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

ENCLOSURE 1

EXAMINATION REPORT

Facility Licensee: Tennessee Valley Authority
500-A Chestnut Street, Tower II
Chattanooga, TN 37401

Facility Docket Nos.: 50-259, 50-260, 50-296

Facility License Nos.: DPR-33, DPR-52, DPR-58

Examinations administered at Browns Ferry Nuclear Plant, Decatur, Alabama

Chief Examiner: ______________________________ ____________
Siegfried Guenther Date Signed

Approved by: ______________________________ ____________
Bruce A. Wilson, Section Chief Date Signed

Summary

Examinations on March 23-30, 1984

Written, oral, and simulator replacement examinations were administered to ten SROs and nine ROs; two SROs and seven ROs passed these examinations.

Written requalification examinations were administered to nine SROs and six ROs; seven SROs and four ROs passed these examinations. Oral requalification examinations were administered to four SROs and two ROs; all individuals passed these examinations.

REPORT DETAILS

1. Persons Examined

SRO Candidates

G. Bradford (W/O)
D. W. Charles
R. L. Dye
R. R. Eades
C. B. Fisher
J. C. Hall
R. F. Hannah (W)
R. D. Higdon
J. E. Lamb
J. W. Martin (W)

SRO Requalification

R. B. Abercrombie (W)
B. E. Baggett (W)
A. L. Burnette (W)
J. E. Duke (O)
D. O. Elkins (W)
E. S. Howard (W)
R. G. Jones (W)
T. G. Jones (W)
T. W. Jordan (O)
J. A. Marbutt (O)
R. H. McDowell (W)
P. L. Walker (W)
J. H. White (O)

RO Candidates

D. L. Branam
G. H. Christopher
J. E. Dollar
T. D. Elms
J. A. Frost
R. E. Harris, Jr.
P. E. Hillis
J. Y. Joe, Jr.
E. R. Scillian

RO Requalification

J. W. Allison (W)
M. W. Ash (W)
M. E. Gann (W)
R. L. Johnson (O)
R. V. Johnston (W)
R. O. Miller (W)
D. M. Olive (W)
E. S. Westfield (O)

NOTE: W/O indicates that only a written and/or oral examination was administered during this visit.

Other Facility Employees Contacted

G. T. Jones, Plant Superintendent, BFNP (2/conference call)
J. Swindell, Assistant Plant Superintendent, Operations (2/conference call)
R. Hunkapillar, Operations Section Supervisor, BFNP (2/conference call)
R. J. Johnson, Chief, Nuclear Training Branch (2)
L. Sain, Assistant Chief, Nuclear Training Branch (2)
C. H. Noe, Supervisor, Operator Training, NTB (1,2)
N. Catron, Simulator Unit Supervisor (1,2)
D. Connors, Section Supervisor (2)
T. Dexter, BWR Engineering Unit Supervisor (1,2)
R. G. Jones, Training Shift Engineer (2)
A. Champion, Training Shift Engineer (1,2)

N. J. Lewis, Simulator Instructor (2)
J. D. Johnson, Simulator Instructor (1,2)
W. D. Dawson, Simulator Instructor (1)
J. D. Glover, Shift Engineer (1)
M. W. Miller, Shift Engineer (1)
J. E. Duke, Assistant Shift Engineer (1)
C. A. Casto, Unit Operator (1)
J. R. Fish, Unit Operator (1)

2. Examiners

S. Guenther, NRC, Chief Examiner (1,2)
J. Munro, NRC (1,2)
C. E. Dodd, EG&G (1)
D. E. Hill, EG&G
J. C. Kvamme, EG&G
R. L. Persons, EG&G

NOTE: (1) Present at exam review.
(2) Present at exit meeting.

3. Examination Review Meeting

At the conclusion of the written examinations, license examiners S. Guenther, J. Munro, C. Dodd, and B. A. Wilson, Chief, Operator Licensing Section, NRC, Region II, met with facility representatives (identified in (1) above) to review the written examinations and answer keys (RO/SRO for replacement/requalification).

The following comments were made by the facility reviewers in reference to the RO and SRO replacement examinations. Where applicable, the comments are also referenced to appropriate requalification examination questions (numbers in parentheses) although they were not specifically addressed by the individuals reviewing those exams.

a. Question 1.01a - Candidates may use a scaling factor to determine "% power" and then calculate the answer.

   Resolution - It is acceptable for the candidate to convert the indicated power to % power by use of a scaling factor. However, since the question specifically asked for "INDICATED" power, the candidate must convert his new % power level back to an indicated power level to receive full credit. No change to the answer key is required.

b. Question 1.04.a (1.3.a) - Critical Quality, boiling length, actual power and CPRLIM should be included in the answer key.

   Resolution - The question specifically asked for three "measurable core parameters" necessary for the process computer to calculate MCPR. Since critical quality, boiling length and CPRLIM are not directly measurab` parameters, they will not be acceptable answers. Actual power is already included in the answer key under "Power, local power, flux, or local flux"; no change to the answer key is required.

c. Question 1.07a - "Power in the core after a scram" should be added to the key as an acceptable answer.

Resolution - This phrase is considered an acceptable equivalent/ substitute for the first segment of the key answer to question 1.07a. The answer key has been changed accordingly.

d. Question 2.01a (2.1.a) - Loss of the open indication light might not lead the candidate to understand that the valve is not full open.

Resolution - The intent of the question is to test the candidates' knowledge of valve/system interlocks related to the loss of open indication for the RWCU inlet isolation valve (FCV-69-1) and not to determine if the candidate has an instrument technician's knowledge of how that valve's position indication is derived. Any candidate who misinterprets the intent of the question will receive appropriate credit based upon his assumptions and answer. No change to the answer key is required. (A post-exam review failed to reveal any misinterpretations in the intent of the question.)

e. Question 2.05 (2.4) - Cal. 154 psig; setpoint 142 psig. (i.e. 142 psig should be an acceptable first stage turbine pressure.)

Resolution - LER 84-014-01, a copy of which is attached to the examination answer key, and a caution plaque mounted on the EHC panel in the control room confirm that the actual setpoint for first stage turbine pressure is 142 psig. The answer key has been changed to accept both 154 psig (per Tech Specs) and 142 psig (actual).

f. Question 2.06d (6.7c) - The HPCI pump may not overspeed, but will go to its maximum pump capacity and speed. The turbine will only overspeed if uncoupled.

Resolution - If this malfunction existed when a turbine start signal is received, then an overspeed would probably occur. If the candidate assumed an overspeed condition, then he/she would be held liable for the full answer as stated in the key. A description of the same malfunction in the Singer Simulator Malfunction Index states that if the failure occurred while the pump was operating that it would increase speed and capacity to maximum but probably would not overspeed. Since the question did not specify when the failure occurred, the answer key has been changed to include the speed/capacity increase as an acceptable answer.

g. Question 2.07/6.03a&b - The candidates may not provide both parts of the answer per the key because the question does not specify a two-part answer.

Resolution - Learning objectives #5 and #6 of BF LP-43, "Automatic Depressurization System," require that the candidate know the bases for the selection of the -114.5" water level and 120-second timer in the ADS logic. The question requests the candidate to "State the Bases for each condition" and both parts of each answer are required for a complete answer. No change to the answer key is warranted.

h. Question 3.01c (3.1c) - The question asks how control rod "position" is determined while the answer key refers to movement and direction, not position.

Resolution - Accurate determination of control rod position requires both direction information from the Rod Movement Control Switch (RMCS) and movement information from the RMCS timer "settle" function as specified in RSCS LP-25, page 8. No change to the answer key is warranted.

i. Question 3.02b (2.3.b./6.2.b) - The answer assumes that the operator placed the breaker in emergency. Use of the term "backup control panel" should refer to "MOV Board."

Resolution - LP-44, "RHR System," page 13 indicates that these valves (24 & 35) have backup control at the remote shutdown panel, a commonly used synonym for Backup Control Panel (BCP). Physical verification, however, revealed that their backup controls are not at the BCP but at the MOV board. The LP should be corrected to reflect the actual plant condition. The answer key has been changed to allow full credit for a statement that no control exists for those valves at the BCP.

j. Question 3.04/6.08.a (3.3.a) - With Max Combined Flow at "0", a signal still goes to the LVG that would allow 50% steam flow or 25% valve opening.

Resolution - Browns Ferry Transient HXY-16 and documentation provided by the facility (BF SMG 147-1.2, p.34 and P24B-AL-4985, p. 20A) substantiate the requested change to the answer key; it has been changed accordingly.

k. Question 3.07a - The "breakers open" and "no overcurrent lockout" are two separate requirements.

Resolution - Browns Ferry Diesel Generator LP-38, P. 11 identifies three conditions required to initiate automatic closure of the D/G output breaker. "All other supply breakers open with no overcurrent lockout" is identified as a single condition. No change to the answer key is warranted.

l. Question 4.01/7.10 - The first action step would be to verify which valve is open by checking the acoustic monitor and TR-1-1.

Resolution - Since the question did not specify a valve number, verification of which relief valve is open could be interpreted by the candidate as a valid action step per the procedure. The answer key has been changed to include that as one of five possible answers, with four of five required for full credit.

m. Question 4.02(4)/7.02d - Failed jet pump flow would decrease if the failure were due to a jet pump riser failure. The question does not specify failure mode.

Resolution - Since the question did not specify a failure mode for the jet pump, the candidate would be free to assume a riser failure. If he/she specifies riser failure as the assumed failure mode in his/her answer, then decrease would be an acceptable answer. The answer key has been changed accordingly.

n. Question 4.03/7.02b (4.2/7.2b) - The critical speeds for the Unit 2 main turbine (determined by telecon with the Operations Section Supervisor) are 900 rpm and 1100-1400 rpm. Written documentation will be provided later.

Resolution - Subsequent research by the facility, documented in a letter to the NRC, revealed a different critical speed(s) for each turbine bearing. The speed ranges of 900-1100 rpm and 1365-1415 rpm were determined as representative for the Unit 2 turbine and were incorporated in the answer key.

o. Question 4.03d/7.02c (4.2/7.2c) - Some candidates may give answer of "70P" rather than "manual voltage regulator."

Resolution - "70P" and "90P" are acceptable equivalents for the manual and automatic voltage regulators respectively.

p. Question 4.09/7.07b (4.7/7.7b) - Some candidates may give 93°F due to modifications on the torus temperature monitor not being completed on Units 2 and 3. Reference BF OI-64, January 12, 1984, page 21.

Resolution - BF OI-64 (January 12, 1984), p. 21 and BF OI-74 (Unit 2, July 29, 1983), p. 8 specify 93°F for initiating torus cooling on Units 2 and 3. Although the question was in reference to GOI-100-11, which specifies 95°F for initiation of torus cooling, either answer will be accepted provided the candidate identifies the unit(s) to which he is referring.

q. Question 4.10/7.11a - The instruments are set at 100 cpm above background; any deflection from background would show contamination. It is more important to be able to recognize contamination than to memorize .05 or 300 cpm.

Resolution - When nuclear plant operators perform a routine self-frisk, they should be aware of what meter reading represents the Maximum Contamination Limit for the skin without having to rely upon an instrument alarm which has a potential for drift or failure. The question is within the scope of NUREG 1021, ES-202, paragraph B.4 in that it concerns radiological control procedures necessary to ensure personnel protection against the hazards of radiation. No change to the answer key is warranted.

r. Question 5.04d (5.2d) - The answer should be decrease per the same reference.

Resolution - An error was made in interpretation of the reference material. Decrease is the correct answer, the key has been changed accordingly.

s. Question 5.11a&b (5.8a&b) - Candidates may use boiling length changes in the answer.

Resolution - A discussion of changes in boiling length when answering the question may be acceptable depending upon the context in which it is used. The question will be graded based upon the candidate's understanding of the concepts involved. No change to the answer key is required.

t. Question 6.04 - "APRM not downscale" may be used for companion APRM's on scale.

Resolution - "Not Downscale" is an acceptable equivalent for "on scale". The answer key has been changed to accept either term for full credit.

u. Question 8.03 - The operators only need to be aware of the limits due to planning requirements to exceed these limits. HP would be the responsible party.

Resolution - The BFN Accelerated Requalification Program lesson plan on Radiological Control and Safety requires the operators to list from memory the extreme emergency exposure guidelines (learning objective #3). No change is warranted.

v. Question 8.08a (8.5.a) - The question may lead to a response of permission being received from the SRO or may get response of actions to reset PCIS.

Resolution - Standard Practice 12.17 does not address the granting of the SRO's permission before resetting the primary containment isolation. The question specifically asked what must be done <u>prior to</u> resetting, not what actions are taken to reset the isolation. No change to the answer key is warranted.

w. Question 8.10b - Reactivity insertion should be included as an acceptable answer per HLM #7, p. 16.

Resolution - Browns Ferry Hot License Manual, Recirc. System LP #7, p. 16, supports the requested addition. The answer key has been changed to include reactivity insertion.

The following comments were made by the facility reviewers in reference to the RO and SRO requalification examinations:

aa. Question 2.1a - The question does not elicit the response as noted on the answer key. The question as written does not represent need to know information.

Resolution - Refer to comment/resolution (d) above. Additionally, the question is within the scope of NUREG-1021, ES-202, paragraph B.2, in that it pertains to the design intent, construction, operation, and interrelationships of a system (RWCU) directly associated with normal nuclear power plant operation. The question is also consistent with the RWCU lesson plan (#13) learning objective #4, i.e., "List the pump automatic trips and their setpoints."

bb. Question 2.2/6.5 - The question does not represent need to know (long term memorization) information for an RO.

Resolution - The following requalification lesson plans and learning objectives address required operator knowledge regarding loss of electrical power to system valves, etc.: HPCI/3&4; ADS/3&4; RCIC/3&4; CS/4&5; and RHR/4&5. Five answers to the question had, therefore, been previously evaluated as need to know information by the facility training staff and would have been sufficient for a full credit answer.

cc. Question 2.3.b.2/6.2.b.2 - The question, due to plant terminology, could elicit an improper response. There are no control switches on panel 25-32.

Resolution - Refer to comment/resolution (i) above.

dd. Question 2.5b/6.4b - The answer per OI-66 would be "high alarm."

Resolution - In view of the fact that a high alarm/trip will already have been actuated when the high-high and triple-high levels are reached, "High Alarm on either post-treatment radiation monitor" will be satisfactory for full credit. The answer key has been changed accordingly.

4. Exit Meeting

At the conclusion of the examination visit, the examiners met with representatives of the plant and training center staffs to discuss the results of the examinations. Those individuals who clearly passed the oral and/or simulator examinations were identified. Several topics concerning generic exam performance and the operator licensing process were discussed during the exit briefing and are summarized as follows:

a. The examiners noted some improvement in the candidates' ability to apply theoretical concepts to plant operations.

b. The examiners noted that the print index in use in the control room is falling apart and should be repaired or replaced. (G. Jones, Plant Superintendent, BFNP, responded that the index had already been replaced.) Some candidates also continue to have difficulty in cross-referencing GE to BFNP prints and in locating appropriate instrumentation prints.

c. The examiners noted that several candidates failed to use procedures during their simulator examinations. Communication between the RO and SRO candidates was also inadequate to keep all examinees aware of changes in plant status.

d. The chief examiner pointed out difficulties encountered in the distribution of required reference material to all examiners. A discussion followed concerning the establishment of appropriate boundaries for NRC examination questioning.

e. Dr. R. J. Johnson expressed his philosophy on items (c) and (d) above and proposed possible meetings between TVA and the NRC to discuss the appropriate use of procedures by plant operators and the establishment of more specific boundaries on the field of questioning for NRC licensing examinations.

f. Dr. R. J. Johnson expressed a desire to have the simulator and oral examinations evaluated as soon as possible so the results could be provided to the facility. The chief examiner responded that it would not be possible to process the simulator and oral results until the written examinations were graded, but that every effort would be made to provide complete exam results within thirty days.

g. The cooperation given to the examiners and the effort to ensure a control room atmosphere conducive to the conduct of oral examinations was appreciated and noted.

5. Summary

Based upon the operator and senior operators' performance on the written and oral requalification examinations, as summarized in Enclosure 2, the Browns Ferry Nuclear Plant 1983 Accelerated Requalification Training Program is evaluated as satisfactory. The lessons learned as a result of the November 18, 1983 requalification examination and the actions taken to strengthen the operator training program have proven effective and should be applied henceforth to further improve operator knowledge and performance.

Discussions held with members of the BFNP training staff in the Region II Office on March 2, 1984 (Meeting Summary of March 15, 1984 refers), and NRC correspondence dated February 15, 1984 (Accelerated Requalification Training Program) expressed our concerns regarding the shortcomings in the Requalification Training Program learning objectives and the narrow scope of the material covered. It is highly recommended that the Tennessee Valley Authority and Browns Ferry Nuclear Plant training staffs continue to evaluate and improve those learning objectives and broaden the scope of material covered. The field of questioning in future requalification examinations administered by the NRC at Browns Ferry (and elsewhere) will continue to follow the guidance of ES-601, paragraph D.3, in that it will be based primarily on the learning objectives of the facility's requalification program when the NRC determines that these objectives encompass the scope of material required by 10 CFR 55, Appendix A.

It was noted that an inordinate number of SRO candidates failed this examination (8 of 10) and the June, 1983 examination (4 of 6). Additionally, one RO was noted to have failed this written examination with an extremely low grade of 55.9 overall. These statistics raise a concern regarding the validity of Tennessee Valley Authority's certification that Browns Ferry license applicants have "learned to operate the controls in a competent and safe manner" pursuant to Title 10, Code of Federal Regulations, Part 55. The NRC intends to examine the Browns Ferry Nuclear Plant license applicant certification process during the training program assessment scheduled for the week of July 16, 1984.

— MASTER —

ENCLOSURE 3

TABLE ES-203-1

Reviewed by:
CH Noe
N Catron
Dexter } thru each
Glover
Fish

U.S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR REQUALIFICATION EXAMINATION

Facility: Browns Ferry 1, 2, 3

Reactor Type: BWR

Date Administered: March 23, 1984

Examiner: S. Guenther

Candidate: ______________

INSTRUCTIONS TO CANDIDATE:

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

Category Value	% of Total	Candidate's Score	% of Category Value	Category
15.5	25.0			1. Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer and Fluid Flow
14.5	23.4			2. Plant Design Including Safety and Emergency Systems
16.5	26.6			3. Instruments and Controls
15.5	25.0			4. Procedures - Normal, Abnormal, Emergency, and Radiological Control
62.0				TOTALS

Final Grade ________ %

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

____
Candidate's Signature

1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW

1.1 Figure 1.1, "Transient Period" illustrates the reactor period and power response to a rod withdrawal performed while exactly critical in the source range.

a. Calculate the stable positive period shown on the center graph assuming a BOL core. Show all assumptions and calculations. (1.0)

b. Assuming no operator action and the following initial conditions:

Rx Power 2 E + 4 CPS
Rx Temperature 281 F
Rx Pressure 35.3 PSIG

What will be the final stable values of reactor power, pressure, and temperature? Show all assumptions and calculations. (1.5)

1.2 a. The heat load on a coolant system containing two identical, variable speed, cent. pumps has increased such that twice the flow is required to remove the heat. If one pump is initially running at half speed, would it be more efficient to double the speed of the running pump or to start the second pump at half speed in parallel with the running pump? (.5)

b. Figure 1.2 illustrates two methods of reducing a system's flow capacity; i.e., by throttling and by pump speed reduction. State which method is preferred and give one reason WHY. (1.0)

1.3 a. Since MCPR is not a directly measurable parameter, what are three (3) measurable core parameters needed by the process computer to calculate MCPR? (1.0)

b. With regard to MAPRAT,

1. What is the relationship between MAPRAT and MAPLHGR? (.75)

2. The process computer prints out a MAPRAT of 1.05. Is this acceptable? (.5)

3. What physical consequence could occur if the MAPRAT limit is exceeded? (.75)

1.4 For each of the following events, STATE WHICH coefficient of reactivity would act FIRST to change reactivity. (No explanation required).

a. Control rod drop at power (.5)
b. SRV opening at power (.5)
c. Loss of shutdown cooling when shutdown (.5)
d. One recirc pump trips while at 50% power (.5)
e. Loss of one feedwater heater (extraction steam isolated) (.5)

1.5 With the plant operating at 100% power, a TURBINE TRIP WITH BYPASS occurs. Answer the following using the transient information on attached figure 1.5 and assuming no operator action:

a. Why does feedwater flow initially decrease (AREA 1) and then increase sharply (AREA 2)? (1.0)

b. Explain what is happening to total steam flow in AREA 3? (.5)

c. Why is there no flux spike (AREA 4) caused by the turbine trip? (.5)

1.6 Answer the following with regard to figure 1.6, which illustrates xenon reactivity vs power level and time.

a. Mark the vertical axis of the graph in units of % delta k/k indicating the approximate values of 100% equilibrium AND peak xenon. (.5)

b. For a scram occurring from 50% equilibrium xenon, how long after the scram would the peak occur? (Select correct answer) (.5)

1. 7 hours
2. 12 hours
3. 10 hours

c. At approximately 20 hours after the scram the reactor is restarted and power is returned to 25% until a new equilibrium xenon level is established. On figure 1.6, draw the xenon transient response to this power change. (1.0)

1.7 HOW and WHY does control rod worth vary for the following changes: (2.0)

a. As moderator temperature increases.

b. As an adjacent rod is inserted.

c. As the rod's axial position inside the core is changed.

2. PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS

2.1 a. The plant is operating at high power when a technician incorrectly lifts a lead resulting in the loss of open indication of RWCU inlet isolation valve (FCV-69-1). What two (2) automatic actions, direct or indirect, will result? (1.0)

b. When using the RWCU system for "Hot Blowdown", the blowdown flow rate must be restricted to avoid exceeding temperature limitations in two locations. WHAT are these locations and WHY does hot blowdown cause you to approach their temperature limits? (1.5)

2.2 During reactor operation at 98% power a loss of 250-VDC Reactor MOV board A occurs. List five (5) different systems and/or MAJOR components that will be affected prior to transferring power, and describe one way in which each one is affected.

Example: If 250 - VDC Rx MOV C had been lost - RCIC - Loss of power to condensate pump. (2.5)

2.3 When operating RHR in Shutdown Cooling Mode:

a. What three (3) conditions (including setpoints) will result in an automatic closure of the 48 valve (S/D cooling suction inbd. isolation valve)? (1.5)

b. An operator attempts to open the 24 & 35 valves (RHR pump torus suction) from the following locations:

1. Control Room

2. Backup Control Panel

IN BOTH CASES EXPLAIN why the valves WILL or WILL NOT open. (1.5)

2.4 Explain the differences between an ATWS recirc pump trip and a Recirculation Pump Trip (RPT). Include in your explanation initiation signal(s), all bypasses, and components actuated. Answer on the attached handout table. (2.5)

2.5 a. Place the following components in proper flow path order. (1.25)

- off gas condenser
- gas reheater
- holdup pipe
- catalytic recombiner
- preheater
- charcoal adsorbers

b. The charcoal adsorber bypass valve (113B) is open during periods of low activity operation. Under what conditions will it automatically close and the adsorber inlet valve (113A) open? (.75)

2.6 When a scram signal occurs at power, describe IN DETAIL how the Control Rod Drive and its associated Hydraulic Control Unit function to insert the control rod. Include which components open, close, energize, and deenergize, and the motive forces for the entire rod travel as a MINIMUM in your answer. (2.0)

3. INSTRUMENTS AND CONTROLS

3.1 a. The RSCS enforces Group Notch Control from ___% rod density to ___% reactor power as sensed by ______________________. (1.0)

Candidates were given IC of all rods in group at same height, then one rod withdrawn one notch.

b. What 2 ROD BLOCKS are used to enforce control rod sequences when in GNC? (1.0)

c. When in GNC with the Sequence Mode Selector (SMS) switch in normal, how does the RSCS determine control rod position? (1.0)

3.2 a. Assuming all APRM channels are initially operable, what effect (if any) will manually bypassing APRM channel B have on BOTH RBM channels? Explain why. (1.0)

b. A control rod surrounded by three (3) strings of LPRM's is selected for movement. How many of the LPRM's could be bypassed and/or failed without giving a RBM trip? Explain. (0.5)

3.3 With the Unit operating a 75% power, an electrical fault causes the Maximum Combined Flow setpoint to drop to zero. How will the following parameters RESPOND after the fault and WHY? Consider their response for ONE MINUTE following the fault. Assume NO OPERATOR ACTION. Attached FIGURE 3.3, EHC Logic, is provided for reference.

a. Turbine control valve position (1.0)

b. Bypass valve position (1.0)

c. Reactor power & pressure (1.0)

3.4 The plant is operating at 80% power with the FWCS in single element control and with LT-3-53 selected as the FWCS input. An instrument technician assigned to perform an instrument calibration on LT-3-60 mistakenly goes to LT-3-53 and begins opening the equalizing valve. Explain WHAT will happen to the plant and WHY; answer on the attached handout page and refer to figure 3.4 as necessary. (3.0)

NOTE: Limit your answer to the effects on and of FW, FWCS, RPS, level indicating and actual vessel level as directed by the handout.

3.5 a. GOI 100-3 directs that when LPRM maintenance is of such a magnitude that any one APRM is made inoperable, the RPS will be placed in the SRM non-coincidence scram mode. EXPLAIN what is meant by "SRM non-coincidence scram mode". Your discussion should include two (2) significant consequences of entering that mode of operation. (1.5)

b. TRUE or FALSE

1. The reactor period indication is no longer valid after the SRM detectors are withdrawn to their lower stop. (.5)

2. An SRM short periou of <30 seconds produces a rod withdrawal block. (.5)

3.6 Regarding diesel generator Control Circuits:

On Panel 9-23 there are two (2) Backfeed Switches. What 2 automatic actions result when these switches are placed in BACKFEED and what manual action is then possible? Be specific. (1.5)

3.7 Answer the following with regard to the Process Radiation Monitoring System:

a. When the reactor building ventilation exhaust reaches its trip level of 100 mR/HR five automatic actions are initiated. List FOUR of these automatic actions. (1.2)

b. List the four process streams which are monitored by the Liquid Radiation Monitoring System. (0.8)

4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL

4.1 OI-68 states that Unit 2's recirculation pumps may experience some speed induced vibration when passing through the critical speed zone of 1100 to 1200 rpm.

a. What precautions are taken on Unit 2 when passing through the critical speed zones to minimize vibration problems? (.75)

b. Why are these precautions necessary on Unit 2 but not on Units 1 and 3? (.75)

4.2 With regard to OI 47 (Turbine Generator):

a. During turbine startup, EXPLAIN WHY you are cautioned not to operate below 800 rpm for greater than five (5) minutes. (1.0)

b. The procedure states "critical speeds of the unit must be known by the operating personnel." WHAT are these speeds for Unit 2 and WHY is operation at these speeds undesirable? (1.0)

c. After synchronizing, the voltage regulator transfer volt meter is reading -5 volts. What is actually measured and HOW is it nulled? (1.0)

4.3 In reference to the Cold Startup Section of GOI-100-1:

a. During the heatup WHAT THREE (3) parameters are required to be recorded every 15 minutes? (1.0)

b. List 3 of 4 action steps required prior to transferring the mode switch to RUN. Include required and/or expected parameter values in your list. (1.5)

NOTE: ACTION STEPS MAY HAVE MULTIPLE ACTION AND/OR CHECKS

4.4 In response to GOI 100-12, Normal Shutdown from Power:

a. If the RSCS is found inoperable at 15% reactor power during the shutdown, WHAT must be done? (0.5)

b. With the unit in cold shutdown and the reactor pressure vessel (RPV) head torqued, WHAT are TWO indications of reactor vessel water stratification, OTHER THAN indication of pressure on the RPV? (2.0)

c. WHAT is the moderator temperature range which must be maintained during cold shutdown? (0.5)

4.5 Regarding EOI-36, Loss of Coolant Accident Inside Drywell:

a. WHAT TWO (2) criteria are to be used to determine the difference between excessive primary coolant leakage and a loss of coolant accident (LOCA)? (1.0)

b. When attempting to locate and isolate the break, the procedure cautions the operator to ensure sufficient reactor coolant inventory to keep the core covered before isolating a line break. WHY is this caution necessary? (1.0)

4.6 According to EOI-41, Methods of Water Makeup to the Reactor, WHAT are FOUR (4) plant systems which can be used to supply HIGH pressure water makeup to the reactor with NO NUCLEAR STEAM AVAILABLE? (2.0)

4.7 Answer the following regarding the operator actions of GOI-100-11 for a REACTOR SCRAM with MSIV's closed:

a. WHY is it preferable to manually operate the MSRVs to control reactor pressure rather than allowing them to cycle on their own? (1.0)

b. At WHAT torus temperature is it necessary to place an RHR loop in torus cooling? (0.5)

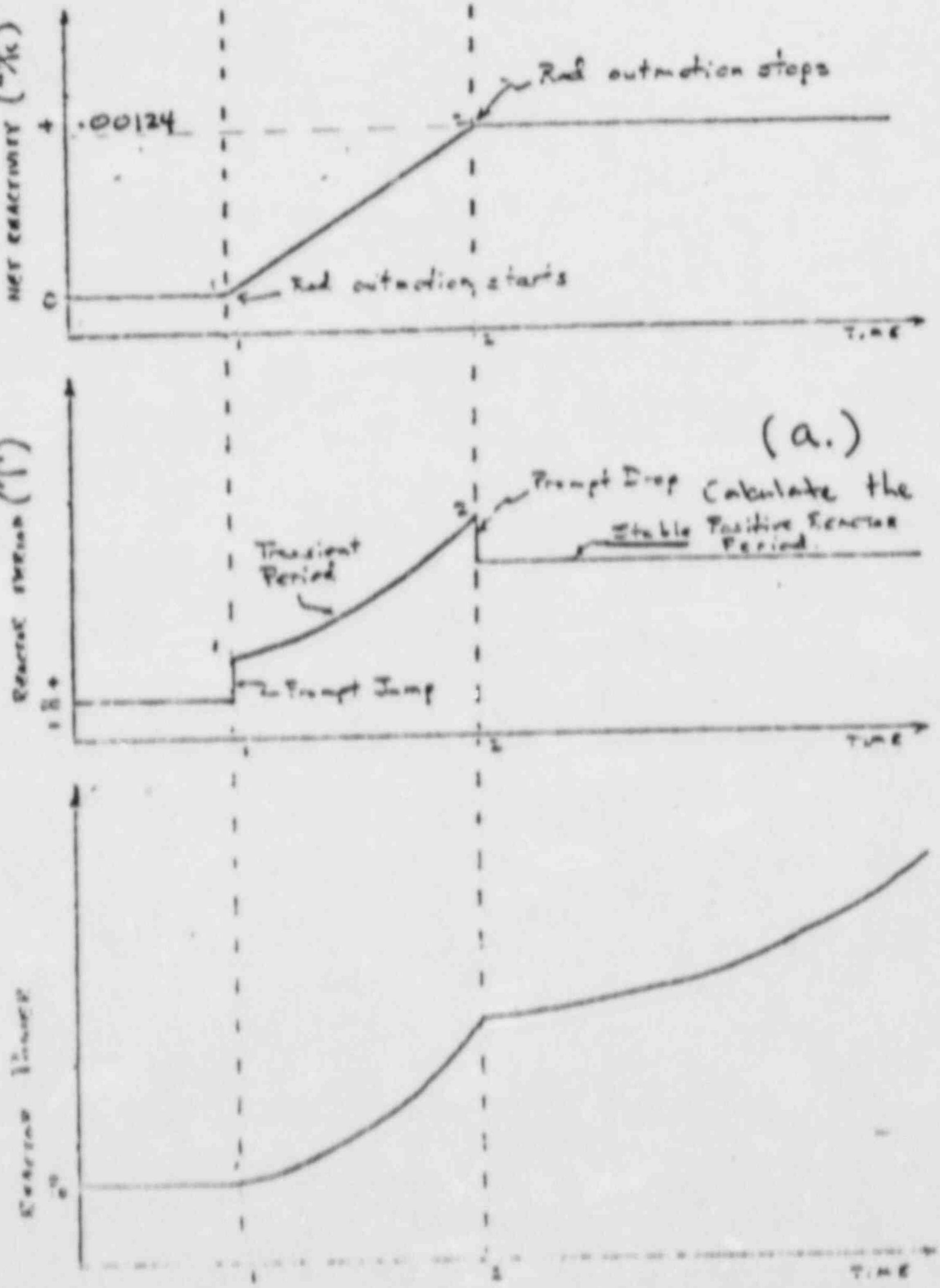


FIGURE 1.1 TRANSIENT PERIOD

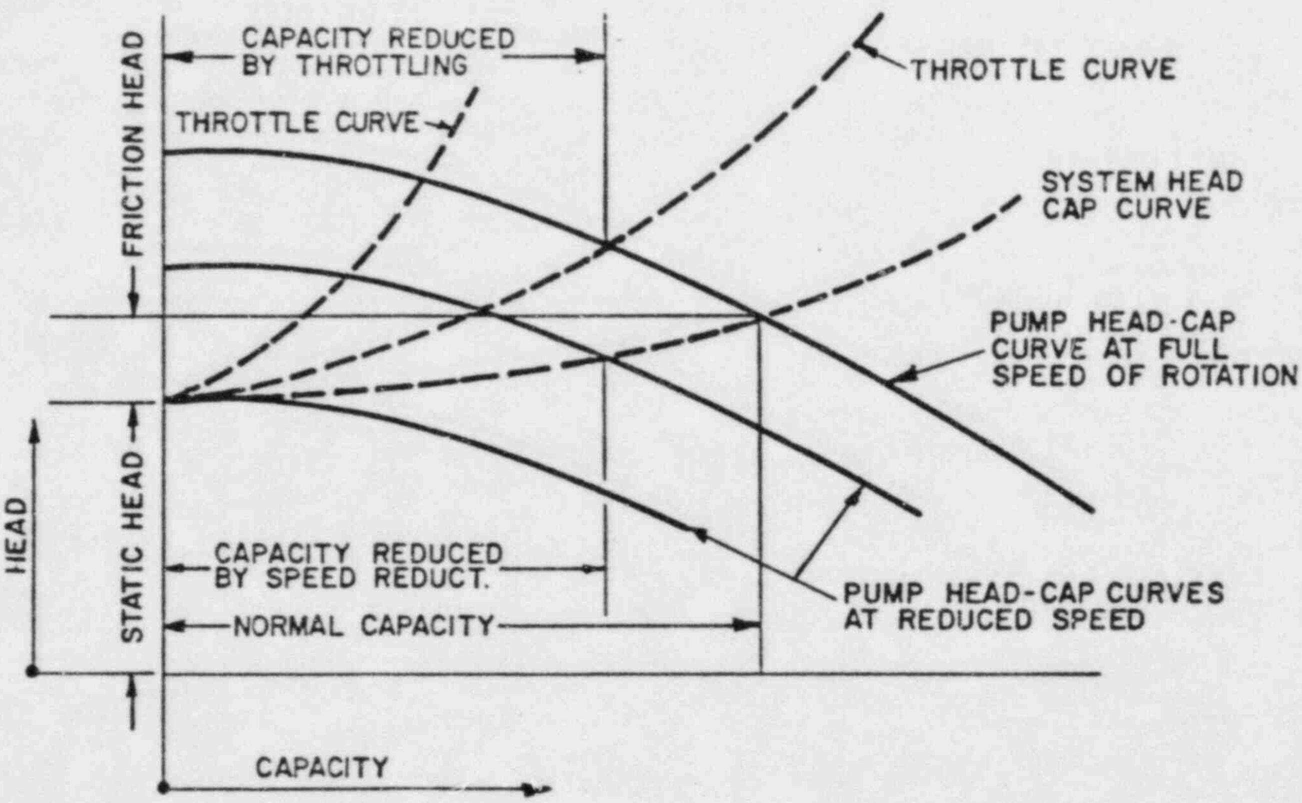
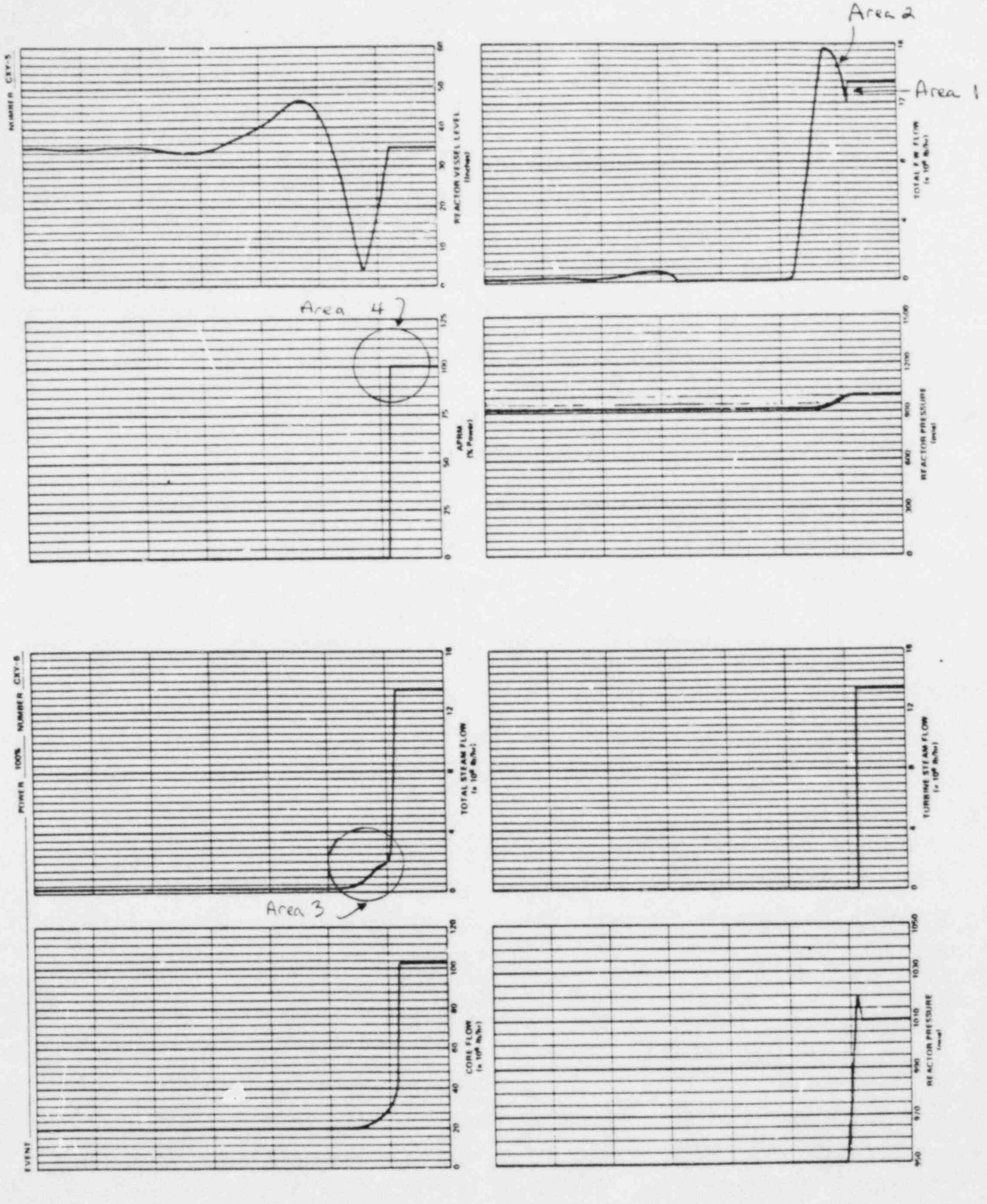


