U. S. NUCLEAR REGULATORY COMMISSION REGION I
OPERATOR LICENSING EXAMINATION REPORT

EXAMINATION REPORT NO. 85-24(OL) and 85-27 (OL)

FACILITY DOCKET NO. 50-272 and 50-311

FACILITY LICENSE NO. DPR-70 and 75

LICENSEE: Public Service Electric and Gas Co.
P. O. Box 236
Hancock's Bridge, New Jersey 08038

FACILITY: Salem 1 and 2

EXAMINATION DATES: November 19-26, 1985

CHIEF EXAMINER: Donald F. Johnson — 1/10/86
Donald F. Johnson, Lead Reactor Engineer (Examiner) — Date

REVIEWED BY: R M Keller — 1/10/86
Robert M. Keller, Chief Projects Section 1C — Date

APPROVED BY: Harry B. Kister — 1/14/86
Harry B. Kister, Chief, Projects Branc' No. 1 — Date

SUMMARY: Seven Reactor Operator (RO) and eight Senior Reactor Operator (SRO) candidates were examined during this period; four RO and six SRO candidates received their licenses. Three of the RO's failed the written exam, two of those three also failed the simulator and oral exams; one SRO failed the written exam and one SRO failed the simulator and oral exams.

REPORT DETAILS

TYPE OF EXAMS: Replacement

EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail
Written Exam	4/3	7/1
Oral Exam	5/2	7/1
Simulator Exam	5/2	7/1
Overall	4/3	6/2

1. CHIEF EXAMINER AT SITE: D. F. Johnson (NRC)

2. OTHER EXAMINERS: R. M. Keller (NRC)
   D. G. Ruscitto (NRC)
   N. F. Dudley (NRC)
   B. S. Norris (NRC)

1. Summary of generic deficiencies noted on simulator/oral exams:

   a. Candidates were unable to perform a simple heat balance on a heat exchanger, nor were they able to locate all of the necessary parameters when prompted.

   b. Candidates were not aware of the sources of radiation for the RCS both shutdown and at 100% power; they were not aware of the relative magnitude for radiation levels on spent fuel; they were not aware of the potential problem associated with the incore nuclear detectors.

   c. Several deficiencies relate to the Candidates use of the EOPs:

      (1.) When transitioning to a new procedure, the Desk Operator (RO) would repeat the same verification steps and wait for a response from the Board Operator (RO) rather than realize that it had just been verified and would not change (Example: "Verify reactor tripped")

      (2.) When transitioning to a new procedure, the Desk Operator did not always ensure that the Shift Supervisor (SRO) verified the kickout parameter.

      (3.) When obviously in the wrong procedure, the Shift Supervisors were slow to stop and rediagnose the situation.

   d. During the simulator examinations, when equipment was reported as not functioning properly, the Control Room personnel would simply accept the report and not question why the equipment was malfunctioning.

2. Summary of generic deficiencies noted from grading of written exams:

   a. SRO Exam

      - Question 5.05.a - Candidates did not know the reasons for the differences in requirements for performance of an Inverse Count Rate Ratio (ICRR) calculation for rod withdrawal and boron dilution.

      - Question 7.09 - AOP-ELEC procedures do not include all of the Technical Specification sections which are applicable to the failure of electrical buses. (Examples: AOP-ELEC-VIB-C, Appendix 1 and AOP-ELEC-125v-C, Appendix 1 are incomplete).

      - Question 8.07 - Candidates were unable to identify the proper type of tagout to be used for maintenance activities.

      - Question 8.11 - Candidates were unable to properly classify events when provided with the E-Plan Classification Guide.

b. RO Exam

- Question 1.08 - Candidates were unable to determine the system flow and operating pressure when running centrifugal pumps in parallel with a positive displacement pump.
- Question 1.13 - Candidates were unable to calculate the new steady state values for Tavg and S/G pressure after closure of one Main Steam Isolation Valve.
- Question 3.0.6.a - Candidates were unable to list the Control Room indications for the failure of the high voltage power supply to a power range detector.
- Question 4.03 - Candidates were unfamiliar with the length of time an On-the-Spot change is valid for; they were also unfamiliar with the approval requirements.

3. Personnel present at Exit Interview:

NRC Personnel

D. F. Johnson, Lead Reactor Engineer (Examiner)
B. S. Norris, Reactor Engineer (Examiner)
T. J. Kenny, Senior Resident Inspector

Facility Personnel

D. Hanson - Manager, Nuclear Training
A. Thompson - General Manager, Nuclear Services
R. Schaeffer - Assistant Manager, Operations Training
J. Gueller - Operations Manager, Salem
J. K. Lloyd - Principal Training Supervisor
P. J. Landers - Principal Training Supervisor
J. M. Zupko, Jr. - General Manager, Salem Operations

4. Summary of NRC Comments made at exit interview:

a. Preliminary results on oral/simulator exams:
One Senior Reactor Operator - marginal on oral
One Reactor Operator - marginal on oral and simulator
One Reactor Operator - failed oral and simulator

b. Salem has only two shutdown initial conditions (ICs) for the simulator, neither of which was functional for this exam. This must be corrected prior to the next scheduled examination.

c. Generic weaknesses (details are contained within the report)

(1) sources of radiation
(2) simple heat balance calculations
(3) slow to rediagnose when obviously following wrong emergency procedure.

5. Summary of facility comments and commitments made at exit interview:

a. Facility agreed to correct the problem of no shutdown ICs prior to the next scheduled requalification training (February, 1986)

6. Changes made to Written Exam during examination review:

a. SRO Exam

See Attachments 3 and 4

b. RO Exam (all changes to answer key)

1.01.b changed to:
"Core exit TCs - stable or decreasing
RCS hot leg temperature - stable or decreasing
S/G pressure - stable or decreasing
RCS cold leg temperature - saturation temperature for S/G pressure

1.12.d(b) Changed to "Feedwater heater outlet"

2.09.g Changed to "FC (MS 169 and 171)"

3.10.b Deleted "Emergency stop pushbutton"
Added "Backup differential"

Attachments:

1. Written Examination and Answer Key (RO)
2. Written Examination and Answer Key (SRO)
3. Facility Comments on SRO Written Exam
4. NRC Resolution of Facility Comments on SRO Exam

Attachment 1

MASTER

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: SALEM 1&2

REACTOR TYPE: PWR-WEC4

DATE ADMINISTERED: 85/11/19

EXAMINER: NORRIS, B. S.

APPLICANT: ______________________

INSTRUCTIONS TO APPLICANT:

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

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

FINAL GRADE ________________%

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

____
APPLICANT'S SIGNATURE

QUESTION 1.01 (2.50)

a. Describe how Thermal Driving Head (TDH) causes natural circulation. [0.50]

b. What indications are used to determine if natural circulation is occuring? [1.00]

c. While performing a natural circulation cooldown without RVLIS, pressurizer level rapidly increases. What is the MOST probable cause? [1.00]

QUESTION 1.02 (2.50)

The reactor is at 15% power, equilibrium Xenon and Samarium, boron concentration is 1200 ppm, BOL, Control Bank C rods are at 200 steps.

Using Figures 1.1 to 1.8:

What will be the final boron concentration if power is raised to 100%, ARO, steady state conditions? State which figure you use as applicable. State all assumptions and show all work.

QUESTION 1.03 (.75)

Moderator temperature coefficient becomes MORE negative from BOL to EOL PRIMARILY because of (choose ONE):

a. Decrease in number of thermal neutrons available for absorption in the moderator.
b. Increase in resonance escape probability per degree change in moderator temperature.
c. Decrease in thermal utilization factor per degree change in moderator temperature.
d. Increase in rod worth due to fuel burnout.

QUESTION 1.04 (.75)

Doppler coefficient becomes MORE negative from BOL to EOL because of: (choose ONE)

a. Increase in effective fuel temperature.
b. Production of Pu-240.
c. Clad creep and fuel pellet swell.
d. Overlapping of resonant peaks.

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.05 (.75)

Control rod worth is GREATEST at (choose ONE):
a. Low boron concentration.
b. High boron concentration.
c. Low moderator temperature.
d. High moderator temperature.

QUESTION 1.06 (1.00)

Differential boron worth becomes MORE negative over core life because there is _____________ (more/less) competition among boron atoms for thermal neutrons, therefore each boron atom is worth _____________ (more/less).

QUESTION 1.07 (1.00)

a. What is the MAJOR reason for ensuring sufficient Net Positive Suction Head (NPSH)?

b. Does the required NPSH increase, decrease, or remain the same if the speed of the pump increases?

QUESTION 1.08 (2.50)

Refer to Figure 1.9:

If all three pumps are running, what will be the total system flow (gpm) and operating pressure (psig)? Show your work.

QUESTION 1.09 (2.50)

a. List the production and removal reactions of Xenon and Samarium. [1.50]

b. Provide TWO reasons for Xenon contributing nore negative reactivity at full power than does Samarium. [1.00]

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.10 (2.00)

Figure 1.10 is a sketch of Xenon concentration vs. time. Using Figures 1.11 & 1.12 as reference, sketch the approximate power history that would be associated with this Xenon concentration plot. Use the same form provided. Assume that all power changes are made as STEP changes.

QUESTION 1.11 (3.25)

The reactor is critical at 2x10E-9 amps; 150 seconds later power is observed to be 7x10E-8 amps.

a. What is the Startup Rate (SUR)? [0.75]

b. Given a delayed neutron fraction of 6x10E-3 and an average delayed neutron life of 12.7 seconds, how much reactivity was added for the SUR in part a? [1.50]

c. What would be the power level if the SUR in part 'a' continued into the power range? Explain. A NUMERICAL ANSWER IS NOT REQUIRED. [1.00]

QUESTION 1.12 (3.00)

Consider each of the following sets of conditions separately and determine whether each is subcooled, saturated, or superheated and by how much:

a. 2235 psig & 610 F [0.50]

b. 1100 psia & 435 F [0.50]

c. 25.55 in Hg vac. & 128.7 F [0.50]

d. At what three locations in the plant would you expect to find the conditions described in parts a, b, & c above? [1.50]

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.13 (2.50)

The plant is at a steady state power level of 33% with the below initial conditions when one of the Steam Generator Main Steam Isolation Valves goes shut due to the Train B dump valve failing. Calculate the final steady state values for the listed parameters. Assume no operator action, no reactor trip, turbine controls in automatic, and rod control in manual. State all assumptions and show all work.

Initial conditions: Tavg = 555 F
Tstm = 538 F
Core Delta T = 22 F

a. Turbine power
b. Tavg in the affected loop
c. Steam Generator pressure in the affected loop
d. Tavg in the non-affected loops
e. Steam Generator pressure in the non-affected loops

(***** END OF CATEGORY 01 *****)

QUESTION 2.01 (3.00)

a. What are three reasons for having Rod Insertion Limits? 1.00 [~~0.75~~]

b. With respect to the Rod Insertion Limits, "...the steamline break accident imposes the highest shutdown margin requirement." Explain why this is a true statement. [1.00]

c. Considering each of the following sets of conditions separately, which will make the Steamline Break accident worse? 1.00 [~~0.75~~]

BOL or EOL
Reactor shutdown or at 100% power
Tavg at 350 F or at 547 F

QUESTION 2.02 (3.00)

a. When must the plant be shifted from the cold-leg recirculation mode to the hot-leg recirculation mode? [0.50]

b. Why must the plant shift to the hot-leg recirculation mode? [0.50]

c. Using Figure 2.1 and the provided hi-liter, show ALL flowpaths when in the hot-leg recirculation mode. [2.00]

QUESTION 2.03 (2.00)

Aproximately 4% of the reactor coolant flow bypasses the core and thus is not available for heat removal.

List the four flow paths which bypass the core.

QUESTION 2.04 (2.00)

a. What is POPS and what is its purpose? [1.00]

b. How is POPS activated? [0.50]

c. ~~d.~~ What is the setpoint for POPS? [0.50]

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.05 (1.75)

a. What limitations are placed on plant operations if one or more Steam Generator code safety valves is declared inoperable? WHY? [1.00]

b. How many code safeties can be declared inoperable and still be allowed power operation? [0.25]

c. Why are the code safeties considered '...the only reliable means of heat removal'? [0.50]

QUESTION 2.06 (1.50)

a. List the normal, alternate, and emergency sources of water available to the auxiliary feed system. [1.00]

b. What is the power supply for the motor driven auxiliary feedwater pumps? [0.50]

QUESTION 2.07 (4.50)

a. Complete the drawing of the containment spray system using Figure 2.2 Include all major components, valves (valve numbers not required), sources of water, and detectors. [3.5]

b. What would be the affect on the containment atmosphere if the Spray Addition Tank contents were NOT injected? [0.5]

c. What auto signals will initiate containment spray? Include setpoints and logic. [0.5]

(***** CATEGORY 02 CONTINUED ON NEXT PAGE *****)

QUESTION 2.08 (3.00)

Answer the following questions regarding the Steam Dump electro-pneumatic control system with either (A) always true, (B) sometime true and sometime false, or (C) always false. Assume no malfunctions exist unless specified. If answer (B) is chosen, briefly explain why.

a. The control air which acts on the steam dump valve diaphragm passes through the positioner and three solenoid valves. [1.00]

b. The logic signal which tells the steam dump valve when to open and how much to open passes through the I/P converter. [1.00]

c. When in the steam pressure mode of operation above the low-low Tave setpoint, a simultaneous failure of both turbine impulse pressure channels (low) will cause steam dump actuation. [1.00]

QUESTION 2.09 (1.75)

How do the control valves in the below systems fail on a loss of Control Air?
(i.e. fail open, fail close, or no affect) Consider each separately.

a. CVCS letdown flow
b. RCP seal flow
c. RHR
d. AFW alternate water sources
e. Pressurizer power operated relief valves
f. Reactor Coolant Drain Tank
g. Main Steam Isolation Valves

QUESTION 2.10 (2.50)

Unit 2 at Salem has three emergency diesel generators, each supplying its own 4160 volt vital bus. List 5 loads on Bus 2A and 5 loads on Bus 2C. (NOTE: two pumps for the same system are considered one load)

(***** END OF CATEGORY 02 *****)

QUESTION 3.01 (3.00)

List the interlocks and/or automatic control features, if any, associated with the below listed valves (refer to Figure 3.1):

a. 2CV277
b. 2CV4
c. 2CV7
d. 2CV21
e. 2CV18
f. 2CV35
g. 2SJ2
h. 2CV71

QUESTION 3.02 (2.50)

List the conditions that will automatically initiate the following ESF signals. Include setpoints and logic as apropriate.

a. Safety Injection ('S')
b. Phase B Containment Isolation ('B')

QUESTION 3.03 (3.00)

a. Explain how and why indicated pressurizer level will change due to a leaking reference leg on a pressurizer level transmitter. [1.00]

b. Explain how and why indicated pressurizer level will change due to a steam leak inside containment. [1.00]

c. The pressurizer master level control channel provides inputs to three CONTROL functions. List two of them. [1.00]

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.04 (1.00)

What is the significance of the following two alarms?

a. Rod Control Urgent Failure

b. Rod Control Non-Urgent Failure

QUESTION 3.05 (2.50)

a. Will the Overtemperature Delta T and the Overpower Delta T setpoints increase, decrease, or not change for each of the below conditions? Consider each separately. [1.50]

1. Tavg increases
2. Pressure increases
3. N41 upper detector fails high

b. Justify your answers for part a.2 above. [1.00]

QUESTION 3.06 (3.50)

The plant is operating at 100% load and you are the Board Operator:

a. What indications would YOU have if the high voltage power supply was lost to ONE of the power range detectors (both upper and lower)? [1.80]

b. What actions must be taken to continue power operation? [1.00]

c. With one channel out-of-service already, may surveillance be performed on a second channel while at power? Explain. [0.70]

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.07 (2.00)

For the below Process Radiation Monitoring Systems, list:
purpose
protection on alarm, if any
type of detector used (Geiger-Mueller, scintillation, ion chamber)

a. Steam Generator Blowdown Liquid Monitors

b. Letdown Line Monitors

QUESTION 3.08 (2.00)

The reactor is at 75% power, CBD is at 176 steps. For each of the below conditions, determine rod direction and speed.
Consider each case separately

a. Bank Selector Switch (BSS) in Auto, Tavg = 564 F, Tref = 565 F
b. BSS in Auto, Tavg = 564 F, Tref = 560 F
c. BSS in Manual, RAISE pushbutton depressed, PT-505 fails low
d. BSS in Manual, RAISE pushbutton depressed, PR-NI41 upper fails high

QUESTION 3.09 (3.00)

Salem's Tavg program is a compromise between two extremes; that is, a constant Tavg program and a constant steam pressure Tstm) program.

This compromise minimizes three disadvantages of the constant programs - what are these three disadvantages?
(NOTE: one of the disadvantages is due to the constant Tavg program, the other two are due to the constant steam pressure program)

QUESTION 3.10 (2.50)

a. What two automatic signals will start the diesel generators? [1.00]

b. What will trip the diesel generators when they are operating in the emergency mode? [1.50]

(***** END OF CATEGORY 03 *****)

QUESTION 4.01 (2.25)

List the nine major IMMEDIATE actions required by EOP-Trip-1 'Reactor Trip or Safety Injection'

QUESTION 4.02 (3.00)

In order to maintain the plant at 100% power, work must be performed inside the containment in a radiation field of 850 mrem/hr gamma, 30 mrad/hr thermal neutron, 20 mrad/hr fast neutron, and 200 mrad/hr beta. The maintenance man selected is 29 years old and has a life time exposure through last quarter of 53 REM on his NRC Form 4; additionally, he has accumulated 0.5 REM this quarter.

a. Assuming the man wears the appropriate protective clothing, how long may he work in this area without exceeding his 10CFR limits? Show all work. [2.00]

b. During a declared emergency, this individual volunteers to enter a high radiation area and perform work necessary to prevent further effluent release. In accordance with the Station Procedures, what is his maximum allowed whole body exposure? [0.50]

c. Whose authorization is needed in part b. [0.50]

QUESTION 4.03 (1.50)

Answer the following with respect to AP-3, 'Document Control Program':

a. How long is an On-The-Spot change valid for? [0.50]

b. Who must approve an On-The-Spot change prior to implementation? [0.50]

c. Who must approve the On-The-Spot change after implementation? When must this aprovel be accomplished by? [0.50]

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.04 (3.50)

a. List the bases for the below statement out of IOP-3, 'Hot Standby to Minimum Load'? [2.50]
'Within fifteen minutes prior to criticality, verify RCS Tavg to be greater than or equal to 541 F'

b. If, during power operation, Tavg drops below 541 F, what action(s) must be taken as per the Technical Specifications? [1.00]

QUESTION 4.05 (2.40)

Answer the following in accordance with IOP-2, 'Cold Shutdown to Hot Standby':

a. Below 250 F, at least one RHR pump or one RCP must be in operation. Why? (List 2 reasons) [1.20]

b. RHR must be isolated from the RCS before temperature reaches 350F or pressure reaches 375 psig. Why? (List a reason for each) [1.20]

QUESTION 4.06 (2.00)

As a certified reactor operator, assigned to a shift as an extra person, you are required to perform the second verification of a valve alignment on the Auxiliary Feedwater Pump #21.

a. May both the first and second verification be performed together? [0.50]

b. How is the position of a manual valve verified? [1.50]

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.07 (3.00)

In accordance with EI-4.10, 'Control Room Evacuation', part of the immediate actions is to station personnel at various locations within the plant to operate equipment or to monitor indications. For the following equipment, state WHERE personnel must go:

a. Pressurizer heater control
b. Auxiliary feedwater tank level indication
c. Charging pump start/stop control
d. Steam generator level indication
e. Main steam stop valve control

QUESTION 4.08 (2.40)

Indicate at what power level each of the below hapens:

a. Enter Mode 1 (power operation)
b. P-7 permissive
c. P-2 permissive
d. Intermediate range rod stop
e. Power range lo range hi level trip
f. P-8 permissive

QUESTION 4.09 (2.20)

a. What is the basis for the limit on Axial Flux Difference (AFD) per the Technical Specifications? [1.00]

b. Given the below information and assuming that it is 0800, 19 Nov 85, at what time (include date) will it be permissible to raise power greater than 50%? State all assumptions and show all work. [1.20]

Date	Time (leave band)	Time (return band)	Power
19 Nov 85	0716	0800	45%
19 Nov 85	0325	0359	65%
18 Nov 85	1802	1830	80%

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.10 (2.75)

Answer the following in accordance with OP-II-3.3.8 'Rapid Boration'

a. What are the preferred and alternate sources of water? [0.50]

b. Flow is to be greater than ________ gpm. [0.75]

c. If only one boric acid pump is available, how long will you have to rapid borate for the following: [1.50]
Show your work and state your assumptions
(1) two rods stuck fully out
(2) an unexpected cooldown of 50 F while shutdown

(***** END OF CATEGORY 04 *****)
(************* END OF EXAMINATION ***************)

Equations

$Q = M\Delta h$

$$\rho = \frac{1}{\tau K_{eff}} + \frac{\beta}{1 + \lambda\tau}$$

$\dot{Q} = \dot{m}c_p\Delta T$

$$SUR = \frac{26\ \rho}{\ell^* + (\beta-\rho)\ \tau}$$

$Q = UA\Delta t$

$C_1(1-K_1) = C_2\ (1-K_2)$

$h_L = K\dot{V}^2$

$P = Po\ 10^{sur\ (t)}$

$\pi = 3.14$

$P = Po\ e^{t/\tau}$

$e = 2.72$

$SUR = \frac{26.06}{\tau}$

$CR = \frac{S}{1-K_{eff}}$

$\tau = \frac{\ell^*}{\rho} + \frac{\beta-\rho}{\rho\bar{\lambda}}$

$\tau = \frac{\rho}{\rho-\beta}$

$\frac{1}{M} = \frac{CR_1}{CR_2}$

$\chi = \frac{h-h_f}{h_{fg}}$

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

$\ell^* = 10^{-5}$ seconds

$\chi = \frac{s-s_f}{s_{fg}}$

Conversions

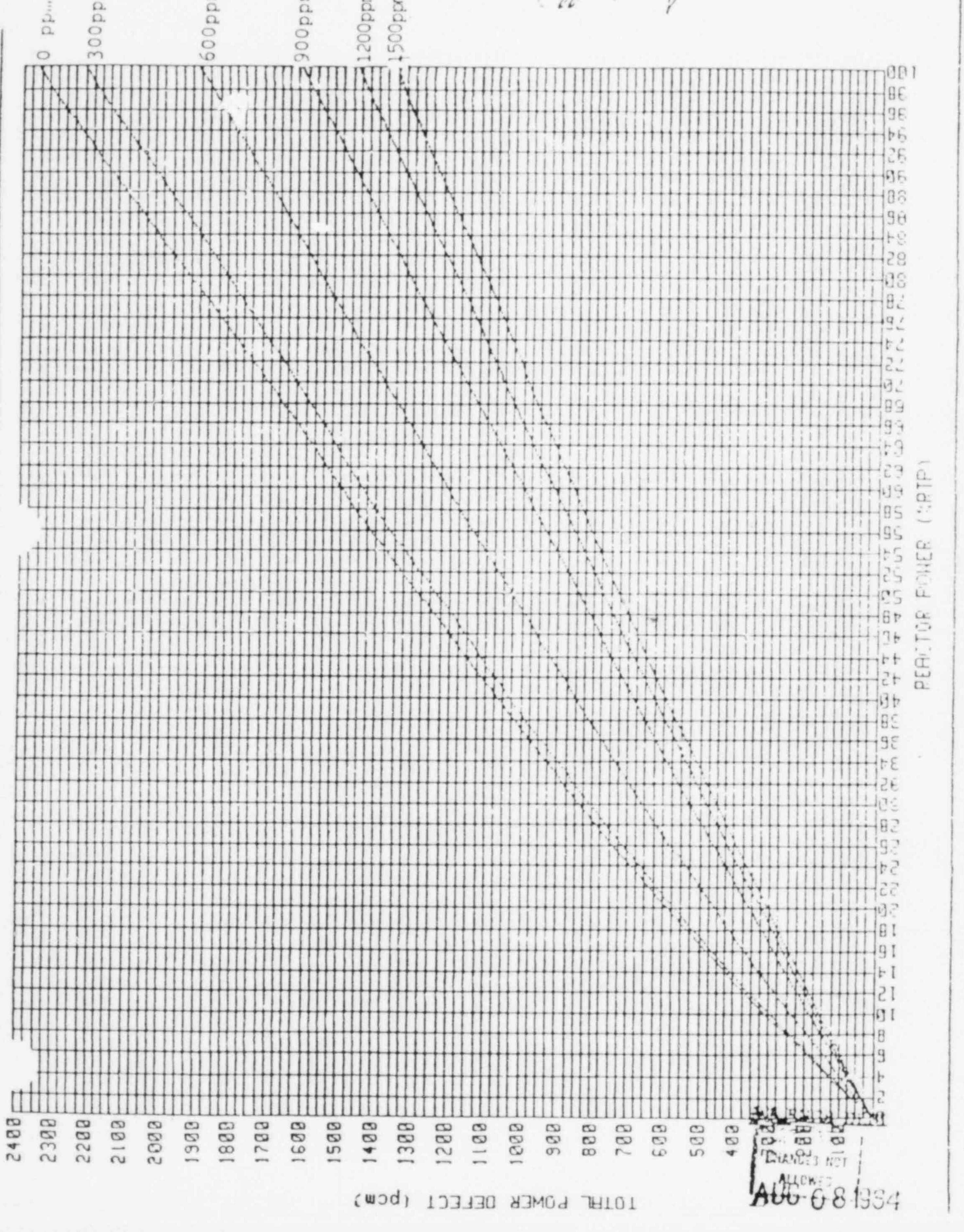
1 curie = 3.7×10^{10} dps	1 kg = 2.2 lbs
1 gal = 3.78 liters	1 gm/cm^3 = 62.4 lbs/ft^3
1 in = 2.54 cm	1 ft^3 = 7.48 gal
1 yr = 2.15×10^7 sec	1 gal = 8.3453 lbm
1 MW = 3.41×10^6 BTU/HR	1 in Hg = ~~2.037 psia~~ = 0.491 psia

FIGURE 1.1

SALEM 1 CYCLE 6

POWER DEFECT Vs. REACTOR POWER

PREPARED BY *James L. Jones, Jr.* TECH ENG *Jeffrey G. Jackson* DATE 8/8/84

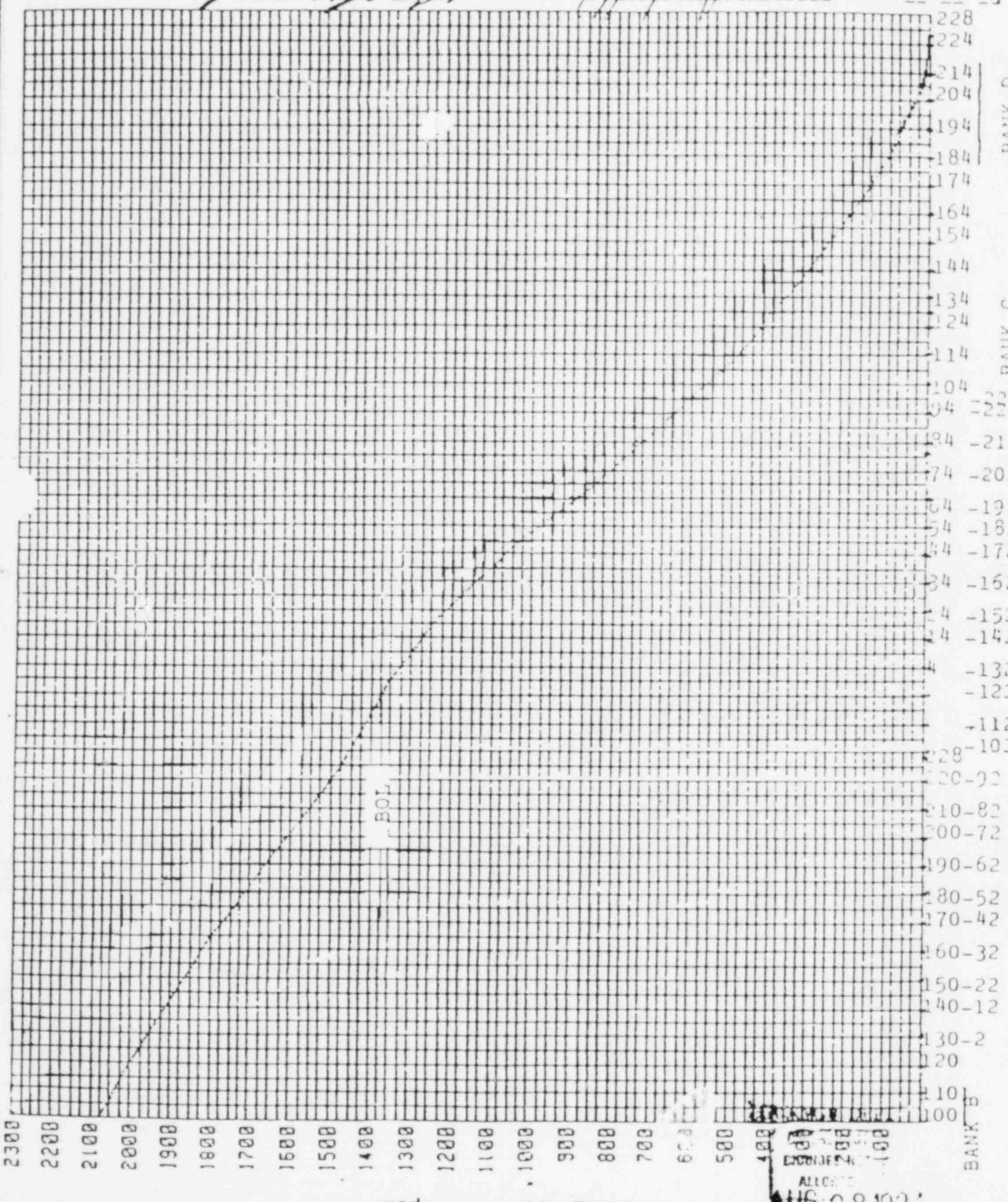


SALEM 1 CYCLE 6

FIGURE 1.2

INTEGRAL ROD WORTH vs. POSITION IN OVERLAP

PREPARED BY [signature] TECH ENG [signature] DATE 8/8/84

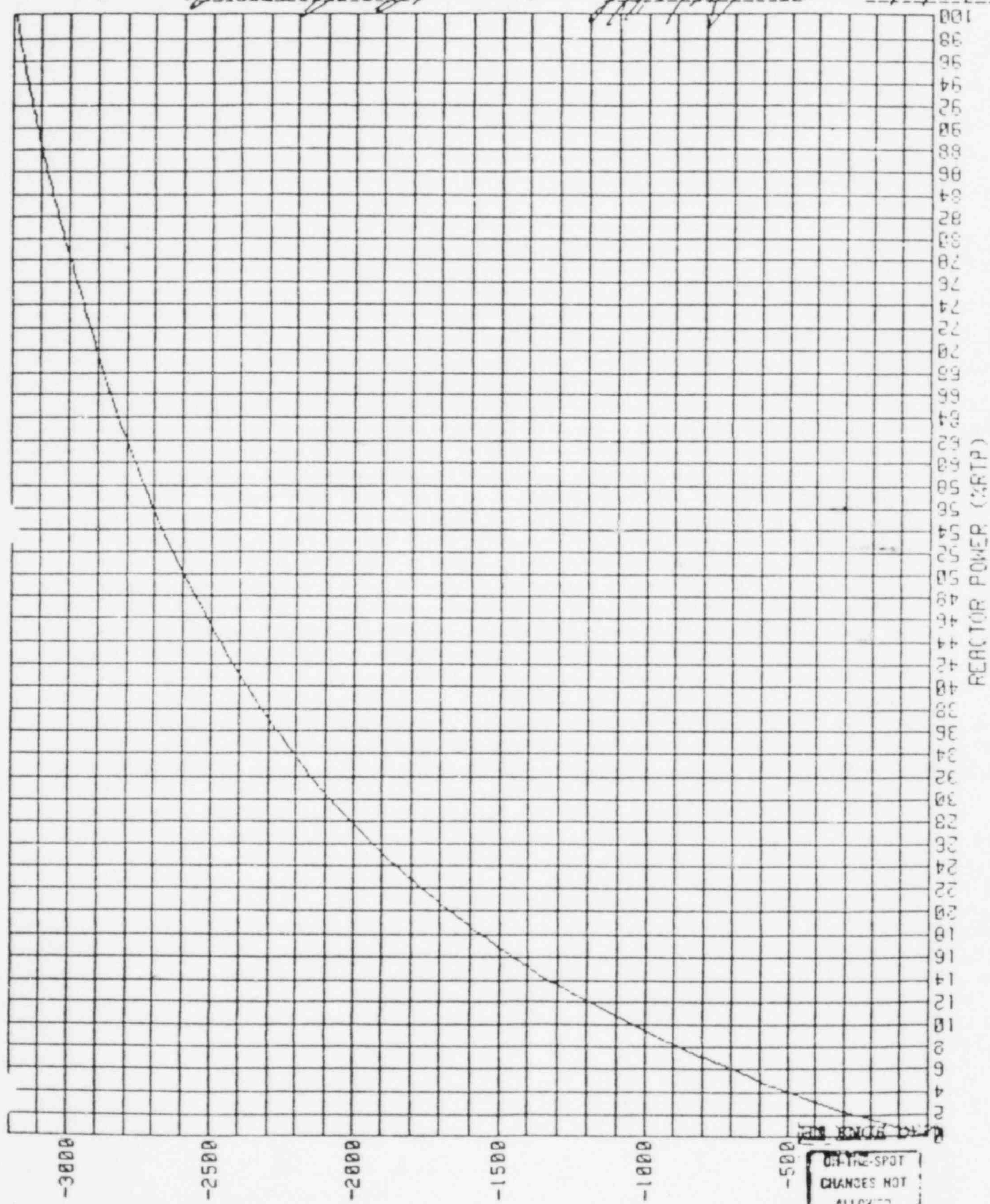


SALEM 1 CYCLE 6

FIGURE 1.3

EQUILIBRIUM XENON REACTIVITY vs REACTOR POWER

Prepared by [signature] TECH Engineer [signature] Date 8/8/84



ON-THE-SPOT CHANGES NOT ALLOWED

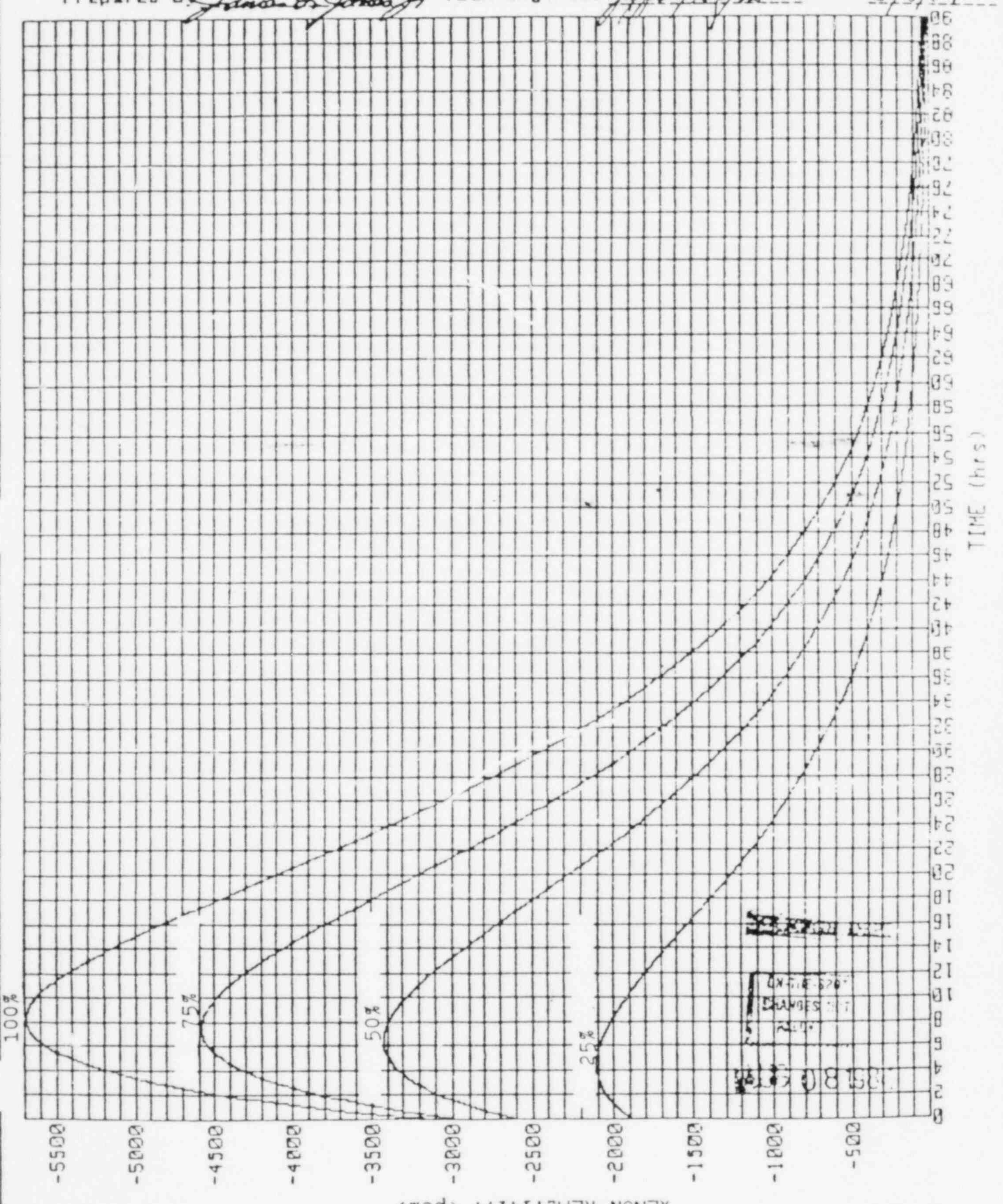
AUG 30 1984

FIGURE 1.4

SALEM 1 CYCLE 6

XENON WORTH VS TIME AFTER TRIP FROM
EQUILIBRIUM CONDITIONS FOR
25%, 50%, 75%, 100% powers

Prepared by [illegible] TECH Engineer [illegible] Date 8/8/84

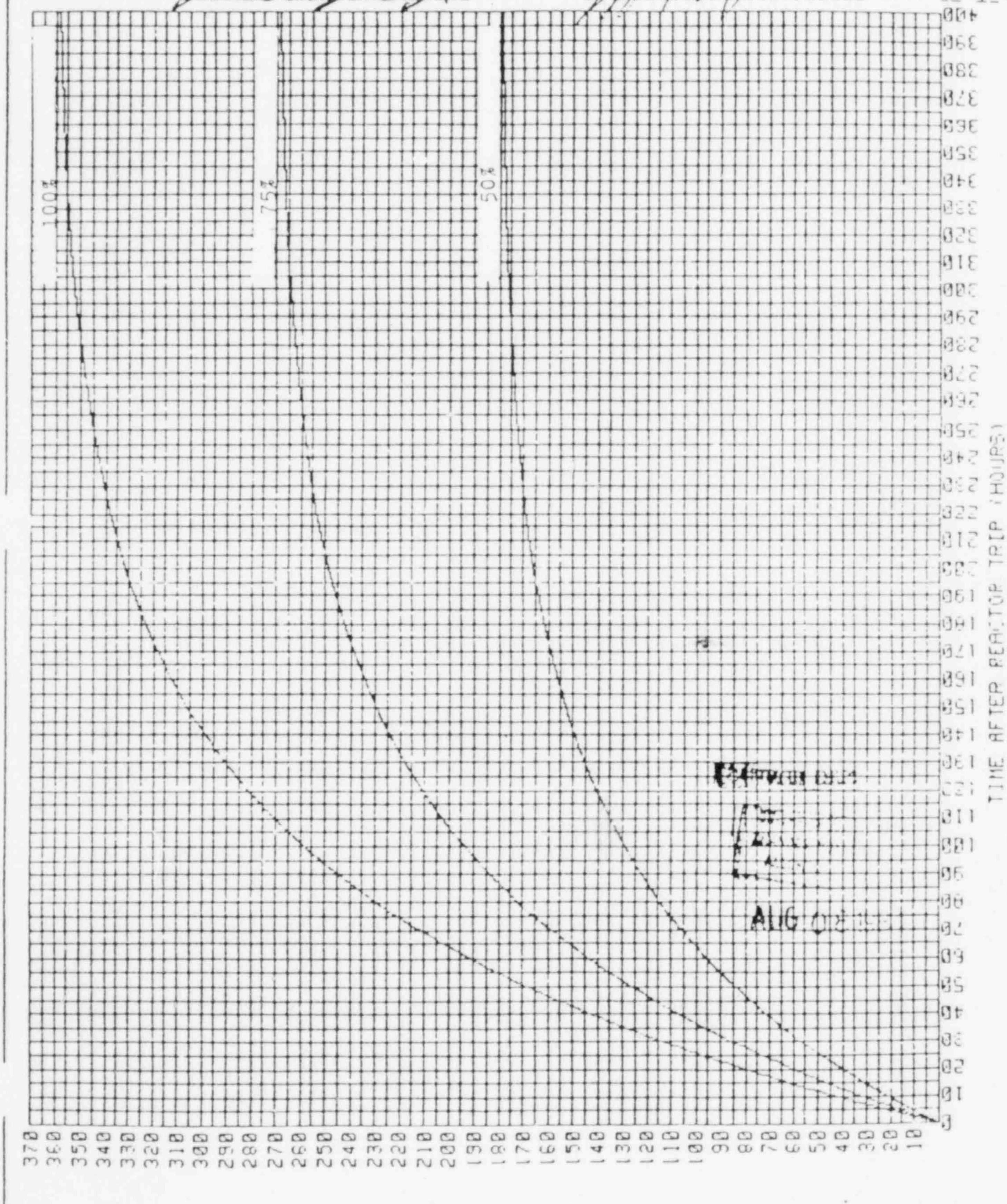


SALEM 1 CYCLE 6 FIGURE 1.5

SAMARIUM REACTIVITY vs. TIME AFTER SHUTDOWN

BUILD UP FROM EQUILIBRIUM CONDITIONS (Eq Sm=588 pcm)

