ENCLOSURE 1

EXAMINATION REPORT - 50-325/OL-85-01

Facility Licensee: Carolina Power and Light Company
411 Fayetteville Street
Raleigh, NC 27602

Facility Name: Brunswick Steam Electric Plant

Facility Docket Nos.: 50-324 and 50-325

Written, oral and simulator examinations were administered at Brunswick Steam Electric Plant near Southport, North Carolina.

Chief Examiner: John F. Munro — Date Signed: 6/11/85

Approved by: Bruce A. Wilson, Section Chief — Date Signed: 6/11/85

Summary:

Examinations on May 21 - 22, 1985

One complete examination (oral, simulator, and written) was administered to an Instructor Certification candidate who passed. Two SRO candidates were administered written reexaminations; both candidates passed.

REPORT DETAILS

1. Facility Employees Contacted:

   *P. Hopkins, Director - Site Training
   *S. Morgan, Senior Specialist - Operator Training
   D. Shaw, Senior Specialist - Operator Training
   M. Magill, Reactor Operator - Operations
   L. Dunlap, Contractor (RTS) Instructor

   *Attended Exit Meeting

2. Examiners:

   *J. Munro, NRC

   *Chief Examiner

3. Examination Review Meeting

   At the conclusion of the written examinations, the examiners met with S. Morgan, and D. Shaw to review the written examination and answer key. The following comment was made by the facility reviewers:

   a. SRO Exam

      1. Question 5.14(d) - The answer key states "Remain the Same." Fuel centerline temperature will increase slightly due to a loss of (some) feedwater heating (extraction steam loss).

         NRC Resolution:

         The Brunswick Steam Electric Plant Simulator confirms the accuracy of the above statement during a simulation of question conditions. The answer key has been changed accordingly.

4. Exit Meeting

   At the conclusion of the site visit the examiners met with representatives of the plant staff to discuss the results of the examination.

   There were no generic weaknesses (greater than 75 percent of candidates giving incorrect answers to one examination topic) noted during the oral examination.

   The cooperation given to the examiner and the effort to ensure an atmosphere in the control room conducive to oral examinations was also noted and appreciated.

   The licensee did not identify as proprietary any of the material provided to or reviewed by the examiners.

TABLE ES-201-7

POWER PLANT EXAMINATION RESULTS SUMMARY SHEET

ENCLOSURE 2

FACILITY	Brunswick 1 & 2
DATE OF WRITTEN	May 21, 1985
DATE OF ORALS	May 22, 1985
DATE OF SIMULATOR	May 22, 1985
EXAMINERS	J. Munro
TYPE OF EXAM	[illegible] (HOT)

Overall Results	Total No.	Passed No.	Passed %	Failed No.	Failed %
Senior Operator	2	2	100	0	0
Reactor Operator	-	-	-	-	-
Instructor Certification	1	1	100	0	0

1. REACTOR OPERATOR
2. SENIOR REACTOR OPERATOR INSTANT
3. SENIOR REACTOR OPERATOR UPGRADE
4. REACTOR OPERATOR RETAKE
5. SENIOR REACTOR OPERATOR RETAKE
6. INSTRUCTOR CERTIFICATION

NAME	DOCKET NO.	TYPE CODE	1	2	3	4	TOTAL	5	6	7	8	TOTAL	Examination Results RO W	RO O	RO S	Examiner's Initials SRO W	SRO O	SRO S
Sosler, L.	55-20226	6						84.6	86.8	92.5	75.9	85.04				P JM	P JM	P JM
Reinsburrow, J.	55-8280	5						93.8	86.6	95.3	83.8	89.8				P JM	W	W
Poulk, R.	55-6661	5						86.2	78.5	84.3	81.3	82.5				P JM	W	W

ENCLOSURE 3

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

Facility: Brunswick
Reactor Type: BWR
Date Administered: 5/21/85
Examiner: Munro, J., Brockman, K
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
26	24.4			5. Theory of Nuclear Power Plant Operation, Fluids, and Thermodynamics
28	26.3			6. Plant Systems Design, Control, and Instrumentation
26.5	24.9			7. Procedures - Normal, Abnormal, Emergency, and Radiological Control
26	24.4			8. Administrative Procedures, Conditions, and Limitations
106.5		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 The fission process in a commercial reactor requires the neutrons that are "born" by fission to be "thermalized." The interaction in the reactor core which is most efficient in thermalizing neutrons for fission occurs with the ... (CHOOSE ONE) (1.0)

a. OXYGEN atoms in the water molecules
b. BORON atoms in the control rods
c. ZIRCONIUM atoms in the fuel cladding
d. HYDROGEN atoms in the water molecules

5.2 Which of the following statements best describes the operating characteristics of an LPRM detector?

NOTE: Consider detector operation only. (1.0)

a. Depletion of the detector's Uranium coating causes both the neutron and the gamma sensitivity to DECREASE with detector age; the resulting neutron to gamma signal ratio remains relatively CONSTANT.

b. Since the detector functions as an ionization chamber and the Argon gas pressure remains relatively CONSTANT, BOTH the neutron and the gamma sensitivity, as well as the neutron to gamma signal ratio, remain relatively CONTANT as the detector ages.

c. Depletion of the detector's Uranium coating causes neutron sensitivity to DECREASE, but has an INSIGNIFICANT effect on gamma sensitivity; this results in a neutron to gamma signal ratio DECREASE as the detector ages.

d. Depletion of the detector's Uranium coating has an insignificant effect on neutron sensitivity, but causes gamma sensitivity to DECREASE; this results in a neutron to gamma signal ratio INCREASE as the detector ages.

5.3 As part of the scram procedure, the operator is directed to insert the SRMs and IRMs.

a. Following a severe LOCA, briefly EXPLAIN how these systems could be used to detect gross core damage (deformation)? (1.0)

b. Briefly EXPLAIN how these systems could be used to provide a crude indication of water level if level could not be confirmed by normal instrumentation. (1.0)

5.4 With respect to the reactivity worth of control rods: A Control Rod will experience its greatest TOTAL worth when ... (CHOOSE ONE) (1.0)

a. ...it is INSERTED individually and all other rods are WITHDRAWN.

b. ...it is INSERTED individually and all other rods are INSERTED.

c. ...it is WITHDRAWN individually and all other rods are INSERTED.

d. ...it is WITHDRAWN individually and all other rods are WITHDRAWN.

5.5 Concerning the behavior of Samarium-149, which one of the following statements is TRUE? (1.0)

a. Once equilibrium Samarium is established, Samarium reactivity does not change regardless of power level changes.

b. 50% equilib-ium Samarium reactivity is equal to 100% equilibrium Samarium reactivity.

c. Samarium is only removed by radioactive decay.

d. Samarium is produced by the decay of Iodine.

5.6 a. After a reactor Scram from power, the shortest STABLE period possible is - 80 seconds. Briefly EXPLAIN why. (1.0)

b. Is the INITIAL period IMMEDIATELY following the above described Scram SHORTER or LONGER than the - 80 seconds. Briefly EXPLAIN. (1.0)

5.7 Attached Figure 152 shows a basic closed loop fluid system with its head vs. flow plot (BOLD LINES). The two pumps are identifical, variable speed, radial, centrifugal pumps. Pump 1 is initially operating at one-half speed to supply flow to component 1, as shown.

a. Component 2 is placed into service, thereby increasing the system heat load. Would total power consumption be less by ...(CHOOSE ONE)

(1) Doubling the speed of Pump 1
(2) Starting Pump 2 at one-half speed (0.5)

b. Which Pump Curve - A, B, C, or D - most accurately shows BOTH PUMPS (Both Running at one-half speed) operating to supply the system flow? (0.5)

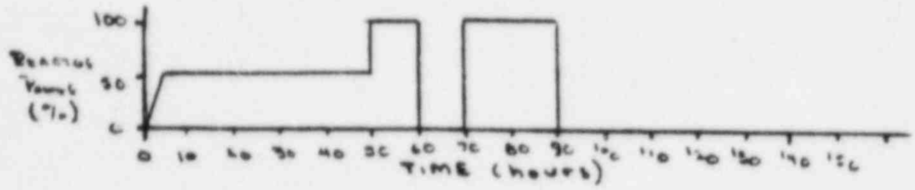
c. With only Pump 1 operating at one-half speed and Component 1 in Service - If component 2 were throttled open from its initial position, would the system flow INCREASE, DECREASE, or REMAIN THE SAME? (COMPONENTS 1 & 2 ARE IDENTICAL!) (0.5)

d. Given one operating pump, that is changed to a POSITIVE DISPLACEMENT pump. Is the correct Pump Curve to reflect this Curve A, B, C, or D? (0.5)

5.8 Which of the following describes the changes to the steam that occurs between the inlet and the outlet of a REAL turbine? (1.0)

a. Enthalpy DECREASES, Entropy DECREASES, Quality DECREASES
b. Enthalpy INCREASES, Entropy INCREASES, Quality INCREASES
c. Enthalpy CONSTANT, Entropy DECREASES, Quality DECREASES
d. Enthalpy DECREASES, Entropy INCREASES, Quality DECREASES

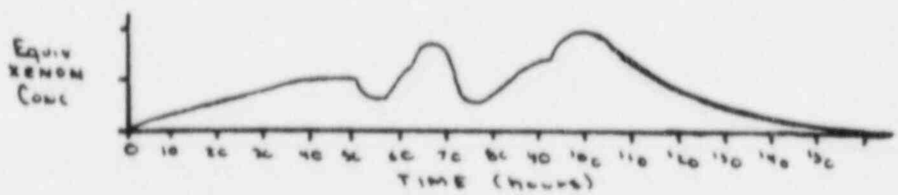
5.9 Given the following Power History:



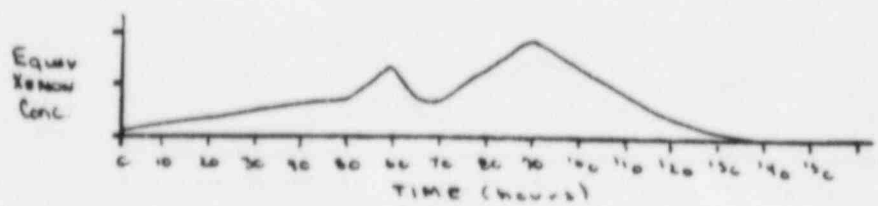
Select the most accurate curve displaying the expected XENON transient. (1.0)

a.

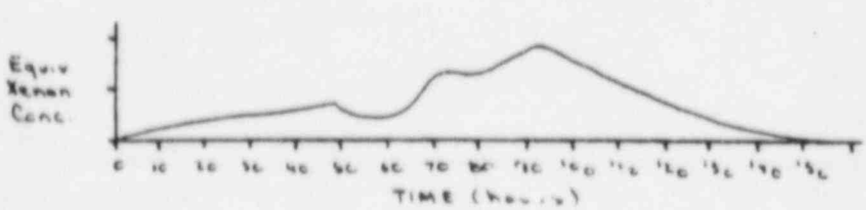
b.



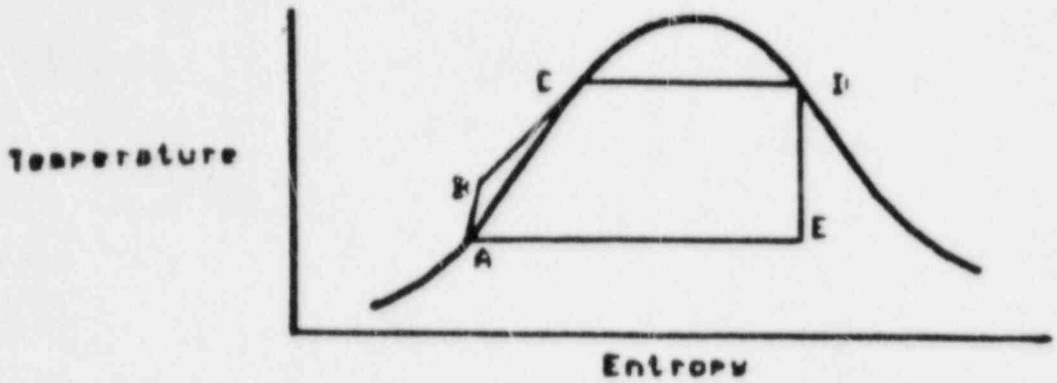
c.



d.



5.10 The BWR is designed to operate like the RANKINE VAPOR CYCLE, shown below. Select the statement which is NOT TRUE, as applied to the REAL BWR cycle. (1.0)



a. Increasing condenser vacuum (25" changed to 29") INCREASES cycle efficiency.

b. Increasing Condensate Depression, which is required for proper plant equipment performance, INCREASES overall thermodynamic efficiency.

c. Feedwater Heating INCREASES overall thermodynamic efficiency.

d. Feedwater pump pressure increase causes the feedwater to be FURTHER from saturation conditions.

5.11 A steam condenser must remove more heat energy to condense... (CHOOSE ONE) (1.0)

a. ...five (5) pounds of steam at 15 psia

b. ...fifteen (15) pounds of steam at 1000 psia

c. ...twenty-five (25) pounds of steam at 2000 psia

d. ...fifty (50) pounds of steam at 3000 psia

5.12 STATE for which condition the reactivity coefficient contribution would be MORE NEGATIVE. Briefly EXPLAIN your choice. (1.0)

Moderator temperature coefficient for a 75% CONTROL ROD DENSITY.

-OR-

Moderator temperature coefficient for a 25% CONTROL ROD DENSITY.

5.13 Concerning General Electr c's Preconditioning Interim Operating Management Recommendatior (PCIOMR):

a. Starting with the .el at a threshold of 11.0 kw/ft, a maximum ramp increase is begun at time 0000 and the final desired power of 13.0 kw/ft is achieved at 2000. At this time, the required soak is performed FOR 10 MINUTES, at which time the load dispatcher directs a power reduction that takes nodal power down to 12.5 kw/ft. SELECT the valid preconditioned value for this node. (1.0)

ASSUME THE MAXIMUM RAMP RATE IS .10 Kw/ft/hr

1) 11.0 kw/ft
2) 11.8 kw/ft
3) 12.5 kw/ft
4) 13.0 kw/ft

b. After seven hours, the Load Dispatcher directs a return to full power. SELECT the minimum time required to get back to 13.0 kw/ft, given the above ramp rate. (1.0)

1) Immediate (Raise to 13.0 kw/ft, w/o restrictions)
2) 5 hours
3) 12 hours
4) 20 hours

5.14 STATE how fuel pin centerline temperature will change (INCREASE, DECREASE, or REMAIN THE SAME) with each of the following conditions.

a. A Feedwater Heater is removed from service causing a reduction in feedwater temperature of 10 deg F. (0.5)

b. The Pressure Set on EHC is raised by 10 psig. (0.5)

c. A fuel pellet experiences "swell". (Gap with clad decreases) (0.5)

d. A HPCI full flow surveillance is conducted. (0.5)

5.15 Which of the following valves would cause the greatest energy loss in the fluid going through it (Highest dP)? (1.0)

a. FULLY OPEN Globe Valve.

b. FULLY OPEN Gate Valve.

c. FULLY OPEN Ball Valve.

d. FULLY OPEN Swing-type Check Valve.

5.16 Attached Figure 215 represents a transient that could occur at a BWR.

Given: (1) One Jet Pump Riser fails at time T=1.2 min.
(2) No operator actions occur
(3) Recorder Speed = 1 division = 1 minute

Briefly EXPLAIN the cause(s) of the following recorder indications: (2.5)

Note: There may be more than 1 cause for each answer.

a. Level INCREASE (Point A)
b. Reactor Power DECREASE (Point B)
c. Total Steam Flow DECREASE (Point C)
d. Feedwater flow DECREASE (Point D)
e. Reactor Pressure DECREASE (Point E)

5.17 Attached Figure 216 represents a transient that could occur at a BWR.

Given: (1) EHC Pressure Regulator Fails to Maximum at Time t = 1.0 min.
(2) No operator actions occur
(3) Recorder Speed = 1 division = 1 minute

Briefly EXPLAIN the cause(s) of the following recorder indications: (2.5)

Note: There may be more than 1 cause for each answer.

a. Level INCREASE (Point A)
b. Reactor Power DECREASE (Point B)
c. Reactor Power DECREASE (Point C)
d. Steam Flow DECREASE (Point D)
e. Pressure FLUCTUATION (Point E at times = 3 to 6 minutes)

5.18 A "Periodic NSS Core Performance Log" (P-1) is attached (Fig. 217) for reference. Which statement is most accurately reflected by this printout? (1.0)

a. Maximum LHGR(s) in the core is 4.92 kw/ft.
b. Maximum LHGR(s) in the core is 6.06 kw/ft.
c. Maximum LHGR(s) in the core is 10.89 kw/ft.
d. Maximum LHGR(s) in the core is 13.40 kw/ft.

6. Plant Systems Design, Control, and Instrumentation

6.1 Backup Scram valves provide a redundant means of venting air from the scram pilot valves and scram discharge valves. These backup valves are.... (CHOOSE ONE) (1.0)

a. ...normally energized and will de-energize upon a RPS Scram signal.

b. ...aligned such that two valves in series, one from each RPS trip channel, must actuate to vent the scram air header.

c. ...designed such that both RPS channels must trip in order for any one of the valves to actuate.

d. ...powered from the RPS Buses A and B.