FIGURE 1.2 PUMP OPERATION IN A SYSTEM



Area 2
Area 1
Area 4
Area 3
REACTOR VESSEL LEVEL
TOTAL F W FLOW
APRM
REACTOR PRESSURE
TOTAL STEAM FLOW
TURBINE STEAM FLOW
CORE FLOW
REACTOR PRESSURE
POWER 100% NUMBER CXY-6
EVENT

Figure 1.5

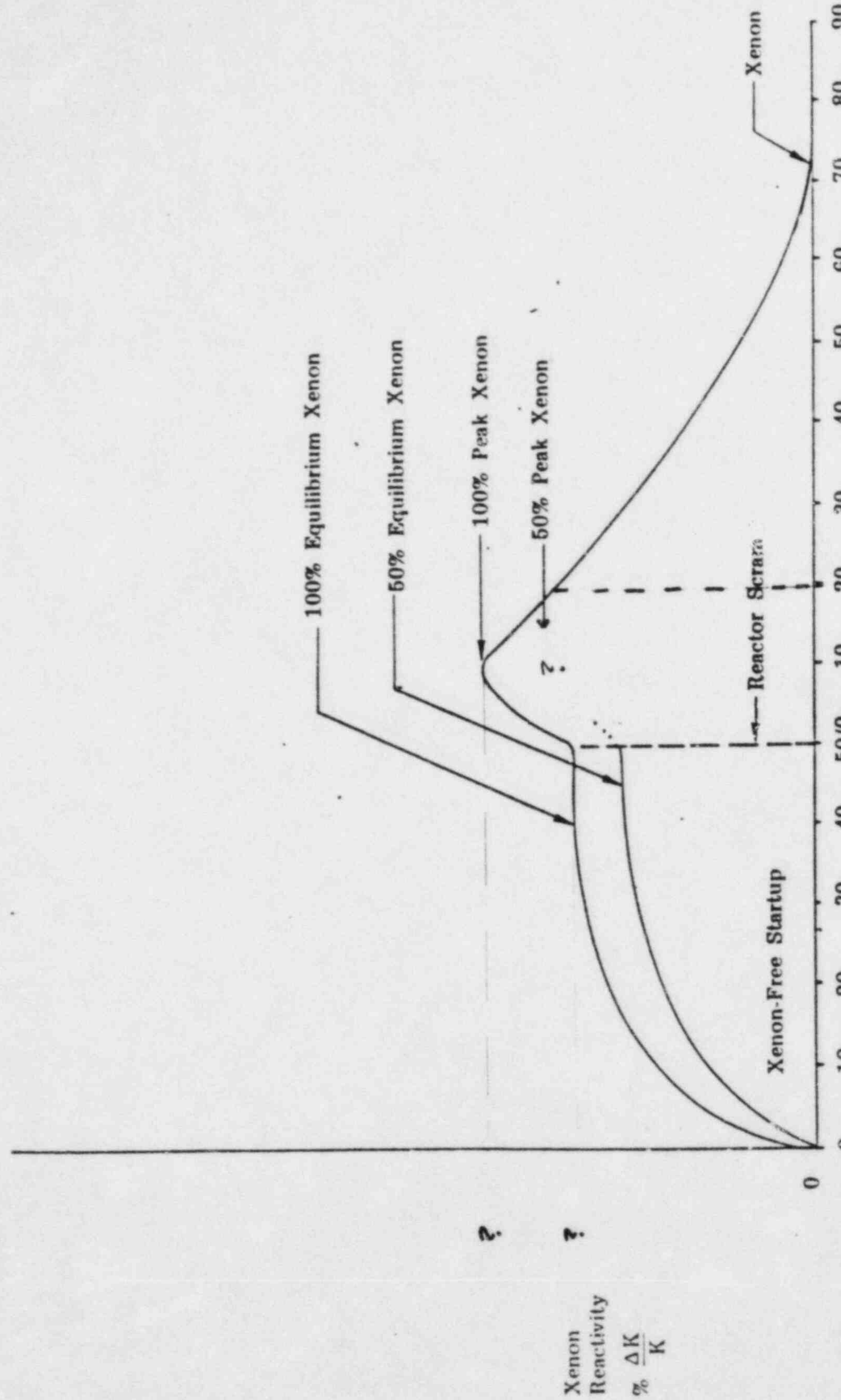


FIGURE 1.6 XENON REACTIVITY VS. POWER LEVEL AND TIME

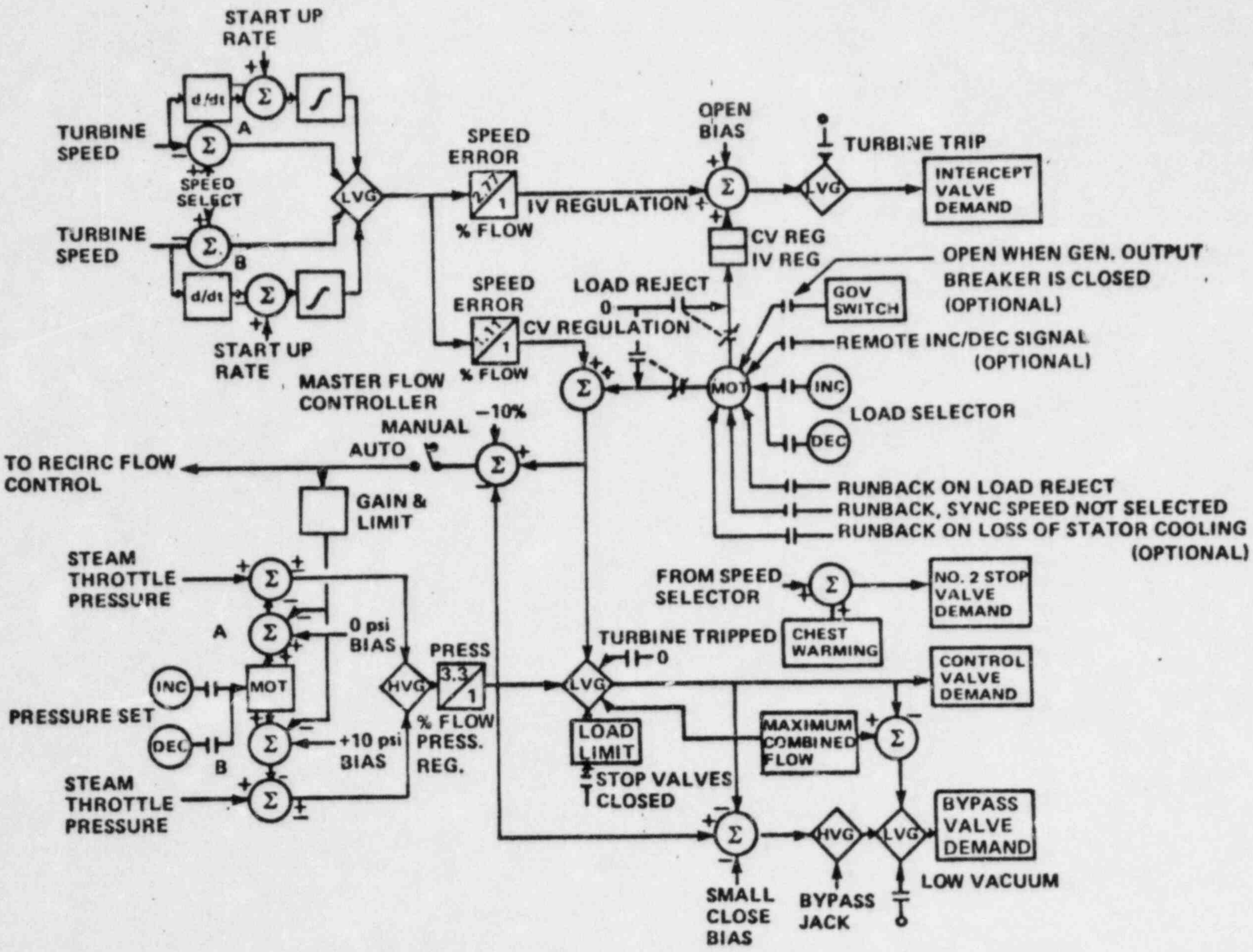


FIGURE 3.3 Electro-Hydraulic Control Logic

FIGURE 3.3 for Question 3.3

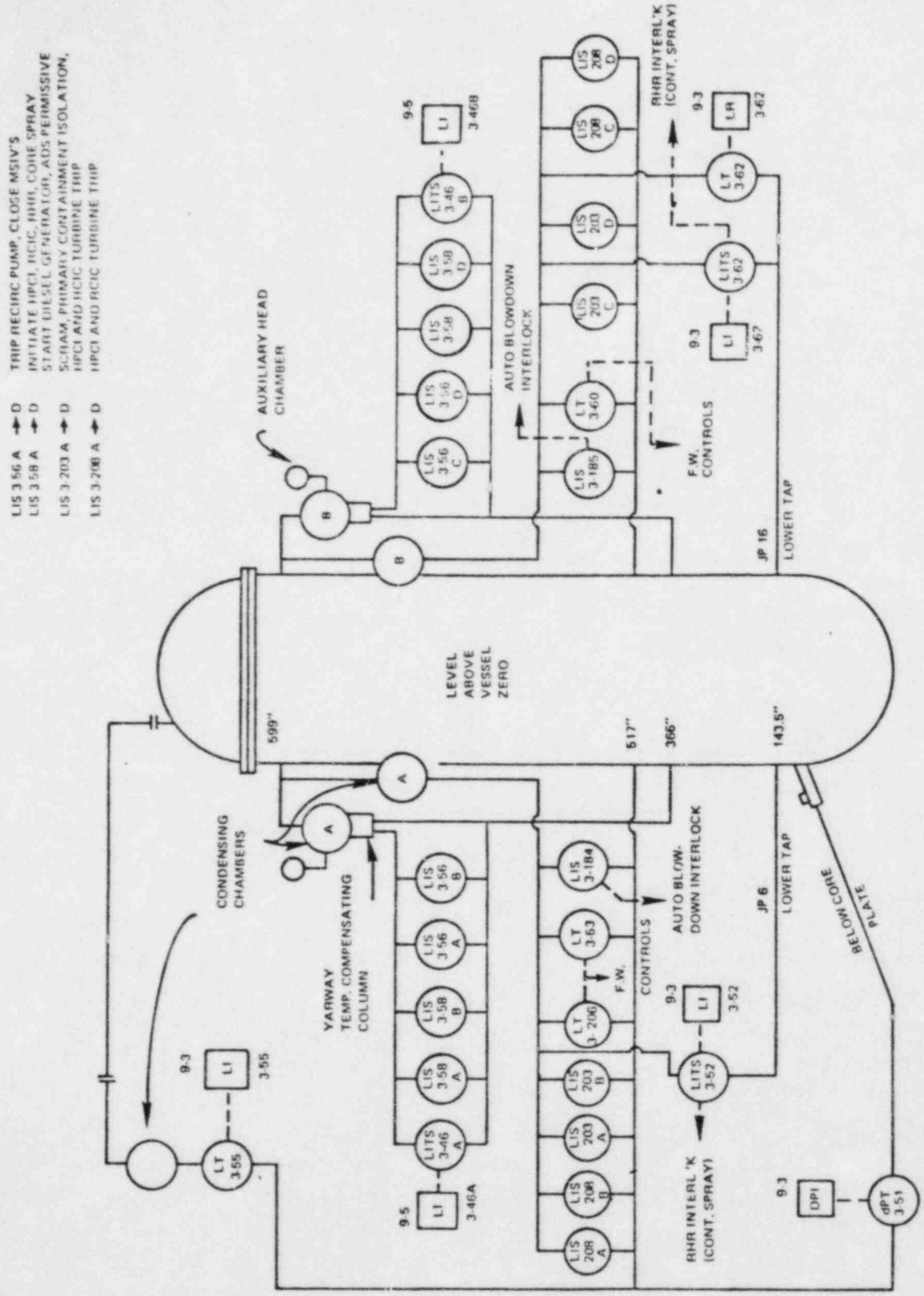


FIGURE 3.4 Reactor Vessel Level Instrumentation

FOR USE IN ANSWERING QUESTION NUMBER 2.04

TRIP	INITIATING SIGNALS WITH SETPOINTS	BYPASSES (IF ANY)	COMPONENTS ACTUATED
ATWS			
RPT			

FOR USE IN ANSWERING QUESTION NUMBER 3.04

1. LT-3-206 INDICATES ______________ (↑, →, ↓)
2. FWCS RESPONSE ______________

   ____
3. RFP ACTION(S) ______________

   ____
4. ACTUAL VESSEL LEVEL ______________ (↑, →, ↓)
5. FUNCTIONING OF RPS INCLUDING REASON(S) AND EFFECT(S) ______________

   ____
6. INDICATION: LITS 3-46A ______________
   LITS 3-52 ______________
   LITS 3-46B ______________ (↑, →, ↓)

Table 2: Saturated Steam: Pressure Table

Abs Press Lb/Sq In	Temp Fahr	Specific Volume			Enthalpy			Entropy			Abs Press Lb/Sq In
p	t	Sat Liquid v_f	Evap v_{fg}	Sat Vapor v_g	Sat Liquid h_f	Evap h_{fg}	Sat Vapor h_g	Sat Liquid s_f	Evap s_{fg}	Sat Vapor s_g	p
0.08865	32.018	0.016022	3302.4	3302.4	0.0003	1075.5	1075.5	0.0000	2.1872	2.1872	0.08865
0.25	59.323	0.016032	1235.5	1235.5	27.382	1060.1	1087.4	0.0542	2.0425	2.0967	0.25
0.50	79.586	0.016071	641.5	641.5	47.623	1048.6	1096.3	0.0925	1.9446	2.0370	0.50
1.0	101.74	0.016136	333.59	333.60	69.73	1036.1	1105.8	0.1326	1.8455	1.9781	1.0
5.0	162.24	0.016407	73.515	73.532	130.20	1000.9	1131.1	0.2349	1.6094	1.8443	5.0
10.0	193.21	0.016592	38.404	38.420	161.26	982.1	1143.3	0.2836	1.5043	1.7879	10.0
14.696	212.00	0.016719	26.782	26.799	180.17	970.3	1150.5	0.3121	1.4447	1.7568	14.696
15.0	213.03	0.016726	26.274	26.290	181.21	969.7	1150.9	0.3137	1.4415	1.7552	15.0
20.0	227.96	0.016834	20.070	20.087	196.27	960.1	1156.3	0.3358	1.3962	1.7320	20.0
30.0	250.34	0.017009	13.7266	13.7436	218.9	945.2	1164.1	0.3682	1.3313	1.6995	30.0
40.0	267.25	0.017151	10.4794	10.4965	236.1	933.6	1169.8	0.3921	1.2844	1.6765	40.0
50.0	281.02	0.017274	8.4967	8.5140	250.2	923.9	1174.1	0.4112	1.2474	1.6586	50.0
60.0	292.71	0.017383	7.1562	7.1736	262.2	915.4	1177.6	0.4273	1.2167	1.6440	60.0
70.0	302.93	0.017482	6.1875	6.2050	272.7	907.8	1180.6	0.4411	1.1905	1.6316	70.0
80.0	312.04	0.017573	5.4536	5.4711	282.1	900.9	1183.1	0.4534	1.1675	1.6208	80.0
90.0	320.28	0.017659	4.8779	4.8953	290.7	894.6	1185.3	0.4643	1.1470	1.6113	90.0
100.0	327.82	0.017740	4.4133	4.4310	298.5	888.6	1187.2	0.4743	1.1284	1.6027	100.0
110.0	334.79	0.01782	4.0306	4.0484	305.8	883.1	1188.9	0.4834	1.1115	1.5950	110.0
120.0	341.27	0.01789	3.7097	3.7275	312.6	877.8	1190.4	0.4919	1.0960	1.5879	120.0
130.0	347.33	0.01796	3.4364	3.4544	319.0	872.8	1191.7	0.4998	1.0815	1.5813	130.0
140.0	353.04	0.01803	3.2010	3.2190	325.0	868.0	1193.0	0.5071	1.0681	1.5752	140.0
150.0	358.43	0.01809	2.9958	3.0139	330.6	863.4	1194.1	0.5141	1.0554	1.5695	150.0
160.0	363.55	0.01815	2.8155	2.8336	336.1	859.0	1195.1	0.5206	1.0435	1.5641	160.0
170.0	368.42	0.01821	2.6556	2.6738	341.2	854.8	1196.0	0.5269	1.0322	1.5591	170.0
180.0	373.08	0.01827	2.5129	2.5312	346.2	850.7	1196.9	0.5328	1.0215	1.5543	180.0
190.0	377.53	0.01833	2.3847	2.4030	350.9	846.7	1197.6	0.5384	1.0113	1.5498	190.0
200.0	381.80	0.01839	2.2689	2.2873	355.5	842.8	1198.3	0.5438	1.0016	1.5454	200.0
210.0	385.91	0.01844	2.16373	2.18217	359.9	839.1	1199.0	0.5490	0.9923	1.5413	210.0
220.0	389.88	0.01850	2.06779	2.08629	364.2	835.4	1199.6	0.5540	0.9834	1.5374	220.0
230.0	393.70	0.01855	1.97991	1.99846	368.3	831.8	1200.1	0.5588	0.9748	1.5336	230.0
240.0	397.39	0.01860	1.89909	1.91769	372.3	828.4	1200.6	0.5634	0.9665	1.5299	240.0
250.0	400.97	0.01865	1.82452	1.84317	376.1	825.0	1201.1	0.5679	0.9585	1.5264	250.0
260.0	404.44	0.01870	1.75548	1.77418	379.9	821.6	1201.5	0.5722	0.9508	1.5230	260.0
270.0	407.80	0.01875	1.69137	1.71013	383.6	818.3	1201.9	0.5764	0.9433	1.5197	270.0
280.0	411.07	0.01880	1.63169	1.65049	387.1	815.1	1202.3	0.5805	0.9361	1.5166	280.0
290.0	414.25	0.01885	1.57597	1.59482	390.6	812.0	1202.6	0.5844	0.9291	1.5135	290.0
300.0	417.35	0.01889	1.52384	1.54274	394.0	808.9	1202.9	0.5882	0.9223	1.5105	300.0
350.0	431.73	0.01912	1.30642	1.32554	409.8	794.2	1204.0	0.6059	0.8909	1.4968	350.0
400.0	444.60	0.01934	1.14162	1.16095	424.2	780.4	1204.6	0.6217	0.8630	1.4847	400.0
450.0	456.28	0.01954	1.01224	1.03179	437.3	767.5	1204.8	0.6360	0.8378	1.4738	450.0
500.0	467.01	0.01975	0.90787	0.92762	449.5	755.1	1204.7	0.6490	0.8148	1.4639	500.0
550.0	476.94	0.01994	0.82183	0.84177	460.9	743.3	1204.3	0.6611	0.7936	1.4547	550.0
600.0	486.20	0.02013	0.74962	0.76975	471.7	732.0	1203.7	0.6723	0.7738	1.4461	600.0
650.0	494.89	0.02032	0.68811	0.70843	481.9	720.9	1202.8	0.6828	0.7552	1.4381	650.0
700.0	503.08	0.02050	0.63505	0.65556	491.6	710.2	1201.8	0.6928	0.7377	1.4304	700.0
750.0	510.84	0.02069	0.58880	0.60949	500.9	699.8	1200.7	0.7022	0.7210	1.4232	750.0
800.0	518.21	0.02087	0.54809	0.56896	509.8	689.6	1199.4	0.7111	0.7051	1.4163	800.0
850.0	525.24	0.02105	0.51197	0.53302	518.4	679.5	1198.0	0.7197	0.6899	1.4096	850.0
900.0	531.95	0.02123	0.47968	0.50091	526.7	669.7	1196.4	0.7279	0.6753	1.4032	900.0
950.0	538.39	0.02141	0.45064	0.47205	534.7	660.0	1194.7	0.7358	0.6612	1.3970	950.0
1000.0	544.58	0.02159	0.42436	0.44596	542.6	650.4	1192.9	0.7434	0.6476	1.3910	1000.0
1050.0	550.53	0.02177	0.40047	0.42224	550.1	640.9	1191.0	0.7507	0.6344	1.3851	1050.0
1100.0	556.28	0.02195	0.37863	0.40058	557.5	631.5	1189.1	0.7578	0.6216	1.3794	1100.0
1150.0	561.82	0.02214	0.35859	0.38073	564.8	622.2	1187.0	0.7647	0.6091	1.3738	1150.0
1200.0	567.19	0.02232	0.34013	0.36245	571.9	613.0	1184.8	0.7714	0.5969	1.3683	1200.0
1250.0	572.38	0.02250	0.32306	0.34556	578.8	603.8	1182.6	0.7780	0.5850	1.3630	1250.0
1300.0	577.42	0.02269	0.30722	0.32991	585.6	594.6	1180.2	0.7843	0.5733	1.3577	1300.0
1350.0	582.32	0.02288	0.29250	0.31537	592.3	585.4	1177.8	0.7906	0.5620	1.3525	1350.0
1400.0	587.07	0.02307	0.27871	0.30178	598.8	576.5	1175.3	0.7966	0.5507	1.3474	1400.0
1450.0	591.70	0.02327	0.26584	0.28911	605.3	567.4	1172.8	0.8026	0.5397	1.3423	1450.0
1500.0	596.20	0.02346	0.25372	0.27719	611.7	558.4	1170.1	0.8085	0.5288	1.3373	1500.0
1550.0	600.59	0.02366	0.24235	0.26601	618.0	549.4	1167.4	0.8142	0.5182	1.3324	1550.0
1600.0	604.87	0.02387	0.23159	0.25545	624.2	540.3	1164.5	0.8199	0.5076	1.3274	1600.0
1650.0	609.05	0.02407	0.22143	0.24551	630.4	531.3	1161.6	0.8254	0.4971	1.3225	1650.0
1700.0	613.13	0.02428	0.21178	0.23607	636.5	522.2	1158.6	0.8309	0.4867	1.3176	1700.0
1750.0	617.17	0.02450	0.20263	0.22713	642.5	513.1	1155.6	0.8363	0.4765	1.3128	1750.0
1800.0	621.02	0.02472	0.19390	0.21861	648.5	503.8	1152.3	0.8417	0.4662	1.3079	1800.0
1850.0	624.83	0.02495	0.18558	0.21052	654.5	494.6	1149.0	0.8470	0.4561	1.3030	1850.0
1900.0	628.56	0.02517	0.17761	0.20278	660.4	485.2	1145.6	0.8522	0.4459	1.2981	1900.0
1950.0	632.22	0.02541	0.16999	0.19540	666.3	475.8	1142.0	0.8574	0.4358	1.2931	1950.0
2000.0	635.80	0.02565	0.16266	0.18831	672.1	466.2	1138.3	0.8625	0.4256	1.2881	2000.0
2100.0	642.76	0.02615	0.14885	0.17501	683.8	446.7	1130.5	0.8727	0.4053	1.2780	2100.0
2200.0	649.45	0.02669	0.13603	0.16272	695.5	426.7	1122.2	0.8828	0.3848	1.2676	2200.0
2300.0	655.89	0.02727	0.12406	0.15133	707.2	406.0	1113.2	0.8929	0.3640	1.2569	2300.0
2400.0	662.11	0.02790	0.11287	0.14076	719.0	384.8	1103.7	0.9031	0.3430	1.2460	2400.0
2500.0	668.11	0.02859	0.10209	0.13068	731.7	361.6	1093.3	0.9139	0.3206	1.2345	2500.0
2600.0	673.91	0.02938	0.09172	0.12110	744.5	337.6	1082.0	0.9247	0.2977	1.2225	2600.0
2700.0	679.53	0.03029	0.08165	0.11194	757.3	312.3	1069.7	0.9356	0.2741	1.2097	2700.0
2800.0	684.96	0.03134	0.07171	0.10305	770.7	285.1	1055.8	0.9468	0.2491	1.1958	2800.0
2900.0	690.22	0.03262	0.06158	0.09420	785.1	254.7	1039.8	0.9588	0.2215	1.1803	2900.0
3000.0	695.33	0.03428	0.05073	0.08500	801.8	218.4	1020.3	0.9728	0.1891	1.1619	3000.0
3100.0	700.28	0.03681	0.03771	0.07452	824.0	167.3	991.3	0.9914	0.1460	1.1373	3100.0
3200.0	705.08	0.04472	0.01191	0.05663	875.5	56.1	931.6	1.0351	0.0482	1.0832	3200.0
3208.2*	705.47	0.05078	0.00000	0.05078	906.0	0.0	906.0	1.0612	0.0000	1.0612	3208.2*

*Critical pressure

EQUATION SHEET

$f = ma$

$v = s/t$

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

$w = mg$

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

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

$W = v\ \Delta P$

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

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

$\Delta E = 931\ \Delta m$

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

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

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

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

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

$Pwr = W_f \Delta h$

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

$TVL = 1.3/\mu$

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

$HVL = -0.693/\mu$

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

$SUR = 26.06/T$

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

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

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

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

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

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

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

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

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

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

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

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

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

1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
ANSWERS

1.1 a. Using: $T = \frac{\beta - \rho}{\lambda \times \rho}$

Assuming: $\beta = .0071$
$\lambda = 0.08/\text{sec}$ for $\lambda = 0.1/\text{sec}$
$T = \frac{(.0071)\ (.00124)}{(.08/\text{sec})\ (.00124)}$ ⇒ 46.45 sec.
$T = \frac{.00586}{.0000992}$ sec = 59.07 sec (1.0)

b. ° Power will stabilize at about 0.1% to 1.0% [.5]

° Assuming moderator coefficient of -1 E -4 delta k/k/F [.2] then a 12.4 F temperature increase will cancel the +0.00124 delta k/k added by the rod withdrawal giving a final temperature of 293.4 F.[.3]

° Interpolating in the steam tables, 293.4 F equates to Psat of 60.7 PSIA [.25] which when converted to gage pressure equals 46 PSIG [.25]. (1.5)

Reference: BFN Reactor Theory Requalification LP, Objectives 14, 16, 17, Problems 14.a; BFN Reactor Physics Review, page 29, Figure 28.

1.2 a. Running the two pumps in parallel at half speed is more efficient. (.5)

b. Reducing pump speed is preferred [.5] because it does not increase system head loss OR less pump head must be developed OR less power is wasted [0.5]. or More efficient system operation (1.0)

Reference: BFN Char. of Cent Pumps Requalification LP, Objective 6

1.3 a. - Power, Local power, Flux, or Local flux
- Flow
- Pressure
- Inlet subcooling
(3 of 4 req. @ 0.333 each) (1.0)

b. 1. MAPRAT is the ratio of MAPLHGR (act) to MAPLHGR (LCO). [0.75]
2. NO [0.5]
3. The clad temperature can exceed 2200 degree F during a DBA LOCA. [0.75] (2.0)

Reference: BRF Plant Performance 10.2
BRF Requalification BWR Thermal Hyd. Review pg. 22-25 & 31-33

1.4 a. Doppler or fuel temperature
b. Void
c. Moderate temperature
d. Void
e. Moderate Temperature
(0.5 each) (2.5)

Reference: BRF Rx Physics Review pg. 31, 39 & 40
BRF Requal Rx Theory pg. 27-30

1.5 a. FWCS cuts back to follow steam flow [.5] and then increases feed flow in response to low Rx level [.5] (1.0)

b. The BPV's are opening to control pressure. (.5)

c. The scram is anticipatory based on TSV position. (.5)

Reference: BFNP Transient CXY-5

1.6 a. 100% equil. xenon = -3% delta k/k [.25]
100% peak xenon = -4% delta k/k [.25] (.5)

b. The peak would occur at about 7 hours. (.5)

c. See figure 1.6 at the end of the answer key. (1.0)

Reference: BFNP Reactor Theory Requal LP Objective 2C

1.7 a. As temperature increases, migration length increases exposing the control rod to increased thermal neutron flux, thus rod worth increases. [0.67]

b. As an adjacent rod is inserted the remaining rods zone of control decreases, the rod is exposed to a reduced thermal flux thus the rod worth will decrease. [0.66]

c. Differential rod worth is greatest as the rod end travels through the peak axial thermal neutron flux. [0.67] (2.0)

Reference: BRF Rx Physics Review pgs. 35-37

Reference: BRF Recirc L.P. & Recirc Flow Control L.P.
BRF Requal Recirc L.P. pgs. 14 & 24, Objective 9
Warning plate on plant EHC panel; LER 84-014-01

2.5 a.
- preheater
- catalytic recombiner
- off gas condenser
- holdup pipe
- gas reheater
- charcoal absorbers (1.25)

b. If either of the two post treatment rad monitors sees a high, (high high, or high high high) alarm. (.75)

Reference: BFNP Off-Gas System LP, Objectives 2,4
BF OI-66

2.6 A scram signal deenergizes the scram pilot valves [0.33], venting air from the scram inlet and outlet valves, allowing them to open [0.33]. This vents water from the overpiston area of the CRD to the SDV [0.33] and applies HCU accumulator water to the underpiston area of the CRD [0.33]. This provides the initial motive force for the rod [0.33]. As accumulator pressure drops below reactor pressure, a ball check valve in the CRD opens to apply reactor pressure to the CRD to complete the scram stroke [0.33], (2.0)

Reference: BFNP CRD Hydraulics Lesson Plan
BFNP CRD Rod Blade and Drive Mechanics LP

3. INSTRUMENTS AND CONTROLS
ANSWERS

3.1 a. <50% rod density [0.33] to 30% power [0.33] as sensed by turbine first stage pressure. [0.33] (1.0)

b. When in GNC, rod blocks are enforced which:

1. Prevent any rod in the group from being moved in opposite direction from the first rod moved. [0.5]

2. Prevent any rod in group from being moved >1 notch in direction of first movement. [0.5] (1.0)

c. Rod position is determined by sensing direction in which the rod movement control switch is moved [0.5] & sensing RMCS timer settle function. [0.5] (1.0)

Reference: BRF RSCS L.P.