PREPARED BY James L. ... Jr. TECH ENG Jeffrey G. Jackson DATE 8/8/84



SALEM 1 CYCLE 6 FIGURE 1.7

DIFFERENTIAL BORON WORTH vs. BORON CONCENTRATION

REACTOR ENGINEERING MANUAL

PREPARED BY [signature] TECH ENG [signature] DATE 8-8-84

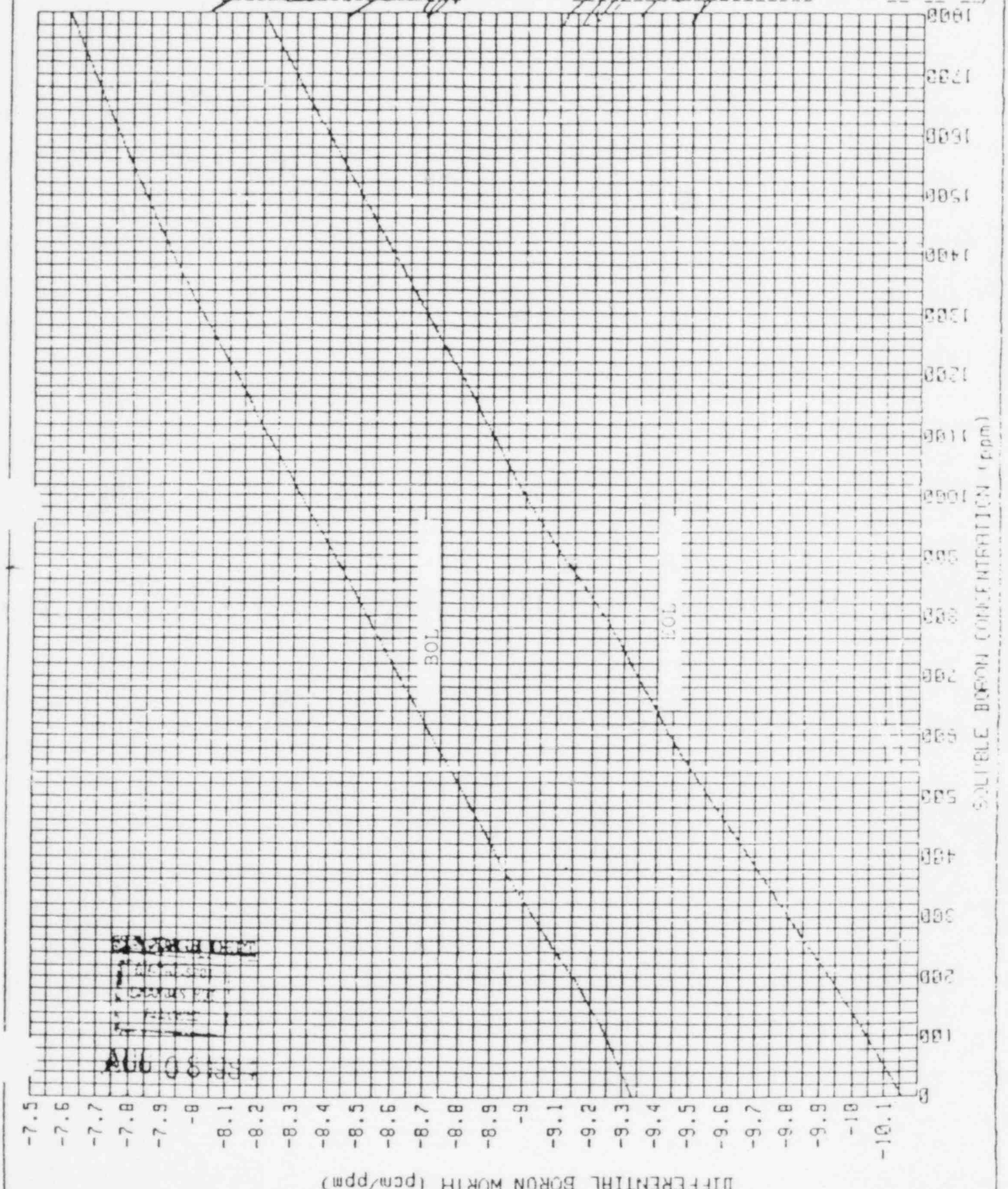
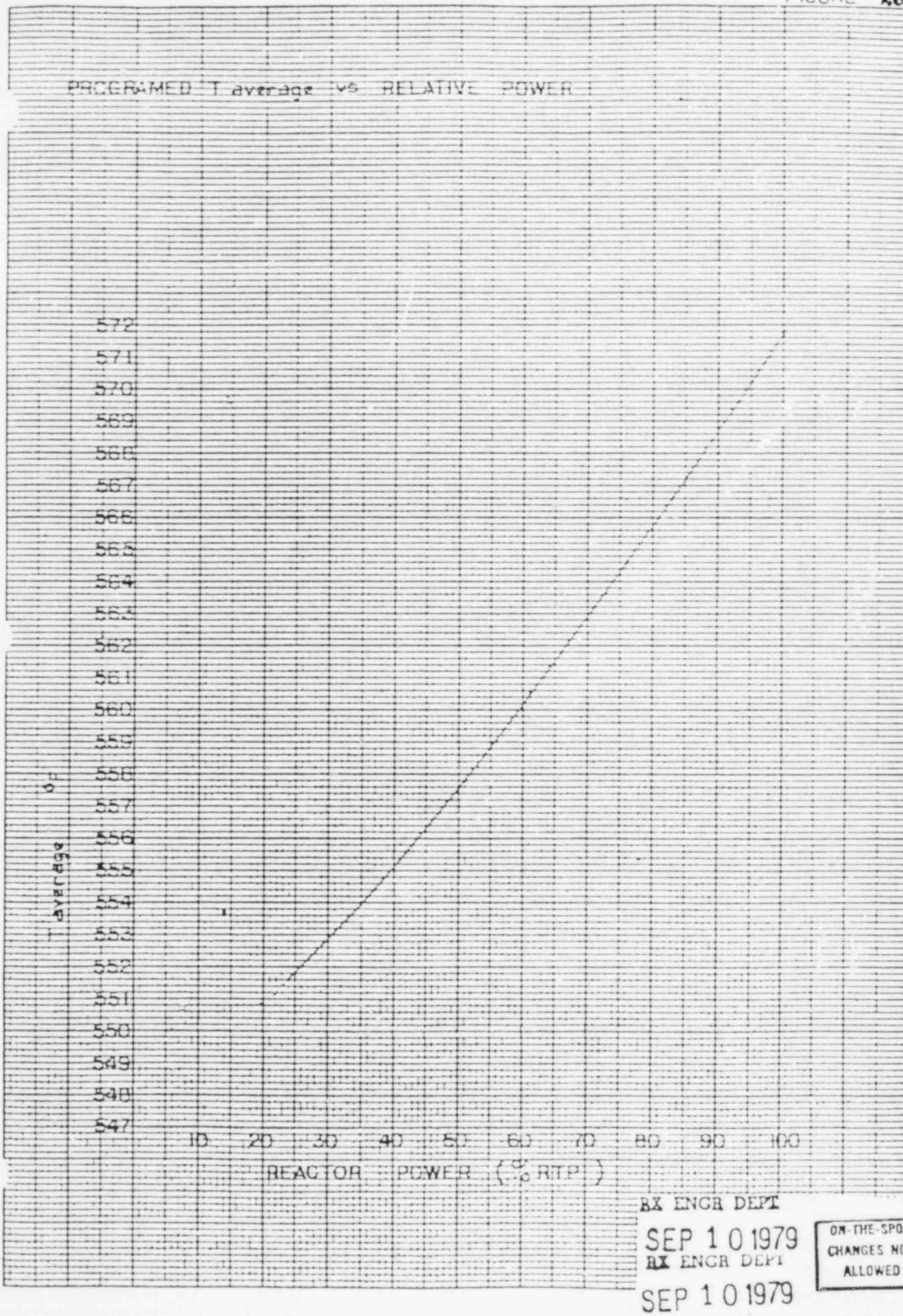


FIGURE 16



ON-THE-SPOT CHANGES NOT ALLOWED

FIGURE 1.8

SALEM 1 CYCLE 6

CONTROL ROD INSERTION LIMITS vs. REACTOR POWER FOR BANK 'B', BANK 'C', & BANK 'D'

PREPARED BY James L. Jones TECH ENG Jeffrey W. Jackson DATE 8/8/84

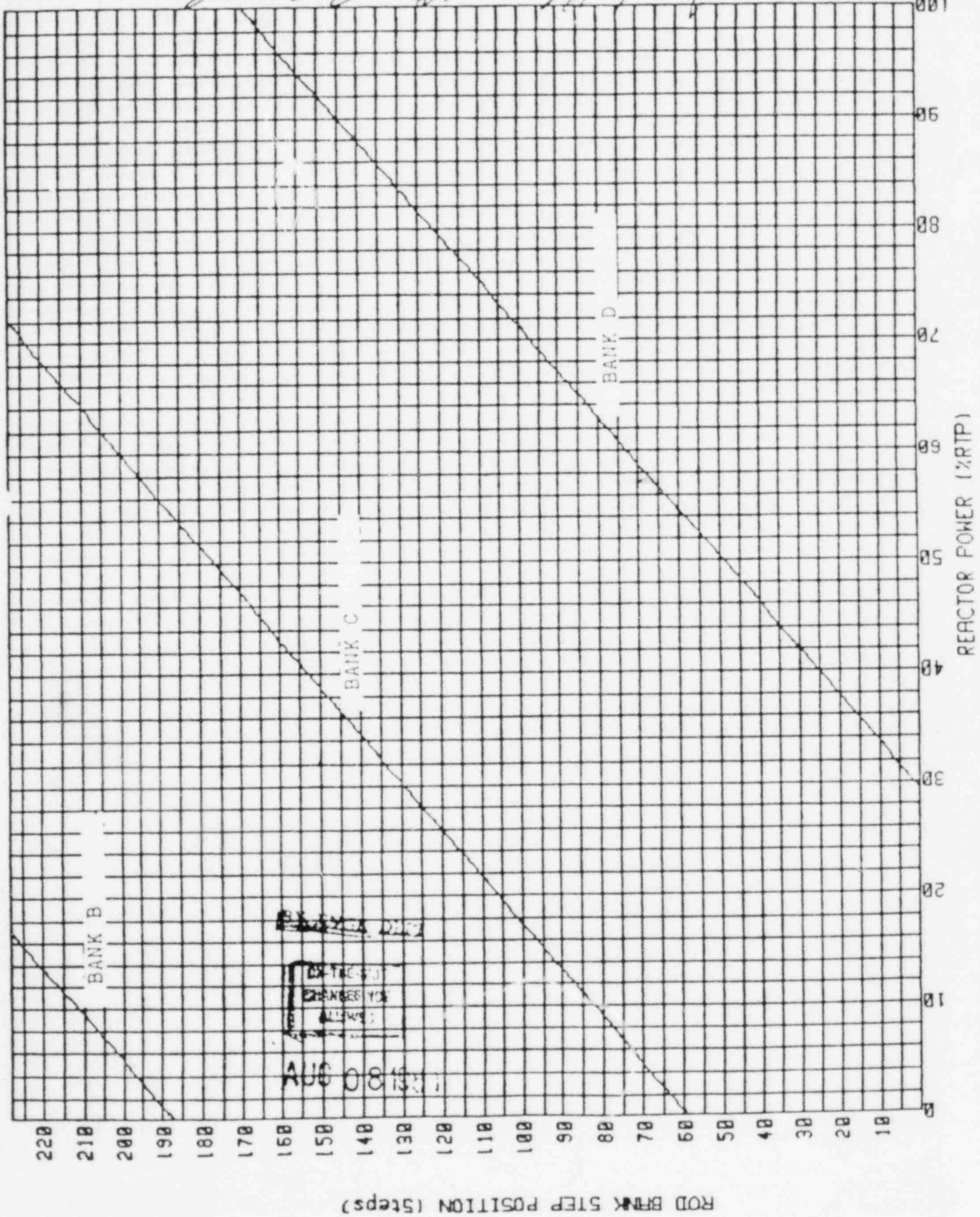
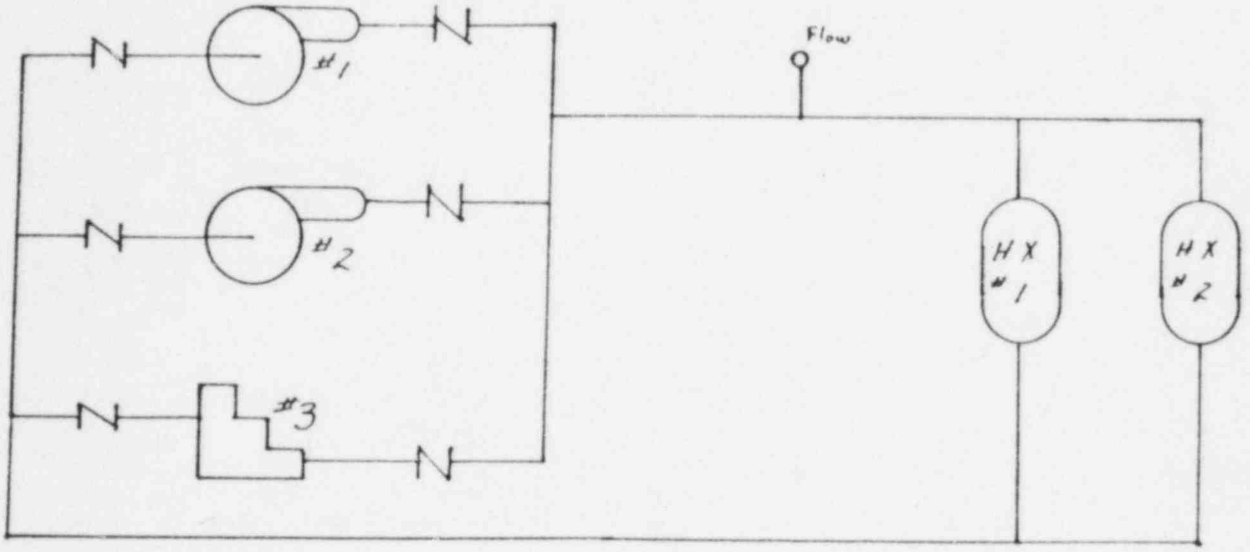
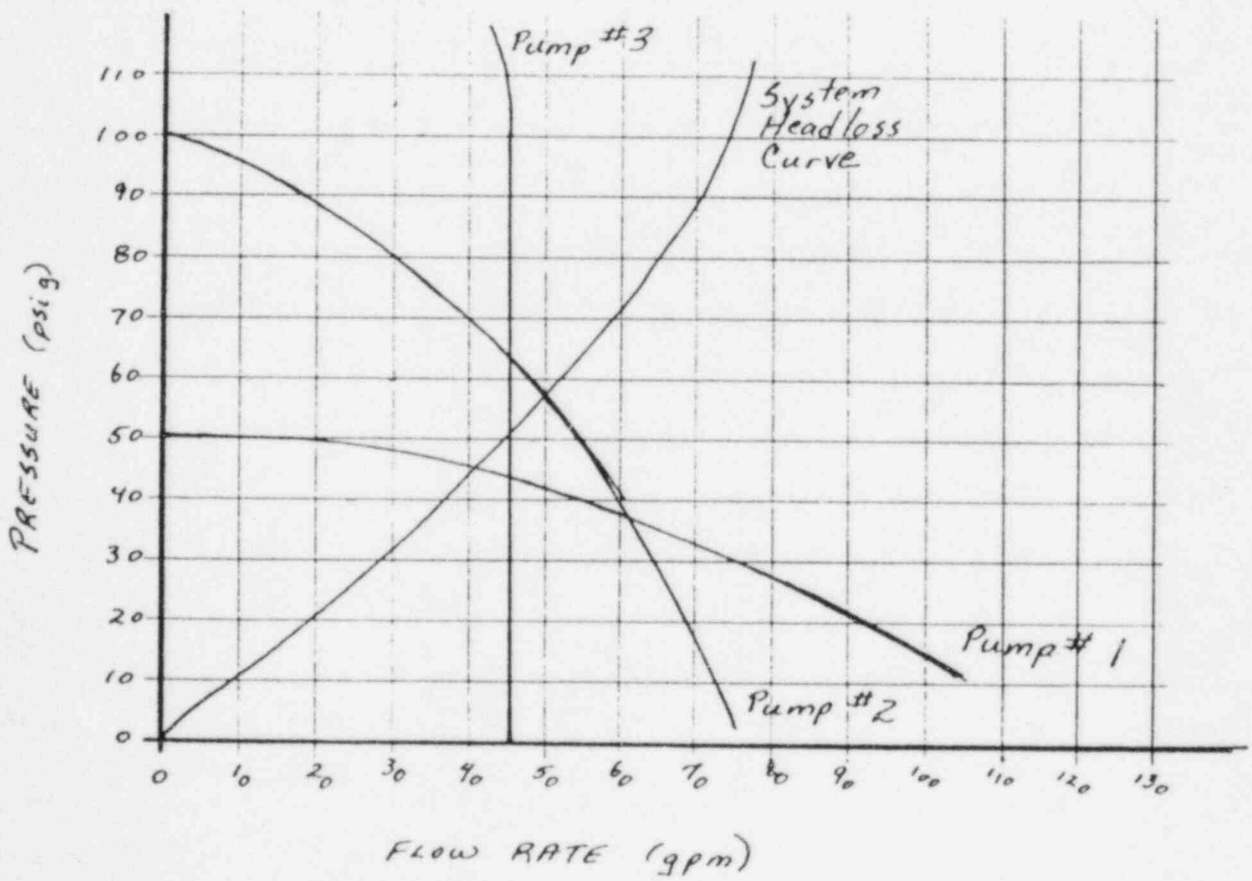


FIGURE 1.9


Flow
#1
#2
#3
HX #1
HX #2


Pump #3
System Headloss Curve
Pump #1
Pump #2
PRESSURE (psig)
FLOW RATE (gpm)
0
10
20
30
40
50
60
70
80
90
100
110
120
130

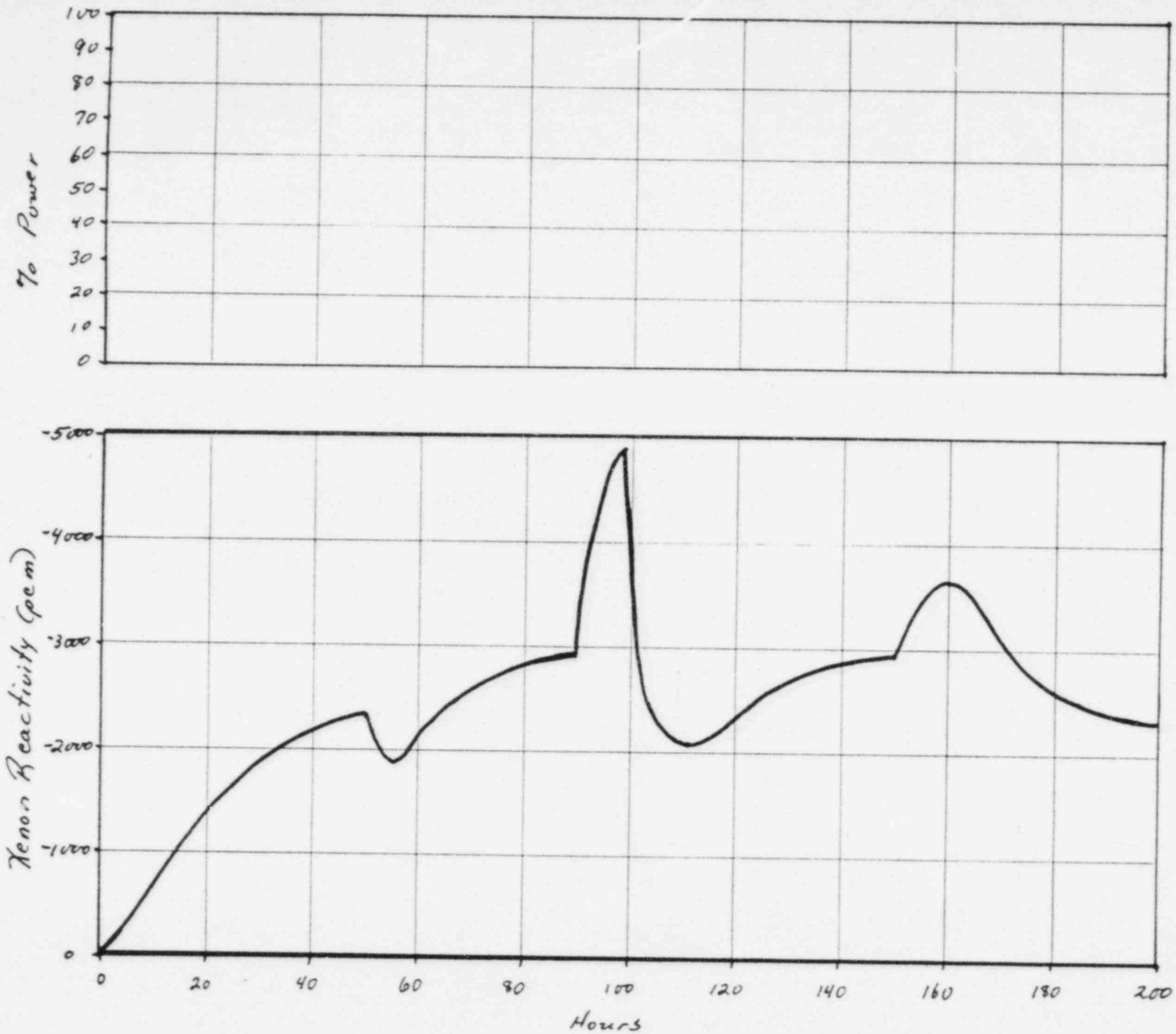
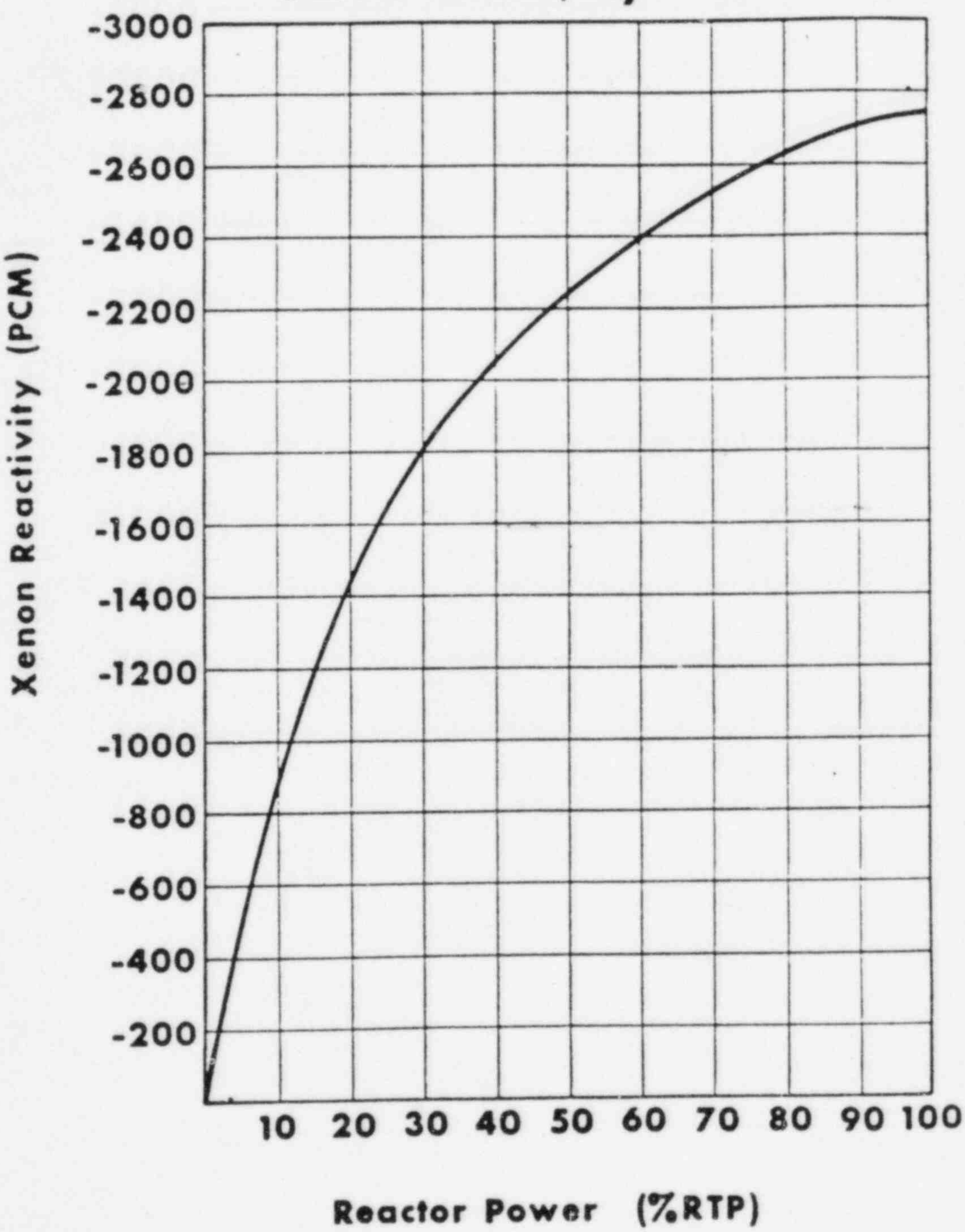

% Power
100
90
80
70
60
50
40
30
20
10
0
Xenon Reactivity (pcm)
-5000
-4000
-3000
-2000
-1000
0
0
20
40
60
80
100
120
140
160
180
200
Hours

Figure 1.10

Figure 1.11

RxTh-TP-38.5

EQUILIBRIUM XENON REACTIVITY
vs
REACTOR POWER
Salem Unit 1, Cycle 3



G83-0182

RxTh-TP-39.2

XENON WORTH vs TIME AFTER TRIP

SALEM UNIT 1, CYCLE 3

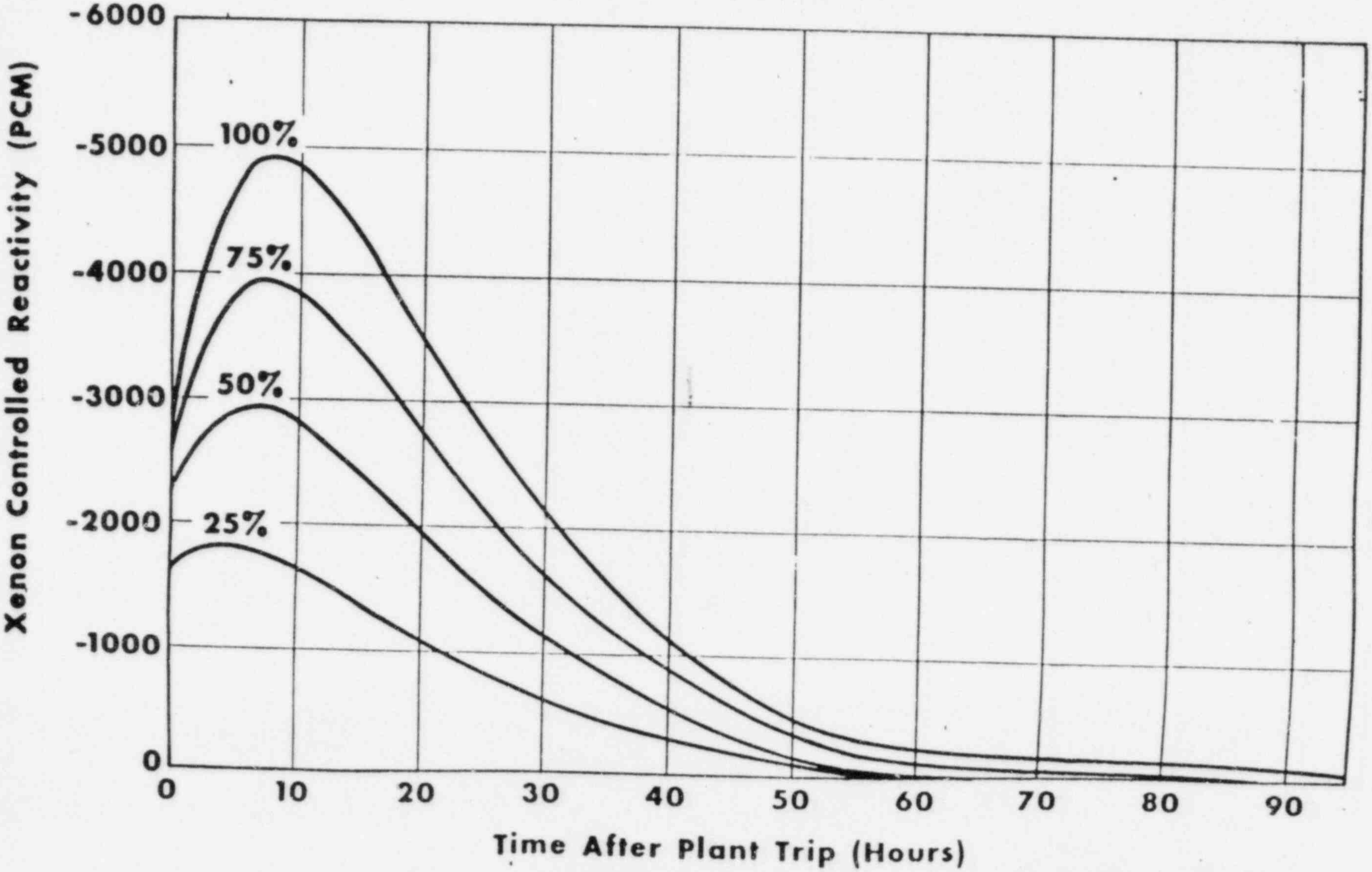


Figure 1.12

EMERGENCY CORE COOLING SYSTEM ONE-LINE DIAGRAM

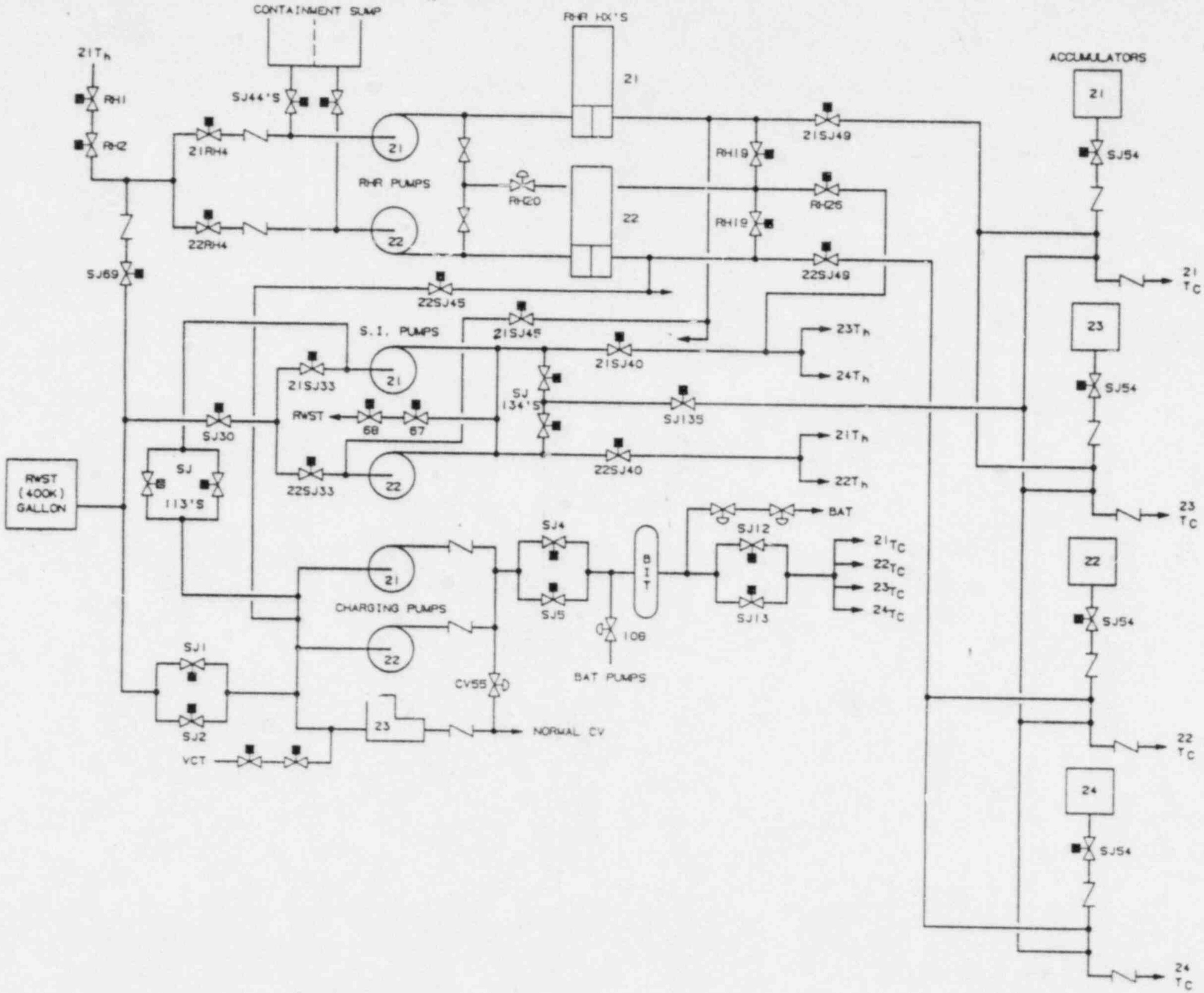


Figure 2.1

CONTAINMENT SPRAY

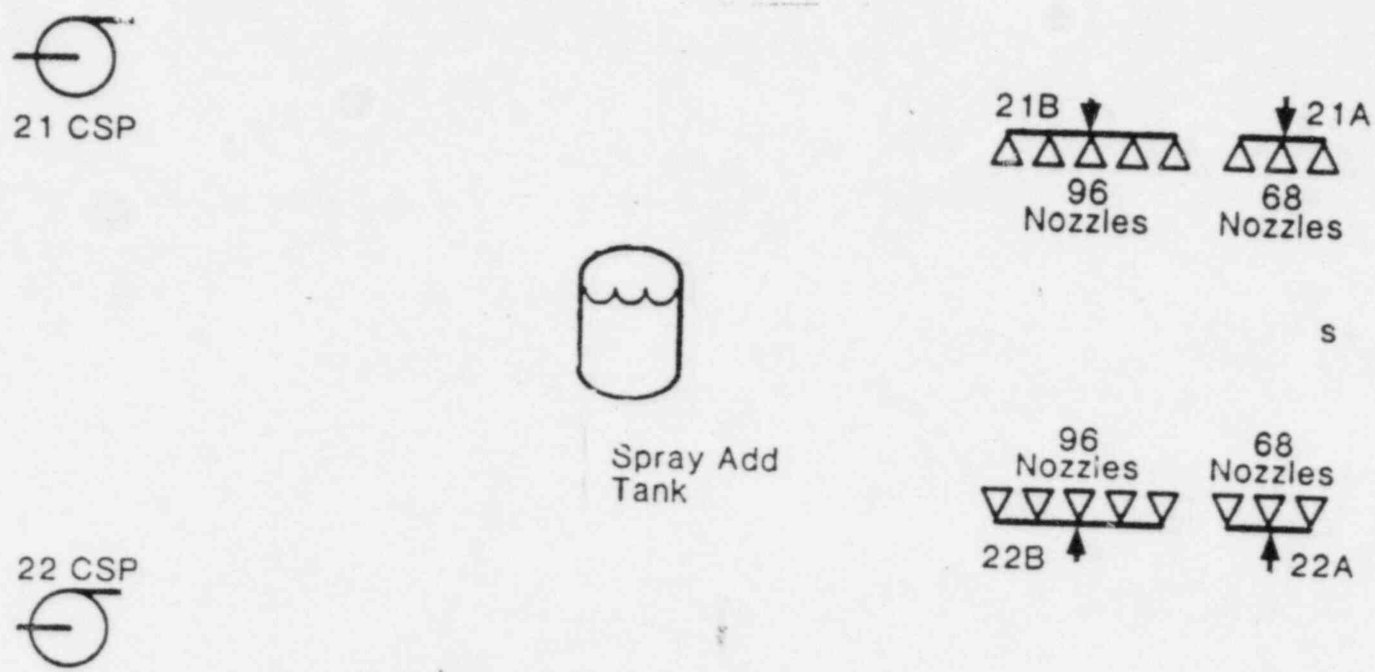
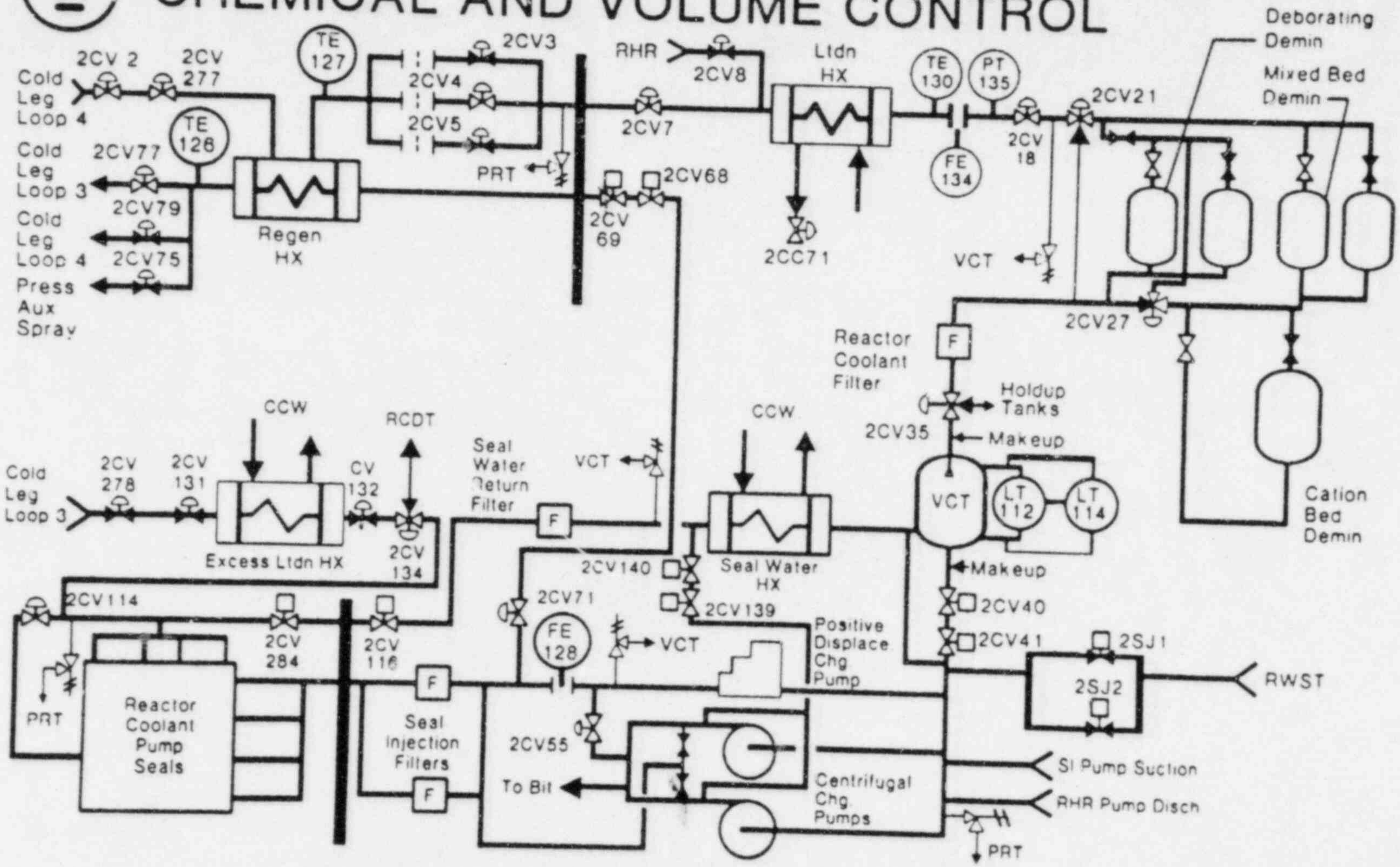


Figure 2.2

Figure 3.1

CHEMICAL AND VOLUME CONTROL



MASTER

ANSWER 1.01 (2.50)

a. Flow is caused by a pressure difference [0.25] due to fluids at different elevations [0.25]. [0.50]

b. ~~RCS Delta T < or = Full Load Delta T~~ ~~[0.33 each]~~
~~Core exit thermocouple (or WR temp.) constant or decreasing~~
~~Steam Generator pressure constant or decreasing consistent with RCS temp~~

c. Voiding in the RCS (during depressurization) [1.00]

REFERENCE
Student Notebook, Chapter 1, Reactor Coolant System, pgs 17-20
EOP-Trip-5, para 3.11 - Caution

K&A 000017EK1.01/IF 4.4

ANSWER 1.02 (2.50)

Reactivity due to power increase (Fig 1.1)
250 - 1450 = -1200 pcm [0.50]

Reactivity due to rod withdrawal (Fig 1.2)
825 - 0 = +825 pcm [0.50]

Reactivity due to Equilibrium Xenon Change (Fig 1.3)
1400 - 3180 = -1780 pcm [0.50]

Boron worth (Fig 1.7)
=~ -8.2 pcm/ppm [0.50]

Total: -1200pcm + 825pcm -1780pcm = -8.2 pcm/ppm(1200ppm - BC)
-2155 pcm = -8.2 pcm/ppm(1200ppm - BC)
BC = (9840 pcm - 2155 pcm)/(8.2 pcm/ppm)
BC = 937.19 ppm [0.50]

REFERENCE
Reactor Theory, pg 152
Reactor Engineering Manual, Part 1 (Figures)

K&A 001010K5.21/IF 3.4

1.01.b Core exit TC's - stable or decreasing [0.25 each]
RCS hot leg - stable or decreasing
S/G Pres - stable or decreasing
RCS cold leg temp at saturation temp for S/G pres
Ref: EOP-LOCA-1, step 3.26

ANSWERS -- SALEM 1&2 -85/11/19-NORRIS, B. S.