6.2 Trace on Fig. 6.2 (<u>or</u> describe in detail) the flowpath of exhaust water from the CRD mechanism following a <u>normal</u> rod insertion. Include the specific component(s) or system section(s) the water travels through until it is no longer in the CRD system. (1.0)

6.3 The Reactor Recirculation Pump seal cartridge assemblies consist of two sets of sealing surfaces and breakdown bushing assemblies. Failure of the #2 seal assembly at rated conditions would result in ... (CHOOSE ONE) (1.0)

a. ...an INCREASE in #2 seal cavity pressure from approximately 500 psig to approximately 1000 psig.

b. ...a DECREASE in #2 seal cavity pressure from approximately 500 psig to approximately 0 psig.

c. ...an INCREASE in #1 seal cavity pressure from approximately 500 psig to approximately 1000 psig.

d. ...a DECREASE in #1 seal cavity pressure from approximately 500 psig to approximately 0 psig.

6.4 The plant is operating at 26% power and both recirc pump M/A transfer stations are in manual and set for 28% speed. The recirc flow A limit annunciator is clear. For each of the following instances, indicate how the speed of recirc pump "A" would change (i.e., increase, decrease, or remain same) and which component(s) of the control system is limiting. See Figure 16 for information.

a. Recirc pump "A" M/A transfer station placed to "Auto" (1.0)

b. Tachometer output feedback signal fails low-contact Y1 opens (1.0)

6.5 Assume the following initial rod position distribution:

All rods in groups 1 through 3 are fully withdrawn, except for one rod in each group - 22-55 in group 1, 46-55 in group 2, and 18-03 in group 3 - all fully inserted.

All rods in groups 4 through 10 are fully inserted to position 0 except for rod 34-27 in group 4 which is fully withdrawn.

Fill in the following table with the Rod/Rod Group number you would expect to see displayed in each RWM window for both situations below (a&b) upon initialization or relatch of the RWM. If nothing will be displayed write "BLANK." (2.0)

RWM Window	(a) Initial Condition (IC)	(b) Same as IC but rod 22-55 withdrawn to 48
Rod Group	________	________
Insert Error	________	________
Insert Error	________	________
Withdraw Error	________	________

6.6 Upon a loss of instrument air, how will the following valves fail? (Closed, open, as is)

a. CRD flow control valve (.5)
b. F&C SU level control valve (.5)
c. Scram discharge volume drain valve (.5)

Assume this is a complete loss of interruptible and non-interruptible instrument air.

6.7 With the plant operating at 100% power (Unit 1), recirc in master manual, an operator inadvertently increases the "Pressure Set" by 5 psig. Which of the following responses most correctly describes the initial response and final status of the following parameters due to this action? (1.0)

Assume 1. No operator action
2. Starting Parameters
- TCVs - 100% steam flow position
- BPVs - 0% steam flow position
- RxPower - 100%
- RxPressure - 1010 psig
3. All EHC control settings are standard

Note: All valve %s are % Steam Flow Position.
See Figure 3.3-15 for information.

	a.	b.	c.	d.
Initial Response				
- TCVs	≈82.5%	≈82.5%	≈82.5%	≈100%
- BPVs	0%	≈17.5%	≈17.5%	≈10%
- RxPower	Increase	100%	100%	<100%
- RxPressure	Increase	1010psig	1010psig	<1010psig
Final Status				
- TCVs	≈100%	≈100%	≈82.5%	≈100%
- BPVs	0%	0%	≈17.5%	0%
- RxPower	>100%	100%	>100%	100%
- RxPressure	>1010psig	1010psig	>1010psig	1010psig

6.8 The core contains 124 LPRM detectors in 31 "detector assemblies" (stainless steel tubes with 4 LPRM detectors each).

Fill in the blanks.

The "detector assemblies" are __________ (wet, dry) tubes and are __________ (symmetrically, assymmetrically) located in the core. (1.0)

6.9 Assume that APRM "B" currently has eleven operable LPRM inputs and is reading 65% power. Which of the following indication(s) and/or action(s) will occur as a result of one LPRM (of the eleven remaining LPRM inputs to APRM "B") failing downscale? Assume no operator action. (1.0)

a. LPRM downscale alarm - APRM "B" reading <65%

b. LPRM downscale alarm - APRM "B" reading >65%

c. LPRM downscale alarm - APRM Inop Alarm - Rod Block - APRM "B" reading 65%

d. LPRM downscale alarm - APRM INOP Alarm - Rod Block - 1/2 scram - APRM "B" Reading 65%

6.10 Which of the following axial location sequences correctly describes the axial locations of LPRMs in the core? (1.0)

Note: All measurements are inches above BAF

a. BAF → "A" @ + 9" → "B" @ + 27" → "C" @ + 45" → "D" @ + 63"

b. BAF → "A" @ + 18" → "B" @ + 54" → "C" @ + 90" → "D" @ + 126"

c. BAF → "D" @ + 9" → "C" @ + 27" → "B" @ + 45" → "A" @ + 63"

d. BAF → "D" @ + 18" → "C" @ + 54" → "B" @ + 90" → "A" @ + 126"

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

6.12 The APRM scram function actually consists of two separate setpoints; i.e., .66w + 54% and a fixed 120% scram.

a. Where, specifically, is/are the sensor(s) located which measure the variable "w"? (0.5)

b. While operating at power, one MSIV fails shut resulting in a brief (1 Second) flux spike to 121% power. Which of the two scram setpoints mentioned above (or both) should initiate a reactor scram? Justify your choice. (1.0)

6.13 Which AGAF value (P-1 Printout) is more conservative? (0.5)

a. .99
b. 1.01

6.14 The main turbine is at 1800rpm in preparation for synchronizing the main generator to the grid (i.e. the 230 kv generator breakers are still open). What will happen if the "All Valves Closed" pushbutton is depressed? (1.0)

a. Nothing will happen since the synchronous speed select signal is sealed in.

b. The turbine control valves and main stop valves will close, but the intercept valves will remain open.

c. All of the control valves (TCVs and IVs) and main stop valves will close.

d. The control valves (TCVs and IVs) will close, but the main stop valves will remain open.

6.15 SELECT which one of the following an operator does to INCREASE VARS. (1.0)

a. INCREASE Generator Speed
b. INCREASE Capacity Factor
c. INCREASE Generator Voltage
d. INCREASE Generator Stator Cooling

6.16 What components receive their cooling water supply from the vital service water header? (1.5)

6.17 Which of the following sequences of components correctly reflects the normal HPCI condensate flow path on Unit 2? (1.0)

a. CST → Booster Pump → Main Pump → "A" FW Line, upstream of FW Flow detector

b. CST → Booster Pump → Main Pump → "A" FW Line, downstream of FW flow detector

c. CST → Main Pump → Booster Pump → "A" FW Line, upstream of FW flow detector

d. CST → Main Pump → Booster Pump → "A" FW Line, downstream of FW flow detector

6.18 The RCIC (Reactor Core Isolation Cooling) System is capable of taking a suction from the CST or the suppression pool. The suppression pool suction valves (F031 and F029) and CST suction valve (F010) are interlocked such that... (CHOOSE ONE) (1.0)

a. ... the suction will automatically transfer from the CST to the suppression pool on high suppression pool water level.

b. ... the CST suction valve will automatically open if the suppression pool suction valves are manually closed while in standby mode.

c. ... the CST suction valve will automatically close if the suppression pool suction valves are opened.

d. ... the CST suction valve and both suppression pool suction valves will automatically close on a Group V (RCIC) isolation signal.

6.19 Which of the following logic signal combinations most correctly detail the complete logic sequence for the automatic initiation of the RHR system in the LPCI mode? (1.0)

a. - Rx vessel low level (LL #3)
   - or -
   - Drywell high pressure

b. - Rx vessel low level (LL #3) with Rx vessel low pressure
   - or -
   - Drywell high pressure with Rx vessel low pressure

c. - Rx vessel low level (LL #3) with Rx vessel low pressure
   - or -
   - Drywell high pressure

d. - Rx vessel low level (LL #3)
   - or -
   - Drywell high pressure with Rx vessel low pressure

6.20 Assume the CS pumps had automatically started in response to a Rx low-low-low level signal. The operator secures the CS pumps when Rx level is restored to 185". Which of the following operator actions must be taken to insure proper automatic start of the CS pumps on any subsequent CS initiation signal? (1.0)

a. No operator actions are required.

b. CS pump control switches must be cycled - i.e., pump re-started and secured.

c. CS "initiation signal sealed in" reset PB(s) must be depressed.

d. CS pump control switches must be taken from the stop position to the auto position.

6.21 The reactor is critical at approximately 5 psig and the "pressurization" phase of GP-02 is being performed. The Normal Control Range GEMAC LIs in the control room read the following "approximate" values:

GEMAC A (N004 A)	187"
GEMAC B (N004 B)	188"
GEMAC C (N004 C)	187"

a. The two Emergency System Range (Yarway) control room level indicators should read approximately... (CHOOSE ONE) (1.0)

1. ...150 inches
2. ...165 inches
3. ...188 inches
4. ...210 + inches

b. The Shutdown Vessel Flooding Range control room level indicator should read approximately... (CHOOSE ONE) (1.0)

1. ...150 inches
2. ...165 inches
3. ...188 inches
4. ...210 + inches

6.22 The plant is operating at 100% power in 3-element control when one steam flow input signal is lost to the FWLC system. Which of the following responses describe the correct system/plant response? Assume <u>no</u> operator action. See Figure 6.22 for information. (1.0)

a. Reactor water level decreases and stabilizes at a lower level.

b. Reactor water level decreases and initiates a reactor scram.

c. Reactor water level increases and stabilizes at a higher level.

d. Reactor water level increases and initiates a turbine trip (w/scram).

6.23 Which one of the following statements correctly describes the operation of the Motor Gear Unit (MGU) in controlling Reactor Feed Pump (RFP) turbine speed? (1.0)

a. The MGU will control the RFP turbine speed only if its speed demand signal is greater than that from the MSC.

b. The MGU can be used to control feed flow rate over a turbine speed range of approximately 0-5500 rpm.

c. The MGU is manually controlled from the control room at either a high or a low speed rate.

d. The MGU will fail "as is" to prevent a ramp response if it loses its signal from the flow controller.

6.24 Attached Figure 2 depicts the UPS power supply line-up and switch contact alignment for Unit 2 UPS with Inverter 2A supplying. Utilizing Figure 2 as a reference, describe what will automatically occur if inverter output 2A is lost, i.e., what is the new source of UPS power and what switch contacts change positions? (1.0)

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

7.1 a. SHELL WARMING is in progress. The internal bypass in 2 TSV is slowly opened to raise shell pressure. With no further operator action a scram results. Briefly EXPLAIN WHY. (1.0)

b. After depressing the 100 rpm speed select push button for a Turbine start-up, you should verify valve motion and light indication. Put the following in the order that you would see them per GP-03, Unit Startup and Sychronization. (1.5)

1. Intercept valves 1 and 3 - open slowly
2. Main stop valves 1, 3, 4 - open slowly
3. Increasing speed light comes on
4. Main stop valve #2 - begins to open
5. Control valves - throttle open
6. Intercept valves #2 and #4 - start to open

c. When bringing the turbine to rated speed, it is recognized that abnormal vibration exists. What action should be taken? (0.5)

7.2 With regard to GP-02, Approach to Criticality and Pressurization:

a. What two actions must be performed if a period of five seconds or less is reached? (1.0)

b. What action should be taken if a single notch withdrawal results in a period of 20 seconds? (0.5)

7.3 Complete the blanks of the following ECP CAUTION:

DO NOT SECURE OR PLACE an ECCS OR RCIC in manual mode UNLESS by at least ______ independent indications: (0.5)

1. ____, (0.5)

OR

2. ____. (0.5)

7.4 With regard to OP-16, RCIC:

a. When placing RCIC in standby, the sequential steps are as follows:

1. Verify that the steam supply outboard isolation valve, E51-F008 is closed.

2. Verify that the steam supply inboard isolation valve E51-F007 is closed.

3. Open the supply drain pot drain bypass valve, E51-F054.

4. Open the steam supply outboard isolation valve, E51-F008.

5. Slowly throttle open the steam supply inboard isolation valve E51-F007.

Why are you performing these steps, and what are the consequences of opening F007 quickly? (1.0)

b. When operating RCIC for Rx pressure control, a caution states that opening the bypass to CST valve may cause turbine speed to decrease below 2000 RPM. Why is RCIC operation below 2000 RPM undesirable (2 reasons)? (1.0)

7.5 Answer the following questions with regard to reactor recirculation pump operational limitations and precautions:

a. GP-05, "Unit Shutdown," states that recirculation pump operation at a suction pressure below 300 psig should be minimized. Why is this recommendation necessary? (0.5)

b. When increasing recirc pump speeds with both controllers in MANUAL, their speeds should normally be maintained within ________________%. The speed differential is limited to _________% when below 75% core flow and _____% when above 75% core flow. (1.5)

7.6 Answer the following questions with regard to the Reactor Water Cleanup System Operating Procedure (OP-14):

a. Reactor Coolant temperature is 300°F and a major portion of the RWCU flow is being rejected to the condenser. Briefly explain why it is recommended that flow back to the reactor vessel be established slowly over a 45 minute period. (1.0)

b. OP-14 cautions the operator to maintain maximum RWCU System flow and temperature when operating at low power. Why is this practice recommended? (1.0)

7.7 An improper RBCCW system lineup could result in possible damage to the pumps and/or heat exchangers as stated in the "CAUTIONS" of the system operating procedure (OP-21). Which of the following lineups/conditions would minimize the likelihood of component damage over an extended operating period? (1.0)

a. Running one RBCCW pump with two RBCCW heat exchangers.
b. Running two RBCCW pumps with two RBCCW heat exchangers.
c. Running two RBCCW pumps with one RBCCW heat exchanger.
d. Running two RBCCW pumps with three RBCCW heat exchangers.

7.8 GP-07, "Preparation for Core Alterations," cautions the operator to suspend fuel movement in the fuel pool near the fuel pool gates while work is in progress in the reactor cavity. Briefly explain why this precaution is necessary. (1.0)

7.9 A LOCA has occurred and a high temperature steam environment exists in the drywell. Briefly explain why drywell sprays must NOT be initiated in the "unsafe" region of figure #16, "Drywell Spray Initiation Pressure Limit" (attached). (1.0)

7.10 Unit 2 is operating at 10% power when the "A" CRD pump seizes due to a failed bearing. Upon starting the "B" CRD pump in accordance with AOP-02.1, "Inability to Move Control Rods," it immediately trips on overload and cannot be restarted. When is a manual reactor scram required per the AOP? (1.0)

a. If reactor pressure is below 800 psig.

b. Immediately.

c. Upon activation of the "CRD HYD TEMP HIGH" annunciator.

d. Upon activation of the "CRD ACCUM LO PRESS/HI LEVEL ALM" annunciator.

7.11 Which of the following is a symptom you would expect to see as a result of a "Jet Pump Failure" (AOP-04.4)? (1.0)

a. Increase in generator megawatt output.
b. Increase in core thermal power.
c. Increase in total core flow.
d. Increase in core plate dp.

7.12 Unit 1 is operating at power when a Safety Relief Valve (SRV) fails and sticks in the open position. Per AOP-30.0, "Safety Relief Valve Failures," when must a manual reactor scram be initiated? (1.0)

a. Prior to the automatic reactor scram at 850 psig.
b. As soon as it is recognized that the SRV will not close.
c. As soon as suppression pool water temperature reaches 120°F
d. Immediately upon discovering that the SRV is open.

7.13 Match the automatic action in column "A" with the system parameter in column "B" which will initiate that action. (2.0)

Column A	Column B
a. Service Air header isolates	1. Service Air header Pressure decreases to 105 psig.
b. Interrubtable Instrument Air header isolates	2. Instrument air header pressure decreases to 103 psig.
c. Standby reactor building air compressors start	3. Instrument air header pressure decreases to 100 psig.
d. Air compressors A, B, and C start and load	4. Non-interruptable instrument air header pressure decreases to 95 psig.

7.14 What are the four abnormal conditions that would be reason to terminate fuel handling operations per Fuel Handling Procedure FH-11? (2.0)

7.15 A reactor cooldown is in progress and neither RHR loop can be placed in the shudown cooling mode. In accordance with AOP-15.0, Alternate Shutdown Cooling has been established to remove decay heat and continue the cooldown. State the condition/status of each of the following components/parameters when operating in this mode of shutdown cooling. (2.5)

a. MSIV's
b. SRV's
c. RHR loops A and B
d. Rx pressure (with respect to suppression chamber pressure)
e. Rx level (provide numerical value or component reference)

7.16 List four of the five entry conditions (parameters and set-points) for the Containment Control Procedure, EOP-01-CCP. (2.0)

8.0 Administrative Procedures, Conditions, and Limitations

8.1 Which of the following individuals represents the minimum level of authority required for cancellation of an LCO per OI-04, LCO Evaluation and Follow-up? (1.0)

a. Regulatory Compliance Specialist
b. Control Operator
c. Shift Foreman
d. Shift Operating Supervisor

8.2 Who is the individual charged to perform the preliminary/initial determination of whether an event requires a red phone or prompt report (as defined in the Technical Specifications) per OI-04, LCO Evaluation and Follow-up? (1.0)

a. Control Operator
b. Shift Foreman
c. Shift Operating Supervisor
d. Shift Technical Advisor

8.3 Until the EOF is activated, the Site Emergency Coordinator can not delegate the responsibility for... (CHOOSE ONE) (1.0)

a. ... directing the combined activities of plant personnel in the CR, TSC, and OSC.

b. ... requesting outside emergency assistance.

c. ... assessing the emergency condition for possible upgrade in classification.

d. ... deciding what protective action recommendations will be made to off-site authorities.