3.2 a. RBM A is not affected [0.5] RBM channel B uses APRM B as its normal reference input but when the APRM is bypassed the alternate APRM D is automatically placed in the circuit. Hence, no adverse effect on either RBM channel [0.5] (1.0)

b. Three (3) [0.25] there are 12 LPRM's, 6 in RBM A & 6 in B. Must have at least 50% operable/channel; hence 3/CHANNEL - 6 total [0.25] (0.5)

Reference: BRF RBM L.P.
BRF Requal RBM L.P. Objective 2, 3 & 5 pg. 10, 15-16 & 21

3.3 a. The TCV's will close to the 50% steam flow/25% valve position ~~fully.~~ [0.5] The TCV LVG passes ~~ing~~ ~~the~~ a MCF signal of 50% rather than the signal from the pressure controller. [0.5] (1.0)

b. The BPV's will remain closed through the transient. [0.5] The MCF summer will send a zero signal to the BPV LVG. [0.5] (1.0)

c. Reactor power & pressure will rapidly increase following the fault. [0.5] The reactor will scram on High Flux &/OR High Pressure. Reactor pressure will be controlled by the ~~SRV's~~ TCV's. [0.5] (1.0)

Reference: BRF EHC L.P.
Transient HXY-16
BF SMG 147 - 1.2, P. 34
P24B-AL-4985, P 20A

3.4 1. increasing level (.5)
2. FWCS will send sgnal to runback the RFP's (.5)
3. runback and then trip on high level (from LT-3-206 & LT-3-63) (.5)
4. actual level is decreasing (.5)
5. RPS scram on low level from LIS-203 C&D OR HIGH LEVEL => TT => Scram (.5)
6. LITS-3-46A decreases (.167)
LITS-3-52 stays same (.167)
LITS-3-46B decreases (.167)

Reference: BFNP FWCS & Inst. LP's
I & E Notice 81-25

3.5 a. 1. Any single SRM, IRM, or APRM trip will result in a trip of both RPS channels, (i.e., a full scram)

2. The SRM high high trip (5 E + 5 CPS) becomes functional. (.75 ea/1.5)

b. 1. False (.5)
2. False (.5)

Reference: BFNP RPS LP, P21, Objective 4
BFNP SRM LP, P24, Objective 2
GOI 100-3, P10

3.6 a. Placing these switches in Backfeed will trip and lockout the normal and alternate supply breakers on the associated Unit board [0.75] and allow closing of the unit board to shutdown bus breakers for backfeed operations. [0.75] (1.5)

Reference: BRF Requal Diese en. L.P.

3.7 a. 1. Isolates Rx Bldg and refueling zone ventilation system
2. Starts the SBGTS
3. Closes the Primary Containment Purge and vent valves
4. Isolates the control bay ventilation
5. Starts CREV. (4 of 5 @ .3 ea/1.2)

b. 1. RHRSW
2. RCW
3. RBCCW
4. RADWASTE (.2 ea/.8)

Reference: BFNP Process Rad Mon LP, P 26, 20-22

4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
ANSWER

4.1 a. (Place the loop flow controllers in manual and) maintain a 6-8% speed differential until pump speeds are past the critical speed zones. (.75)

b. Unit 2 recirc. pipe inspections showed indications of fatigue induced cracks on some (sweepolet-to-manifold) welds. (.75)

Reference: BFNP OI-68, P3, 23, ATTACHMENT B

4.2 a. Operation at low speed may result in large rotor bows being generated by rubbing with no indication of high vibration from the turbine supervisory instruments. OR - Low steam flow to last stages during s/u ⇒ Significant heating of last stage blading and inner casing [not per LP but ok for .7 partial credit] (1.0)

900-1100 and 1345-1415

b. ~~CAF on Unit 2 critical speeds~~. [0.5] Operation at critical speed would be to operate at the point of maximum vibration. [0.5] (1.0)

c. The transfer voltmeter reads the difference between the Auto (90P) and Manual voltage regulator outputs. [0.5] By adjusting the manual voltage regulator (70P) the meter is nulled. [0.5] (1.0)

Reference: BRF Turbine Generator OI 47 ; BFNP Main Turbine LP, P.14
Letter from Browns Ferry NP ; BFNP Main Gen. & Aux LP

4.3 a. Parameters

1. Steam dome pressure [0.33]
2. Reactor bottom head drain temperature [0.33]
3. Recirc loop A & B temperature [0.33] (1.0)
4. Moderator Temperature
5. Vessel Shell Temp. adjacent to Flange

b. Checks prior to transfer to RUN:

1. APRMs >5% & <12%
2. Inboard and outboard MSIVs open
3. Condenser vacuum >24" Hg vacuum, and condenser A, B, & C low vacuum annunciators cleared
4. Reactor pressure >850 PSIG and low reactor pressure annunciators cleared

(3 of 4 required at 0.5 each) (1.5)

Reference: BRF GOI-100-1, & BRF Q&A 5-4 & 5-127
BRP Tech. Spec./SI - 3.6.A.1/4.6.A.1

4.4 a. Manually scram the reactor. (0.5)

b. ° Temperature differential of >50 F between bottom vessel head and "A" FW sparger when moderate is >150 F.

° Temperature differential of >75 F between bottom vessel and "A" FW sparger when moderate is <150 F.

° FW sparger temperature of >200 F on either "A" or "B" FW sparger when recirc. pumps and shutdown cooling are OOS.

(2 of 3 at 1.0 each) (2.0)

c. Between 180 degrees F and 200 degrees F (.5)

References: BF GOI 100-12, pp. 1, 7, & 14
BFNP Requal Lesson Plan, "GOI-100-12," (Objectives 1, 3 & 4).

4.5 a. A break large enough to result in a reactor scram on High Drywell Pressure [0.5] or Low Reactor Water Level [0.5]. (1.0)

b. Isolation of line breaks may cause reactor pressure to increase rapidly [0.5] resulting in level shrink uncovering the core. [0.5] (1.0)

Reference: BFNP Requal Lesson Plan, "EOI-36" (Objective 1 & 4)

4.6 ° CRD
° SLC
° RCIC
° HPCI (0.5 each) (2.0)

Reference: BF EOI-41, pp. 1 & 2
BFNP Requal Lesson Plan, "EOI-41" (Objective 1).

4.7 a. To sequence the opening order per the MSRV position chart in order to minimize torus hot spots. (1.0)

b. >95 F. for Unit 1 ; >93°F for Unit 2 & 3 (either temp & unit) (0.5)

Reference: BF GOI-100-11, pp. 5-7.
BF OI-74, p-t
BF OI-64, p21

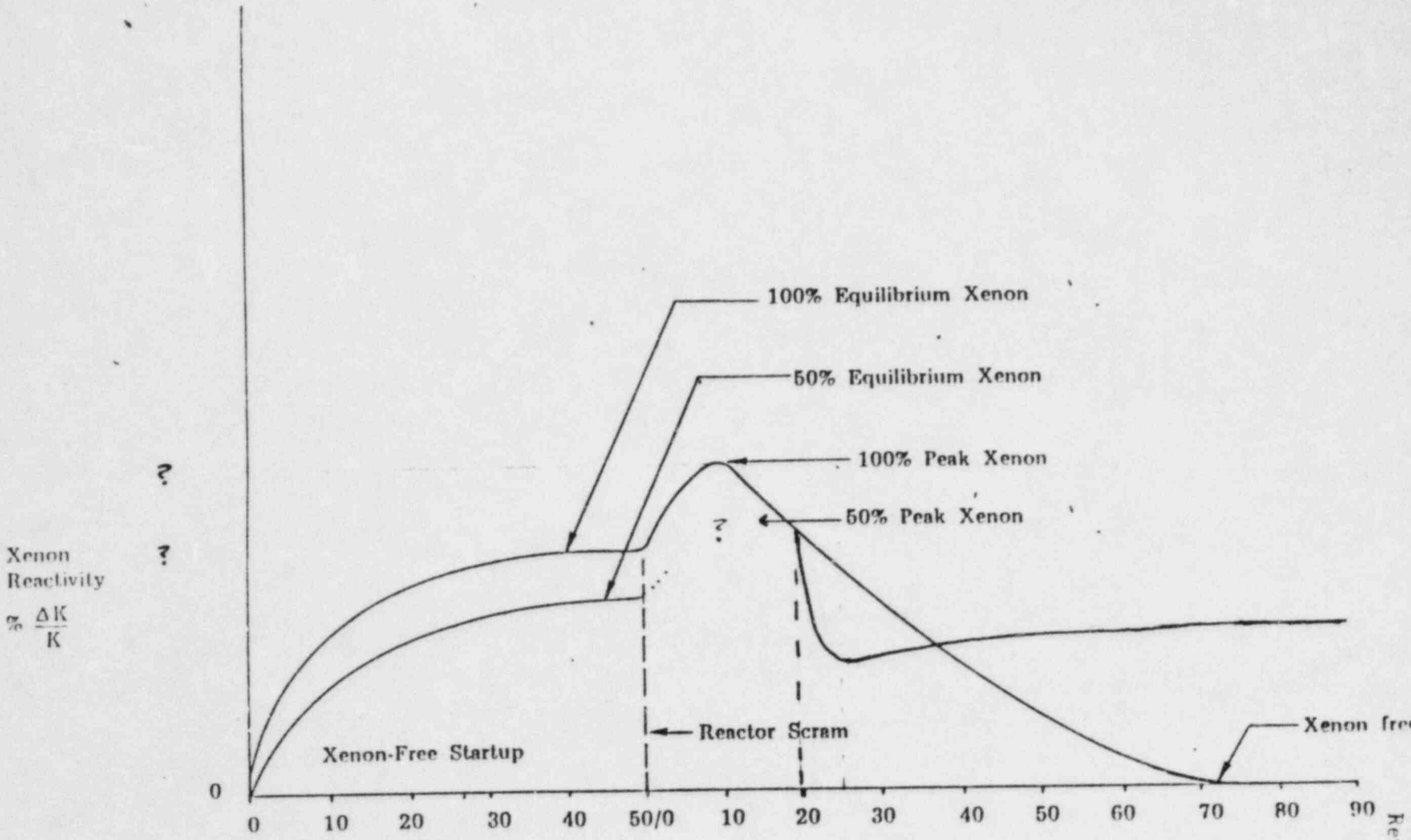


FIGURE [illegible] XENON REACTIVITY VS. POWER LEVEL AND TIME

—MASTER—

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

Reviewed by:
C H Noe
N Catron
Dexter] 1 hr. each
Miller]
Glover

Facility: Browns Ferry 1, 2, 3
Reactor Type: BWR
Date Administered: March 23, 1984
Examiner: S. Guenther
Candidate: ____________________

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%.

Category Value	% of Total	Candidate's Score	% of Category Value	Category
15.5	23.9			5. Theory of Nuclear Power Plant Operation, Fluids, and Thermodynamics
17.5	27.0			6. Plant Systems Design, Control, and Instrumentation
15.5	23.9			7. Procedures - Normal, Abnormal, Emergency, and Radiological Control
16.25	25.1			8. Administrative Procedures, Conditions, and Limitations
64.75	______ Final Grade			Totals

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

____
Candidate's Signature

5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS

5.1 Assume the reactor is shutdown with a shutdown margin of 1.8%. If the control rods are withdrawn until the count rate increased by a factor of 20 with the reactor still subcritical, WHAT is the new K-eff? SHOW ALL WORK. (2.0)

5.2 The tabulation below illustrates REACTIVITY COEFFICIENT VARIATIONS due to increases in several core parameters. PLACE ARROWS in the squares labeled (a) through (i) indicating how the ABSOLUTE VALUE of that coefficient varies if the indicated parameter is increased. (1.80)

CORE PARAMETER-> COEFFICIENT	^ Moderator Temp.	^ Core Voiding	^ Rod Density	^ Fuel Temp.	^ Core Age
Void Coefficient	->	(a)	(b)	(c)	(d)
Moderator Temp. Coefficient	(e)	->	^	^	(f)
Fuel Temperature Coefficient	^	(g) ↑	->	(h) ↓	(i) ↑

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

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

b. 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)

5.4 Referring to attached FIGURE 5.4, Operating Map for Units 2 & 3:

a. WHY does core flow increase with constant recirculation pump speed from Point 7 to Point 4? (1.0)

b. The APRM Rod Block Monitor Line provides protection from exceeding WHICH core thermal limit? (0.5)

5.5 With the plant 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 5.5:

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)

5.6 With regard to the MAPLHGR thermal limit:

a. Briefly, WHAT is the reason, or basis for having a MAPLHGR thermal limit? (.75)

b. WHICH TWO of the following four parameters affect the MAPLHGR LIMIT? (0.5)

1. Moderator Temperature
2. Type of fuel
3. Fuel exposure
4. Reactor pressure

c. If an OD-6, Option 4 is selected on the Process Computer, the program provides, among other things, MAPRAT. WHAT is the relationship between MAPRAT and MAPLHGR? (.75)

5.7 Attached FIGURE 5.7 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 show 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)

5.8 Answer the following regarding transient effects on core boiling heat transfer when operating at power?

a. Briefly EXPLAIN the immediate (instantaneous) effect of a sudden core flow INCREASE on the amount of NUCLEATE boiling at the clad surface. (1.0)

SECTION 5 CONTINUED ON NEXT PAGE

b. A sudden flow (INCREASE, DECREASE) in the core could cause film boiling. Choose the correct answer and briefly JUSTIFY your choice. (0.7)

6. PLANT SYSTEMS DESIGN, CONTROL AND INSTRUMENTATION

6.1 The plant is operating at 80% power with the FWCS in single element control and with LT-3-53 selected as the FWCS input. An instrument technician assigned to perform an instrument calibration on LT-3-60 mistakenly goes to LT-3-53 and begins opening the equalizing valve. Explain WHAT will happen to the plant and WHY; answer on the attached handout page and refer to figure 3.4 as necessary. (3.0)

NOTE: Limit your answer to the effects on and of FW, FWCS, RPS, level indicating and actual vessel level as directed by the handout.

6.2 When operating RHR in Shutdown Cooling Mode:

a. What three (3) conditions (including setpoints) will result in an automatic closure of the 48 valve (S/D cooling suction inbd. isolation valve)? (1.5)

b. An operator attempts to open the 24 & 35 valves (RHR pump torus suction) from the following locations:

1. Control Room

2. Backup Control Panel

IN BOTH CASES EXPLAIN why the valves WILL or WILL NOT open. (1.5)

6.3 a. GOI 100-3 directs that when LPRM maintenance is of such a magnitude that any one APRM is made inoperable, the RPS will be placed in the SRM non-coincidence scram mode. EXPLAIN what is meant by "SRM non-coincidence scram mode". Your discussion should include two (2) significant consequences of entering that mode of operation. (1.5)

b. TRUE or FALSE

1. The reactor period indication is no longer valid after the SRM detectors are withdrawn to their lower stop. (.5)

2. An SRM short period of <30 seconds produces a rod withdrawal block. (.5)

6.4 a. Place the following components in proper flow path order. (1.25)

- off gas condenser
- gas reheater
- holdup pipe
- catalytic recombiner
- preheater
- charcoal adsorbers

b. The charcoal adsorber bypass valve (113B) is open during periods of low activity operation. Under what conditions will it automatically close and the adsorber inlet valve (113A) open? (.75)

6.5 During reactor operation at 98% power a loss of 250-VDC Reactor MOV board A occurs. List five (5) different systems and/or MAJOR components that will be affected prior to transferring power, and describe one way in which each one is affected.

Example: If 250 - VDC Rx MOV C had been lost - RCIC - Loss of power to condensate pump. (2.5)

6.6 a. Assuming all APRM channels are initially operable, what effect (if any) will manually bypassing APRM channel B have on BOTH RBM channels? Explain why. (1.0)

b. A control rod surrounded by three (3) strings of LPRM's is selected for movement. How many of the LPRM's could be bypassed and/or failed without giving a RBM trip? Explain. (0.5)

6.7 With regard to the High Pressure Coolant Injection (HPCI) System:

a. WHAT would be of immediate concern if the HPCI minimum flow control valve FAILED TO SHUT following a HPCI turbine trip? (1.0)

b. With the system in "standby," HOW is the pump discharge piping maintained full of water? (1.0)

c. DESCRIBE the response of the HPCI system IF the HPCI pump discharge flow element output signal to the HPCI flow controller failed to a zero output, following a valid automatic HPCI initiation. (1.0)

7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL

7.1 OI-68 states that Unit 2's recirculation pumps may experience some speed induced vibration when passing through the critical speed zone of 1100 to 1200 rpm.

a. What precautions are taken on Unit 2 when passing through the critical speed zones to minimize vibration problems? (.75)

b. Why are these precautions necessary on Unit 2 but not on Units 1 and 3? (.75)

7.2 With regard to OI 47 (Turbine Generator):

a. During turbine startup, EXPLAIN WHY you are cautioned not to operate below 800 rpm for greater than five (5) minutes. (1.0)

b. The procedure states "critical speeds of the unit must be known by the operating personnel." WHAT are these speeds for Unit 2 and WHY is operation at these speeds undesirable? (1.0)

c. After synchronizing, the voltage regulator transfer volt meter is reading -5 volts. What is actually measured and HOW is it nulled? (1.0)

7.3 In reference to the Cold Startup Section of GOI-100-1

a. During the heatup WHAT THREE (3) parameters are required to be recorded every 15 minutes? (1.0)

b. List 3 of 4 action steps required prior to transferring the mode switch to RUN. Include required and/or expected parameter values in your list. (1.5)

NOTE: ACTION STEPS MAY HAVE MULTIPLE ACTION AND/OR CHECKS

7.4 In response to GOI 100-12, Normal Shutdown from Power:

a. If the RSCS is found inoperable at 15% reactor power during the shutdown, WHAT must be done? (0.5)

b. With the unit in cold shutdown and the reactor pressure vessel (RPV) head torqued, WHAT are TWO indications of reactor vessel water stratification, OTHER THAN indication of pressure on the RPV? (2.0)

c. WHAT is the moderator temperature range which must be maintained during cold shutdown? (0.5)

7.5 Regarding EOI-36, Loss of Coolant Accident Inside Drywell:

a. WHAT TWO (2) criteria are to be used to determine the difference between excessive primary coolant leakage and a loss of coolant accident (LOCA)? (1.0)

b. When attempting to locate and isolate the break, the procedure cautions the operator to ensure sufficient reactor coolant inventory to keep the core covered before isolating a line break. WHY is this caution necessary? (1.0)

7.6 According to EOI-41, Methods of Water Makeup to the Reactor, WHAT are FOUR (4) plant systems which can be used to supply HIGH pressure water makeup to the reactor with NO NUCLEAR STEAM AVAILABLE? (2.0)

7.7 Answer the following regarding the operator actions of GOI-100-11 for a REACTOR SCRAM with MSIV's closed:

a. WHY is it preferable to manually operate the MSRVs to control reactor pressure rather than allowing them to cycle on their own? (1.0)

b. At WHAT torus temperature is it necessary to place an RHR loop in torus cooling? (0.5)

8. ADMINISTRATIVE PROCEDURES, CONDITIONS AND LIMITATIONS

8.1 Answer the following relative to the applicable Section Instruction Letter (SIL):

a. According to SIL-45, "Main Steam Isolation Valve Operation," WHAT restriction is placed on MSIV operation? (0.5)

b. According to SIL-73, "Medical Treatment, Rescue, and Evaluation," WHAT are four responsibilities of the Fire Brigade Leader? (2.0)

8.2 According to the TECHNICAL SPECIFICATIONS for the CONTROL ROD SYSTEM:

a. WHEN must the RWM be operable? (1.0)

b. Define a limiting control rod pattern? (1.0)

c. WHAT TWO restrictions are placed on rod withdrawal when a limiting control rod pattern exists? (1.0)

8.3 STATE the POWER TRANSIENT fuel cladding integrity safety limit. (1.5)

8.4 A Surveillance Instruction (SI) CANNOT be completed on the day it has been scheduled. The Assistant Shift Engineer notes this in his daily journal and completes a Form BF-49, "Data Cover Sheet for Surveillance Instructions Not Performed."

According to SIL-29, "Surveillance Instructions," DESCRIBE the method used by the control room operating staff to ensure subsequent shifts are alerted to perform this particular SI at the earliest opportunity. (2.0)

8.5 Regarding Standard Practice BF 12.17, Administrative Controls for Plant Operation:

a. WHAT must be done prior to resetting a primary containment isolation? (1.0)

b. Individual system panel checklists for core cooling systems must be reviewed by WHOM? (2 required) (1.0)

c. HOW is the review in Part (b) documented? (0.25)

d. The shift engineer of each shift will review ____________ logs and temporary ____________ from the preceding shifts to ensure that no out-of-normal equipment configurations have resulted from application of these procedures. FILL IN THE BLANKS. (1.0)

8.6 According to SIL-56, Abnormal Voltage Conditions on Safety-Related Auxiliary Power Systems:

WHAT are the Minimum Voltage Limits on the 4-kV AND 480-V shutdown boards? (1.0)

8.7 Answer each of the following with regard to RECIRCULATION PUMP STARTING LIMITATIONS. Be specific.

a. The pump in an idle recirculation loop shall not be started unless the temperature of the coolant within the idle and operating recirc loops are within 50 F of each other. WHY? (1.0)

b. WHAT restriction is placed on the operating recirc pump speed when restarting an idle recirc pump? WHY? (1.0)

8.8 Following a TIP Trace, the "C" TIP Ball valve did not auto close. The TIP machine requires extra jogging to close this valve due to a sticking "in-shield" limit switch. The Ball valve is presently closed. Is Primary Containment Integrity satisfactory? If so, WHY? If not, WHY NOT and HOW can it be made satisfactory? (1.0)

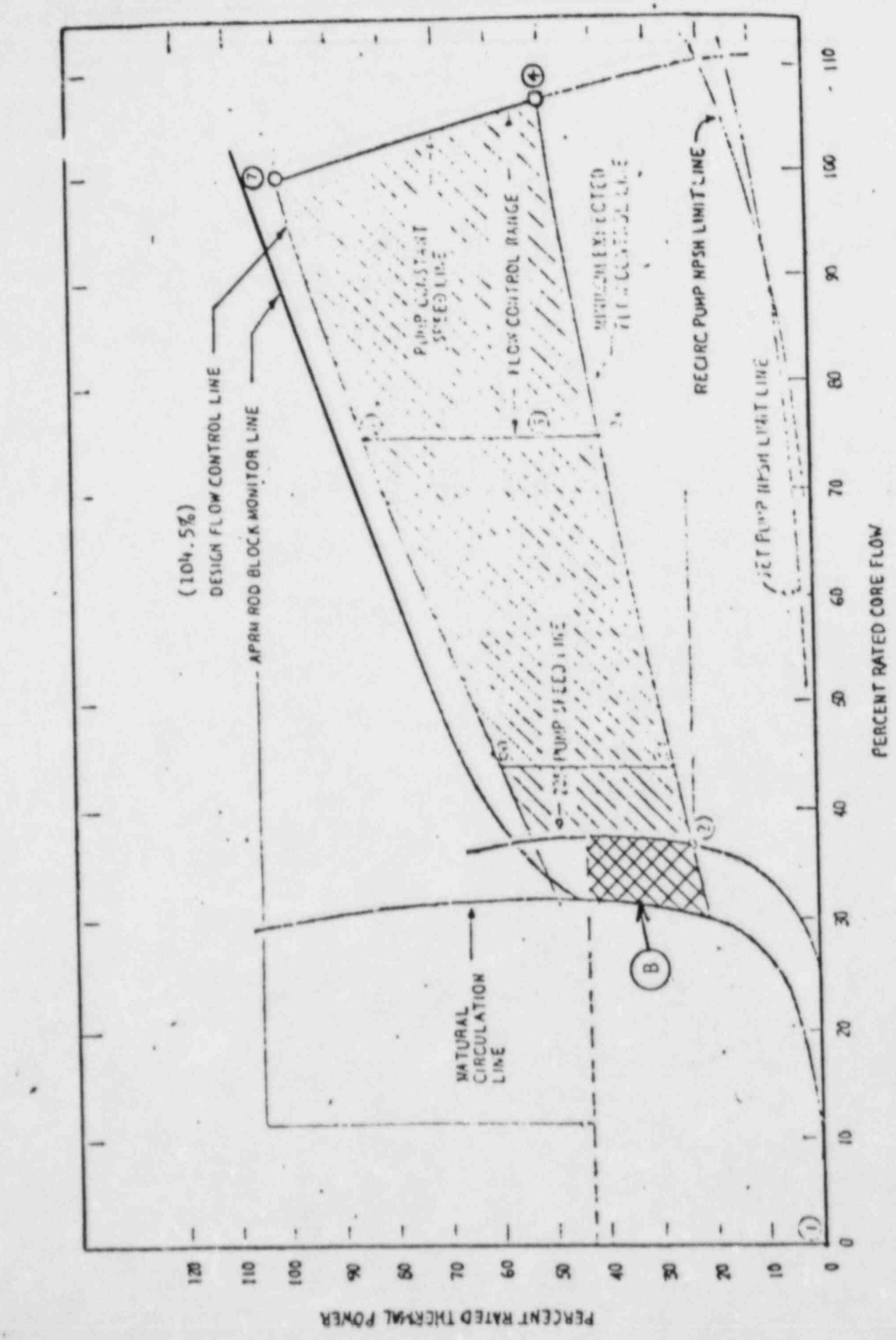
QUESTION 5.2

CORE PARAMETER->	^ Moderator Temp.	^ Core Voiding	^ Rod Density	^ Fuel Temp.	^ Core Age
COEFFICIENT					
Void Coefficient	->	(a)	(b)	(c)	(d)
Moderator Temp. Coefficient	(e)	->	^	^	(f)
Fuel Temperature Coefficient	^	(g)	->	(h)	(i)

FOR USE IN ANSWERING QUESTION NUMBER 6.1

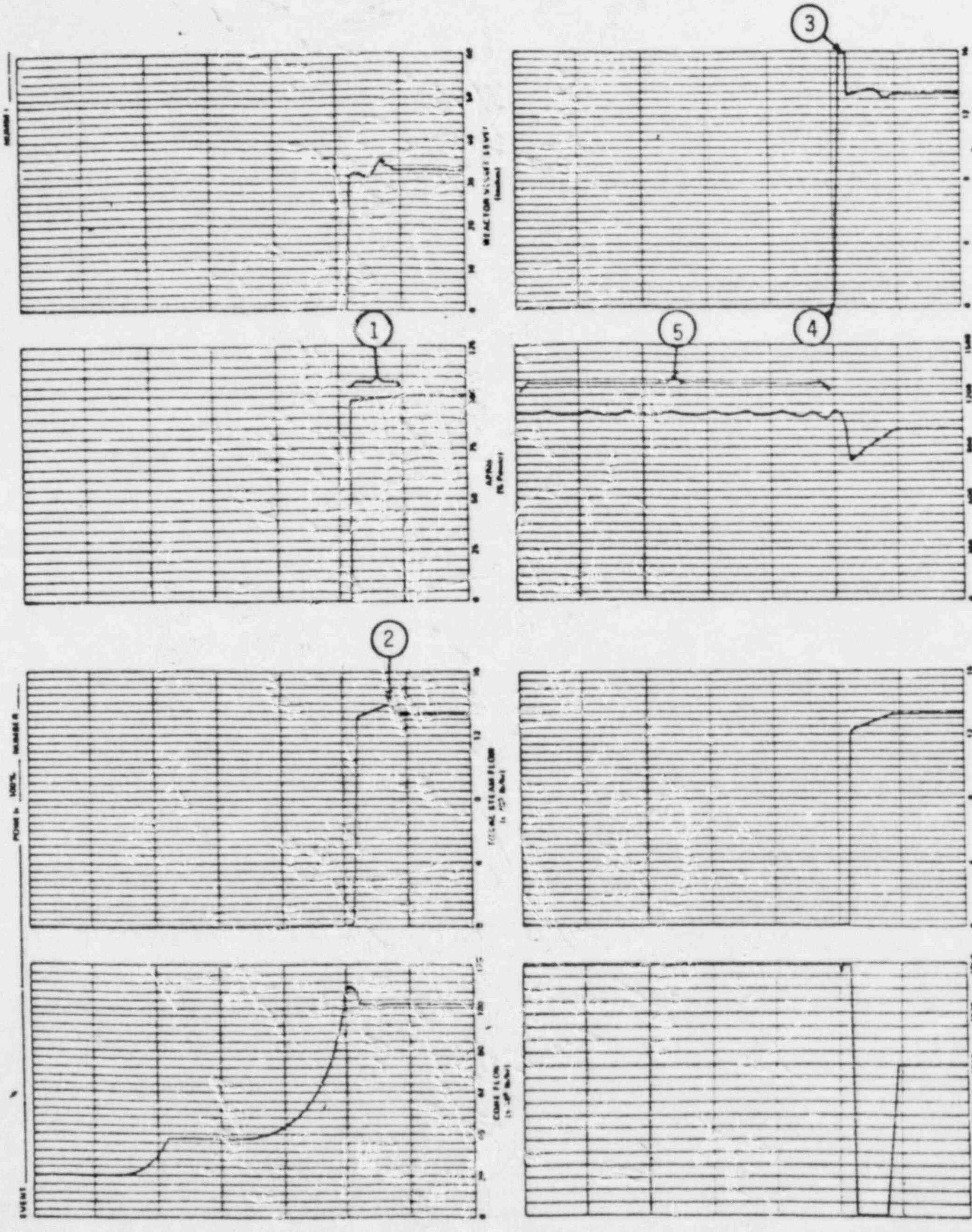
1. LT-3-206 INDICATES ______________ (↑, →, ↓)
2. FWCS RESPONSE ______________
3. RFP ACTION(S) ______________
4. ACTUAL VESSEL LEVEL ______________ (↑, →, ↓)
5. FUNCTIONING OF RPS INCLUDING REASON(S) AND EFFECT(S) ______________
6. INDICATION: LITS 3-46A ______________
   LITS 3-52 ______________ (↑, →, ↓)
   LITS 3-46B ______________

FIGURE 5.4



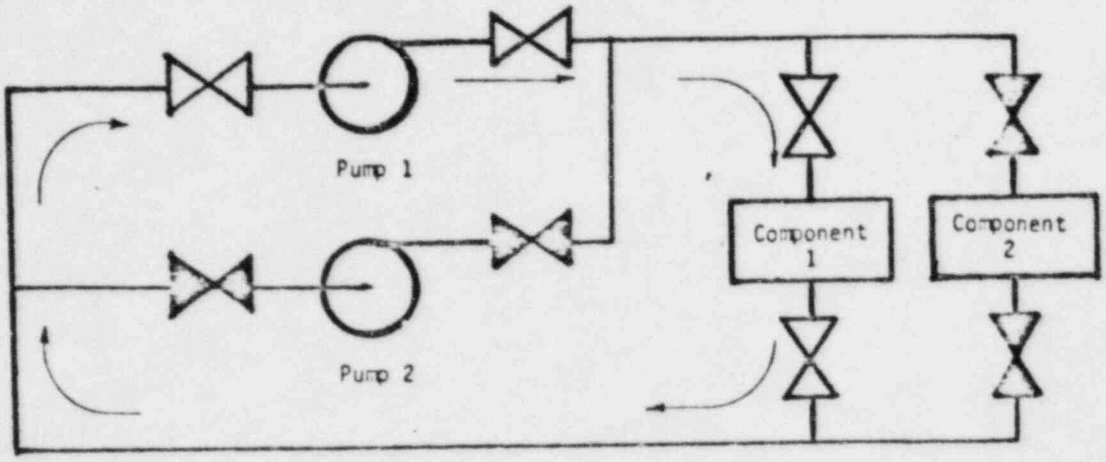
OPERATING MAP

FIGURE 5.5

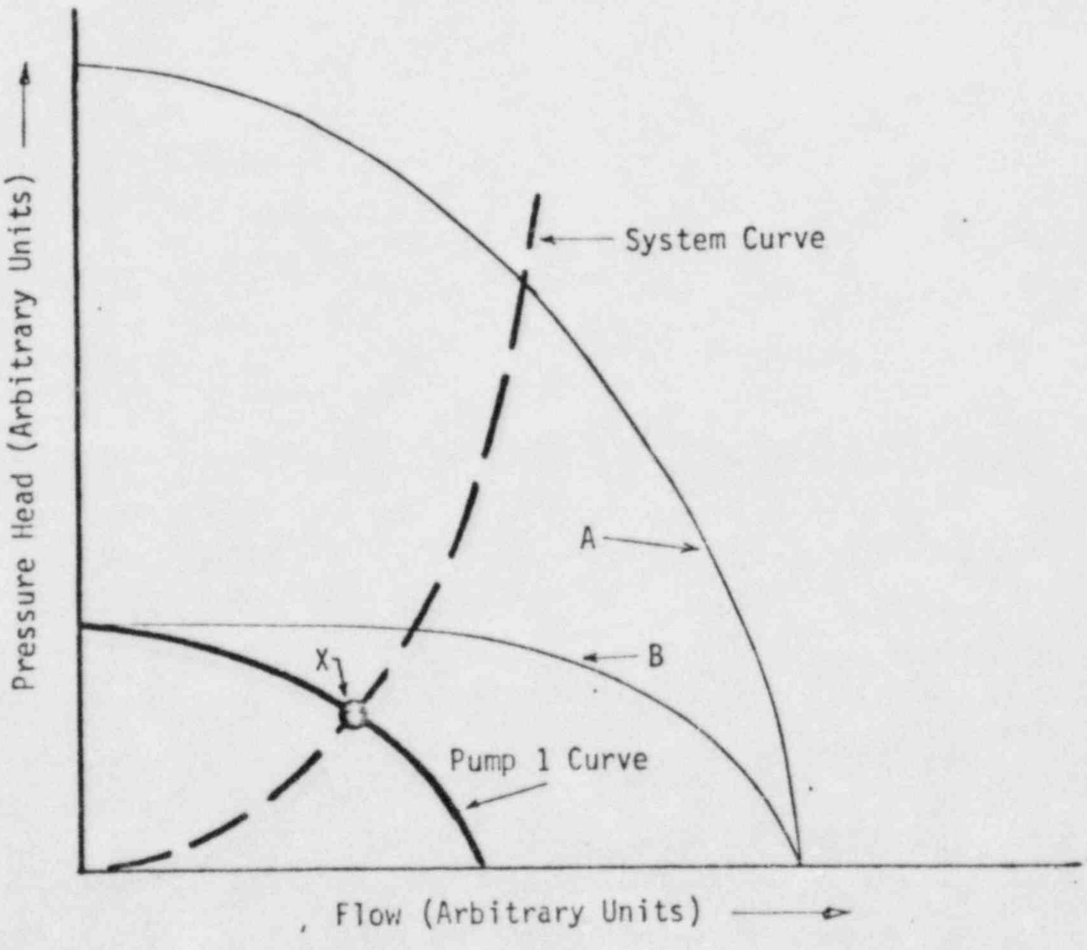


EHC PRESSURE SETPOINT DECREASE

FIGURE 5.7



SYSTEM



SYSTEM HEAD VS. FLOW PLOT

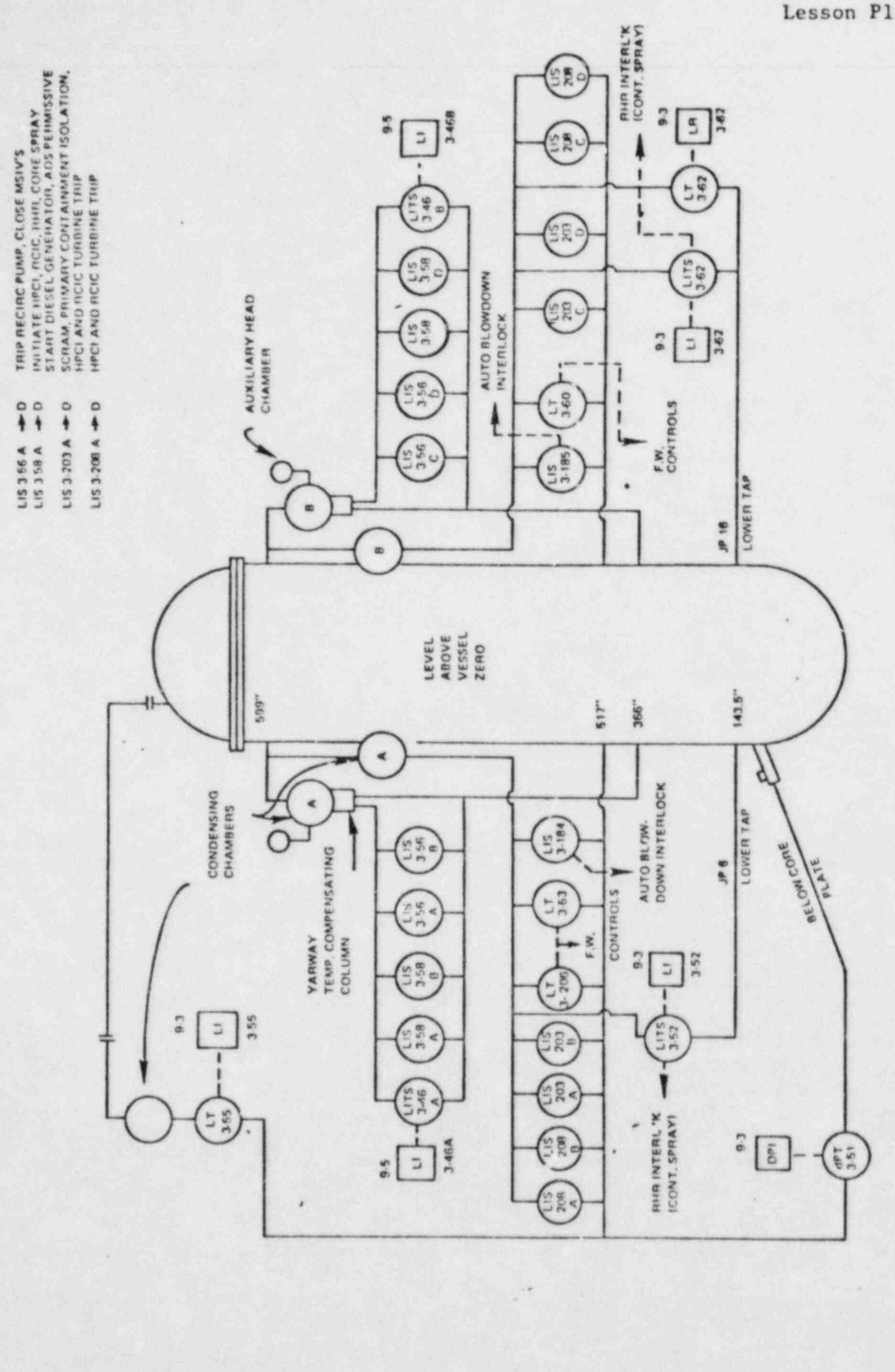


FIGURE 3.4 Reactor Vessel Level Instrumentation

EQUATION SHEET

$f = ma$

$v = s/t$

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

$w = mg$

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

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = e/t$

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

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

$W = v\ \Delta P$

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

$\Delta E = 931\ \Delta m$

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

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

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

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

$Pwr = W_f \Delta h$

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

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

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

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

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

$SUR = 26.06/T$

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

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

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

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

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

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

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

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

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

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

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

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

5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS

ANSWERS

5.1 Reactivity (dK/K) = -1.8% (0.25)

Initial Keff = $\frac{1}{1 - dK/K} = \frac{1}{1 + 0.018} = .982$ (0.5)