ANSWER 1.03 (.75)

C [0.75]

REFERENCE
Student Notebook, Chapter 46, Transient Analysis, pgs 6-9
Reactor Theory, pgs 150-152

K&A 001000K5.26/IF 3.3

ANSWER 1.04 (.75)

B [0.75]

REFERENCE
Student Notebook, Chapter 46, Transient Analysis, pgs 10-11
Reactor Theory, pgs 169-170

K&A 001000K5.48/IF 3.3

ANSWER 1.05 (.75)

D [0.75]

REFERENCE
Reactor Theory, pg 203

K&A 001000K5.02/IF 2.9

ANSWER 1.06 (1.00)

Less [0.50]
More [0.50]

REFERENCE
Reactor Theory, pg 191

K&A 001000K5.30/IF 2.9

ANSWER 1.07 (1.00)

a. To prevent cavitation of the pump. [0.50]

b. Increase [0.50]

REFERENCE
General Physics HT/T & FF Fundamentals, pgs 319-320

K&A Pumps (pg A-9)/IF 3.4

ANSWER 1.08 (2.50)

Refer to Figure 1.9:

Pump #1 will have zero flow since the system is operating at a pressure greater than its shutoff head; total flow is found by adding pumps #2 & #3. [1.00]

Flow rate = 68 (+or- 3) gpm [0.75]
Operating pressure = 86 (+or- 3) psig [0.75]

REFERENCE
General Physics HT/T & FF, pgs 324 - 332

K&A Pumps (pg A-9)/IF 2.4

ANSWER 1.09 (2.50)

a. Production: I-135 -> Xe-135 [0.25 each]
Xe directly from fission
Pm-149 -> Sm-149
Removal: Xe-135 + n -> Xe-136
Xe-135 -> Cs-135
Sm-149 + n -> Sm-~~146~~150

b. 1. Higher fission yield for Xenon and its precursors than for Samarium's precursors. [0.50]

2. The microscopic cross-section of absorption is greater for Xenon. [0.50]
(Xe 2.7x10E6 barns; Sm 5600 barns)

REFERENCE
Reactor Theory, pgs 219-233
Nuclear Reactor Engineering, Glasstone & Sesonske, Tables A.2 & A.3

K&A 001000K5.33/IF 3.2
001000K5.38/IF 3.5

ANSWER 1.10 (2.00)

See attached Figure 1.10

REFERENCE
Student Notebook, Chapter 46, Transient Analysis, pgs TA-12 & TA-19
Reactor Theory, pgs Rx-Th-TP-38.5, & 39.2-39.5

K&A 001000K5.13/IF 3.7

ANSWER 1.11 (3.25)

a. $P = Po10^{SUR(t)}$ [0.25]

$7\times10E-8 = (2\times10E-9)10^{SUR(2.5)}$ [0.25]

$35 = 10^{SUR(2.5)}$

1.544 = 2.5(SUR)

SUR = 0.618 dpm [0.25]

b. Beta-eff = 0.006 [0.25]
Lambda-eff = 1/12.7 = 0.08 [0.25]
T = 26.06/SUR [0.25]
= 26.06/0.618
= 42.17 sec [0.25]
T = (Beta - rho)/rho(lambda) [0.25]
42.17 = (0.006 - rho)/rho(0.08)
rho = 0.006/4.37 = 0.00137 = 137 pcm [0.25]

"Power Defect" acceptable

c. At power levels above POAH [0.25], Doppler[0.25] and MTC[0.25] will turn power such that the power will increase to a level where the reactivity added by the SUR is equal to the negative reactivity added due to Doppler and MTC[0.25].

REFERENCE
Student Notebook, Chapter 46, Transient Analysis, pgs 42-44
Reactor Theory, pgs 134-135

K&A 001000A1.06/IF 4.1

ANSWERS -- SALEM 1&2 -85/11/19-NORRIS, B. S.

ANSWER 1.12 (3.00)

a. Tsat at 2250 psia = 652.7 F
652.7 - 610 = 42.7 F [0.25]
subcooled [0.25]

b. Tsat at 1100 psia = 556.28 F
556.28 - 435 = 121.28 F [0.25]
subcooled [0.25]

c. 25.55 in Hg (0.491) = 12.55 psia
14.7 - 12.55 = 2.15 psia [0.25]
*if use Table 1: Tsat at 2.15 psia = 128.7 F
saturated [0.25]
OR*if use Table 2: Tsat at 2.15 psia = 119.1 F
128.7 - 119.1 = 9.6 F [0.10]
superheated [0.15]

d. (a) RCS hot leg or core exit [0.50]
(b) ~~High pressure condenser heater outlet~~ Feedwater heater outlet (S/G inlet) [0.50]
(c) Condenser (shell side) [0.50]

REFERENCE
Student Notebook, Chapter 1, Reactor Coolant System, pg 35
Student Notebook, Chapter 34, Condensate & Feed System, pg 55
General Physics HT/T & FF, pgs 83-88

K&A 001000K5.56/IF 4.2

c) 25.55 in Hg (0.491) = 52.05 psia
Tsat @ 52 psia = 283.36
283.36° - 128.7° = 154.66°
subcooled

ANSWER 1.13 (2.50)

a. Turbine power
~~slightly reduced due to the reduced Pstm~~ remain constant since governor valves will open as Pstm↓ [0.50]

b. Tavg (affected)
final: goes to Thot of 566 F [0.50]

c. S/G pressure (affected)
initial: Psat at 538 F = 947 psia
final: Psat at 566 F = 1189 psia [0.50]

d. Tavg (non-affected)
Delta T must increase by 1/3 => final Delta T = 29.3F
final: Tavg = Th - (Delta T)/2 = 566 - 29.3/2 = 551.33 F [0.50]

e. S/G pressure (non-affected)
(Tavg - Tstm) must increase by 1/3 => final Tavg - Tstm = 22.67 F
therefore, Tstm = 551.33 - 22.67 = 528.66 F
final: Pstm at 528.66 F = 901.3 psia [0.50]

REFERENCE
Student Notebook, Chapter 1, Reactor Coolant System, pg 35
Student Notebook, Chapter 5, Steam Generator, pg 33
Student Notebook, Chapter 22, Rod Control, pg RS-3

K&A 035010K1.09/IF 3.8

ANSWER 2.01 (3.00)

a. To ensure adequate shutdown margin [~~0.25~~ 0.33 each]
To minimize the reactivity effect of a rod ejection accident
To minimize radial flux peaking factors (maintain pwr. dist limits)

b. Due to the large value of positive reactivity inserted by MTC during the resulting uncontrolled RCS cooldown. [1.00]

c. EOL [~~0.25~~ 0.33 each]
Shutdown
547 F

REFERENCE
Student Notebook, Chapter 23, Rod Position Indication, pgs 20-21

K&A 001000k5.08/IF 3.9

ANSWER 2.02 (3.00)

a. 1. When cold-leg recirc has been in operation for 22.5 hours [0.25]
2. When boron concentration decreases [0.25]

b. 1. To quench the steam bubble in the vessel head [0.25]
2. To flush the boron off the fuel rods and back into solution [0.25]

c. See Figure 2.1 [2.00]

REFERENCE
Student Notebook, Chapter 10, Emergency Core Cooling System, pgs 53-64

K&A 006030K4.03/IF 3.4
006030A1.03/IF 3.6

ANSWER 2.03 (2.00)

1. Nozzel bypass flow [0.50 each]
2. Control rod & thimble bypass flows
3. Baffle wall bypass flow
4. Head cooling bypass flow

REFERENCE
Student Notebook, Chapter 3, Reactor Vessel & Internals, pg 19

ANSWERS -- SALEM 1&2 -85/11/19-NORRIS, B. S.

K&A 002000K6.13/IF 2.3

ANSWER 2.04 (2.00)

a. Pressurizer Overpressure Protection System [0.50]
provided to prevent brittle fracture of the RCS at low pressure [0.50]

b. The system must be manually activated by the operator [0.50]

c. 375 psig (+/- 10 psig) [0.50]

REFERENCE
Student Notebook, Chapter 25, Pressurizer Pressure & Level Control, pgs 14-16

K&A 002000K4.10/IF 4.2
010000K4.03/IF 3.8

ANSWER 2.05 (1.75)

a. Plant power is restricted [0.50]
due to the reduced heat removal capability of the secondary [0.50]

b. 3 per generator [0.25]

c. the code safeties required no outside motive force for actuation [0.50]

REFERENCE
Student Notebook, Chapter 5, Steam Generator System, pg 16
Student Notebook, Chapter 33, Main Steam, pgs 11-12

K&A 035010SG#5/IF 3.1

ANSWER 2.06 (1.50)

a. normal - auxiliary feed water storage tank [0.25 each]
   alternate - demineralized water storage tank
   emergency - fresh water/fire protection storage tanks
   - service water

b. MDAFWP #21 - 4160 vital bus 2A [0.25 each]
   MDAFWP #22 - 4160 vital bus 2B

REFERENCE
Student Notebook, Chapter 11, Auxiliary Feed System, pgs 15, 51, AF-5
Student Notebook, Chapter 40, Electrical Distribution, pg 70

K&A 061000K1.07/IF 3.6
061000K2.02 & 2.03/IF 3.7
061000K4.05/IF 4.5

ANSWER 2.07 (4.50)

a. See attached Figure 2.2 [3.50]

b. The iodine removal process would not be as effective due to the lack of the sodium-hydroxide (NaOH raises pH from 9.5 to 11.0) [0.50]

c. High-High containment pressure [0.30]
   23.5 psig [0.10]
   2/4 sensors [0.10]

REFERENCE
Student Notebook, Chapter 12, Containment & Containment Spray, pgs 1(#9), 13, 20-21, & CS-7

K&A 026020A1.01/IF 3.1
026020SG#9/IF 3.6

ANSWER 2.08 (3.00)

a. B (when energized, the positioner is bypassed and supply air pops-open the trip-open valve) [1.00]

b. B trip open logic bypasses the I/P converter [1.00]

c. C (turbine impulse pressure is not an input in the Pstm mode) [1.00]

REFERENCE
Student Notebook, Chapter 26, Steam Dump System, pgs 11 & SD-2

K&A 041020K4.17/IF 3.7
041020K6.03/IF 2.7

ANSWER 2.09 (1.75)

a. FC (CV2 & CV277) [0.25 each]
b. FO (CV104)
c. FO (RH18)
d. FC (AF52)
e. FC (PR1 & 2)
f. FC (WL12)
g. ~~No affect (electro-hydraulic)~~
FC (MS 169 & 171)

REFERENCE
Student Notebook, Chapter 6, CVCS, pg 8
4, RCP, pg 22
8, RHR, pg 16
11, AFW, pg 18
25, Pzr Pres & Lvl Cntrl, pg 12
29, Rad Liq Waste, pg WL-2
33, Mn Stm, pg 16

K&A 078000K3.02/IF 3.4

ANSWER 2.10 (2.50)

(five required from each Bus) [0.25 each]

Bus 2A	Bus 2C
Auxiliary Feedwater Pump (#21)	Containment Spray Pump (#22)
Containment Spray Pump (#21)	Service Water Pumps (#25 & 26)
Service Water Pumps (#21 & 22)	Safety Injection Pump (#22)
Residual Heat Removal Pump (#21)	Component Cooling Pump (#23)
Safety Injection Pump (#21)	Centrifugal Charging Pump (#22)
Component Cooling Pump (#21)	460 volt Vital Bus 2C
460 volt Vital Bus 2A	230 volt Vital Bus 2C
230 volt Vital Bus 2A	

REFERENCE
Student Notebook, Chapter 42, Electrical Distribution, pg 70

K&A 064000K3.03/IF 3.6

ANSWER 3.01 (3.00)

a. -not able to open unless Pzr level > 17% [0.20 each]
   -closes if Pzr level < 17%

b. -not able to open unless at least one charging pump is running
   -not able to open unless Pzr level > 17%
   -not able to open unless 2CV2 AND 2CV277 open
   -closes if Pzr level < 17%
   -closes if Phase A isolation
   -closes if 2CV2 OR 2CV277 closes

c. -closes on Phase A isolation

d. -diverts letdown flow around the demineralizers on high temp (120 F) out of the letdown heat exchanger

e. -throttles to maintain pressure (350 psig) in the letdown heat exchanger

f. -modulates letdown flow to maintain level in the VCT

g. -opens on low level in the VCT
   -opens on an 'S' signal

h. -none

REFERENCE
Student Notebook, Chapter 6, Chemical & Volume Control System, pgs 8-19

K&A 0040x0K4.xx/IF 3.0

ANSWERS -- SALEM 1&2   -85/11/19-NORRIS, B. S.

ANSWER 3.02 (2.50)

a. 1. Low Pressurizer Pressure [0.20]
< 1765 psig [0.20]
2/3 sensors [0.10]

2. High Containment Pressure [0.20]
> 4.0 psig [0.20]
2/3 sensors [0.10]

3. High Steamline Differential Pressure [0.20]
> 100 psi ~~above~~ below remaining S/Gs [0.20]
2/3 sensors in one S/G [0.10]

4. High Steamline Flow (coincident with) [0.10]
LowLow Tavg OR [0.05]
Low Steamline Pressure [0.05]
1/2 ~~2/4~~ Steamflow; 2/4 Tavg; 2/~~3~~ 4 Steam pres. sensors [0.10]
on 2/4 Stmlines
Steamflow > 40% when at 0 - 20% load
40 - 110% when at 20 - 100% load [0.10]
Tavg < 543 F [0.05]
Steam Pressure < 500 psig [0.05]

b. High High Containment Pressure [0.20]
23.5 psig [0.20]
2/4 sensors [0.10]

REFERENCE
Student Notebook, Chapter 10, Emergency Core Cooling System, pgs 8-9

K&A 013000K1.01/IF 4.2

ANSWER 3.03 (3.00)

a. Pzr level will indicate higher than actual [0.50]
because the delta P across the bellows will be decreasing [0.50]

b. Pzr level will indicate higher than actual [0.50]
due to the steam heating the reference leg which causes the bellows to open which appears as a loss of fluid [0.50]

c. CVCS charging flow [any two .t 0.50 each]
Back/up heaters
Level alarms

REFERENCE
Student Notebook, Chapter 25, Pressurizer Pressure & Level Control, pgs 25-26

K&A 011000K1.xx/IF 3.7
000028EA2.12/IF 3.1

ANSWER 3.04 (1.00)

a. operational problems exist in the logic and/or power cabinets [0.30]
it affects the system ability to hold or move control rods [0.20]

b. problems within the auxiliary power supply to the logic and/or power cabinets [0.30]
no immediate effect [0.20]

REFERENCE
Student Notebook, Chapter 22, Rod Control System, pg 47

K&A 001000K6.04/IF 2.4

ANSWER 3.05 (2.50)

a. [0.25 each]

OTDT	OPDT
Decrease	Decrease
Increase	No change
Decrease	No change

b. OTDT - as pressure increases, you are further away from DNB [0.50]
OPDT - pressure has no effect on power

REFERENCE
Student Notebook, Chapter 24, RCS Temperature Indication System, pgs 9-13

K&A 012000K4.02/IF 3.9

ANSWER 3.06 (3.50)

a. 'PR Loss of Detector Voltage' alarm (window D-7) [0.30 each]
'Upper (& Lower) Section Deviation above 50%' alarm (windows D-37/45)
'Power Range Channel Deviaiton' bistable (window D-39)
'Power Range High Neutron Flux Rate (Neg)' bistable (wnidow D-38)
Indication to zero (meter & recording pen)
Power-on lamp out on the Excore NI Cabinet

b. Bypass trip the failed channels for control [0.~~50~~ .25] and protection [0.~~50~~ .25]
within one hour [0.50]

c. Yes [0.20]
T.S. allows one additional channel to be bypassed for up to two hours for surveillance testing [0.50]

REFERENCE
Student Notebook, Chapter 20, Excore Nuclear Instrumentation, pgs 17-24
Emergency Instruction I-4.19, pgs 6-7
Technical Specification 3/4.3.1, pg 3/4 3-5

K&A 015000K3.01/IF 3.9
015000A2.01/IF 3.5

ANSWER 3.07 (2.00)

a. purpose - primary to secondary leak detection [0.40]
protection - B/D isolation valve will colse on the alarming S/G [0.40]
detector - scintillation [0.20]

b. purpose - detection of failed fuel [0.40]
protection - none [0.40]
detector - UNIT 1 - scintillation [0.10]
UNIT 2 - ion chamber [0.10]

REFERENCE
Student Notebook, Chapter 17, Radiation Monitoring System, pgs 23-24

K&A 068000A4.04/IF 3.8
073000K4.01/IF 4.0
073000K4.02/IF 3.3

ANSWER 3.08 (2.00)

a. No motion (in the deadband) [0.50 each]

b. IN at 40 spm (acceptable for speed if answer midule of ramp)

c. OUT at 48 spm (blocked in auto only)

d. No motion (overpower rod stop)

REFERENCE
Student Notebook, Chapter 22, Rod Control, pgs 25-29, 69-71, RS-10 & RS-11

K&A 001000K4.02/IF 3.8

ANSWER 3.09 (3.00)

1. Constant Tavg program causes poor turbine performance [1.00]
(due to the low secondary steam pressure at high power levels)

2. Constant steam pressure program requires a very large pressurizer [1.00]
(due to the wide range that Tavg must cover)

3. Constant steam pressure program also causes Tavg to approach [1.00]
limits (that could cause bulk boiling or excessive fuel
centerline temperatures)

REFERENCE
Student Notebook, Chapter 22, Rod Control, pgs 6-7

K&A 002000K5.11/IF 4.0

ANSWER 3.10 (2.50)

a. safety injection signal [0.50]
loss of all offsite power to the 4160 volt vital buses (BLACKOUT) [0.50]

b. generator differential [0.40]
engine overspeed [0.40]
engine low low oil pressure [0.40]
~~emergency stop pushbutton (local)~~ Backup differential [0.30]

REFERENCE
Student Notebook, Chapter 41, Diesel Generator, pgs 36 & 50

K&A 064000K4.02/IF 3.9
064000A3.01/IF 4.1

ANSWERS -- SALEM 1&2 -85/11/19-NORRIS, B. S.

ANSWER 4.01 (2.25)

1. Manually trip the reactor [0.25 each, all 9 required]
2. Check all turbine stop valves closed
3. Check total AFW flow > 440,000 lbm/hr
4. Confirm reactor trip
5. Check at least two 4Kv vital buses energized
6. Check if SI required
7. Check for SEC loading for energized vital buses
8. Check safeguards valves positioned properly
9. Check containment pressure < 23.5 psig

REFERENCE
EOP-Trip-1, Reactor Trip or Safety Injection, pgs 2-5

K&A 000007EA2.02/IF 4.3

ANSWER 4.02 (3.00)

a. 5(N-18) = 55 REM [0.25]
Total lifetime to date = 53 + 0.5 = 53.5 rem
Total lifetime available = 55 - 53.5 = 1.5 rem [0.25]
Total this quarter available = 3 - 0.5 = 2.5 rem [0.25]
Lifetime is more restrictive than quarterly limit

gamma 850 mrem/hr = 850 mrem/hr [0.25]
th neut 30 mrad/hr x 3QF = 90 mrem/hr [0.25]
f neut 20 mrad/hr x 10QF = 200 mrem/hr [0.25]
* beta negligible due to protective clothing [0.25]

total =1140 mrem/hr total dose rate

1.5 rem/(1.14 rem/hr) = 1.32 hr = 79 min [0.25]

b. 25 REM whole body one time exposure [0.50]

c. General Manager, Salem Operations [0.50]

REFERENCE
10 CFR 20
AP No. 24, Radiological Protection Program, pgs 5 & 14

K&A Plant Wide Generic #15/IF 3.4

if make assumption of "high-energy β", calculation will change as such:
β 200 mrad/hr x 1QF = 200 mrem/hr => 1340 mrem/hr total

$\Rightarrow \frac{1.5 \text{ Rem}}{1.34 \text{ Rem/hr}} = 1.1 \text{ hr} = 66 \text{ min}$

ANSWER 4.03 (1.50)

a. Permanent changes - valid until incorporated into next review [0.25]
Temporary changes - valid up to the effective date on the change notice [0.25]

b. Supervisor in charge on the work [0.25]
Senior Shift Supervisor or Shift Supervisor on duty [0.25]

c. Same level as original procedure [0.25]
within 14 days [0.25]

REFERENCE
AP No. 3, Document Control Program, pgs 3-4

K&A Plant Wide Generic # 21/IF 3.8

ANSWER 4.04 (3.50)

a.1. MTC is within the analyzed temperature range [0.50 each]
2. Protective instrumentation is within normal operating range
3. P-12 (Tavg > 543 F) above its setpoint
4. Pressurizer is operable
5. Reactor vessel is above minimum temperature (RT-NDT)

b.1. Restore Tavg > 541 within 15 minutes, or [0.50]
2. Be in hot standby witnin next 15 minutes [0.50]

REFERENCE
IOP-3, Hot Standby to Minimum Load, pg 6
TS, pgs 3/4 1-6 & B3/4 1-2

K&A 001000K5.15/IF 3.4
001000K5.16/IF 3.4

ANSWERS -- SALEM 1&2 -85/11/19-NORRIS, B. S.

ANSWER 4.05 (2.40)

a. cooling [any 2 of 3] [0.60]
reliable temperature indication [0.60]
boron stratification
b. temperature - heat removal capability of RHR HX [0.60]
pressure - design pressure of RHR system (when combined with discharge pressure of RHR pump) [0.60]

REFERENCE
IOP-2, Cold Shutdown to Hot Standby, pg 2
Student Notebook, Chapter ~~8,~~ Residual Heat Removal, pgs 5-6
T.S. B 3/4 9-2

K&A 000002EK3.02/IF 3.03

ANSWER 4.06 (2.00)

a. No [0.50]

b. verified closed - attempt to close [0.50 each]
verified open - move in closed direction, then reopen
verified throttled - fully close, then open required number of turns

REFERENCE
OD-7, Valve Operations & Systems Alignment, pg 2

K&A Plant Wide Generic/IF 3.7

ANSWER 4.07 (3.00)

a. Panels 2GP & 2EP - Pressurizer Heaters [0.60 each]
(electrical penetration - elev 78')
b. Panel 379 - Auxiliary Feedwater Storage Tank Panel
(outside auxiliary building - elev 100')
c. Panel 213 - Hot Shutdown Panel
(auxiliary building - elev 84')
d. Panel 213 - Hot Shutdown Panel
(auxiliary building - elev 84')
e. Panel 687 - MSSV Local Control Station
(~~north & south~~ outer & inner penetration areas - elev 100')

[Panel Number not required]

ANSWERS -- SALEM 1&2 -85/11/19-NORRIS, B. S.

REFERENCE
EI-4.10, Control Room Evacuation, pgs TBL 1-1 to TBL 1-3

K&A 000068EA1.12/IF 4.4

ANSWER 4.08 (2.40)

a. 5% [0.40 each]
b. 10%
c. 15%
d. 20%
e. 25%
f. 36%

~~k. 50% (+/- 2%)~~
~~l. prior to 50% (+ 0%/- 2%)~~
~~m. > 85% (+/- 2%)~~
~~n. >90%~~

REFERENCE
IOP-4, Power Operation, pgs 3 & TBL 1-1 to 1-2

K&A System Wide Generic #12/IF 3.5

ANSWER 4.09 (2.20)

a. Assures that FQ(Z) [0.50] (upper bound envelope of 2.32 times the normalized axial peaking factor) is not exceeded during either normal [0.25] or in the event of xenon [0.25] redistribution following power changes.

b. 19 Nov (0800 - 0716) x (0.5) = 22 penalty minutes [0.25]
19 Nov (0359 - 0325) x (1.0) = 34 penalty minutes [0.25]

60 - 22 - 34 = 4 penalty minutes available [0.25]

1830 - 4 = 1826 on 18 Nov 85 started current 60 minutes [0.25]

Can go greater than 50% anytime after 1826 on 19 Nov 85 [0.20]

REFERENCE
TS, pgs B 3/4 2-1 & 2-2

K&A 001000A3.03/IF 3.6

ANSWER 4.10 (2.75)

a. BAST [0.25]
RWST [0.25]

b. 70 gpm [0.75]

c. with one pump it takes 1 minute to inject 200 pcm [0.50]

(1) (2 rods)x(1500 pcm/rod)x(1 min/200 pcm) = 15 min [0.50]

(2) (50 F)x(20 pcm/1 F)x(1 min/200 pcm) = 5 min [0.50]

REFERENCE
OP-II-3.3.8, Rapid Boration, pgs 1-3

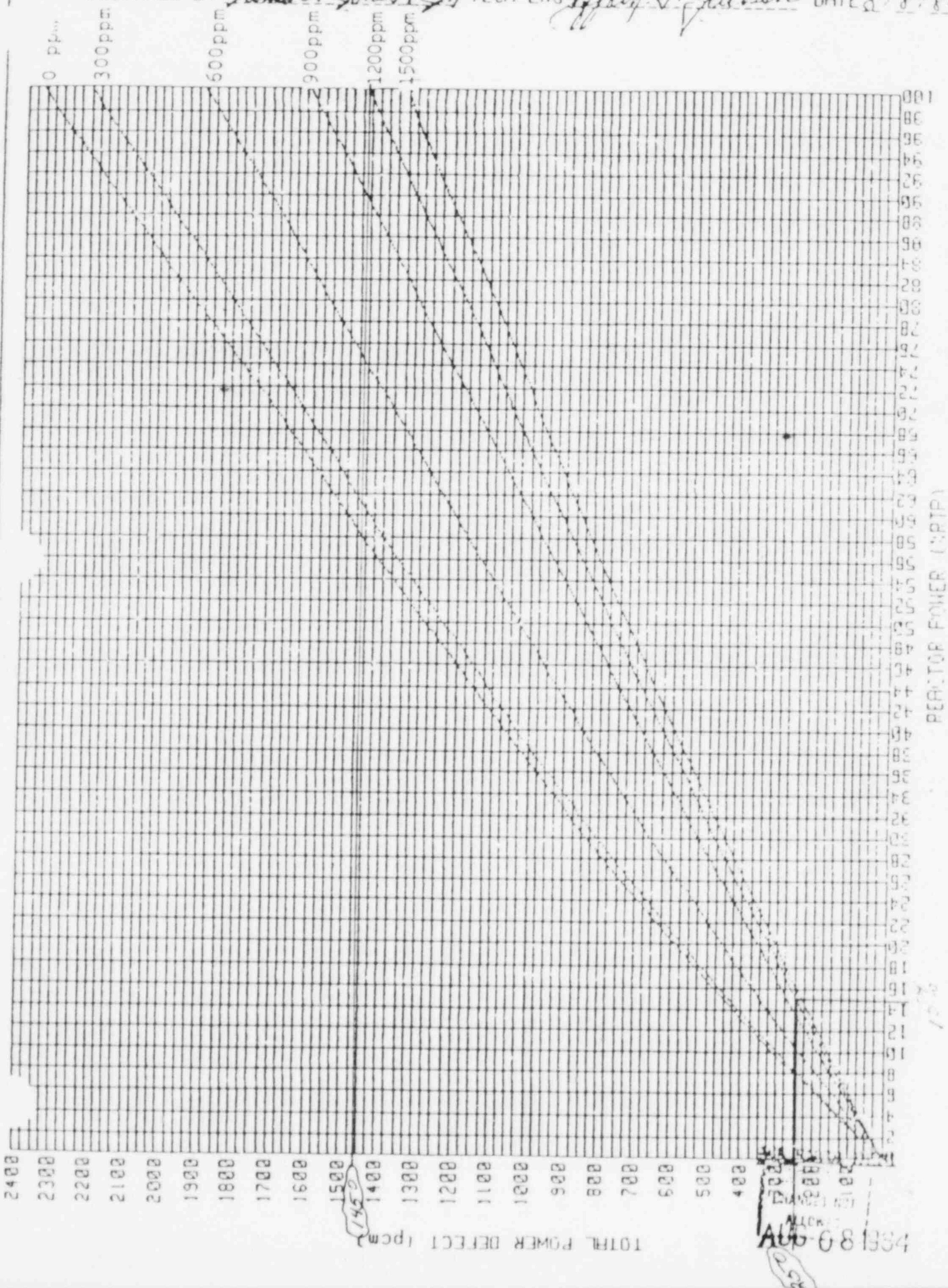
K&A 000024EA2.05/IF 3.3
000024EK1.01/IF 3.4
000029EK3.11/IF 4.2

FIGURE 1.1

SALEM 1 CYCLE 6

POWER DEFECT Vs. REACTOR POWER

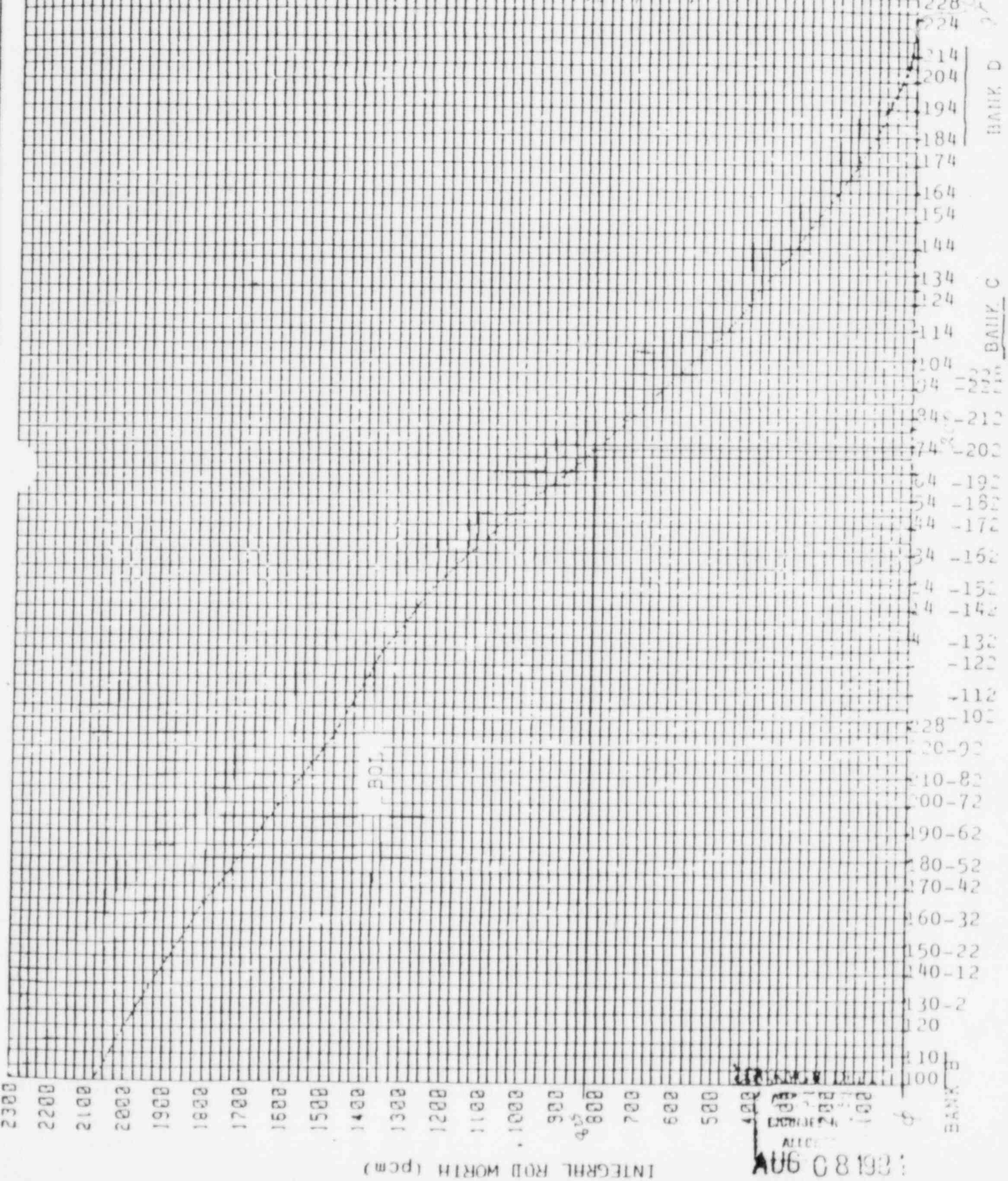
PREPARED BY Samuel L. Jones, Jr. TECH ENG Jeffrey G. Jackson DATE 8/8/84



SALEM 1 CYCLE 6 FIGURE 1.2

INTEGRAL ROD WORTH vs. POSITION IN OVERLAP

PREPARED BY [signature] TECH ENG [signature] DATE 8/8/84

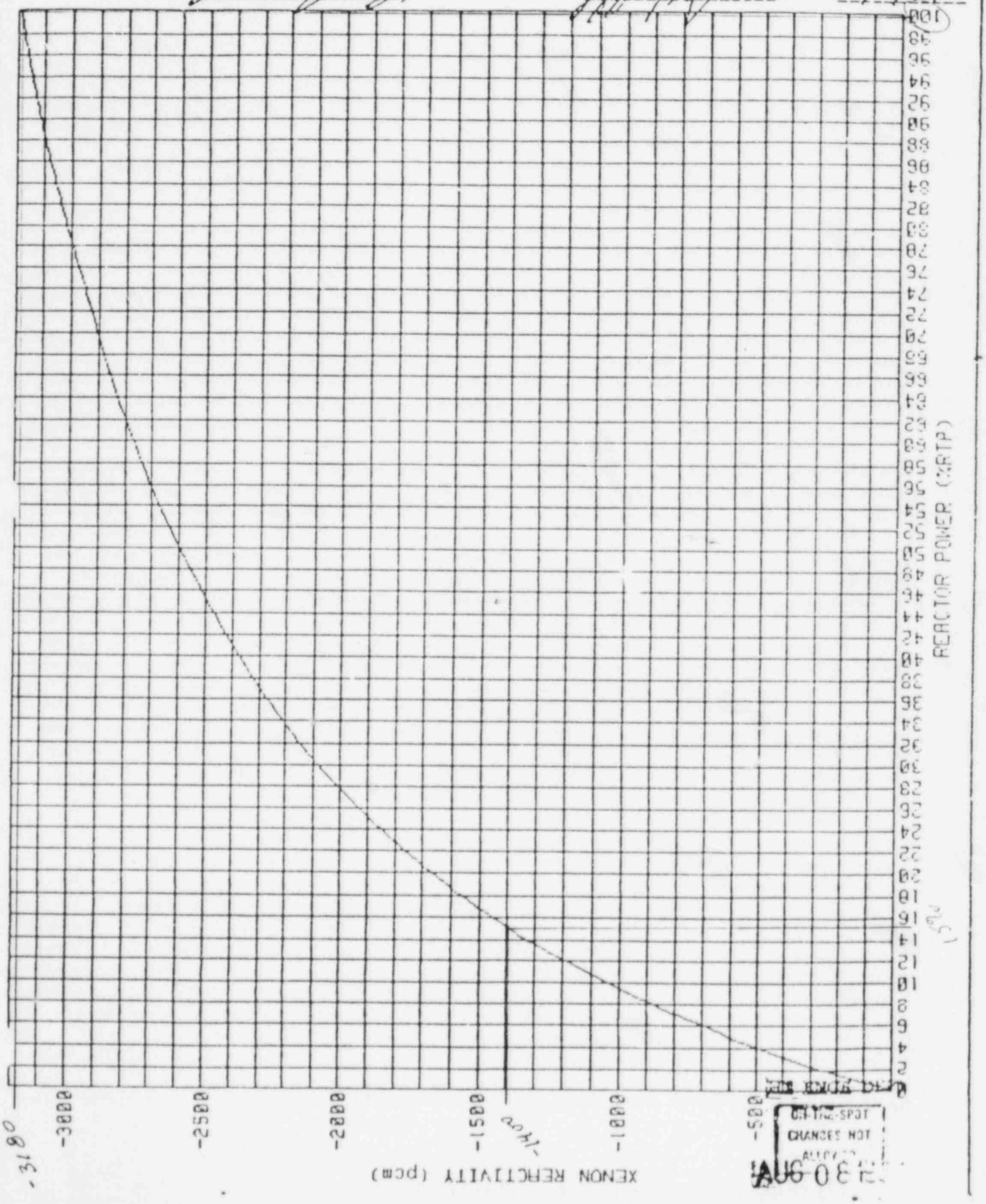


SALEM 1 CYCLE 6

FIGURE 1.3

EQUILIBRIUM XENON REACTIVITY vs REACTOR POWER

Prepared by [signature] TECH Engineer [signature] Date 8/8/84



SALEM 1 CYCLE 6 FIGURE 1.5

SAMARIUM REACTIVITY vs. TIME AFTER SHUTDOWN

BUILD UP FROM EQUILIBRIUM CONDITIONS (Eq Sm=588 pcm)

