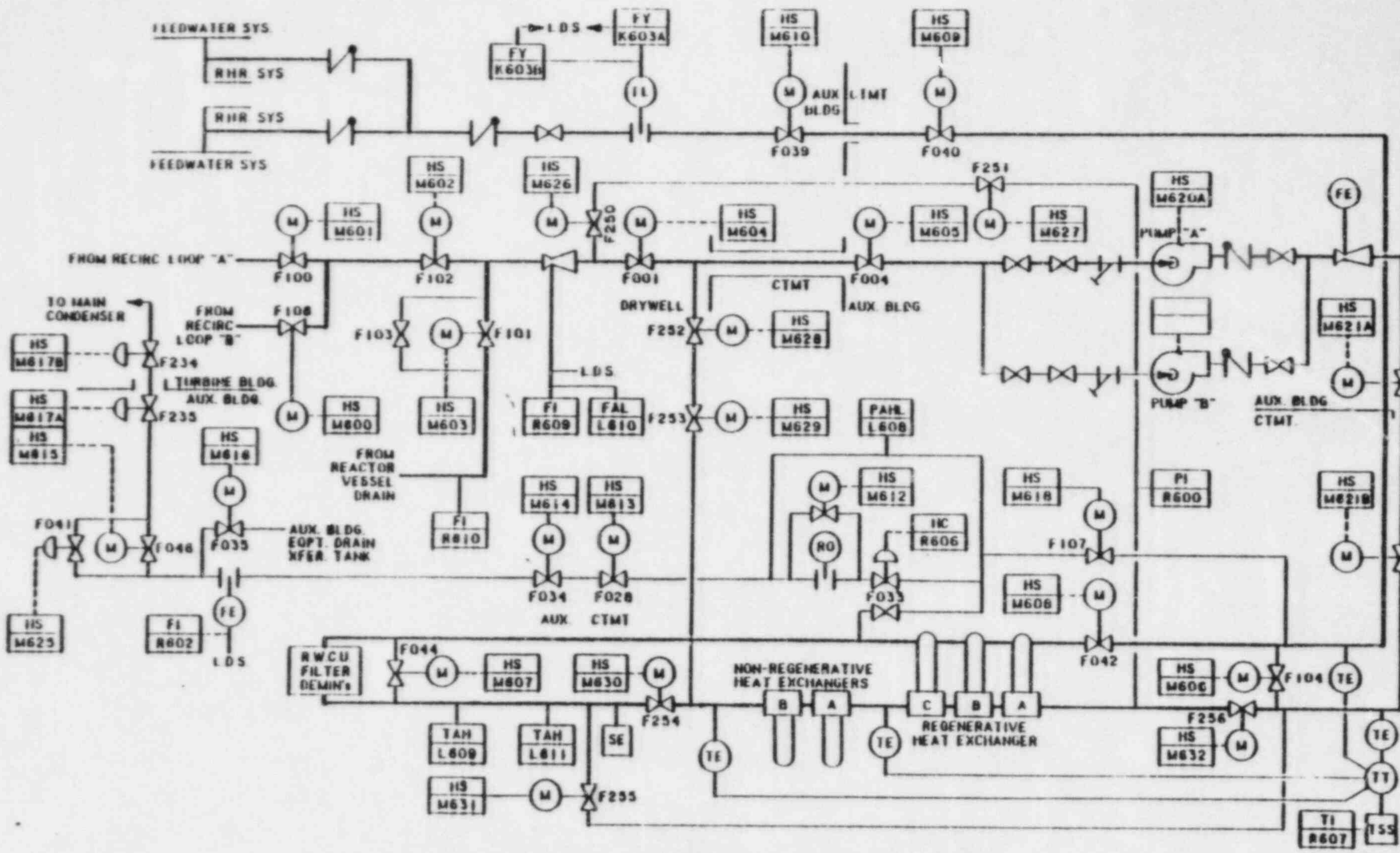


FIGURE 1 REACTOR WATER CLEANUP SYSTEM

VI. C. <u>Trips, Permissives and Interlocks</u>

3. LPCS Pump

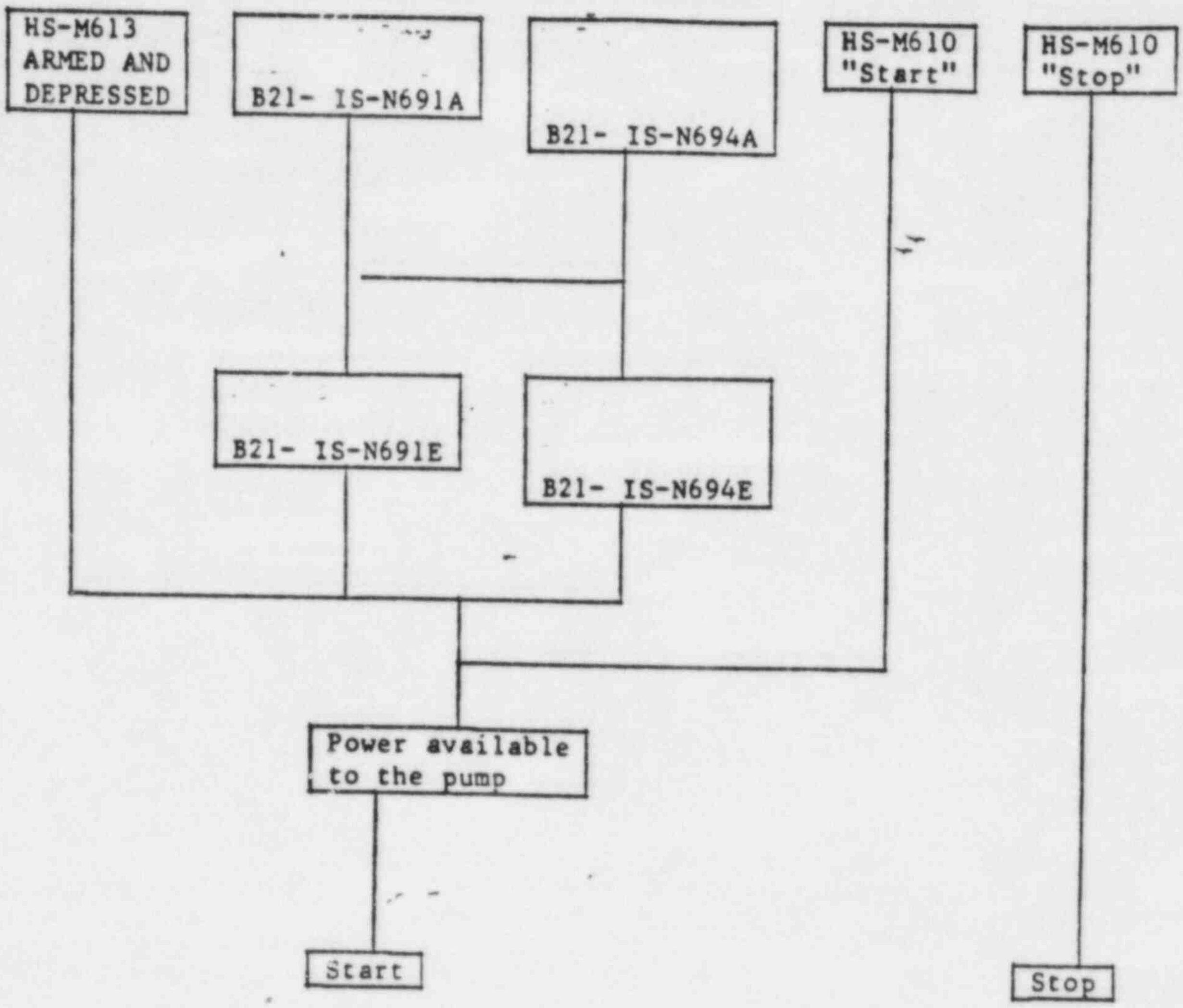
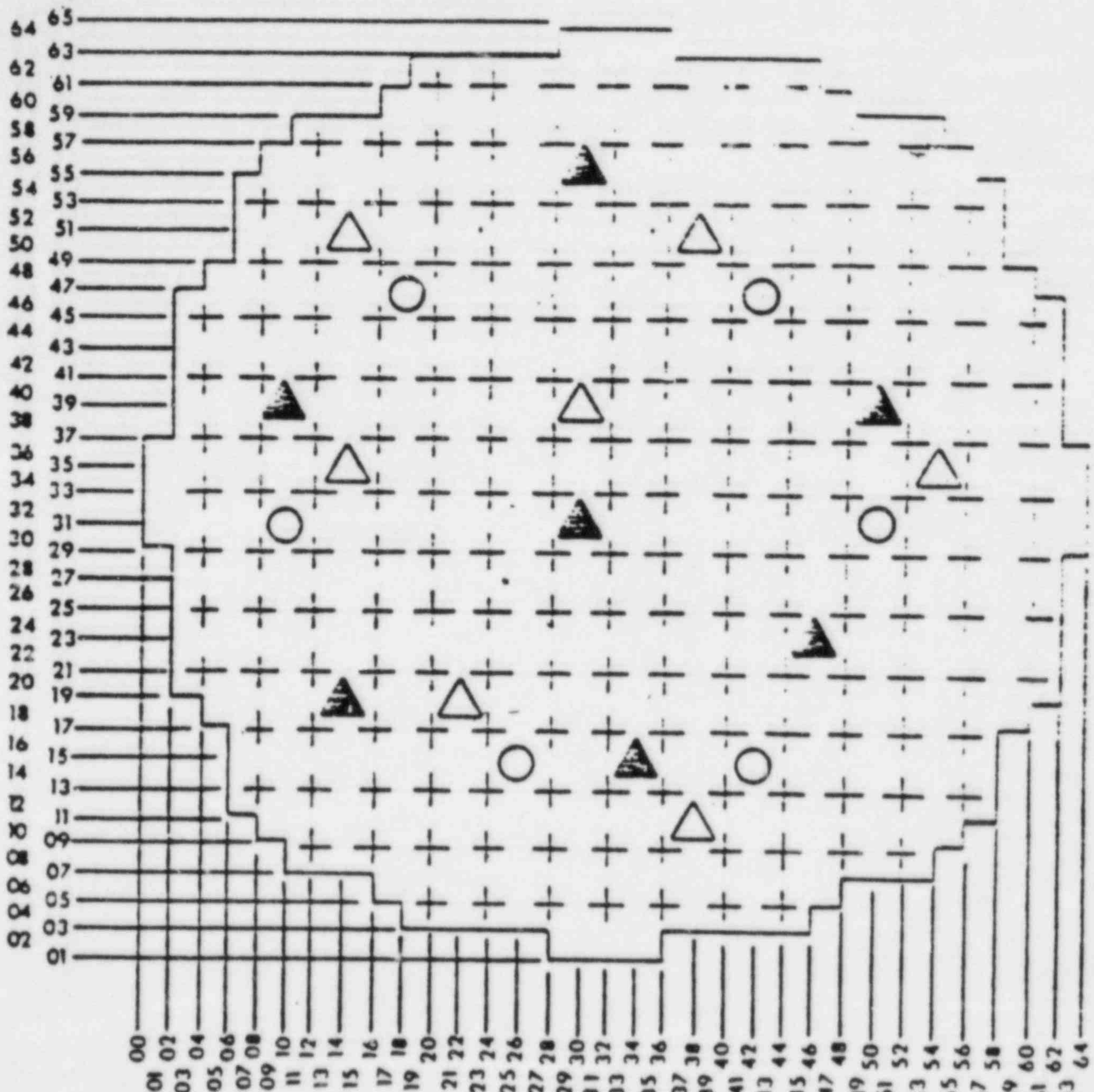


Figure 2.06



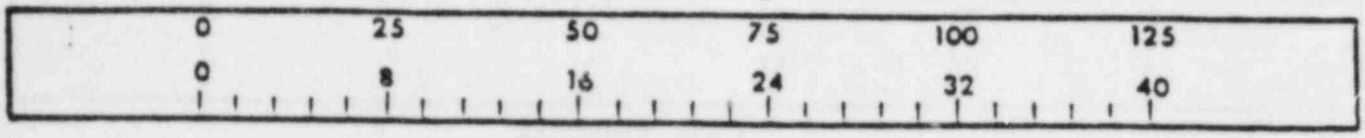
○ SRM DETECTOR LOCATIONS

▲ NEUTRON SOURCE LOCATIONS

△ SPARE NEUTRON SOURCE LOATIONS (FOR POSSIBLE FUTURE USE)

Figure 3.0(2) Source And SRM Detector Assembly Core Positions

Figure 3.05 (b)