8.4 Which of the following statements correctly describes a Priority 1 work request per the "Corrective Maintenance" Procedure, MP-14? (1.0)

a. It may be worked on a 24-hr/day, 7-day/week schedule upon approval by the General Manager.

b. It must be approved by the Shift Operating Supervisor and the Maintenance Supervisor.

c. The WR&A must be completed and signed prior to commencing the maintenance activity.

d. Priority 1 shall be assigned to all failures of safety-related equipment requiring immediate plant shutdown.

8.5 What is the difference between a DEPARTURE and a DEVIATION from an established procedure? (1.0)

8.6 Procedure OI-01, Operating Principles and Philosophy, states that during the performance of normal evolutions by two persons at different locations, both persons should have a copy of the procedure. In which of the following situations would it be allowable per OI-01 for only one individual to have a copy of the procedure? (1.0)

a. During an evolution that requires only a limited number of manipulations by an individual under the direction of the controlling individual. Only the individual controlling the evolution need have a copy of the procedure.

b. During an evolution that requires only a limited number of manipulations by an individual under the direction of the controlling individual. Only the individual performing the manipulations need have a copy of the procedure.

c. During an evolution in a contamination area that will be completed within one hour by the individual performing the manipulations irregardless of the number (of manipulations) involved. Only the individual controlling the evolution need have a copy of the procedure.

d. During an evolution in a contamination area that will be completed within one hour by the individual performing the manipulations irregardless of the number (of manipulations) involved. Only the individual performing the manipulations need have a copy of the procedure.

8.7 Match the following four emergency classes to the appropriate definition. (2.0)

a. Unusual Events	1. Events are in process or have occurred which involve actual or likely major failures of the plant functions needed for protection of the public.
b. Alert	2. Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant.
c. Site Emergency	3. Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.
d. General Emergency	4. Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.

8.8 In accordance with 10 CFR 55, "if a licensee has not been actively performing the functions of an operator or senior operator for a period of __*__ months or longer, he shall, prior to resuming activities licensed pursuant to this part, demonstrate to the Commission that his knowledge and understanding of facility operation and administration are satisfactory."
*Fill in the blank with one of the following times. (1.0)

a. 4
b. 6
c. 12
d. 24

8.9 In which of the following situations would it be possible to utilize a human red tag in lieu of a properly issued clearance per AI-58, Equipment Clearance Procedure? (1.0)

a. During the processing of a clearance boundary extension for Priority 1 maintenance, if the Clearance Tag Sheet (BSEP 20) will be completed within one hour of the human red tag assignment and it is approved by the Shift Operating Supervisor and the Maintenance Supervisor.

b. During maintenance on equipment that has been rendered safe by the placement of approved wire removal tags, if the maintenance will be complete within one day and it is approved by the Shift Foreman and the Maintenance Supervisor.

c. During maintenance on Inoperable Safety/Technical Specification related equipment which, if not returned to operability status, will require a plant shutdown within one hour. The approval of the Shift Operating Supervisor or the Shift Foreman is required.

d. During an emergency situation as determined by the Shift Operating Supervisor or the Shift Foreman.

8.10 Clearances may be transferred from one clearance holder to another provided both persons agree to the transfer. Which one of the following individuals must also give his/her consent per AI-58, Equipment Clearance Procedure? (1.0)

a. Supervisor of original clearance holder.
b. Applicable unit Control Operator.
c. Applicable unit Shift Foreman.
d. Shift Operating Supervisor.

8.11 __*__ clearances. (definition) - Issued to two or more individuals which have the same boundary for work that may may or may not be related. *Fill in the blank with one of the following terms. (1.0)

a. Local
b. Individual
c. Multiple
d. Master

8.12 Unit 2 Technical Specifications define SHUTDOWN MARGIN as ...

"SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor would be subcritical assuming..."

List the three (3) conditions which complete the definition of SHUTDOWN MARGIN. (1.5)

8.13 Given the following conditions on Unit 2:

Mode Switch	-	Refuel
Temperature	-	180°F
Pressure	-	0 psig
Level	-	184 inches
RHR	-	SDC Mode

Head bolts to the RPV are DETENSIONED

STATE the above described Operational Condition. (0.5)

8.14 A __*__ shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indication and/or status derived from independent channels measuring the same parameter. *Fill in the blank with one of the following TS terms. (1.0)

a. Channel Calibration
b. Channel Check
c. Channel Functional Test
d. Logic System Functional Test

8.15 The APRM Trip Setpoint Formula is (.66W + 54%) T. What is the definition of variable "T"? When is it applied to the Formula? (1.0)

8.16 Unit 2 is at 75% rated thermal power, with two outstanding deficiencies:

ADS	1 ADS Valve INOP (1 Day)
CS	CST Suction INOP (3 Days)

The Auto - swap of the HPCI suction upon receiving CST low level is determined to be UNSATISFACTORY. The suction is MANUALLY switched to the Suppression Pool, and the suction to the CST is ISOLATED.

Which of the following actions most correctly detail the allowances and/or limitations imposed by the Technical Specifications in this instance? (1.0)

NOTE: Applicable TSs are enclosed for reference.

a. No new limitations or TS Operational Condition restrictions are initiated by this re-alignment.

b. Be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to less than or equal to 113 psig within the following 24 hours.

c. Be in at least HOT SHUTDOWN within six hours and in COLD SHUTDOWN within the following 30 hours.

d. Be in at least HOT SHUTDOWN within six hours and in COLD SHUTDOWN within the following 24 hours.

8.17 Unit 2 is at 75% rated thermal power with only one outstanding LCO:

- HPCI has been INOP for nine days due to an in-progress repair.

Ten minutes into the shift, RHR Pump "A" fails to start twice during the performance of a scheduled surveillance and is declared INOP.

Which of the following actions most correctly detail the allowances and/or limitations imposed by the Technical Specifications in this instance? (1.0)

Note: Applicable TSs are enclosed for reference.

a. Power Operation may continue for five days; and then, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. Power Operation may continue for seven days; and then, be in at least HOT SHUTDOWN within the next 12 hours and in Cold Shutdown within the following 24 hours.

c. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. Be in at least HOT SHUTDOWN within six hours and in COLD SHUTDOWN withi the following 30 hours.

8.18 Unit 2 is in COLD SHUTDOWN during a reactor startup with no outstanding deficiencies. The Containment Atmosphere Dilution (CAD) system becomes INOP. It is anticipated that repairs will be complete within two weeks.

With regard to the reactor startup, which of the following actions most correctly detail the allowances and/or limitations imposed by the Technical Specifications? (1.0)

NOTE: Applicable TSs are enclosed for reference.

a. Startup activities may continue; Operational Condition 1 may be entered with no restriction on power, but the CAD system must be returned to an Operable status within 32 days of exceeding 15% power.

b. Startup activities may continue; Operational Condition 1 may be entered but Thermal Power is limited to 15%.

c. Startup activities may continue; Operational Condition 2 may be entered but not exceeded.

d. Startup activities may continue; however, Operational Condition 4 must be maintained. (Entry into Operational Condition 5 is acceptable.)

8.19 Unit 1 is shutdown and in a long term outage. Unit 2 has been recently shutdown and placed in Cold Shutdown - cooldown completed last shift. The shutdown and cooldown of Unit 2 was necessitated by a requirement to drain and visually inspect the Suppression Pool.

The following plant conditions/requirements have been established:

- CS system is aligned to the CST with an Operable Flow Path capable of transferring water through the spray sparger to the reactor vessel.
- Reactor Mode Switch is locked in the Shutdown position.
- No operations affecting the reactor vessel or with the possibility of draining the vessel are in progress or planned.

There is only one outstanding deficiency:

- DG #4 is INOP due to in-progress repairs.

The Shift Operating Supervisor directs that the draining of the Suppression Pool will commence as soon as all of the TS requirements are met.

Which of the following actions most correctly detail the allowances and/or limitations imposed by the Technical Specifications in this instance? (1.0)

NOTE: Applicable TSs are enclosed for reference.

a. Commence Suppression Pool draining as soon as practical since all TS LCO requirements are met.

b. Commence Suppression Pool draining only after DG #4 is repaired and declared operable.

c. Commence Suppression Pool draining as soon as you insure that No Positive Reactivity changes will occur during this condition.

d. Commence Suppression Pool draining as soon as you insure that No Positive Reactivity changes will occur <u>and</u> that one LPCI subsystem is operable.

8.20 Which of the following individuals must perform a Control room board walk down prior to assuming the shift position per OI-02, Shift Turnover Checklist? (1.0)

a. Control Operator
b. Shift Foreman
c. Shift Operating Supervisor
d. All of the above

8.21 The determination is made that an Invalid Multiple Input Annunciator Condition is being caused by one failed sensor input. Which of the following steps should be taken per OI-05, Abnormal Annunciator Status? (1.0)

a. Remove annunciator card and identify annunciator window with a "Red dot".

b. Remove annunciator card - Defeat invalid sensor input - Replace annunciator card.

c. Remove annunciator card - Defeat invalid sensor input - Replace annunciator card. Identify annunciator window with a "Red dot".

d. Remove existing annunciator card and replace with "special slow window flash" annunciator card. Identify window with a "yellow dot".

8.22 Briefly explain the reason for the following caution from OI-13.

"When performing valve checks or line-ups on systems that are normally operated at high temperatures, valves should NOT be positioned on their backseat." (1.0)

8.23 The following data resulted from the DSR Drywell leakage calculations required to be made every 4 hours. The data was taken on Unit 2 during a single day of operation at Operational Condition 1 (The unit has been in Operational Condition 1 for 2 weeks). Only final data is presented; Preliminary calculational data is not supplied.

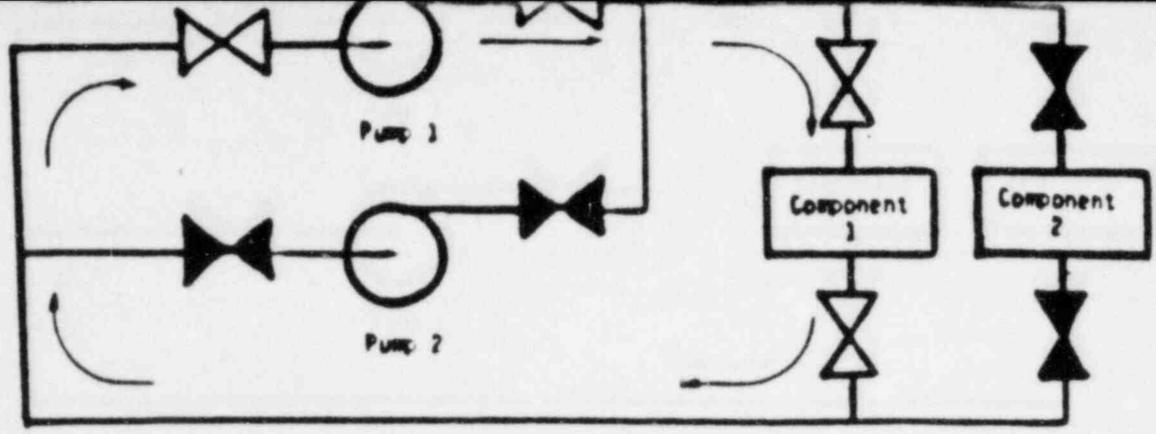
SHIFTS

	00-04	04-08	08-12
Floor Drain Leak Rate	2.52 gpm	4.25 gpm	3.75 gpm
Equipment Drain Leak Rate	20.91 gpm	20.58 gpm	21 gpm
Leak Rate to Drywell	23.43 gpm	24.83 gpm	24.75 gpm

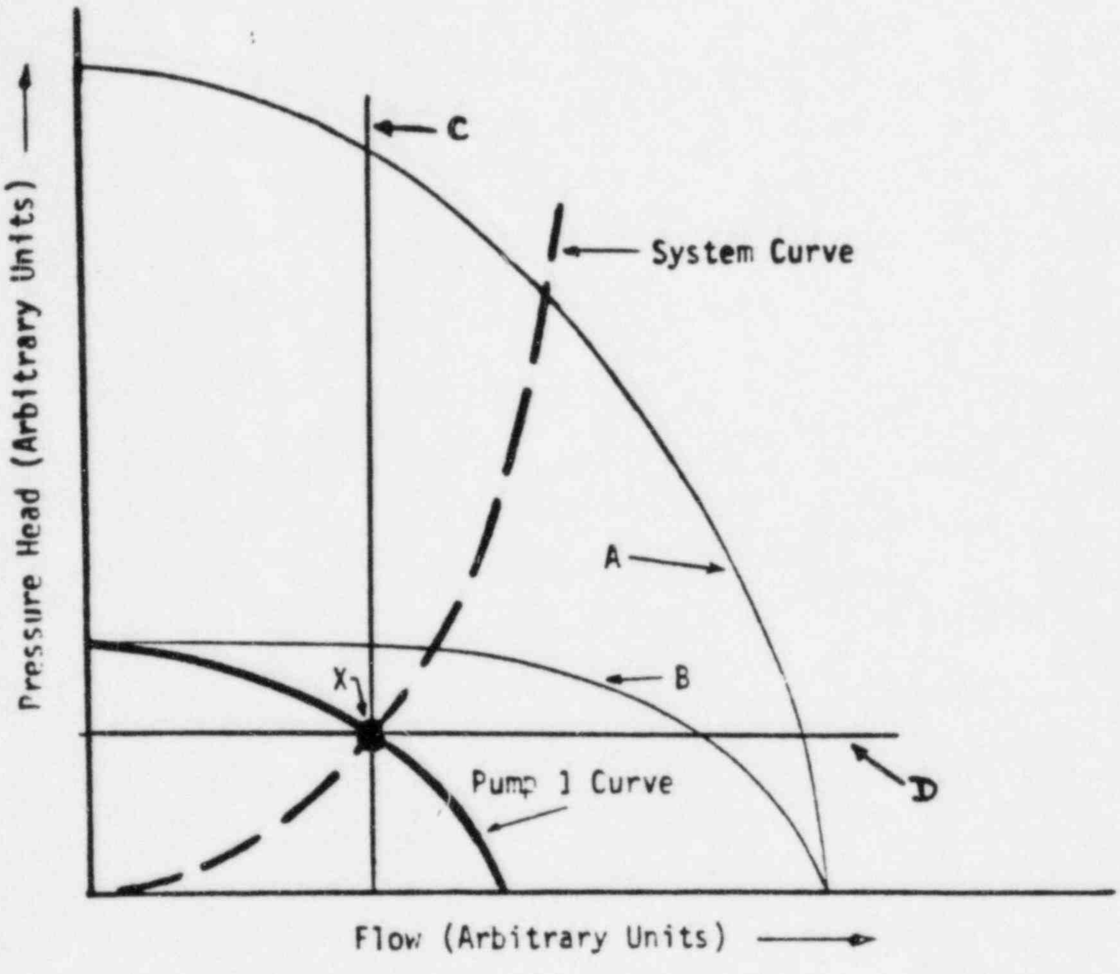
	12-16	16-20	20-24
Floor Drain Leak Rate	4.3 gpm	4.48 gpm	5.2 gpm
Equipment Drain Leak Rate	22.25 gpm	24.33 gpm	19.33 gpm
Leak Rate to Drywell	26.55 gpm	28.91 gpm	24.53 gpm

State the 4 TS Operation Leakage LCO limit(s) applicable in this plant condition and Identify any that were exceeded (as indicated by the above data). (3.0)

NOTE: DSR Drywell leakage calculation sheet is provided as information - applicable limits have been deleted.