20 = Final Count Rate (CR-f)/Initial Count Rate (CR-i), and

$CR = \frac{S}{1 - K\text{-}eff}$ (0.5)

Substituting,

$20 = \frac{CR\text{-}f}{CR\text{-}i} = \frac{S/1 - K[f]}{S/1 - K[i]} = \frac{1 - K[i]}{1 - K[f]} = \frac{(1 - 0.982)}{(1 - K[f])}$ (0.5)

where: S -- Source Strength
K[i] -- Initial K-eff
K[f] -- Final K-eff

Therefore,

20(1 - K[f]) = 0.018, and K[f] = 1 - (0.018/20) = .999 (0.25)

Reference

BFNP Hot Lic. LP #1, PP 22 & 23; OBJ #17
BFNP Requal. LP, "Rx Theory" pp. 20 & 21; OBJ 12, 13

5.2 a. Increases, (Page 40 and Figure 47)
b. Increases, (Page 40 and Figure 48)
c. Increases, (Page 40)
d. ~~Increases~~, (Page 40 and Figure 48) Decreases - Incorrect interpretation of Ref.; Decrease is correct answer.
e. Increases, (Page 31 and Figure 30)
f. Decreases, (Page 32 and Figure 32)
g. Increases, (Page 39 and Figure 45)
h. Decreases, (Page 39 and Figure 44)
i. Increases, (Page 39 and Figure 46)
(0.2 each) (1.8)

Reference:

BFNP Hot License Lesson Plan 1, pages per answer.
BFNP Requal Lesson Plan, "Reactor Theory," Figure 11, p. 47
(Objective 18).

5.3 a. Peripheral rod worth will increase [0.3] because the highest xenon concentration will be in the center of the core [0.3] where the highest flux existed previously [0.3]. This will suppress the flux in the center of the core [0.3] and increase the flux in the area of the peripheral rods, (thereby increasing their worth). [0.3] (1.5)

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

Reference:

BFNP Hot License Lesson Plan 1, p. 46, and BF GOI 100-1, p. 9

5.4 a. As reactor power is decreased from Point 7 to Point 4 core voiding decreases [0.5] resulting in a <u>decrease in two-phase flow resistance in the core</u> [0.5]. (1.0)

b. Critical Power, CPR, or MCPR. Underline acceptable for full credit (0.5)

Reference:

BFNP Hot License Lesson Plans No. 8, p. 18, & No. 22, p. 27, and BF GOI 100-1, p. 22.

5.5 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)

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:

BFNP Transient Hxy-12

5.6 a. Minimize fuel damage during a DBA LOCA by limiting the peak clad temperature (to <2200 F) -OR- limiting bundle stored energy. (.75)

b. 2 and 3. (0.5)

c. MAPRAT = MAPLHGR/LIMLHGR -or- = MAPLHGR/HAPLHGR limit -or- = (MAPLHGR) actual/(MAPLHGR) LCO max (.75)

Reference:

BFNP Thermodynamics, p. 10.2-7, BF GOI 100-5, p. 44D, and
BF Unit 1 Tech Specs, pp. 168 & 168A.

BFNP Requal Lesson Plans, "Thermal Hydraulics," pp. 32 & 33, and
"Thermal Limits," p. 5, 6; OBJ 3, 6

5.7 a. System operating point. (0.5)

b. Curve B. (0.5)

c. Right. (0.5)

Reference:

BFNP Thermodynamics, pp. 6.1-6, 6.2-4, 6.4-6 & 7.

BFNP Requal Lesson Plan, "Pump Characteristics," Figures 4, 5,
& 6. OBJ 5, 6; Sample Problem (1)

5.8 a. The sudden flow increase causes the clad surface temperature to decrease [0.33] due to more efficient convection heat transfer [0.33], decreasing the amount of nucleate boiling [0.33]. (1.0)

b. Decrease [0.2]. Due to increased clad surface temperature [0.5]. (0.7)

Reference:

BFNP Thermodynamics, pp. 3.4-3 & 3.4-4.

6. PLANT SYSTEMS DESIGN, CONTROL AND INSTRUMENTATION

ANSWERS

6.1
1. increasing level (.5)
2. FWCS will send sgnal to runback the RFP's (.5)
3. runback and then trip on high level (from LT-3-206 & LT-3-63) (.5)
4. actual level is decreasing (.5)
5. RPS scram on low level from LIS-203 C&D or High level → TT → Scram (.5)
6. LITS-3-46A decreases (.167)
   LITS-3-52 stays same (.167)
   LITS-3-46B decreases (.167)

Reference: BFNP FWCS & Inst. LP's
I & E Notice 81-25

6.2 a. MOV 48 will auto close on the following:

1. Rx Pressure = or > 100 psig [0.5]
2. DW Pressure = or > 2.45 psig [0.5]
3. Rx Water Level = or < 11 inches [0.5] (1.5)

b.
1. No they will not open [0.25] the valves are interlocked closed when the 25 & 26 valves are open. [0.5]
2. Yes they will open, [0.25] Automatic logic & Equipment interlocks are not operable when equipment is controlled from the backup control panel. [0.5] (or MOV Board) (1.5)

OR NO [.25] these valves do not have control switches on the BCP [.5]

Reference: BRF RHR L.P. pg. 18 and Control Room Abandonment L.P.
BRF Requal RHR L.P. Objective 7, pgs. 11, 12 & 14
Verified physically at the Facility

6.3 a.
1. Any single SRM, IRM, or APRM trip will result in a trip of both RPS channels, (i.e., a full scram)
2. The SRM high high trip (5 E + 5 CPS) becomes functional. (.75 ea/1.5)

b.
1. False (.5)
2. False (.5)

Reference: BFNP RPS LP, P21, Objective 4
BFNP SRM LP, P24, Objective 2
GOI 100-3, P10

6.4 a.
- preheater
- catalytic recombiner
- off gas condenser
- holdup pipe
- gas reheater
- charcoal absorbers (1.25)

b. If either of the two post treatment rad monitors sees a high, (high high, or high high high)alarm. (.75)

Reference: BFNP Off-Gas System LP, Objectives 2,4
BF OI-66

6.5 (1) HPCI - Loss of power to valves, pumps and Division II logic
(2) CS - System II will not auto initiate if needed
(3) RHR - System II will not auto initiate if needed
(4) RCIC - Loss of Division II logic
(5) MSRV's - Loss of operability & ind. on some valves
(6) Recirc - "a" MG loss of emerg. oil pump, loss of speed control and resulting lockup of its scoop tube
(7) RPS - Loss of backup scram capability
(8) MSIV's - The outboard MSIV DC solenoids will deenergize.
PCIS - loss of power to DC isol valves / will not shut

(5 of 8 required at 0.5 each) (2.5)

ADS - two valves lose normal power & auto swap to Rmov-B

Reference: BRF OI 57 pgs. 75-78 & BRF DC Elec. Dist. LP
BRF Requal ECCS LPs, Obj. 3,4
BRF ADS Requal LP, p.7
BRF PCIS LP

6.6 a. RBM A is not affected [0.5] RBM channel B uses APRM B as its normal reference input but when the APRM is bypassed the alternate APRM D is automatically placed in the circuit. Hence, no adverse effect on either RBM channel [0.5] (1.0)

b. Three (3) [0.25] there are 12 LPRM's, 6 in RBM A & 6 in B. Must have at least 50% operable/channel; hence 3/CHANNEL - 6 total [0.25] (0.5)

Reference: BRF RBM L.P.
BRF Requal RBM L.P. Objective 2, 3 & 5 pg. 10, 15-16 & 21

6.7 a. Draining the CST to the suppression pool. (1.0)

b. The CST being sufficiently elevated above the HPCI gravity fills the discharge piping. (1.0)

c. HPCI will start and operate, but it will overspeed due to low flow feedback. [0.7] After it overpeeds, it will then reset itself and the sequence of events will be repeated [0.3] (1.0)

OR - HPCI will and operate at maximum speed [1.0]

Reference:

BFNP Hot License Lesson Plan 42, pg. 11, 17 - 19, & Figure 1.
BFNP OI-73
BFNP Requal HPCI L.P. Objective 1 & 2, pg. 5, 8
Singer Simulator Malfunction Description

7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL

ANSWERS

7.1 a. (Place the loop flow controllers in manual and) maintain a 6-8% speed differential until pump speeds are past the critical speed zones. (.75)

b. Unit 2 recirc. pipe inspections showed indications of fatigue induced cracks on some (sweepolet-to-manifold) welds. (.75)

Reference: BFNP OI-68, P3, 23, ATTACHMENT B

7.2 a. Operation at low speed may result in large rotor bows being generated by rubbing with no indication of high vibration from the turbine supervisory instruments. (1.0)

OR - Low steam flow to last stages during s/u ⇒ significant heating of last stage blading and inner casing [not per OI but ok for .7 partial credit]

900-1100
1365-1415

b. ~~CAF on Unit 2 critical speeds~~. [0.5] Operation at critical speed would be to operate at the point of maximum vibration. [0.5] (1.0)

c. The transfer voltmeter reads the difference between the Auto (90P) and Manual voltage regulator outputs. [0.5] By adjusting the manual voltage regulator (70P) the meter is nulled. [0.5] (1.0)

Reference: BRF Turbine Generator OI 47 ; Letter from BFNP (attached)
BRF Main Turbine LP, P.14 ; BRF Main Gen & Aux LP

7.3 a. Parameters

1. Steam dome pressure [0.33]
2. Reactor bottom head drain temperature [0.33]
3. Recirc loop A & B temperature [0.33] (1.0)
4. Moderator Temperature
5. Vessel shell Temp adjacent to Flange

b. Checks prior to transfer to RUN:

1. APRMs >5% & <12%
2. Inboard and outboard MSIVs open
3. Condenser vacuum >24" Hg vacuum, and condenser A, B, & C low vacuum annunciators cleared
4. Reactor pressure >850 PSIG and low reactor pressure annunciators cleared

(3 of 4 required at 0.5 each) (1.5)

Reference: BRF GOI-100-1, & BRF Q&A 5-4 & 5-127
BRF Tech Spec. 1SI - 3.U.A.1 / 4.U.A.1

7.4 a. Manually scram the reactor. (0.5)

b. ° Temperature differential of >50 F between bottom vessel head and "A" FW sparger when moderate is >150 F.

° Temperature differential of >75 F between bottom vessel and "A" FW sparger when moderate is <150 F.

° FW sparger temperature of >200 F on either "A" or "B" FW sparger when recirc. pumps and shutdown cooling are OOS.

(2 of 3 at 1.0 each) (2.0)

c. Between 180°F and 200°F. (0.5)

References: BF GOI 100-12, pp. 1, 7, & 14
BFNP Requal Lesson Plan, "GOI-100-12," (Objectives 1, 3 & 4).

7.5 a. A break large enough to result in a reactor scram on High Drywell Pressure [0.5] or Low Reactor Water Level [0.5]. (1.0)

b. Isolation of line breaks may cause reactor pressure to increase rapidly [0.5] resulting in level shrink uncovering the core. [0.5] (1.0)

Reference: BFNP Requal Lesson Plan, "EOI-36" (Objective 1 & 4)

7.6 ° CRD
° SLC
° RCIC
° HPCI (0.5 each) (2.0)

Reference: BF EOI-41, pp. 1 & 2
BFNP Requal Lesson Plan, "EOI-41" (Objective 1).

7.7 a. To sequence the opening order per the MSRV position chart in order to minimize torus hot spots. (1.0)

b. >95 F. for Unit 1 ; >93°F for Unit 2 & 3 (either temp & unit) (0.5)

Reference: BF GOI-100-11, pp. 5-7.
BF OI-74, P.8
BF OI-64., P.21

8. AMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS

ANSWERS

8.1 a. Do not operate when no steam is flowing through the valves. (0.5)

b.
- Proceed to the scene of the emergency.
- Appraise situation.
- Notify the shift engineer of the condition.
- Verify at lease two auxiliary operators are at the emergency equipment storage room for level II support (except for cooling tower auxiliary).
- Take whatever action necessary to implement the plan of corrective actions.
- If auxiliary operators are not available, dispatch an AUO to the emergency equipment storage room for Level II support. [4 of 6 at 0.5 each] (2.0)

Reference:

BFNP SILs 45 & 73.

BFNP Requal Lesson Plan, "Discussion of Selected Section Instruction Letters," (Objectives 1 & 3).

8.2 a. Whenever the reactor is in the startup or run mode below 20% rated thermal power. (1.0)

b. A pattern which results in the core being on a thermal hydraulic limit (i.e., MCPR or LHGR). (1.0)

c. Both RBM channels must be operable or control rod withdrawal shall be blocked. (1.0)

Reference:

BF TS, pp. 123, 124, & 131.

BFNP Requal Lesson Plan, "Rod Block Monitor System," pp. 23 & 24 (Objective 6). (Applicable to Parts (b) and (c).)

8.3 Each required scram shall be initiated by its expected scram signal [0.5]. The thermal power safety limit shall be assumed to be exceeded [0.5] when scram is accomplished by other than the expected scram signal [0.5]. (1.50)

Reference:

BFNP Unit 1 TS, p. 11.

BFNP Requal Lesson Plan, "Unit 1 Tech Specs," pp. 3 & 4 (Objective 2.a.2).

8.4 The SI is to be flagged by a marker [~~0.66~~ 1.0] in the rolodex card file located on the unit operator's desk [~~0.66~~ 1.0]. ~~Each oncoming shift is required to survey the card file for any SIs that have been flagged by a marker [0.66].~~ Not specifically asked for in the question. (2.00)

Reference:

BFNP SIL-29.

8.5 a. An evaluation shall be made to ensure that inadvertent transfer of significant amounts of containment fluids will not occur upon resetting the isolation. (1.0)

b. The Unit Operator [0.5] and Assistant Shift Engineer [0.5]. (1.0)

c. On the shift turnover cover sheet. (0.25)

d. Clearance, alterations. (1.0)

Reference:

BFSP 12, 17, pp. 2 & 3.

BFNP Requal Lesson Plan, "Selected Browns Ferry Standard Practice" (Objectives 6 & 7 for Parts [a] and [d])

8.6
- 3,950 Volts (4-kV)
- 440 Volts (480-V) (0.5 each) (1.0)

Reference:

BF SIL-56.

BFNP Requal Lesson Plan, "Core Spray System," pp. 9 & 10 (Objective 6).

8.7 a. Prevents a large uncontrolled thermal stress on the pump casing OR limits cold water reactivity addition OR limits thermal stress at the reactor vessel nozzles and and bottom head region. (1.0)

b. Operating pump speed must be reduced to less than 50% rated speed [0.5] to prevent excessive jet pump riser vibration [0.5]. (1.0)

Reference:

BFNP Hot License Lesson Plan No. 7, p. 18, BF OI-68, pp. 2-4, and TS, p. 216.

BF OI-68 p. 3

8.8 No, by definition of primary containment integrity the Ball valve must be <u>deactivated</u> in it's isolated position. (1.0)

Reference:

BF Hot Lic. LP 16
BF TS p. 4
BF Requal LP-PCIS OBJ. #3

MASTER COPY

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

Reviewed by: Champion
Crosto
J.D. Johnson
Dexter
Glover] this week

FACILITY: _BRF_______________

REACTOR TYPE: _BWR_______________

DATE ADMINISTERED: _84/03/23___________

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	25.00	________	________	1.	PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
25.50	25.50	________	________	2.	PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
24.50	24.50	________	________	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 (2.00)

Three (3) minutes following a reactor scram from high power, indicated reactor power is 75 on range 4 and decreasing.

a. WHAT will INDICATED power be one (1) minute later? (Show calculations) (1.0)

b. Explain WHY power decreased at this rate. (1.0)

QUESTION 1.02 (2.00)

A variable speed centrifugal pump is operating at 3000 RPM, and is pumping 30,000 gpm, with a discharge head of 200 psig. The speed is now changed and the pump has a discharge pressure of 50 psig. (assume ideal conditions i.e. no head loss.) (SHOW CALCULATIONS AND STATE ALL ASSUMPTIONS MADE) (2.0)

a. WHAT is the new pump speed?
b. WHAT is the new flow rate?
c. By WHAT ratio has the power requirement for the pump changed?

QUESTION 1.03 (2.00)

Your reactor has just scrammed from extended full power operation. Ten (10) hours later cooldown is complete, and the SDM is determined to be 1% dk/k, since all rods did not insert. EXPLAIN the changes to the SDM AND any possible adverse consequences for the next 20 hours. (2.0)

QUESTION 1.04 (3.00)

Concerning THERMAL LIMITS:

a. Since MCPR is not a directly measurable parameter, WHAT are THREE (3) measurable core parameters needed by the process computer to calculate MCPR? (1.0)

b. With regard to MAPRAT. (2.0)
   1. WHAT is the RELATIONSHIP between MAPRAT & MAPLHGR?
   2. The process computer prints out a MAPRAT of 1.05. Is this acceptable?
   3. WHAT physical consequence could occur if the MAPRAT limit is exceeded?

QUESTION 1.05 (2.50)

For each of the following events, STATE WHICH coefficient of reactivity would act FIRST to change reactivity.

a. Control rod drop at power (0.5)
b. SRV opening at power (0.5)
c. Loss of shutdown cooling when shutdown (0.5)
d. One recirc pump trips while at 50% power (0.5)
e. Loss of one feedwater heater (extraction steam isolated) (0.5)

QUESTION 1.06 (2.50)

T-S diagrams of real plant cycles show a small amount of "Condensate Depression"(Subcooling) in the condenser.

a. HOW & WHY would CYCLE EFFICIENCY be affected if subcooling is DECREASED? (2.0)

b. Give one (1) example HOW an operator could INCREASE the amount of subcooling. (0.5)

QUESTION 1.07 (2.00)

a. WHAT is decay heat and HOW is it produced? (1.0)

b. Does this power INDICATE on the Source Range nuclear instrumentation following a scram? WHY or WHY NOT? (1.0)

QUESTION 1.08 (3.00)

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? (1.5)
c. HOW do delayed neutrons contribute to the control capability of a commercial reactor? (1.0)

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 on a power increase with "NO CHANGE IN RECIRC FLOW" if this feature were eliminated. (2.5)

QUESTION 1.10 (1.50)

WHY are installed neutron sources no longer required at BFNP during reactor startups? Your answer should include THREE sources o` source neutrons currently present at BFNP. (1.5)

QUESTION 1.11 (2.00)

HOW and WHY does control rod worth vary for the following changes: (2.0)

a. As moderator temperature increases.

b. As an adjacent rod is inserted.

c. As the rod's axial position inside the core is changed.

QUESTION 2.01 (2.50)

a. The plant is operating at high power when a technician incorrectly lifts a lead resulting in the loss of open indication of RWCU inlet isolation valve (FCV-69-1). WHAT 2 automatic actions, direct or indirect, will result. (1.0)

b. When using the RWCU system for "Hot Blowdown" the blowdown flow rate must be restricted to avoid exceeding temperature limitations in two locations. WHAT are these locations and WHY does hot blowdown cause you to approach their temperature limits? (1.5)

QUESTION 2.02 (3.00)

With regard to the RHRSW and EECW systems:

a. List the RHRSW pumps that are normally assigned to EECW and which 4160-V shutdown board powers each pump. (1.0)

b. WHAT 2 automatic actions take place if the RCW pressure at the inlet to the RBCCW HX's drops to <15 psig? (1.0)

c. RHRSW cools the RHR HX's. WHICH fluid should be at the greater pressure? WHY? (1.0)

QUESTION 2.03 (3.00)

During reactor operation at 98% power a loss of the 250-VDC Reactor MOV board A occurs. LIST 6 systems and/or MAJOR components that will be affected prior to transferring power, and DESCRIBE one way in which each one is affected.
EXAMPLE: If 250 - VDC Rx MOV C had been lost - RCIC - Loss of power to condensate pump. (3.0)

QUESTION 2.04 (3.00)

a. WHEN a scram signal occurs at power, DESCRIBE IN DETAIL how the Control Rod Drive and its associated Hydraulic Control Unit function to insert the control rod. INCLUDE which components open, close, energize, deenergize, and the motive forces for the entire rod travel as a MINIMUM in your answer. (2.0)
b. Following the above scram WHEN will the accumulators fully recharge? WHY? (1.0)

QUESTION 2.05 (2.50)

Explain the difference between an ATWS recirc pump trip and a Recirculation Pump Trip (RPT). Include in your explanation initiation signal(s), all bypasses, and components actuated. (2.5)

QUESTION 2.06 (3.50)

With regard to the High Pressure Coolant Injection (HPCI) System:

a. WHICH HPCI isolation signal does NOT seal in (setpoint NOT required)? (0.5)

b. WHAT would be of immediate concern if the HPCI minimum flow control valve FAILED TO SHUT following a HPCI turbine trip? (1.0)

c. With the system in "standby," HOW is the pump discharge piping maintained full of water? (1.0)

d. DESCRIBE the response of the HPCI system IF the HPCI pump discharge flow element output signal to the HPCI flow controller failed to a zero output, following a valid automatic HPCI initiation. (1.0)

QUESTION 2.07 (2.00)

STATE THE BASES for each of the following conditions required for ADS (Automatic Depressurization System) initiation.

a. The -114.5" water level signal. (1.0)

b. The 120 second time delay (i.e., WHY 120 seconds?). (1.0)

QUESTION 2.08 (3.00)

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. (1.5)

QUESTION 2.09 (3.00)

Two Core Spray (CS) System alarms (and their associated setpoints) are listed below. For each, provide the ABNORMAL CONDITION being indicated by receipt of the alarm in the Control Room AND briefly EXPLAIN HOW the abnormal condition could prevent the CS System from accomplishing its design function when required.

a. Core Spray System Header to Core Plate HIGH dP (with a setpoint of 2 PSID decreasing). (1.5)

b. High Pressure Valve Leakage Test (with a setpoint of 450 PSIG increasing). (1.5)

QUESTION 3.01 (3.00)

a. The RSCS enforces Group Notch Control from ______% rod density to _____% reactor power as sensed by ___________________________. (1.0)

Candidates were given IC of all rods at same height, then one rod withdrawn one notch.

b. WHAT 2 ROD BLOCKS are used to enforce control rod sequencing when in GNC? (1.0)

c. WHEN in GNC with the Sequence Mode Selector (SMS) switch in normal, HOW does the RSCS determine control rod position? (1.0)

QUESTION 3.02 (3.50)

Operating with RHR in the Shutdown Cooling mode:

a. What 3 conditions (including setpoints) will result in an automatic closure of the 48 valve (S/D cooling suction inbd. isolation valve)? (1.5)

b. An operator attempts to open the 24 & 35 valves (RHR pump torus suction) from the following locations:
1. Control Room
2. Backup Control Panel
IN BOTH CASES EXPLAIN why the valves WILL or WILL NOT open. (1.5)

c. If the 24 & 35 valves in part b. were to open while in S/D cooling what would be of immediate concern to the operator? (0.5)

QUESTION 3.03 (2.50)

a. List three (3) different ways in which one or both Rod Block Monitor (RBM) channels is/can be bypassed. (1.0)

b. Assuming all APRM channels are initially operable, what effect (if any) will manually bypassing APRM channel B have on BOTH RBM channels? Explain why. (1.0)

c. A control rod surrounded by three (3) strings of LPRM's is selected for movement. How many of the LPRM's could be bypassed and/or failed without giving a RBM trip? Explain. (0.5)

QUESTION 3.04 (3.00)

With the Unit operating at 75% power, an electrical fault causes the Maximum Combined Flow Setpoint to drop to zero. How will the following parameters RESPOND after the fault and WHY? Consider thier response for ONE MINUTE following the fault. Assume NO OPERATOR ACTION. Attached FIGURE 7, EHC Logic, is provided for reference.

a. Turbine control valve position (1.0)

b. Bypass valve position (1.0)

c. Reactor power & pressure (1.0)

QUESTION 3.05 (3.00)

The plant is operating at 80% power with the Feedwater Control System (FWCS) in 3-element control and Channel "A" reactor level detector selected. Referring to attached FIGURE 6, "Reactor Vessel Level Instrumentation," answer the following with the rupture shown in the figure. Assume NO OPERATOR ACTION after the rupture occurs.

a. HOW will the associated control room indicator respond for the level transmitter with the rupture (FULL SCALE or DOWNSCALE)? (0.5)

b. Explain HOW the FWCS will respond to the change in Channel "A" indicated level due to the rupture [Part (a) above] and HOW the FWCS response will affect ACTUAL vessel level. (1.5)

c. WHAT, if any, automatic actions would you anticipate occurring in the plant as a result of the ruptured transmitter line? (1.0)

QUESTION 3.06 (2.00)

a. WHAT two (2) conditions cause the IRM Hi-Hi Flux scram to be bypassed? (1.0)

b. TRUE or FALSE The IRM detector wrong position interlock will not allow IRM withdrawal prior to placing the mode switch in run. EXPLAIN your choice. (1.0)

QUESTION 3.07 (3.00)

Regarding diesel generator control circuits:

a. A diesel generator has auto-started on degraded voltage on its respective 4160 V S/D board. What three (3) conditions must be satisfied for the control circuit to close the breaker to the S/D board? (1.5)

b. On Panel 9-23 there are two (2) Backfeed Switches. What 2 automatic actions result when these switches are placed in BACKFEED and what manual action is then possible? Be specific. (1.5)

QUESTION 3.08 (2.00)

With regard to the Local Power Range Monitors (LPRM's):

a. True or False The Trip Reset Pushbutton on the meter (LPRM) on the front of the 9-14 Panel, will reset LPRM alarms on the Full Core Display. (0.5)

b. Describe HOW Uranium depletion in an LPRM detector is compensated for over detector lifetime. Be Specific (1.5)

QUESTION 3.09 (2.50)

With regard to the Reactor Recirculation Speed Control System:

a. Briefly EXPLAIN the purpose of the 75% speed limiter. Include parameters and setpoints which place the limiter in effect. (1.5)

b. With the plant operating at 23% power and minimum flow, an operator inadvertently shifts the M/A transfer station for recirc pump "A" from "Manual" to "Auto." Assuming NO further operator action, briefly EXPLAIN WHAT will happen to the speed of "A" recirc pump. Continue your discussion to the final steady state speed. (1.0)

QUESTION 4.01 (2.00)

Unit 3 is operating at 100% power when annunciation of "Auto Blowdown Relief Valves Open", is received and a drop in main generator MWe is noted, indicating an open relief valve. ~~WHAT~~ four (4) action steps would you as the operator have taken prior to torus temperature reaching 115 degrees F?. LIST OF 5 (2.0)

NOTE: ACTION STEPS MAY HAVE MULTIPLE ACTIONS AND/OR CHECKS

QUESTION 4.02 (2.50)

Five (5) indications of a failed jet pump per BF OI-68(Abnormal Operations Section) are listed below. Indicate on your answer sheet whether these indications INCREASE, DECREASE, OR REMAIN APPROXIMATELY THE SAME.