RECORDER SCALE

EQUATION SHEET

$f = ma$

$w = mg$

$E = mc^2$

$KE = 1/2\ mv^2$

$PE = mgh$

$V_f = V_o + at$

$W = v\ \Delta P$

$\Delta E = 931\ \Delta m$

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

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

$Pwr = W_f\Delta h$

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

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

$SUR = 26.06/T$

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

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

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

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

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

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

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

$\Sigma = \sigma N$

$v = s/t$

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

$a = (V_f - V_o)/t$

$\omega = \theta/t$

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

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

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

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

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

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

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

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

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

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

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

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

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

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

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

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

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

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

$I_1 d_1 = I_2 d_2$

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

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

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

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

ANSWER 1.01 (2.50)

a. 1. is the correct answer [0.5]. The large rod worth is due to the high flux created at the rods location, while the average neutron flux remains very low [1.0].

b. An increase in voids results in a decrease in rod worth[0.5] due to changes in Ls and Lf.[0.5]

REFERENCE
Grand Gulf Rx Physics pg-39

ANSWER 1.02 (3.50)

```
                        ------[1.0]------.--[1.0]-.--------[1.0]--------
          100% Peak -                    .        .
 XENON                                   .        .
                                         .        .
 CONC.     100% Eq. -                    .        .
            50% Eq. -                    .        .
                                         .        .
                                         .        .
                    0 -  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .
                         0  1  2  3  4  5  6  7  8  9  10 11 12 13 14 15 16
                                    - -  TIME IN HOURS (x10)
```

[0.5] for the axes.

REFERENCE
Grand Gulf Rx Physics pg. 41-44

ANSWER 1.03 (3.00)

A. T = B-p/ p so p = B/ T + 1

Assume B̄ = 0.0072 (BOL)[0.1]

p = 0.0072/(100)(0.1) + 1 = 6.545 X 10 E-4 DK/K[0.9]

B. = Ln 2/t$_{1/2}$ = 0.693/55.6 = 0.0125 sec E-1

T = 1/- = 1/-0.0125 = -80 sec.
After the initial prompt drop, power cannot decrease faster than the longest lived delayed neutron appears.[1.0]
(Calculation not required for full credit.)

C. Yes,[0.5] the initial drop in power will only be due to prompt neutrons.[0.5] (and could be calculated by T = %1/p)

REFERENCE
Grand Gulf Rx Physics pg. 31-36

ANSWER 1.04 (3.00)

	Failure Mechanism	Limited Condition
A. LHGR	2	1
B. APLHGR	3	3
C. MCPR	1	2

(6 answers req. @ 0.5 each)

REFERENCE
Grand Gulf Mitigation of Core Damage L.P. ch.2

ANSWER 1.05 (1.00)

a. Larger[0.5]

b. Longer[0.5]

REFERENCE
Grand Gulf Rx Physics pg. 27

ANSWER 1.06 (2.00)

Following a turbine trip at 29% power there will be no scram. The turbine bypass valves will handle all the generated steam[0.5]. However, due to a loss of extraction steam there will be a gradual but marked decrease in feedwater temperature.[0.5] Due to the moderator temperature coefficient, reactor power will begin to increase[0.5]. As reactor power increases pressure will increase as during normal operation.[0.5]
(DAF) High power/pressure scram is possible if power exceeds 40%
(capacity of BPV's + aux. steam loads)

REFERENCE
Grand Gulf RPS, Rx Theory L.P.'s and Procedure 03-1-01-1

ANSWER 1.07 (2.50)

a. .64% [0.5]
b. Decreases[0.5] due to the production of Plutonium[0.5] which has a lower delayed neutron fraction than U-235.
c. Delayed neutrons increase the average neutron generation time[0.75] (by a factor of more than 5000) Increasing the control time of the reactor.[0.25]

REFERENCE
Grand Gulf Rx Physics pg. 31-34

ANSWER 1.08 (1.50)

o Convert pressures in 'psig' to 'psia':

985 psig + 14.7 psi = 1000 psia, and
385 psig + 14.7 psi = 400 psia. [0.5]

o Obtain corresponding temperatures from the Steam Tables:

1000 psia = 544.6 F, and 400 psia = 444.6 F. [0.5]

o Determine the temperature change for the hour:
544.6 F - 444.6 F = 100 F/hr. [0.5]

REFERENCE
Grand Gulf Steam Tables L.P.

ANSWER 1.09 (2.50)

Will accept core orificing OR orificed fuel support pieces.[0.5] As power increases the amount of boiling (two-phase flow) increases. [0.5] The amount of power generated in a peripheral bundle is < (approximately half) that of a center bundle; therefore boiling is greatest in the core center.[0.5] Two-phase flow restricts cooling water flow due to the boiling action.[0.5] This would cause the higher powered bundles to receive less cooling water, as their higher resistance to flow would divert flow to lower power fuel bundles [0.5] starving the higher power bundles.

REFERENCE
Grand Gulf Thermal Limits L.P. pg. 20-22

ANSWER 1.10 (1.50)

The detachment of steam bubbles from the heated surface agitates and breaks up the relatively stagnant fluid boundary layer,[1.0] thus promoting better fluid mixing,[0.5] improving core HT characteristics.

REFERENCE
Grand Gulf Core Cooling Mech., L.P. pg. 28

ANSWER 1.11 (2.00)

a. The subcooling of condensate below the saturation temperature.[0.5]

b. Without Condensate Depression, the condensate pumps would cavitate[1.0] (due to the water at the 'eye' of the pump being at saturation temperature.)

c. Cycle efficiency would be increased[0.5] by a decrease in condensate depression.

REFERENCE
Grand Gulf Thermo L.P.

ANSWER 2.01 (3.00)

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

b. Control rod drifting in(0.5) or CRD High temperature(0.5) (1.0)

c. See attached Fig. 2.01 (1.0)

REFERENCE
GGNS Control Rod Drive Hydraulic System Description

ANSWER 2.02 (3.50)

a. Two possible initiation paths, EITHER of the following initial combinations

LOW-LOW suppression pool level[0.3] AND a LOCA has occured[0.3]
OR
A LOCA has occurred[0.3](Low Rx water level/High DW pressure)
AND a 30 min. time delay.[0.3]

and the following:

SPMU mode selector switch ~~in~~ AUTO[0.3]

b. High level in the suppression pool could cause overpressurization of the drywell[0.5] during the initial phase of a LOCA[0.5]. An inadvertant dump while irradiated fuel is elevated for transit could present a radiation hazard to personnel on the refuel floor[1.0].