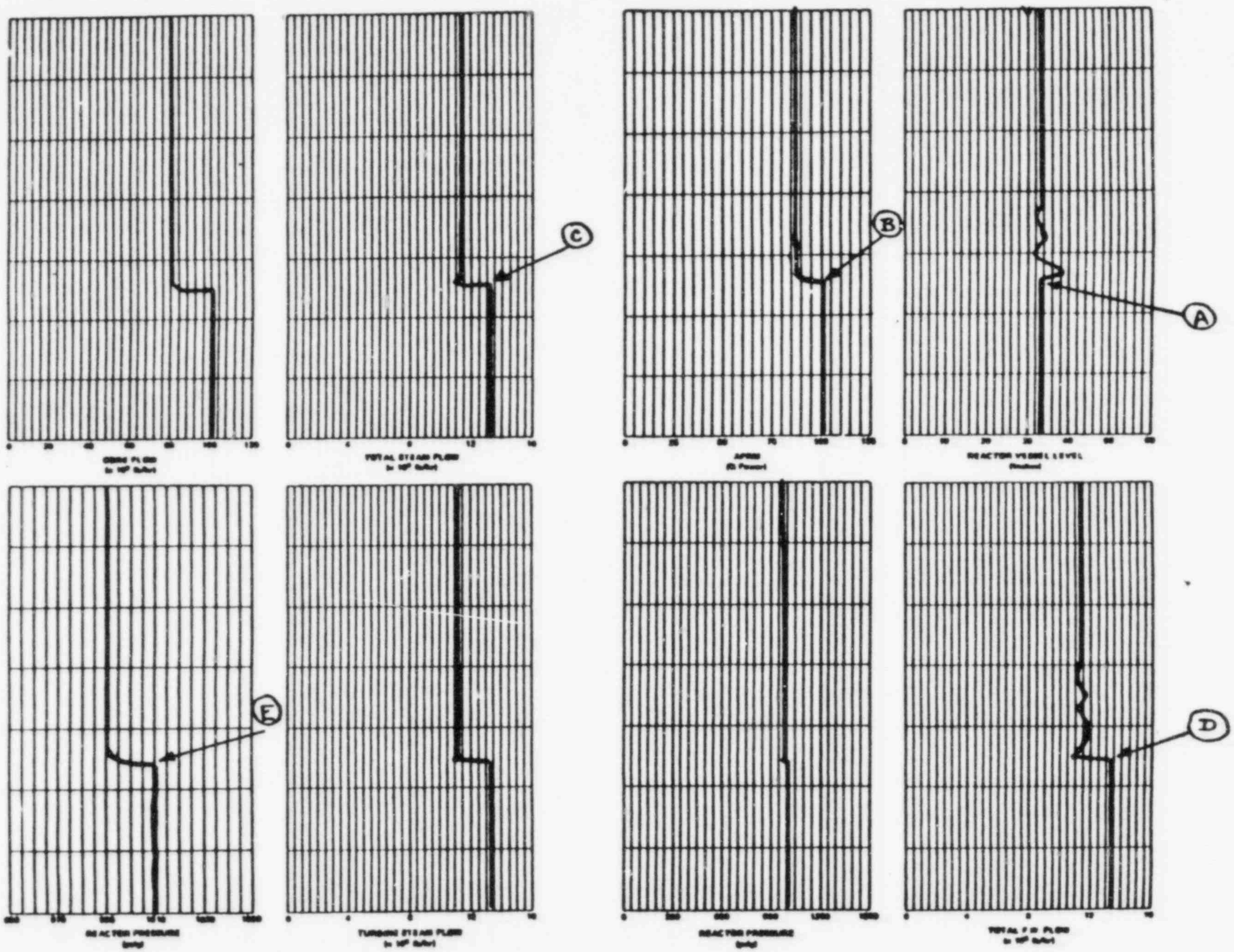
SYSTEM



SYSTEM HEAD VS. FLOW PLOT

FIGURE FOR 152

JET PUMP RISER FAILURE



Finure 215

PRESSURE REGULATOR FAILS HIGH

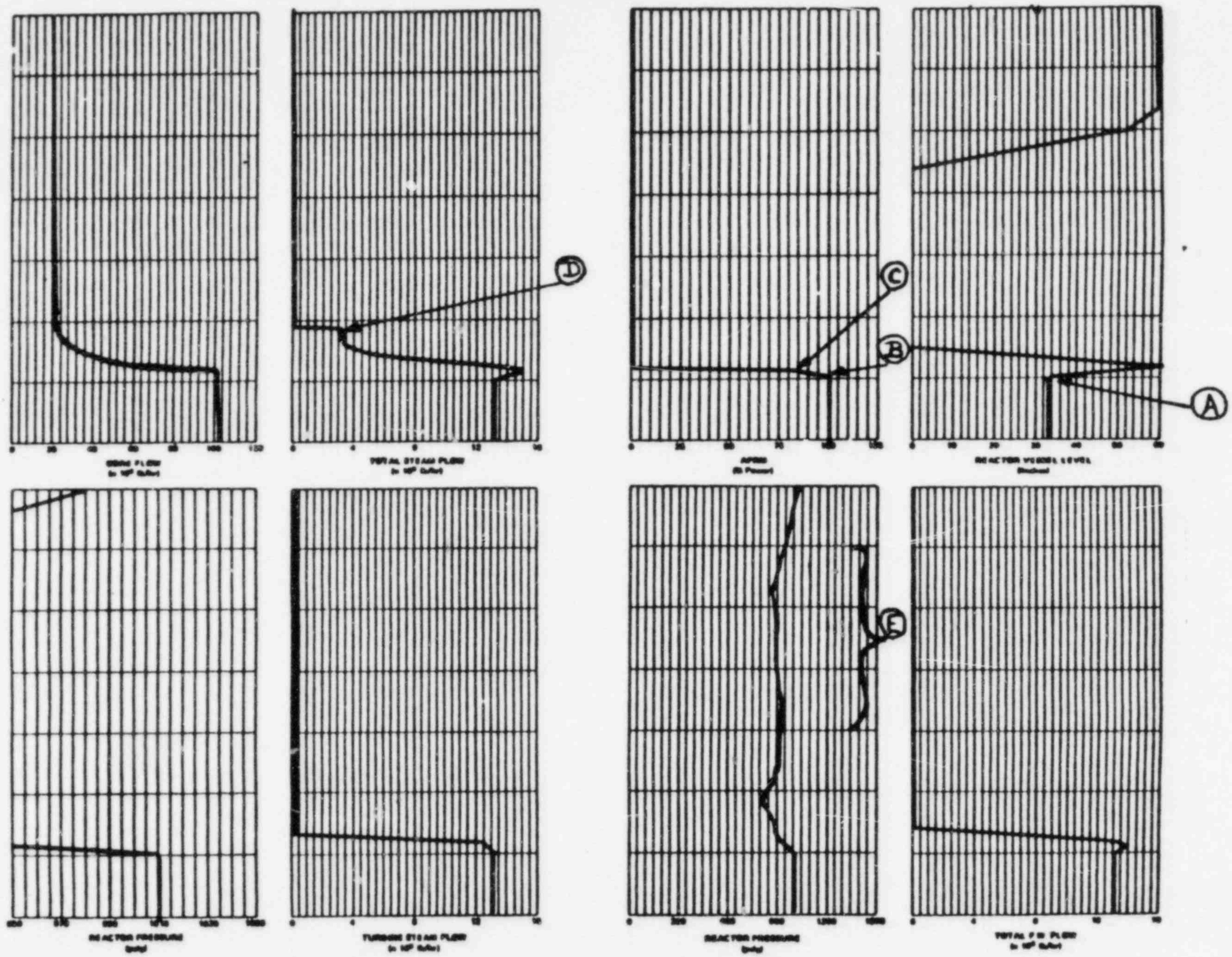


Figure 216

PERIODIC NSS CORE PERFORMANCE LOG

LOCATION	1	2	3	4	5	6	7	8	9	10	11	12
AXIAL REL PWR	0.46	0.98	1.16	1.17	1.14	1.14	1.12	1.18	1.21	1.11	0.85	0.49
REGION REL PWR	0.90	1.00	0.90	1.06	1.18	1.06	0.90	1.00	0.90			
RING REL PWR	0.89	1.28	1.16	1.09	1.08	1.09	0.73					
APRM GAF	0.95	0.97	0.92	0.93	0.97	0.93						

REGION	1	2	3	4	5	6	7	8	9
MFLCPR	0.591	0.579	0.590	0.628	0.642	0.628	0.590	0.579	0.591
LOC	11-14	23-18	41-14	11-22	31-28	41-32	11-40	29-36	41-40
FLOW	0.0599	0.0601	0.0599	0.0588	0.0589	0.0588	0.0599	0.0601	0.0599
PKF	1.26	1.23	1.25	1.33	1.31	1.33	1.25	1.23	1.26
MFLPD	0.422	0.440	0.422	0.452	0.451	0.452	0.422	0.440	0.422
LOC	13-18-16	23- 8- 5	39-18-16	5-28- 6	27-22-16	47-28- 6	13-36-16	29-46- 5	39-36-16
PKFL	1.87	1.95	1.87	2.00	2.00	2.01	1.87	1.95	1.87
MAPRAT	0.426	0.425	0.426	0.432	0.452	0.432	0.426	0.425	0.426
LOC	13-18-18	23- 8- 5	39-18-18	13-32- 5	27-22-16	39-22- 5	13-36-18	29-46- 5	39-36-18
PKFS	1.53	1.57	1.53	1.61	1.63	1.61	1.53	1.57	1.53

CMWT	1320.
PCT PWR	54.2
GMWE	402.0
CMFCP	0.642
CMFLPD	0.452
CMAPR	0.452
CMPF	2.005
CAEQ	0.149
CAQA	0.079
CAVF	0.365
CAPD	26.637
CRD	0.145
CRSYM	2.
PR	963.45
DPC-M	5.47
DPC-C	9.07
RWL	185.51
DHS	28.20
WFW	5.22
WD	15.76
WTSUB	37.35
WTHB	-1.00
WT	35.72
PCTWTR	46.4
WTFLAG	2.0
ITER	1.0
IREC	0.0
IEQL	1.0
IXYFLG	0.0

FAILED SENSORS 2 3 4 5 6 8 18 34

FAILED LPRM LIST

BASE CRIT CODE

2005.A.2	2805.A.2	3605.A.2	1213.C.2
2013.C.2	2813.A.2	4413.C.2	2821.B.2
3629.C.2	0437.B.2	2037.C.2	2837.B.1
2837.D.2	4437.D.2	2045.B.1	2045.C.1
2845.A.2	2845.C.1		

THE 12 MOST LIMITING BUNDLES

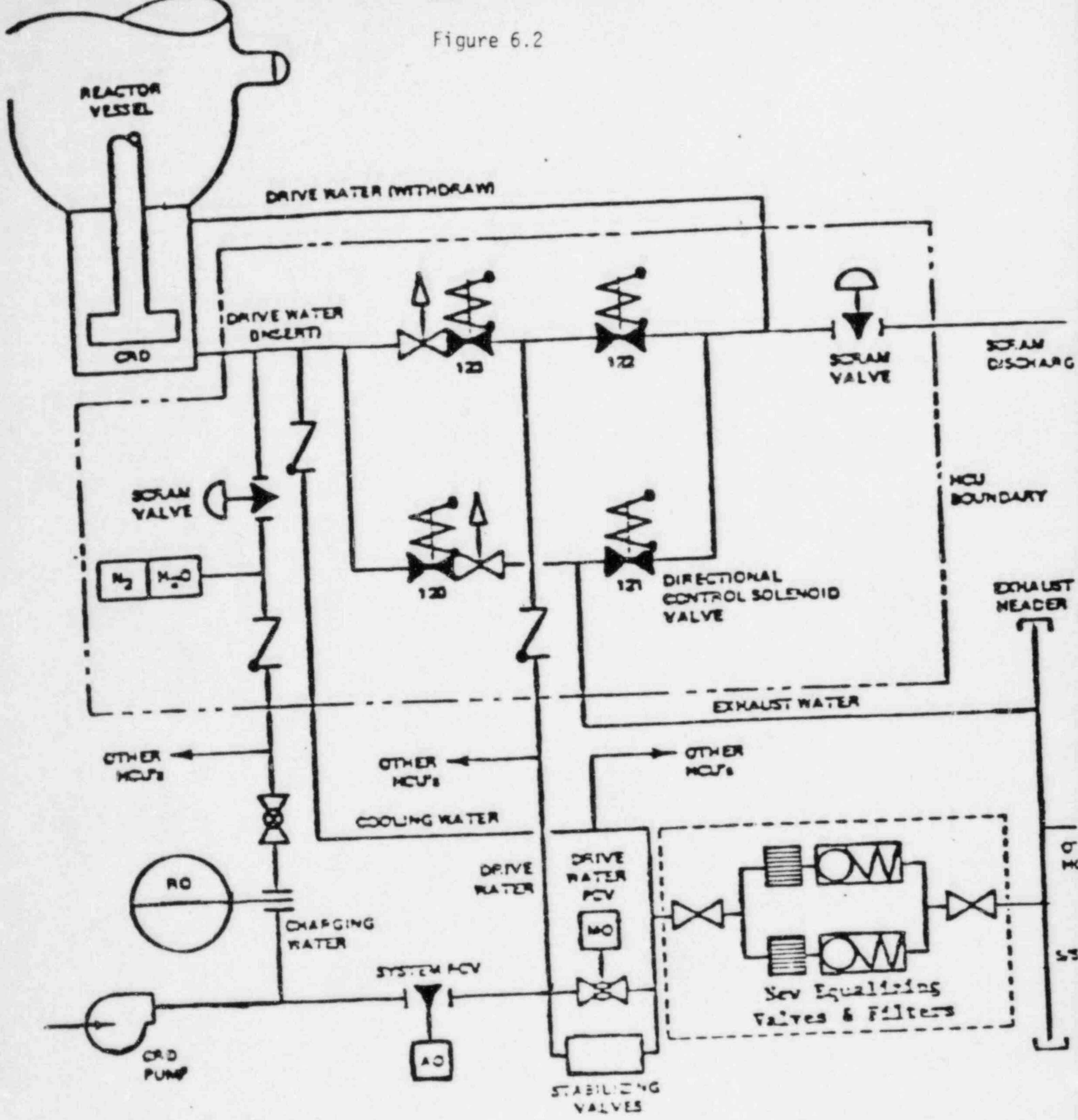
FOR MFLCPR				FOR MFLPD				FOR MAPRAT			
MFLCPR	LOC	MCPR	CPRLIM	MFLPD	LOC	MRPD	RPDLIM	MAPRAT	LOC	MAPLHGR	LIMLHGR
0.642	31-28	2.175	1.396	0.452	47-28- 6	6.06	13.40	0.452	27-22-16	4.92	10.89
0.642	31-26	2.175	1.396	0.452	47-26- 6	6.05	13.40	0.452	25-32-16	4.92	10.89
0.642	21-28	2.175	1.396	0.452	5-28- 6	6.05	13.40	0.452	27-32-16	4.92	10.89
0.642	21-26	2.175	1.396	0.452	5-26- 6	6.05	13.40	0.452	25-22-16	4.92	10.89
0.636	29-28	2.195	1.396	0.451	13-32- 5	6.05	13.40	0.451	31-28-16	4.91	10.88
0.636	29-26	2.196	1.396	0.451	39-22- 5	6.05	13.40	0.451	31-26-16	4.91	10.88
0.636	23-28	2.196	1.396	0.451	13-22- 5	6.04	13.40	0.451	21-28-16	4.91	10.88
0.632	33-30	2.210	1.396	0.451	39-32- 5	6.04	13.40	0.451	21-26-16	4.91	10.88
0.632	33-24	2.210	1.396	0.451	27-22-16	6.04	13.40	0.446	29-20-18	4.86	10.88
0.632	19-30	2.210	1.396	0.451	25-32-16	6.04	13.40	0.446	23-34-18	4.86	10.80
0.632	19-24	2.210	1.396	0.451	27-32-16	6.04	13.40	0.446	23-20-18	4.86	10.88
0.630	27-30	2.217	1.396	0.451	25-22-16	6.04	13.40	0.446	29-34-18	4.86	10.88

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

Figure 217

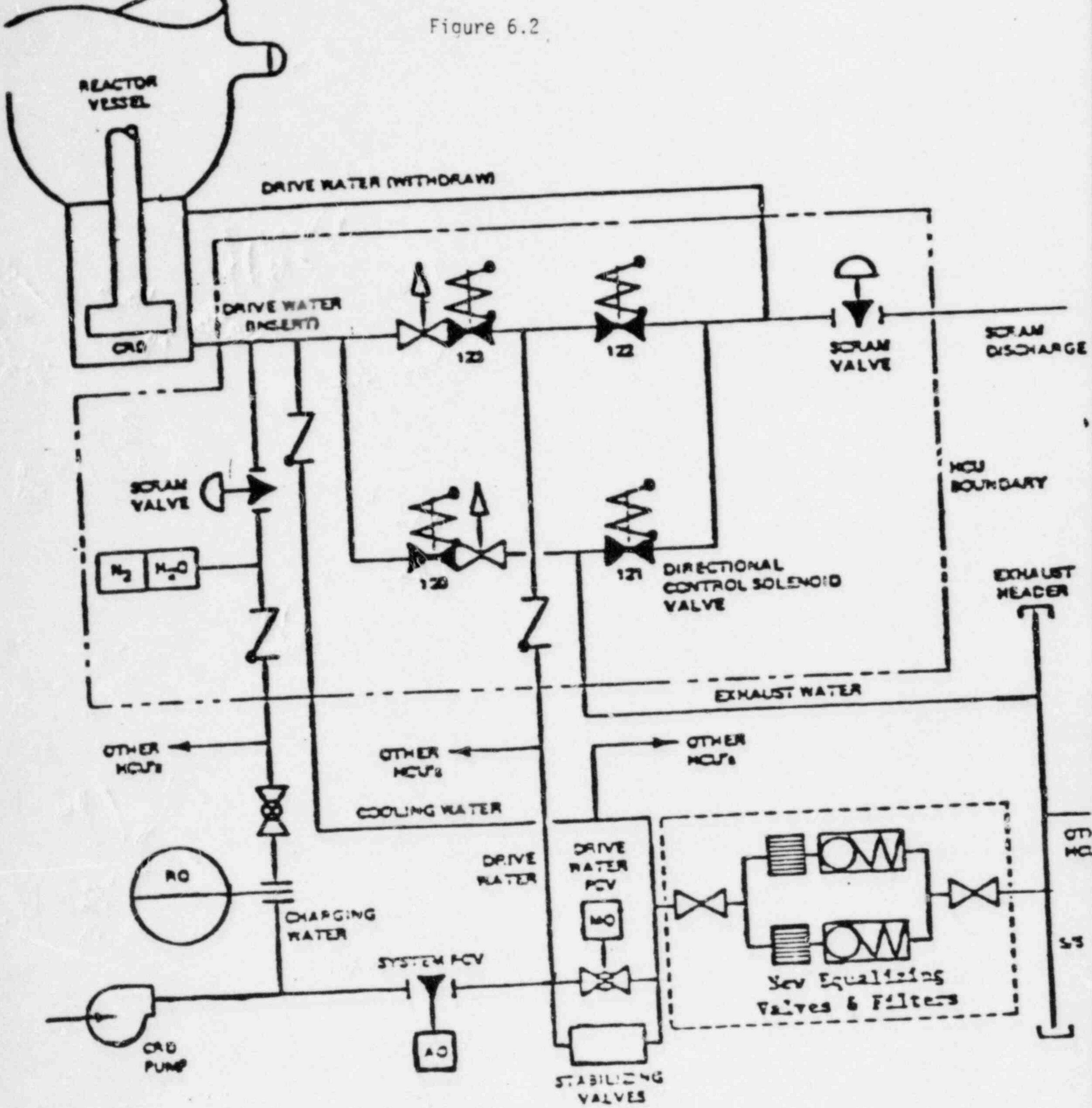
CRD HYDRAULIC SYSTEM

Figure 6.2



SIMPLIFIED SCHEMATIC OF MODIFIED
CRD HYDRAULIC SYSTEM

Figure 6.2



RECIRCULATION SYSTEM FLOW CONTROL NETWORK (A or B)