1. Reactor power as indicated on the APRM's
2. Core Differential pressure
3. Core indicated flow
4. Failed jet pump flow
5. Failed jet pump loop flow (0.5 ea)

QUESTION 4.03 (4.00)

With regard to OI 47 (Turbine Generator):

a. During turbine startup, EXPLAIN WHY you are cautioned not to operate below 800 rpm for greater than five (5) minutes. (1.0)

b. The procedure states "critical speeds of the unit must be known by the operating personnel". WHAT are these speeds for Unit 2 and WHY is operation at these speeds undesirable? (1.0)

c. who's permission is needed to manually synchronize the generator and which synchronizing mark (Red or Green) is used? (1.0)

d. After synchronizing, the voltage regulator transfer volt meter is reading -5 volts. WHAT is actually measured and HOW is it nulled? (1.0)

QUESTION 4.04 (2.50)

In reference to the Cold Startup section of GOI-100-1:

a. During the heatup WHAT THREE (3) parameters are required to be recorded every 15 minutes? (1.0)

b. List 3 of 4 action steps required prior to transferring the mode switch to RUN. Include required and/or expected parameter valves in your list. (1.5)

NOTE: ACTION STEPS MAY HAVE MULTIPLE ACTIONS AND/OR CHECKS

QUESTION 4.05 (3.00)

In reference to GOI 100-12, Normal Shutdown from Power:

a. If the RSCS is found inoperable at 15% reactor power during the shutdown, WHAT must be done? (0.5)

b. With the unit in cold shutdown and the reactor pressure vessel (RPV) head torqued, WHAT are TWO indications of reactor vessel water stratification, OTHER THAN indication of pressure on the RPV? (2.0)

c. WHAT is the moderator temperature range which must be maintained during cold shutdown? (0.5)

QUESTION 4.06 (2.00)

Regarding EOI-36, Loss of Coolant Accident Inside Drywell:

a. WHAT TWO (2) criteria are to be used to determine the difference between excessive primary coolant leakage and a loss of coolant accident (LOCA)? (1.0)

b. When attempting to locate and isolate the break, the procedure cautions the operator to ensure sufficient reactor coolant inventory to keep the core covered before isolating a line break. WHY is this caution necessary? (1.0)

QUESTION 4.07 (2.00)

According to EOI-41 (Methods of Water Makeup to the Reactor), WHAT are FOUR (4) plant systems which can be used to supply HIGH pressure water makeup to the reactor with NO NUCLEAR STEAM AVAILABLE? (2.0)

QUESTION 4.08 (3.00)

Other than supervisor notification, WHAT are the SIX (6) immediate operator action steps required by EOI-47 (Failure of Reactor to Scram When Required or Failure of Control Rods to Fully Insert During Scram)? (3.0)

NOTE: ACTION STEPS MAY HAVE MULTIPLE ACTIONS AND/OR CHECKS

QUESTION 4.09 (2.50)

Answer the following regarding the operator actions of GOI-100-11 (Reactor Scram) for a REACTOR SCRAM WITH MSIVs CLOSED:

a. WHY is it preferable to manually operate the MSRVs to control reactor pressure rather than allowing them to cycle on their own? (1.0)

b. At WHAT torus temperature is it necessary to place an RHR loop in torus cooling? (0.5)

IF MSIVs CANNOT BE REOPENED:

c. Briefly, HOW is HPCI used to control reactor pressure if it is not required for level control? (0.5)

d. WHY fire an auxiliary boiler if the MSIVs cannot be re-opened? (0.5)

QUESTION 4.10 (1.50)

When exiting from a controlled zone and performing the required self frisk:

a. At WHAT instrument meter reading have you exceeded the Maximum Contamination Limit for Skin? (0.75)

b. WHAT would you do if you exceeded the above limit? (0.75)

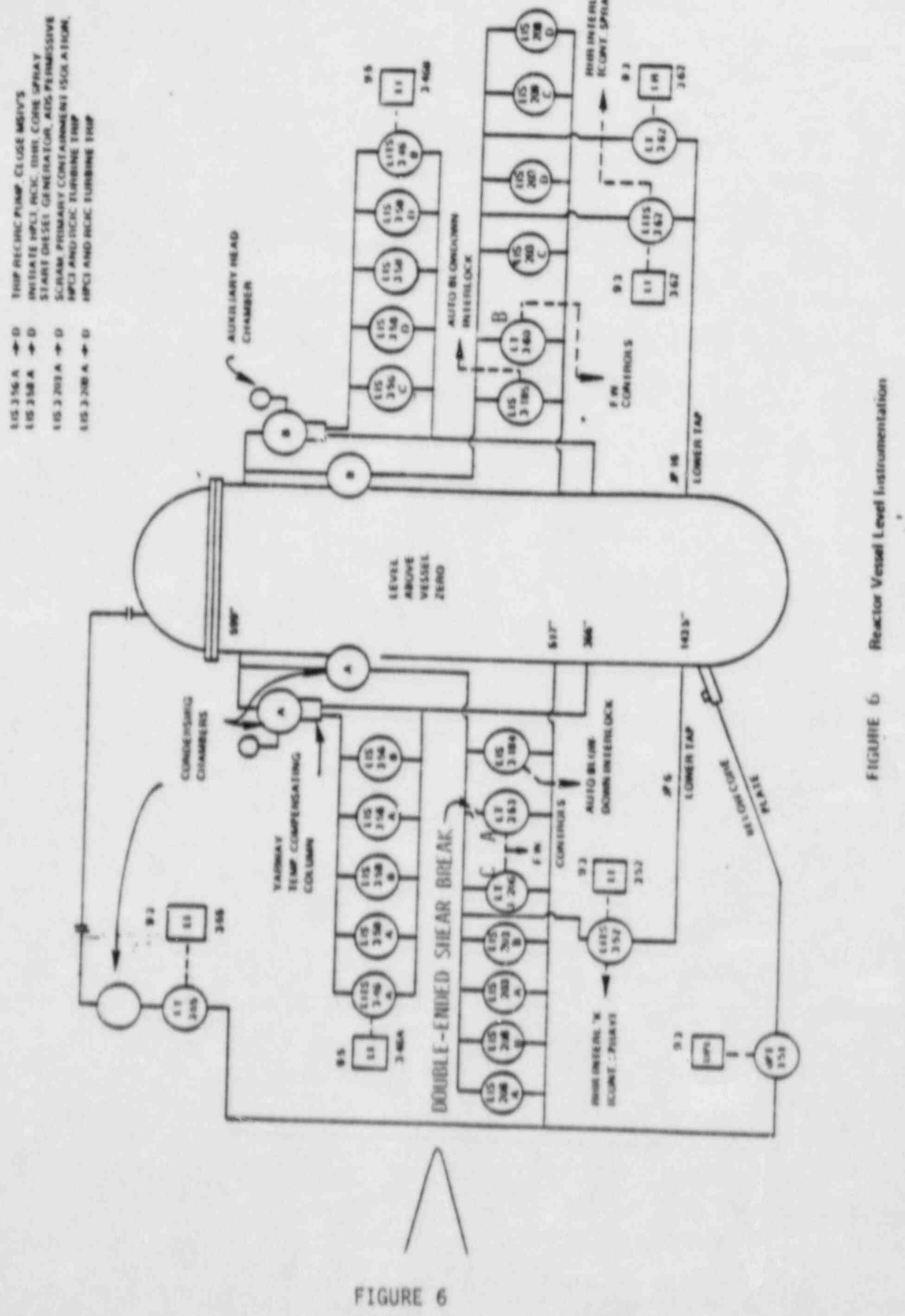
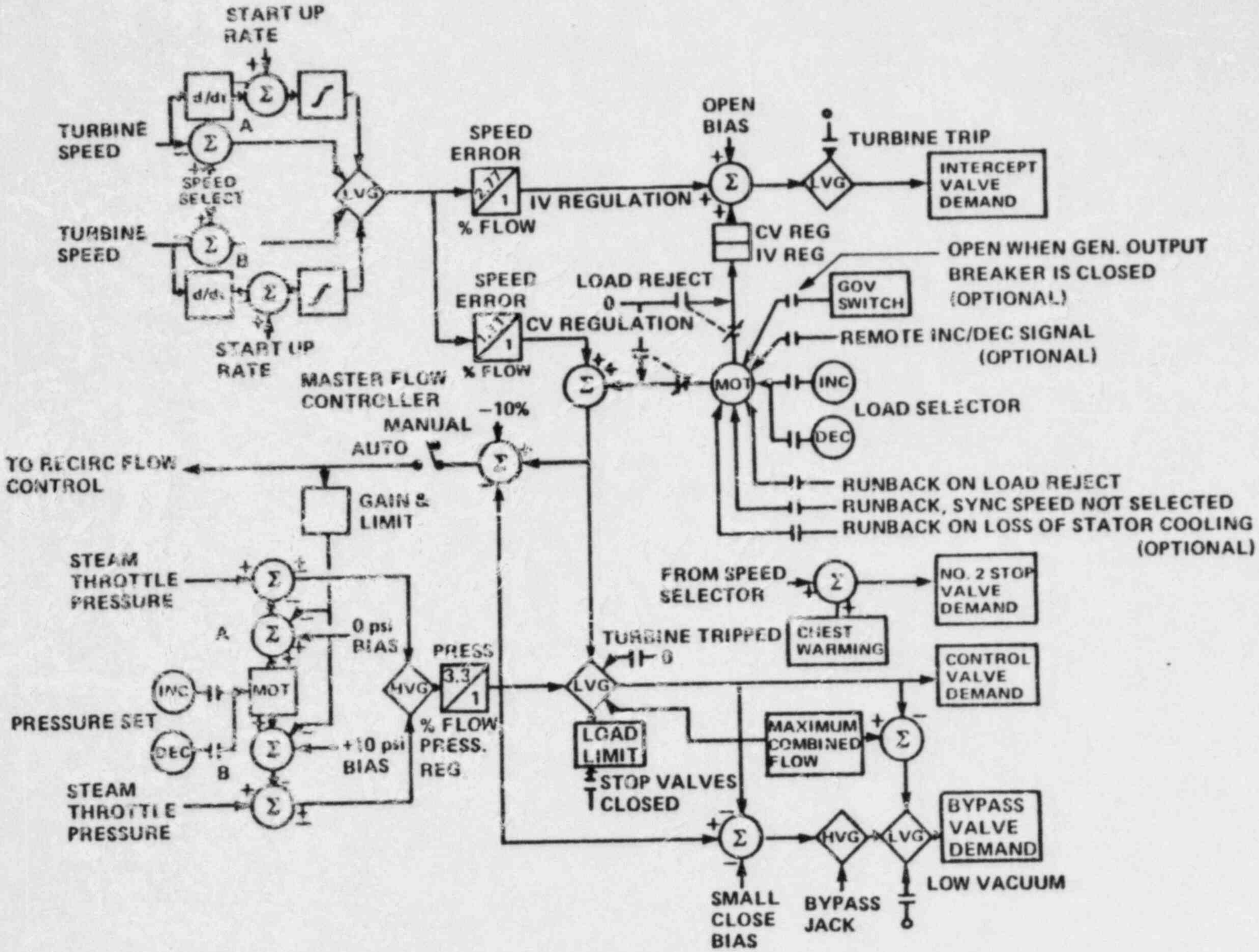

DOUBLE-ENDED SHEAR BREAK
CONDENSING CHAMBERS
AUXILIARY HEAD CHAMBER
LEVEL ABOVE VESSEL ZERO
LOWER TAP
AUTO BLOWDOWN INTERLOCK
F.W. CONTROLS
FIGURE 6 Reactor Vessel Level Instrumentation

FIGURE 6

FIGURE 7



Electro-Hydraulic Control Logic

EQUATION SHEET

$f = ma$

$v = s/t$

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

$w = mg$

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

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

$W = v\ \Delta P$

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

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

$\Delta E = 931\ \Delta m$

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

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

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

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

$Pwr = W_f \Delta h$

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

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

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

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

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

$SUR = 26.06/T$

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

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

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

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

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

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

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

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

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

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

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

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

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

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

$I_1 d_1 = I_2 d_2$

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

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

$\Sigma = \sigma N$

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

ANSWER 1.01 (2.00)

a. Using P = Po e to the t/T then P = 75 e to 60/-80
P = 75 e to -0.75 = 35 on Range 4 [1.0]

b. On down-power transients, the rate of power change is limited by the rate of decay of the longest lived precursors,thus retarding the rate of power decrease.[1.0]

REFERENCE
BRF Rx Physics Review pg. 23&24 CED 259
BRF Requal Rx Theory pg. 25&26

ANSWER 1.02 (2.00)

Using : Capacity is prop. to speed, Pump Head is prop. to speed squared, and Power prop. to speed cubed.

Hd2/Hd1 = 50/200 = 1/4

1/4 Hd prop. speed squared

a. Therefore 1/2 speed 1 = speed 2 = 1500 RPM [0.67]

b. 1/2 vol. 1 = 1/2 speed 1 = volume 2 = 15000 gpm [0.67]

c. P1/P2 = (speed 1/speed 2) cubed

P2 = (speed 2/speed 1) cubed X P1 = (1500/3000) cubed X P1

= 1/8 P1 [0.66]

REFERENCE
BRF Pumps & Fluid Flow 6.2 pg. 5 CED 260
BRF Requal Cent. Pumps pg. 5

ANSWER 1.03 (2.00)

Since the reactor was shut down by 1% dk/k as determined at the time of peak Xenon, then the SDM will decrease as Xenon decays.[1.0] Since Xenon (peak) is greater than the 1% dk/k a reactor restart would occur.[1.0] (2.0)

REFERENCE
BRF Rx Physics Review pg. 43-47 CED 261
BRF Requal Rx Theory pg. 20-32

ANSWER 1.04 (3.00)

a.- Power, Local power, Flux, or Local flux
- Flow
- Pressure
- Inlet subcooling
(3 of 4 req. @ 0.333 each) (1.0)

b. 1. MAPRAT is the ratio of MAPLHGR(act) to MAPLHGR(LCO).[0.75]
2. NO [0.5]
3. The clad temperature can exceed 2200 deg. F during a DBA LOCA.[0.75] (2.0)

REFERENCE
BRF Plant Performance 10.2 CED 262
BRF Requal BWR Thermal Hyd. Review pg. 22-25 & 31-33

ANSWER 1.05 (2.50)

a. Doppler or fuel temperature
b. Void
c. Moderator temperature
d. Void
e. Moderator temperature

(0.5 each) (2.5)

REFERENCE
BRF Rx Physics Review pg. 31, 39 & 40 CED 263
BRF Requal Rx Theory pg. 27-30

ANSWER 1.06 (2.50)

a. Cycle efficiency would be increased by a decrease in subcooling.[0.5] As less heat is rejected to the condenser, the returning condensate requires less reactor heat to produce steam.[1.5] Therefore cycle efficiency will increase.

b. By decreasing the temp or increasing the flow rate of the cooling water to the condenser, the operator can directly increase subcooling. [0.5]

REFERENCE
BRF Plant Performance 5.3 pg. 1 & 7.3 pg. 2 CED 264

ANSWER 1.07 (2.00)

a. Heat produced at some time after the fission event [0.5] is decay heat. It is produced by the radioactive decay of the fission products [0.5] Will accept: Power in core after a scram [0.5]

b. No.[0.5] The nuclear instrumentation indicates neutrons.[0.5]

REFERENCE
GE Rx Fundamentails NEDO 10806 CED 266

ANSWER 1.08 (3.00)

a. .64% (will accept .6 to .7 %)[0.5]
b. Decreases[0.5] due to the production of Plutonium[0.5] which has a lower delayed neutron fraction than U-235.[0.5]
c. Delayed neutrons increase the average neutron generation time.[0.75] (by a factor of more than 1000) Increasing the control time of the reactor.[0.25]

REFERENCE
BRF Rx Physics Review pg. 24&25 CED 267
BRF Requal Rx Theory pg. 7-11

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
BRF Plant Performance 9.3 pg. 2 CED 268
BRF Requal BWR Thermal Hyd. Review pg. 17-20

ANSWER 1.10 (1.50)

After power operation, the gamma and deuterium[0.5] concentrations are high enough to produce significant numbers of source neutrons, along with the alpha-oxygen reaction[0.5] and spontaneous fission of Uranium and Plutonium.[0.5] (1.5)

ALTERNATE ANSWER: Spontaneous fission of Cm-242 [0.5] and Cm-244 [0.5] (for low and high exposure fuel respectively, provide sufficient numbers of source neutrons.)

REFERENCE
BRF Rx Physics Review pg. 4 CED 269

ANSWER 1.11 (2.00)

a. As temperature increases, migration length increases exposing the control rod to increased thermal neutron flux, thus rod worth increases.[0.67]

b. As an adjacent rod is inserted the remaining rods zone of control decreases, the rod is exposed to a reduced thermal flux thus the rod worth will decrease.[0.66]

c. Differential rod worth is greatest as the rod end travels through the peak axial thermal neutron flux.[0.67]

REFERENCE
BRF Rx Physics Review pg. 35-37 CED 270

ANSWER 2.01 (2.50)

a. The cleanup recirc pumps will trip[0.5] and the hold pumps will auto start when flow decreases.[0.5]

b. To stay within temperature limitations on the F/D inlet [0.5] and RBCCW NRHX outlet [0.5] due to discharged water is unavailable to cool the Regen HX. [0.5]

REFERENCE
BRF RWCU L.P. pg. 13 & 16 CED 271

ANSWER 2.02 (3.00)

a.	Pump	Power	
1.	A3	3EA	(0.25)
2.	B3	C	(0.25)
3.	C3	3EB	(0.25)
4.	D3	D	(0.25)

b. The pumps assigned to EECW will auto start.[0.5] The RBCCW system FCV's open to send EECW to the RBCCW HX's.[0.5]

c. The RHRSW should be at the greater pressure[0.5] this is to ensure that any leakage is into, not out of a radioactive system.[0.5] OR will accept to prevent an uncontrolled radioactive release to the environment.[0.5]

REFERENCE
BRF RHR L.P. pg. 10 & EECW L.P. CED 272

BRF Requal RHR L.P. pg. 10, 23, 24, 26
RHRSW L.P. Objective 2
EECW L.P. Objective 1 pg. 1

ANSWER 2.03 (3.00)

1. HPCI - Loss of power to valves, pumps and Division II logic
2. CS - System II will not auto initiate if needed
3. RHR - System II will not auto initiate if needed
4. RCIC - Loss of Division II logic
5. MSRV's - Loss of operability & ind. on some valves ADS - two valves lose normal power to auto scram to RMOV-B
6. Recirc - "A" MG loss of emerg. oil pump, loss of speed control and resulting lockup of its scoop tube
7. RPS - Loss of backup scram capability
8. MSIV's - The outboard MSIV DC solenoids will deenergize.

(6 of 8 required at 0.5 each)

PCIS - Loss of power to outboard isol. valves

REFERENCE
BRF OI 57 pg. 75-78 & BRF DC Elec. Dist. L.P. CED 273
BRF ADS Regnal LP P.7
BRF PCIS LP

ANSWER 2.04 (3.00)

a. A scram signal deenergizes the scram pilot valves[0.33], venting air from the scram inlet and outlet valves, allowing them to open[0.33]. This vents water from the overpiston area of the CRD to the SDV[0.33] and applies HCU accumulator water to the underpiston area of the CRD[0.33]. This provides the initial motive force for the rod[0.33]. As accumulator pressure drops below reactor pressure, a ball check valve in the CRD opens to apply reactor pressure to the CRD to complete the scram stroke[0.33].

b. The accumulators will not fully recharge until the scram is reset.[0.5] Drive seal leakage is greater than CRD pump capacity.[0.5]

REFERENCE
BRF CRD Hydraulics L.P. & Control Rod Blade and Drive Mech. CED 277
pg. 18 for part b.

ANSWER 2.05 (2.50)

The ATWS trip is initiated by reactor pressure reaching 1120 psig [0.5] and results in a trip of the drive motor breaker. This trip cannot be bypassed.[0.5]
The RPT is initiated on TCV fast closure OR TSV closure.[0.5]
The RPT is bypassed automatically by turbine first stage pressure @ <30% load or 154 psig.[0.5] The Rpt opens two (2) redundant breakers between the MG & recirc pump motor.[0.5]
will accept 142 psig

REFERENCE
BRF Recirc L.P. & Recirc Flow Control L.P. CED 279
PLAQUE ON UNIT 1 EHC PNL. & LER 84-014-01
BRF Requal Recirc L.P. pg. 14 & 24

ANSWER 2.06 (3.50)

a. Low reactor pressure. (0.5)

b. Draining the CST to the suppression pool. (1.0)

c. The CST being sufficiently elevated above the HPCI gravity fills the discharge piping. (1.0)

d. HPCI will start and operate, but it will overspeed due to low flow feedback.[0.7] After it overspeeds, it will then reset itself and the sequence of events will be repeated.[0.3] (1.0)
OR - HPCI will start and operate at maximum speed [1.0]

REFERENCE
BFNP Hot License Lesson Plan 42, pg. 11, 17 - 19, & Figure 1. CED 285
BFNP OI-73
BFNP Requal HPCI L.P. Objective 1 & 2, pg. 5, 8
Singer Simulator Malfunction Description

ANSWER 2.07 (2.00)

a. Depressurizes the reactor vessel in time to allow fuel cooling by CS and LPCI systems [0.5] (following a LOCA) if the other makeup systems (feedwater, CRDH, RCIC, and HPCI) fail to maintain vessel level [0.5]. (1.0)

b. Long enough so that HPCI has time to start [0.5] and yet not so long that CS and LPCI are unable to adequately cool the fuel if HPCI should fail to start [0.5]. (1.0)

REFERENCE
BFNP Hot License Lesson Plan No. 43, "ADS," pp. 4 & 5. CED 286

ANSWERS -- BRF -- 84/03/23 -- DODD, C.

BFNP Requal Lesson Plan, "ADS," pp. 5 & 6, (Objective 2.b).

ANSWER 2.08 (3.00)

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]. water hammer and torus integrity (1.5)

b. Increases [0.5]. The open vacuum breaker provides a direct path to the drywell for the steam entering the SRV discharge line [1.0]. (1.5)

REFERENCE
BFNP Hot License Lesson Plan No. 9, p. 5, and NUREG/BR-005/Vol. 5, No. 4, Power Reactor Events, January 1984, p. 5, "Uncontrolled Leakage of Reactor Coolant Outside Primary Containment - Update from Vol. 5, No. 1," (event at Hatch Unit 2 on August 25, 1982). CED 288

ANSWER 2.09 (3.00)

a. Indicates possible CS pipe break inside the reactor vessel (between shroud and vessel wall -- NOT REQUIRED for full credit) [0.75] which could cause CS injection flow to go into the vessel annulus instead of on top of the core [0.75]. (1.5)

b. Indicates leakage back from the reactor through the testable check valve and the injection valve [0.75] which could cause overpressurization and possible rupture of the low pressure portion of the CS discharge piping [0.75]. (1.5)

REFERENCE
BFNP Hot License Lesson Plan No. 45, p. 8 and Figure 3. CED 289

ANSWERS -- BRF -- 84/03/23 -- DODD, C.

ANSWER 3.01 (3.00)

a. <50% rod density[0.33] to 30% power[0.33] as sensed by turbine first stage pressure.[0.33]

b. When in GNC, rod blocks are enforced which:
   1. Prevent any rod in the group from being moved in opposite direction from the first rod moved.[0.5]
   2. Prevent any rod in group from being moved >1 notch in direction of first movement.[0.5]

c. Rod position is determined by sensing direction in which the rod movement control switch is moved[0.5] & sensing RMCS timer settle function.[0.5]

REFERENCE
BRF RSCS L.P. CED 274

ANSWER 3.02 (3.50)

a. MOV 48 will auto close on the following:
   1. Rx Pressure = or > 100 psig [0.5]
   2. DW Pressure = or > 2.45 psig [0.5]
   3. Rx Water Level = or < 11 inches [0.5]

b. 1. No they will not open.[0.25] The valves are interlocked closed when the 25 & 26 valves are open.[0.5]
   2. Yes they will open.[0.25] Automatic logic & Equipment interlocks are not operable when equipment is controlled from the backup control panel.[0.5] (on MOV Board) CD
   OR- No [.25] No control at backup control pnl - (Control at mov Board) [.5]

c. A drain path from the reactor to the torus would be established.[0.5]

REFERENCE
BRF RHR L.P. pg. 18 & Control Rm. Abandonmet SP BF 2.1 CED 275

BRF Requal RHR L.P. Objective 7, pg. 11, 12 & 14

Physical verification at facility

ANSWERS -- BRF -- 84/03/23 -- DODD, C.

ANSWER 3.03 (2.50)

a. Auto bypassed: a<30% power.[0.33] Selecting an edge rod.[0.33]
Manual BP of either RBM with joystick.[0.33]
ALSO GAVE CREDIT FOR DESELECT ALL RODS

b. RBM A is not effected.[0.5] RBM channel B uses APRM B as its normal reference input but when the APRM is bypassed the alternate APRM D is automatically placed in the circuit. Hence, no adverse effect on either RBM channel.[1.0] [0.5] (1.0)

c. (3) [0.25] There are 12 LPRM's, 6 in RBM A & 6 in B. Must have at least 50% operable/channel; hence 3/CHANNEL - 6 total.[0.25]

REFERENCE
BRF RBM L.P. CED 276

BRF Requal RBM L.P. Objective 2, 3 & 5 pg. 10, 15-16 & 21

ANSWER 3.04 (3.00)

a. The TCV's will close ~~fully.~~ TO 25% OPEN OR 50% FLOW [0.5] The TCV LVG passes ~~passing the~~ a MCF signal of ~~0%~~ 50% rather than the signal from the pressure controller.[0.5]

b. The BPV's will remain closed through the transient.[0.5] The MCF summer will send a zero signal to the BPV LVG.[0.5]

c. Reactor power & pressure will rapidly increase following the fault.[0.5] The reactor will scram on High Flux &/OR High Pressure. Reactor pressure will be controlled by the ~~SRV's.~~ TCV's [0.5]

REFERENCE
BRF EHC L.P. CED 283
BRF Transient Hxy-16
BF SMG 147-1.2, p.34
P248-AL-4985, p.20A

ANSWERS -- BRF -- 84/03/23 -- DODD, C.

ANSWER 3.05 (3.00)

a. Full scale. (0.5)

b. {If Part (a) is answered correctly} -- The FWCS detects an erroneous high level (Ch. "A") [0.5] which sends a negative (decrease) signal to the RFP speed controller [0.5] resulting in a decreasing actual reactor water level [0.5].

-OR-

{If Part (a) is answered incorrectly} -- Everything opposite of above. (1.5)

c. 2 out of 3 coincidence (Ch. "A" & "C" failing high due to rupture) RFP and main turbine trips causing a turbine trip scram.

-OR-

Low level scram from LIS 203s. (1.0)

REFERENCE
BFNP Hot License Lesson Plan No. 12, "Feedwater Level Control," and Transient EXY-6. CED 290

ANSWER 3.06 (2.00)

a. 1. Mode switch in "Run."[0.5]
2. Companion APRMs on scale.[0.5]

b. False.[0.5] This interlock does not stop detector, only initiates a rod block.[0.5]

REFERENCE
BFNP Hot License Lesson Plan No. 20, "IRMs," p. 18 & 19 CED 291

ANSWERS -- BRF -- 84/03/23 -- DODD, C.

ANSWEF 3.07 (3.00)

a. Breaker closure follows:
   1. Diesel 870 rpm close permissive signal.[0.5]
   2. All other supply breakers open with NO overcurrent lockout.[0.5]
   3. Under voltage exists on 4160 V board.[0.5]

b. Placing these switches in Backfeed will trip and lockout the normal and alternate supply breakers on the associated Unit board [0.75] and allow closing of the unit board to shutdown bus breakers for backfeed operations.[0.75]

REFERENCE
BRF Diesel Gen. L.P. pg. 11 & 14 CED 293

BRF Requal Diesel Gen. L.P. Objective F & pg. 22, 25

ANSWER 3.08 (2.00)

a. False [0.5]

b. Over LPRM detector lifetime the amplifier gain potentiometer[0.75] is used in conjunction with the 3 range positions [0.75](LO, MED & HIGH GAIN)

REFERENCE
BRF LPRM L.P. pg. 10, 11, 13, 17 CED 299

BRF Requal LPRM L.P. pg. 10, 11, 13 & 16

ANSWER 3.09 (2.50)

a. Limits recirc pump speed such that the feedwater control system can maintain or recover reactor vessel water level upon loss of a reactor feed pump [0.75]. Limits speed when one or more RFPs are at <20% rated flow [0.375] and reactor vessel level is <+27 inches [0.375]. (1.5)

b. Pump speed will increase to 45% at which time the Master Controller low speed limiter will be limiting. (1.0)

REFERENCE
BFNP Lesson Plan No. 8, "Recirculation Flow Control," pp. 5 & 9. CED 320

ANSWERS -- BRF -- 84/03/23 -- DODD, C.

ANSWER 4.01 (2.00)

1 CHECK TR-1-1 & ACCOUSTIC MONITOR TO DETERMINE WHICH VLV. IS OPEN

2 ~~1~~. Operate relief valve control switch 2 or 3 times.~~[0.5]~~

3 ~~2~~. Initiate suppression pool cooling.~~[0.5]~~

4 ~~3~~. Change EHC pressure regulator setpoint.~~[0.5]~~

5 ~~4~~. Once determined valve will not close and/or prior to exceeding 110 deg. F torus temperature, reduce reactor power and uppon Shift Supervisors approval manually scram the reactor.~~[0.5]~~

REFERENCE (4 of 5 REQ @ 0.5 each)

BRF OI-1 IV.C 1-5 pg. 6 & 6A CED 280

ANSWER 4.02 (2.50)

1. decreases
2. decreases
3. increases
4. increases - WILL ACCEPT DECREASE IF RISER FAILURE IS SPECIFIED
5. remains approx the same or small decrease

REFERENCE

Browns Ferry OI-68 pg 29 CED 281

ANSWER 4.03 (4.00)

OR: Low steam flow to last stages during s/u ⇒ signif. heating of last stage blading and inner casing [not per OI but ok for .7 partial credit]

a. Operation at low speed may result in large rotor bows being generated by rubbing with no indication of high vibration from the turbine supervisory instruments.[1.0]

900-1100 RPM & 1365-1415

b. CAF on Unit 2 critical speeds.[0.5] Operation at critical speed would be to operate at the point of maximum vibration.[0.5]

c. The Operating Supervisor permission is required.[0.5] and the Green mark is used for manual sync.[0.5].

d. The transfer voltmeter reads the difference between the Auto (90P) & Manual Voltage Regulator outputs.[0.5] By adjusting the manual voltage regulator (70P) the meter is nulled.[0.5]

ANSWERS -- BRF -- 84/03/23 -- DODD, C.

REFERENCE
BRF Turbine Generator OI 47 CED 282
BRF Main Gen & Aux LP
Letter from BFNP
BRF Main Turbine LP P.14

ANSWER 4.04 (2.50)

a. Parameters:
1. Steam dome pressure [0.33]
2. Reactor bottom head drain temperature [0.33]
3. Recirc loop A & B temperature [0.33]
4 Moderator Temperature
5. Vessel Shell Temp adjacent to Flange

b. Checks prior to transfer to RUN:
1. APRMs > 5% & < 12%
2. Inboard and outboard MSIVs open
3. Condenser vacuum > 26" Hg vacuum, and condenser A, B, & C low vacuum annunciators cleared
4. Reactor pressure > 850 PSIG and low reactor pressure annunciators cleared

(3 of 4 required at 0.5 each) (1.5)

REFERENCE
BRF GOI-100-1. & BRF Q&A 5-4 & 5-127 CED 284
BFNP Tech. Spec. /SI - 3.6.A.1 /4.6.A.1

ANSWER 4.05 (3.00)

a. Manually scram the reactor. (0.5)

b. o Temperature differential of > 50 F between bottom vessel head and "A" FW sparger when moderator is > 150 F.

o Temperature differential of > 75 F between bottom vessel head and "A" FW sparger when moderator is < 150 F.

o FW sparger temperature of > 200 F on either "A" or "B" FW sparger when recirc. pumps and shutdown cooling are OOS.

(2 of 3 at 1.0 each) (2.0)

c. Between 180 F and 200 F. (0.5)

REFERENCE
BF GOI 100-12, pp. 1, 7, & 14. CED 294

BFNP Requal Lesson Plan, "GOI-100-12," (Objectives 1, 3, & 4).

ANSWER 4.06 (2.00)

a. A break large enough to result in a reactor scram on High Drywell Pressure [0.5] or Low Reactor Water Level [0.5]. (1.0)

b. Isolation of line breaks may cause reactor pressure to increase rapidly [0.5] resulting in level shrink uncovering the core.[0.5] (1.0)

REFERENCE
BF EOI-36, pp. 1 & 5.

BFNP Requal Lesson Plan, "EOI-36" (Objectives 1 & 4). CED 295

ANSWER 4.07 (2.00)

o CRD
o SLC
o RCIC
o HPCI (0.5 each)

REFERENCE
BF EOI-41, pp. 1 & 2. CED 296

BFNP Requal Lesson Plan, "EOI-41" (Objective 1).

ANSWER 4.08 (3.00)

o If scram does not occur when a scram setpoint is reached, manually scram the reactor.

o Verify existing condition by multiple indications.

o Verify all automatic actions have occurred. If not, place controls in manual and make corrective manipulations.

o Trip recirc pumps.

c Place mode switch in shutdown. Place scram discharge volume high water level bypass switch to bypass.

o Reset scram (verify scram discharge vents and drains open) and manually scram the reactor, reset and repeat if rod motion is observed until all control rods are fully inserted.

(0.5 each)

REFERENCE
BF EOI-47, pp. 1 & 2. CED 297

BFNP Requal Lesson Plan, "EOI-47" (Objective 1).

ANSWER 4.09 (2.50)

a. To sequence the opening order per the MSRV position chart in order to minimize torus hot spots. (1.0)

b. > 95 F. UNIT 1 OR 93° UNIT 2 & 3 - MUST SPECIFY UNIT FOR FULL CREDIT (0.5)

c. In test mode with flow to the CST. (0.5)

d. To return steam seals and SJAEs to service. (0.5)

REFERENCE
BF GOI-100-11, pp. 5-7. CED 298
BF OI-64, P.21; BF OI-74, P.8
BFNP Requal Lesson Plan, "GOI-100-11" (Objective 4, for Part [d]).

ANSWERS -- BRF -- 84/03/23 -- DODD, C.

ANSWER 4.10 (1.50)

a. The maximum reading is 300 cpm which is equal to the limit of 0.05 mrad/hr.[0.75]

b. Notify Health Physics.[0.75]

REFERENCE
BRF RCI pg. 8 & 9 CED 321

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

Reviewed by: Champion
J. D. [illegible]
Duke
[illegible]
Dept [illegible]

FACILITY: BRE

REACTOR TYPE: BWR

DATE ADMINISTERED: 84/03/23

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
24.00	24.12	________	________	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.50	25.63	________	________	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.00	25.13	________	________	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.13	________	________	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
99.50	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 (3.00)

Refer to the attached FIGURE 1, Integral Rod Worth, to answer the following:

a. HOW MUCH reactivity will be added to the core by withdrawing the rod from notch position 4 to notch position 16? (0.5)

b. From attached Figure 1, SKETCH a DIFFERENTIAL rod worth curve with appropriately labled axes. (1.5)

c. Assuming all other parameters remain constant, HOW would rod worth be affected (INCREASED, DECREASED, or UNCHANGED) by an increase in the moderator temperature from cold to hot at 1% power? Briefly explain WHY? (1.0)

QUESTION 5.02 (2.00)

Assume the reactor is shut down with a shutdown margin of 1.8%. If the control rods are withdrawn until the count rate increased by a factor of 20 with the reactor still subcritical, WHAT is the new K-eff? SHOW ALL WORK.

QUESTION 5.03 (2.50)

With the plant operating at 100% power, a TOTAL LOSS OF FEEDWATER flow occurs. Answer the following using the transient information on attached FIGURE 2:

a. WHY does reactor POWER initially decrease [AREA 1] AND subsequently decrease more rapidly [AREA 2]? (1.0)

b. WHAT is causing the reactor LEVEL increase several minutes after the loss of feed [AREA 3]? (0.5)

c. WHAT caused core FLOW to decrease initially [AREA 4] AND subsequently [AREA 5]? (1.0)

QUESTION 5.04 (1.80)

The tabulation below illustrates REACTIVITY COEFFICIENT VARIATIONS due to increases in several core parameters. PLACE ARROWS in the squares labeled (a) through (i) indicating how the ABSOLUTE VALUE of that coefficient varies if the indicated parameter is increased.

CORE PARAMETER→ / COEFFICIENT↓	↑ Moderator Temp.	↑ Core Voiding	↑ Rod Density	↑ Fuel Temp.	↑ Core Age
Void Coefficient	→	(a) ↑	(b) ↑	(c) ↑	(d) ↓
Moderator Temp. Coefficient	(e) ↑	→	↑	↑	(f) ↓
Fuel Temperature Coefficient	↑	(g) ↑	→	(h) ↓	(i) ↑

QUESTION 5.05 (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 5.06 (1.50)

Referring to attached FIGURE 3, Operating Map for Units 2 & 3:

a. WHY does core flow increase with constant recirculation pump speed from Point 7 to Point 4? (1.0)

b. The APRM Rod Block Monitor Line provides protection from exceeding WHICH core thermal limit? (0.5)

QUESTION 5.07 (3.00)

With the plant 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 4:

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 5.08 (2.00)

With regard to the MAPLHGR thermal limit:

a. Briefly, WHAT is the reason, or basis for having a MAPLHGR thermal limit? (0.75)

b. WHICH TWO of the following four parameters affect the MAPLHGR LIMIT? (0.5)

1. Moderator Temperature
2. Type of fuel
3. Fuel exposure
4. Reactor pressure

c. If an OD-6, Option 4 is selected on the Process Computer, the program provides, among other things, MAPRAT. WHAT is the relationship between MAPRAT and MAPLHGR? (0.75)

QUESTION 5.09 (1.00)

WHY does taking steam and heat (extraction steam) from the turbine to heat the feedwater increase overall efficiency of the plant cycle?