REFERENCE
GGNS SPMU System Description, Drawings Nos E-1220-01 & 02, Rev 12 & 13
OMn 7/6/84

ANSWER 2.03 (2.50)

A. The Turbine Stress Evaluator(0.5) and the reference value set by the SP DEMAND buttons(0.5). The lower of the two values is controlling(0.5). (1.5)

B. Load rejection(0.5)
Turning off the Load Reference Controller(0.5) (1.0)

REFERENCE
GGNS Pressure/Turbine Control System Description

ANSWER 2.04 (2.50)

A. HPCS D/G Jacket water cooler(0.5)
HPCS room cooler(0.5) (1.0)

B. Low reactor water level
High drywell pressure
HPCS D/G running
HPCS pump running
(0.25 each) (1.0)

C. The auto start signal must be reset (0.5)

REFERENCE
GGNS SSW System Description

ANSWER 2.05 (1.50)

SSW, Radwaste Effluent, CCW (0.5 each) (1.5)

REFERENCE
GGNS Process Radiation Monitoring System Description

ANSWER 2.06 (3.50)

a. 1. True[0.5]
2. False[0.5]

b. 1. LPCS pump start
2. LPCS room cooler start
3. F012 Test return valve close
4. F005 Injection valve open
5. F011 Minimum flow valve open

(4 required @ 0.25 each) (1.0)

c. See attached figure 2.06 (1.5)

REFERENCE
GGNS Sys Description Book 3 Sys E-21 pg 2,4,5,6
and SOI 04-1-01-E21-1 (LPCS)

ANSWER 2.07 (2.25)

a. 1. No [0.25] The interlock between F068 and F045-A will prevent F045-A from opening unless F068 is full open.[0.5]

2. Yes [0.25] The flow element would not detect any flow and therfore the pump would go to maximum speed.[0.5]

b. ~~1. Yes [0.25] The electrical trips can be reset from the control room.[0.5]~~

1. ~~2.~~ No [0.25] The overspeed is a mechanical trip and it must be reset locally.[0.5]

deleted from examination - See Report JHm 7/6/84

REFERENCE
GGNS System Description Book 7 Sys E-51 pg 5,7,8,11 and figure 1
Drawings Nos. E-C1185-06 (Rev A), E-1185-06 (Rev 6), -34 (Rev 5), -35 (Rev 3A), & -42 (Rev 5C)

ANSWER 2.08 (3.00)

a. 1. Electric FP - Fire Water supply header pressure of 125 psig.[0.66]

2. "A" Diesel FP - Fire Water @ 125 psig for 15 seconds.[0.67]

3. "B" Diesel FP - Fire Water @ 125 psig for 30 seconds.[0.67]

b. (Dry piping requires air pressure to keep the deluge valve closed.) For system actuation the fuzed nozzle melts, OR heat rate of rise detector activated; allowing instrument air to bleed off and the deluge valve opens.[1.0]

REFERENCE
GGNS Fire Prot. L.P.

ANSWER 2.09 (3.00)

a. See attached figures (2.0)

b. See attached figures OR valves F001 & F251 (1.0

REFERENCE
GGNS RWCU Sys Description & 05-1-02-III-5 Rev. 15

ANSWER 3.01 (3.00)

a. Causes reactor level to DECREASE [0.25] due to the Level Control System having a STEAM FLOW/FEED FLOW ERROR, STEAM FLOW < FEED FLOW [0.375] resulting in a SIGNAL to DECREASE the SPEED OF THE REACTOR FEED PUMPS. [0.375] (1.0)

b. Causes reactor level to INCREASE [0.25] due to the Level Control System having a LEVEL ERROR, with NO compensating FLOW ERROR [0.375] resulting in a SIGNAL to INCREASE the SPEED OF THE REACTOR FEED PUMPS [0.375] (1.0)

c. Reactor level should REMAIN CONSTANT [0.25] because the 'B' FEED PUMP Turbine Control Unit [0.375] will lock the pumps at the speed at the time of failure.[0.375] (1.0)

REFERENCE
GGNS Feedwater Flow Control System Description

ANSWER 3.02 (4.00)

a. High pressure in the SRV discharge piping (1.0)

b. 1. Safety(0.5)
2. No(0.5) (1.0)

c. High drywell pressure
Low reactor water level
time delay
(.33 each) (1.0)

d. All power is from 125 VDC (1.0)

REFERENCE
GGNS ADS System Description

ANSWER 3.03 (2.50)

a. rod block
b. half-scram
c. rod block
d. full scram
e. no trip
(0.5 each) (2.5)

ANSWERS -- GRAND GULF 1   -84/06/05-DODD, C.

REFERENCE
GGNS Reactor Protection System, APRM System Description

ANSWER 3.04 (2.50)

a. Containment spray is being initiated.[0.5] ~~Ten~~ 10.85 minutes after a LPCI initiation signal is received;[0.5] with ~~5~~ 7.84 psig in the containment[0.5] and ~~2~~ 1.39 psig in the drywell,[0.5] containment spray (A loop) initiates shutting F042A.

JPM
7/6/84

b. Yes.[0.5]

REFERENCE
GGNS RHR L.P. & Drawing E-1181-43 Rev. 6 and -67 Rev. 7A
Procedure 05-S-01-EP-3 TCN #1

JPM
7/6/84

ANSWER 3.05 (3.50)

a. 1. See attached figure.[0.6]
   2. About 15 inches[0.2] above the core midplane.[0.2]

b. 2.5 on range 7[0.5] NO auto actions will result.[0.5]

c. #1 - .99[0.5] is the more conservative

d. 1. APRM INOP[0.33]
   2. Rod Block[0.33]
   3. 1/2 Scram[0.34]

REFERENCE
GGNS NI Sys description (C-51), GGNS Theory Rev., NEDO(24810)

ANSWER 3.06 (4.50)

a. A pressure of 2 psig in the drywell results in transfer of both loop flow controllers to manual[0.5] and locks the flow control valves.[0.5]

b. Vessel Thermal Shock Interlocks:
   1. Loop/loop suction differential of <50 deg. F [0.5]
   2. Steam dome/loop suction differential of <100 deg. F [0.5]
   3. Steam dome/bottom head drain differential of <100 deg. F [0.5]

c. The EOC RPT trip is actuated if >30% power [0.33] and a TCV fast closure trip [0.33] or TSV trip is activated.[0.34]

   This is to insure sufficient negative reactivity is added in conjunction with the control rods to ensure thermal hydraulic limits (MCPR) are met.[1.0]

REFERENCE
GGNS Recirc Flow Control and Recirc System Descriptions

ANSWER 3.07 (3.00)

a. 1. Indicates that there is a loss of rod position data or erroneous data.[0.5]