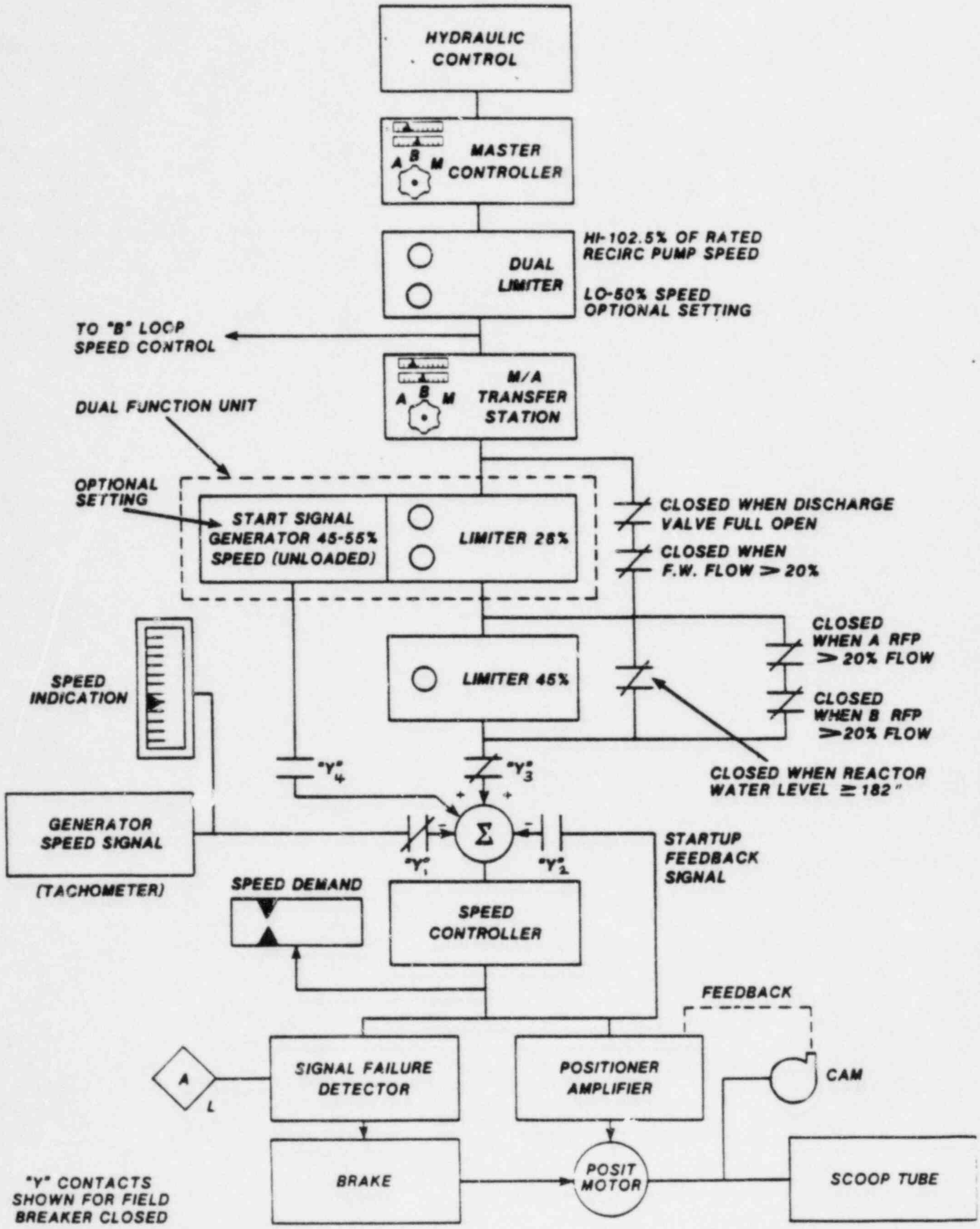


Figure 16

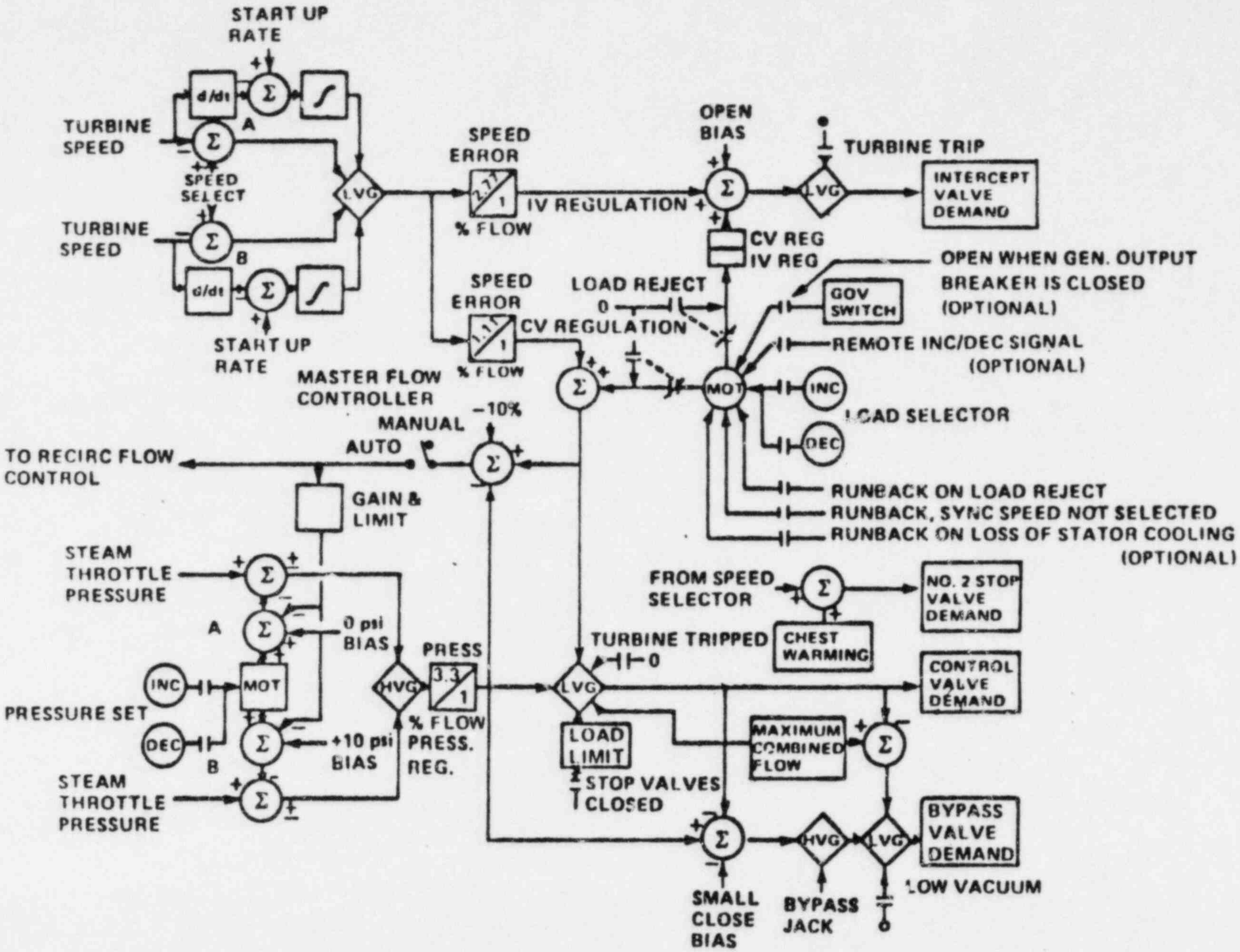
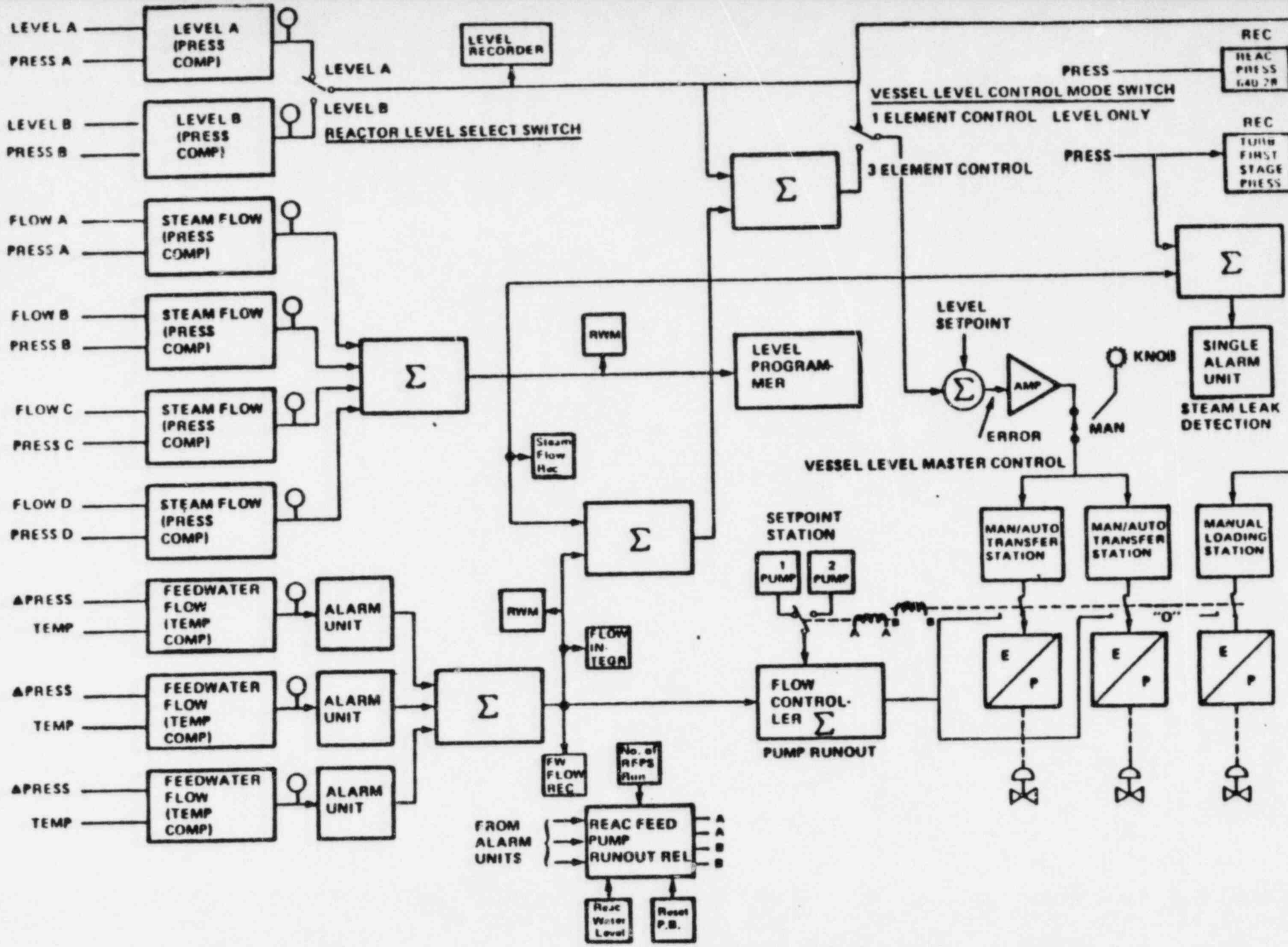