QUESTION 5.10 (1.50)

Attached FIGURE 5 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 5.11 (1.70)

Answer the following regarding transient effects on core boiling heat transfer when operating at power:

a. Briefly EXPLAIN the immediate (instantaneous) effect of a sudden core flow INCREASE on the amount of NUCLEATE boiling at the clad surface. (1.0)

b. A sudden flow (INCREASE, DECREASE) in the core could cause film boiling. Briefly, JUSTIFY your choice. (0.7)

QUESTION 5.12 (1.00)

HOW is condensate depression affected (INCREASED or DECREASED) by the following changes in the circ water flowing through the condenser:

a. Flow decreases. (0.5)

b. Temperature increases. (0.5)

QUESTION 6.01 (3.00)

With regard to the High Pressure Coolant Injection (HPCI) System:

a. WHICH HPCI isolation signal does NOT seal in (setpoint NOT required)? (0.5)

b. WHAT would be of immediate concern if the HPCI minimum flow control valve FAILED TO SHUT following a HPCI turbine trip? (1.0)

c. With the system in "standby," HOW is the pump discharge piping maintained full of water? (1.0)

d. WHY is it necessary to maintain the pump discharge piping filled for the system to be considered operable? (0.5)

QUESTION 6.02 (3.00)

The plant is operating at 80% power with the Feedwater Control System (FWCS) in 3-element control and Channel "A" reactor level detector selected. Referring to attached FIGURE 6, "Reactor Vessel Level Instrumentation," answer the following with the rupture shown in the figure. Assume NO OPERATOR ACTION after the rupture occurs.

a. HOW will the associated control room indicator respond for the level transmitter with the rupture (FULL SCALE or DOWNSCALE)? (0.5)

b. EXPLAIN, HOW the FWCS will respond to the change in Channel "A" indicated level due to the rupture [Part (a) above] and HOW the FWCS response will affect ACTUAL vessel level? (1.5)

c. WHAT, if any, automatic actions would you anticipate occurring in the plant as a result of the ruptured transmitter line? (1.0)

QUESTION 6.03 (2.00)

STATE THE BASES for each of the following conditions required for ADS (Automatic Depressurization System) initiation.

a. The -114.5" water level signal. (1.0)

b. The 120 second time delay (i.e., WHY 120 seconds?). (1.0)

QUESTION 6.04 (1.00)

WHAT TWO conditions cause the IRM Hi-Hi Flux scram to be bypassed? (1.0)

QUESTION 6.05 (2.50)

With regard to the Reactor Recirculation Speed Control System:

a. Briefly EXPLAIN the purpose of the 75% speed limiter. Include parameters and setpoints which place the limiter in effect. (1.5)

b. With the plant operating at 23% power and minimum flow, an operator inadvertently shifts the M/A transfer station for recirc pump "A" from "Manual" to "Auto." Assuming NO further operator action, briefly EXPLAIN WHAT will happen to the speed of "A" recirc pump. Continue your discussion to the final steady state speed. (1.0)

QUESTION 6.06 (3.00)

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. (1.5)

QUESTION 6.07 (3.00)

Two Core Spray (CS) System alarms (and their associated setpoints) are listed below. For each, provide the ABNORMAL CONDITION being indicated by receipt of the alarm in the Control Room AND briefly EXPLAIN HOW the abnormal condition could prevent the CS System from accomplishing its design function when required.

a. Core Spray System Header to Core Plate HIGH dP (with a setpoint of 2 PSID decreasing). (1.5)

b. High Pressure Valve Leakage Test (with a setpoint of 450 PSIG increasing). (1.5)

QUESTION 6.08 (3.00)

With the Unit operating at 75% power, an electrical fault causes the Maximum Combined Flow Setpoint to drop to zero. HOW will the following parameters RESPOND after the fault and WHY? Consider their response for ONE MINUTE following the fault. Assume NO OPERATOR ACTION. Attached FIGURE 7, EHC Logic, is provided for reference.

a. Turbine control valve position (1.0)

b. Bypass valve position (1.0)

c. Reactor power AND pressure (1.0)

QUESTION 6.09 (3.00)

During reactor operation at 98% power a loss of the 250 VDC Reactor MOV board "A" occurs. LIST 6 systems and/or MAJOR components that will be affected prior to transferring power, and DESCRIBE one way in which each one is affected. (3.0)

EXAMPLE: (If 250 VDC Reactor MOV board "C" was lost)
RCIC - Loss of power to condensate pump.

QUESTION 6.10 (2.00)

a. WHAT TWO (2) automatic actions take place if the RCW pressure at the inlet to the RBCCW heat exchangers drops to < 15 psig? (1.0)

b. RHRSW cools the RHR heat exchangers. WHICH fluid should be at the greater pressure? WHY? (1.0)

QUESTION 7.01 (3.00)

In reference to GOI 100-12, Normal Shutdown from Power:

a. If the RSCS is found inoperable at 15% reactor power during the shutdown, WHAT must be done? (0.5)

b. With the unit in cold shutdown and the reactor pressure vessel (RPV) head torqued, WHAT are TWO indications of reactor vessel water stratification, OTHER THAN indication of pressure on the RPV? (2.0)

c. WHAT is the moderator temperature range which must be maintained during cold shutdown? (0.5)

QUESTION 7.02 (3.00)

With regard to OI-47, Turbine Generator:

a. During turbine startup, EXPLAIN WHY you are cautioned not to operate below 800 rpm for greater than five (5) minutes. (1.0)

b. The procedure states "critical speeds of the unit must be known by the operating personnel". WHAT are these speeds for Unit 2 and WHY is operation at these speeds undesirable? (1.0)

c. After synchronizing, the voltage regulator transfer volt meter is reading -5 volts. WHAT is actually being measured by this meter and HOW is it nulled? (1.0)

QUESTION 7.03 (2.50)

Five (5) indications of a failed jet pump per BF OI-68, Abnormal Operations Section, are listed below. Indicate on your answer sheet whether these indications INCREASE, DECREASE, OR REMAIN APPROXIMATELY THE SAME.

a. Reactor power as indicated on the APRM's. (0.5)

b. Core differential pressure. (0.5)

c. Core indicated flow. (0.5)

d. Failed jet pump flow. (0.5)

e. Failed jet pump loop flow. (0.5)

QUESTION 7.04 (2.00)

Regarding EOI-36, Loss of Coolant Accident Inside Drywell:

a. WHAT TWO (2) criteria are to be used to determine the difference between excessive primary coolant leakage and a loss of coolant accident (LOCA)? (1.0)

b. When attempting to locate and isolate the break, the procedure cautions the operator to ensure sufficient reactor coolant inventory to keep the core covered before isolating a line break. WHY is the caution necessary? (1.0)

QUESTION 7.05 (2.00)

According to EOI-41 (Methods of Water Makeup to the Reactor), WHAT are FOUR (4) plant systems which can be used to supply HIGH pressure water makeup to the reactor with NO NUCLEAR STEAM AVAILABLE?

QUESTION 7.06 (3.00)

Other than supervisor notification, WHAT are the SIX (6) immediate operator action steps required by EOI-47 (Failure of Reactor to Scram When Required or Failure of Control Rods to Fully Insert During Scram)?

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

QUESTION 7.07 (2.50)

Answer the following regarding the operator actions of GOI-100-11 (Reactor Scram) for a SCRAM WITH MSIVs CLOSED:

a. WHY is it preferable to manually operate the MSRVs to control reactor pressure rather than allowing them to cycle on their own? (1.0)

b. At WHAT torus temperature is it necessary to place an RHR loop in torus cooling? (0.5)

IF MSIVs CANNOT BE REOPENED:

c. Briefly, HOW is HPCI used to control reactor pressure if it is not required for level control? (0.5)

d. WHY fire an auxiliary boiler if the MSIVs cannot be re-opened? (0.5)

QUESTION 7.08 (1.50)

According to BF RCI-9, Special Work Permits (SWPs):

a. WHAT is the NORMAL lifetime limitation on an SWP? (0.5)

b. Should it be deemed necessary to extend an SWP, whose PERMISSION is required, and WHAT is the MAXIMUM lifetime for which an SWP can be extended? (1.0)

QUESTION 7.09 (2.00)

Regarding the Control Room Abandonment procedure:

a. If sufficient time is available to initiate a scram from the control room, WHAT are FOUR (4) of 6 ACTIONS to be carried out AFTER the reactor has been scrammed and all rods have been verified inserted, JUST PRIOR to abandoning the control room? (1.0)

b. WHAT precaution must be taken when transferring valve control switches to the emergency position at the backup control center panel? (1.0)

QUESTION 7.10 (2.00)

Unit 3 is operating at 100% power when annunciation of "Auto Blowdown Relief Valves Open," is received and a drop in main generator MWe is noted, indicating an open relief valve. ~~WHAT~~ List FOUR (4) action steps should the operator take prior to torus temperature reaching 115 degrees F? (2.0)

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

QUESTION 7.11 (1.50)

When exiting from a controlled zone and performing the required self frisk:

a. At WHAT instrument meter reading have you exceeded the Maximum Contamination Limit for Skin? (0.75)

b. WHAT would you do if you exceeded the above limit? (0.75)

QUESTION 8.01 (2.50)

Answer the following relative to the applicable Section Instruction Letter (SIL):

a. According to SIL-45, "Main Steam Isolation Valve Operation," WHAT restriction is placed on MSIV operation? (0.5)

b. According to SIL-73, "Medical Treatment, Rescue, and Evaluation," WHAT are four responsibilities of the Fire Brigade Leader? (2.0)

QUESTION 8.02 (1.25)

Refer to attached FIGURE 8, the Technical Specifications plots of "Reactor Pressure in PRV Top Head vs Minimum Temperature," for the following:

a. WHICH side (RIGHT or LEFT) of Curve #3 must the reactor pressure vs temperature condition be prior to initiating control rod withdrawal to critical? (0.25)

b. WHY were the curves shifted 30°F to the right, as noted in the right hand margin of the figure? (1.0)

QUESTION 8.03 (3.00)

LIST the three (3) emergency conditions covered by the Extreme Emergency Exposure Guidelines and the maximum recommended WHOLE BODY DOSE for each condition.

QUESTION 8.04 (3.00)

According to the TECHNICAL SPECIFICATIONS for the CONTROL ROD SYSTEM:

a. WHEN must the RWM be operable? (1.0)

b. WHAT is a limiting control rod pattern? (1.0)

c. WHAT TWO restrictions are placed on rod withdrawal when a limiting control rod pattern exists? (1.0)

QUESTION 8.05 (1.50)

STATE the POWER TRANSIENT fuel cladding integrity safety limit.

QUESTION 8.06 (2.00)

A Surveillance Instruction (SI) CANNOT be completed on the day it has been scheduled. The Assistant Shift Engineer notes this in his daily journal and completes a Form BF-49, "Data Cover Sheet for Surveillance Instructions Not Performed."

According to SIL-29, "Surveillance Instructions," DESCRIBE the method used by the control room operating staff to ensure subsequent shifts are alerted to perform this particular SI at the earliest opportunity.

QUESTION 8.07 (2.50)

Unit 1 is operating at 100% power. Standby Gas Treatment System train "C" has been inoperable for two (2) days and will require three (3) more days to complete the maintenance work on it.

Using the excerpts from the Technical Specifications attached to the back of the exam, answer the following:

a. HOW is operability of the two remaining SBGT trains assured? (0.5)

b. HOW MANY DAYS can SBGTS train "C" remain inoperable without affecting Unit 1 operation? (0.5)

c. Refer to FIGURE 9. In addition to SBGTS Train "C" being inoperable, the following (as shown in FIGURE 9) are also INOPERABLE:

(1) The "3EB" 4160v Shutdown Board.

(2) The 480v feeder breaker to the "A" 480v Diesel Auxiliary Board from the "A" 4160v Shutdown Board. (The alternate feeder from the "B" 4160v Shutdown Board is closed, supplying the "A" Diesel Auxiliary Board.)

With both remaining power supplies to the "B" 4160v Shutdown Board operable, WOULD you permit maintenance work requiring Diesel Generator "B" to be inoperable for four (4) hours? WHY or WHY NOT?
NOTE: The SBGTS Train "A" Blower is supplied by the "A" 480v Diesel Auxiliary Board. (1.5)

QUESTION 8.08 (3.25)

Regarding Standard Practice BF 12.17, Administration Controls for Plant Operation:

a. WHAT must be done prior to resetting a primary containment isolation? (1.0)

b. Individual system panel checklists for core cooling systems must be reviewed by WHOM? (2 required) (1.0)

c. HOW is the review in Part (b) documented? (0.25)

d. The shift engineer of each shift will review ____________ logs and temporary ____________ from the preceding shifts to ensure that no out-of-normal equipment configurations have resulted from application of these procedures. FILL IN THE BLANKS. (1.0)

QUESTION 8.09 (4.00)

According to SIL-56, Abnormal Voltage Conditions on Safety-Related Ausiliary Power Systems:

a. WHAT are the Minimum Voltage Limits on the 4-kV AND 480-V shutdown boards? (1.0)

b. If any safety-related bus voltage falls below the limits in Part (a), WHAT are THREE (3) of the four required immediate corrective actions? (3.0)

QUESTION 8.10 (2.00)

Briefly explain WHY each of the following RECIRCULATION PUMP STARTING LIMITATIONS are necessary. Be specific.

a. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirc loops are within 50 °F of each other. (1.0)

b. If the temperature of the water in the lower head is more than 145°F below vessel saturation temperature, the recirc pump shall NOT be started. (1.0)

EQUATION SHEET

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$f = ma$

$v = s/t$

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

$w = mg$

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

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

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

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

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

$W = v\ \Delta P$

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

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

$\Delta E = 931\ \Delta m$

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

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

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

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

$Pwr = W_f \Delta h$

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

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

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

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

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

$SUR = 26.06/T$

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

$\Sigma = \sigma N$

$I_1 d_1 = I_2 d_2$

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

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

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

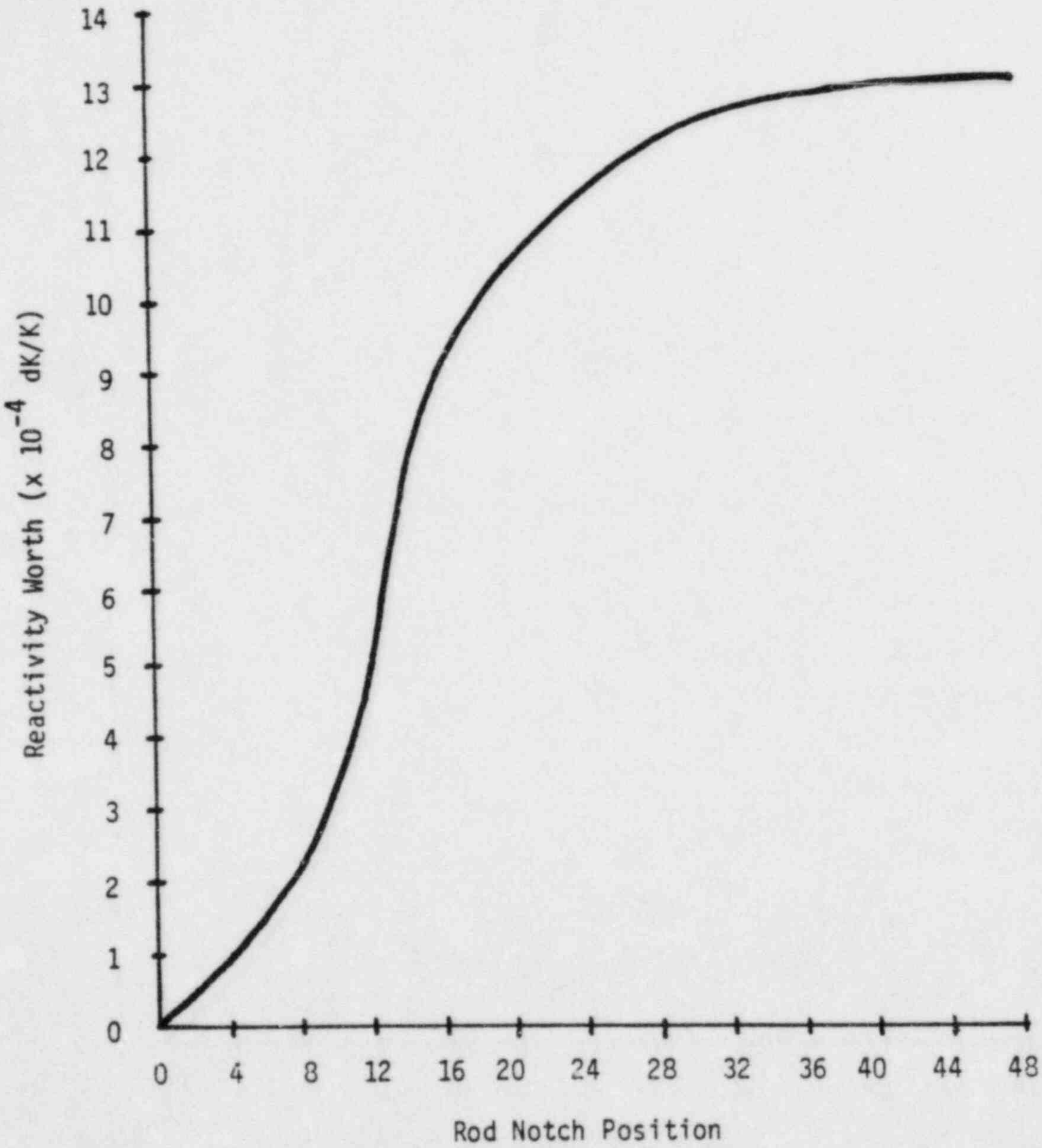
Water Parameters

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.

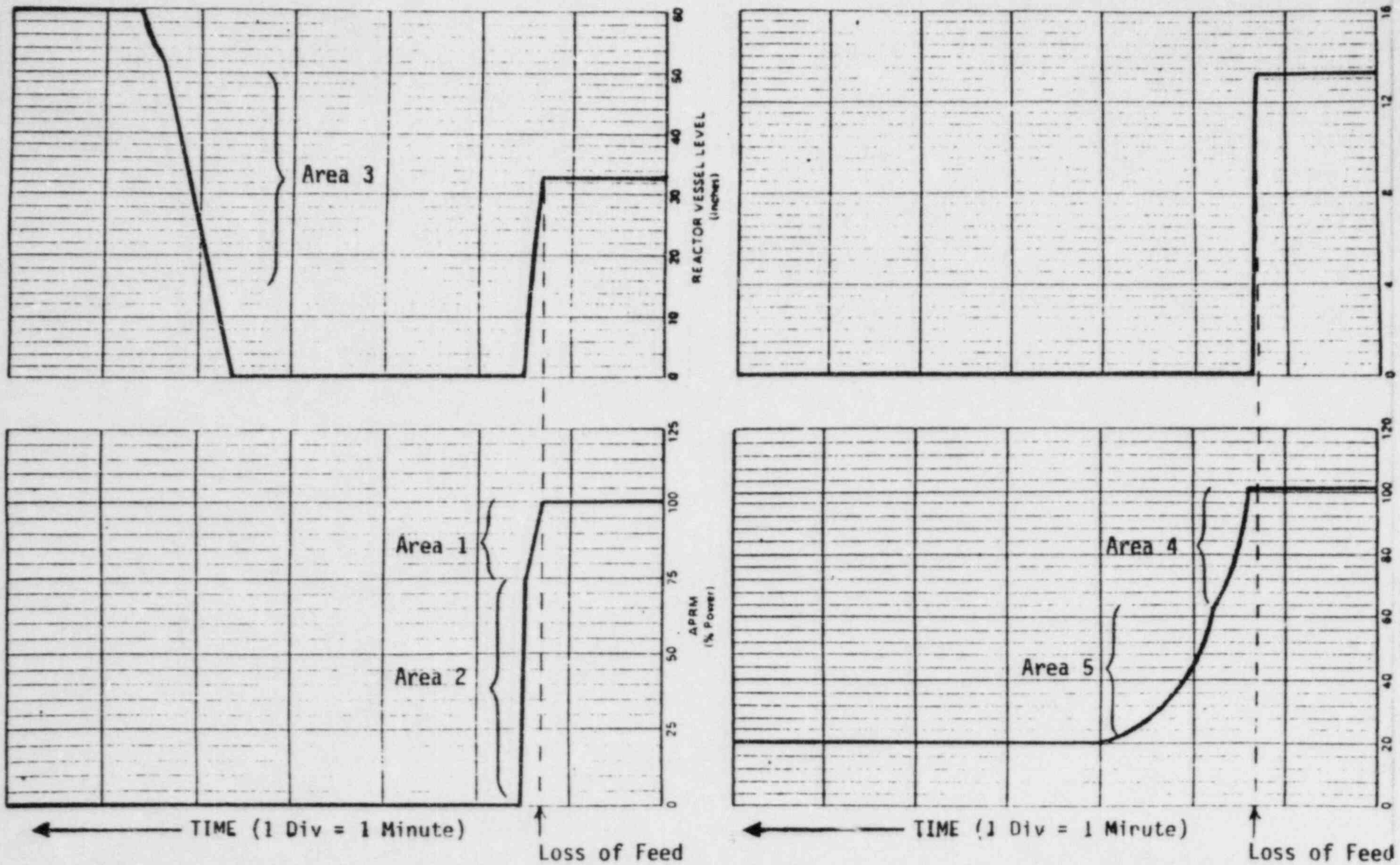
Miscellaneous Conversions

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
1in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)
1 BTU = 778 ft-lbf