   2. Ind. that all scram valves are not in the same position.[0.5]

   3. Ind. that the RGDS finds disagreement between the signals received.[0.5]

   4. Ind. that the withdrawn rod must be fully inserted before any other control rod can be selected.[0.5]

b. 2 notches[0.5], yes[0.5]

REFERENCE
GGNS Question C11-R-8 & RC&IS sys description

ANSWER 3.08 (2.00)

When activated - The source is positioned to irradiate the detector causing an upscale meter deflection and a HI Rad alarm.[1.0]

When deactivated - There is sufficient leakage to cause a background level reading, so that a channel failure would be indicated by a downscale alarm.[1.0]

REFERENCE
GGNS ARM sys. description

ANSWER 4.01 (3.50)

a. Recommended that the reactor NOT be taken critical in continuous withdrawl;[0.5] and that continuous withdrawal NOT be used between rod positions 08 and 24 [0.5] from the point of criticality [0.5] until above the Low Power Set Point.[0.5]

b. Mode switch to run:
   1. Main steam line pressure >850 psig[0.375]
   2. Main steam line low pressure alarm cleared[0.375]
   3. All operable APRM's indicate >5% power[0.375]
   4. All APRM downscale alarms cleared[0.375]

REFERENCE
IOI 03-1-01-1 Cold S/D to Gen Carrying Min Load pg. 2, & 36-37

ANSWER 4.02 (3.50)

a. Symptoms/Indications:

   1. Possible CW pump(s) tripped.
   2. Failed steam seal regulator / improper steam seal pressure.
   3. Increasing condensate temperature out of hotwell.
   4. Increasing turbine exhaust temperature.
   5. Condenser vacuum breaker not full shut indication.
   6. Increased Off Gas System Flow, due to condenser inleakage.
      ((4 required at 0.5 each))

b. Auto actions
   1. Turbine Trip (21"Hg)
   2. RFP Trip (16"Hg)
   3. BPV Closure (12"Hg)
   4. MSIV Closure (9"Hg)

   (3 actions req @ 0.3 each and 0.2 each for proper order)

REFERENCE
ONEP 05-1-02-V-8 LOSS OF CONDENSER VACUUM Rev.11 pg. 1

ANSWER 4.03 (1.50)

Plant operation may continue until the second accumulator(0.5) associated with a withdrawn control rod is declared inoperable(0.5). The plant is shutdown by placing the Mode Switch to Shutdown(0.5). (1.5)

REFERENCE
ONEP 05-1-02-IV-1 Control Rod/Drive Malfunctions

ANSWER 4.04 (3.00)

a. 1. Place the Reactor Mode Switch in Shutdown.[0.66]
   2. Verify all control rods are fully inserted.[0.67]
   3. Manually initiate RCIC sys. time permitting.[.67]

b. RCIC, SSW & CRD (0.333 each)

REFERENCE
GGNS 05-1-02-II-1 S/D from the Remote S/D Panel

ANSWER 4.05 (2.50)

a. 1. RCIC
   2. RHR (Steam Condensing Mode)
   3. SRV's
   4. RWCU (Recirculation Mode)
   5. RWCU (Blowdown Mode) (4 of 5 required @ 0.5 each)
   Note: Where operating mode is included, mode is 1/2 credit

b. Condensate/feedwater system[0.5]

REFERENCE
05-S-01-EP-2, pg. 2,4-6.

ANSWER 4.06 (2.50)

1. DW pressure >1.23 psig
2. DW temperature >135 deg. F
3. Suppression Pool temp. >95 deg. F
4. Suppression Pool level >18.83'
5. Suppression Pool level <18.45'
6. Containment Temp. >90 deg. F

(5 of 6 required at 0.5 each)

REFERENCE
GGNS 05-S-01-EP-3 and TCN-1

ANSWER 4.07 (4.00)

a. 1. RWC [0.5]
2. Fuel Pool Heat Exchanger[0.5]

b. Scram the reactor.[0.67] Manually trip recirc pumps [0.33] within 1 minute [.167], or any noted increase in recirc pump or motor winding temperature [.167]. Secure any running CCW pumps[0.33] if surge tank level is low and cannot be restored.[0.33]

c. When Recirc Pump temperatures cannot be maintained below the alarm setpoint.[1.0]

REFERENCE
05-I-02-V-1 Loss of Component Cooling Water, pg. 1-2

ANSWER 4.08 (5.00)

a. CRD drive mechanisum damage could result.[1.0]

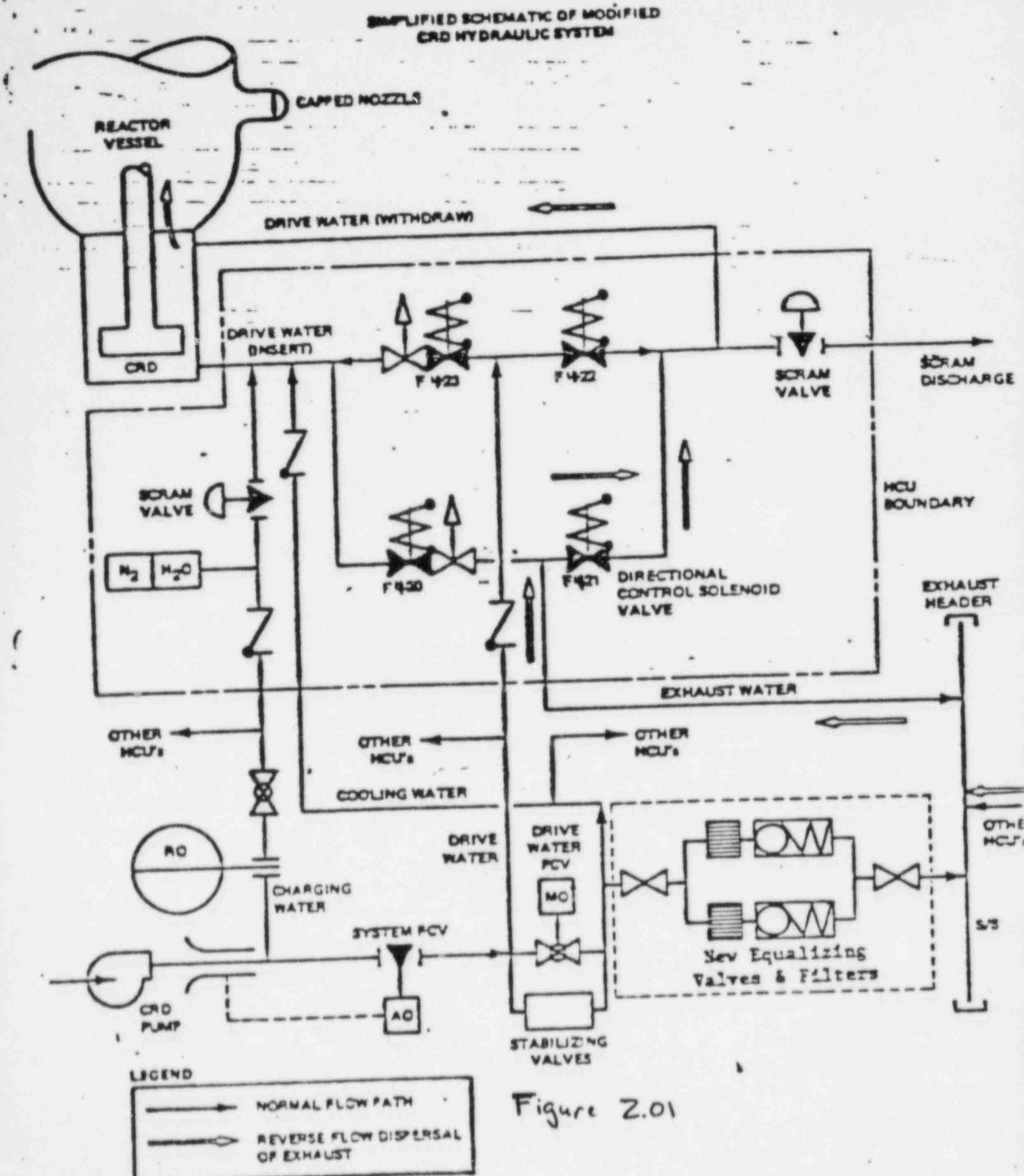
b. To ensure adequate NPSH at the RWCU pumps.[1.0]