PREPARED BY James D. [illegible] Jr. TECH ENG. Jeffrey G. Jackson DATE 8/8/83

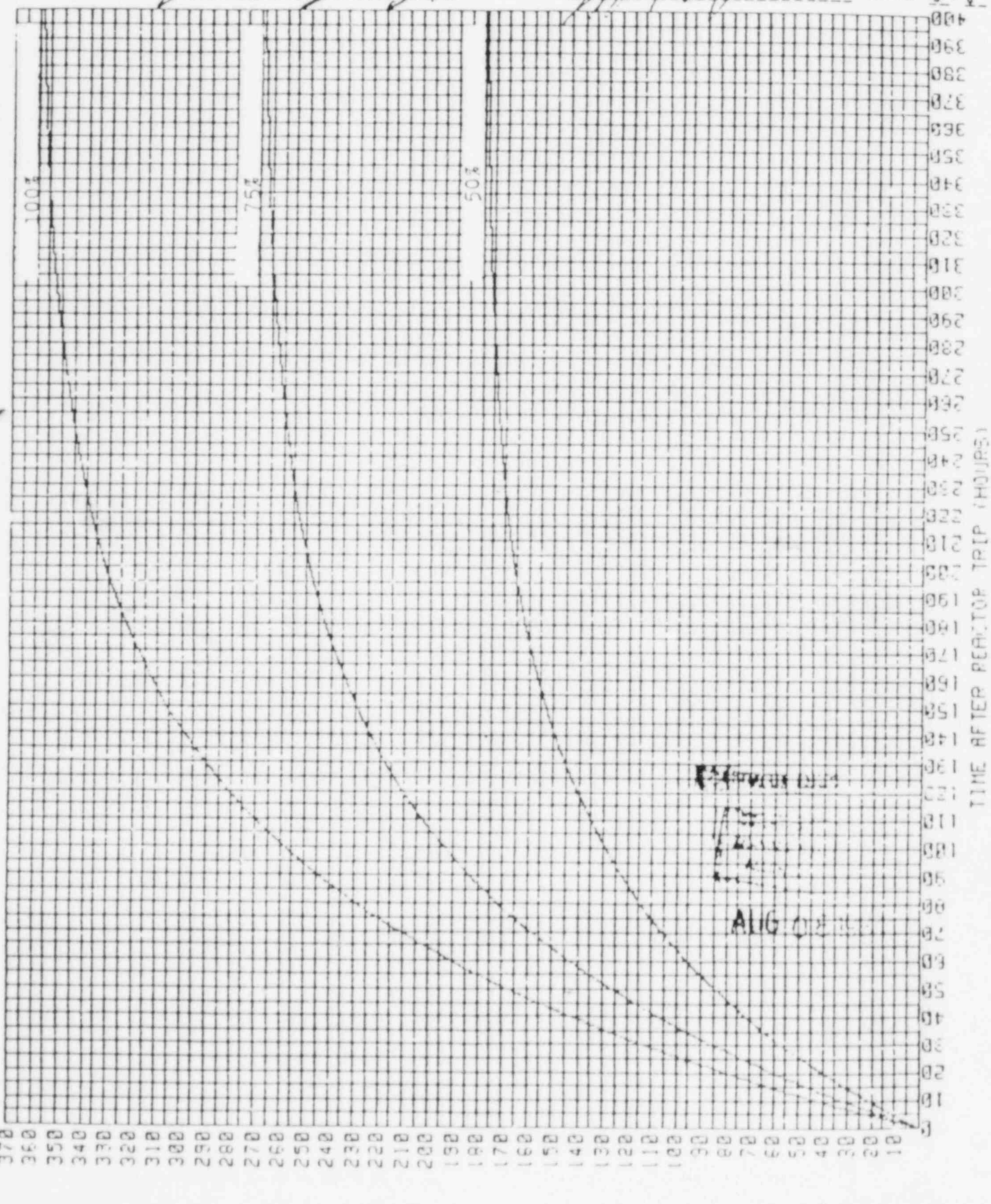
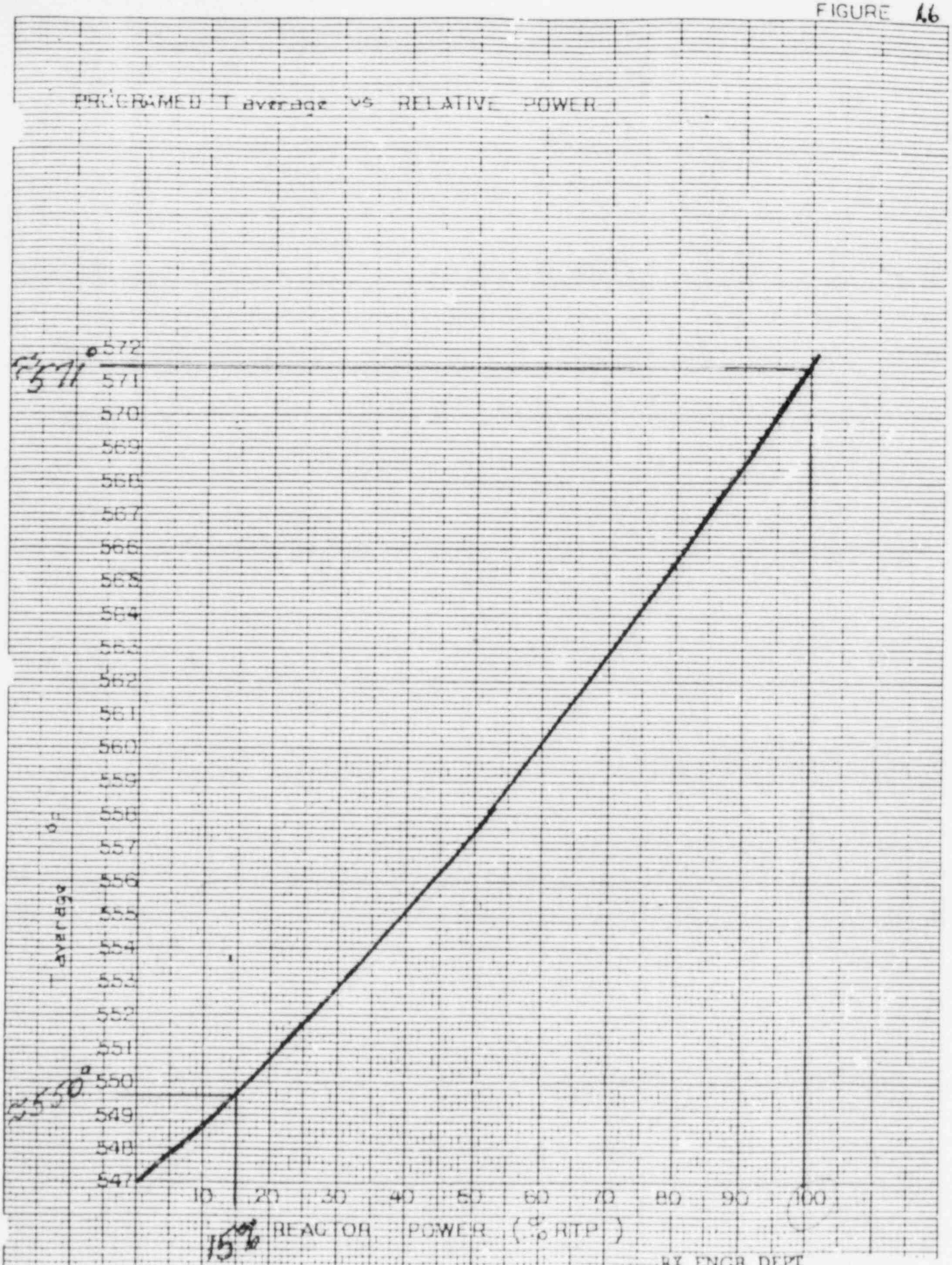


FIGURE 16



RX ENGR DEPT
SEP 10 1979

RX ENGR DEPT
SEP 10 1979

ON-THE-SPOT CHANGES NOT ALLOWED

FIGURE 1.7

SALEM 1 CYCLE 6

DIFFERENTIAL BORON WORTH vs. BORON CONCENTRATION

REACTOR ENGINEERING MANUAL

PREPARED BY [illegible] TECH ENG [illegible] DATE 8-8-84

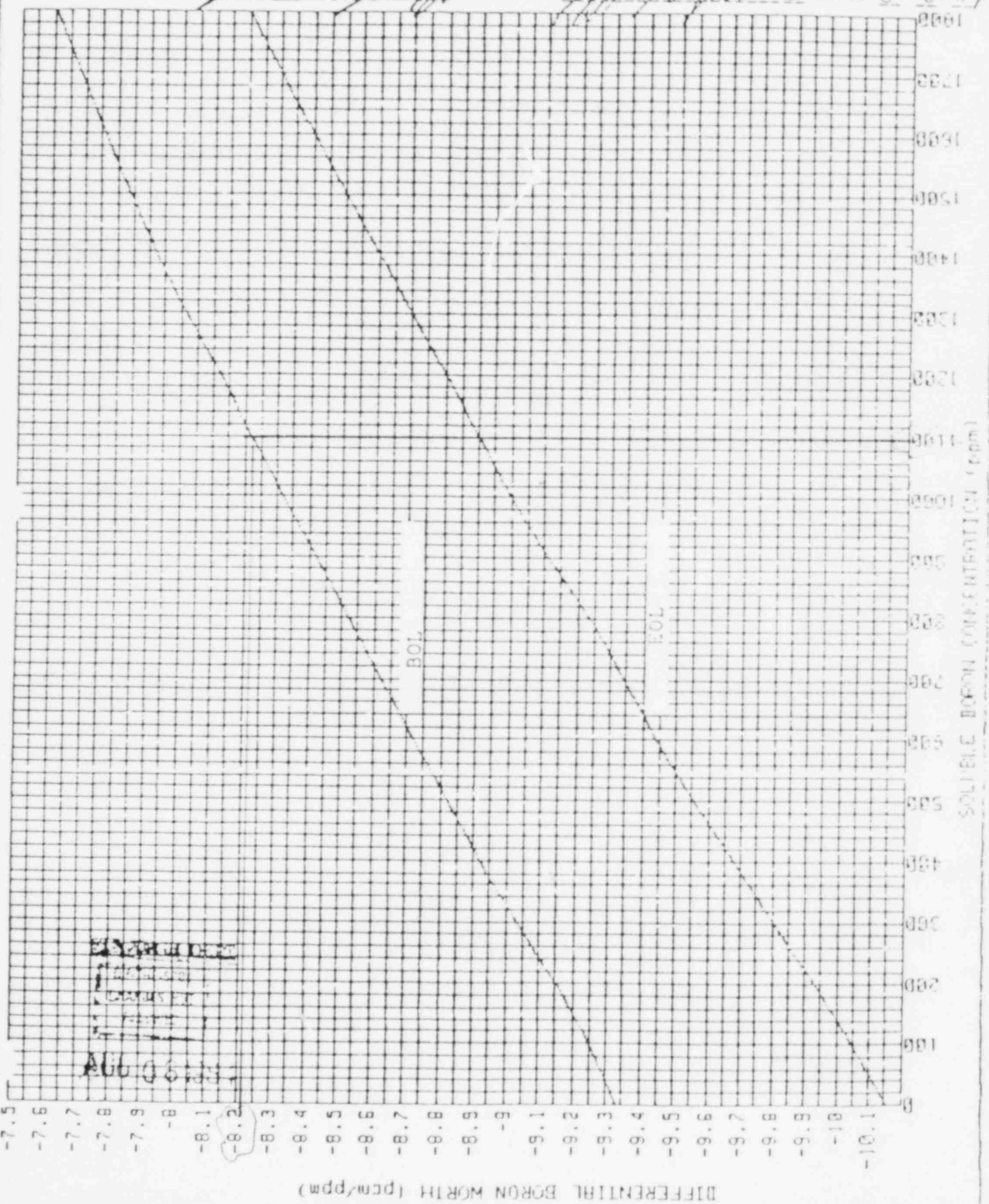
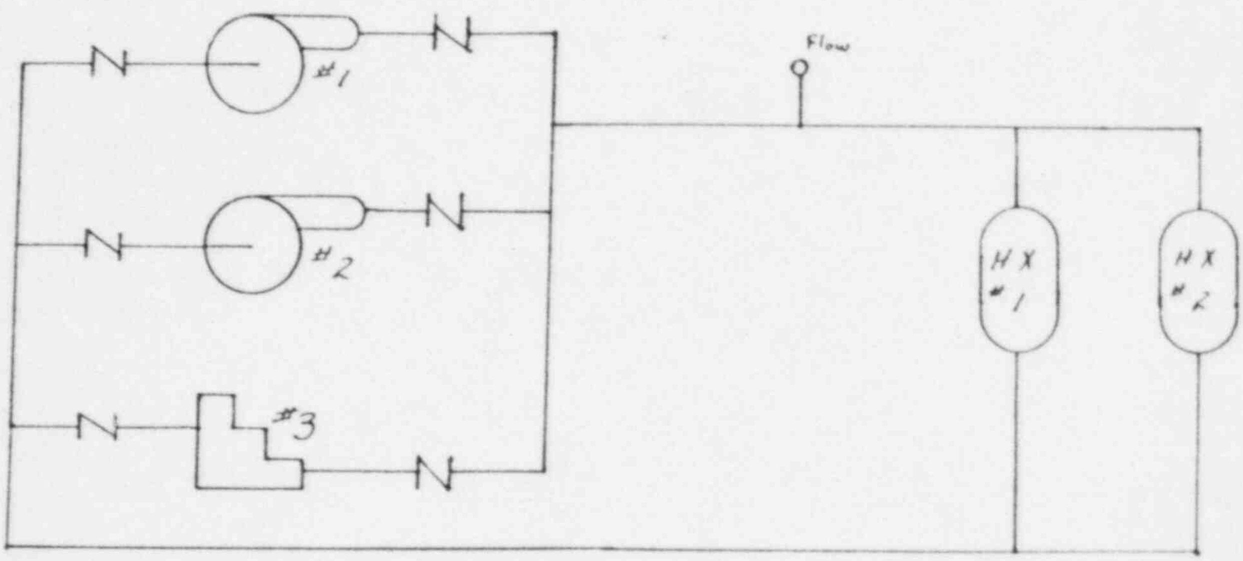
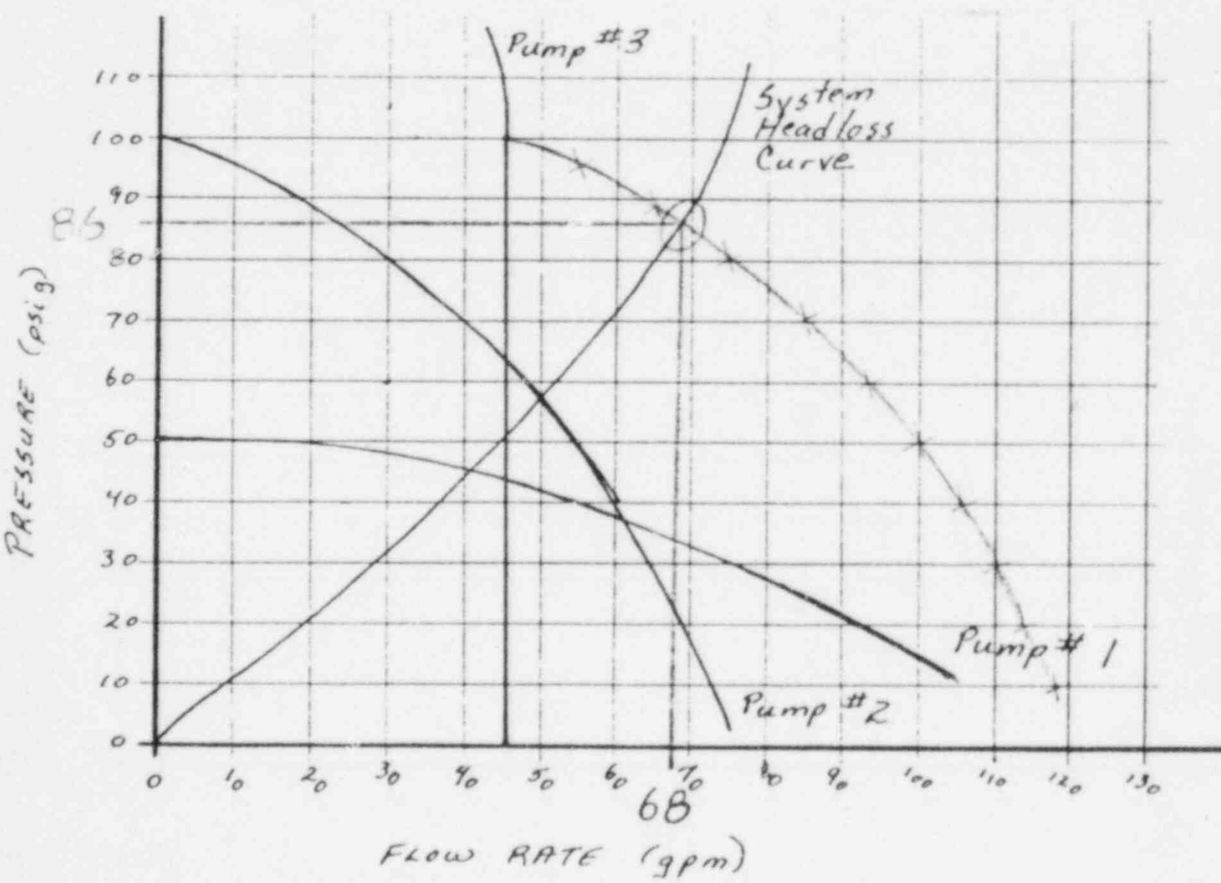


FIGURE 1.9



#1
#2
#3
Flow
HX #1
HX #2



Pump #3
System Headloss Curve
Pump #1
Pump #2
86
68
PRESSURE (psig)
FLOW RATE (gpm)

Figure 1.10

% Power
Xenon Reactivity (pcm)
Hours
0
20
40
60
80
100
120
140
160
180
200
-5000
-4000
-3000
-2000
-1000
0
10
30
50
70
90
50%
100%

EMERGENCY CORE COOLING SYSTEM ONE-LINE DIAGRAM

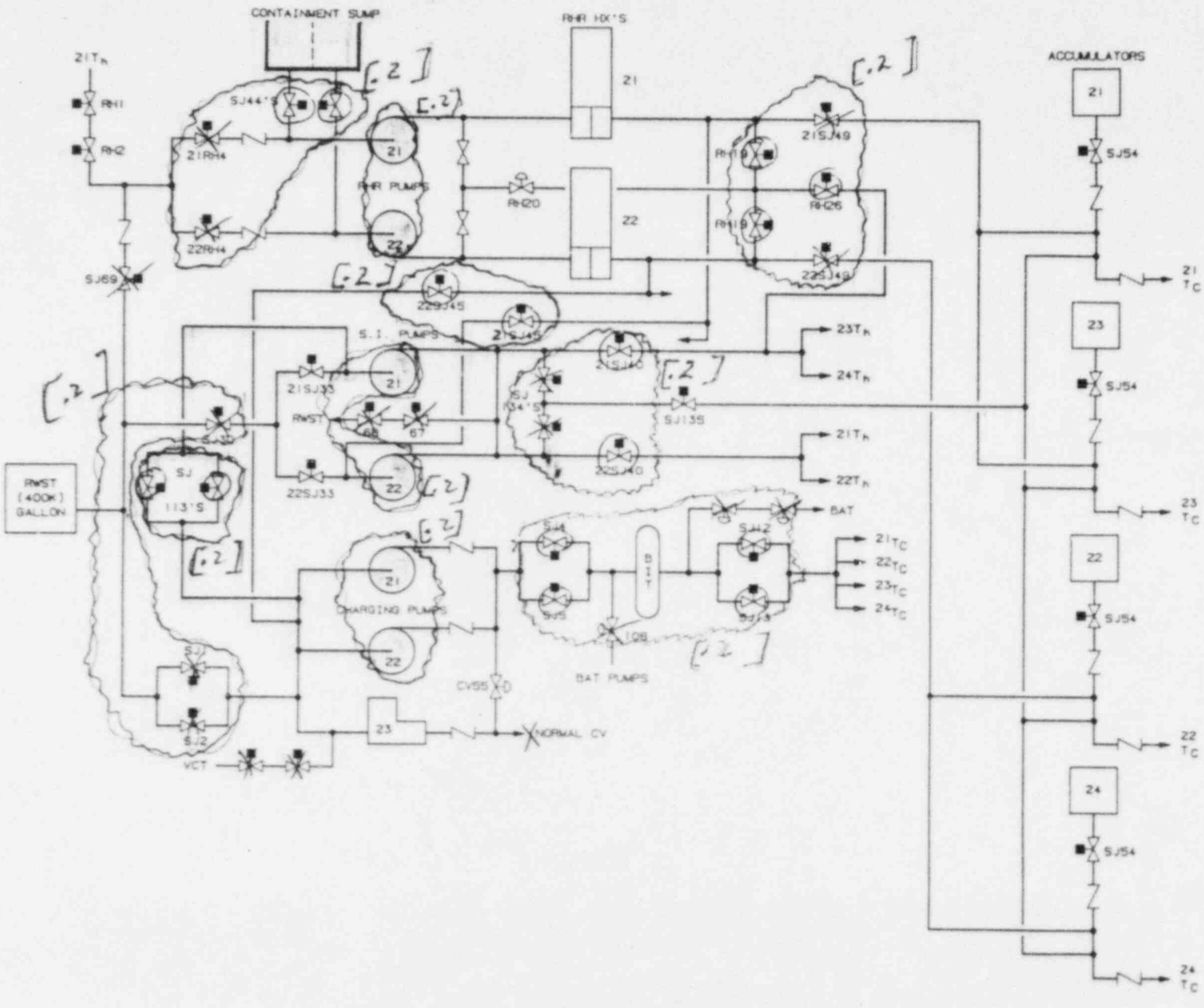
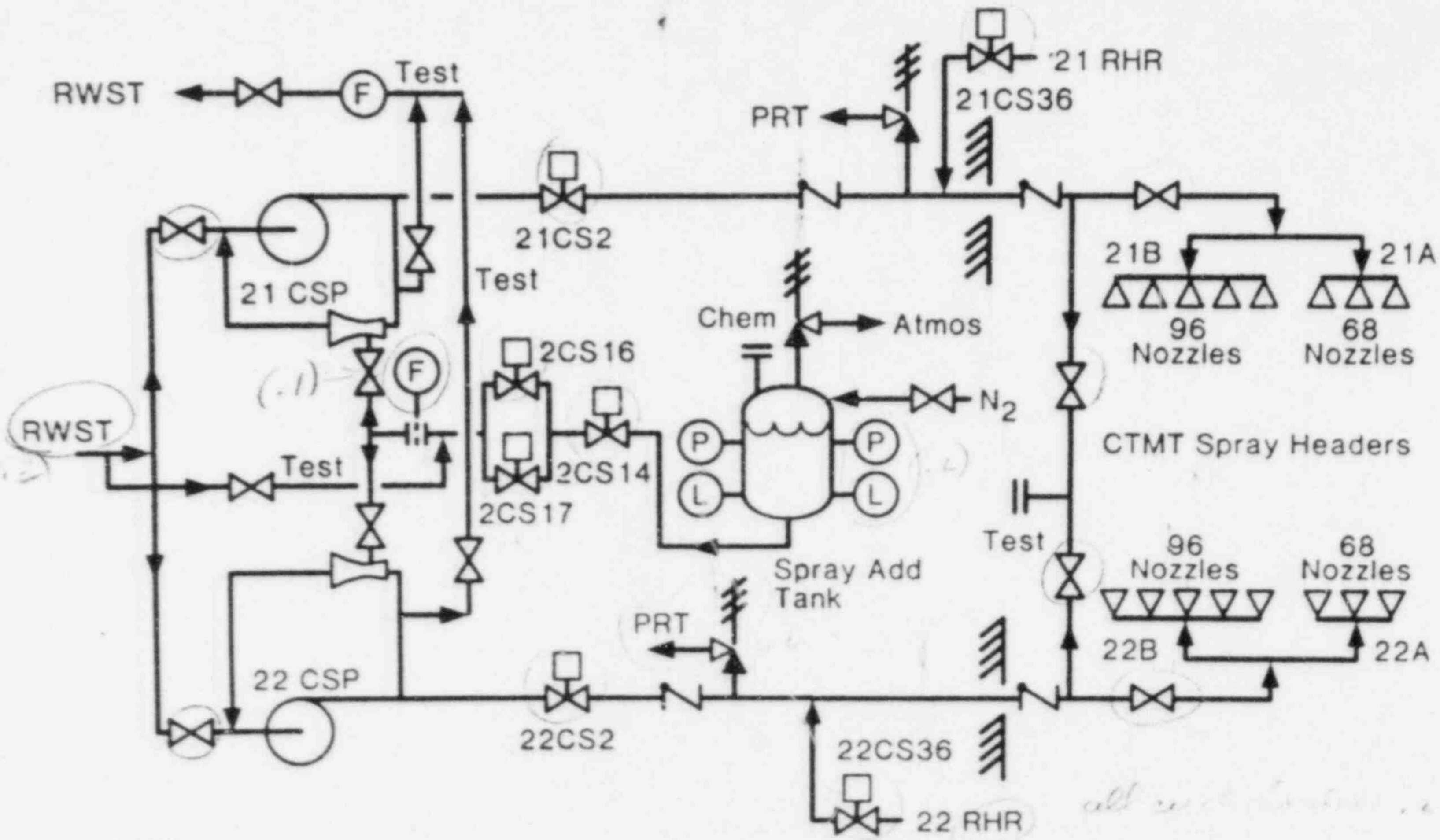


Figure 2.1

CONTAINMENT SPRAY



155 122682 006

Attachment 2

MASTER COPY

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

FACILITY: SALEM 1&2

REACTOR TYPE: PWR-WEC4

DATE ADMINISTERED: 85/11/19

EXAMINER: DUDLEY

APPLICANT: ______________________

INSTRUCTIONS TO APPLICANT:

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

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	25.00	________	________	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.00	25.00	________	________	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
23 ~~25~~.00	25.00	________	________	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.00	________	________	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
100.00	100.00	________	________	TOTALS

FINAL GRADE ________________%

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

____
APPLICANT'S SIGNATURE

QUESTION 5.01 (2.00)

a. List two causes of waterhammer. (0.8)

b. Give two examples of how waterhammer can be minimized. (1.2)

QUESTION 5.02 (2.50)

The Unit 1 reactor is operating at 50% power, BOL, when a steam dump fails open. Assume rods are in manual, no operator action is taken, and no reactor trip occurs. Explain HOW and WHY reactor power and Tave will change.

QUESTION 5.03 (2.00)

a. As fuel temperature increases the reasonance absorption peaks for U-238 become lower in height and the bands broaden but the area under the curve remains theoretically constant. Why then is heatup of the fuel a negative reactivity effect?

b. Does the doppler COEFFICIENT become more or less negative as fuel temperature increases? EXPLAIN.

QUESTION 5.04 (3.00)

For each of the parameters listed below, provide the desired indication or trending that would be expected for natural circulation cooling and what might result if the parameter was not trending as expected.

a. Th
b. Subcooling
c. Steam Generator Pressure
d. Steam Generator Level
e. Pressurizer Level

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.05 (2.50)

a. Explain why an ICRR is required for withdrawal of shutdown banks if count rate triples and is required during boron dilution when the countrate only doubles? (1.0)

b. When is an ICRR required during withdrawal of control banks? (1.5)

QUESTION 5.06 (3.00)

a. What will happen to the current drawn by a reactor coolant pump as the system is heated from 200 deg F to 564 deg F? Explain your answer.

b. What would happen to the current drawn by a reactor coolant pump if voids form in the reactor core during accident conditions? Explain your answer.

QUESTION 5.07 (3.00)

Indicate at which time in core life (BOL or EOL) the following accidents are more severe or result in a longer time spent at higher power. Justify your answer.

a. Main steam line break

b. Total loss of coolant flow

c. Rod withdrawal accident from low in the source range prior to any significant reactor coolant temperature increase.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.08 (3.00)

After operating for several weeks at 100% power, a reactor at BOL is shutdown for maintenance. An Estimated Critical Position (ECP) is calculated for a startup to be conducted 12 hours after the shutdown. EXPLAIN why the Actual Critical Position (ACP) is HIGHER than, LOWER than, or the SAME as the ECP if the following occur: (consider each independently)

a. The Steam Dump Pressure Controller is changed to 900 psia after the ECP calculation.

b. The startup is delayed 4 hours.

c. The rod worth curves used in the ECP apply for HFP not HZP.

QUESTION 5.09 (4.00)

Assume Unit 1 has just tripped after operating at 75% power for three days. The control rods were at 154 steps on Bank D, the boron concentration remains constant during the event at 1200 ppm, temperataure is being controlled at 547 F, and the core is near the beginning of cycle. Using the graphs provided answer the following questions, state all additional assumptions and show all calculations.

a. What is the margin by which the plant is shutdown. (1.5)

b. What is the difference between the margin by which the plant is shut down and shutdown margin as defined in Technical Specifications? (1.0)

c. How much makeup water must be added if the reactor is to be restarted 8 hours after the trip? Assume rods will be at at 154 steps on Bank D. (1.5)

(***** END OF CATEGORY 05 *****)

QUESTION 6.01 (1.50)

Answer the following questions concerning the Reactor Coolant Pumps:

a. List two design features which protect the RPCCW system from a failure/rupture in the thermal barrier cooling coil? (1.0)

b. Explain why VCT pressure is important to the operation of the RCP seals. (0.5)

QUESTION 6.02 (2.50)

The reactor is operating at a steady state 25% power, all control systems are in automatic. Turbine load is increased to 100% and the steam pressure detector for #1 S/G sticks at the 25% value. Explain the signal processing including all steps of the resulting transient and ending at the final stable conditions assuming no operator actions.

QUESTION 6.03 (2.50)

Answer the following questions:

a. What limits are the following RCS trips designed to protect against?
   1) OTDelta T
   2) OPDelta T
   3) Low pressurizer pressure
   4) Hi pressurizer pressure [0.4 each]

b. For a given RCS temperature, how does the low Pressurizer pressure trip serve to limit the range of OTDelta T setpoint? [0.9]

QUESTION 6.04 (2.00)

For each of the following Unit 1 area radiation monitors indicate what interlocks, if any, are actuated at the high alarm set points.

a. Control room (1R1A)
b. Fuel storage area (1R9)
c. Containment (1R21; 1R44A, B)
d. Fuel handling crane (1R23A, B)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.05 (2.00)

Indicate on the attached figure the expected electrical lineup for each of the following conditions.

a. No. 2 Main Generator Synchronized
b. Both Main Generators off the line
c. Blackout

QUESTION 6.06 (3.00)

a. What three pieces of equipment are protected by D.C. oil pumps during a loss of power accident? (1.0)

b. What five support systems are needed for cooldown after a loss of power accident? ASSUME ALL VITAL BUSES ARE AVAILABLE, (2.0)

QUESTION 6.07 (2.00)

What would be the approximate leak rate if a Loss of Coolant Accident occurred, all automatic safety injection systems functioned properly, pressurizer level stabalized, and RCS pressure stabalized at 1700 psig? Justify your answer.

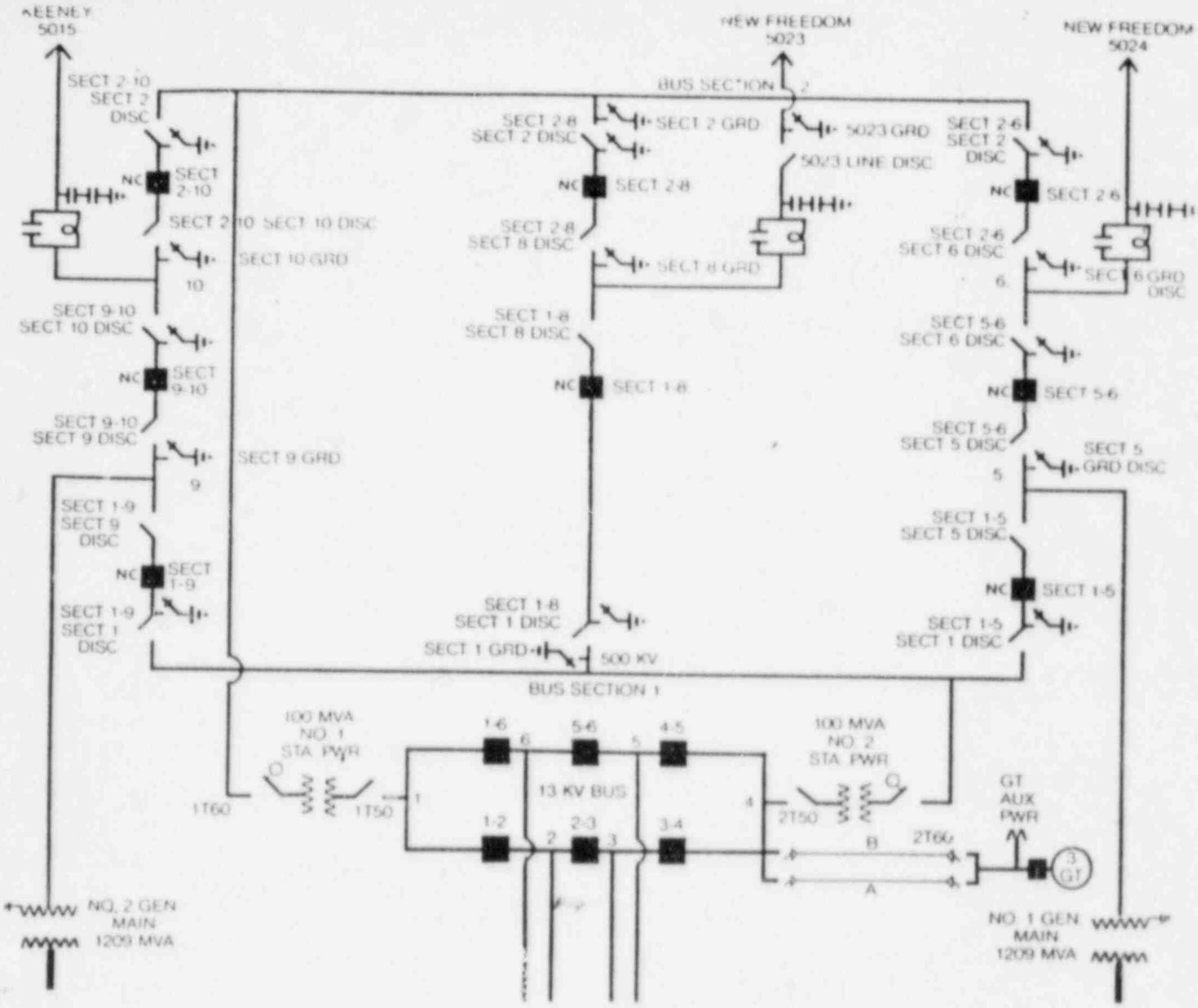
QUESTION 6.08 (3.00)

Answer the following questions concerning the Power Range NI's:

a. What happens when N41 is selected on the power mismatch bypass switch? List 3 systems which are affected.

b. List two reasons why the detector current comparator alarm function is automatically defeated when power is below 50%.

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)



LINEUP NUMBER	CIRCUIT BREAKER NUMBERS									
	1-5	5-6 (1)	2-6 (1)	2-8 (2)	1-8 (2)	1-9	9-10 (3)	2-10 (3)	500KV 2T60	500KV 1T60

QUESTION 6.09 (3.50)

What are the inputs or conditons which would cause actuation of the following Safeguard Equipment Control (SEC) signals. Provide set points and coincidence were applicable.

a. Mode 1; Accident only (2.7)
b. Mode 2; Blackout only (0.8)

QUESTION 6.10 (3.00)

a. In what positions of the Rod Control System Bank Selector Switch is 'Bank Overlap' in service? (0.5)

b. At what Bank B position should Bank C rods begin to be withdrawn during sequence operation? (0.5)

c. BRIEFLY EXPLAIN why the Rod Control System Startup Pushbutton is not used when recovering from a dropped control rod at power. (0.5)

d. Indicate the direction of rod motion (IN, OUT OR NONE) if the following instrument failures occur with RCCS in AUTOMATIC control at 50% power.

1) Loop 3 Tcold input to Tave fails LOW.
2) Turbine impulse pressure fails LOW.
3) Power range input channel fails high instantaniously. (1.5)

(***** END OF CATEGORY 06 *****)

QUESTION 7.01 (1.50)

After criticality is achieved, loop 1 Tave is noticed to be less than the minimum required temperature for critical operations. All other temperature channels are above this limit. What actions, if any, should be taken?

QUESTION 7.02 (.60)

If two rods do not fully insert on a reactor trip, how long should the reactor operator be directed to rapid borate? (Choose the most correct answer.)

A. Until RCS boron level increases by 150 ppm
B. Until Keff is 0.95
C. Until directed otherwise by Shift Supervisor
D. Until 16 minutes have passed

~~QUESTION 7.03 (2.00)~~

~~What TWO conditions might result if 21 RCP and 24 RCP were the only reactor coolant pumps operating during cooldown below 350 F?~~

QUESTION 7.04 (2.00)

What are FOUR plant conditions which place the plant on a RED PATH and requires the operator to utilize the status tree?

QUESTION 7.05 (2.40)

a. Explain why RCS pressure may decrease and stabilize at a new equilibrium value when Safeguards Pumps are stopped.

b. Explain why RCS subcooling requirements increase as the number of running Safeguards Pumps are decreased.