FIGURE 1 for Question 5.01

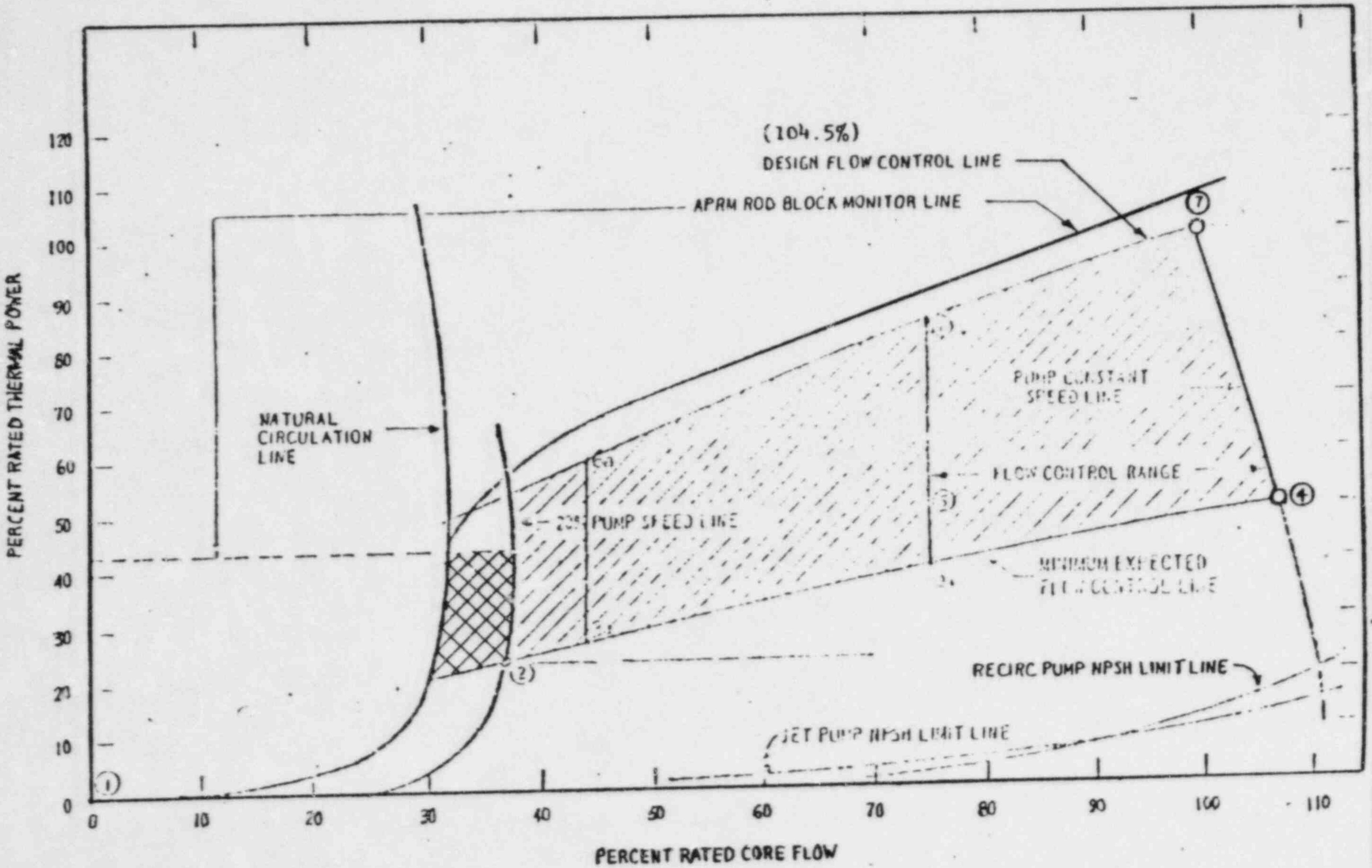


INTEGRAL ROD WORTH



STRIP CHART TRACES FOR TOTAL LOSS OF FEED

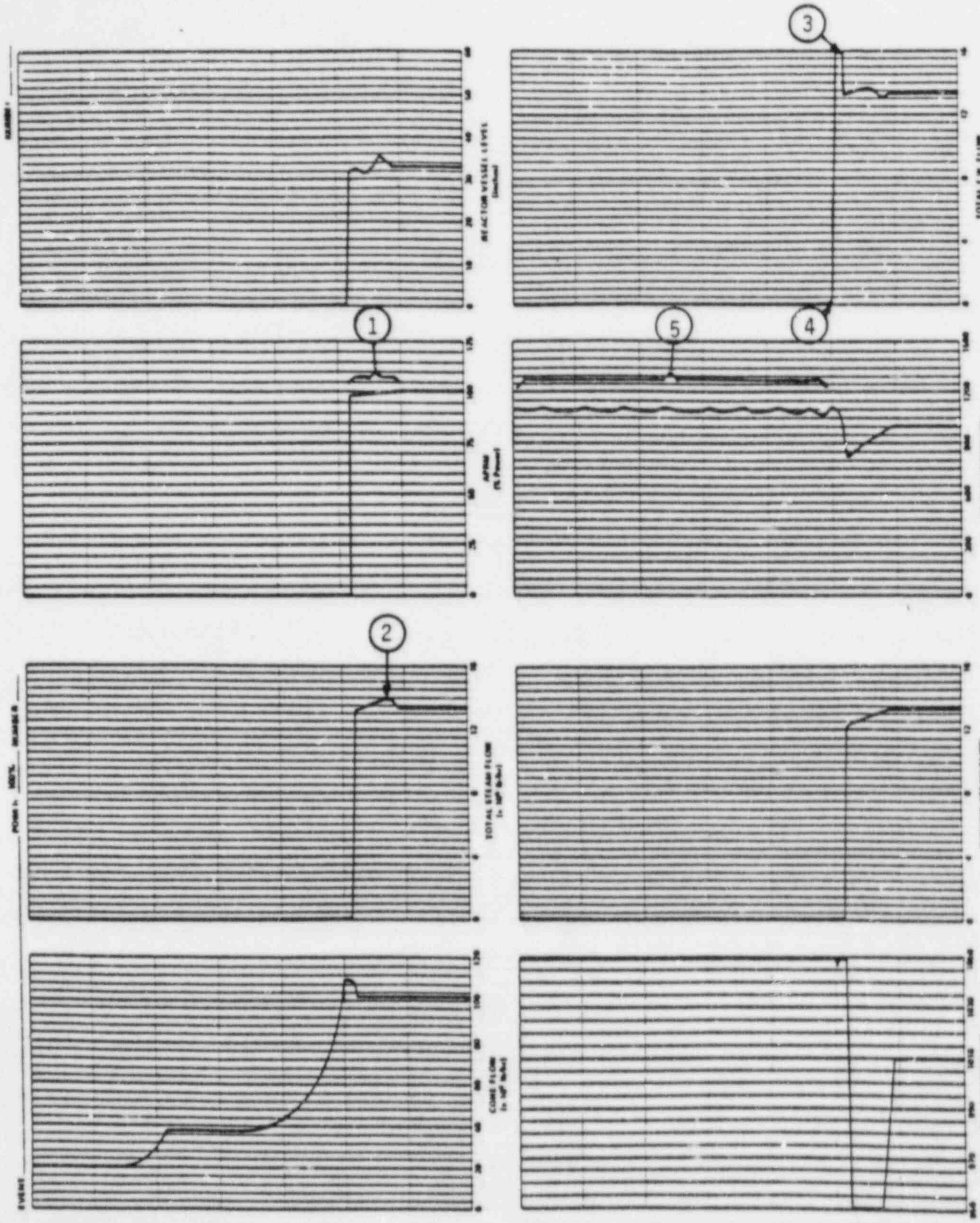
FIGURE 2 for Question 5.03



OPERATING MAP

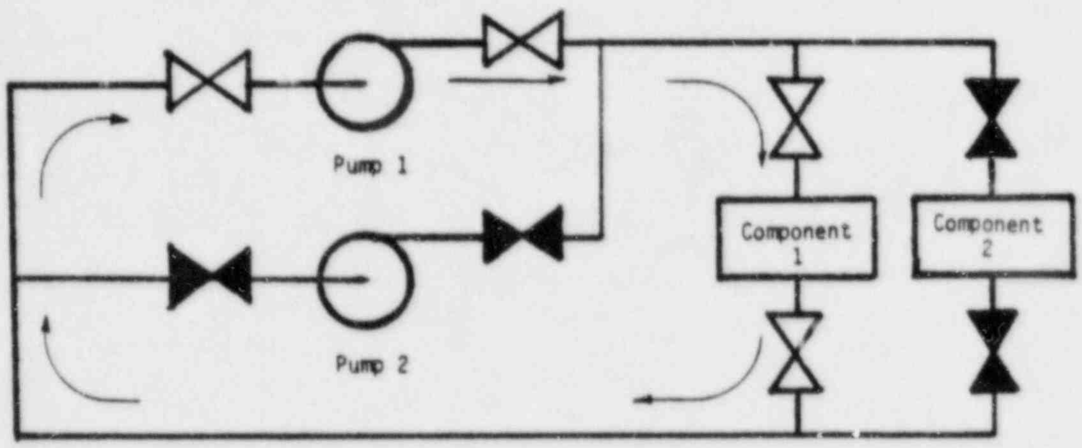
FIGURE 3 for Question 5.06

FIGURE 4 for Question 5.07

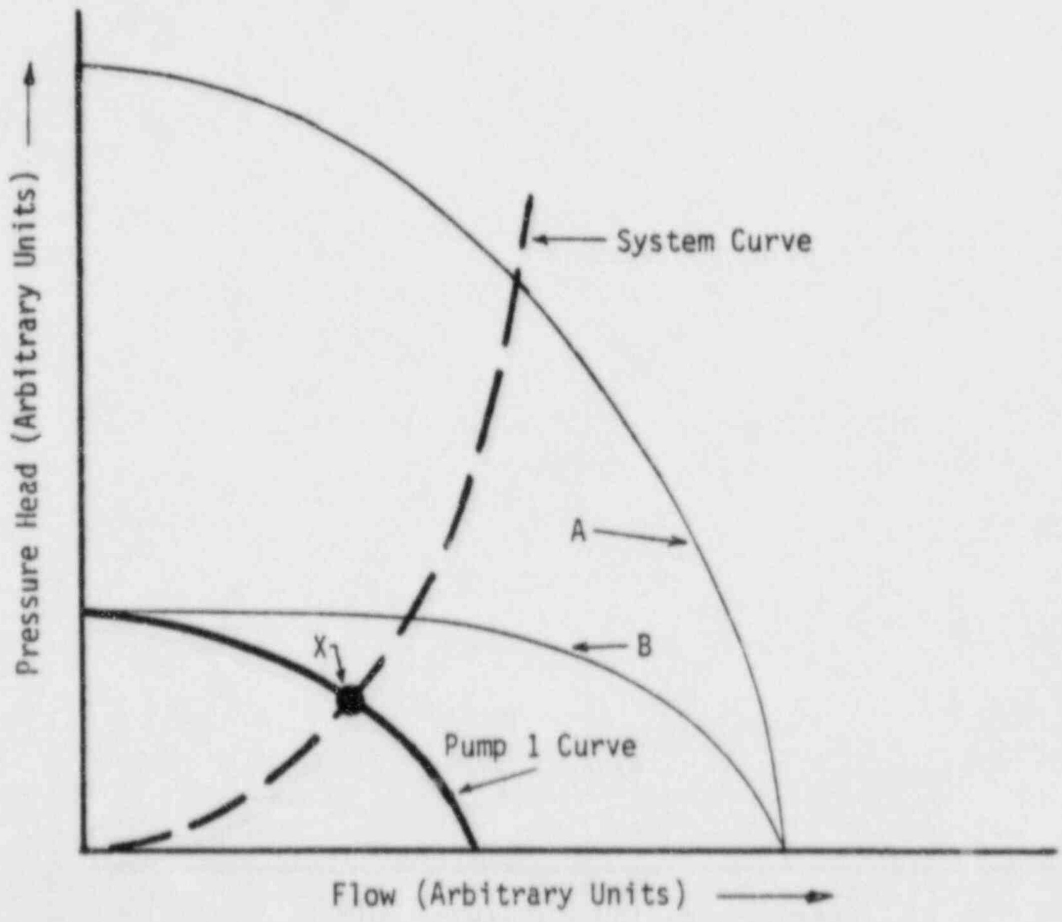


EHC PRESSURE SETPOINT DECREASE

FIGURE 5 for Question 5.10



SYSTEM



SYSTEM HEAD VS. FLOW PLOT

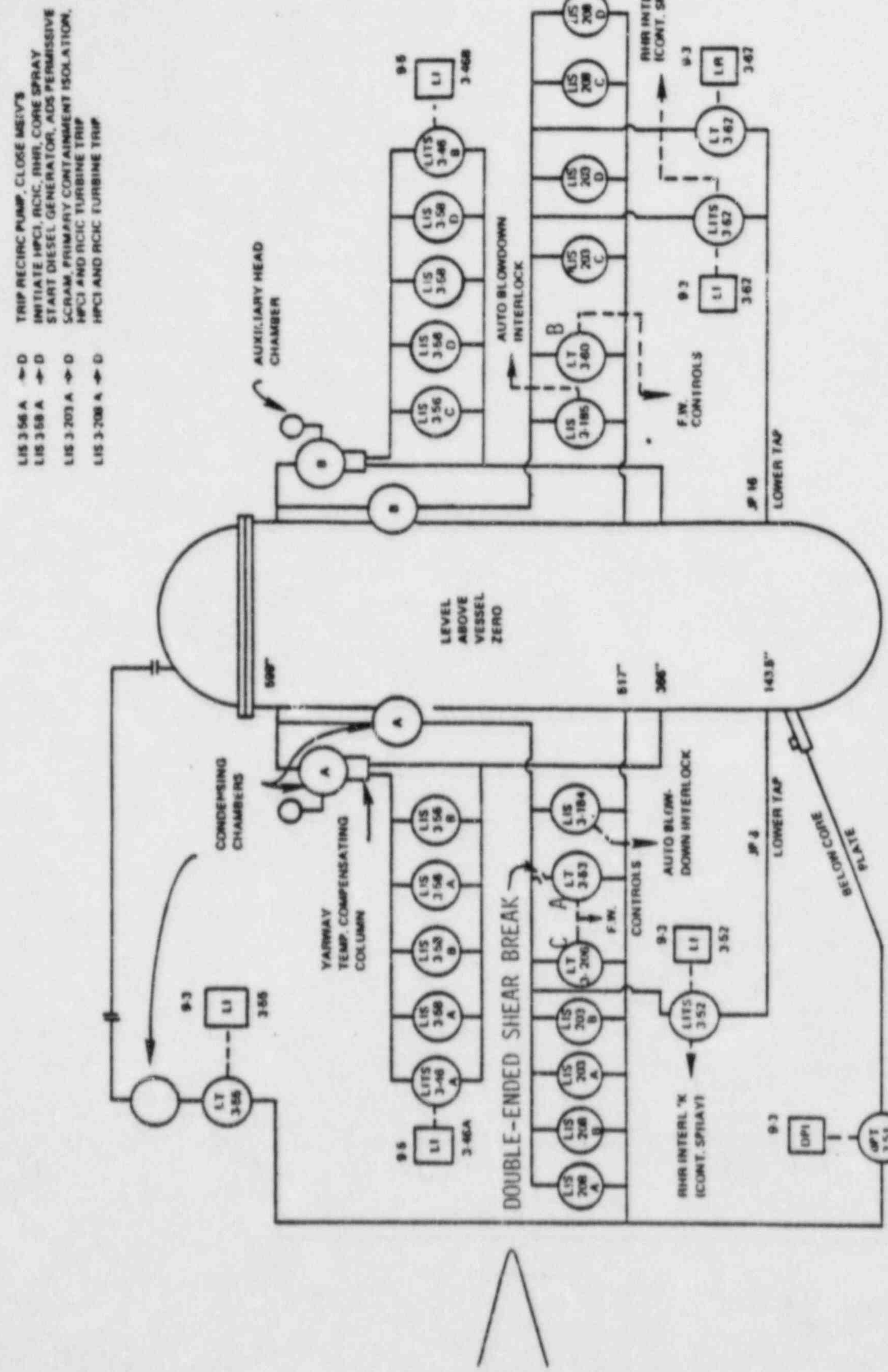


FIGURE 6 Reactor Vessel Level Instrumentation

For Question 6.02

FIGURE 7 for Question 6.08

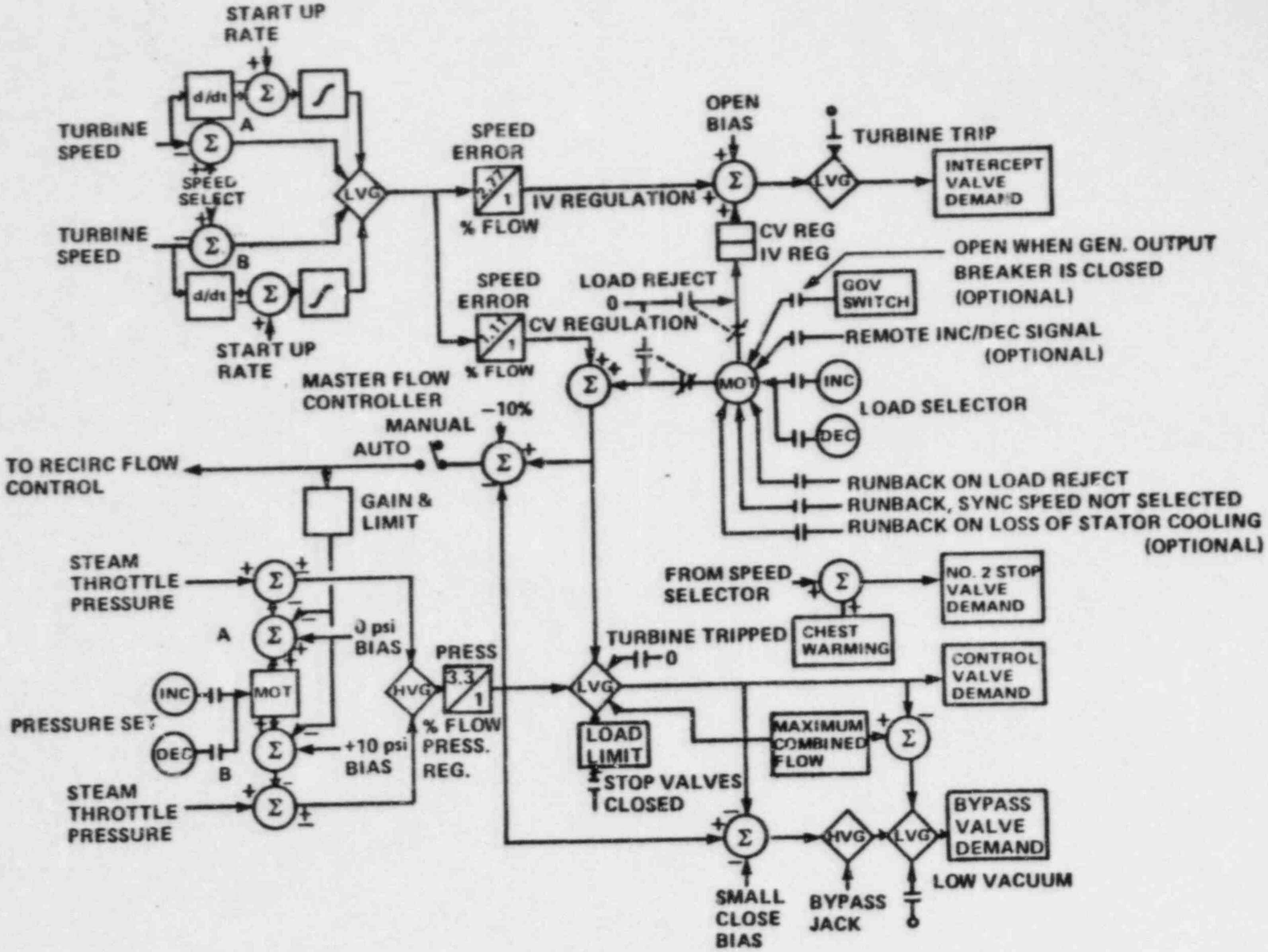
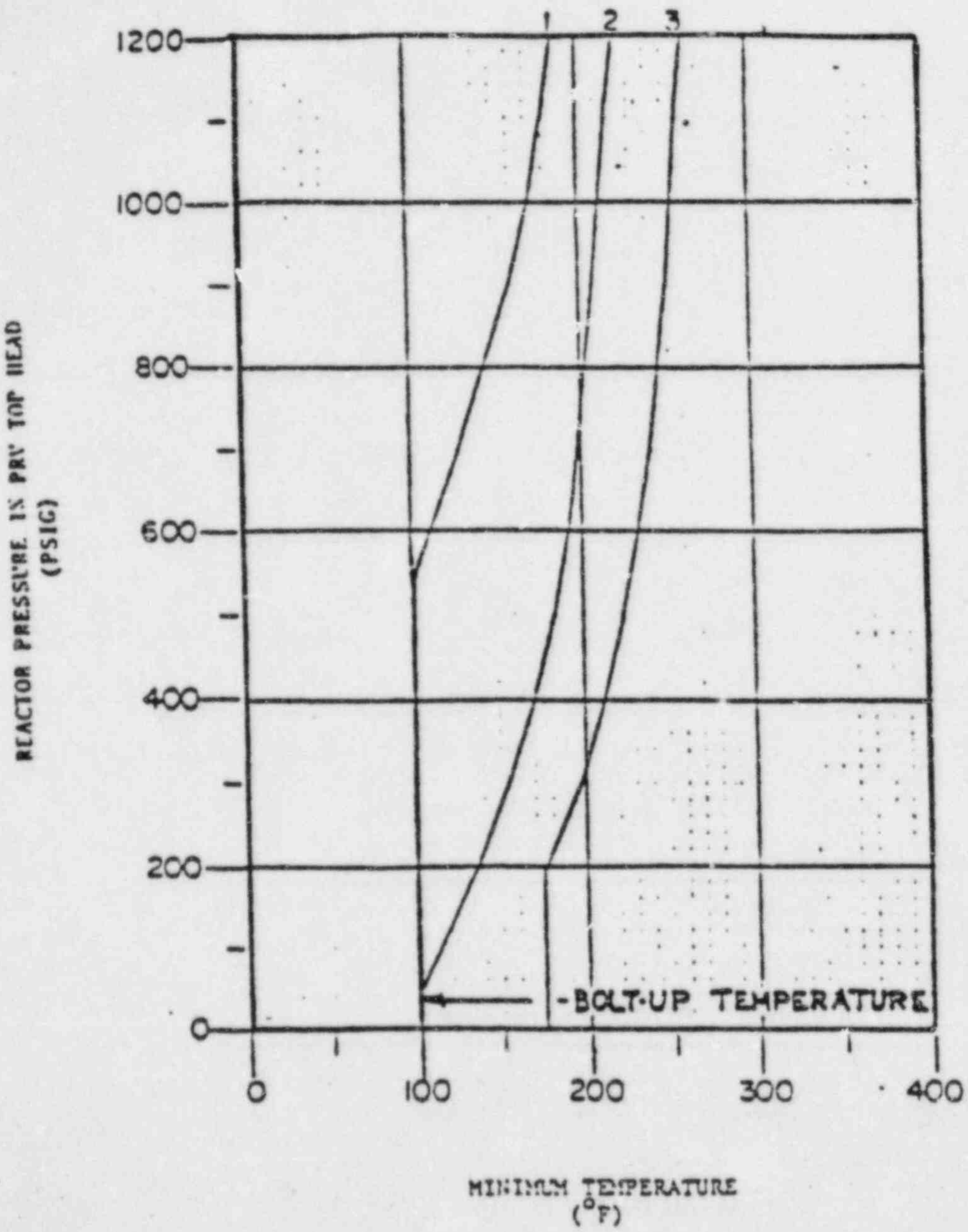


FIGURE 3.3-15 Electro-Hydraulic Control Logic

FIGURE 8 for Question 8.02 MAY 24 1982



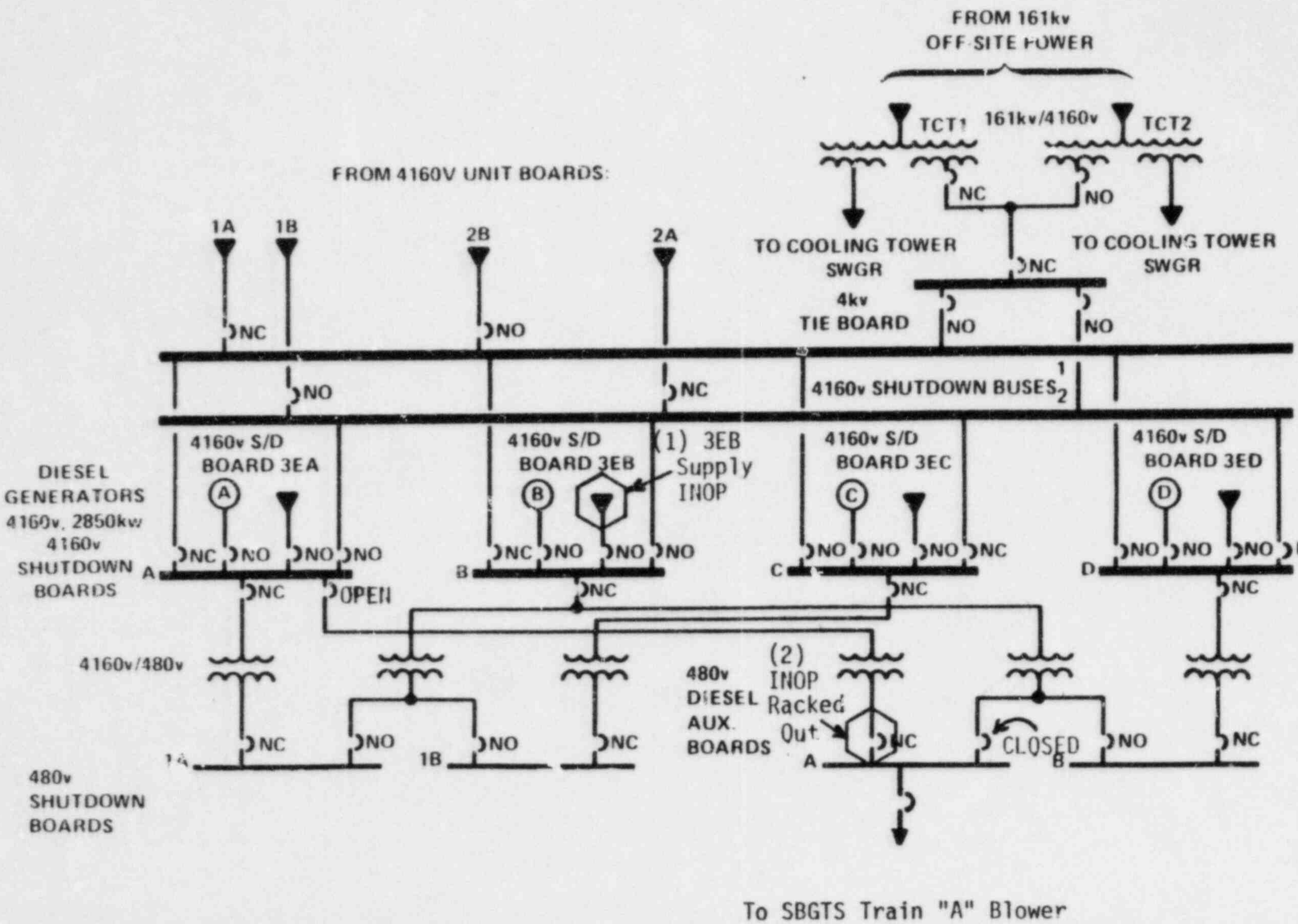
Curve #1
Minimum temperature for pressure tests such as required by Section XI.

Curve #2
Minimum temperature for mechanical heat up or cooldown following nuclear shutdown.

Curve #3
Minimum temperature for core operation (criticality) Includes additional margin required by 10CFR50 Appendix G, Par. IV A.2.C.

Notes
These curves are shifted 30°F to the right of the original set of curves to include a ΔRT_{NDT} of 30°F.

FIGURE 9 for Question 8.07



NOTE: For Question 8.07, all breakers designated NC are CLOSED and those designated NO are OPEN.

FIGURE 9 Standby Auxiliary Power Distribution

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

B. Limiting Safety System Setting (LSSS) - The limiting safety system setting are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

C. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

1. In the event a Limiting Condition for Operation and/or associated requirements cannot be satisifed because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible discovery or until the reactor is placed in an operational condition in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications. This provides actions to be taken for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. For example, if a specification calls for two systems (or subsystems) to be operable and provides for explicit requirements if one system (or subsystem) is inoperable, then if both systems (or subsystems) are inoperable the unit is to be in at least Hot Standby in 6 hours and in Cold Shutdown within the following 30 hours if the inoperable condition is not corrected.

1.0 DEFINITIONS (continued)

2. When a system, subsystem, train, component or device is determined to be inoperable solely because its onsite power source is inoperable, or solely because its offsite power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition For Operation, provided:
(1) its corresponding offsite or diesel power source is operable; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are operable, or likewise satisfy these requirements. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least Hot Standby within 6 hours, and in at least Cold Shutdown within the following 30 hours. This is not applicable if the unit is already in Cold Shutdown or Refueling. This provision describes what additional conditions must be satisfied to permit operation to continue consistent with the specifications for power sources, when an offsite or onsite power source is not operable. It specifically prohibits operation when one division is inoperable because its offsite or diesel power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason. This provision permits the requirements associated with individual systems, subsystems, trains, components or devices to be consistent with the requirements of the associated electrical power source. It allows operation to be governed by the time limit of the requirements associated with the Limiting Condition For Operation for the offsite or diesel power source, not the individual requirements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its offsite or diesel power source.

D. DELETED

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P. Secondary Containment Integrity

1. Secondary containment integrity means that the reactor building is intact and the following conditions are met:

   a) At least one door in each access opening to the turbine building, control bay and out-of-doors is closed.

   b) The standby gas treatment system is operable and can maintain 0.25 inches of water negative pressure in those areas where secondary containment integrity is stated to exist.

   c) All reactor building ventilation system automatic isolation valves are operable or deactivated in the isolated position.

2. Reactor zone secondary containment integrity means the unit reactor building is intact and the following conditions are met:

   a) At least one door between any opening to the turbine building, control bay and out-of-doors is closed.

   b) The standby gas teatment system is operable and can maintain 0.25 inches water negative pressure on the unit zone.

   c) All the unit reactor building ventilation system automatic isolation valves are operable or deactivated in the isolated position. If it is desirable for operational considerations, a reactor zone may be isolated from the other reactor zones and the refuel zone by maintaining at least one closed door in each common passageway between zones.* Reactor zone safety related features are not compromised by openings between adjacent units or refuel zone, unless it is desired to isolate a given zone.

3. Refuel zone secondary containment integrity means the refuel zone is intact and the following conditions are met:

   a) At least one door in each access opening to the out-of-doors is closed.

   b) The standby gas treatment system is operable and can maintain .25 inches water negative pressure on the refuel zone.

   c) All the refuel zone ventilation system automatic isolation valves are operable or deactivated in the isolated position. If it is desirable for operational considerations, the refuel zone may be isolated from the reactor zones by maintaining all hatches in place between refuel floor and reactor zones and at least one closed door in each access between the refuel zone and the reactor building.*

   Refuel zone safety related features are not compromised by openings between the reactor building unless it is desired to isolate a given zone.

* To effectively control zone isolation, all accesses to the affected zone will be locked or guarded to prevent uncontrolled passage to the unaffected zones.

3.7 CONTAINMENT SYSTEMS

B. Standby Gas Treatment System

1. Except as specified in Specification 3.7.B.3 below, all three trains of the standby gas treatment system and the diesel generators required for operation of such trains shall be operable at all times when secondary containment integrity is required.

4.7 CONTAINMENT SYSTEMS

B. Standby Gas Treatment System

1. At least once per year, the following conditions shall be demonstrated.

   a. Pressure drop across the combined HEPA filters and charcoal adsorber tanks is less than 6 inches of water at a flow of 9000 cfm (± 10%).

   b. The inlet heaters on each circuit are capable of an output of at least 40 kW when tested in accordance with ANSI N510-1975.

   c. Air distribution is uniform within 20% across HEPA filters and charcoal adsorbers.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7 CONTAINMENT SYSTEMS	**4.7 CONTAINMENT SYSTEMS**
2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at ≥ 10% design flow on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.	2. a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once per operating cycle or once every 18 months whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.
b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal when tested in accordance with ANSI N510-1975 (130°C, 95% R.H.).	b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
c. System shall be shown to operate within ±10% design flow.	c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

Unit 1 TS for Question 8.07

3.7 CONTAINMENT SYSTEMS	4.7 CONTAINMENT SYSTEMS
	d. Each train shall be operated with the heaters on a total of at least 10 hours every month.
	e. Test sealing of gaskets for housing doors shall be performed utilizing chemical smoke generators during each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a.
3. From and after the date that one train of the standby gas treatment system is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding 7 days unless such circuit is sooner made operable, provided that during such 7 days all active components of the other two standby gas treatment trains shall be operable.	3. a. At least once per year automatic initiation of each branch of the standby gas treatment system shall be demonstrated from each unit's controls. b. At least once per year manual operability of the bypass valve for filter cooling shall be demonstrated.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7 CONTAINMENT SYSTEMS	4.7 CONTAINMENT SYSTEMS
	c. When one train of the standby gas treatment system becomes inoperable the other two trains shall be demonstrated to be operable within 2 hours and daily thereafter.
4. If these conditions cannot be met, the reactor shall be placed in a condition for which the standby gas treatment system is not required.	

MASTER COPY

ANSWERS -- BRF -- 84/03/23 -- PERSONS, R.

ANSWER 5.01 (3.00)

a. Rod worth at notch 4 = 1 x 10 E-4 dK/K.
Rod worth at notch 16 = 9 x 10 E-4 dK/K.

Reactivity added is = (9 - 1) x 10 E-4 dK/K = 8 x 10 E-4 dK/K. (0.5)

b. See attached sketch [0.5 for axes, 1.0 for curve]. (1.5)

c. INCREASES [0.25]. The neutron leakage from the fuel cell to the volume around the control rod increases, exposing the control rod to a higher thermal neutron flux [0.75]. (1.0)

REFERENCE
BFNP Hot License Lesson Plan 1, pp 35 & 36, and Figures 35, 36, & 39.

ANSWER 5.02 (2.00)

Reactivity (dK/K) = -1.8% (0.25)

$$\text{Initial K-eff} = \frac{1}{1 - dK/K} = \frac{1}{1 + 0.018} = .982 \qquad (0.5)$$

20 = Final Count Rate (CR-f)/Initial Count Rate (CR-i), and

$$CR = \frac{S}{1 - \text{K-eff}} \qquad (0.5)$$

Substituting,

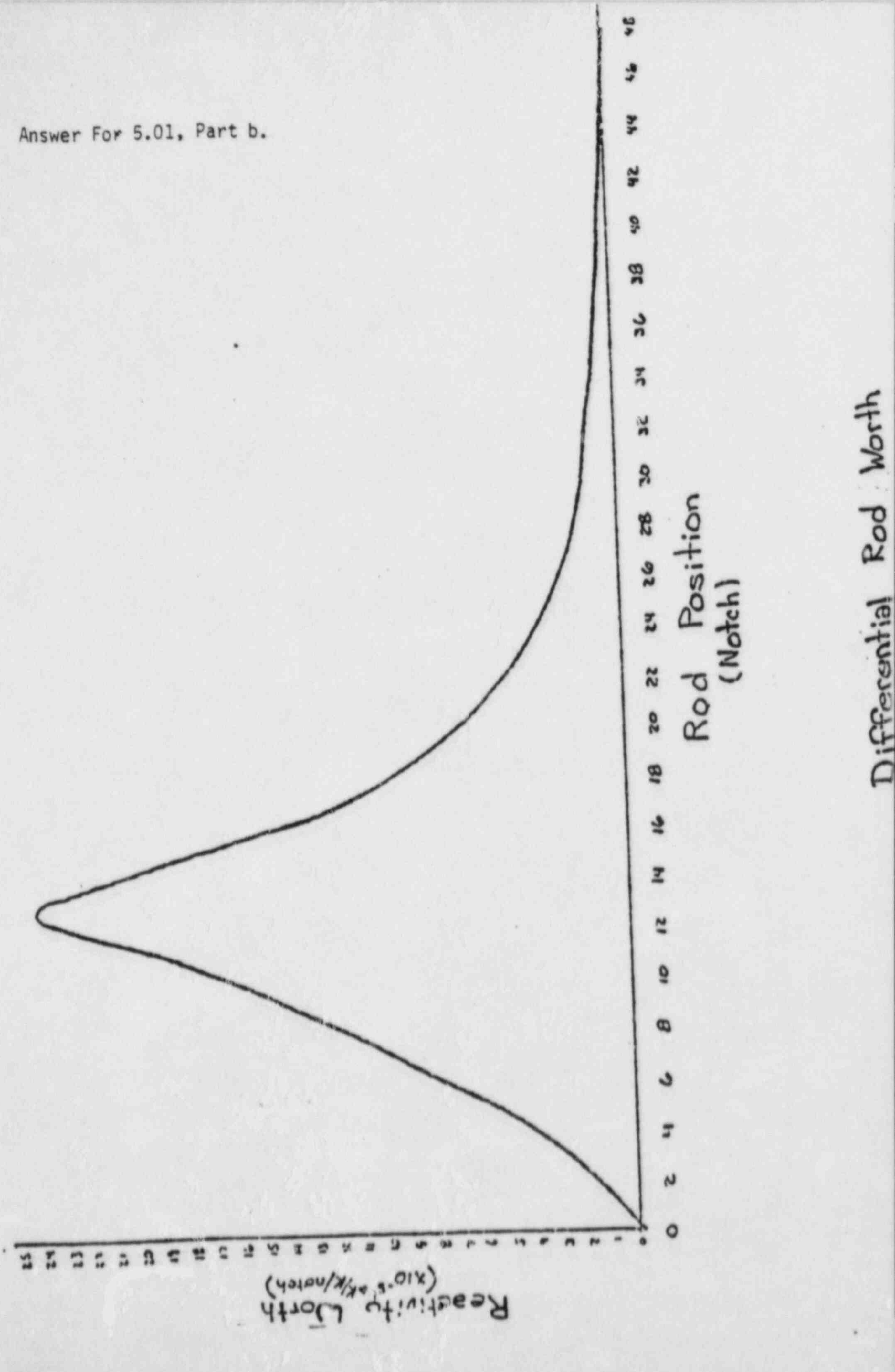
$$20 = \frac{\text{CR-f}}{\text{CR-i}} = \frac{S/1 - K[f]}{S/1 - K[i]} = \frac{1 - K[i]}{1 - K[f]} = \frac{(1 - 0.982)}{(1 - K[f]} \qquad (0.5)$$

where: S -- Source Strength
K[i] -- Initial K-eff
K[f] -- Final K-eff

Therefore,

20(1 - K[f]) = 0.018, and K[f] = 1 - (0.018/20) = .999 (0.25)

Answer For 5.01, Part b.



ANSWERS -- BRF -- 84/03/23 -- PERSONS, R.

ANSWER 5.03 (2.50)

a. Area 1 -- Decrease in core inlet subcooling due to loss of feed flow to vessel. (0.5)

Area 2 -- Reactor scram at low-level setpoint (10"). (0.5)

b. Area 3 -- HPCI and RCIC injection. (0.5)

c. Area 4 -- Recirc pump runback due to low feed flow intlk (< 20%). (0.5)

Area 5 -- Recirc pump trip at low-low level setpoint (-51.5"). (0.5)

(Setpoints NOT REQUIRED for full credit.)

REFERENCE
BFNP Transient EXY-6.

ANSWER 5.04 (1.80)

a. Increases. (Page 40 and Figure 47)
b. Increases. (Page 40 and Figure 48)
c. Increases. (Page 40)
d. ~~Increases.~~ DECREASES (Page 40 and Figure 48) — INCORRECT INTERPRETATION OF REF., DECREASES IS CORRECT. [signature] 4-4-84
e. Increases. (Page 31 and Figure 30)
f. Decreases. (Page 32 and Figure 32)
g. Increases. (Page 39 and Figure 45)
h. Decreases. (Page 39 and Figure 44)
i. Increases. (Page 39 and Figure 46)

(0.2 each)

REFERENCE
BFNP Hot License Lesson Plan 1, pages per answer.

BFNP Requal Lesson Plan, "Reactor Theory," Figure 11, p. 47 (Objective 18).

ANSWER 5.05 (3.00)

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

b. Peripheral rod worth will increase [0.3] because the highest xenon concentration will be in the center of the core [0.3] where the highest flux existed previously [0.3]. This will suppress the flux in the center of the core [0.3] and increase the flux in the area of the peripheral rods, thereby, increasing their worth [0.3]. (1.5)

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

REFERENCE
BFNP Hot License Lesson Plan 1, p. 46, and BF GOI 100-1, p. 9.

ANSWER 5.06 (1.50)

a. As reactor power is decreased from Point 7 to Point 4 core voiding decreases [0.5] resulting in a decrease in two-phase flow resistance in the core [0.5]. (1.0)

b. Critical Power, CPR, or MCPR. ACCEPTABLE FOR FULL CREDIT (0.5)

R. L. [signature] 4-11-84

REFERENCE
BFNP Hot License Lesson Plans No. 8, p. 18, & No. 22, p. 27, and BF GOI 100-1, p. 22.

ANSWER 5.07 (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)

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
BFNP Transient HXY-12.

ANSWER 5.08 (2.00)

a. Minimize fuel damage during a DBA LOCA by limiting the peak clad temperature (to < 2200 °F) -OR- limiting bundle stored energy. (0.75)

b. 2 and 3. (0.5)

c. MAPRAT = MAPLHGR/LIMLHGR -or- = MAPLHGR/MAPLHGR limit -or- = (MAPLHGR) actual/ (MAPLHGR) LCO max (0.75)

REFERENCE
BFNP Thermodynamics, p. 10.2-7, BF GOI 100-5, p. 44D, and BF Unit 1 Tech Specs, pp. 168 & 168A.

BFNP Requal Lesson Plans, "Thermal Hydraulics," pp. 32 & 33, and "Thermal Limits," p. 5.

ANSWER 5.09 (1.00)

Using the latent heat (wasted in the condenser) to heat the feedwater instead of losing it to the condenser circ water makes the cycle more efficient.

REFERENCE
BFNP Thermodynamics, p. 5.4-1.

ANSWER 5.10 (1.50)

a. System operating point. (0.5)

b. Curve B. (0.5)

c. Right. (0.5)

REFERENCE
BFNP Thermodynamics, pp. 6.1-6, 6.2-4, 6.4-6 & 7.

BFNP Requal Lesson Plan, "Pump Characteristics," Figures 4, 5, & 6.

ANSWERS -- BRF -- 84/03/23 -- PERSONS, R.

ANSWER 5.11 (1.70)

a. The sudden flow increase causes the clad surface temperature to decrease [0.33] due to more efficient convection heat transfer [0.33], decreasing the amount of nucleate boiling [0.33]. (1.0)

b. Decrease [0.2]. Due to increased clad surface temperature [0.5]. (0.7)

REFERENCE
BFNP Thermodynamics, pp. 3.4-3 & 3.4-4.

ANSWER 5.12 (1.00)

a. Decreases. (0.5)

b. Decreases. (0.5)

REFERENCE
BFNP Thermodynamics, p. 5.3-2.

ANSWER 6.01 (3.00)

a. Low reactor pressure. (0.5)

b. Draining the CST to the suppression pool. (1.0)

c. The CST being sufficiently elevated above the HPCI gravity fills the discharge piping. (1.0)

d. To prevent waterhammer. (0.5)

REFERENCE
BFNP Hot License Lesson Plan 42, pp. 18, 19, & Figure 1.

ANSWER 6.02 (3.00)

a. Full scale. (0.5)

b. {If Part (a) is answered correctly} -- The FWCS detects an erroneous high level (Ch. "A") [0.5] which sends a negative (decrease)signal to the RFP speed controller [0.5] resulting in a decreasing actual reactor water level [0.5].

-OR-

{If Part (a) is answered incorrectly} -- Everything opposite of above. (1.5)

c. 2 out of 3 coincidence (Ch. "A" & "C" failing high due to rupture) RFP and main turbine trips causing a turbine trip scram.

-OR-

Low level full scram (from LIS 203 A & B). (1.0)

REFERENCE
BFNP Hot License Lesson Plan No. 12, "Feedwater Level Control," and Transient EXY-6.

ANSWER 6.03 (2.00)

a. Depressurizes the reactor vessel in time to allow fuel cooling by CS and LPCI systems [0.5] (following a LOCA) if the other makeup systems (feedwater, CRDH, RCIC, and HPCI) fail to maintain vessel level [0.5]. (1.0)

b. Long enough so that HPCI has time to start [0.5] and yet not so long that CS and LPCI are unable to adequately cool the fuel if HPCI should fail to start [0.5]. (1.0)

REFERENCE
BFNP Hot License Lesson Plan No. 43, "ADS," pp. 4 & 5.

BFNP Requal Lesson Plan, "ADS," pp. 5 & 6, (Objective 2.b).

ANSWER 6.04 (1.00)

o Mode switch in "Run."

o Companion APRMs on scale. OR NOT DOWNSCALE (ACCEPTABLE, MEANS SAME THING) [0.5 each] 4-4-84 (1.0)

REFERENCE
BFNP Hot License Lesson Plan No. 20, "IRMs," p. 19.

ANSWER 6.05 (2.50)

a. Limits recirc pump speed such that the feedwater control system can maintain or recover reactor vessel water level upon loss of a reactor feed pump [0.75]. Limits speed when one or more RFPs are at <20% rated flow [0.375] and reactor vessel level is <+27 inches [0.375]. (1.5)

b. Pump speed will increase to 45% at which time the Master Controller low speed limiter will be limiting. (1.0)

REFERENCE
BFNP Lesson Plan No. 8, "Recirculation Flow Control," pp. 5 & 9.

ANSWER 6.06 (3.00)

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]. (water hammer and torus integrity.) (1.5)

b. Increases [0.5]. The open vacuum breaker provides a direct path to the drywell for the steam entering the SRV discharge line [1.0]. (1.5)

REFERENCE
BFNP Hot License Lesson Plan No. 9, p. 5, and NUREG/BR-005/Vol. 5, No. 4, Power Reactor Events, January 1984, p. 5, "Uncontrolled Leakage of Reactor Coolant Outside Primary Containment - Update from Vol. 5, No. 1," (event at Hatch Unit 2 on August 25, 1982).

ANSWER 6.07 (3.00)

a. Indicates possible CS pipe break inside the reactor vessel (between shroud and vessel wall -- NOT REQUIRED for full credit) [0.75] which could cause CS injection flow to go into the vessel annulus instead of on top of the core [0.75]. (1.5)

b. Indicates leakage back from the reactor through the testable check valve and the injection valve [0.75] which could cause overpressurization -OR- possible rupture of the low pressure portion of the CS discharge piping [0.75]. (1.5)

REFERENCE
BFNP Hot License Lesson Plan No. 45, p. 8 and Figure 3.

ANSWERS -- BRF -- 84/03/23 -- PERSONS, R.

PER ADDITIONAL REF. MTL BELOW,

ANSWER 6.08 (3.00)

a. The TCV's will close ~~fully~~ TO 50% STEAM FLOW. (OR 25% OPEN) [0.5]. The TCV LVG passing the MCF signal of 50% rather than the signal from the pressure controller [0.5]. (1.0)

ES A V

b. The BPV's will remain closed through the transient [0.5]. The MCF summer will send a zero signal to the BPV LVG [0.5]. (1.0)

c. Reactor power & pressure will rapidly increase following the fault [0.5]. The reactor will scram on High Flux &/OR High Pressure. Reactor pressure will be controlled by the SRV's TCVs [0.5]. (1.0)

REFERENCE
BFNP Hot License Lesson Plan No. 14, BF SMG 147-1.2, p. 34,
TRANSIENT HXY-16
~~P#6~~ -AL-4985, P.20a

ANSWER 6.09 (3.00)

1. HPCI - Loss of power to valves, pumps and Division II logic
2. CS - System II will not auto initiate if needed
3. RHR - System II will not auto initiate if needed
4. RCIC - Loss of Divison II logic
5. MSRV's - Loss of operability & ind. on some valves ADS- two valves lose normal power & auto swap to Rmov B
6. Recirc - "A" MG loss of emerg. oil pump, loss of speed control and resulting lockup of its scoop tube
7. RPS - Loss of backup scram capability
8. MSIVs - Outboard MSIV DC solenoid will de-energize.
(6 of 7 required at 0.5 each)
PCIS - Loss of power to outboard isol. valves

REFERENCE
BRF OI 57 pp. 75-78 and DC Electrical Distribution Lesson Plan.
BRF ADS Required LP, P.7
BRF PCIS LP

ANSWER 6.10 (2.00)

a. The pumps assigned to EECW will auto start [0.5]. The RBCCW system FCV's open to send EECW to the RBCCW HX's [0.5]. (1.0)

b. The RHRSW should be is at the greater pressure [0.5] this is to ensure any leakage is into, not out of a radioactive system. -OR- (will accept) to prevent an uncontrolled radioactive release to the environment [0.5]. (1.0)

REFERENCE
BFNP Hot License Lesson Plans Nos. 44 & 51.

ANSWERS -- BRF -- 84/03/23 -- PERSONS, R.