c. Impingement damage to installed core instruments could occur.[1.0]

d. Diesel Generator auto S/D features are inhibited.[1.0]

e. Flow below this setpoint would opn the min. flow valve sending RPV water to the suppression pool.[1.0]

REFERENCE
SEE GGNS Procedures listed in question

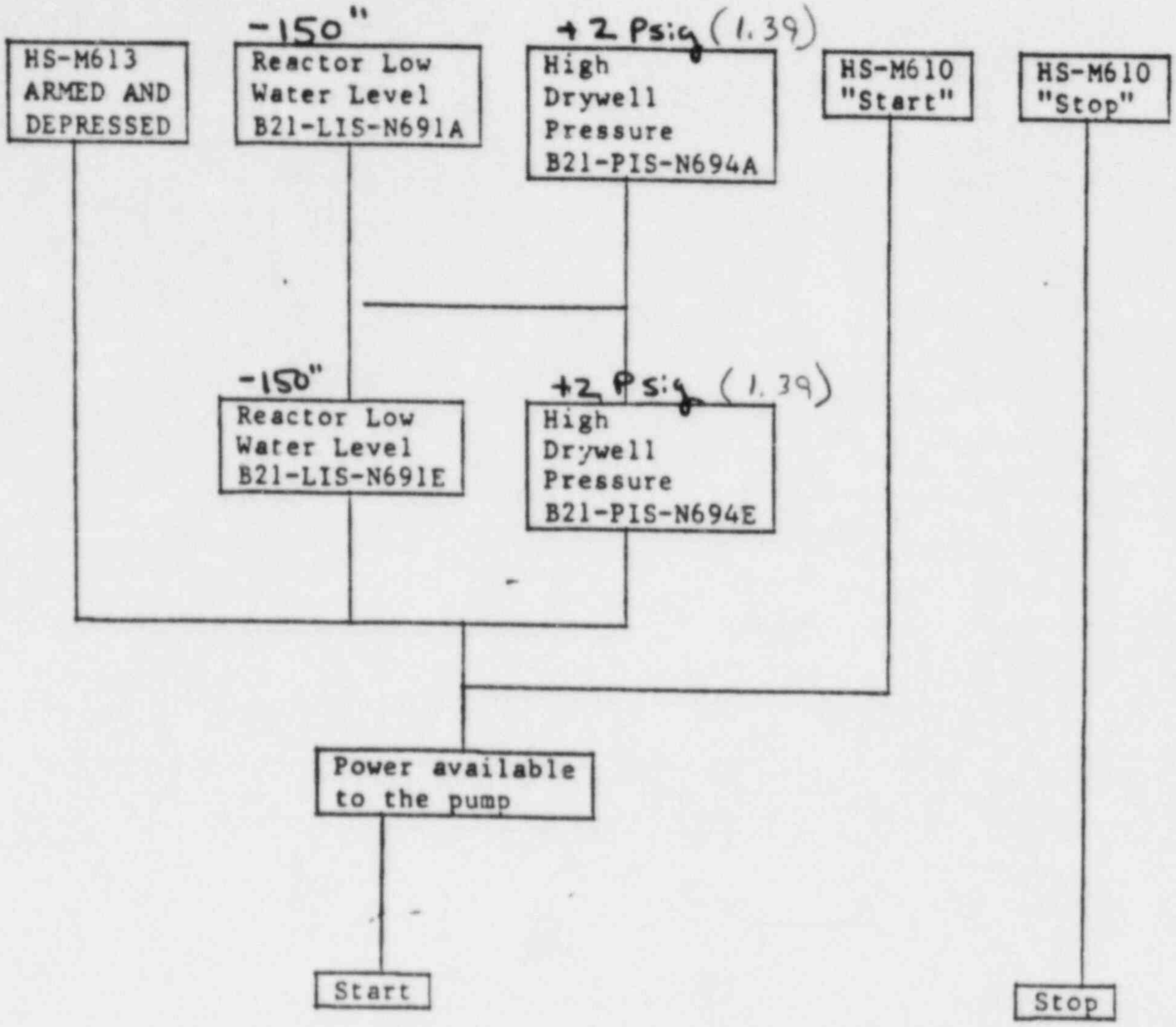


SIMPLIFIED SCHEMATIC OF MODIFIED
CRD HYDRAULIC SYSTEM
CAPPED NOZZLS
REACTOR
VESSEL
DRIVE WATER (WITHDRAW)
DRIVE WATER
(INSERT)
CRD
F 123
F 122
SCRAM
VALVE
SCRAM
DISCHARGE
SCRAM
VALVE
HCU
BOUNDARY
N_2
H_2O
F 120
F 121
DIRECTIONAL
CONTROL SOLENOID
VALVE
EXHAUST
HEADER
EXHAUST WATER
OTHER
HCU's
OTHER
HCU's
OTHER
HCU's
COOLING WATER
DRIVE
WATER
DRIVE
WATER
PCV
MO
RO
CHARGING
WATER
SYSTEM PCV
New Equalizing
Valves & Filters
OTHE
HCU's
S/S
CRD
PUMP
AO
STABILIZING
VALVES
LEGEND
NORMAL FLOW PATH
REVERSE FLOW DISPERSAL
OF EXHAUST