FIGURE 3.3-15 Electro-Hydraulic Control Logic



FEEDWATER CONTROL SYSTEM

Figure 6.22

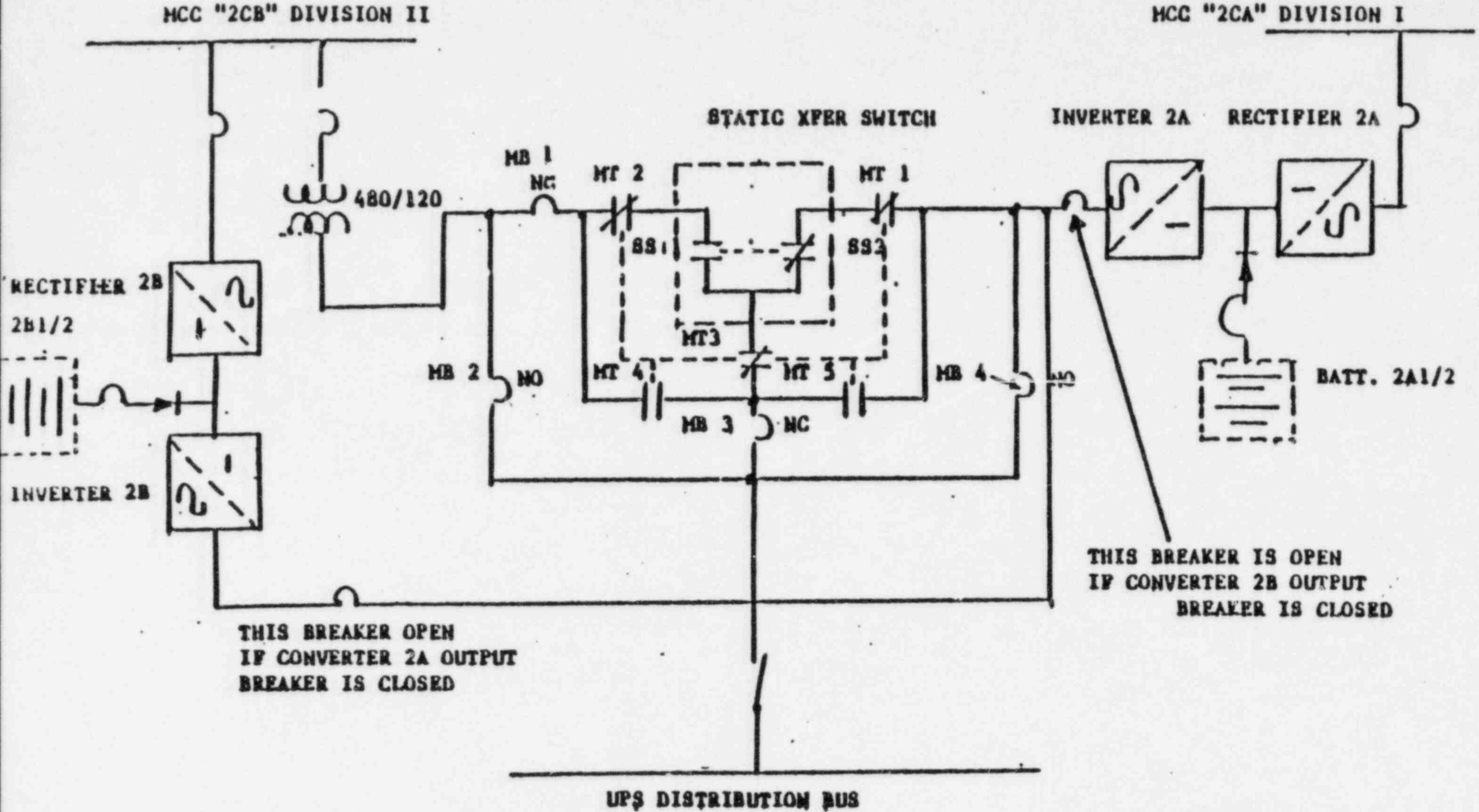


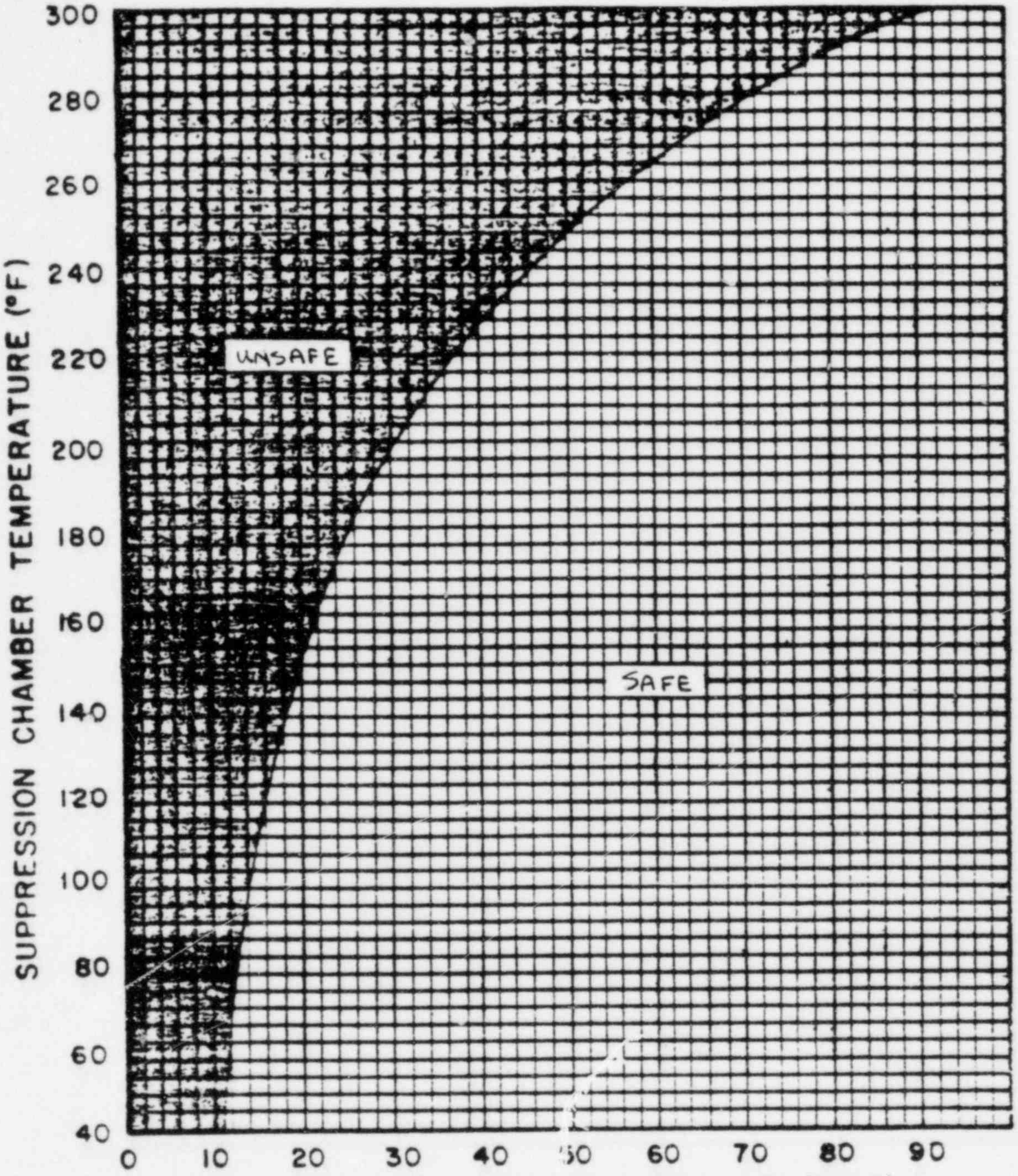
FIGURE 2
UNIT 2 UPS SYSTEM

NOTE

SUPPRESSION CHAMBER AIR SPACE TEMPERATURE IS DETERMINED BY:

UNIT ONE ONLY
THE AVERAGE OF POINTS 17, 18, 19 AND 20 ON CAC-TR-1258 OR THE AVERAGE OF COMPUTER POINTS W106, W107, W115 AND W116.

UNIT TWO ONLY
THE AVERAGE OF POINTS 2 AND 3 ON CAC-TR-4426-1 AND POINTS 2 AND 3 ON CAC-TR-4426-2 OR THE AVERAGE OF COMPUTER POINTS W106, W107, W115 AND W116.



DRYWELL SPRAY INITIATION PRESSURE LIMIT

FIGURE #16

DRYWELL LEAKAGE CALCULATION

	SAT	SUN	MON	TUES	WED	THURS	FRI
BASE VALUE LEAKAGE = ____________ gpm*							
SUMP FILL TIMER SETTING = __________ minutes*							
MANUALLY PUMP FLOOR DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP).							
CALCULATE TIME INTERVAL (MINUTES) SINCE SUMP WAS LAST MANUALLY PUMPED.							
RECORD INTEGRATOR READING.							
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM TIME SUMP WAS LAST MANUALLY PUMPED (LEAKAGE).							
CALCULATE FLOOR DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).**							
MANUALLY PUMP EQUIPMENT DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP).							
CALCULATE TIME INTERVAL (MINUTES) SINCE SUMP WAS LAST MANUALLY PUMPED.							
RECORD INTEGRATOR READING.							
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM TIME SUMP WAS LAST MANUALLY PUMPED (LEAKAGE).							
CALCULATE EQUIPMENT DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).							
FLOOR DRAIN LEAK RATE. (If zero, see xxx below)	2.52 gpm						
EQUIPMENT DRAIN LEAK RATE. (If zero, see xxx below)	20.91 gpm						
LEAK RATE TO DRYWELL.	23.43 gpm						
INITIALS: (Control Operator/Shift Foreman)	/	/	/	/	/	/	/

* NOTE: Base value leakage and sump fill timer setting will be determined by 16-24 shift, Friday. Base value leakage will normally be the weekly average leakage, except after an outage, in which case the last base value determined at normal operating pressure will be used until the first 24-hour average leakage can be obtained after return to normal operating pressure. After determination of the new base value leakage (BVL), calculate and readjust the floor drain sump fill timer setting to either 350/(BVL + 2) or 150 minutes, whichever is less. These values should be carried forward to the new week's DSR.

Calculate drywell leakage daily at approximately 0900.
Sump leak calculations required when greater than 212°F coolant temperature and irradiated fuel in reactor vessel (conditions 1, 2, and 3) - reference - Technical Specification 4.4.3.2a and AOP-01.

xxx A channel functional test is required once/31 days per Technical Specification 4.4.3.1.b in condition 1, 2, or 3. At least one value other than zero during three consecutive readings constitutes an acceptable functional test.

SHIFT 00-04

UNIT 2

BRUNSWICK STEAM ELECTRIC PLANT
DAILY SURVEILLANCE REPORT
CONTROL OPERATORS

WEEK OF __________ TO __________

DRYWELL LEAKAGE CALCULATION

	SAT	SUN	MON	TUES	WED	THURS	FRI
BASE VALUE LEAKAGE = ____________gpm*							
SUMP FILL TIMER SETTING = __________ minutes*							
MANUALLY PUMP FLOOR DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP).							
CALCULATE TIME INTERVAL (MINUTES) SINCE SUMP WAS LAST MANUALLY PUMPED.							
RECORD INTEGRATOR READING.							
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM TIME SUMP WAS LAST MANUALLY PUMPED (LEAKAGE).							
CALCULATE FLOOR DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).**							
MANUALLY PUMP EQUIPMENT DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP).							
CALCULATE TIME INTERVAL (MINUTES) SINCE SUMP WAS LAST MANUALLY PUMPED.							
RECORD INTEGRATOR READING.							
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM TIME SUMP WAS LAST MANUALLY PUMPED (LEAKAGE).							
CALCULATE EQUIPMENT DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).							
FLOOR DRAIN LEAK RATE. (If zero, see xxx below)	4.25 gpm						
EQUIPMENT DRAIN LEAK RATE. (If zero, see xxx below)	20.58 gpm						
LEAK RATE TO DRYWELL.	24.83 gpm						
INITIALS: (Control Operator/Shift Foreman)	/	/	/	/	/	/	/

* NOTE: Base value leakage and sump fill timer setting will be determined by 16-24 shift, Friday. Base value leakage will normally be the weekly average leakage, except after an outage, in which case the last base value determined at normal operating pressure will be used until the first 24-hour average leakage can be obtained after return to normal operating pressure. After determination of the new base value leakage (BVL), calculate and readjust the floor drain sump fill timer setting to either 350/(BVL + 2) or 150 minutes, whichever is less. These values should be carried forward to the new week's DSR.

Calculate drywell leakage daily at approximately 0900.
Sump leak calculations required when greater than 212°F coolant temperature and irradiated fuel in reactor vessel (conditions 1, 2, and 3) - reference - Technical Specification 4.4.3.2a and AOP-01.

xxx A channel functional test is required once/31 days per Technical Specification 4.4.3.1.b in condition 1, 2, or 3. At least one value other than zero during three consecutive readings constitutes an acceptable functional test.

UNIT ____2

SHIFT 04-08

BRUNSWICK STEAM ELECTRIC PLANT
DAILY SURVEILLANCE REPORT
CONTROL OPERATORS

WEEK OF__________TO__________

DRYWELL LEAKAGE CALCULATION

	SAT	SUN	MON	TUES	WED	THURS	FRI
BASE VALUE LEAKAGE = ____________gpm*							
SUMP FILL TIMER SETTING = __________ minutes*							
MANUALLY PUMP FLOOR DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP).							
CALCULATE TIME INTERVAL (MINUTES) SINCE SUMP WAS LAST MANUALLY PUMPED.							
RECORD INTEGRATOR READING.							
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM TIME SUMP WAS LAST MANUALLY PUMPED (LEAKAGE).							
CALCULATE FLOOR DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).**							
MANUALLY PUMP EQUIPMENT DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP).							
CALCULATE TIME INTERVAL (MINUTES) SINCE SUMP WAS LAST MANUALLY PUMPED.							
RECORD INTEGRATOR READING.							
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM TIME SUMP WAS LAST MANUALLY PUMPED (LEAKAGE).							
CALCULATE EQUIPMENT DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).							
FLOOR DRAIN LEAK RATE. (If zero, see xxx below)	3.75 gpm						
EQUIPMENT DRAIN LEAK RATE. (If zero, see xxx below)	21 gpm						
LEAK RATE TO DRYWELL.	24.75 gpm						
INITIALS: (Control Operator/Shift Foreman)	/	/	/	/	/	/	/

* NOTE: Base value leakage and sump fill timer setting will be determined by 16-24 shift, Friday. Base value leakage will normally be the weekly average leakage, except after an outage, in which case the last base value determined at normal operating pressure will be used until the first 24-hour average leakage can be obtained after return to normal operating pressure. After determination of the new base value leakage (BVL), calculate and readjust the floor drain sump fill timer setting to either 350/(BVL + 2) or 150 minutes, whichever is less. These values should be carried forward to the new week's DSR.

Calculate drywell leakage daily at approximately 0900.
Sump leak calculations required when greater than 212°F coolant temperature and irradiated fuel in reactor vessel (conditions 1, 2, and 3) - reference - Technical Specification 4.4.3.2a and AOP-01.

xxx A channel functional test is required once/31 days per Technical Specification 4.4.3.1.b in condition 1, 2, or 3. At least one value other than zero during three consecutive readings constitutes an acceptable functional test.

SHIFT 08-12

BRUNSWICK STEAM ELECTRIC PLANT
DAILY SURVEILLANCE REPORT
CONTROL OPERATORS

UNIT 2

WEEK OF __________ TO __________

DRYWELL LEAKAGE CALCULATION

	SAT	SUN	MON	TUES	WED	THURS	FRI
BASE VALUE LEAKAGE = ____________ gpm*							
SUMP FILL TIMER SETTING = __________ minutes*							
MANUALLY PUMP FLOOR DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP).							
CALCULATE TIME INTERVAL (MINUTES) SINCE SUMP WAS LAST MANUALLY PUMPED.							
RECORD INTEGRATOR READING.							
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM TIME SUMP WAS LAST MANUALLY PUMPED (LEAKAGE).							
CALCULATE FLOOR DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).**							
MANUALLY PUMP EQUIPMENT DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP).							
CALCULATE TIME INTERVAL (MINUTES) SINCE SUMP WAS LAST MANUALLY PUMPED.							
RECORD INTEGRATOR READING.							
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM TIME SUMP WAS LAST MANUALLY PUMPED (LEAKAGE).							
CALCULATE EQUIPMENT DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).							
FLOOR DRAIN LEAK RATE. (If zero, see xxx below)	4.3 gpm						
EQUIPMENT DRAIN LEAK RATE. (If zero, see xxx below)	22.25 gpm						
LEAK RATE TO DRYWELL.	26.55 gpm						
INITIALS: (Control Operator/Shift Foreman)	/	/	/	/	/	/	/

* NOTE: Base value leakage and sump fill timer setting will be determined by 16-24 shift, Friday. Base value leakage will normally be the weekly average leakage, except after an outage, in which case the last base value determined at normal operating pressure will be used until the first 24-hour average leakage can be obtained after return to normal operating pressure. After determination of the new base value leakage (BVL), calculate and readjust the floor drain sump fill timer setting to either 350/(BVL + 2) or 150 minutes, whichever is less. These values should be carried forward to the new week's DSR.

Calculate drywell leakage daily at approximately 0900.
Sump leak calculations required when greater than 212°F coolant temperature and irradiated fuel in reactor vessel (conditions 1, 2, and 3) - reference - Technical Specification 4.4.3.2a and AOP-01.

xxx A channel functional test is required once/31 days per Technical Specification 4.4.3.1.b in condition 1, 2, or 3. At least one value other than zero during three consecutive readings constitutes an acceptable functional test.

SHIFT 12-16 ________

BRUNSWICK STEAM ELECTRIC PLANT
DAILY SURVEILLANCE REPORT
CONTROL OPERATORS

UNIT 2

WEEK OF __________ TO __________

DRYWELL LEAKAGE CALCULATION

BASE VALUE LEAKAGE = ______________ gpm* SUMP FILL TIMER SETTING = __________ minutes*	SAT	SUN	MON	TUES	WED	THURS	FRI
MANUALLY PUMP FLOOR DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP).							
CALCULATE TIME INTERVAL (MINUTES) SINCE SUMP WAS LAST MANUALLY PUMPED.							
RECORD INTEGRATOR READING.							
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM TIME SUMP WAS LAST MANUALLY PUMPED (LEAKAGE).							
CALCULATE FLOOR DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).**							
MANUALLY PUMP EQUIPMENT DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP).							
CALCULATE TIME INTERVAL (MINUTES) SINCE SUMP WAS LAST MANUALLY PUMPED.							
RECORD INTEGRATOR READING.							
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM TIME SUMP WAS LAST MANUALLY PUMPED (LEAKAGE).							
CALCULATE EQUIPMENT DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).							
FLOOR DRAIN LEAK RATE. (If zero, see xxx below)	4.48 gpm						
EQUIPMENT DRAIN LEAK RATE. (If zero, see xxx below)	24.33 gpm						
LEAK RATE TO DRYWELL.	28.91 gpm						
INITIALS; (Control Operator/Shift Foreman)	/	/	/	/	/	/	/

* **NOTE**: Base value leakage and sump fill timer setting will be determined by 16-24 shift, Friday. Base value leakage will normally be the weekly average leakage, except after an outage, in which case the last base value determined at normal operating pressure will be used until the first 24-hour average leakage can be obtained after return to normal operating pressure. After determination of the new base value leakage (BVL), calculate and readjust the floor drain sump fill timer setting to either 350/(BVL + 2) or 150 minutes, whichever is less. These values should be carried forward to the new week's DSR.

Calculate drywell leakage daily at approximately 0900.
Sump leak calculations required when greater than 212°F coolant temperature and irradiated fuel in reactor vessel (conditions 1, 2, and 3) - reference - Technical Specification 4.4.3.2a and AOP-01.

xxx A channel functional test is required once/31 days per Technical Specification 4.4.3.1.b in condition 1, 2, or 3. At least one value other than zero during three consecutive readings constitutes an acceptable functional test.

SHIFT 16-20 ________

BRUNSWICK STEAM ELECTRIC PLANT
DAILY SURVEILLANCE REPORT
CONTROL OPERATORS

UNIT 2

WEEK OF __________ TO __________

DRYWELL LEAKAGE CALCULATION

	SAT	SUN	MON	TUES	WED	THURS	FRI
BASE VALUE LEAKAGE = ____________ gpm*							
SUMP FILL TIMER SETTING = __________ minutes*							
MANUALLY PUMP FLOOR DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP).							
CALCULATE TIME INTERVAL (MINUTES) SINCE SUMP WAS LAST MANUALLY PUMPED.							
RECORD INTEGRATOR READING.							
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM TIME SUMP WAS LAST MANUALLY PUMPED (LEAKAGE).							
CALCULATE FLOOR DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).**							
MANUALLY PUMP EQUIPMENT DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP).							
CALCULATE TIME INTERVAL (MINUTES) SINCE SUMP WAS LAST MANUALLY PUMPED.							
RECORD INTEGRATOR READING.							
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM TIME SUMP WAS LAST MANUALLY PUMPED (LEAKAGE).							
CALCULATE EQUIPMENT DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).							
FLOOR DRAIN LEAK RATE. (If zero, see xxx below)	5.2 gpm						
EQUIPMENT DRAIN LEAK RATE. (If zero, see xxx below)	19.33 gpm						
LEAK RATE TO DRYWELL.	24.53 gpm						
INITIALS; (Control Operator/Shift Foreman)	/	/	/	/	/	/	/

* **NOTE**: Base value leakage and sump fill timer setting will be determined by 16-24 shift, Friday. Base value leakage will normally be the weekly average leakage, except after an outage, in which case the last base value determined at normal operating pressure will be used until the first 24-hour average leakage can be obtained after return to normal operating pressure. After determination of the new base value leakage (BVL), calculate and readjust the floor drain sump fill timer setting to either 350/(BVL + 2) or 150 minutes, whichever is less. These values should be carried forward to the new week's DSR.