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.06 (2.00)

a. When must Reactor Coolant Pump 21 be secured if all component cooling water is lost to the pump while Unit 2 is operating at 80% power. (1.5)

b. What is the maximum amount of time a reactor coolant pump can be operated if both seal injection and component cooling water are lost and no temperature limits are exceeded? (0.5)

QUESTION 7.07 (2.00)

a. What is the definition of a partial loss of Reactor Coolant? (0.8)

b. At what point during a partial loss of Reactor Coolant accident should Safety Injection be initiated and the "Safety Injection Initiation" procedure, EI-4.0, be entered? (1.2)

QUESTION 7.08 (3.00)

An entry into the containment is required while at 100% power and will result in an estimated whole body dose of 120 mrem. The following four candidates are equally qualified to perform the task. Which candidates may be allowed to perform the task in accordance with administrative procedures?

Explain your reasons for accepting or rejecting each candidate. No waivers can be obtained.

CANDIDATE	1	2	3	4
SEX	male	male	female	male
AGE	20	38	24	27
WK/EXPOSURE	100mrem	30mrem	0mrem	150mrem
QT/EXPOSURE	1900mrem	800mrem	20mrem	?
ACCUM LIFE EXPOSURE	9900mrem	4000mrem	2200mrem	?
REMARKS	none	None	3 months pregnant	History unavailable

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.09 (3.00)

If Unit 2 is at 100% power, what Technical Specifications should be consulted for immediate (one hour) action statements for each of the following:

a. Loss of 2C 115 V Vital Instrument Bus (THREE REQUIRED) (1.5)
b. Loss of 2C 125 VDC Bus (ONE REQUIRED) (0.5)
c. Loss of 2C 4 KV Vital Bus (TWO REQUIRED) (1.0)

QUESTION 7.10 (3.50)

For each of the following situations indicate whether the reactor should be tripped, shutdown, operated at reduced power, or operated at 100% power. Also indicate what deteriorated conditions, if any, would require an immediate reactor trip. Assume plant is at 100% power and consider each situation seperately.

a. Turbine output is 90% when reactor power is 100% due to a steam leak downstream of the MSIV.

b. Circulating water flow to the condenser is reduced due to pump failures. Vacuum has decreased to 23 inches.

c. Pressurizer heaters are deenergized and cannot be energized from their normal power supply.

d. Control air pressure drops to 80 psig.

e. Both seals of the ~~emergency escape~~ EQUIPMENT hatch are found to be leaking excessively.

QUESTION 7.11 (3.00)

In accordance with EOP-TRIP-1, what steps should be taken if a reactor trip is not confirmed?

(***** END OF CATEGORY 07 *****)

QUESTION 8.01 (1.50)

During a backshift work is being done to replace and reposition supports for the discharge piping of an HPSI pump. The workers cannot find the supports specified in the work package and have determined that it is impossible to install the supports at the required locations. The workers have located some pipe hangers in the shop and want permission to install them as close as possible to the specified locations. What actions, if any, should the shift supervisor take? Support your answer.

QUESTION 8.02 (2.50)

What two people must the Senior Shift Supervisor contact or notify following a reactor trip and what type of notifications or requests should be made?

QUESTION 8.03 (2.00)

Following watch relief, may the SNSS order the oncoming SRO (NSS) from Unit 1 to relieve the off-going SRO (NSS) from Unit 2 if the relief for the Unit 2 SRO is expected to arrive within half an hour. Justify your decision assuming both plants are at power and the STA's are unlicensed.

QUESTION 8.04 (2.00)

What four actions or reports must be completed prior to allowing Unit 1 to return to unrestricted operations following a loss of feedwater transient which resulted in indicated RCS pressure of 2750 psig.

QUESTION 8.05 (2.00)

What are four of the five basis for the requirement to maintain RCS temperature above 541 F during critical operations?

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.06 (2.00)

a. What person or organization is responsible for scheduling Technical Specification surveillance items? (0.5)

b. What extensions, if any, are allowed for completing Technical Specification surveillance items beyond the specified time interval? (1.5)

QUESTION 8.07 (2.50)

What type of safety or jumper/lifted lead tags, if any, are required for each of the following situations?

a. Leads need to be lifted in accordance with an installation procedure for a new input to the annuniator panel.

b. A blind flange is to be installed in the CVCS system while a relief valve is being calibrated.

c. Work is to be conducted on the Reactor Coolant Pump motor which will require jogging the pump.

d. Work is being done on the ventilation system which requires making damper adjustments with no flow and then checking that the required flow has been established.

e. Following an outage it is discovered that the hand switches for the spray valves have been incorrectly wired so that each switch operates the opposite spray valve. Work is to be done to correct the wiring problem.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.08 (2.50)

Should each of the following requests be allowed to be conducted? Justify your decision. Unit 1 is in mode 6 and Unit 2 is at 100% power. Consider each request separately.

a. A request to conduct an approved modification on a safety nuclear instrument channel which will require deenergizing the detector. Over Power Delta T channel C is tripped. UNIT 2

b. A request to commence the procedure for replacement of the reactor vessel head while ~~both~~ ALL pressurizer code safety valves are inoperable.

c. A request to replace the governor on Unit 2 Emergency Diesel Generator number 2.

d. A maintenance request to replace the seals on both air lock doors on Unit 1.

e. A maintenance request to replace the HEPA filters in the Auxiliary Building ventilation exhaust system. UNIT 2

QUESTION 8.09 (2.00)

Technical Specification 3.4.5 states 'two power operated relief valves (PORV's) and their associated block valves shall be operable'. For the following situations state what actions are required to be taken within an hour if operations at power are to continue?

a. One or more PORV's inoperable

b. One or more block valves inoperable

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.10 (3.00)

The plant is operating at 55% power and the latest leak rate data shows:

6.2 gpm - Total leakage
3.1 gpm - Leakage due to a leaking charging pump relief valve
1.2 gpm - Leakage into the RC Drain Tank
1.5 gpm - Leakage through a RHR discharge valve
0.8 gpm - Total primary to secondary leakage
4.2 gpm - Leakage past RCP seals

a. What actions, if any, are required in Unit 1? Justify your answer. (1.0)

b. What actions, if any, are required in Unit 2? Justify your answer. (2.0)

QUESTION 8.11 (3.00)

Classify each of the following occurances using the Event Classification sheets provided. Identify what factor was used to determine the final classification.

a. A small plane crashes on the river bank across from the plant. The fuel tanks of the plane have ruptured and the radioactive medical isotopes the plane was carrying cannot be located.

b. While working with the polar crane in the containment during Mode 5, the brake on the hook fails, dropping the hook onto a piping run. The sample line from the pressurizer liquid space ruptures and area monitors R41B and R41C increase readings to 1.7E3 cpm.

c. An accident in the containment results in an injury to a worker. The worker is in contaminated anti-C clothing and has broken his leg. Due to the accident the worker receives 2 Rem of radiation Previously he had accumulated 500 mrem for the quarter and year.

d. Following operations at 100% power for six months a loss of coolant accident occurs which initiates containment spray. During the transient five relief valves on the steam generator lift and two fail to reseat.

(***** END OF CATEGORY 08 *****)
(************* END OF EXAMINATION ***************)

ANSWER 6.01 (1.50)

a. A downstream valve automatically closes if hi-flow is sensed. ~~[0.5]~~
An upstream check valve prevents backflow ~~[0.5]~~

b. VCT pressure is kept greater than 15 psig to ensure #2 seal is not robbed of a trickle cooling flow. [0.5]

REFERENCE
SNGS Student Notes Chap. 4, RCP, p 20, 25

ANSWER 6.02 (2.50)

As power increases, steam flow detector delta-P increases [0.4] and steam pressure decreases [0.4].
Since steam flow uses a squre root extractor corrected for steam pressure and steam pressure is failed to 24%, indicated flow will be higher than actual flow. [0.5]
Flow signal will open the FWRV. [0.4]
As level increases, the level signal will close FWRV. [0.4]
Eventually, the level error will cancel the flow error and steam flow will equal feed flow at a higher level [0.4]

REFERENCE
SNGS Student Notes, Chap 27, SG Water Level Control, p 21, 22, 26

ANSWER 6.03 (2.50)

a. 1) DNB
2) excessive fuel power and temperatures
3) DNB
4) protects RCS integrity from overpressure [0.4 each]

b. Since RCS pressure is an input to OTDelta T and lower pressure lowers the setpoint, the low pressurizer pressure reactor trip places a lower limit on OTDelta T range at a given RCS temperature. [0.9]

REFERENCE
SNGS Student Notebook Chap. 28, RPS, p 12

ANSWER 6.04 (2.00)

a. Control room ventilation isolation [0.5]
b. ~~Fuel handling area alarm [0.1]~~ (ALERT + WARNING)
FHB exhaust ventilation shifts No. 22 filter unit [0.4] 5
c. None [0.5]
d. ~~Emergency warning light [0.1]~~
Prevents crane hoist up operations [0.4] 5

REFERENCE
System Operations Procedures, IV-11.3.1, Table 1
SNGS Student Notebook Chap. 17, Radiation Monitoring System, p 26-28

ANSWER 6.05 (2.00)

X - CLOSED
O - OPEN

LINEUP NUMBER	CIRCUIT BREAKER NUMBERS									
	1-5	5-6 (1)	2-6 (1)	2-8 (2)	1-8 (2)	1-9	9-10 (3)	2-10 (3)	500KV 2T60	500KV 1T60
a	O	O	X	X	X	X	X	X	X	X
b	O	O	X	X	X	O	O	X	X	X
c	O	O	X	O	O	O	O	O	O or X DEPENDING	O or X ON STATED

ASSUMPTION

REFERENCE
System Operating Procedure, IV-1.3.1

ANSWER 6.06 (3.00)

a. SG feed pumps [0.4]
Turbine generator [0.3] (EMERGENCY OIL PUMP)
Generator (Emergency seal oil pump) [0.3]

b. RHR system
CCW system
Primary makeup system
AFST SG makeup
Charging and letdown ~~[0.4] EACH]~~
SW / MS-10 [any 5 @ 0.4 EACH]

REFERENCE
EOP - LOPA-1, p 2, 14

ANSWER 6.07 (2.00)

Leakage would be the sum of high ~~and intermediate~~ head pumps discharge flow rate. [0.6]
and positive displacement pump

High head flow	2 X ~400 gpm =	800 gpm	[0.6]
~~Intermediate head~~	~~2 X 425 gpm =~~	~~850 gpm~~	~~[0.6]~~
Positive displacement pump		75 gpm	[0.6]
	LEAK RATE =	~~1650~~ 875 gpm	[0.2]

REFERENCE
SNGS Student Notebook Chap , ECCS, Table 1, Fig.

ANSWER 6.08 (3.00)

a. Input from PR channel N41 is defeated to the auctioneering units [0.~~6~~7] of the Rod Control System [0.~~9~~7], Axial Flux Distribution Monitoring System [0.3], ~~and SGWLC System [0.3].~~

b. 1) Below 50% power, heat generation is sufficiently low to assure safety [0.75]
   2) Since the setpoint is 1.02 times the average detector output, at low powers even slight differences would cause the alarm [0.75]

REFERENCE
SNGS Student Notebook Chap. 20, Excore NI, p 24, 25

ANSWER 6.09 (3.50)

a. Low PZR pressure [0.3]	1765 psig [0.1]	2/3 [0.1]
High Containment pressure [0.3]	4 psig [0.1]	2/3 [0.1]
High Steamline Dif. pressure [0.3]	100 psid [0.1]	1 to 2 [0.1]
High Steam flow [0.1]	40-110% power [0.1]	with low-low Tavg or low Steam line Press. [0.3]
MANUAL [0.2]		
b. UV on 4 KV buses [0.3]	70% for 3.5 sec. [0.2] 90% for 10.5 sec. [0.2]	2/3 [0.1]

REFERENCE
SNGS Student Notebook Chap 10, ECCS, p 8, 9
SNGS Student Notebook Chap 13, SEC System, p 3, 35

ANSWER 6.10 (3.00)

a. - AUTOMATIC [0.25]
   - MANUAL [0.25]

b. ~~256 steps or 50% withdrawal of bank B [0.5]~~
   128 STEPS ON BANK B [0.5]

c. The pushbutton will rezero all rods (only the affected group rods require reset). [0.5]

d. 1) NONE
   2) IN
   3) ~~NONE~~ IN [0.5 each]

REFERENCE
SGNS Student Notebook, Chap 22, RCS, p 20-21, 26, 27, 31, RS-15

ANSWERS -- SALEM 1&2 -85/11/19-DUDLEY

ANSWER 7.01 (1.50)

Restore Tave within limits within 15 minutes by using rods, boron, or steam demand. [1.0]
If unable to restore temperature be in hot standby within the next 15 minutes. [0.5]

REFERENCE
IOP-3, p 2
TS, p 3/4 1-6

ANSWER 7.02 (.60)

D. Until 16 minutes have passed

REFERENCE
EOP - LOPA-2, p 13

DELETE
~~ANSWER 7.03 (2.00)~~

~~Uneven RCS temperatures among the loops. [1.0]~~
~~Conditions in the SG's leading to inadvertent SI created by SG differencial pressure. [1.0]~~

REFERENCE
IOP-6, p 12

ANSWER 7.04 (2.00)

Nuclear power > 5% after reactor trip
Core exit Tc > 1200 F
Core exit Tc > 15 700 F and RVLIS full range < ~~13~~ 50 and no RCPs running
NR SG level < ~~14~~% and feed flow < ~~15 gpm~~ 22E4 lbm/hr
Tc decrease > 100 F in 60 minutes and ~~Tc < 16 F~~ Left of limit curve
CTMT press > ~~17~~ 47 psig
[any 4 @ 0.5 each]

REFERENCE
Westinghouse Owner Group Emergency Response Guidelines, Executive Summary; Generic Issue Foldout Page Items, Fig 1

ANSWERS -- SALEM 1&2 -85/11/19-DUDLEY

ANSWER 7.05 (2.40)

a. Flow is reduced.
Head loss due to friction in the pipes decreases.[0.6]
Therefore discharge head of pump decreases. [0.6]

BREAK FLOW AND INJECTION FLOW [illegible] EQUAL [illegible] AT [illegible]

b. ~~Since less energy is being removed from the core greater subcooling must be maintained to ensure no boiling in the core. [1.2]~~ IF BREAK FLOW IS GREATER THAN [illegible] PUMP FLOW PRESSURE WILL DROP TO SATURATION. THEREFORE A GREATER MARGIN TO SATURATION IS ~~[illegible]~~ REQUIRED. [1.2]

REFERENCE
EOP - FRHS-1, p 16
General Physics, Thermodynamics and Fluid Flow Fundamentals, p 302

ANSWER 7.06 (2.00)

a. Within 5 minutes. [0.5]
or a high motor bearing temperature of 175 F. [0.5]
Pump radial bearing temperature reaches 210 F. [0.5]

b. 1 minute [0.5]

REFERENCE
System Operating Instructions, II-1.3.1, p 3, 4

ANSWER 7.07 (2.00)

a. A small RCS break which can be compensated for by the available charging pumps. [0.7]

b. If pressurizer pressure and/or level continues to decrease [0.7] after starting additional charging pumps [0.2], reducing letdown flow [0.2] and energizing pressurizer heaters. [0.1]

REFERENCE
I-4.17, p 10

ANSWER 7.08 (3.00)

EXCEED 2000 mr/qt

1. No [0.35] because he would exceed 5(N-18) limit. [0.4]
2. Yes [0.35] exposure would be less than ~~2100~~ 1900 mrem/qt [0.4]
3. Yes [0.35] because she would not exceed limit of 500 mrem/~~qt~~ [0.4]
4. No [0.35] because he ~~would~~ HAS exceed ~~2000~~ 100 mrem/qt [0.4] gestation period

REFERENCE
Salem AD-24, p 13

ANSWER 7.09 (3.00)

a. Reactor Trip Instrumentation [0.5]
   PORV Operability [0.5]
   A.C. Electrical System [0.5]
   ESF INSTRUMENTATION
b. RCS Loop-Startup and Operation [0.5]
   OTHER A.C. SOURCES OPERABLE
c. PORV Operability [0.5]
   A.C. Electrical System [0.5]

REFERENCE
AOP-ELEC-VIB-C, p APPX 1-1
AOP-ELEC-125V-C, p APPX 1-1
AOP-ELEC-4KV-C, p APPX 1-1

ANSWER 7.10 (3.50)

a. Shutdown reactor. [0.4]
   No requirement for a reactor trip [0.3]
b. Reduce power to maintain vacuum. [0.4]
   If turbine vibration reaches 7 mils trip turbine. [0.3]
c. Shutdown reactor. [0.4]
   If PZR pressure decreases below 1865 [0.3]
d. Continue to operate. [0.4]
   If pressure reaches 65 psig trip reactor. [0.3]
e. Shutdown reactor. [0.7]
   RESTORE ~~[illegible]~~ CONTAINMENT INTEGRITY

REFERENCE
I-4.26 Secondary plant leak, p 1
I-4.13 Loss of circulating water/loss of condenser vacuum, p 2
I-4.24 Pressurizer pressure control malfunction, p 10, 11
I-4.18 Loss of control air, p 2
I-4.23 Loss of containment integrity, p 2

ANSWER 7.11 (3.00)

Trip reactor with second trip handle. [0.5]
Open both reactor trip breakers. [0.5]
Open 2E6D and 2G6D. [0.5]
Dispatch operator to open Reactor Trip breakers and trip Rod MG set. [0.7]
Manually insert control rods. [0.5]
Go to EOP (FRSM-1) response to nuclear power generation. [0.3]

REFERENCE
EOP - TRIP-1, p 3

ANSWERS -- SALEM 1&2 -85/11/19-DUDLEY

ANSWER 8.01 (1.50)

~~Initiate a Revision Request Form, AD-1-A-1 and forward to the Senior Operations Technicial Supervisor (SOTS). [0.6]~~
~~Job cannot be handled as On-the-Spot Changes. [0.9]~~

NOT ALLOW WORK
DEVIATES FROM ORIGINAL WORK PACKAGE
REQUIRES INITIATION OF SYSTEM DESIGN CHANGE [0.5 each]

REFERENCE
AD-1, p 1, 8

ANSWER 8.02 (2.50)

Controls Engineer [0.5] request assistance for review of sequence of events print out [0.7]

Operations Manager [0.5] notification of the event [0.4] and subsequent findings of the review report. [0.4]
STA, LOCAL COMMUNITIES, NRC

REFERENCE
AD-16, Post Reactor Trip/ Safety Injection Review, p 1-2

ANSWER 8.03 (2.00)

No. [0.7] The off going Unit 1 NSS must stay to met the T.S. manning requirement. [0.8] Action statements cannot be entered for operating convience. [0.5]

REFERENCE
Salem T.S., p 6-4, 6-5

ANSWER 8.04 (2.00)

Unit must be placed in hot standby in one hour. [0.~~7~~5]
NRC is informed within one hour. [0.~~7~~5]
Safety Limit Violation Report shall be prepared. [0.~~3~~5]
Report is submitted to NRSC and Upper Management (within 14 days) [0.~~3~~5]

REFERENCE
Salem 1 T.S. p 6-12a

ANSWER 8.05 (2.00)

MTC is within normal analyzed range.
Protective instrumentation is within its normal operating range.
P-12 is above its setpoint
The prssurizer is capable of being opeable with a steam bubble
The reactor vessel is above minimum RT NDT

[any 4 @ 0.5 each]

REFERENCE
Salem T.S., p B3/4 1-2

ANSWER 8.06 (2.00)

a. (CAF) Operations department. [0.5]

b. Extension of up to 25% of interval [0.75]
Three consecutive intervals shall not exceed 3.25 times the normal interval. [0.75]

REFERENCE
salem TS, p 3/4 0.2

ANSWER 8.07 (2.50)

a. None
b. Jumper/lifted lead tag
c. Red blocking tag ~~and Yellow Permissive tag~~ WITH TEMPORARY RELEASE
d. RED ~~Worker~~ blocking tag WITH TEMPORARY RELEASE
e. ~~White caution tag.~~ NONE

[0.5 each]

REFERENCE
Salem AD-13, p 1, 2
Salem AD-15, p 1-3

ANSWER 8.08 (2.50)

a. ~~No~~ Yes [0.2] ~~will result in trip on 2/4 on OPDT.~~ No Bistable for OPΔT from N.I. [0.3]

b. No [0.2] cannot enter another Mode with reliance on action statement [0.3]

c. ~~No [0.2] do not intentionally enter action statement for maintenace. [0.3].~~ Yes [0.2] Maintenance and entry into action statement has been specified by licensing supervisor. [0.3]

d. Yes [0.2] if only one seal is replaced at a time and other door is kept shut. [0.3]

e. Yes [0.2] if one train at a time is taken out of service. [0.3]

REFERENCE

Salem TS, p 3/4 3-5
3/4 4-4, 3/4 0-2
3/4 8-1
3/4 9-4
3/4 7-22

ANSWER 8.09 (2.00)

a. Either restore the PORV(s) to operable status [0.5] or close the associated block valve(s) and remove power from the block valve [0.5].

b. Either restore the block valves to operable status [0.5] or close the block valves and remove power from the block valves [0.5].

REFERENCE

Unit 2: TS, p 3/4 4-8

ANSWER 8.10 (3.00)

a. No action required since no leakage limits exceeded. [1.0]

b. Close 2 manual valves or deactivate auto valves to isolate HP from LP part of system [0.7] or be in cold shutdown [0.5] (hot standby in six hours or cold shutdown in 12 hours.)
1 gpm leakage from RCS has been exceeded. [0,8]

REFERENCE

Salem 1 T.S., p 3/4 4-15
Salem 2 T.S., p 3/4 4-17

ANSWER 8.11 (3.00)

a. No report [0.5] does not effect plant ops, was not initiated by plant, and does not effect local travel patterns. [0.25]
b. Unusual event [0.5] due to leak rate [0.25] or ATT. 19 [illegible] DUE TO PLANT VENT.
c. Unusual event [0.5] due to transport of contaminated individual. [0.25]
d. Site Emergency [0.5] LOCA greater than makeup capacity. [0.25]

REFERENCE
Event Classification/Notification/Reporting guide, Sec 1, p 1
Sec 7, p 1
Sec 19, p 1

ANSWER 5.01 (2.00)

a. Valve operation, opening or closing
Pump starting or stopping
Oscillation of auto control valves [any 2 @ 0.4 each]

b. Slowly opening of valves between voided and full systems
Proper venting of components
Adequate level on tanks in systems where the tanks provide supply or surge function
Proper use of steam traps and vents
Proper sequencing of valves in pressurized systems [any 2 @ 0.6 each]
ALLOW TIME FOR THERMAL EQUALIZATION

REFERENCE
Nuclear Energy Training, Thermodynamics, NET 4-2, pg. 2.1-4,5

ANSWER 5.02 (2.50)

T ave decreases since more energy is being removed. [0.7]
Rx Power increases due to the positive reactivity added through MTC [0.5]
~~and doppler. [0.8]~~. DOPPLER WOULD WOULD ADD NEGATIVE REACTIVITY [0.3]
Power stabilizes at a higher value. [0.5]
Tave stabilizes at a lower value. [0.5]

REFERENCE
SNGS Student Notebook Chap 47, Accident Analysis, p 24

ANSWER 5.03 (2.00)

a. Due to decrease in self shielding of the fuel pellet [0.5] as the range of neutron energies in the absorption band increases [0.5].

b. Less negative [0.5] due to increased overlapping of resonance absorption bands [0.5] OR decreased self-shielding [0.5]

REFERENCE
SNGS Student Notebook Chap 46, Trainsient Analysis, p 10, TA-7

ANSWER 5.04 (3.00)

a. Th stable or decreasing [0.3]
   Loss of matural circulation flow. [0.3]
b. 10 F subcooling [0.3]
   Voiding in core or hot leg which would interrupt flow. [0.3]
c. SG pressure tracking ~~Tave~~ Tc saturation pressure. [0.3]
   SG not removing heat. [0.3]
d. SG level in Narrow Range [0.3]
   SG no longer available as heat sink. [0.3]
e. PZR level 50% [0.3]
   Voiding of hot leg which would interrupt flow. [0.3]
~~d. FCT increases [0.3] because the clad heat transfer capacity is reduced.~~
~~(increase temperature due to lower conductivity of crud)[0.4]~~

REFERENCE
General Physics; Heat Transfer Thermodynamics and Fluid Flow Fundamentals, p. 356-357.

ANSWER 5.05 (2.50)

a. Shut down bank withdrawal occurs when the reactor is further away from criticality then when boron dilution occurs. [0.5] Prevents unexpected criticality. [0.5] ~~Doubling the count rate halves the the amount by which reactor is shutdown.[0.5]~~

1 SD BANK MAY SHADOW SR DETECTORS

2) RATE OF REACTIVITY ADDITION IS DIFFERENT

[0.25 each]

b. ECP different from last criticality by greater than the tolerances on the work sheet. [1.0] (>650 PCM DUE TO RODS, >1000 PCM DUE TO XE, 500 PCM PAST ECP WITH NO CRITICALITY)

REFERENCE
Rx Eng Man, Part 1, p 1
Part 4.1, p 1
Part 4.2, p 1

CAF

ANSWER 5.06 (3.00)

a. CURRENT ~~Power~~ decreases [0.5]
The pump is pumping the same volumetric flow rate but the density of the water decreases [0.5] which causes a lower mass flow and less pump CURRENT ~~power~~. [0.5]

b. CURRENT ~~Power~~ decreases. [0.5]
The pump is pumping against a lower discharge head due to the reduction of head losses in the area of the void. [1.0]

REFERENCE
Heat Transfer and Thermodynamics, Part A, Chap 2, Sec III, p 332

ANSWER 5.07 (3.00)

a. EOL - MTC is more negative and thus imparts greater positive reactivity from the drop in coolant temperature. [0.5]
Doppler only Power coefficient (DOPC) is MORE ~~less~~ negative and thus imparts MORE ~~less~~ reactivity from the power increase. [0.5]

b. EOL ~~BOL~~ - ~~MTC is less negative and thus imparts less negative reactivity from the coolant heat-up. [0.5]~~ CLOSER TO DNBR [0.5]
DOPC is more negative and thus imparts more positive reactivity as power drops which tends to hold power up. [0.5]

c. BOL ~~EOL~~ - Beta-eff is less, making SUR greater. [0.5]
DOPC is less negative and thus imparts less negative reactivity after the fuel temperature begins to increase. [0.5]

REFERENCE
SNGS Student Notebook Chap 47, Accident Analysis, p 28, 47, 56

ANSWER 5.08 (3.00)

a. ~~An increased~~ A DECREASED pressure setting ~~raises~~ LOWERS Tave. Assuming a negative MTC, this adds ~~negative~~ POSITIVE reactivity so the ACP is ~~HIGHER.~~ LOWER (1.0)

b. A 4 hour delay adds positive reactivity since Xenon is decaying beyond its ECP value. The ACP is LOWER. (1.0)

c. Rod worth increases with temperature, so the HFP worth is greater than the HZP worth. This implies less positive reactivity has been added so the ACP is HIGHER. (1.0)

REFERENCE
SNGS Student Notebook Chap 46, p 6, 24, 38

ANSWER 5.09 (4.00)

a. Power Defect (Fig. 2)	+1125 pcm		[0.4]
IRW (Fig. 4, 15)	-3200 pcm	(-3480 + 280)	[0.6]
SD rod worth (Fig. 16)	-4470 pcm		[0.4]

	-6545 pcm		[0.1]

b. Differs by the worth of the most reactive rod being withdrawn. [0.6]
From Fig. 17, 1290 pcm. [0.4] OR 1.6 %Δk/k - 6.5% Δk/k = 4.9% Δk/k

c. Xenon worth (Fig. 8) = -4600 + 3000
= -1600 pcm [0.6]

ppm change = Xenon worth + POWER DEFECT / INITIAL Differencial Boron Worth (Fig. 12)
= -1600 pcm / -8.15 pcm/ppm
= ~~196 ppm~~ 59 PPM [0.6]

Gallons of demin. (Fig. 101) ~ ~~12000~~ 3000 gal [0.3]

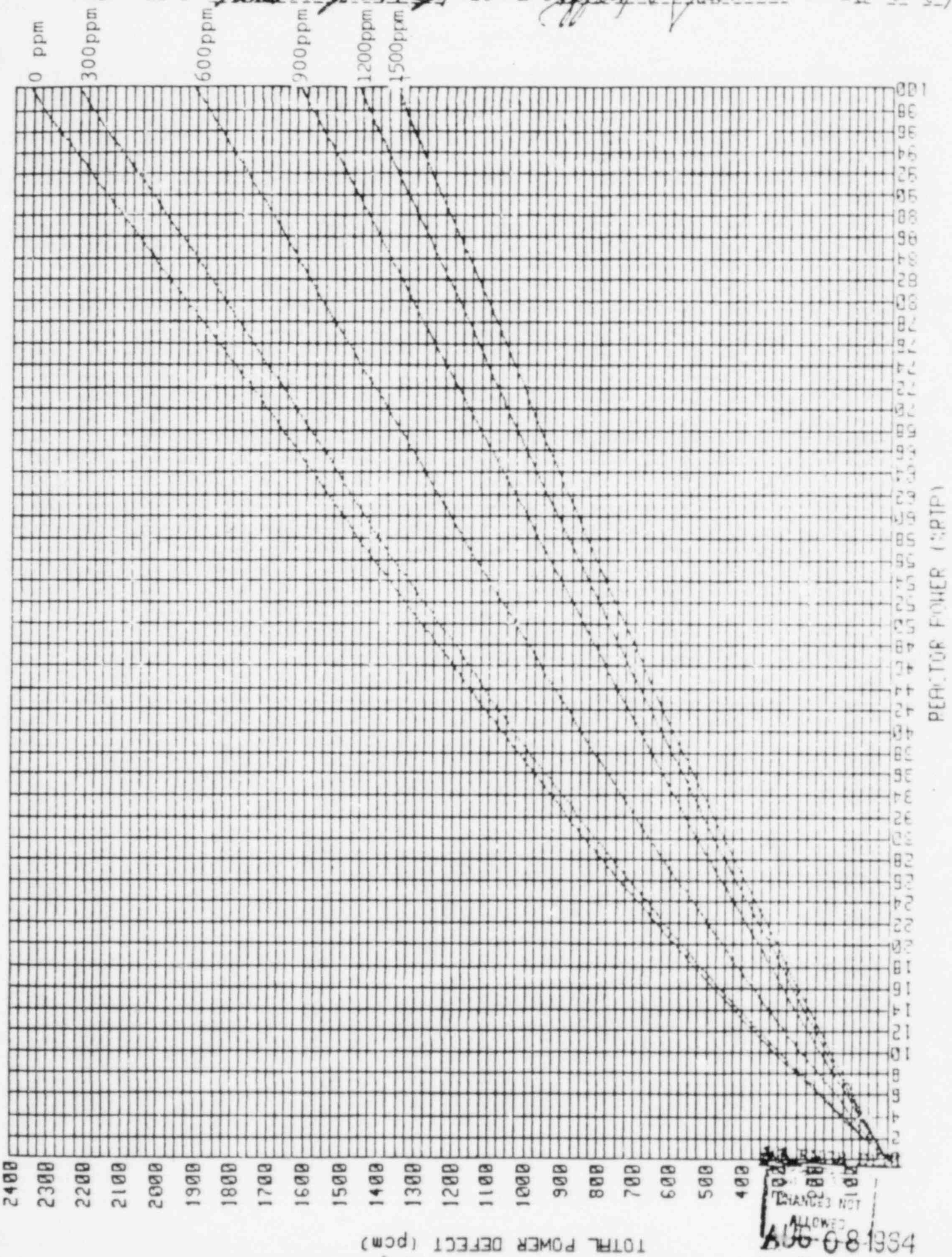
REFERENCE
Rx ENG Man, Part 3, Sec 3.7 (b), p 5, 6