Figure 2.01

VI. C. Trips, Permissives and Interlocks

3. LPCS Pump



+.25 for each signal (2 for .5)

+.25 for each Setpoint (2 for .5)

Figure 2.06

+.5 for correct logic placement

Ref: GGNS LPCS E21 SD

Gave credit for new high DW press setpoint

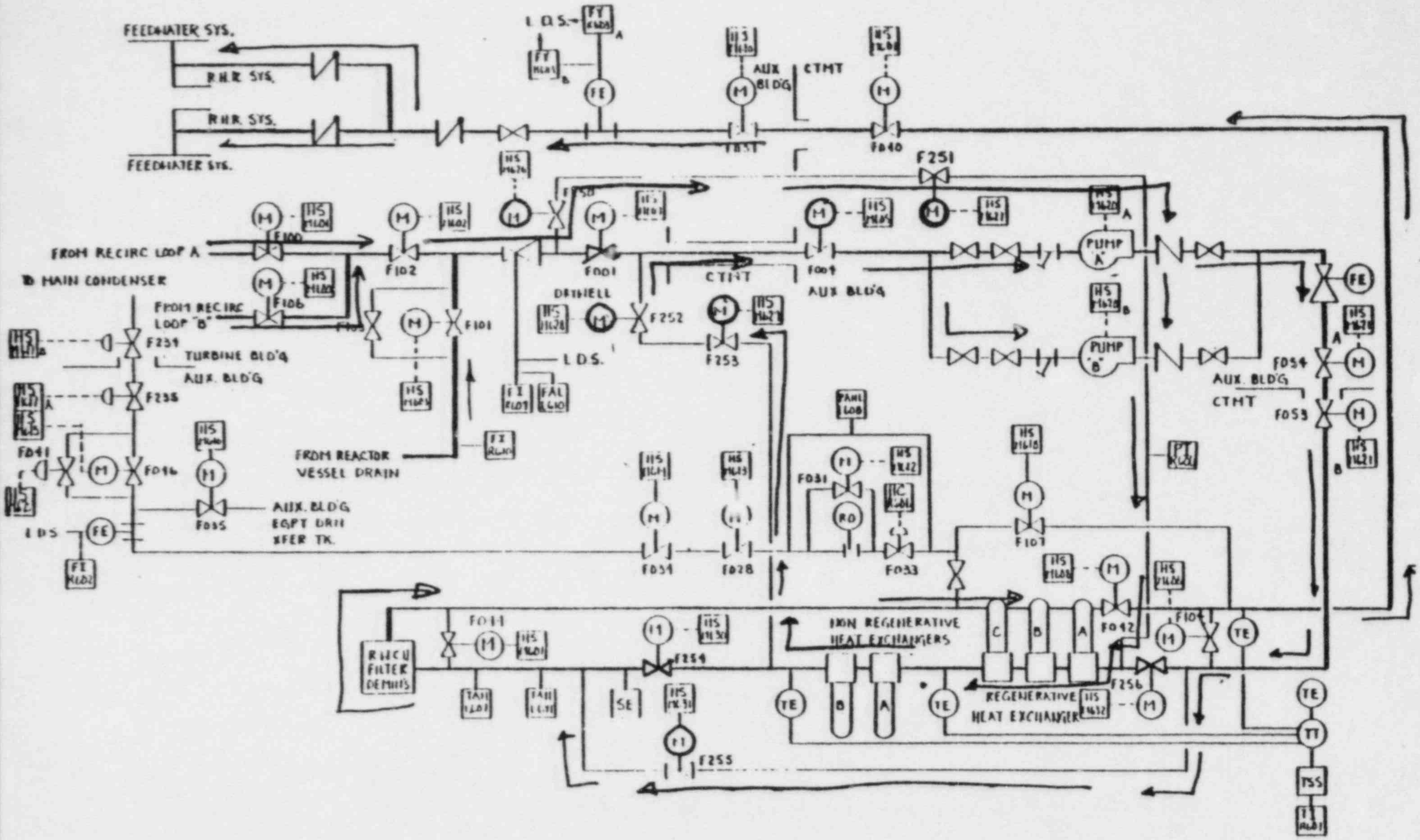


FIGURE 1. REACTOR WATER CLEANUP SYSTEM