Calculate drywell leakage daily at approximately 0900.
Sump leak calculations required when greater than 212°F coolant temperature and irradiated fuel in reactor vessel (conditions 1, 2, and 3) - reference - Technical Specification 4.4.3.2a and AOP-01.

xxx A channel functional test is required once/31 days per Technical Specification 4.4.3.1.b in condition 1, 2, or 3. At least one value other than zero during three consecutive readings constitutes an acceptable functional test.

UNIT ____2____

SHIFT ___20-24___

BRUNSWICK STEAM ELECTRIC PLANT
DAILY SURVEILLANCE REPORT
CONTROL OPERATORS

WEEK OF __________ TO __________

ATTACHED ARE THE FOLLOWING TECHNICAL SPECIFICATIONS EXCERPTS FOR USE IN ANSWERING QUESTIONS 8.16 thru 8.19

1. 3/4.0 Applicability
   - 3.0.1 thru 3.0.5

2. 3/4.5 Emergency Core Cooling Systems
   - 3.5.1 HPCI
   - 3.5.2 ADS
   - 3.5.3.1 CS
   - 3.5.3.2 LPCI
   - 3.5.4 Suppression Pool

3. 3/4.6 Containment Systems
   - 3.6.2.1 Suppression Chamber
   - 3.6.2.2 Suppression Pool Cooling
   - 3.6.6.2 CAD System
   - 3.6.6.3 Oxygen Concentration

4. 3/4.8 Electrical Power System
   - 3.8.1.2 AC Power (SD)
   - 3.8.2.2 AC Distribution (SD)

LIMITING CONDITION FOR OPERATION

3.0.1 Limiting Conditions for Operation and ACTION requirements shall be applicable during the OPERATIONAL CONDITIONS or other states specified for each specification.

3.0.2 Adherence to the requirements of the Limiting Condition for Operation and associated ACTION within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

3.0.3 In the event a Limiting Condition for Operation and/or associated ACTION requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours unless corrective measures are completed that permit operation under the permissible ACTION statements for the specified time interval as measured from initial 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.

3.0.4 Entry into an OPERATIONAL CONDITION or other specified applicability state shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION statements unless otherwise excepted. This provision shall not prevent passage through OPERATIONAL CONDITIONS required to comply with ACTION requirements.

3.0.5 When a system, subsystem, train, component, or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal 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 normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystems(s), train(s), component(s), and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least HOT SHUTDOWN within 6 hours, and in at least COLD SHUTDOWN within the following 30 hours. This specification is not applicable in Conditions 4 or 5.

3/4.5.1 HIGH PRESSURE COOLANT INJECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.1 The High Pressure Coolant Injection (HPCI) system shall be OPERABLE with:

a. One OPERABLE high pressure coolant injection pump, and

b. An OPERABLE flow path capable of taking suction from the suppression pool and transferring the water to the pressure vessel.

APPLICABILITY: CONDITIONS 1, 2, and 3 with reactor vessel steam dome pressure > 113 psig.

ACTION:

a. With the HPCI system inoperable, POWER OPERATION may continue provided the ADS, CSS, and LPCI systems are OPERABLE; restore the inoperable HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the surveillance requirements of Specification 4.5.1 not performed at the required frequencies due to low reactor steam pressure, the provisions of Specification 4.0.4 are not applicable provided the appropriate surveillance is performed within 48 hours after reactor steam pressure is adequate to perform the tests.

SURVEILLANCE REQUIREMENTS

4.5.1 The HPCI shall be demonstrated OPERABLE:

a. At least once per 31 days by:

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

TSI 84-07
RETYPED TECH. SPECS.
Updated Thru. Amend. 78

SURVEILLANCE REQUIREMENTS (Continued)

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

b. At least once per 92 days, by verifying that the system develops a flow of at least 4250 gpm for a system head corresponding to a reactor pressure ≥ 1000 psig when steam is being supplied to the turbine at 1000, +20, -18, psig.

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel is excluded from this test.

2. Verifying that the system develops a flow of at least 4250 gpm for a system head corresponding to a reactor pressure of ≥ 165 psig when steam is being supplied to the turbine at 165, ± 15, psig.

3. Verifying that the suction for the HPCI system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank low water level signal or suppression pool high water level signal.

See TSI 84-07
RETYPED TECH. SPECS.
Updated Thru. Amend. 78

LIMITING CONDITION FOR OPERATION

3.5.2 The Automatic Depressurization System (ADS) shall be OPERABLE with at least seven OPERABLE ADS valves.

APPLICABILITY: CONDITIONS 1, 2, and 3 with reactor vessel steam dome pressure > 113 psig.

ACTION:

a. With one ADS valve inoperable, POWER OPERATION may continue provided the HPCI, CSS, and LPCI systems are OPERABLE; restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With two or more ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

c. With the Surveillance Requirement of Specification 4.5.2.b not performed at the required interval due to low reactor steam pressure. the provisions of Specification 4.0.4 are not applicable provided the appropriate surveillance is performed within 12 hours after reactor vessel steam pressure is adequate to perform the tests.

SURVEILLANCE REQUIREMENTS

4.5.2 The ADS shall be demonstrated OPERABLE at least once per 18 months by:

a. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

b. Manually opening each ADS valve when the reactor steam dome pressure is ≥ 100 psig and observing that either;

   1. The control valve or bypass valve position responds accordingly, or

   2. There is a corresponding change in the measured steam flow.

RETYPED TECH. SPECS.
Updated Thru. Amend.

11

CORE SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. One pump, and

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

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

2. In CONDITION 4 or 5*:

a) From the suppression pool, or

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

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

ACTION:

a. In CONDITION 1, 2, or 3:

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

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

* The core spray system is not required to be OPERABLE when the suppression pool is inoperable, provided that the reactor vessel head is removed and the cavity is flooded, the spent fuel pool gates are removed, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

RETYPED TECH. SPECS.
Updated Thru. Amend.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

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

b. In CONDITION 4 or 5*:

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

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

3. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.5.3.1 Each CSS subsystem shall be demonstrated OPERABLE:

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

b. At least once per 31 days by:

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

* The core spray system is not required to be OPERABLE when the suppression pool is inoperable provided that the reactor vessel head is removed and the cavity is flooded, the spent fuel pool gates are removed, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS (Continued)

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

c. **At least once per 92 days* by:**

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

2. Performing a CHANNEL CALIBRATION of the core spray header ΔP instrumentation (E21-dPIS-N004A,B) and verifying the setpoint to be 5, ±1.5, psid greater than the normal indicated ΔP.

d. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel is excluded from this test.

* The surveillance test required by this license in Appendix A, paragraph 4.5.3.1.C.1, regarding the flow test of the core spray system may be postponed during the current refueling outage (Reload 5) until within 48 hours after restoration of the suppression chamber to operable status but in any case no later than November 15, 1984.

LIMITING CONDITION FOR OPERATION

3.5.3.2 Two independent Low Pressure Coolant Injection (LPCI) subsystems of the residual heat removal system shall be OPERABLE with each subsystem comprised of:

a. Two pumps,

b. An OPERABLE flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

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

ACTION:

a. In CONDITION 1, 2, or 3:

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

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

3. With the LPCI system cross-tie valve open or power not removed from the valve operator, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

b. In CONDITION 4* or 5* with one or more LPCI subsystems inoperable, take the ACTION required by Specification 3.5.3.1. The provisions of Specification 3.0.3 are not applicable.

*Not applicable when two CSS subsystems are OPERABLE per Specification 3.5.3.1.

4.5.3.2 Each LPCI subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by:

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

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

3. Verifying that the subsystem cross-tie valve is closed with power removed from the valve operator.

b. At least once per 92 days by verifying each pair of LPCI pumps discharging to a common header can be started from the control room and develops a total flow of at least 17,000 gpm against a system head corresponding to a reactor vessel pressure of $\geq$ 20 psig.

c. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel is excluded from this test.

LIMITING CONDITION FOR OPERATION

3.5.4 The suppression pool shall be OPERABLE with a minimum water level ≥-31 inches except the suppression pool may be inoperable:

a. In OPERATIONAL CONDITION 4, provided that:

1. The reactor mode switch is locked in the Shutdown position, and

2. The core spray system is OPERABLE per Specification 3.5.3.1 with an OPERABLE flow path capable of taking suction from the OPERABLE condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

b. In OPERATIONAL CONDITION 5, provided that:

1. The reactor mode switch is locked in the Refuel position, and

2. The core spray system is OPERABLE per Specification 3.5.3.1 with an OPERABLE flow path capable of taking suction from the OPERABLE condensate storage tank, and transferring the water through the spray sparger to the reactor vessel, or

3. The reactor vessel head is removed and the cavity is flooded, the spent fuel pool gates are removed, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

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

ACTION:

a. In CONDITIONS 1, 2, or 3 with the water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In CONDITIONS 4 or 5, with the suppression pool inoperable and the above conditions not satisfied, suspend all operations in the reactor vessel, all positive reactivity changes, and all operations that have a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The suppression pool shall be determined OPERABLE by verifying the water level to be within the limit at least once per 12 hours.

4.5.4.2 The above conditions shall be verified to be satisfied prior to making the suppression pool inoperable and at least once per 12 hours thereafter until the suppression pool is restored to OPERABLE status.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 87,600 ft^3 and 89,600 ft^3, equivalent to a level between -27 inches and -31 inches, and a

2. Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

   a) 105°F during testing which adds heat to the suppression chamber.

   b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

   c) 120°F with the main steam line isolation valves closed following a scram.

b. Two OPERABLE suppression chamber water temperature instrumentation channels with a minimum of 11 operable RTD inputs per channel.

c. A total leakage from the drywell to the suppression chamber of less than the equivalent leakage through a 1-inch diameter orifice at a differential pressure of 1 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than 95°F, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

LIMITING CONDITIONS FOR OPERATION (Continued)

ACTION: (Continued)

1. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression chamber average water temperature greater than 110°F manually scram the reactor and operate at least one residual heat removal loop in the suppression pool cooling mode.

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

c. With one suppression chamber water temperature instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or verify suppression chamber water temperature to be within the limits at least once per 12 hours.

d. With both suppression chamber water temperature instrumentation channels inoperable, restore at least one inoperable temperature instrumentation channel to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:

1. At least once per 5 minutes during testing which [illegible] heat to the suppression chamber, by verifying the suppres[illegible] chamber average water temperature to be less than or eq[illegible] 105°F.

2. At least once per hour when suppression chamber average water temperature is greater than 95°F, by verifying:

a) Suppression chamber average water temperature to be less than or equal to 110°F, and

b) THERMAL POWER to be less than or equal [illegible] RATED THERMAL POWER after suppression chamber average water temperature has exceeded 95°F for more than 24 hours.

3. At least once per 30 minutes following a scram [illegible] suppression chamber average water temperature greater [illegible] by verifying suppression chamber average water temperature less than or equal to 120°F.

c. By an external visual examination of selected emergency [illegible] cooling system suction line penetrations of the suppression chamber enclosure prior to taking the reactor from COLD SHUTDOWN after safety/relief valve operation with the suppression chamber average water temperature greater than or equal to 160°F and reactor coolant system pressure greater than 200 psig.

d. By verifying at least two suppression chamber water temperature instrumentation channels OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours.

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 18 months (550 days).

with the temperature alarm setpoint for high water temperature less than or equal to 95°F. (CAC-TE-4426-2 thru 13: CAC-TY-4426-1; CAC-TR-4426-1) (CAC-TE-4426-15 thru 26; [illegible] CAC-TR-4426-2)

e. At least once per 18 months by:

1. A visual inspection of the [illegible] suppression chamber and [illegible] enclosure.

SURVEILLANCE REQUIREMENTS (Continued)

2. Conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 1 psig and verifying that the differential pressure does not decrease by more than 0.25 inches of water per minute for a 10 minute period.

3.6.2.2 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent cooling loops, each loop consisting of two pumps and one heat exchanger.

APPLICABILITY: CONDITIONS 1, 2, and 3.

ACTION:

a. With one RHR suppression pool cooling loop inoperable, operation may continue and the provisions of Specification 3.0.4 are not applicable; restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both RHR suppression pool cooling loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 92 days by verifying that each RHR pump can be started from the control room and develops a flow of at least 10,300 gpm against a system head corresponding to a reactor pressure of $\geq$ 20 psig on recirculation flow.

CONTAINMENT ATMOSPHERE DILUTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.2 The containment atmosphere dilution (CAD) system shall be OPERABLE with:

a. An OPERABLE flow path capable of supplying nitrogen to the drywell, and

b. A minimum supply of 4350 gallons of liquid nitrogen.

APPLICABILITY: CONDITION 1*.

ACTION:

With the CAD system inoperable, restore the CAD system to OPERABLE status within 31 days or be in at least STARTUP within the next 8 hours. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.6.2 The CAD system shall be demonstrated to be OPERABLE:

a. At least once per 31 days by verifying that:

1. The system contains a minimum of 4350 gallons of liquid nitrogen, and

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

b. At least once per 18 months by:

1. Cycling each power-operated (excluding automatic) valve in the flow path through at least one complete cycle of full travel, and

2. Verifying that each automatic valve in the flow path actuates to its correct position on a Group 2 and 6 isolation test signal.

*When oxygen concentration is required to be < 4% per Specification 3.6.6.3.

LIMITING CONDITION FOR OPERATION

3.6.6.3* The primary containment atmosphere oxygen concentration shall be less than 4% by volume during the period from:

a. Within 24 hours after THERMAL POWER > 15% of RATED THERMAL POWER, to

b. Within 24 hours prior to a scheduled reduction of THERMAL POWER to < 15% of RATED THERMAL POWER.

APPLICABILITY: CONDITION 1.

ACTION:

With the oxygen concentration in the primary containment exceeding the limit, be in at least START-UP within 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.3 The oxygen concentration in the primary containment shall be verified to be within the limit within 24 hours after THERMAL POWER > 15% of RATED THERMAL POWER and at least once per 7 days thereafter.

*For the period commencing at 0630 on June 29, 1981, a temporary exemption is allowed to operate BSEP-2 in Condition 1 with containment oxygen concentration exceeding 4% by volume for 72 hours.

SHUTDOWN OF BOTH UNITS

LIMITING CONDITION FOR OPERATION

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

a. One circuit per Unit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two diesel generators, as required to operate ECCS systems in accordance with Specifications 3.5.3.1 and 3.5.3.2:

1. Each with a separate:

   a) Engine-mounted fuel tank containing a minimum of 100 gallons of fuel,

   b) Day fuel tank containing a minimum of 22,650 gallons of fuel, and

   c) Fuel transfer pump.

2. With a fuel storage tank containing a minimum of 37,000 gallons of fuel.

APPLICABILITY: CONDITIONS 4 and 5.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving irradiated fuel handling, CORE ALTERATIONS, positive reactivity changes, or operations that have the potential of draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE per surveillance requirements of Specifications 4.8.1.1.1 and 4.8.1.1.2, except for the requirement of 4.8.1.1.2.a.5.

A.C. DISTRIBUTION - SHUTDOWN OF BOTH UNITS

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical buses shall be OPERABLE for each unit but aligned to an OPERABLE diesel generator, as required to operate ECCS systems in accordance with Specifications 3.5.3.1 and 3.5.3.2:

1 - 4160-volt Emergency Bus,

1 - 480-volt Emergency Bus,

2 - 120-volt A.C. Vital Buses.

APPLICABILITY: CONDITIONS 4 and 5.

ACTION:

With less than the above complement of A.C. buses OPERABLE, suspend all operations involving irradiated fuel handling, CORE ALTERATIONS, positive reactivity changes, or operations that have the potential of draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. buses shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.

RETYPED TECH. SPECS.
Updated Thru. Amend.