FIGURE 2

SALEM 1 CYCLE 6

POWER DEFECT Vs. REACTOR POWER

PREPARED BY James L. Jones, Jr TECH ENG Jeffrey G. Jubon DATE 8-8-84



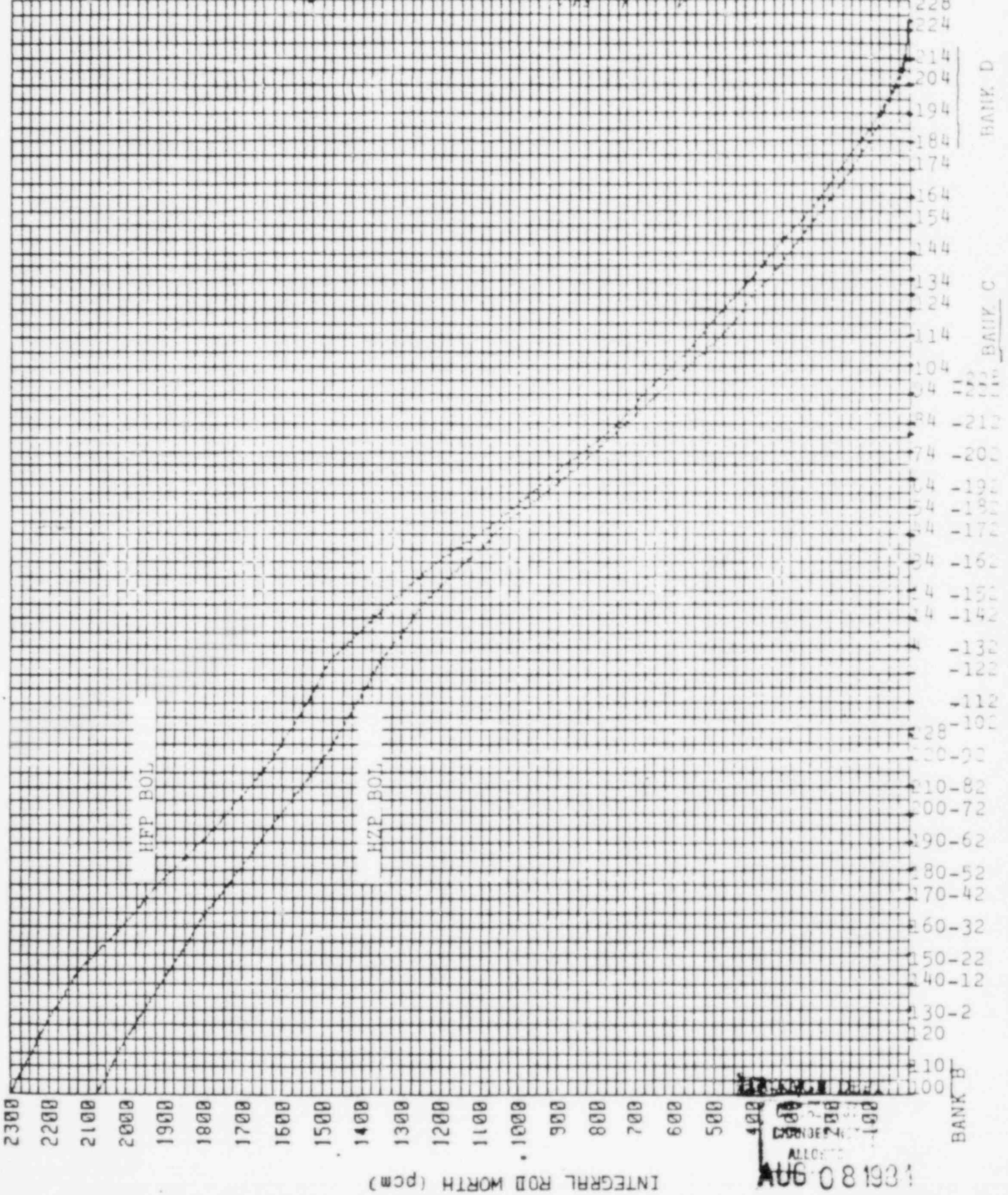
CHANGES NOT ALLOWED

AUG 08 1984

SALEM 1 CYCLE 6 FIGURE 4

INTEGRAL ROD WORTH vs. POSITION IN OVERLAP
FOR HFP & HZP ROD WORTHS

PREPARED BY James L. Jones TECH ENG Jeffrey D. Jackson DATE 8/8/84



SALEM 1 CYCLE 6 FIGURE 6

EQUILIBRIUM XENON REACTIVITY vs REACTOR POWER

Prepared by [signature] TECH Engineer [signature] Date 8/8/84

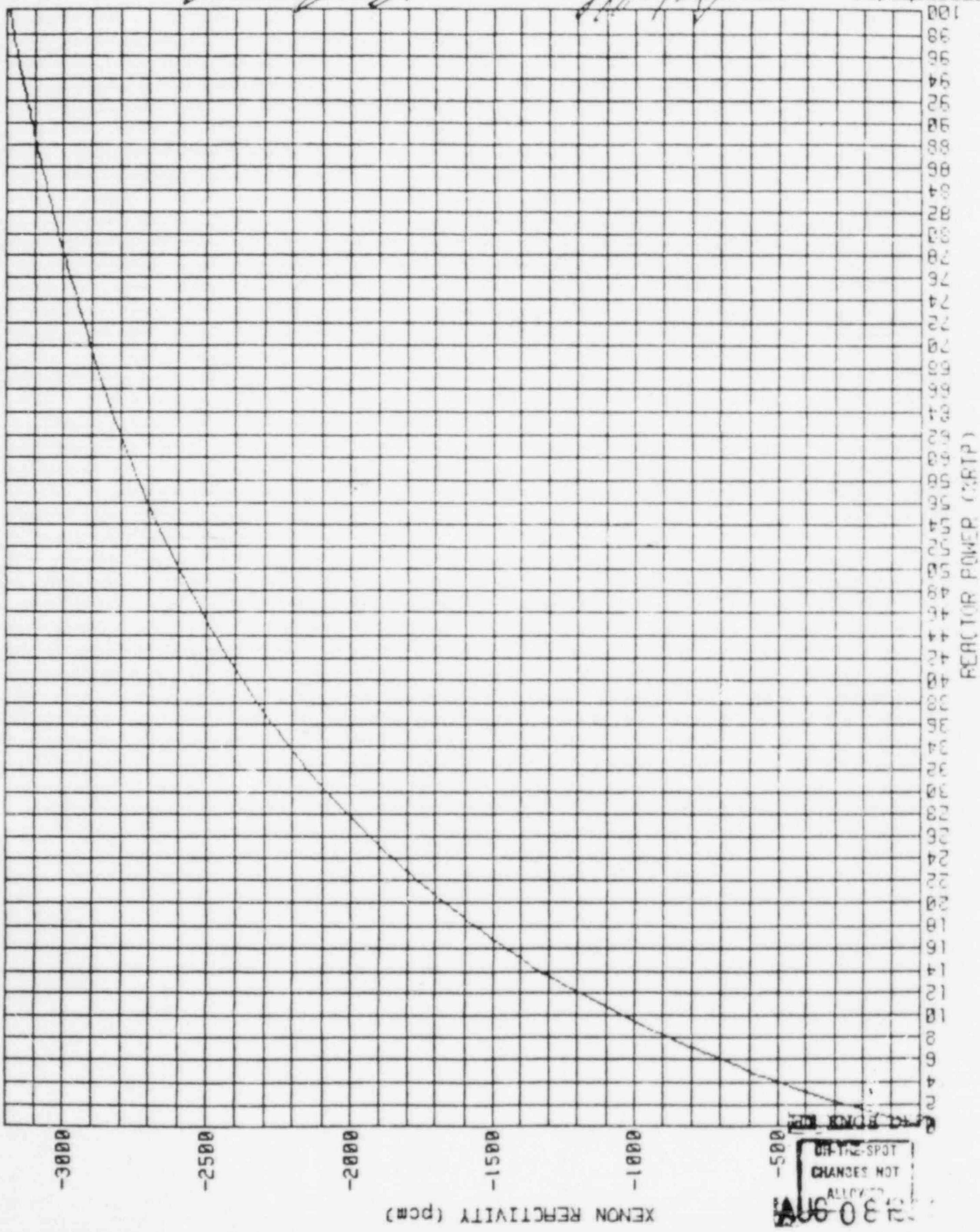


FIGURE 6

SALEM 1 CYCLE 6

XENON WORTH VS TIME AFTER TRIP FROM

EQUILIBRIUM CONDITIONS FOR

25%,50%,75%,100% powers

Prepared by TECH Engineer Date 8/8/84

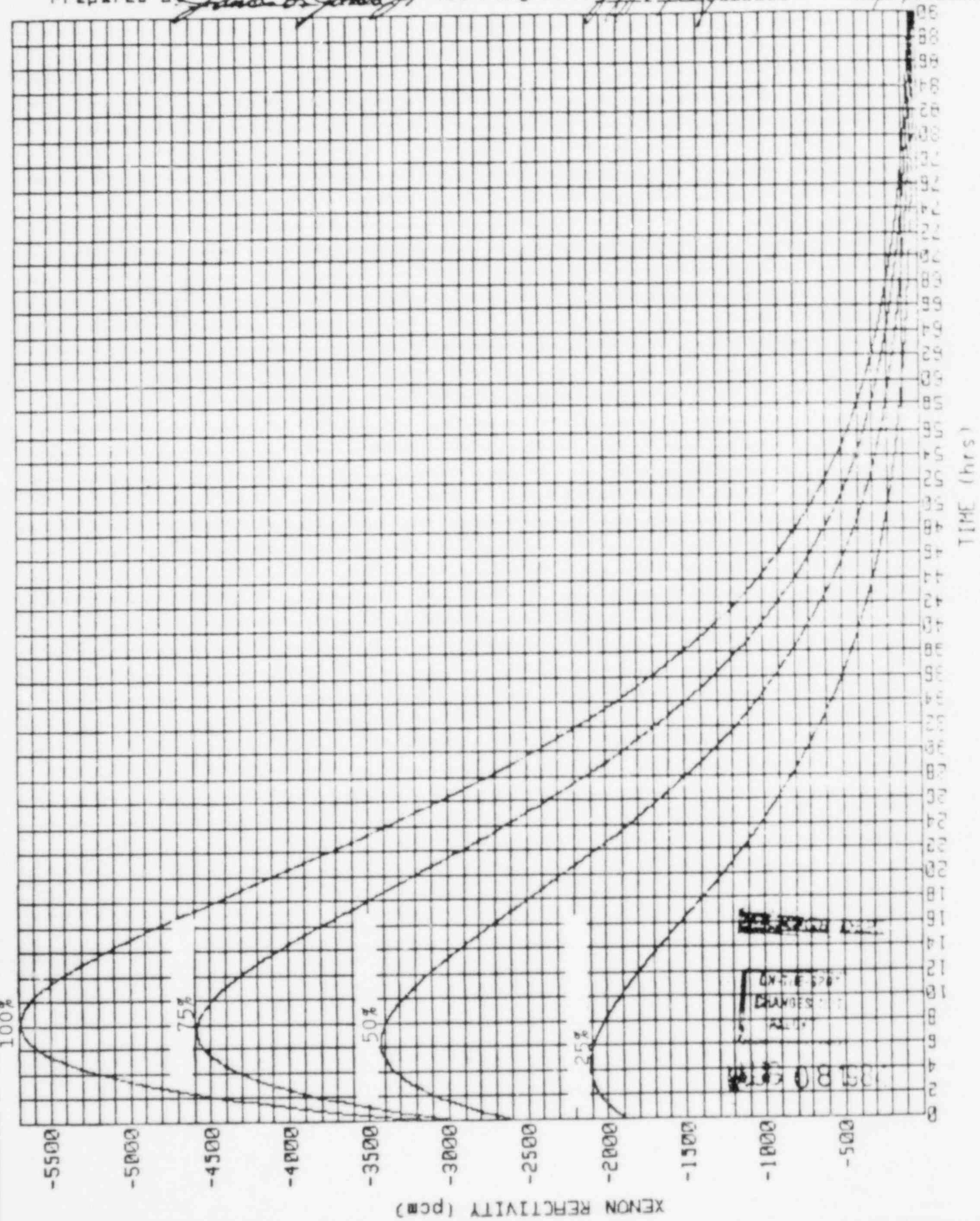


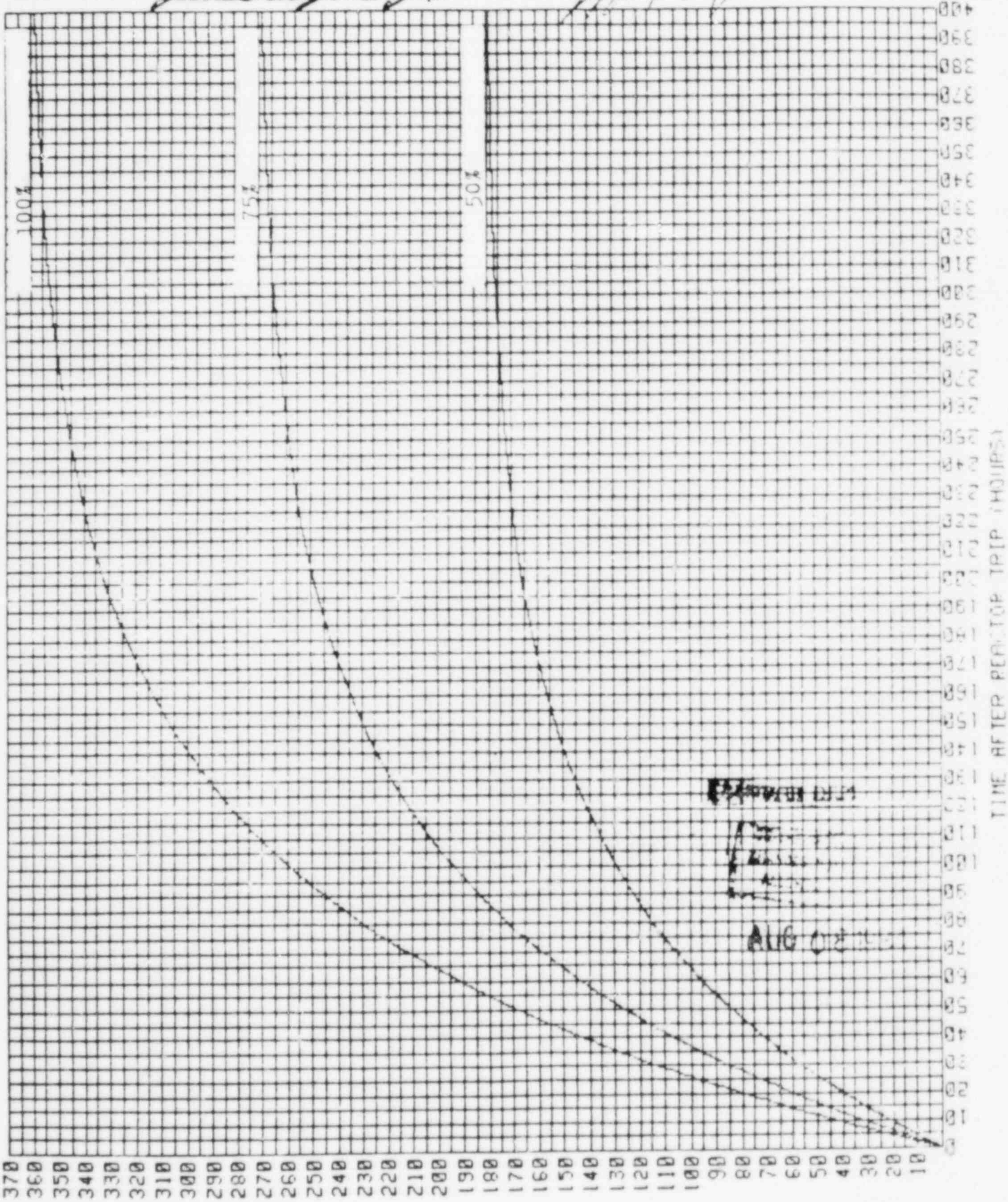
FIGURE 10

SALEM 1 CYCLE 6

SAMARIUM REACTIVITY vs. TIME AFTER SHUTDOWN

BUILD UP FROM EQUILIBRIUM CONDITIONS (Eq Sm=588 pcm)

PREPARED BY James L. Jones Jr. TECH ENG Jeffrey G. Jackson DATE 8/5/84



SALEM 1 CYCLE 6 FIGURE 12

DIFFERENTIAL BORON WORTH vs. BORON CONCENTRATION

REACTOR ENGINEERING MANUAL

PREPARED BY James B. Jones Jr. TECH ENG Jeffrey G. Jacobson DATE 8-8-84

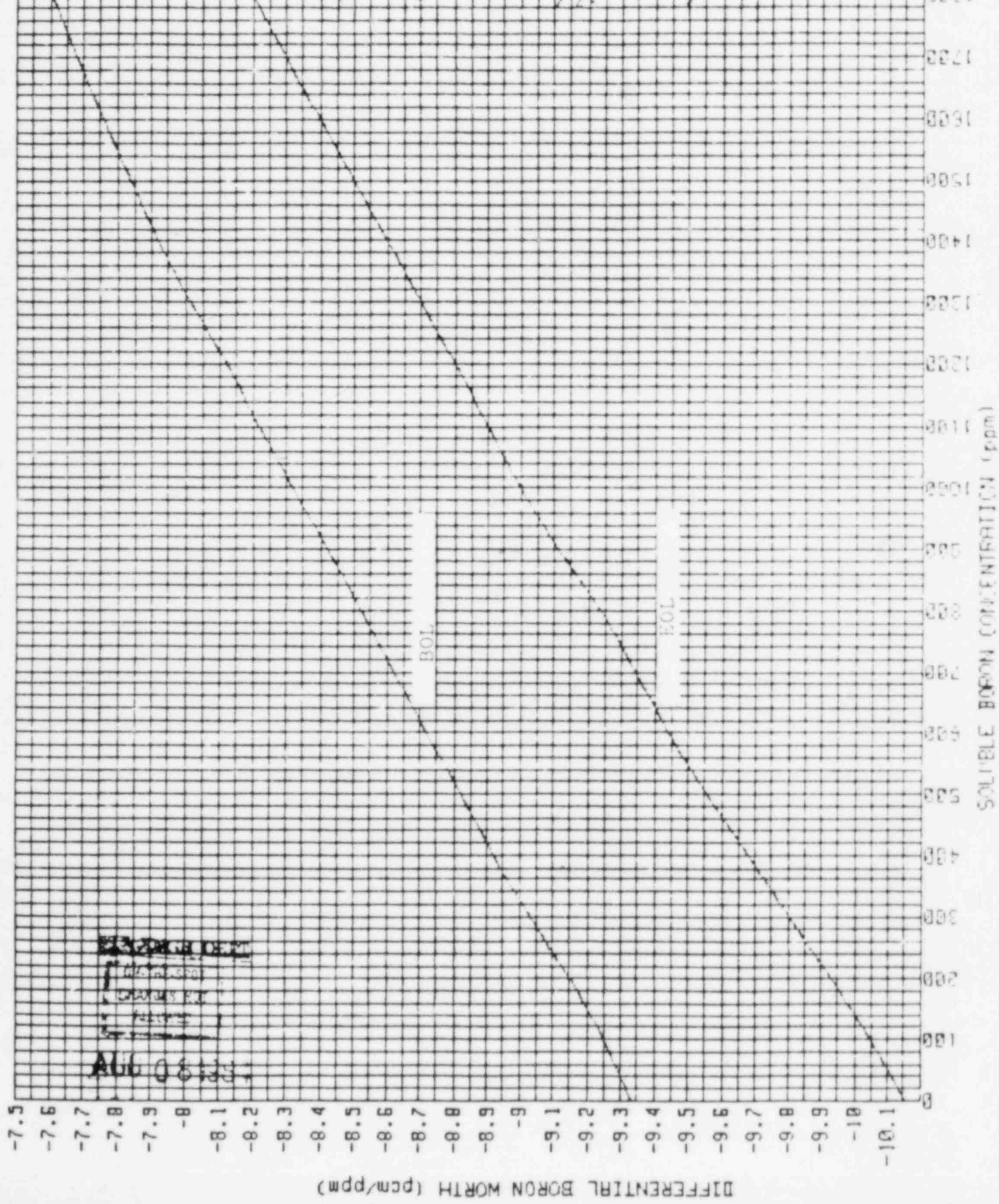


FIGURE 15

SALEM 1 CYCLE 6

CONTROL BANK WORTH vs. CORE EXPOSURE

FOR HZP & CZP CONDITIONS

PREPARED BY TECH ENG DATE 8-8-84

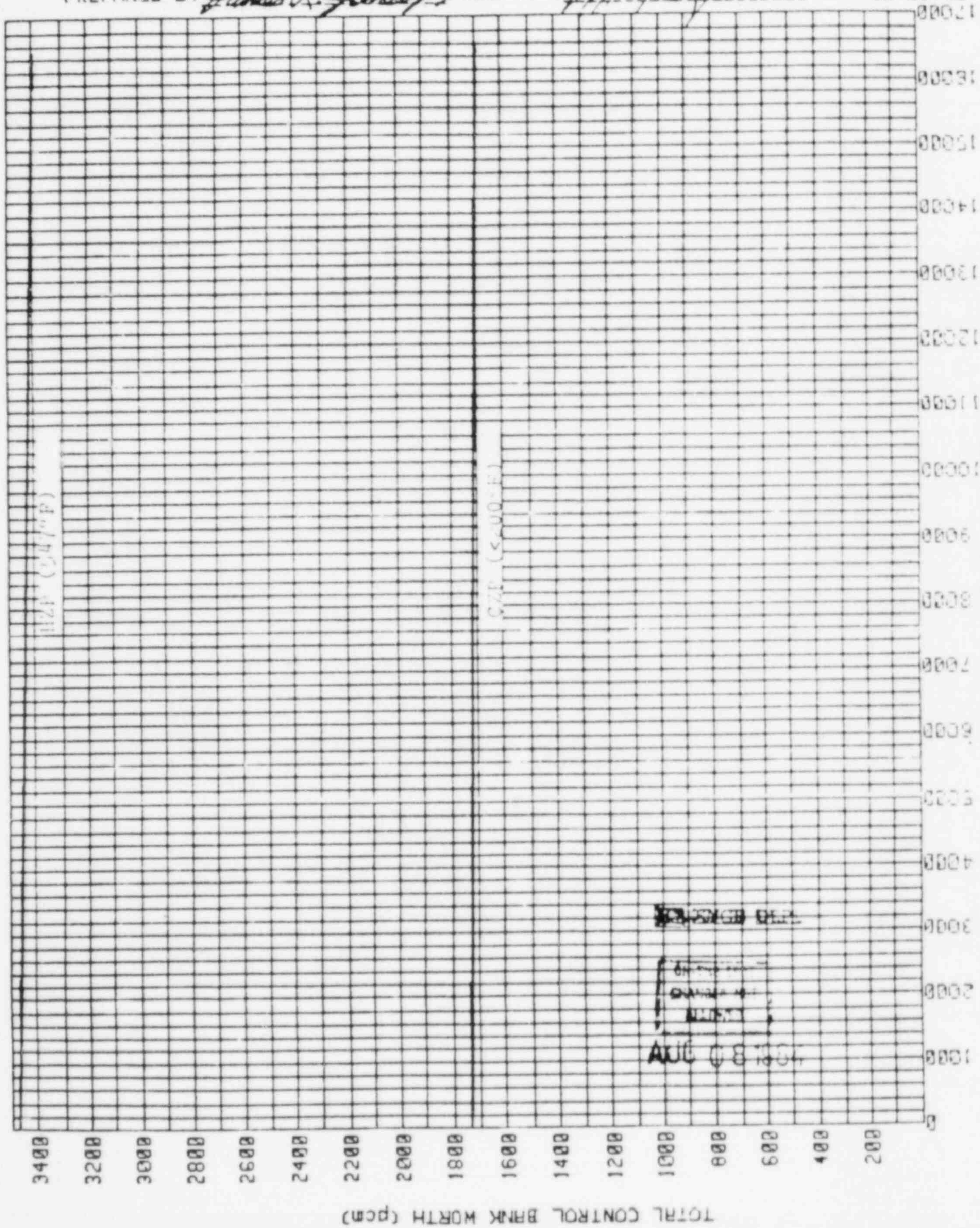


FIGURE 16

SALEM 1 CYCLE 6

TOTAL SHUTDOWN BANK WORTH vs. CORE EXPOSURE FOR HZP & CZP CONDITIONS

PREPARED BY James L. Ronaye Jr. TECH ENG Jeffrey G. Jackson DATE 8-8-84

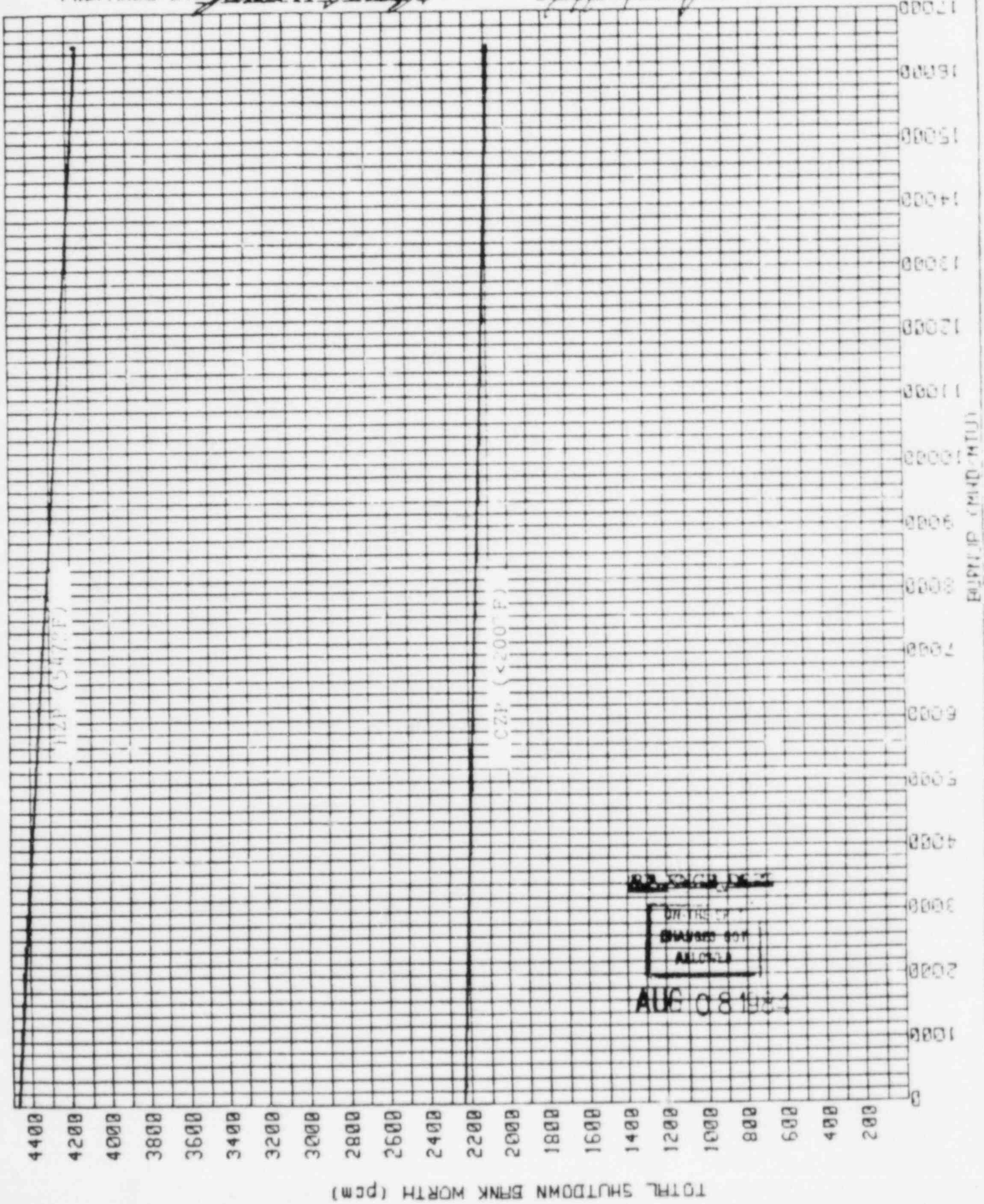


FIGURE 17

SALEM 1 CYCLE 6

MOST REACTIVE STUCK ROD WORTH vs. CORE EXPOSURE FOR HZP & CZP CONDITIONS

PREPARED BY James R. Jones Jr TECH ENG Jeffrey G. Aulson DATE 10/8/84

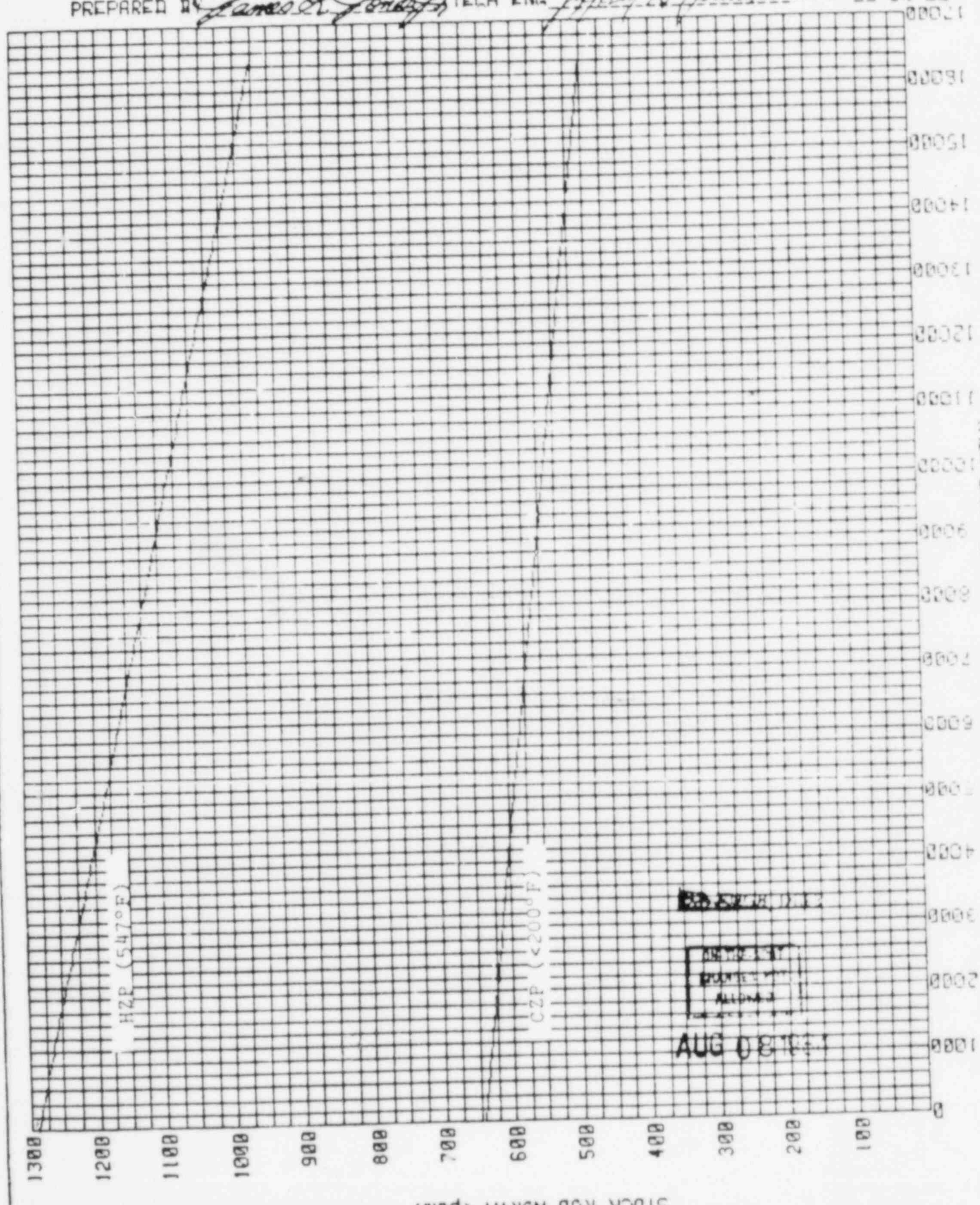


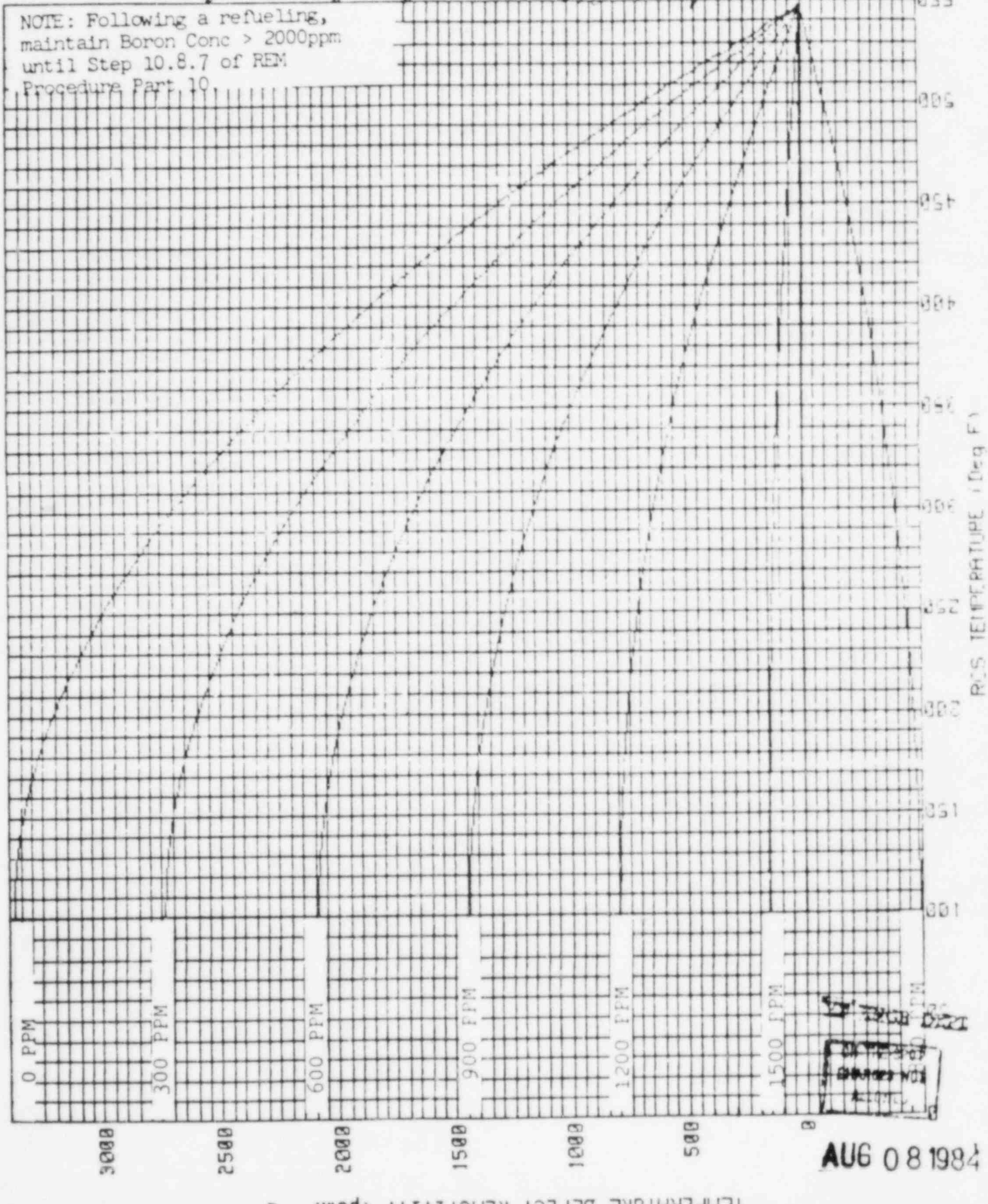
FIGURE 18

SALEM 1 CYCLE 6

ISOTHERMAL TEMPERATURE DEFECT vs. RCS TEMPERATURE FOR HZP CONTITIONS AT VARIOUS BORON CONCENTRATIONS

PREPARED BY James L. Jones Jr. TECH ENG Jeffrey G. Jackson DATE 8/8/84

NOTE: Following a refueling, maintain Boron Conc > 2000ppm until Step 10.8.7 of REM Procedure Part 10



AUG 0 8 1984

SALEM 1 CYCLE 6 FIGURE 20b

HOT STANDBY BORON CONCENTRATIONS

THIS CURVE ASSUMES NO XENON, PEAK SAMARIUM, & RCS TEMP=547 Deg F

PREPARED BY James L. [illegible] TECH ENG Jeff Jackson DATE 8-8-84

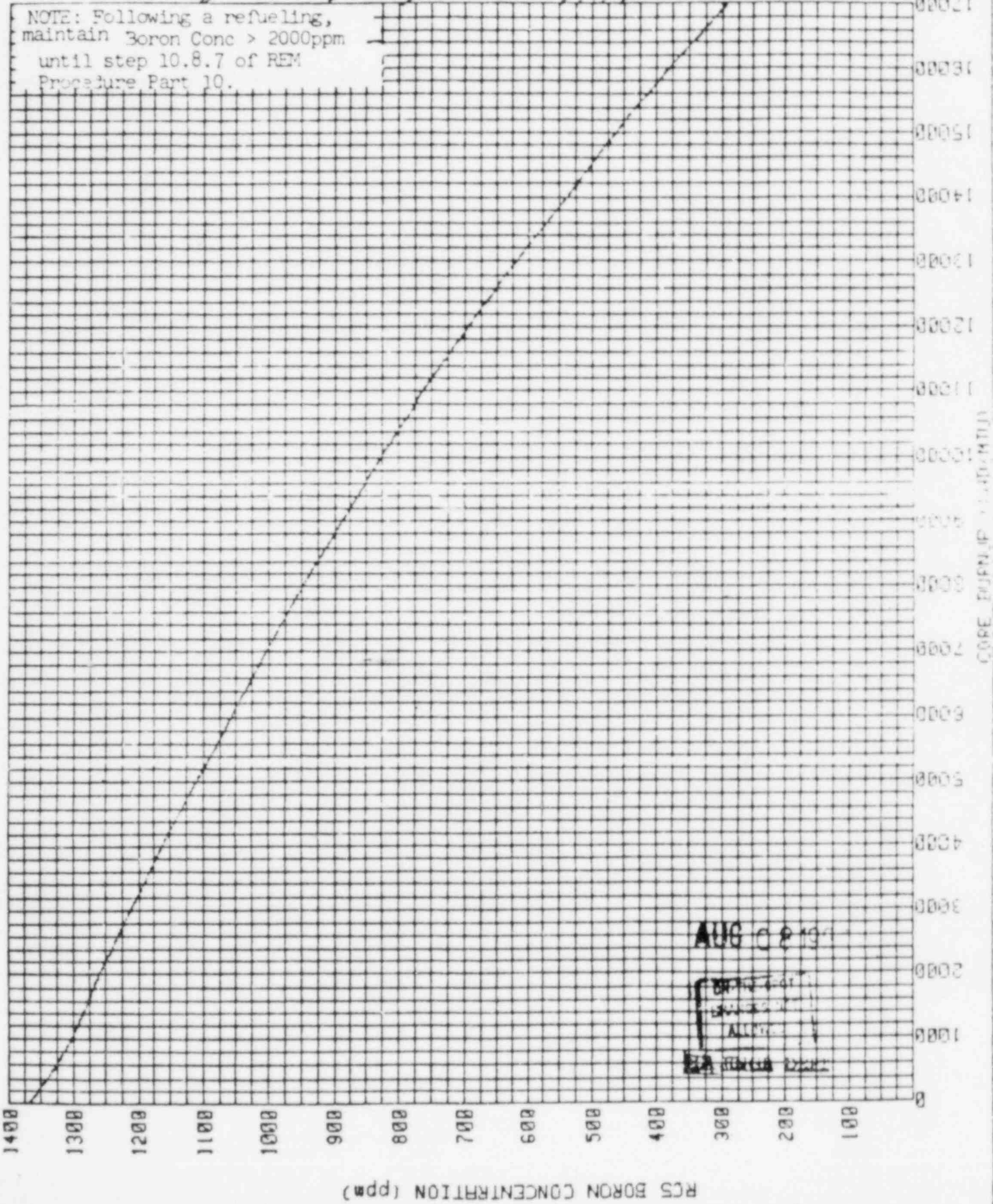
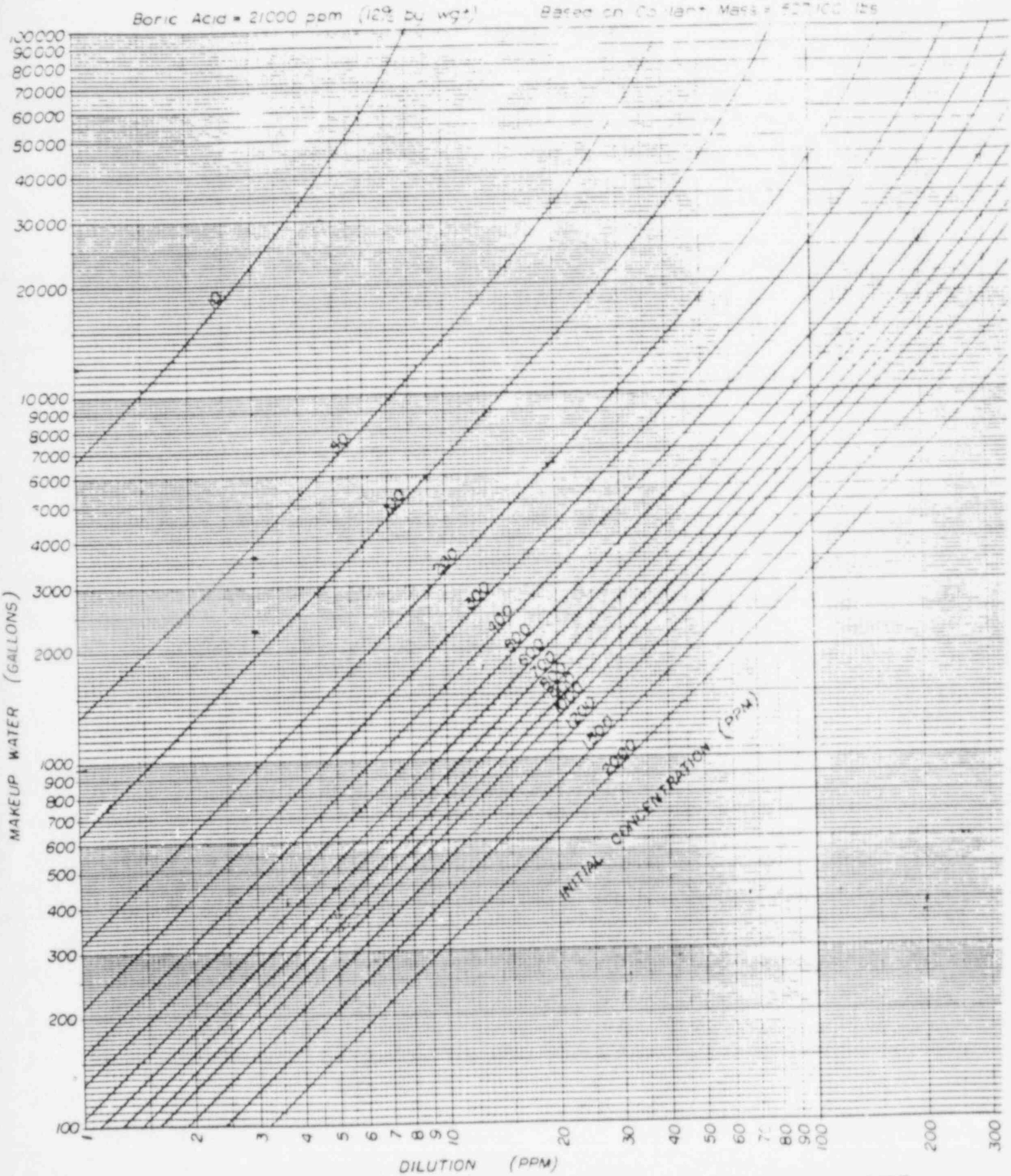
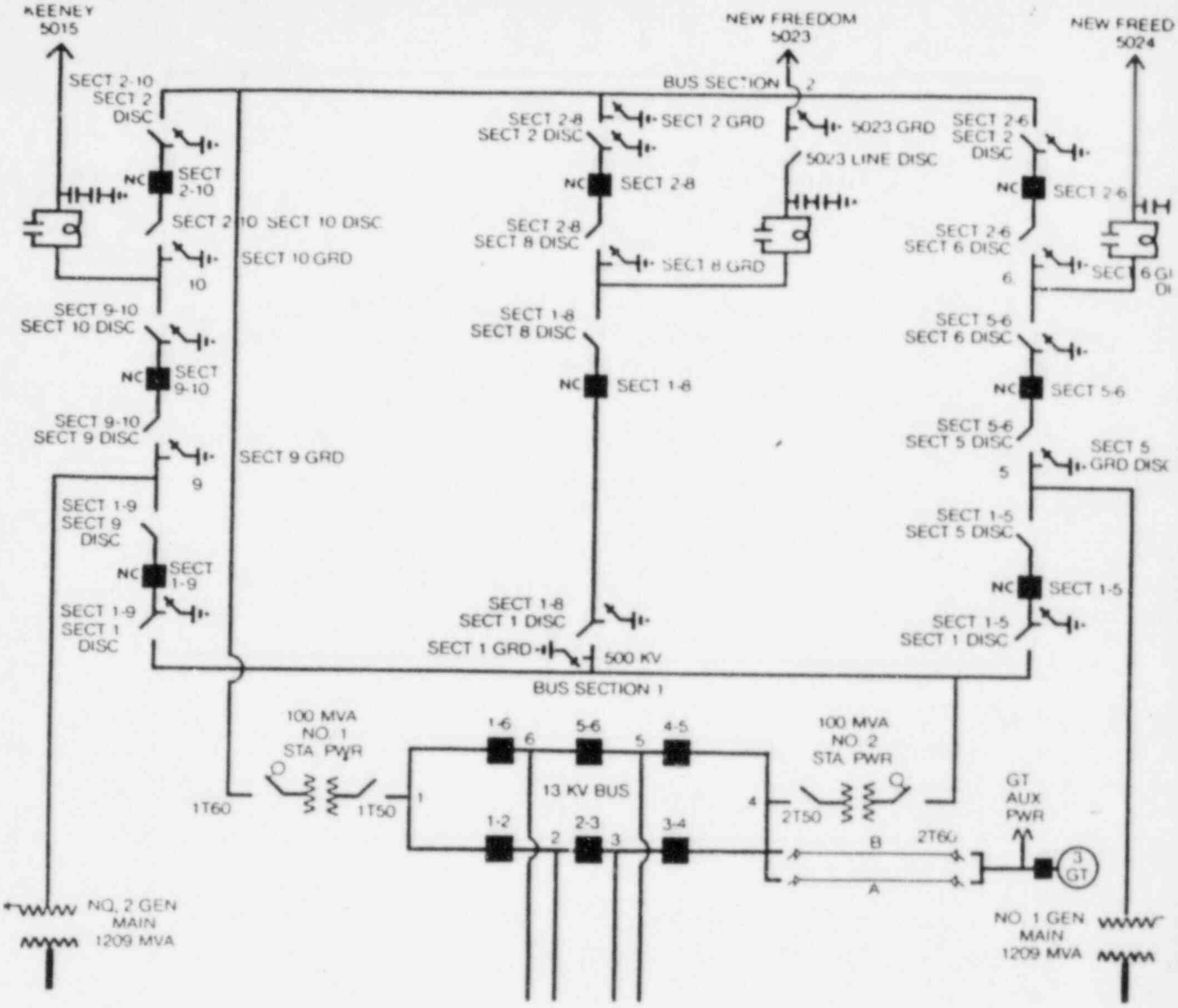