Post-Pump Mode

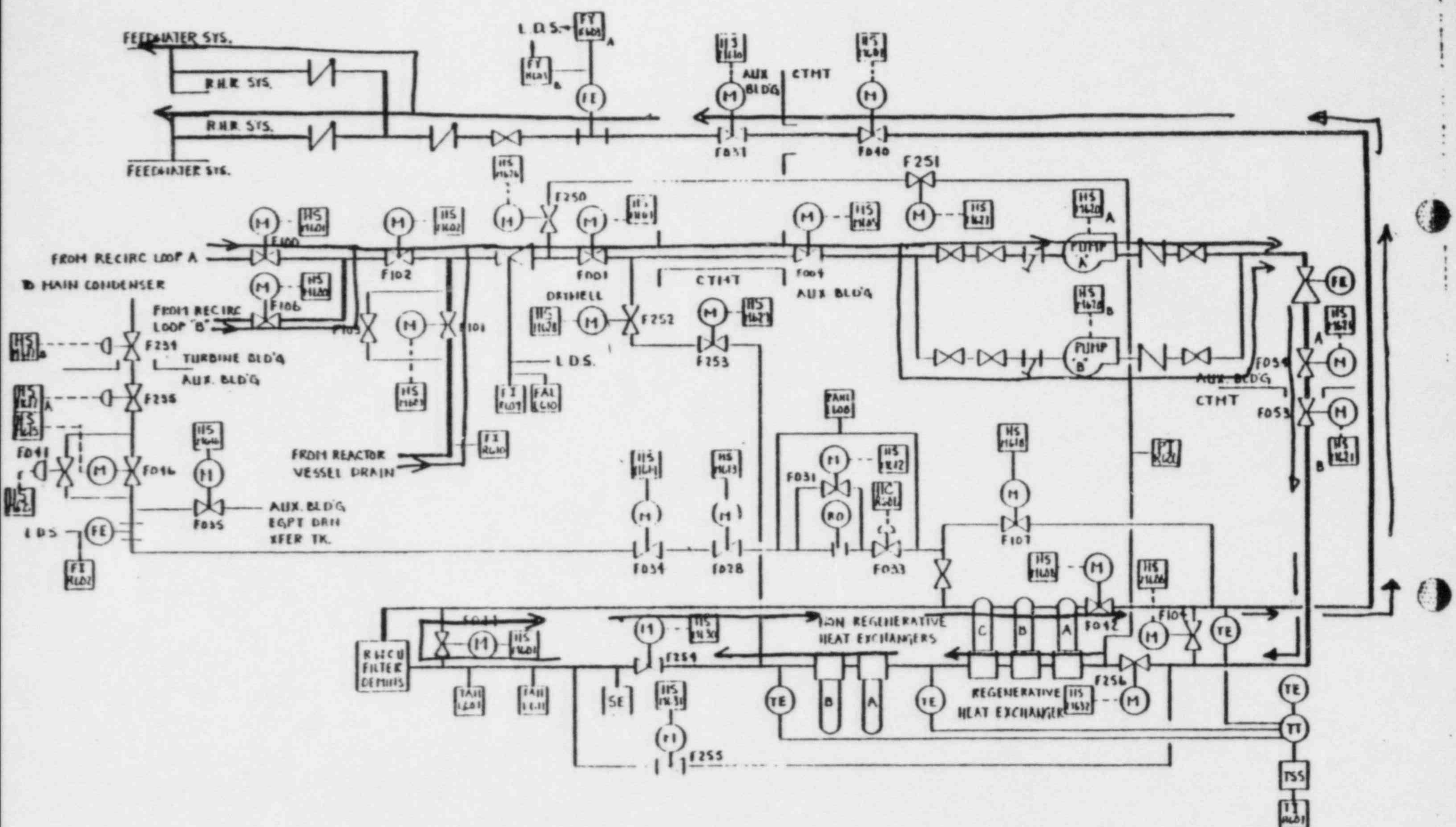
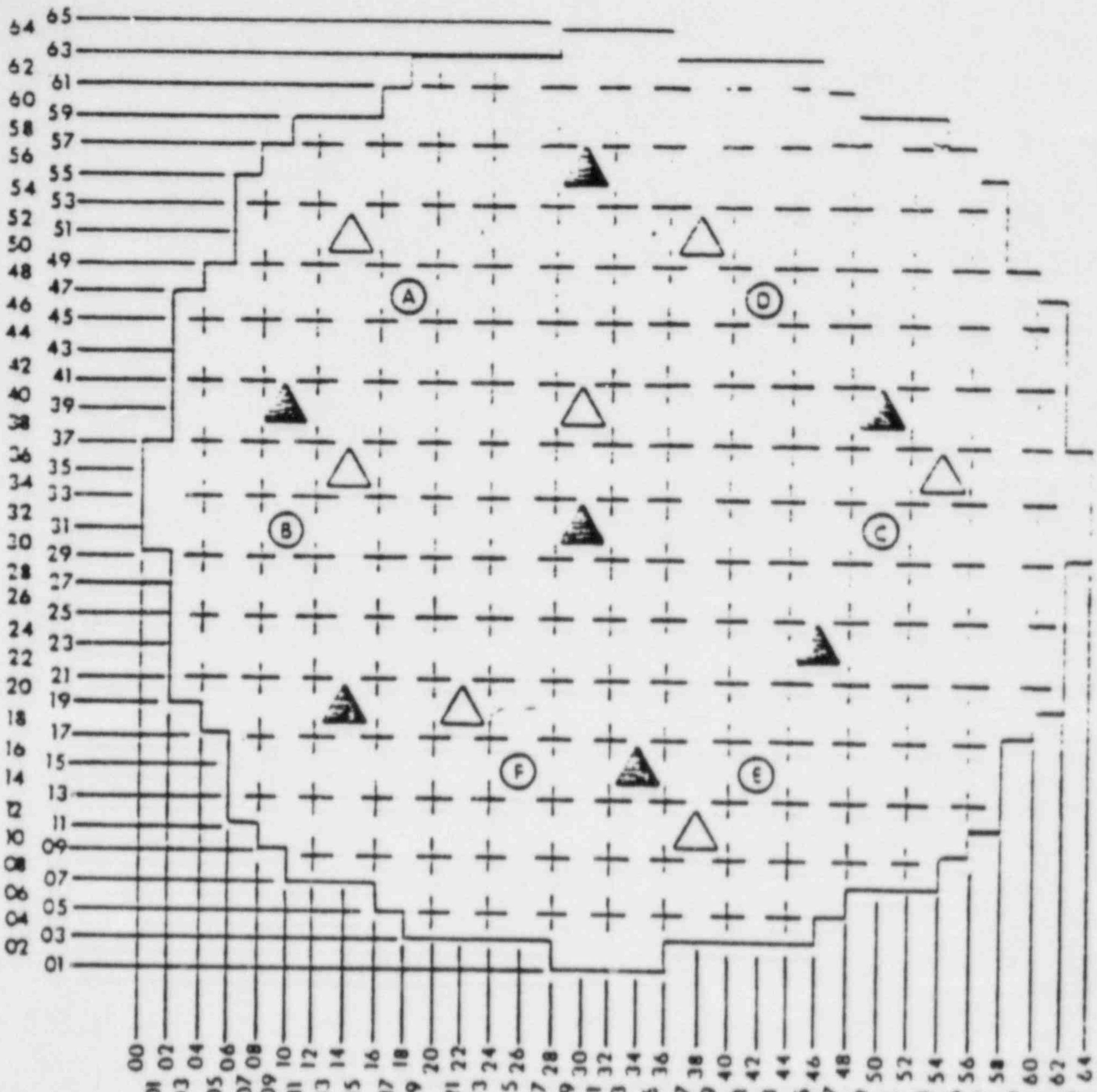


FIGURE 1. REACTOR WATER CLEANUP SYSTEM

Pre - Pump Mode



○ SRM DETECTOR LOCATIONS

▲ NEUTRON SOURCE LOCATIONS

△ SPARE NEUTRON SOURCE LOATIONS (FOR POSSIBLE FUTURE USE)

Figure 3.05(a) Source And SRM Detector Assembly Core Positions