ANSWERS - SECTION 5

5.1 d (1.0)

REF:
EIH: L-RQ-602 (9)
BSEP: L/P 02-2/3-A, pp 54, 83

5.2 c (1.0)

REF:
EIH: GPNT, Vol. II, Chapter 3.E; Nuclear Power Reactor Instrumentation Systems Handbook, Harrer & Beckerly, p 44.
BSEP: L/P 25-2/3-C, pp 3, 4

5.3 a. By observing the Full-in and Full-out travel lights (the operator could determine if geometric distortion had occurred. Inability to conduct full detector movement would indicate that internal misconfiguration had occurred). (1.0)

b. By observing the neutron level while moving the nuclear instrumentation. A significantly HIGHER (approximately 300 times) count rate would be seen for the UNVOIDED areas of the core as opposed to the VOIDED. (1.0)

REF:
EIH: L-RQ-540 (M.8)
BSEP: L/P 05-2/3-C, p 25; L/P 02-2/3-A

5.4 c (1.0)

REF:
EIH: GPNT, Vol. VII, Chapter 10.1-78; HNP-2-1001
BSEP: 02-2/3-A, pp 156-159

5.5 b (1.0)

REF:
EIH: L-RQ-606 (5)
BSEP: L/P 02-2/3-A, pp 176-177

5.6 a. lambda = Ln 2/$_{T_{1/2}}$ = .693/55.6 = .0125 sec E -1

(CALC NOT REQUIRED)

T=1/-lambda = 1/-.0125 = - 80 sec

After the initial prompt drop, power cannot decrease faster than the longest lived delayed neutrons. (1.0)

b. Shorter [0.5] The initial prompt drop will only be due to prompt neutrons [0.5] -OR- Decay of short lived precursors [0.5] (1.0)

REF:
EIH: GPNT, Vol. VII, Chapter 10.1
BSEP: L/P 02-2/3-A, pp 122, 134, Fig 51

5.7 a. (2) - Starting Pump 2 (0.5)
b. Curve B (0.5)
c. INCREASE (0.5)
d. Curve C (0.5)

REF:
Pump Laws

5.8 d (1.0)

REF:
Mollier Diagram (Steam Tables)

5.9 b (1.0)

REF:
EIH: GPNT, Vol. VII, Chapter 10.1-83-86
BSEP: L/P 02-2/3-A, pp 172-176

5.10 b (1.0)

REF:
EIH: Thermodynamics L/P, pp 52-56
BSEP: L/P 04-2/3-E, pp 66-78

5.11 c (1.0)

REF:
Steam Tables

5.12 75% CONTROL ROD DENSITY (0.5)

EIH: (The increased Control Rod Density causes greater competition for the thermal neutrons; this necessitates greater pin power for the same net power output.) Higher pin power results in a greater Void Fraction which causes a more negative coefficient.

BSEP: With a greater rod density, a greater number of neutrons are "lost" to the control rods (increased leakage). Thus, a change in rod density affects reactivity more, by allowing increased absorbtion by other fuel bundles. (Can also explain why low rod density does not have a large reactivity effect, since the leakage to other fuel bundles is already so large.) (0.5)

REF:
EIH: Reactor Physics L/P, pp 1.7-9, 10, & 13.
BSEP: L/P 02-2/3-A, pp 141-143

5.13 a. 2 (1.0)
b. 2 (1.0)

REF:
General Electric NEDE 21493 (Rev. 5)
EIH: GPNT, STA Training Manual, Section 9
BSEP: L/P 06-2/3-B, p 1-16

5.14 a. INCREASE
b. INCREASE
c. DECREASE
d. ~~REMAIN THE SAME~~ JFM INCREASE ; BSEP Simulator (0.5 ea)
indicates an ≅ 100,000 #/HR steam Flow Decrease to Mn. Turbine and a Decrease of 1°F in FW Temperature on 5/25/85

REF:
EIH: GPNT, Vol. VII, Chaper 10.2
BSEP: GPC, "Heat Transfer, Thermodynamics, Fluid Flow", pp 235-241

5.15 a (1.0)

REF:
BSEP: L/P 04-2/3-B, pp 72-75

5.16 a. Due to Increased voiding [.25] and flow escaping from the riser to the annulus [.25].
b. Due to reduced core flow.
c. Follows steam flow (as EHC closes CVs to control pressure).
d. Follows steam flow decrease [.25] and level increase [.25].
e. Follows power reduction. (0.5 ea.)

REF:
BSEP: Transient L/P, Transient HXY-7

5.17 a. "Swell" due to increased voiding (from pressure decrease).
b. Due to increased voiding.
c. SCRAM due to turbine trip.
d. GROUP I Isolation on low steam pressure (in RUN Mode).
e. Effects of HPCI (0.4) and RCIC (0.1) injection. (0.5 ea.)

REF:
BSEP: Transient L/P, Transient HXY-7

5.18 b (1.0)

REF:
General Electric, NEDE-24810 (Jun 81)

ANSWERS - SECTION 6

6.1 c (1.0)

Ref: BSEP SD-03 Fig. 3-3

6.2 See attached trace - Figure 6.2A (1.0)

Ref: BSEP RTN 2A, Plant Mod 81-291; GE SIL 200 (Suppl. 2)

6.3 b (1.0)

Ref: BSEP SSM 10-3-A, Figure 5

6.4 a. Increase (50%) [0.5]; Dual/Master limiter [0.5] (1.0)

b. Increase [0.5]; scoop tube positioning unit (full range, LS/mech. stop) [0.5] (1.0)

Ref: BSEP SSM 10-3-A, RTN 010

6.5 a. 03 b. 04

a.	b.
03	04
22-55	18-03
46-55	46-55
34-27	Blank

(.25 each/2.0)

Ref: BSEP SSM 27-2-B

6.6 a. Closed (0.5)

b. As Is (0.5)

c. Closed (0.5)

Ref: BSEP RTN F&C; BSEP NRC Exams 4/83, Requal & 5/84

6.7 a (1.0)

Ref: BSEP RTN 033, 012; SD 26.2

6.8 - wet (0.5)
- Assymmetrically (0.5)

Ref: BSEP SSM 25-2-C, RTN 029

6.9 a (1.0)

Ref: BSEP SSM 25-2-D and 25-2-C, RTN 029

6.10 b (1.0)

Ref: BSEP SSM 25-2-C

6.11 - 50% rod density (.33)
- to 22% power (.33)
- turbine first stage pressure (.34)

Ref: BSEP SSM 27-2-C

6.12 a. Recirc loop flow elements (pump discharge) (0.5)

b. Only the 120% fixed scram [0.5] because the flow biased scram incorporates a time delay (≈6 seconds, representative of fuel temperature transient time) [0.5] (1.0)

REF: BSEP SSM 25-2-D

6.13 a .99 (0.5)

Ref: BSEP SSM 25-2-D

6.14 c (1.0)

Ref: BSEP GP-03, SD-26.2

6.15 c (1.0)

Ref: BSEP GP-03

6.16 - RHR pump room cooler (0.5)
- RHR pump seal Hxers (0.5)
- Core Spray Pump Room Coolers (0.5)

Ref: BSEP SD-43

6.17 b (1.0)

Ref: BSEP RTN 022, SD-19

6.18 c (1.0)

Ref: BSEP SD-16, 12

6.19 d. (1.0)

Ref: BSEP SSM 14-3-D

6.20 c (1.0)

Ref: BSEP SSM 14-3-E

6.21 a. - 4 (1.0)

b. - 3 (1.0)

REF: BSEP RTN 008, SSM 08-2-A

6.22 a (1.0)

Ref: BSEP RTN 028

6.23 d (1.0)

Ref: BSEP RTN 026

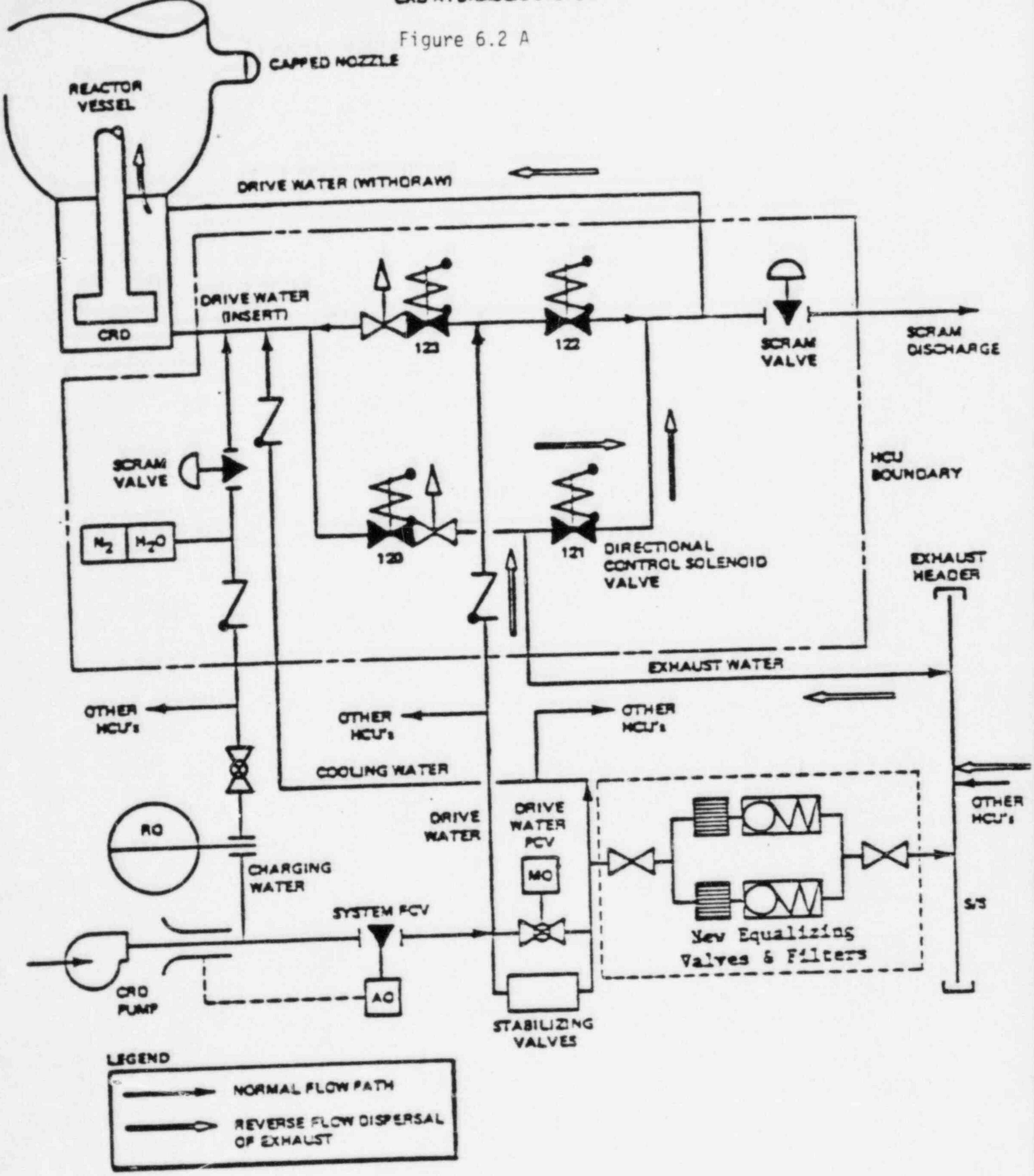
6.24 - Alternate power source (480/120 VAC transformer) (0.5)

- SS1 shuts [.25] and SS2 opens [.25] (0.5)

Ref: BSEP SSM 20-2-F

SIMPLIFIED SCHEMATIC OF MODIFIED
CRD HYDRAULIC SYSTEM

Figure 6.2 A



ANSWERS - Section 7

7.1 a. (As the shell warms), first stage pressure increases [0.5]. When shell pressure reaches 155 psig, the reactor scrams on TSV closure at >30% indicated power [0.5]. (1.0)

b. 4, 2, 1, 6, 5, 3 (.25 ea/1.5)

c. Trip the turbine instantly [0.5] and place on the turning gear. (0.5)

REF: BSEP GP-02, 03; OP-26

7.2 a. - the reactor shall be shutdown (0.5)
- (the reactor SU shall be discontinued until) an assessment can be performed by the NE (and approved by SOS) (0.5)

b. - control rod inserted to achieve a stable period of $\geq$100 seconds (0.5)

REF: BSEP GP-02

7.3 - 2 (0.5)

- Misoperation in the automatic mode is confirmed (0.5)

- Adequate core cooling is assured (0.5)

REF: BSEP EOP-01

7.4 a. - Warming up and pressurizing steamlines (0.5)
- Water hammer in the steam lines (0.5)

b. Operation <2000 RPM should be minimized to ensure adequate oil pressure to operate the governing valve [0.5] and to prevent exhaust check valve chattering [0.5]. (1.0)

REF: BSEP OP-16

7.5 . Such operations can shorten seal life. (0.5)

b. 1%, 5%, 10% (1.5)

REF: BSEP GP-04, P.4
BSEP GP-05, P.5

7.6 a. To prevent regenerative Hx tube damage [0.5] due to rapid cooldown rate [0.5]. (1.0)

b. To minimize thermal duty on the feedwater nozzles. (1.0)

REF: BSEP OP-14, P.11, 31

7.7 b. (1.0)

REF: BSEP OP-21

7.8 The Fuel Pool gates by themselves provide no shielding for personnel working in the Reactor Cavity Area. (1.0)

REF: BSEP GP-07

7.9 Because spraying the drywell may decrease containment pressure below atmospheric at a rate beyond the capacity of the RB-to-suppression chamber vacuum breakers, resulting in negative containment pressures in excess of design. (1.0)

REF: BSEP EOP-01/UG

7.10 a.

REF: BSEP AOP-02.1 (1.0)

7.11 c

REF: BSEP AOP-04.4 (1.0)

7.12 b

REF: BSEP AOP-30.0 (1.0)

7.13 a.-1
b.-3
c.-4
d.-2 (0.5 ea./2.0)

REF: BSEP AOP-20.0

7.14
- indications of criticality observed on SRMs
- loss of communications between Control Room and Refuel Floor
- malfunctioning or failure of > one SRM or IRM channel
- accidental dropping or damaging of a fuel element (0.5 ea./2.0)

REF: BSEP FH-11

7.15 a. closed (0.5)

b. 1 (or 2) open (0.5)

c. 1 RHR loop injecting to Rx (.25)
1 RHR loop in suppression pool cooling (.25)

d. 107 to 164 psig > suppression chamber pressure (0.5)

e. Rx level to main steam line elevation or >254" (0.5)

REF: BSEP AOP-15.0

7.16
- Suppression Pool temperture [.4] > 95°F [.1] (.5)
- DW average temperature [.4] > 135°F [.1] (.5)
- DW pressure [.4] > 2 psig [.1] (.5)
- Suppression Pool level [.4] <-31" [.1] (.5)
- Suppression Pool level [.4] >-27" [.1] (.5)

(4 req. at .5 ea 2.0)

REF: BSEP EOP-01-CCP

ANSWERS - SECTION 8

8.1 c. (1.0)

REF: BSEP OI-04

8.2 b. (1.0)

REF: BSEP OI-04

8.3 d. (1.0)

REF: BSEP PEP 02.2

8.4 b. (1.0)

REF: BSEP MP-14

8.5 A departure may change the intent of the procedure [0.5] while a deviation may not [0.5]. (1.0)

REF: BSEP AP - Vol. 1

8.6 a. (1.0)

REF: BSEP OI-01

8.7 a. - 2 (0.5)
b. - 4 (0.5)
c. - 1 (0.5)
d. - 3 (0.5)

REF: BSEP - REP, GET

8.8 a. (1.0)

REF: 10 CFR 55.31(e)

8.9 d. (1.0)

REF: BSEP AI-58

8.10 c. (1.0)

REF: BSEP AI-58

8.11 c. (1.0)

REF: BSEP AI-58

8.12 - All control rods capable of insertion are fully inserted except for the analytically determined highest worth rod [.25] which is assumed to be fully withdrawn [.25] (0.5)

- Cold (68 degrees F) (0.5)

- Xenon-free (0.5)

REF: BSEP TS Definitions

8.13 Operational Condition 5 (Refueling) (0.5)

REF: BSEP TS Table 1.2

8.14 b. (1.0)

REF: BSEP TS Definitions

8.15 "T" = lowest value of the ratio of design TPF divided by the MTPF (obtained for any class of fuel in the core) [0.5]

"T" is applied only if less than or equal to 1.0 [0.5] (1.0)

REF: BSEP TS 3.2.2

8.16 a. (1.0)

REF: BSEP TSs 3.5.1, 3.5.2, 3.5.3.1

8.17 d. (1.0)

REF: BSEP TSs 3.0.3, 3.5.1, 3.5.3.2, B 3.0.3

8.18 a. (1.0)

REF: BSEP TSs 3.6.6.2, 3.6.6.3, 3.0.4

8.19 b. (1.0)

REF: BSEP TSs 3.5.4, 3.5.3.1, 3.0.1

8.20 a. (1.0)

REF: BSEP OI-02

8.21 b. (1.0)

REF: BSEP OI-05

8.22 Thermal expansion (of valve internals on heat up) may cause valve binding and/or damage. (1.0)

REF: BSEP OI-13

8.23 - No Pressure Boundary Leakage (.5)

- 5 gpm Unidentified leakage [.4] averaged over any 24 hour period [.1] (.5)

- 25 gpm total leakage [.4] averaged over any 24 hour period [.1] (.5)

- 2 gpm increase in Unidentified leakage [.4] within any 24 hour period [.1] (.5)

- Total leakage exceeded 25 gpm over 24 hours (25.50) (.5)

- Unidentified Leakage increased from 2.52 gpm at 00-04 to a 4.58 gpm and 5.2 gpm rate on shifts 16-20 and 20-24 respectively, thus exceeding 2 gpm increase limit. (.5)

REF: BSEP TS 3.4.3.2