ANSWER 7.01 (3.00)

a. Manually scram the reactor. (0.5)

b. o Temperature differential of > 50°F between bottom vessel head and "A" FW sparger when moderator is > 150°F.

o Temperature differential of > 75°F between bottom vessel head and "A" FW sparger when moderator is < 150°F.

o FW sparger temperature of > 200°F on either "A" or "B" FW sparger when recirc. pumps and shudown cooling are OOS.

(2 of 3 at 1.0 each) (2.0)

c. Between 180°F and 200°F. (0.5)

REFERENCE
BF GOI 100-12, pp. 1, 7, & 14.

BFNP Requal Lesson Plan, "GOI-100-12," (Objectives 1, 3, & 4).

ANSWER 7.02 (3.00) → OR Low steam flow to last stages during S/D → Signif heating of last stage blading & inner casing [not per OI but ok for .7 partial credit]

a. Operation at low speed may result in large rotor bows being generated by rubbing with no indication of high vibration from the turbine supervisory instruments. (1.0)

R L Persons 4-9-84

→ 900-[illegible]-1415 RPM

b. CAF on Unit 2 critical speeds.[0.5] Operation at critical speed would be to operate at the point of maximum vibration.[0.5] (1.0)

c. The transfer voltmeter reads the difference between the Auto (90P) & Manual Voltage Regulator outputs.[0.5] By adjusting the manual voltage regulator (OR "70P") the meter is nulled.[0.5] (1.0)

↳ ACCEPTABLE TERM FOR MAN. VOLT. REG.

REFERENCE
BF OI-47, pp. 21-24.

BF Main Turbine LP, P.14

BF Main Gen. & Aux LP

Letter from BFNP

R L Persons 4-4-84

ANSWERS -- BRF -- 84/03/23 -- PERSONS, R.

ANSWER 7.03 (2.50)

a. Decreases
b. Decreases
c. Increases
d. Increases
e. Remains approximately the same or decreases slightly.

(0.5 each)

Will accept 'Decreases', if specified as the "other jet pump on failed riser". Per same reference. 4-4-84

REFERENCE
BF OI-68, p. 29.

ANSWER 7.04 (2.00)

a. A break large enough to result in a reactor scram on High Drywell Pressure [0.5] or Low Reactor Water Level [0.5]. (1.0)

b. Isolation of line breaks may cause reactor pressure to increase rapidly [0.5] resulting in level shrink uncovering the core [0.5]. (1.0)

REFERENCE
BF EOI-36, pp. 1 & 5.

BFNP Requal Lesson Plan, "EOI-36" (Objectives 1 & 4).

ANSWER 7.05 (2.00)

o CRD
o SLC
o RCIC
o HPCI (0.5 each)

REFERENCE
BF EOI-41, pp. 1 & 2.

BFNP Requal Lesson Plan, "EOI-41" (Objective 1).

ANSWERS -- BRF -- 84/03/23 -- PERSONS, R.

ANSWER 7.06 (3.00)

- o If scram does not occur when a scram setpoint is reached, manually scram the reactor.
- o Verify existing condition by multiple indications.
- o Verify all automatic actions have occurred. If not, place controls in manual and make corrective manipulations.
- o Trip recirc pumps.
- o Place mode switch in shutdown. Place scram discharge volume high water level bypass switch to bypass.
- o Reset scram (verify scram discharge vents and drains open) and manually scram the reactor, reset and repeat if rod motion is observed until all control rods are fully inserted.

(0.5 each)

REFERENCE
BF EOI-47, pp. 1 & 2.

BFNP Requal Lesson Plan, "EOI-47" (Objective 1).

ANSWER 7.07 (2.50)

a. To sequence the opening order per the MSRV position chart in order to minimize torus hot spots. (1.0)

b. > 95°F (U-1); > 93°F (U 2 & 3) (either temp. limit w/ applicable unit # ok for full credit) (0.5)

c. In test mode with flow to the CST. (0.5)

d. To return steam seals and SJEAs to service. (0.5)

REFERENCE
BF GOI-100-11, pp. 5-7. & BF OI 74, p. 8. (ATTACHED) [signature] 4-4-84
BF OI 64, p. 21
BFNP Requal Lesson Plan, "GOI-100-11" (Objective 4, for Part [d]).

ANSWERS -- BRF -- 84/03/23 -- PERSONS, R.

ANSWER 7.08 (1.50)

a. One shift. (0.5)

b. Plant Health Physics Supervisor, or his representative [0.5]. One week [0.5]. (1.0)

REFERENCE
BF RCI-9, p. 10.

ANSWER 7.09 (2.00)

a. o Close MSIVs.
o Trip RFPs.
o Trip main turbine.
o Trip condensate pumps.
o Trip condensate booster pumps.
o Start diesel generators. (4 of 6 at 0.25 each) (1.0)

b. Verify the affected control switch is in the desired position. (1.0)

REFERENCE
BF Emergency Plans Manual, Procedure 2.1, p. 2.

ANSWER 7.10 (2.00)

1. CHECK TR-1-1 & ACCOUSTIC MONITOR TO DETERMINE WHICH VALVE IS OPEN.

2. Operate relief valve control switch 2 or 3 times. (0.5)

3. Initiate suppression pool cooling. (0.5)

4. Change EHC pressure regulator setpoint. (0.5)

5. Once determined valve will not close and/or prior to exceeding 110 deg. F torus temperature, reduce reactor power and upon Shift Supervisors approval manually scram the reactor. (0.5)

(4 OF 5 AT 0.5 EACH)

REFERENCE
BRF OI-1, IV.C 1-5, pp. 6 & 6A.

PER REFERENCE AND FACILITY REVIEW COMMENT

4-4-84

ANSWERS -- BRF -- 84/03/23 -- PERSONS, R.

ANSWER 7.11 (1.50)

a. The maximum reading is 300 cpm which is equal to the limit of 0.05 mrad/hr. (0.75)

b. Notify Health Physics. (0.75)

REFERENCE
BF RCI-1, pp. 8 & 9.

ANSWER 8.01 (2.50)

a. Do not operate when no steam is flowing through the valve. (0.5)

b. o Proceed to the scene of the emergency.
   o Appraise situation.
   o Notify the shift engineer of the condition.
   o Verify at least two auxiliary operators are at the emergency equipment storage room for level II support (except for cooling tower auxiliary).
   o Take whatever action necessary to implement the plan of corrective action.
   o If auxiliary operators are not available, dispatch an AUO to the emergency equipment storage room for Level II support.

   [4 of 6 at 0.5 each] (2.0)

REFERENCE
BFNP SILs 45 & 73.

BFNP Requal Lesson Plan, "Discussion of Selected Section Instruction Letters," (Objectives 1 & 3).

ANSWER 8.02 (1.25)

a. Right. (0.25)

b. Due to the neutron flux from the reactor causing neutron [0.5] embrittlement of the reactor vessel metal [0.5]. (1.0)

REFERENCE
BFNP Unit 1 TS, pp. 194 & 215.

ANSWER 8.03 (3.00)

o Planned exposure during an emergency situation [0.75], 10 REM [0.25].

o To prevent serious damage to plant or hazard to personnel [0.75], 25 REM [0.25].

o To save a life [0.75], 100 REM [0.25].

REFERENCE
BF RCI-1, p. 11.

ANSWERS -- BRF -- 84/03/23 -- PERSONS, R.

BFNP Requal Lesson Plan, "Radiological Control and Safety," p. 6 (Objective 3).

ANSWER 8.04 (3.00)

a. Whenever the reactor is in the startup or run mode below 20% rated thermal power. (1.0)

b. A pattern which results in the core being on a thermal hydraulic limit (i.e., MCPR or LHGR). (1.0)

c. Both RBM channels must be operable or control rod withdrawal shall be blocked. (1.0)

REFERENCE
BF TS, pp. 123, 124, & 131.

BFNP Requal Lesson Plan, "Rod Block Monitor System," pp. 23 & 24 (Objective 6). {Applicable to Parts (b) and (c).}

ANSWER 8.05 (1.50)

Each required scram shall be initiated by its expected scram signal [0.5]. The thermal power safety limit shall be assumed to be exceeded [0.5] when scram is accomplished by other than the expected scram signal [0.5].

REFERENCE
BFNP Unit 1 TS, p. 11.

BFNP Requal Lesson Plan, "Unit 1 Tech Specs," pp. 3 & 4 (Objective 2.a.2).

ANSWER 8.06 (2.00)

The SI is to be flagged by a marker [0.~~66~~ 1.0] in the rolodex card file located on the unit operator's desk [~~0.66~~ 1.0]. ~~Each oncomming shift is required to survey the card file for any SIs that have been flagged by a marker [0.66].~~

NOT SPECIFICALLY ASKED IN QUESTION. [signature] 4-10-84

REFERENCE
BFNP SIL-29.

ANSWER 8.07 (2.50)

a. The other two trains are demonstrated to be operable daily. (0.5)

b. Five more days (for a total of 7). (0.5)

c. No [0.5]. Because even though the offsite power supply to the blower is still operable, all redundant systems (SBGTS train "C") are not operable making train "A" inop also [1.0]. (1.5)
[per TS definition 1.C.2]

REFERENCE
BF TS Definitions and pp. 236-239.

ANSWER 8.08 (3.25)

a. An evaluation shall be made to ensure that inadvertent transfer of significant amounts of containment fluids will not occur upon resetting the isolation. (1.0)

b. The Unit Operator [0.5] and Assistant Shift Engineer [0.5]. (1.0)

c. By signature [.125] On the shift turnover cover sheet. [.125] Question asked "How" (0.25)

d. Clearance, alterations. (1.0)

REFERENCE
BFSP 12.17, pp. 2 & 3.

BFNP Requal Lesson Plan, "Selected Browns Ferry Standard Practice" (Objectives 6 & 7 for Parts [a] and [d])

ANSWER 8.09 (4.00)

a. o 3,950 Volts (4-kV)
   o 440 Volts (480-V) (0.5 each) (1.0)

b. o If any unit is on the line and the safety-related boards are being fed from the unit station service transformer, the generator voltage shall be increased to bring the safety-related board voltage back up.

   o Notify WLD, THEN PLACE 161-kV capacitor banks 1 and 2 in service.

   o Verify CCWP and cooling tower lift pump capacitor banks are in service.

   o If sufficient voltage cannot be established from unit or common (reserve) station service, the onsite standby power supplies shall be initiated and used to supply the safety-related auxiliary power system until the normal supply of voltage is restored to within limits.

(3 of 4 at 1.0 each)

REFERENCE
BF SIL-56.

BFNP Requal Lesson Plan, "Core Spray System," pp. 9 & 10 (Objective 6).

ANSWER 8.10 (2.00)

a. Prevents a large uncontrolled thermal stress on the pump casing OR limits cold water reactivity addition OR limits thermal stress at the reactor vessel nozzles and bottom head region. (1.0)

b. Limits thermal stress on CRD housing to stub tube welds and reactor vessel to support skirt welds. AND THE RESULTING REACTIVITY TRANSIENT. (1.0)

4-4-84

REFERENCE
BFNP Hot License Lesson Plan No. 7, p. ~~18~~ 16, BF OI-68, pp. 2-4, and TS, p. 216.

(SEE ATTACHED COPY)

NRC Form 366A

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Browns Ferry - Unit 1	0 5 0 0 0 2 5 9	8 4	— 0 1 4 —	0 1	0 2 OF 0 2

TEXT (If more space is required, use additional NRC Form 366A's) (17)

On February 22, 1984, with unit 1 operating at 7 percent power, unit 2 in startup and unit 3 in a refueling outage, the unit 1 reactor scrammed during startup when the turbine high-pressure first stage pressure exceeded 142 psig with the turbine stop valve (SHV) closed. The A2 and B1 reactor trip actuators were picked up and initiated the scram, indicating that only the setpoints of PT-1-81B and -91A (turbine first stage pressure permissive) (JC) were reached. Subsequent review indicates that the turbine first stage pressure permissive setpoint is reduced 12 psig for instrument accuracy and the actual setpoint is 142 psig vs. 154 psig specified in the Technical Specifications.

The cause of this scram is considered to be procedural inadequacy. Operating instruction OI-47 will be revised to keep the actual turbine first stage pressure from exceeding 135 psig during turbine stop valve closure.

This scram was normal and had no safety significance. There was no operation of the high pressure coolant injection system (BJ), the reactor core isolation cooling system (BH), main steam isolation valves (ISV) or main steam relief valves (RV).

Responsible Plant Section - OP

Previous Similar Events - None

3.e (For HL Q = 2.05 & RQ Q = 2.4)

TABLE I.C-1
SYSTEM MALFUNCTIONS

MALF NO.	MALFUNCTION TITLE/RANGE/CAUSE & EFFECT	FOI NO.

105 HPCI SPEED CONTROL FAILS HIGH

TYPE:DISCRETE

CAUSE:HPCI SPEED CONTROLLER OUTPUT (R-612) FAILS MAXIMUM IN AUTO.

PLT STA:LOSS OF FEEDWATER

EFFECTS:IF THE HPCI SYSTEM IS IN OPERATION WHEN MALFUNCTION IS ACTIVATED, THE TURBINE WILL INCREASE IN SPEED TO ITS HIGH SPEED LIMIT. SYSTEM PRESSURE AND FLOW WILL REFLECT THE TURBINE SPEED INCREASE. REACTOR LEVEL WILL BEGIN TO INCREASE AT A FASTER RATE. IF THE TURBINE IS STARTED AFTER MALFUNCTION IS ACTIVATED, THE TURBINE WILL INCREASE IN SPEED AND TRIP DUE TO OVERSPEED.

THE OPERATOR MAY PLACE THE SPEED CONTROLLER IN MANUAL AND MANUALLY ADJUST TURBINE SPEED TO THE DESIRED VALUE.

REMOVAL OF MALFUNCTION WILL RETURN THE SPEED CONTROLLER OUTPUT TO NORMAL OPERATION.

REF: H-27670

3.f (For HL Q # 2.06d & RQ Q # 6.7C)

MAXIMUM COMBINED FLOW SETPOINT IS DECREASED

POWER LEVEL 100%

I/C NUMBER 16

M/F NUMBER Manually lowered MCF POT. setting at an even rate until at simulated minimum value of 50%.

1. As the MCF setting is decreased nothing happens until the setpoint becomes < 100%. Then further decrease of the setpoint causes the CV's to close while inhibiting the BPV's from opening independent of pressure. This causes pressure to increase until the scram point is reached.

2. Vessel level initially decreases due to void collapse as pressure increases. FWCS responds by increasing feedwater flow after initially decreasing feedwater flow in an initial attempt to follow steam flow. Vessel level decreases sharply after the scram due to void collapse. FWCS recovers level and then feeds as required to maintain level.

3. After the scram pressure drops rapidly, causing EHC to close the CV's faster the MCF setpoint is being decreased. When the minimum value of MCF (~50%) is reached it limits steam flow for a period until pressure is reduced by the EHC System. EHC then controls pressure at 920 psig using the CV's.

4. Core flow increases after the scram due to the change in 2 phase flow and then later decreases as the recirculation pumps run back to minimum speed at < 20% feedwater flow.

3.j (For HL @ 3.04a,
@ 6.08a,
& RO @ 3.3a)

8. Procedure (Continued)

8.7 Bypass Valves Amplifier and Combined Max Flow Limit (Continued)

z. Check the following control valve LVG inputs and verify that each is in positive saturation (voltage reading is greater than +6.00 volts).

1. Load limit op/amp A34 TP2 to TP11.
2. Control valve amp/op/amp A49 TP2 to TP11.
3. Pressure load gate op/amp A59 TP2 to TP11.
4. Combined max flow limit op/amp, A64, TP2 to TP11.

aa. Check that the output of the control valve LVG is more positive than 5 volts at A48 TP8.

ab. Check that the output of the bypass valve LVG is more positive than 5 volts at A60 TP3.

ac. Decrease the setting of the combined max flow limit pot on panel 9-7 to 0.0.

ad. The output of the CV LVG should be about 2.500 volts ± 0.200 volts at A48 TP8.

ae. The output of the DPV LVG should be 0.000 volts ± 0.002 volts at A60 TP3. If necessary, adjust R3 on A64 and/or R3 on A65 to obtain 0.000 volts ± 0.002 volts A60 TP3.

af. Set the combined max flow limit pot on panel 9-7 to 5.00 (50%).

ag. Check that the output of the CV LVG is 5.000 volts ± 0.002 volts at A48 TP8.

ah. Check that the output of the BPV LVG remains at 0.000 volts ± 0.002 volts at A60 TP3.

3.j (for HL Q 3.04 a, Q 6.08 a, & RQ Q 3.3a)

8. Procedure (Continued)

8.7 Bypass Valves Amplifier and Combined Max Flow Limit (Continued)

ai. Turn the bypass valve jack to the fully closed position.

aj. Adjust power supply connected at A54 TP2 and TP11 so that voltage measured at A48 TP8 to TP11 is 5.000 volts ± 0.002 volts. Check A60 TP3 to TP11 for 0.000 volts ± 0.002 volts.

ak. Remove power supply from A54.

al. Verify on data sheet bypass valve amplifier and combined max flow limit has been calibrated.

3.j (For Hc Q 3.04a, Q 6.08a, & RQ Q 3.3a)

VI. PRESSURE CONTROL (continued)

D. (continued)

7. Check to see that the output of the CVA is +5.0V. Then increase the BPV jack to obtain 5.0 at TP3 of BPVA. If not, adjust R1 of BPVA. Check that KT102 is closed. Some adjustment of voltage at TP2 of pressure amplifier may be necessary to obtain this setting.

8. Turn the Combined Max. Flow Limit pot to the full CW (non-controlling) position. Insert the Comb. Max. Flow Limit board and its op/amps.

9. Turn down the Combined Max. Flow Limit pot. The voltage at TP3 of BPVA should decrease to 0 volts. Also, the voltage at TP3 of the Control Valve Amp should decrease to approximately +2.5 volts. Observe that VC P005 operates to indicate comb. max. flow limit in control.

10. Return the Comb. Max. Limit to the CW position, decrease the jack to zero, ground Pin 35 of BPVA and decrease the voltage to TP2 of Pressure Amp to obtain -0.125 at Pin 29 of BPVA and observe +5.0 at TP3 of BPVA.

11. Remove all grounds and test voltages.

E. Automatic Load Following

1. Check TP2 for +0.5 if not adjust (R19). Note setting of recirculation pot ________.

2. Apply -0.5 volts to pin 06 (TP1 of Load Gate Amplifier Board).

3. Run the recirculation pots full CW. Ground pin 03.

4. Apply -5.00 volts to pin 05 (and ground TP2).

5. Check TP3 of 0.0 volts and trim R20 if necessary.

6. Increase voltage at pin 06 to -1.5 volts. Check that voltage at TP3 doesn't increase above 3.125 volts. Adjust R21 if necessary.

7. Run the recirculation pots fully CCW. Check that TP3 voltage decrease, if not, consult factory.

8. Return recirculation pot CW.

9. Remove voltage from pin 6 and ground pin 6.
Remove ground from pin 03 and TP2 apply +1.125V. Read zero at TP3.

Abnormal Operation (Continued)

B. Annunciation - "Main Steam Line Steam Leak Temperature High" 160°F (Panel 9-3). Refer to OI-64.IV.Q.

1. Check reactor building ventilation supply and exhaust fans on.

C. Relief Valve stuck open

1. An open relief valve will be indicated by one or more of the following indications:

a. Ann. of auto blowdown relief valves open.

1) Check TR 1-1 on panel 9-47 for temperature increase downstream of stuck open valve(s). Alarm setpoint is 300oF on U-1 and U-3 and 320oF on U-2.

2) Check accoustic monitor valve positions for indication of flow proportional to the amount of valve opening. Alarm setpoint is … flow.

NOTE: Upon loss of accoustic monitor and temperature indicatio on a SRV refer to TS 3.2.F.

* b. MWE drop.

* c. Mismatch between feedwater flow and main steam flow.

* d. On panel 9-3 check TR 64-161 & TR 64-162 for torus temperature increa
* Use selector buttons to locate possible hot spot in torus.

2. Operate relief valve control switch two or three times in an attempt to close the valve.

3. Initiate suppression pool cooling to maintain torus temperature below 95o and to prevent local hot spots.

4. Change pressure regulator setpoint in an effort to close the valve.

NOTE: Rapid or large changes in Rx. pressure could cause a Group 1 (MS isolation.

*Revision

3.l (For HL Q 4.01 + Q 7.10)

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT
PO Box 2000
Decatur, Alabama 35602

APR 1 0 1984

Mr. Bruce Wilson
Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, GA 30303

Dear Mr. Wilson:

As requested per telephone conversation between Mr. Fred Gunther and Mr. R. G. Jones, BFNP, the following information is being supplied:

Unit #1 Turbo Generator Critical Speed occurs between 900 and 1550 RPMs.

Unit #2 Turbo Generator Critical Speed occurs between 900 and 1415 RPMs. Specific speeds per bearing are as follows.

1 - 1360	980	6 - 1000
2 - 1415	1000	7 - 1100
3 - 1000		8 - 1060
4 - 1000		9 - 900
5 - 1030		10 - 930

This information is being supplied as per telephone conversation with TENNESSEE VALLEY AUTHORITY DIVISION ON NUCLEAR POWER, Mechanical Branch.

J E Swindell
for G. T. Jones
Power Plant Superintendent, BFNP

Enclosure

3.n (For HL Q 4.03b,
Q 7.02b,
RQ Q 4.2b,
& Q 7.2b)

An Equal Opportunity Employer

F. TURBINE AUXILIARY SYSTEMS

1. Exhaust Hood Spray System

a. Purpose

for HL Q 4.03a, Q 7.02a RO Q 4.2a, & Q 7.2a

During machine startup or at low loads, steam flow to the last few stages of the low pressure section is so low that:

1) Little cooling of the blading is provided by the steam and

2) The blades in the last 1 - 2 sections are actually pumping the steam through the machine (not designed as pumps).

This result in significant heating of the last stage blading and inner casing.

Some cooling must be provided in order to prevent distortion of radial and axial shaft to casing clearances.

Problem is worsened if there are significant non-condensables in steam.

b. Brief Description

System consists of:

1) Temperature sensors in the A & C low pressure hoods to detect high temperature conditions. The highest reading detector controls the automatic spray system.

2) An air operated, temperature controlled automatic water spray valve that controls the flow of demineralized water from the condensate system. Maximum flow is 138 gpm at 100 psig with no load on the turbine.

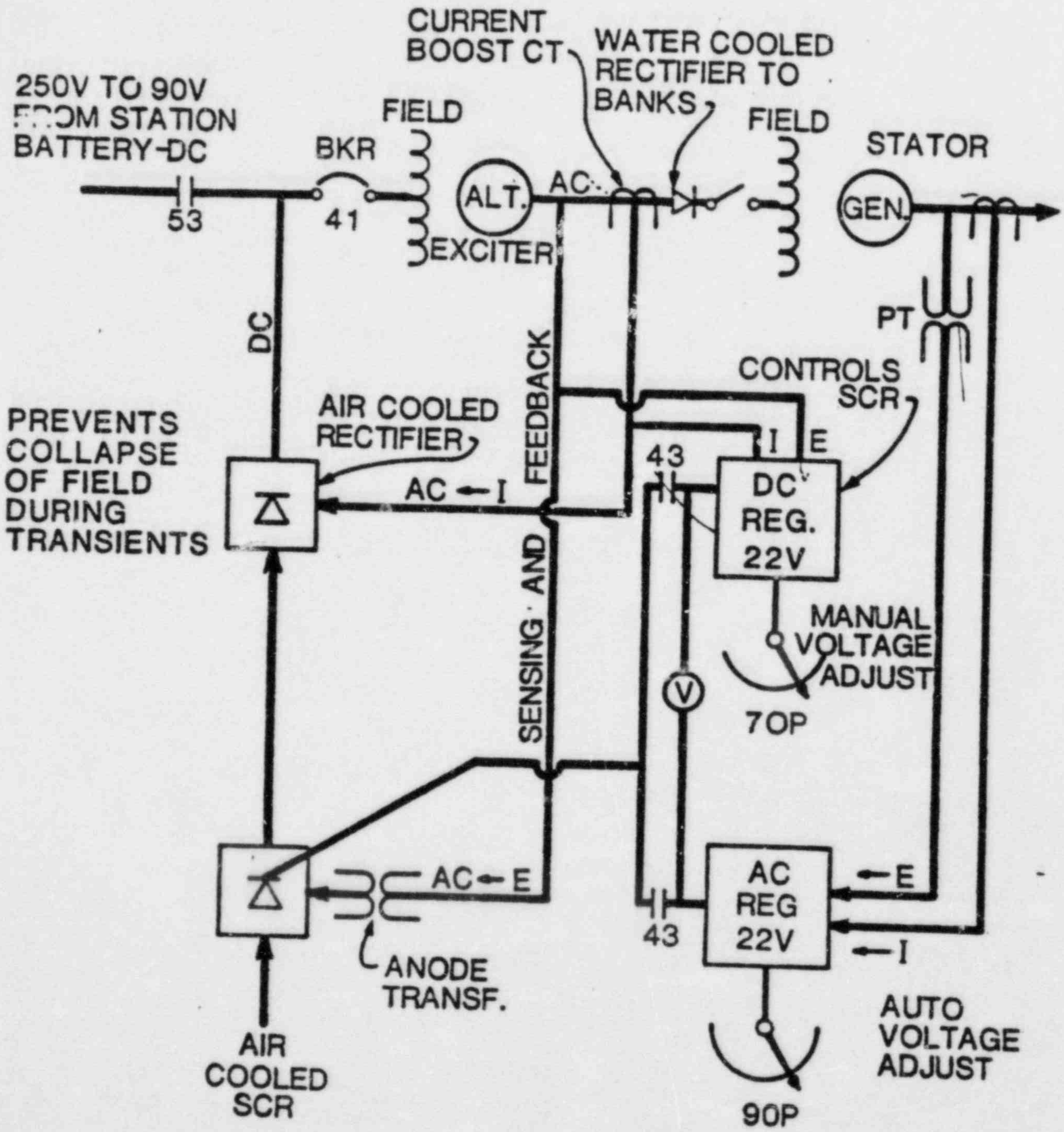
3) A motor operated bypass valve is provided for bypassing the automatic spray valve in the event of its failure.

Do not use bypass valve if exhaust hood temperature is > 135°F and turbine-generator is loaded.

4) Spray nozzles that spray down into the turbine exhaust hood, not the blading or casing, and thus provide indirect cooling of the blading.

5) Instrument air is used for system control.

EXCITATION SYSTEM



3.0 (For HL Q 4.03d,
Q7.02c,
RO Q 4.2c,
Q7.2c)

IV. Abnormal Operations (Continued)

* F. SUPPRESSION CHAMBER WATER TEMPERATURE ABNORMAL (Continued)

* 3. SUPPRESSION CHAMBER WATER TEMPERATURE HIGH (93°F)(U 2&3)

* a. See IV.F.1.A-g

G. SUPPRESSION CHAMBER VACUUM RELIEF VALVE OPEN

1. Check torus pressure (PT-64-50, panel 9-3).

2. Check temperature (TI-64-161, TR-64-161, TI-64-162, TR-64-162).

3. Close isolation valve FCV-64-20 or -21 if torus pressure is not .5 psi less than reactor building pressure (refer to Tech. Spec. 3.7.A).

H. DRYWELL LEAK DETECTION RADIATION HIGH

NOTE: Upon receipt of alarm, immediate action will be taken to confirm the alarm and assess the possibility of increased leakage, notify shift engineer.

1. Have lab take sample for test and log results in daily journal.

2. Scan drywell with television for source of radiation.

3. Check drywell sumps flow (OI-40).

I. DRYWELL LEAK DETECTION RADIATION DOWNSCALE

1. Have instrument checked by instrument section.

2. Check isolation valves not closed.

J. PERSONNEL AIR LOCK OPEN - Have air lock checked.

K. TORUS/DRYWELL ISOLATION VALVES AUTO CLOSURE BYPASSED - Check key lock handswitch, panel 9-5, 9-54, and 9-55.

*Revision

3p (for HL Q 4.09b, 7.07b, RU Q 4.7b & 7.7b)

JUL 29 1983

III. Operating Instructions (Continued)

E. Initiation of Primary Containment Cooling System

CAUTION: IF RHR SEAL WATER TEMPERATURE IS $\geq$ 160°F, THE CONTAINMENT COOLING MODE OF RHR IS CONSIDERED TO BE INOPERABLE.

1. The containment cooling mode of the RHR system will be placed in service whenever necessary to maintain pressure suppression pool
* temperature $\leq$ 93°F or whenever necessary to limit primary containment pressure and temperature (refer to III.D.7).

2. Start an RHR pump. Note minimum flow valve OPEN.

3. Start corresponding RHRSWP.

4. Position FCV on discharge of heat exchanger to adjust cooling water flow to approximately 3,000 gpm.

   NOTE: Cooling water flow may be decreased or increased to a maximum of 4,500 gpm, as the condition dictates.

5. For suppression pool cooling:

   NOTE: Containment spray inboard isolation valve (System I - FCV-74-61, System II - FCV-74-75) must be fully CLOSED.

   NOTE: During suppression pool cooling operation, maintain RHR pump flow range for single pump 7,000-10,000 gpm and for two (2) pumps 14,000-20,000 gpm.

   a. OPEN suppression pool spray and recirculation valve (System I - FCV-74-57, System II - FCV-74-71).

   b. OPEN suppression pool recirculation and test isolation valve (System I - FCV-74-59, System II - FCV-74-73).

3. p (for HL Q4.09b, Q7.07b, RO Q4.7b, + Q7.7b)

*Revision

Lesson Outline	Instructor Notes

3. The two ADS valves fed from the 250V Rx MOV Board A have automatic transfer to an alternate source on loss of 250V DC to their respective control circuits.

For Q HL 2.03
6.09
RQ 2.2
6.5

E. OPERATIONAL SUMMARY

1. System operation during a small liquid or steam line break.

   a. Assuming concurrent loss of feedwater flow and failure of the HPCI system, a small break results in a loss of inventory without a significant reduction in reactor pressure.

      (1) If a small (< 0.1 ft^2) liquid or steam line breaks, inventory will begin to flow into the drywell.

      (2) As inventory decreases, reactor pressure will tend to decrease.

      (3) Since the reactor system is saturated, a reduction in pressure causes a portion of the coolant to flash to steam, which tends to maintain system pressure.

      (4) In addition, heat from the fuel, vessel internals, and the vessel itself will be added to the coolant. This also tends to maintain system pressure.

      (5) The net result is a reduction of coolant inventory (reactor water level) without a significant reduction in reactor pressure.

      (6) Assistnace from either HPCI or ADS is required to prevent possible uncovering of fuel.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6 PRIMARY SYSTEM BOUNDARY	4.6 PRIMARY SYSTEM BOUNDARY
Applicability	Applicability
Applies to the operating status of the reactor coolant system.	Applies to the periodic examination and testing requirements for the reactor coolant system.
Objective	Objective
To assure the integrity and safe operation of the reactor coolant system.	To determine the condition of the reactor coolant system and the operation of the safety devices related to it.
Specification	Specification
A. Thermal and Pressurization Limitations	A. Thermal and Pressurization Limitations
1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100° F/hr when averaged over a one-hour period.	1. During heatups and cooldowns, the following parameters shall be recorded and reactor coolant temperature determined at 15-minute intervals until 3 successive readings at each given location are within 5° F.
(for HL Q 4.04a; & RQ Q 7.3a; Q 4.3a)	a. Steam Dome Pressure (Convert to upper vessel region temperature)
	b. Reactor bottom drain temperature
	c. Recirculation loops A and B
	d. Reactor vessel bottom head temperature
	e. Reactor vessel shell adjacent to shell flange
2. During all operations with a critical core, other than for low level physics tests, except when the vessel is vented, the reactor vessel shell and fluid temperatures shall be at or above the temperature of curve #3 of figure 3.6-1.	2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange, shall be recorded at least every 15 minutes during inservice hydrostatic or leak testing when the vessel pressure is > 312 psig.

BF Hot License Recirc System L.P. #7

For SRO Answer 8.10b

3. w

a) Potentially high differential temperature can exist in the vessel bottom head region due to the CRD cooling water, injected at 70 - 90°F

b) It is desired to keep the ΔT between the saturation temperature and the water from the vessel bottom head drain drain to the cleanup system at a value <145°F and the Temp. between an idle and an operating recirculation loop at a value <50°F

2) To accomplish the above:

a) The minimum recirc pump speed is procedurally limited to 28% even though the fluid coupler could operate down to 20%

b) The vessel bottom drain line was connected to the cleanup system for accurate temperature indication in the bottom head region.

(1) Drain valve has a drilled disc to allow flow at all times and prevent stagnation

3) A ΔT between saturation temperature and the bottom head temperature is not limiting in itself. The stresses occur when starting an idle pump or increasing flow. Hot water now sweeps out the cold, producing an uncontrolled heatup.

a) Regions of primary concern are:

(1) CRD housing to stub tube welds

(2) RPV to RPV skirt welds

b) In addition, the cold water is swept up and through the core, producing a reactivity transient

b. Maximum Speed Operation

1) The recirculation pumps are sized and designed for pumping reactor water at rated conditions, i.e.: 546°F

2) Rated core mass flow 102.5×10^6 lb/hr, can be achieved at lower temperatures. However, current limits on the MG set drive to motor and/or generator will be encountered at approximately 50 HZ if pumping cold water

Operating Instructions (Continued)

C. Startup (Establishing Condenser Vacuum) (Continued)

7. Catalytic recombiners and charcoal adsorbers. (Continued)

c. To place A or B off-gas system in service: (Continued)

6) Place HS-66-113 in the AUTO position.

NOTE: The preferred method of operation is in the parallel mode.

a) Open train 2 bypass (HS-66-118).
b) Open train 1 bypass (HS-66-117).
c) Close train 1 discharge (HS-66-116).

7) With HS-66-113 in the AUTO position and HS-66-117 in the OPEN position, a high radiation in either of the post treatment process radiation monitoring channels will automatically open FCV-66-113A, close FCV-66-113B, and open FCV-66-117, thus routing parallel gas flow through the adsorber beds. TREAT position of the mode switch places the absorbers in service, in parallel manually opening FCV-66-113A, closing FCV-66-113B, and opening FCV-66-117.

RQ #2.5b & 6.4b

CAUTION: DURING POWER OPERATION, DO NOT CHANGE CHARCOAL BED ALIGNMENT. ANY CHANGE IN OFF-GAS FLOW RATES OR DIRECTIONS DISTURBS BED EQUILIBRIUM. AS A DIRECT RESULT OF ANY MAJOR LOSS OF BED EQUILIBRIUM, INCREASED AMOUNTS OF ACTIVITY ARE RELEASED UNTIL BED EQUILIBRIUM IS REESTABLISHED (8 TO 12 DAYS).

8) Maintain the standby catalytic recombiner temperature 275°-360°F.

9) If recombiner temperature is < 275°F:

a) Check cal-rod heater switches in the ON position with power supply on (HS-66-76 and -90).