FIGURE 101

BORON DILUTION GRAPH SALEM UNIT



SEP 10 1979 RX ENGR DEPT



LINEUP NUMBER	CIRCUIT BREAKER NUMBERS									
	1-5	5-6 (1)	2-6 (1)	2-8 (2)	1-8 (2)	1-9	9-10 (3)	2-10 (3)	500KV 2T60	500KV 1T60

EVENT CLASSIFICATION SECTION 1

1. Primary Leakage

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Primary leak a) no PRESSURE BOUNDARY leakage b) 1 gpm UNIDENTIFIED leakage c) 10 gpm IDENTIFIED leakage d) 40 gpm CONTROLLED leakage e) RCS PRESS ISOL VALVES (See T/S)	1. Exceeding action statements of T/S LCO 3.4.6.2 (Unit 1) or 3.4.7.2 (Unit 2) See Tech Spec Action Requirements	ATT 1 UE
B. Primary leak > 50 gpm	1. One chg pmp cannot maintain level	ATT 2 A
C. Known LOCA greater than total makeup capacity	1. Low/decreasing PZR level with maximum charging flow	ATT 3 SAE

Refer to Section 5 prior to classification

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
D. PZR safety/PORV failure to reseat	1. PZR press >2200 psig & POPS not armed, or PZR press <375 psig & POPS armed; AND 2. PORV/safety valve tailpipe hi temp, or PRT temp, press, or level increasing	ATT 1 UE
E. Pipe cracks in stagnant borated water systems	1. Cracks in weld areas of safety related piping (as reported by Engineering or ISI/MIET) IE Information Notice - one hour	ATT 6 (50.72b-1Hr)

2. Primary to Secondary Leakage

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Primary to secondary leak a) 1 gpm TOTAL thru all S/Gs b) 500 gpd thru any 1 S/G	1. Exceeding action statements of T/S LCO 3.4.6.2 (Unit 1) or 3.4.7.2 (Unit 2): See Tech Spec Action Statements	ATT 1 UE
B. S/G tube rupture (several hundred gpm)	1. Rx trip/SI on LO PZR PRESS; AND 2. R15 or any R19 alarm; AND 3. PZR press does not recover > S/G press	ATT 2 A
C. Steam break with primary to secondary leakage	1. a) STMLINE ISOL SI on HI STM Flow with LO Tave or LO STM PRESS; or b) SI on STMLINE HI DIFF PRESS; AND 2. R15 or any R19 alarm	ATT 2 A
D. Single S/G tube failure with loss of offsite power	1. RX trip/SI on LO PZR PRESS; 2. R15 or any R19 alarm; AND 3. PZR press recovers > S/G press; AND 4. Offsite power loss	ATT 2 A
E. Steam break with >50 gpm primary to secondary leak and indicated fuel damage	1. a) STMLINE ISOL SI on HI STM FLOW with LO Tave or LO STM PRESS; or b) SI on STMLINE HI DIFF PRESS; AND 2. R15 or any R19 alarm; AND 3. 1R31C (2R31) in HI ALARM (offscale); AND 4. a) RCS analysis shows failed fuel increase >1%/30 min, total >5%, or I-131 equiv >300 uCI/cc OR b) Survey of main steam lines indicates high activity (SSS/EDO juugement) being released to atmosphere OR c) One or more of the following: R46A $\geq$ 0.14 uCI/cc R46B $\geq$ 0.14 uCi/cc R46C $\geq$ 0.14 uCi/cc R47D $\geq$ 0.14 uCi/cc	ATT 3 SAE

2. Primary to Secondary Leakage (con'd)

*** Refer to Section 5 prior to classification***

F. S/G tube rupture (several hundred gpm) with loss of offsite power	1. Rx Trip/SI on LO PZR PRESS; **AND** 2. R15 or any R19 alarm; AND 3. PZR press does not recover > S/G press; AND 4. Offsite power loss	ATT 3 SAE

*** Refer to Section 5 prior to classification***

3. Secondary Leakage

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Secondary line break causing a rapid uncontrolled secondary depressurization	1. HI STM FLOW or abnormal increases in feed flow* to 1 or 2 S/Gs; AND 2. LO S/G press, LO TAVE, or decreasing RCS press	ATT 1 UE
B. S/G safety/PORV failure to reseat	1. Visual/audible indication at vent stacks after S/G press < setpoint; OR 2. Excessive feed or steam flow for affected S/G	ATT 1 UE
C. Steam break with primary to secondary leakage	1. a) STMLINE ISOL/SI on HI STM FLOW with LO TAVE or LO STM PRESS; or b) SI on STMLINE HI DIFF PRESS; AND 2. R15 or any R19 alarm	ATT 2 A
D. Steam break with >50 gpm primary to secondary leakage and indicated fuel damage	1. a) STMLINE ISOL/SI on HI STM FLOW with LO TAVE or LO STM PRESS; AND b) SI on STMLINE HI DIFF PRESS AND 2. R15 or any R19 alarm 3. 1R31C (2R31) in HI ALARM (offscale); AND 4. a) RCS analysis shows failed fuel increase >1%/30 min, total >5%, or I-131 equiv >300 uCI/cc OR b) Survey of main steam lines indicates high activity (SSS/EDO judgement) bein released to atmosphere) OR c) One or more of the following: R46A $\geq$ 0.14 uCi/cc R46B $\geq$ 0.14 uCi/cc R46C $\geq$ 0.14 uCi/cc R46D $\geq$ 0.14 uCi/cc	ATT 3 SAE

*** Refer to Section 5 prior to classification ***

4. Fuel Damage/Degraded Core

Initiating Event Condition	Emergency Action Level	Notification/ Reporting
A. Fuel damage	1. 1R31C (2R31) alarm; **AND** a) RCS sample shows failed fuel increase of > 0.1%/30 min; OR b) RCS sample shows activity > limits of T/S LCO 3.4.8 (U/1) or 3.4.9 (U/2)	ATT 1 UE
B. Severe loss of fuel cladding	1. RCS sample shows I-131 equiv >300 uCi/cc; OR 2. 1R31C (2R31) HI ALARM (off-scale) and RCS sample shows failed fuel increase of > 1%/30 min or total >5%	ATT 2 A
C. Fuel damage accident with release to FHB or containment	1. R5, R9, or R29 and R41B or R41C alarm; OR 2. 2/4 of R2, R7, R10A, or R10B alarm and R21 $\geq$ 1 R/hr; AND 3. Fuel handling problem with possible fuel damage	ATT 2 A

*** Refer to Section 5 prior to classification ***

Initiating Event Condition	Emergency Action Level	Notification/ Reporting
D. RCP seizure leading to fuel failure	1. RCP motor current spike to locked rotor value then zero, or sudden flow decrease in affected loop, or RCP vib alarm and loose parts monitor alarm; **AND** 2. Letdown monitor 1R31C (2R31) alarm; **AND** 3. Analysis shows RCS activity > limits of T/S LCO 3.4.8 (U/1) or 3.4.9 (U/2)	ATT 2 A
E. Major damage to spent fuel in containment or Fuel handling building	1. R5, R9, or R29 alarm; OR 2. 2/4 of R2, R7, R10A, or R10B alarm, and R21 $\geq$ 1 R/hr; **AND** 3. Confirmed fuel damage or loss of water level to below fuel level	ATT 3 SAE

4. Fuel Damage/Degraded Core (con'd)

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
F. Degraded core with possible loss of Coolable geometry	1. ≥5 core exit T/C >1200 F; OR 2. Loss of Main and Aux feed, no S/G WR level, and no S/G press; OR 3. 1R31A (2R31) in HI ALARM (off-scale) and RCS analysis shows failed fuel increase ≥1%/30 min, total >5% or I-313 equiv >300 uCi/cc; OR 4. ≥ 2 WR hot leg RTDs > 700 F, and any R41 alarm, containment sump level > 81'3", or 2/4 containment press >4.0 psig	ATT 3 SAE

Refer to Section 5 prior to classification

5. Fission Product Boundary Failures

The following conditions indicate failure of a fission product boundary. Any two boundary failures, with the possibility of a third, represents a **GENERAL EMERGENCY** condition and Attachment 5 must be implemented immediately. Monitor for these conditions after any single boundary failure.

A. DEGRADED CORE

> **NOTE**
>
> When SSS/EDO judgement indicates probable failed fuel, completion of RCS analysis shou.d not restrict/delay of emergency classification.

1. Letdown monitor 1R31A (2R31) in HI ALARM (offscale) <u>and</u> RCS anaylsis shows failed fuel:
   a) increase >1%/30 min, or
   b) total >5%, or
   c I-131 equiv > 300 uCi/cc

   OR
2. RCS Analysis shows failed fuel:
   a) increase > 1% / 30 min, or
   b) total > 5%, or
   c) I-131 equiv > 300 uCi/cc

   OR
3. Engineering analysis indicates that a degraded core condition exists (as presented to the Emergency Coordinator by Engineering Support Staff).

   OR
4. ≥ 5 core exit T/Cs ≥ 1200 F, or ≥2 WR Thot > 700F and RCS delta T rapidly increasing (Thot increasing), or decreasing to zero (Thot = Tcold)

OR

B. LOSS OF PRIMARY COOLANT

1. PZR level decreasing with maximum charging flow

   OR
2. 2/4 cont press >4.0 psig and accumulator discharge (Modes 1, 2, 3)

   OR
3. Inadequate RCS subcooling (P250 or manual calculation; Modes 1, 2, 3)

   OR
4. R 44 > 20 R/hr and 2/4 of R2, R7, R10A, or R10B alarm

   OR
5. 2/5 CFCU drainage alarms or 2/4 cont press >4.0 psig <u>and</u> Cont sump level >81'3" (no indication of in-cont stm break)

OR

2/3 — ATT 5 GE

C. CONTAINMENT FAILURE

1. Containment H_2 concentration >4%

   OR
2. 2/4 cont press >47 psig and increasing

   OR
3. No cont spray capability with <5 CFCUs available, or 1 cont spray train with <3 CFCUs available and Containment press >23.5 psig and increasing

   OR
4. Steam break, which cannot be isolated, outside containment with indications of primary-to-secondary leakage

   OR
5. Cont penetration isolation valve(s) or cont hatch failure (any breach in containment as indicated by the significant rise in airborne activity in the area of concern).

OR

6 Radiological Releases

Note: Action levels listed are for valid RMS channel indications. The validity of the indication should be confirmed by sample analysis or other means as necessary.

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Accidental, unplanned, or uncontrolled gaseous release, that exceeds 2 times the applicable concentrations of the limits specified in Appendix B, Table II of Part 20, to unrestricted areas (averaged over 60 min)	1. R41C > 1.5E3 cpm for > 60 min; OR 2. R41B > 5.7E3 cpm increase in 60 min	ATT 29 (50.72b - 4Hr)
B. Accidental, unplanned, or uncontrolled liquid release	1. Any unplanned or uncontrolled liquid release outside the controlled access area.	ATT 29 (50.72b - 4Hr)
C. Liquid release that exceeds T/S limits for ≥ 15 min	1. R18 alarm and no isolation **AND** Confirmed analysis of liquid waste effluent indicating discharge exceeding T/S limits. OR 2. Any R19 alarm and blowdown to 12(22) S/G blowdown tank	ATT 7 UE
D. Gaseous release that exceeds T/S limits for > 60 min	1. R41C > 1.1E5 for > 60 min; OR 2. R41B > 2.1E4 cpm increas in 60 min;	ATT 7 UE
E. Gaseous release that exceeds 10 times T/S limits for > 15 min	1. R41C > 1.1E6 cpm for > 15 min; OR 2. R41B > 5.3E4 cpm increase in 15 min; OR 3. R45B > 7.1E-3 uCi/cc for > 15 min	ATT 8 A

F. Dose Rate at Minimum Exclusion Area (MEA) equivalent to 500 mR/hr WB or 2500 mR/hr thyroid for > 2 min (MEA is defined as any monitoring location greater than 0.79 miles away from the affected unit	1.a) Field team measures whole body dose rates greater than 50 mrem/hr for 30 min or greater than 500 mrem/hr for 3 min at the MEA **OR** b) Field team measures, at the MEA, Thyroid dose rates (equiv I-131 concentrations) greater than: 250 mrem/hr (1.0E-7 uCi/cc) for 30 min or 2500 mrem/hr 1.0E-6 uCi/cc for 3 min; **OR** 2. Calculation of 500 mrem/hr dose rate WB or 2500 mrem/hr thyroid dose rate for > 2 min based on EP IV-111, EP IV-113 or EP IV-114 and actual meteorology; **OR** 3. When methods 1 and 2 above are not available based on valid gaseous effluent monitors: R41C > 2.3E6 cpm, OR R41B > 2.0E6 cpm, OR R45B > 0.14 uCi/cc, OR R43 > 7.5 mR/hr, OR One or more of the following: R46A > 0.14 uCi/cc R46B > 0.14 uCi/cc R46C > 0.14 uCi/cc R46D > 0.14 uCi/cc	ATT 9 SAE

G. Dose rate at Minimun Exclusion Area (MEA) equivalent to 1R/hr WB or 5 rad/hr thyroid (MEA is defined for the purpose of field monitoring teams as any distance greater than 0.79 miles away from the affected unit)	1. Field survey team measures at the MEA 1 R/hr W.B. or 5 rad/hr Thyroid (I-131 concentration of 2.0 E-6 uCi/cc); **OR** 2. Calculation of 1R/hr WB or 5 rad/hr thyroid using EP IV-111, EP IV-113 or EP IV-114 and actual meteorology; **OR** 3. When methods 1 and 2 above are not available based on valid gaseous effluent monitors: R41C > 4.6 E6 cpm, OR R41B > 4.0 E6 cpm, OR R45B > 0.28 uCi/cc, OR R45C > 0.28 uCi/cc, OR R43 > 15.0 mR/hr, OR One or more of the following: R46A > 0.28 uCi/cc R46B > 0.28 uCi/cc R46C > 0.28 uCi/cc R46C > 0.28 uCi/cc	ATT 5 GE
H. Loss or theft of licensed material in such quantities and under such circumstances that it appears to the licensee that a substantial hazard may result to persons in unrestricted areas. (Refer to 10CFR 20.402)	1. Judgement of SSS/EDO.	ATT 35 (50.72b-1Hr)
I. Loss, theft or diversion of any special nuclear material	1. Any loss, theft or diversion of any special nuclear material known or suspected (eg. fuel elements or incore detectors see 10CFR 70.52) either in transit to or from this facility or at this facility	ATT 12 UE
J. Abnormal degradation of systems designed to contain radioactive material (not fuel clad, RCS, or cont)	1. Excluding normal valve packing or gasket leakage Example: a) through-wall leak in WHUT	ATT 13 OTHER
K. Radiological sabotage	1. Judgement of SSS/EDO (attempted or substantiated)	ATT 25 UE

7. Radiological expcsure/contamination

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Transport of a contaminated individual from site to offsite medical facility	1. As stated	ATT 14 UE
B. Increase in measured or calculated dose rate (mR/hr) by ≥ 1000 times (indication of severe degradation in control of radioactive materials)	1. Measured or calculated dose rates increased ≥ 1000 times a) On installed or portable monitors; OR b) Unit 1 increase of analog strip chart reading over 20 min; OR c) Unit 2 rad mon computer-trend increase	ATT 8 A
C. Dose rate at Minimum Exclusion Area (MEA) equivalent to 500 mR/hr WB or 2500 mR/hr WB or 2500 mR/hr thyroid for > 2 min (MEA is defined as any monitoring location greater than 0.79 miles away from the affected unit	1. a) Field team measures whole body dose rates greater than 50 mrem/hr for 30 min or greater than 500 mrem/hr OR b) Field team measures, at the MEA, Thyroid dose rates (equiv I-131 concentrations) greater than: 250 mrem/hr (1.0E-7 uCi/cc for 1.0E uCi/cc for 3 min; OR 2. Calculation of 500 mrem/hr dose rate WB or 2500 mrem/hr thyroid dose rate for > 2 min based on EP IV-111, EP IV-113 or EP IV-114 and actual meteorology: OR 3. When methods 1 and 2 above are not available based on valid gaseous effluent monitors: R41C > 2.3E6 cpm, OR R41B > 2.0E6 cpm, OR R45B > 0.14 uCi/cc, OR R43 > 7.5 mR/hr, OR One or more of the following: R46A > 0.14 uCi/cc R46B > 0.14 uCi/cc R46C > 0.14 uCi/cc R46D > 0.14 uCi/cc	ATT 9 SAE

D. Dose rate at Minimum Exclusion AREA (MEA) equivalent to 1R/hr WB or 5 rad/hr thyroid (MEA is defined for the purpose of field monitoring teams an any distance greater than 0.79 miles away from the affected unit)	1. Field survey team measures at the MEA 1 R/hr W.B. or 5 rad/hr Thyroid (I-131 concentration of 2.0 E-6 uCi/cc; OR 2. Calculation of 1R/hr WB or 5 rad/hr Thyroid using EP IV-111, EP IV-113 or EP IV-114 and actual meteorology; OR 3. When methods 1 and 2 above are not available based on valid gaseous effluent monitors: R41C > 4.6 E6 cpm, OR R41B > 4.0 E6 cpm, OR R45B > 0.28 uCi/cc, OR R45C > 0.28 uCi/cc, OR R43 > 15.0 mR/hr One or more of the following: R46A > 0.28 uCi/cc R46B > 0.28 uCi/cc R46C > 0.28 uCi/cc R46D > 0.28 uCi/cc	ATT 5 GE
E. Any incident involving byproduct, source, or special nuclear material causing any of the listed results	1. Receipt by an indivdual of: a) WB exposure ≥ 5 rem, or b) skin exposure ≥ 30 rem, or c) extremity exposure ≥ 75 rem; OR 2. Loss of ≥ one day of operation; OR 3. Damage to property ≥ $2,000	ATT 10 (50.72b-4Hr
F. Any event involving byproduct, source or special nuclear material causing any of the listed results	1. Receipt by an individual of: a) WB exposure ≥ 25 rem, OR b) skin exposure ≥ 150 rem, OR c) extremity exposure ≥ 375 rem; OR 2. Loss of ≥ one working week of operation; OR 3. Damage to property ≥ $200,000	ATT 11 (50.72b-4Hr
G. Any personnel overexposure	1. Reported by HP: a) > 10CFR 20.101 limits b) > 10CFR 20.103 limits c) > 10CFR 20.104 limits	ATT 13 Other

8. Nonradioactive leak/release

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Service water leak in cont. when cont. intergrity is required	1. Fan coil unit or SW piping leak in containment	ATT 6 (50.72b-1HR)
B. Unmonitored discharge of nonradioactive liquid	1. Observation of non-rad waste basin overflowing	ATT 15 Other
C. On-land oil spill	1. Observation of on-site oil spill	ATT 16 Other
D. Oil spill into river	1. Observation of a spill from on-site into Delaware River; OR 2. Observation of oil slick on Delaware River	ATT 16 Other
E. Toxic or flammable gas release that threatens plant personnel	1. Observation of a release threatening plant personnel; OR 2. Warning from offsite of a release that may travel onsite	ATT 7 UE
F. Toxic or flammable gas release entering vital areas	1. Observation/measurement of gases exceeding flammability and/or toxicity limits after entering CR or aux bldg ventilation system	ATT 8 A
G. Entry of uncontrolled topic gases into vital areas where lack of access to the area constitutes a safety problem.	1. Toxic gases entering vital areas affecting safe operation of the plant	ATT 3 SAE
H. Pipe cracks in stagnant borated water systems	1. Cracks in weld areas of safety relating piping (per Engineering or ISI/MIET)	ATT 6 (50.72b-1HR)

9. Fire

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Fire lasting > 10 min that affects plant operations.	1. Observation of a fire lasting > 10 min that causes a mode change or power reduction, or that hampers site personnel in the performance of duties necessary for the safe operation of the plant.	ATT 17 UE
B. Fire potentially compromising the function of one or more safety systems	1. Fire in an area potentially affecting a safety system: a) containment b) control room/protection racks c) relay room d) auxiliary building e) service water intake structure f) penetration areas g) fuel handling building h) switchgear rooms (64' or 84' el); OR 2. Fire that, in the judgement of the SSS/EDO, could affect a safety system	ATT 18 A
C. Fire compromising the function of one or more safety systems	1. Fire in an area that has affected a safety system: a) containment b) control room/protection racks c) relay room d) auxiliary building e) penetration areas g) fuel handling building h) switchgear rooms (64' or 84' el); OR 2. Fire that, in the judgement of the SSS/EDO, has affected a safety system	ATT 19 SAE
D. Any major internal or external events (eg. fires, earthquakes, substantially beyond design basis) which would cause massive common demage to plant systems which result in a General Emergency	1. Any fire exceeding the design abilities of the Fire Protection System and unable to terminate by additional Fir Company support such that in the judgement of the SSS/EDO will result in the Emergency Action Levels of Section 5, Section 6G, Section 7D, Section 15D, Section 17D, Section 17E	ATT 5 GE

10. Earthquake/Severe Weather

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Earthquake/seismic event felt in-plant or detected on station seismic instrumentation	1. OHA B43 (Unit 1 CR) alarms; **AND** 2. Seismic monitor is recording; **AND** 3. Seismic disturbance level:	
	a) $\geq$ 0.02 g	ATT 20 UE
	b) > OBE levels ($\geq$ 0.1 g)	ATT 21 A
	c) > DBE levels ($\geq$ 0.2 g) (call National Earthquake Information Center 303-234-3994 for verification)	ATT 22 SAE
B. Floods	1. Tide level recorder shows:	
	a) $\geq$ 97.5 feet	ATT 7 UE
	b) $\geq$ 99.0 feet	ATT 2 A
	c) $\geq$ 100.5 feet	ATT 3 SAE
C. Low water levels	1. Tide Level recorder shows:	
	a) $\leq$ 83.1 feet	ATT 7 UE
	b) $\leq$ 81.0 feet	ATT 8 A
	c) $\leq$ 78.4 feet	ATT 9 SAE
D. Hurricane or unusual wind conditions	1. As indicated by 33', 150', or 300' elevation wind speed channels on the Meteorological Tower:	
	a) sustained winds $\geq$ 90 mph	ATT 7 UE
	b) sustained winds $\geq$ 95 mph	ATT 8 A
	c) sustained winds $\geq$ 100 mph	ATT 9 SAE
E. Tornado on site	1. Observed tornado funnel cloud within Minimum Exclusion Area (MEA)	ATT 7 UE
F. Tornado striking facility	1. Observed tornado funnel cloud within Security Boundary	ATT 8 A

10. Earthquake/Severe Weather (con'd)

G. Tornado on site that affects safety structures	1. Tornado affecting: a) turbine building b) service building c) auxiliary building d) containment e) service water intake structture f) RWST, PWST, or AFWST g) fuel handling building	ATT 3 SAE
H. Any major internal or external events (e.g., fires, earthquakes, substantially beyond desing basis) which could cause massive common damages to plant systems which would result in a General Emergency	1. Any severe weather conditions causing damage to plant systems that in the judgement of the SSS/EDO will result in the Emergency Action Levels contained in Section 5, Section 6G or Section 7D.	ATT 5 GE

11. Site Hazards (explosions, crashes, etc.)

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Aircraft crash occurring nearsite	1. Crash within Minimum Exclusion Area (MEA)	ATT 7 UE
B. Aircraft crash occurring onsite	1. Crash within Security Boundary	ATT 8 A
C. Aircraft crash affecting plant structures	1. Crash causing damage or fire in: a) turbine building b) service building c) auxiliary building d) containment e) service water intake structure f) RWST, PWST, or AFWST g) fuel handling building	ATT 19 SAE
D. Turbine rotating component failure	1. Turbine trip	ATT 7 UE
E. Turbine rotating component failure causing casing penetration	1. Turbine trip **AND** 2. Observation of penetrations through outer casing	ATT 2 A
F. Missile impact onsite	1. Missile impact, from any source, causing severe structural damage to any building within the Security Boundary	ATT 8 A
G. Missle impact onsite damaging a vital structure	1. Missile impact, from any source, causing structural damage to: a) turbine building b) service building c) auxiliary building d) containment e) service water intake structure f) RWST, PWST, or AFWST g) fuel handling building	ATT 9 SAE
H. Unplanned explosion affecting plant operations	1. Explosion/combustion or its consequences within the security boundary that causes a power reduction or a mode change; or that hampers site personnel in the performance of duties necessary for the safe operation of the plant	ATT 17 UE

I. Unplanned explosion potentially compromising the function of one or more safety systems or normal operation of the plant	1. Explosion/combustion in one or more of the following areas: a) containment b) control room/protection racks c) relay room d) auxiliary building e) service water intake structure f) penetration areas g) fuel handling building h) switchgear rooms (64' or 84' el that could have, in the judgement of the SSS/EDO, affected a safety system or affecting normal plant operation	ATT 18 A
J. Unplanned explosion compromising the function of one or more safety systems	1. Explosion/combustion in one or more of the following areas: a) containment b) control room/protection racks c) relay room d) auxiliary building e) service water intake structure f) penetration areas g) fuel handling building h) switchgear rooms (64' or 84' el) that has, in the judgement of the SSS/EDO, affected a safety system	ATT 19 SAE
K. Any major internal or external events (e.g., fires, earthquakes, substantially beyond design basis) which would could result in a General Emergency.	1. Any event such as an aircraft crash, explosion or missile impact causing damage to plant systems that in the judgement of the SSS/EDO will result in the General Emergency Levels contained in Section 5, Section 6G, Section 7D, Section 15D, Section 17D, Section 17E	ATT 5 GE

12. Personnel Emergencies

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Transport of a contaminated individual from site to offsite medical facility	1. As stated	ATT 14 UE
B. Any serious injury occurring onsite	1. As judged by SSS/EDO	ATT 24 Other
C. Any fatality occurring onsite	1. As stated	ATT 23 (50.72b - 4Hr)
D. Any personnel overexposure	1. Reported by HP: a) > 10 CFR 20.101 limits b) > 10 CFR 20.103 limits c) > 10 CFR 20.104 limits	ATT 13 Other
E. Any significant personnel over-exposure	1. Any individual receives: a) WB exposure $\geq$ 5 rem b) skin exposure $\geq$ 30 rem c) extremity exposure $\geq$ 75 rem	ATT 10 (50.72b - 4Hr)
F. Any serious personnel overexposure	1. Any individual receives: a) WB exposure $\geq$ 25 rem b) skin exposure $\geq$ 150 rem c) extremity exposure $\geq$ 375 rem	ATT 11 (50.72b - 4Hr)

13. Security Events

Reference listings of
Security Contingency Procedure

Implementation of appropriate Security Contingency Procedure (SCP) - (see below).

SCP 1	Loss or Degradation of Physical Security System	
SCP 2	Loss of Security Computer Power..	
SCP 3	Loss or Degradation of Communcation Systems	
*SCP 4	Loss or Degradation of Security Force	
*SCP 5	Threat Against the Station	
*SCP 6	Discovery of Intruders or Attack	
SCP 7	Internal Disturbance	
*SCP 8	Hostage Situation	
*SCP 9	Fire, Explosion or Other Catastrophe	
*SCP 10	Discovery of Sabotage Devices or Evidence of Sabotage	
SCP 11	Civil Disturbance	
*SCP 12	Security Alert	
*SCP 13	Tamper Alarm Annunciation .... (if actual tampering has occured)	

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Security Alert	SCP 12 - Only	ATT 25 UE
B. Substantiated threat, attempted entry or attempted sabotage:	SCP 1 And SCP 10, or SCP 2 And SCP 10, or SCP 3 And SCP 10, or SCP 4 - Only, or SCP 5 - Only, or SCP 7 - Only, or SCP 8 - Only (Hostage Held Offsite), or SCP 11 - Only, or SCP 11 And SCP 5	ATT 25 UE
C. Substantiated threat, attemped entry or attempted sabotage with Security Alert declared.	SCP 3 And SCP 12, or SCP 8 And SCP 12, or SCP 9 And SCP 12, or SCP 11 And SCP 12	ATT 26 A

* Notify NRC

13. Security Events

D. Ongoing security compromise	SCP 6 - Only (Intruder unarmed and non-violent) - Only, or SCP 5 - And SCP 10, or SCP 8 - Only (Hostage Held onsite/non-vital area), or SCP 9 - And SCP 10 (Device, Fire in non-vital area) or SCP 10- Only (Device in Vital Area)	ATT 26 A
E. Ongoing security compromise involving imminent loss of physical control of the plant	SCP 6 - Only (Intruder Armed or Violent in non-vital Area) SCP 9 - And SCP 10 (Device or Fire in vital area) SCP 10- (Device in Vital Area)	ATT 27 SAE
F. Ongoing security compromise resulting in the loss of physical control of the plant	SCP 6 - Only (Intruder Armed) or Violent in Following Vital Areas: 1. Control Room 2. Hot Shut down Panel SCP 9 - And SCP 10 (Device or fire with loss of vital equipment) in vital following vital areas: 1. Control Room 2. Hot shut down panel	ATT 28 GE
G. Loss, theft or diversion of any special nuclear material	1. Any loss, theft or diversion of any special nuclear material known or suspected (eg. fuel elements or incore detectors see 10CFR 70.52) either in transit to or from this facility or at this facility	ATT 12 UE

14. Technical Specification Items

Initiating Event/ Condition	Emegency Action Level	Notification/ Reporting
A. Any event or condition requiring unit shutdown to comply with any T/S LCO	1. Shutdown initiated	ATT 1 UE
B. Primary leak a) no PRESSURE BOUNDARY leakage b) 1 gpm UNIDENTIFIED leakage c) 10 gpm IDENTIFIED leakage d) 40 gpm CONTROLLED leakage e) RCS PRESS ISOL VALVES (See T/S)	1. Exceeding action statements of T/S LCO 3.4.6.2 (Unit 1) or 3.4.7.2 (Unit 2) See Tech Spec Action Requirements	ATT 1 UE
C. Complete loss of any function needed for cold shutdown	1. Entering Tech Spec 3.5.3 Action a or Action b	ATT 2 A
D. Exceeding any T/S safety limit	1. Exceeding T/S LCO 2.1.1 or 2.1.2	ATT 30 (50.72b - 1HR)
E. Failure of any Rx trip system to initiate or complete protective function(s) as required	Examples: a) PZR press > 2385 psig without automatic Rx trip b) inability to insert sufficient control rods to achieve required SDM	ATT 2 A
F. Operation less conservative than a T/S LCO or prohibited by T/S	Examples: a) Shutdown required by an A/S not begun within specified time b) entering T/S 3.0.3	ATT 13 Other
G. Any reactivity anomaly	1. Disagreement with predicted steady state reactivity balance during power ops $\geq$ 1000 pcm (per Rx Engr); OR 2. Calculated SDM < required; OR 3. SUR $\geq$ 5.2 dpm; OR 4. Unplanned positive reactivity insertion > 500 pcm	ATT 13 Other
H. Any unplanned criticality	1. As stated	ATT 12 UE

I. Any event or condition that alone <u>could have</u> prevented the function of safety structures or systems, such as: 1. Personnel error/ procedure violation 2. Equipment failure 3. Design, analysis, construction, or procedural deficiency 4. Loss of a single-train system 5. Functionally redundant components could fail by the same machanism 6. Loss/degradation of a service or input system necessary for reliable or long-term operation of a safety system	Examples: a) clogged fuel line resulting in no fuel to emergency D/GS; b) multiple instrument drift resulting in loss of protection function; c) failure to restore a safety system to operability following maintenance or testing; d) improper procedure allowing incorrect valve lineup, resulting in functional loss of redundant ECCS subsystems	ATT 29 (50.72b - 4Hr
J. Abnormal degradation of fuel cladding, RCS pressure boundary, or containment function or integrity during operation	1. Excluding valve packing or gasket leakage within T/S limits Examples: a) thru-wall failure of RCS piping/pressure boundary components; b) welding/material defects > allowed by applicable codes (per ISI/MIET); c) containment leak rates (e.g. Type B or Type C) > authorized limits; d) loss of containment .sol valve function; e) loss of containment cooling capability f) loss of RCS relief and/or safety valve operability	ATT 2 A
K. Natural or man-made events that require plant shutdown, safety system operation, or other protective actions IAW T/S	Examples: a) threatened civil disturbance requiring shutdown; b) damage caused by fire, tornado, earthquake, etc.; c) entering T/S A/S 3.7.5.1a (U1), 3.7.5a (U2) for flooding (if shutdown required, see 14A also 10B)	ATT 6 (50.72b - 1
L. Errors discovered in transient or accident analyses	1. Per Engineering notification	ATT 29 (50.72b - 4

M. Performance of any system or component requiring corrective action to prevent operation less conservative than assumed in the UFSAR	Examples: a) SI pumps fail to deliver flow rate assumed in UFSAR; b) safety related breaker fails to trip on instantaneous overcurrent; c) RCS thermal shock from inadvertent SI (per Engr)	ATT 13 Other
N. RPS or ESF instrument setpoints less conservative than required by T/S but not preventing the fulfillment of the functional requirements of the affected system	1. Excluding surveillance testing or preventative maintenance Examples: a) 1/4 cont press channels fail to actuate cont spray during surveillance test b) 1/4 PZR press channels cause Rx trip at 1850 psig instead of 1865 psig during surveillance test	ATT 13 Other
O. Conditions leading to operation in a degraded mode permitted by an LCO	1. Excluding surveillance testing or planned maintenance Examples: a) SI pump failed to start following system initiation, redundant pump tested SAT; b) CCP inoperable due to a faulty bearing, redundant pump tested SAT	ATT 13 Other
P. Inadequately implemented procedural or administrative controls which threaten to reduce ESF or RPS redundancy	Examples: a) D/G tripped on hi temp due to incorrect SW lineups to cooler, other SAT; b) failure to perform surveillance testing at required frequency	ATT 13 Other
Q. Abnormal degradation of systems designed to contain radioactive material (not fuel clad, RCS, or cont)	1. Excluding normal valve packing or gasket leakage Example: a) through-wall leak in WHUT	ATT 13 Other

R. Manual or automatic ECCS actuation with discharge to the vessel	1. ECCS actuation indicated: a) ECCS trip setpoint reached or manual initiation; or b) Lit logic lights for any single or conicidence initiation RP4; **AND** 2. Discharge to vessel is verified by control console indication (flow, valve positions, tank levels, etc.)	ATT 1 UE
S. Inoperable snubbers	1. As reported by ISI/MIET.	ATT 34 Other
T. Special Report required	Examples:	
	a) Positive MTC (T.S. 3.1.1.4 U1, 3.1.1.3 U2)	ATT 15 Other
	b) Seismic monitoring instrumentation inoperable > 30 days (U1 T.S. 3.3.3.3)	ATT 15 Other
	c) Meteorological monitoring instrumentation inoperable > 7 days (U1 T.S. 3.3.3.4)	ATT 15 Other
	d) Fire detection instrumentation inoperable > 14 days (T.S. 3.3.3.6)	ATT 15 Other
	e) RCS activity exceeds limits (T.S. 3.4.8 U1, 3.4.9 U2)	ATT 6 (50.72b - 1Hr)
	f) POPS or RCS vent(s) used to mitigate RCS pressure transient (T.S. 3.4.9.3 U1, 3.4.10.3 U2)	ATT 15 Other
	g) Abnormal degradation of containment structure (T.S. 3.6.1.6)	ATT 6 (50.72b - 1Hr)
	h) **ONE** fire suppression water system inoperable > 7 days (T.S. 3.7.10.1)	ATT 15 Other
	i) **BOTH** fire suppression water systems inoperable (T.S. 3.7.10.1)	ATT 6 (50.72b-1Hr)
	j) Spray and/or sprinkler systems inoperable > 14 days (T.S. 3.7.10.2)	ATT 15 Other
	k) Low pressure CO2 system inoperable > 14 days (T.S. 3.7.10.3)	ATT 15 Other
	l) Fire hose stations inoperable > 14 days (T.S. 3.7.10.4)	ATT 15 Other
	m) Fire barrier penetrations inoperable > 7 days (T.S. 3.7.11)	ATT 15 Other

15. Electrical/Power Failures

Initiating Event/ Conditon	Emergency Action Level	Notification/ Reporting
A. Loss of offsite power/ loss of onsite AC power capability	1. Shutdown IAW T/S A/S 3.8.1.1a, 3.8.1.1b 3.8.1.1c, or 3.8.1.1d; **OR** 2. Loss of 500 kV, 13 kV, and 4 kV group buses; OR 3. Rx trip on loss of 4 kV group buses; OR 4. Loss of 4 kV vital buses with inability to energize from emergency D/Gs	ATT 1 UE
B. Loss of offsite power/ loss of all onsite AC power	1. Loss of 500 kV, 13 kV, and 4 kV group buses; **AND** 2. Loss of 4 kV vital buses with inability to energize from emergency D/Gs	ATT 2 A
C. Loss of offsite power/ loss of all onsite AC power for > 15 min	1. Loss of 500 kV, 13 kV, and 4 kV group buses; **AND** 2. Loss of 4 kV vital buses with inability to energize from emergency D/Gs for > 15 min	ATT 3 SAE
D. Loss of offsite and onsite power with total loss of aux feed capability for several hours	1. Loss of 500 kV, 13 kV, and 4 kV group buses; **AND** 2. Loss of 4 kV vital buses with inability to energize from emergency D/Gs for 2 hrs; **AND** 3. Aux feed shows no flow for > 2 hrs	ATT 5 GE
E. Loss of all onsite DC power	1. Loss of all 125VDC buses; AND 2. Loss of all 28VDC buses	ATT 2 A
F. Loss of all onsite DC power for > 15 min	1. Loss of all 125VDC buses; AND 2. Loss of all 28VDC buses; AND 3. DC power cannot be restored for > 15 min	ATT 3 SAE

16. Loss of Annunciators/Control Room Evacuation

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Loss of all OHAs for > 15 min due to an unknown cause	1. As stated	ATT 8 A
B. Loss of all OHAs for > 1 hr and plant transient initiated or in progress	1. As stated	ATT 9 SAE
C. Evacuation of Control Room anticipated or required and control of S/D systems established locally	1. As stated	ATT 8 A
D. Evacuation of Control Room required and control of S/D systems not established locally within 15 min	1. As stated	ATT 9 SAE

17. Loss or Failure of Engineered Safeguards

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Any problem with the Rx trip breakers	1. As judged by SSS/EDO Example: a) bkr does not actuate as demanded	ATT 31 (50.72b - 1Hr
B. Failure of the RPS to initiate and complete a trip which brings the Rx subritical	1. Receipt of Rx protection logic input on RP4; AND 2. Not all rod bottom lights lit or NIs indicate Rx not subcritical	ATT 2 A
C. Failure of the RPS to automatically, or by operator action to manually, initiate and complete a trip which brings the Rx subcritical	1. Receipt of Rx protection logic input on RP4; AND 2. Not all rod bottom lights lit or NIs indicate Rx not subcritical; AND 3. No boration capabilities	ATT 3 SAE
D. Transient requiring operation of shutdown systems with failure to scram, resulting in core damage or additional failure of core cooling and makeup systems	1. Rx remains critical/returns to critical after trip; AND 2. No flow shown on SI/RHR systems or pumps not running with SI initiated	ATT 5 GE
E. Transient initiated by loss of feed and cond. followed by failure of aux feed for extended period; core damage possible in several hours	1. Rx trip on lo S/G feed flow; AND 2. WR S/G levels decreasing towards offscale lo on all S/Gs; AND 3. No aux feed flow shown or pumps not running 2 min after required; AND 4. Aux feed cannot be restored within 30 min	ATT 5 GE
F. Complete loss of any function needed for cold shutdown	1. RHR system fails to attain/maintain primary system <200F; OR 2. IAW T/S A/S 3.5.3a or 3.5.3b	ATT 2 A
G. Complete loss of any function needed for hot shutdown	1. Loss of main and aux feed; OR 2. Loss of steam dumps and all S/G Safeties and PORVs	ATT 3 SAE

18. Operational Status Changes

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Any event or condition during operation that results in the condition of the plant, including principal safety barriers, being seriously degraded	1. As judged by the SSS/EDO	ATT 2 A
B. Any event or condition during operation that results in the plant being: 1) in an unanalyzed condition that significantly compromises plant safety; 2) in a condition outside the design basis; 3) in a condition not covered by operating or emergency procedures	Examples: a) accumulation of voids that could inhibit the ability to adequately remove heat from the reactor core b) voiding in instrument lines, resulting in erroneous indication, causing the operator to misunderstand the true condition of the plant	ATT 6 (50.72b - 1H:
C. Any event, found while shutdown, that had it been found during operation would have resulted in the plant, including principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety	1) As judged by SSS/EDO	ATT 29 (50.72b - 4H
D. Any deviation from T/S or license condition in an emergency when action is needed to protect the public health and safety (action <u>must</u> be approved at least by a licensed SRO)	1. Action required because no action consistent with license and T/S can provide adequate or equivalent protection	ATT 6 (50.72b - 1H

18. Operational Status Change (con'd)

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
E. Event that results (or should have resulted) in ECCS discharge as a result of a valid signal (manual or automatic)	1. ECCS automatic initiation set point reduced or manual initiation; OR 2. Lit logic lights for initiation on RP4; **AND** 3. Discharge to RCS is called for or verified by control console indication (flow, valve positions, tank levels, etc.)	ATT 1 UE
F. Event that results in manual or automatic actuation of Engineered Safety Features (ESF), including the Reactor Protection System (RPS)	1. As stated, except resulting from and part of a planned test Examples: a) SEC Mode Operation b) Rx trip c) SI (without water into RCS) d) Containment isolation	ATT 29 (50.72b - 4Hr
G. Scheduled shutdown for testing, maintenance refueling	1. Any planned shutdown	ATT 32 Other
H. Derating caused by regulatory action	1. Upon official notification from site management or NRC	ATT 32 Other
I. Any reactivity anomaly	1. Disagreement with predicted steady state reactivity balance during power ops $\geq$ 1000 pcm; OR 2. Calculated SDM < required; OR 3. SUR $\geq$ 5.2 dpm; OR 4. Unplanned positive reactivity insertion > 500 pcm	ATT 13 Other
J. Any unplanned criticality	1. As stated	ATT 12 UE
K. Any event or condition requiring unit shutdown to comply with any T/S LCO	1. Shutdown initiated	ATT 1 UE

18. Operational Status Change (con'd)

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
L. Major loss of emergency assessment capability, offsite response capability, or communications capability	1. Loss of significant portion of control room indication or plant monitors necessary for accident assessment; OR 2. Loss of emergency communications, including ENS; OR 3. Loss of Public Prompt Notification System (sirens)	ATT 6 (50.72b - 1Hr
M. Excessive heatup/ cooldown rates in RCS or PZR	1. Exeeding limits of T/S 3.4.9.1 or 3.4.9.2	ATT 34 Other
N. Event resulting in challenge to PORV or safety valve	1. As stated - valve(s) actuate or should have actuated	ATT 15 Other
O. Event resulting in a safety valve discharge	1. As stated on actual discharge of safety valves	ATT 34 Other

19. Public Interest Items

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Any plant conditions that warrant increased awareness on the part of STATE/LOCAL authorities	1. As judged by the SSS/EDO	ATT 7 UE
B. Any plant conditions that warrant precautionary activation of the TSC and placing EOF and other key emergency personnel on standby	1. As judged by the SSS/EDO	ATT 8 A
C. Any plant conditions that warant precautionary activation of the TSC and EOF and/or notification to the general public	1. As judged by the SSS/EDO	ATT 9 SAE
D. Unusual movements of equipment or personnel which may significantly affect local traffic patterns	1. As judged by SSS/EDO	ATT 32 Other
E. Onsite events which involve alarms, sirens or other sources of noise which may be heard offsite	1. As judged by SSS/EDO	ATT 32 Other
F. Transportation of hi or lo level radioactive material through LAC	1. As judged by site management	ATT 32 Other
G. Unusually large fish kills	1. As judged by SSS/EDO	ATT 32 Other
H. Protected aquatic species impinges on CW or SW intake screens (ex. sea turtle, sturgeon)	1. As reported by Icythological Associates or other site personnel	ATT 33 Other
I. Major loss of communications capability (off-site sirens or telephone system)	1. As judged by SSS/EDO	ATT 6 (50.72b - 1Hr)

ATTACHMENT 3

Public Service
Electric and Gas
Company

Corbin A. McNeill, Jr.
Vice President -
Nuclear

Public Service Electric and Gas Company P.O. Box 236, Hancocks Bridge, NJ 08038 609 339-4800

November 26, 1985

Mr. Don Johnson
Chief Examiner
U.S. Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, PA 19406

Dear Mr. Johnson:

SRO EXAM REVIEW

Attached please find our comments on the SRO written examination. There are numerous comments on the KEY - some of which are in complete disagreement with the key answer. There are four questions that could be interpreted by the candidate in more ways than the key reads:

5.01.a The key lists evolutions which can cause waterhammer. The question does not really ask for that.

5.07 The word "or" tends to allow an interpretation or answer which may not be based on reactivity coefficients only.

6.01.b Asks a conceptual question rather than a minimum value or basis.

8.11.b A correction (area to process monitors) was made for only one candidate.

Section 8 contains an overwhelming point value of questions which require "from memory interpretation" of Technical Specificiations and various other procedures. This is in direct contrast with the actual and good practices associated with the SRO position. In the Control Room and SSS Office, all reference materials necessary to make these decisions are available. When faced with perplexing or questionable situations, the SRO is encouraged to solicit qualified advice. In no case is a decision based on memory or assumed familiarity with a procedure. Questions 8.07 and 8.08 alone contain 20% of the point value of the section and in most cases represent what our reservations are with this section - references would be used to make a decision (real world), but on this examination, you are right or wrong.

This is not meant to negate or detract from the need for license holders to memorize those items that are required and necessary, and be familiar with approved procedures. Consideration should be given during the grading of this section to the fact that substantially more information is available (on-the-job) to the SRO for evaluating these situations. Otherwise, candidates could score <70% on this section and really not be unqualified for the job/position.

Our comments on the key are attached with their references (where appropriate). My training staff is available to discuss these issues with you, if you desire.

Sincerely,

Attachment

SRO Licensing Exam Key Corrections/Comments

Category 5

5.01.a Answer indicates operational evolutions which could cause waterhammer. However, question asks for causes of waterhammer which is: 2-phase flow (stm/water air/water) thru an abrupt change in flow direction (open valve, start pump, pipe bend).

.b Answer should also include:

Allow time for thermal equalization

5.02 Answer is correct except for last part of second line, "Rx power increases due to the positive reactivity added through MTC and Doppler". As power increases, Doppler is adding negative reactivity to balance with positive reactivity due to MTC.

5.03.a OK

.b OK

5.04.a-e Trainees provided natural circ. parameters as specified in the EOP's not the General Physics Heat Transfer Book as referenced on ANSWER KEY (There are some differences such as value for Pzr level).

Also, Part c answer should say "S/G Pressure Tracking Tc saturation pressure not Tang (Tang is not a viable indication without RCP's in service).

Answer key has an extra answer marked d. which has no corresponding question.

5.05.a Answer key has (3) parts

First - The comparison of rod movement vs dilution

Second - Why we do an ICCR

Third - A statement about doubling count rate and having distance to criticality.

The question only asks to compare two different reactivity addition methods and explain why the limits for conducting an ICCR are different. "First" statement on ANSWER KEY does this. Other acceptable answers should be:

S/D banks moving out could uncover a source range detector

and cause count rate to increase rapidly so band is bigger.

Rate of reactivity addition (K/k) for rods is different than boron dilution so different limits.

"Second" answer was not asked for in question "third" answer is irrelevent to question.

.b Answer key states "...tolerances on the work sheet". Should be:

>650 pcm due to rods
>1000 pcm due to xenon
500 pcm past ECP with no criticality

5.06.a&b Both questions ask for current yet ANSWER KEY talks about power (in part a the current would follow ppower with constant voltage so merely substitute current for power).

.b Answer indicated DECREASES, which implies it is steady at some lower value. With voids in system, the motor current would most likely fluctuate or increase and decrease.

Ref: Reference given on KEY does not really apply to a voided condition but to normal ops.

5.07.a Answer is correct except that for the present cycle cores at Salem, the doppler only power coefficient is <u>more</u> negative at EOL not less. (Doesn't matter to overall answer since MTC is more dominant).

.b Another acceptable answer.

AT EOL, we run closer t[illegible]NB (lower DNBR) due to flux shift to top of core[illegible]r pressure at top of core, higher temp. at top [illegible]e. Loss of flow causes a shift in curve which would [illegible]ke the more limiting time at EOL.

.c Answer is incorrect for current Salem core cycle due to doppler only power coefficient being <u>more</u> negative at EOL.

5.08.a Answer is incorrect for Salem. Normal pressure setting for stm. dump at Salem is (1005-1035) by lowering setpoint to 900 psia you would dump <u>more</u> steam and cause tavg to <u>decrease</u> adding <u>positive reactivity</u> to the actual critical position (ACP) would be <u>lower</u>.

.b OK

.c OK

5.09.a Since question asked for "Beginning of Cycle" whihch is underfined and due to picking off numbers from a graph the ANSWER KEY should reflect a range of acceptable answers, not just one answer.

.b Since question indicated that Rx had already tripped with all rods in than the credit for the stuck out, most reactive rod is not applicable. Answer should be the difference between answer in part a and Tech Spec limit (1.6% K/k or 1600 pcm)

.c ECP also considers reactivity effects due to the following:

Samarium

Power Defect (Since we were initially at 75% and now S/D, the power defect would add +p due to S/D. Going critical does not remove this +p addition so it must be taken into consideration.)

6.01.a Additional answers:

Design sizing of cc surge tak RMS monitors closing vent valve.

.b ANSWER KEY gives 15 psig valve, question didn't ask for this. Answer should be simply, "causing sufficient flow/cooling to #2 seal".

6.02 The first part of answer (.4 pts) isn't necessary to answer question, only that indicated flow is higher than actual flow.

6.03.a Answers are OK, but the reference should be the Tech Specs not SNGS Notebook.

.b First part of answer is unnecessary. Only the second part is needed (limits range OT T must operate in due to low pzr pressure Rx trip.)

6.04.a Actually shifts the ventilation to "accident inside air" (which is still a control room ventilation isolation)

.b Should be Hepa plus charcoal. Also, must alarm name be mentioned since question only asks for any interlock?

.c OK

.d OK

6.05.a Questions ask for 500 KV BKR's yet ANSWER KEY is for 13 KV ring Bus BKR's.

Also, question only gives status of #2 generator (not #1) so Unit 1 output BKR's may be shown as open or closed depending on whether trainee considered Unit 1 on or off line.

.b OK

.c 1T60 and 2T60 ciruit switchers may be shown open or closed for a blackout condition. The only auto open signal

for these is a ground fault on 13 KV Bus Section 1 or 4, so a blackout condition would leave them closed unless the operators manually opened them.

6.06.a May see these for components:

SGFP
Turbine (emergency oil pump)
Generator (emergency seal oil pump)

.b Question asks for "cooldown after a loss of power accident." The exam proctor stated that the vital busses were still to be considered energized (i.e., not a loss of all AC) for this to occur the Emergency Diesel Generators must be the source of power to the vital Busses. With this in mind, additional support systems needed would be:

Emergency Diesel Generators
Service Water (to diesel generators)
Atmosphere Steam Relief (MS-10 or main steam safeties) to allow you to cooldown to PNR initiation point of 350°F.

ANSWER KEY reference is LOPA 1 which is for loss of all AC (less than 2 vital Busses energized which is not the situation given in the question.)

6.07 Answer:

Hl Head SI pump flow rate is between 150 gpm - 550 gpm, shutoff to runout. For 1700 psig would be about 350-400 gpm.

350/400 x 2 = 700 to 800 gpm

Intermediate head SI pumps have a shutoff head of about 1550 psig and would not be injecting but running on recirc.

The positive displacement charging pump would remain running if already in that condition (no vital Bus component tripping for Mode 1 sec actuation 50 could see additional 75 gpm)

Answer should be in the 700-900 gpm range.

6.08.a Power range NI's are not an input to the SGWLC System at Salem.

.b OK

6.09.a Also manual SI actuation

.b OK

1.10.a OK

b. Answer. 256 is for bank overlap unit total counter not Bank B step counter. Bank C starts to move out when Bank B is

at 128 steps. Also, this <u>does</u> <u>not</u> correspond to 50% withdrawn.

.c OK

.d

(1) OK

(2) OK

(3) Rods would step in for some time frame due to the powe mismatch circuit associated with auto rod control.

7.01 OK

7.02 OK

7.03 The pump combination listed on question is an <u>acceptable</u> combination to prevent the inadvertent SI created by 5/6 Diff. Press. The only correct answer is: "uneven RCS temps among the loops"

7.04 ANSWER KEY references the Westinghouse Owner Group Emergency Response Guidelines and not the Salem Specific EOP's so the parameters and setpoints are not entirely correct. Should be:

Nuclear Power >5% after Rx trip

Core Exit Tc >1200°F

Core Exit Tc >700°F and RVLIS full range not >50% and no RCP's running and <10°F SWBC001

NR S6 Level <15% and feed flow <22E4

Tc decrease >100°F in 60 min and to left of limit A on graph (not Tc<16°F)

CTMT press >47 psig (not 17 psig)

7.05.a Break flow and injection flow, find new equalibrium at a new lower value.

.b As pumps are taken out of service, the effect on total flow (% flow change) is greater <u>and</u> if when a pump is taken out of service, the break f' w is now > injection flow, the press. would decrease an.. subcooling would decrease.

7.06.a OK

.b OK

7.07.a OK

.b OK

7.08

1. Also, would exceed 2000 mr/qtr which requires a dose extension.

2. ANSWER KEY should be 1000 mr/qtr not 2100 mr/qtr.

3. Answer should show limit as 500 mr/gestation period

4. Also, PBR our AP this individual would have already exceeded the 100 mr/qtr limit which would have required some dose extension.

7.09.a Also, ESF instrumatation (not listed in the AOP)

.b 2c Diesel generator is inoperable so verify other AC sources operable within (1) hr.

.c OK

7.10.a OK

.b OK

.c Shutdown reactor only (if 1865 psig auto Rx trip)

.d OK

.e Restore containment integrity within (1) hour or commence shutdown (per Teach Specs)

7.11 OK

8.01 This is not a problem that can be corrected by a procedure change. Correct answer should be:

Workers would not be allowed to install these supports since it deviates from original work package. Would need a system design change initiated with appropriate QA, Enginneering, etc., review and approval.

8.02 Also PER "Event Classification Guide Attachment 29 Notifications"

STA
PRI/SEC Communicators
NRC
Operations Manager or Operations Engineer
Lower Alloways Creek (local municipality)

8.03 OK

8.04 Vice President Nuclear Notification within 24 hours (additional answer)

8.05 OK

8.06.a OK

.b OK

8.07.a Could need red blocking tag to lift lead (leads are lifted de-energized)

If the leads were left in the circuit at completion of job (even though it was in accordance with a procedure) a lifted leads/jumper tag would be needed.

.b Red blocking tags to isolate system to remove relief valve, if system was to be placed back in-service, install lifted leads/jumper tag and remove red blocking tags.

.c Red blocking tag with a temporary release to jog pump. Could not jog pump with a yellow permissive tag since yellow permissive tags must be used in conjunction with red blocking tags.

.d Red blocking tag with a temporary release <u>or</u> none.

Worker blocking tags are only authorized to be used by T&D personnel (Transmission & Distribution) per AP-15.

.e For a field switch - OK

For a control room switch - No, notation is shift logs

8.08.a Yes, there is no bistable for OP T from PR NIS (set to zero)

.b Question could be answered (2) ways:

Yes, because lifting head on by itself does not constitute a mode change)Do not enter Mode 5 until last bolt on head is tensioned)

No, procedure to lift head also includes steps to tension studs so permission would be given to complete procedure not just a position.

Bottom line is, did the trainee know that you can't change modes in an action statement?

.c Answer is totally wrong!

Should be yes. If you check redundant equip. for operability first; and you feel tht work can be completed within action time frame. How else could you repair anything.

.d OK

.e OK

8.09.a OK

.b OK

8.10.a If the PRI -> SEC leak is thru a S/G leak (1) only (1) S/G, then you will exceed the 500 gal/day Tech Spec limit

and would take appropriate action

(.8 gmp)(60 min) = 48 gal/hr

(48 gal/hr)(24 hrs) = 1152 gal/day

.b OK

8.11.a OK

.b Question incorrectly states area monitors R41B, C (These are process monitors). Exam proctor did correct this for (1) trainee who questioned it, however, no general correction statement was made to group.

The question doesn't provide sufficient information to be specific as to classification. In addition to ANSWER KEY, other possible answers would be:

1. Plant vent >1.5E3

Section 6, Item A
Attachment 29 notification

2. >50 gpm leakage

Section 1, Item B
Attachment 2

3. Based on "area" RMS monitor

Section 18, Item C
Attachment 29

c OK

.d OK

ACTIONS	COMMENTS/CONTINGENCY ACTIONS
3.23 Is average RCS temp less than 554°F?	
YES / NO	a. PERFORM step 3.23.1 when average temp less than 554°F.
	b. GO TO step 3.24.
3.23.1 CLOSE:	
a. 11-14BF19.	Feedwater Control Valve.
b. 11-14BF40.	Feedwater Control Bypass.
c. 11-14BF22.	SG Inlet Stop/Check.
3.24 Are both Reactor Trip breakers open?	
YES / NO	a. DISPATCH Operator to locally open both Trip Breakers.
	b. IF Trip Breaker will not open, THEN request the I&C Department jumper in a P-4 signal for that train.
3.25 CONTROL natural circulation with AFW flow and dumping steam.	
3.26 MONITOR natural circulation with the following trended parameters:	
3.26.1 Core Exit TCs stable or decreasing.	
3.26.2 RCS Hot Leg temp stable or decreasing.	
3.26.3 SG press stable or decreasing.	
3.26.4 RCS Cold Leg temp at saturation temp for SG press.	

MASTER

ESTIMATED CRITICAL POSITION WORKSHEET

3.0 LIMITS ON CRITICAL ROD POSITION

3.1 INSERTION LIMIT (FIG.14): BANK C__________STEPS

3.2 INTENDED ROD POSITION +500 PCM

(ITEM 2.4 +500) = __________PCM
ROD POSITION AT THIS WORTH (FIG. 4, HZP): BANK C__________STEPS

BANK D__________STEPS

3.3 INTENDED ROD POSITION -500 PCM:

(2.4)-500 = __________PCM
ROD POSITION AT THIS WORTH (FIG. 4, HZP): BANK D__________STEPS

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4.0 REACTIVITY CHANGES AND SUM

4.1 CONTROL RODS (ITEM 1.6 - ITEM 2.4) = __________PCM

4.2 POWER DEFECT: (AT POWER IN 1.3 AND BORON CONCENTRATION IN 1.4) (FIGURE 2) = +__________PCM

ON-THE-SPOT CHANGES NOT ALLOWED

4.3 XENON REACTIVITY

(A) XENON REACTIVITY AT TIME IN 1.2: (FIG. 6) (-)__________PCM

(B) ELAPSED TIME FROM 1.8 TO 2.2: __________HRS.

(C) XENON REACTIVITY AT TIME IN 2.2: (FIG. 8) (-)__________PCM

(D) REACTIVITY CHANGE (ITEM C - ITEM A) = __________PCM

4.4 SAMARIUM REACTIVITY

(A) ELAPSED TIME 4.3 (B): __________HRS.

(B) REACTIVITY CHANGE (FIG. 10): (-)__________PCM

4.5 REACTIVITY CHANGE SUM:

__________ + __________ + __________ + __________ = __________PCM
(4.1) (4.2) (4.3D) (4.4B)

NOTE: IF THE ABSOLUTE MAGNITUDE OF ITEM 4.1 IS GREATER THAN 650 PCM OR, IF THE ABSOLUTE MAGNITUDE OF ITEM 4.3D IS GREATER THAN 1000 PCM, THEN USE AN ICRR PLOT TO GUIDE CRITICAL APPROACH.

5.13.1 The Control Bank height at Criticality is below the Rod Insertion Limit for zero power.

___ a. NOTIFY the Senior Shift Supervisor/Shift Supervisor.

___ b. INSERT all Control Banks.

___ c. CALCULATE a Shutdown Margin — Tech Spec 3.1.1.1

___ d. RECALCULATE the ECP

___ e. DETERMINE AND CORRECT the error

5.13.2 The Control Bank height at Criticality is 500 PCM below the ECP but above the Rod Insertion limit.

___ a. NOTIFY the Senior Shift Supervisor/Shift Supervisor.

___ b. INSERT all Control Banks.

c. RECALCULATE the ECP

d. DETERMINE AND CORRECT the error

5.13.3 Criticality has not been achieved with the Control Banks withdrawn to an equivalent of 500 PCM past the ECP.

___ a. NOTIFY the Senior Shift Supervisor/Shift Supervisor.

___ b. INSERT the Control Banks to the position for 500 PCM below criticality

___ c. RECALCULATE the ECP

___ d. DETERMINE AND CORRECT the error. If no error was found,

Mark N/A the steps not required.

___ 1. REQUEST confirming boron sample

___ 2. NOTIFY Reactor Engineering

___ 3. FULLY INSERT all Control Banks

___ 4. PULL rods for criticality using an ICCR plot

___ 5.14 When the P-6 (Source Range Permissive) light is energized, BLOCK the Source Range High Flux Trip by depressing both the BLOCK SOURCE RANGE "A" and BLOCK SOURCE RANGE "B" pushbuttons on the console.

___ 5.14.1 VERIFY the Source Range Trains A&B TRIP BLOCKED light on RP4 and the SR Detector Voltage Trouble Overhead Annunciator alarm are energized.

___ 5.14.2 When P-6 is exceeded and the Source Range Channels are blocked, the Second Pen on the NR-45 should be switched to the other Intermediate Range Channel.

___ 5.15 LEVEL off the Reactor Neutron level at 1.0E-8 amps in the Intermediate Range.

___ 5.15.1 VERIFY Critical Rod Position is not less than the limits of Tech Spec 3.1.3.5 and Figure 14 of the Reactor Engineering Manual within 4 hours of achieving criticality.

5.07 a:c

Core Parameters Data Sheet (FROM Rx ENGINEERING DEPT.)

Item/Parameter	Unit 1 Cycle 6	Unit 2 Cycle 3
1. Core Rating	3338 MW_t	3411 MW_t
2. Full Length Control Rods	53	53
3. Control Rod Worth:		
a. Control Banks (BOL)	3475 pcm	4350 pcm
b. Shutdown Banks (BOL)	4475 pcm 7590	2650 pcm 7000
4. Enrichments: (w/o)		
a. Region 3	—	3.12
b. Region 4	—	3.41
c. Region 5A	2.80	3.80
d. Region 5B	3.40	3.40
e. Region 6	3.40	—
f. Region 7	3.40	—
g. Region 8A	3.40	—
h. Region 8B	3.80	—
5. BPR's (12.5 w/o B_2O_3)	1660 Fresh	1664 Fresh
6. βeff (Effective Delayed Neutron Fraction)		
a. BOL	.006116	.006292
b. EOL	.004994	.005014

300-18:23

Item/Parameter	Unit 1 Cycle 6	Unit 2 Cycle 3
7. Xenon:		
a. 100% (Equilibrium)	-3175 pcm	-2725 pcm
b. 50% (Equilibrium)	-2600 pcm	-2275 pcm
c. 100% (Peak)	-5700 pcm	-5200 pcm
d. 50% (Peak)	-3425 pcm	-3100 pcm
8. Samarium		
a. Equilibrium	588 pcm	588 pcm
b. 100% (Peak)	944 pcm	944 pcm
9. Doppler Only Power Coefficient		
a. HZP (BOL)	-15.0 pcm/%	-14.2 pcm/%
b. HFP (BOL)	- 9.8 pcm/%	- 9.6 pcm/%
Average (BOL)	≈ -12 pcm/%	≈ -12 pcm/%
c. HZP (EOL)	-22.5 pcm/%	-24.2 pcm/%
d. HFP (EOL)	- 9 pcm/%	- 9.6 pcm/%
Average (EOL)	≈ -16 pcm/%	≈ -17 pcm/%
10. Moderator Temperature Coefficient		
a. BOL (ARO, HFP, 571°F/1000 PPM)	-12.1 pcm/°F	-17 pcm/°F
b. EOL (ARO, HFP, 571°F/0 PPM)	-32.1*pcm/°F	-33*pcm/°F
	*Note: This value by Tech Spec is allowed to be -38. PPM point 300 PPM point must not be more NEG. than -29.	*Note: This value by Tech Spec is allowed to be -40. PPM point 300 PPM point must not be more NEG. than -31.

Item/Parameter	Unit 1 Cycle 6	Unit 2 Cycle 3
11. Power Coefficient		
a. BOL (HZP 1000 ppm)	-17.6 pcm/%	-21.5 pcm/%
b. BOL (HFP 1000 ppm)	-15.0 pcm/%	-16.2 pcm/%
Average BOL	≃ -16 pcm/%	≃ -19 pcm/%
Power Coefficient (Cont.)		
c. EOL (HZP, 0 ppm boron)	-33.0 pcm/%	-34.5 pcm/%
d. EOL (HFP, 0 ppm boron)	-22.5 pcm/%	-24.2 pcm/%
Average (EOL)	≃ -28 pcm/%	≃ -29 pcm/%
12. Doppler Only Power DEFECT		
a. BOL	-1200 pcm	-1150 pcm
b. EOL	-1310 pcm	-1400 pcm
13. Power Defect		
a. BOL (900 ppm)	-1625 pcm	-1750 pcm
b. EOL (0 ppm)	-2350 pcm	-2550 pcm
14. Differential Boron Worth		
a. BOL (800 ppm)	≃ -8.4 pcm/ppm	≃ -8.45 pcm/ppm
b. EOL (0 ppm)	≃ -10.2 pcm	≃ -10 pcm/ppm

	Unit 1	Unit 2
Item/Parameter	Cycle 6	Cycle 3
THUMB RULES		
Most Reactive Stuck Rod Worth	1285 pcm	900 pcm
Boration Factor	3 gal/ppm	3 gal/ppm
Boron Worth	10 pcm/ppm	10 pcm/ppm

5.0 PROCEDURE

5.1 Operation During Plant Startup

5.1.1 VERIFY that the Condenser Vacuum permissive light on 1RP4 is ON.

5.1.2 If Tavg block is bypassed (Tavg Bypass light ON), DEPRESS the Train "A" and "B" OFF-RESET Bypass Tavg pushbuttons to reset.

5.1.3 PLACE Steam Dump Interlock Train "A" and "B" controllers to ON.

5.1.4 If the Turbine Impulse Chamber Pressure permissive light is ON, DEPRESS the RESET LOAD REJECTION pushbutton. VERIFY that permissive light goes out.

5.1.5 ADJUST Main Steam Pressure setpoint to desired value. This is 1005 psig for a no load Tavg of 547°F.

5.1.6 VERIFY that all of the Steam Dump Valves are closed.

5.1.7 PLACE system in Main Steam Pressure Control Mode.

5.1.8 VERIFY that the Block Cooldown and Block Non-Cooldown lights are ON. (This indicates that the valves are blocked). As Tavg increases to greater than 543°F, the Block Cooldown and Block Non-Cooldown lights will go OFF.

5.1.9 PLACE the Main Steam Pressure Controller in AUTO.

5.1.10 OBSERVE that the Steam Dump Valves open and MAINTAIN Main Steam Pressure as plant heatup to 547°F is completed.

5.2 Operation While At Power

5.2.1 VERIFY that the loss of load interlock is reset by observing that the Turbine Impulse Chamber Pressure permissive light on the status panel is OUT.

5.2.2 If the loss of load interlock is not reset, PUSH the Reset Load Rejection pushbutton. VERIFY reset by Turbine Impulse Chamber Pressure light going OUT.

5.09 c

SECTION 1.7

ESTIMATED CRITICAL POSITION WORKSHEET

1.0 PREVIOUS CRITICAL CONDITIONS (FROM CONTROL ROOM LOG)

1.1 DATE: ____________

1.2 TIME: ____________

1.3 % POWER: ____________

1.4 BORON CONCENTRATION: ____________ PPM

1.5 CONTROL BANK POSITION: BANK C ____________ STEPS

BANK D ____________ STEPS

1.6 INTEGRAL ROD WORTH (FIG. 4) ____________ PCM

1.7 CORE EXPOSURE (MWD/MTU) ____________ MWD/MTU

1.8 TYPE OF SHUTDOWN

(A) RX TRIP AT DATE: ____________ TIME: ____________

(B) ORDERLY SHUTDOWN DATE: ____________ TIME: ____________

APPROXIMATE SHUTDOWN RATE: ____________ %/MIN.

2.0 INTENDED CRITICAL CONDITIONS

2.1 DATE: ____________

2.2 TIME: (1) ____________

2.3 CONTROL BANK POSITION: BANK C ____________ STEPS

BANK D ____________ STEPS

2.4 INTEGRAL ROD WORTH (FIG. 4 HZP CURVE): ____________ PCM

(1) THIS ECP MUST BE COMPLETED WITHIN FOUR (4) HOURS OF GOING CRITICAL, IAW TECHNICAL SPECIFICIATION 4.1.1.1.1(c)

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ON-THE-SPOT CHANGES NOT ALLOWED

ESTIMATED CRITICAL POSITION WORKSHEET

3.0 LIMITS ON CRITICAL ROD POSITION

3.1 INSERTION LIMIT (FIG.14): BANK C__________STEPS

3.2 INTENDED ROD POSITION +500 PCM

(ITEM 2.4 +500) = __________PCM
ROD POSITION AT THIS WORTH (FIG. 4, HZP): BANK C__________STEPS

BANK D__________STEPS

3.3 INTENDED ROD POSITION -500 PCM:

(2.4)-500 = __________PCM
ROD POSITION AT THIS WORTH (FIG. 4, HZP): BANK D__________STEPS

4.0 REACTIVITY CHANGES AND SUM

4.1 CONTROL RODS (ITEM 1.6 - ITEM 2.4) = __________PCM

4.2 POWER DEFECT: (AT POWER IN 1.3 AND BORON CONCENTRATION IN 1.4) (FIGURE 2) = +__________PCM

4.3 XENON REACTIVITY

(A) XENON REACTIVITY AT TIME IN 1.2: (FIG. 6) (-)__________PCM

(B) ELAPSED TIME FROM 1.8 TO 2.2: __________HRS.

(C) XENON REACTIVITY AT TIME IN 2.2: (FIG. 8) (-)__________PCM

(D) REACTIVITY CHANGE (ITEM C - ITEM A) = __________PCM

4.4 SAMARIUM REACTIVITY

(A) ELAPSED TIME 4.3 (B): __________HRS.

(B) REACTIVITY CHANGE (FIG. 10): (-)__________PCM

4.5 REACTIVITY CHANGE SUM:

__________ + __________ + __________ + __________ = __________PCM
(4.1) (4.2) (4.3D) (4.4B)

NOTE: IF THE ABSOLUTE MAGNITUDE OF ITEM 4.1 IS GREATER THAN 650 PCM OR, IF THE ABSOLUTE MAGNITUDE OF ITEM 4.3D IS GREATER THAN 1000 PCM, THEN USE AN ICRR PLOT TO GUIDE CRITICAL APPROACH.

INSTRUCTIONAL CONTENT:

6.01a 2.2 COMPONENTS

2.2.1 Surge Tank

There is one Component Cooling Surge Tank provided in this system. The surge tank is a horizontally mounted carbon steel cylinder. It is connected to the suction header of the cooling water pumps via two 4-inch lines. The surge tank has a capacity of 2000 gallons with a normal water volume of 1000 gallons. The tank is designed with an internal baffle to divide the tank into two separate volumes which provides redundancy for a passive failure during recirculation following a loss of coolant accident.

The Component Cooling Surge Tank capacity permits:

a. The surge tank to accomode surges in system resulting from thermal expansion and contraction.

b. The normal air volume in the tank to accomodate the amount of reactor coolant entering the component cooling loop following a rupture of a reactor coolant pump thermal barrier cooling coil for a period of three (3) minutes.

c. The normal water volume in the tank provides an intermediate source of makeup to the loop if a leak develops in the system. The water volume is separated into two parts by a baffle which protects against complete drainage of the tank in the case of a leak or failure in one "half" of the system.

The surge tank has a flanged connection at the top for additions of chemical corrosion inhibitor to the component cooling loop. For the purpose of homogenizing this chemical a one-inch recirculation line from the pump discharge is provided. The tank is normally vented to the ABV exh. fan disch via air-operated valve CC149, . The relief valve CC147 on the surge tank is

INSTRUCTIONAL CONTENT:

The tank is vented to the disch. duct of the ABV EXH. via valve CC149. The valve automatically closes upon a high radiation level. Scintillation-type radiation detectors located downstream of each CCW heat exchanger sensing an activity of 0.5 decade above their minimum sensitivity closes CC149 and actuates the alarm "21(22) CC HEADER HIGH ACTIVITY" on the Surge Tank Vent Valve bezel. This vent valve cannot be opened until the radiation level returns to normal. Water withdrawn for radiation detection is recirculated back to the pump suction header through a 3/4"line. The radiation detectors (R-17A and R-17B) are located in the Component Cooling Water Heat Exchanger Room, 84' elevation of the associated Auxiliary Bldg.

The surge tank is designed to Section VIII of the ASME Boiler and Pressure Vessel Code. The tank parameters are as follows:

Number	1
Type	Horizontal, with divider plate
Design Pressure:	
Internal, psig	100
External, psig	15 (vacuum breaker provided) CC148
Design temperature, ^{o}F	200
Normal operating pressure, psig	Atmospheric
Total volume, gal.	2000
Normal water volume; gal.	1000
Material	Carbon Steel

6.03 a.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above 1.30 for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor for all single or multiple dropped rods.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Delta T

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Operation with a reactor coolant loop out of service below the 4 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature delta T setpoint. Three loop operation above the 4 loop P-8 setpoint is permissible after resetting the K1, K2 and K3 inputs to the Overtemperature delta T channels and raising the P-8 setpoint to its 3 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower Delta T

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

6.04a

OPERATING PROCEDURE
II-17.3.2
CONTROL ROOM VENTILATION OPERATION

1.0 PURPOSE

1.1 This instruction describes the operation of the Control Area Air Conditioning System in the following modes of operation:

1.1.1. Normal

1.1.2 Accident Inside Air

1.1.3 Accident Outside Air

1.1.4 Fire Inside Control Area

1.1.5 Fire Outside Control Area

2.0 INITIAL CONDITIONS

2.1 The Control Room Ventilation System is aligned as follows:

2.1.1 TRIS lineup II-17.3.2 has been completed.

OR

2.1.2 A Components Off Normal List for TRIS lineup II-17.3.2 has been generated and all components listed have been evaluated for effects on normal system operation.

2.2 RMS channels 1R1A and 1R1B are in service IAW IV-11,3.1, "Area Radiation Monitors-Normal Operation" and IV-11.3.3, "Process Radiation Monitors- Normal Operation".

2.3 The normal Supply Fan "ROLL-O-MATIC" filters are selected to AUTO.

2.4 Heating water is available to the supply unit heating coils during cold weather operation.

2.5 Chilled water is available to the cooling coils on the normal and emergency supply units.

2.6 The Fire Protection System is pressurized, and 11FP80 is open.

MASTER

5.2.2 If fire is outside of the areas serviced by the Control Area Air Conditioning System, proceed as follows:

a. At 1RP2, DEPRESS the FIRE OUTSIDE CMC pushbutton module.

b. OBSERVE the dampers listed on Table 1 shift to the positions listed in OUTSIDE CONTROL AREA column.

5.2.3 Dampers 1CA201 thru 1CA207 are normally open. A fire in the proximity of any one of these dampers will cause them to close. These dampers must be reset by the fire brigade after the fire has been extinguished either at Panel 796-1, Control Room Aux. Equip Room el. 122', or locally at the damper itself.

5.2.4 If fire is in the Control Area Relay Room, the Halon System will be actuated. OBSERVE the dampers listed on Table 1 shift to the positions listed under Control Area Relay Room.

5.2.5 After the fire is extinguished, RETURN to NORMAL Control Area Air Conditioning. IAW Section 5.1.

5.3 Operation During Accident Conditions

NOTE

This mode is normally automatically initiated by Safety Injection or alarm of RMS channels 1R1A or 1R1B for unit 1 or 2. (A signal on either unit isolates both units).

5.3.1 If both red lights on the "A" and "B" Control Area Isolation CMC switches have not illuminated, DEPRESS the ACCIDENT-INSIDE AIR CMC switch or 1RP2.

NOTE

Control Room Intake Duct Isolation of either unit will isolate both units Control Room Intake Ducts.

5.3.2 VERIFY that the dampers of Table 1 are in the positions listed in ACCIDENT-INSIDE AIR column.

MASTER

TABLE 1
500KV BREAKER LINEUPS

Lineup #1 - No. 1 Main Generator Synchronized. ONLY !!

Lineup #2 - No. 2 Main Generator Synchronized.

Lineup #3 - #1 and #2 Main Generators Synchronized.

Lineup #4 - Both Main Generators Off The Line.

Lineup #5 - Blackout

LINEUP NUMBER	CIRCUIT BREAKER NUMBERS									
	1-5	5-6 (1)	2-6 (1)	2-8 (2)	1-8 (2)	1-9	9-10 (3)	2-10 (3)	500KV 2T60	500KV 1T60
1	X	X	X	X	X	O	O	X	X	X
2	O	O	X	X	X	X	X	X	X	X
3	X	X	X	X	X	X	X	X	X	X
4	O	O	X	X	X	O	O	X	X	X
5	O	O	O	O	O	O	O	O	O	O

NOT TRUE UNLESS ACCOMPLISHED MANUALLY (SEE ATTACHMENT)

Notes

1. Open when 5024 Line is not in service.
2. Open when 5021 Line is not inservice.
3. Open when 5037 Line is not in service.

CONTENT/SKILLS

d. The neutral bushing on the high side of each of the three single phase transformers are connected directly to station ground.

01 2. Circuit Breakers (Nos. 1-5, 5-6, 2-6, 2-8, 1-8, 2-10, 9-10, 1-9)

a. These breakers provide the automatic isolation of the appropriate sections of the 500 kV switchyard when a fault/malfunction is detected. These breakers also provide the capability to alter the alignment of the 500 kV switchyard from a remote location (the control room).

b. Ratings

1) Westinghouse - SF_6 type

2) 3000 amps

3) 1800 kV

04 c. More detailed information about these circuit breakers is available in section IV. C of this document and in Attachment 1 of this handout.

3. Load Interruptng Switches (1T60, 2T60)

01 a. These switches will automatically open when a ground fault is detected on 13 kV Bus Sections 1 or 4. Thus isolating the 13 kV Bus Section from the 500 kV ring bus.

CONTENT/SKILLS

d. The neutral bushing on the high side of each of the three single phase transformers are connected directly to station ground.

01 2. Circuit Breakers (Nos. 1-5, 5-6, 2-6, 2-8, 1-8, 2-10, 9-10, 1-9)

a. These breakers provide the automatic isolation of the appropriate sections of the 500 kV switchyard when a fault/malfunction is detected. These breakers also provide the capability to alter the alignment of the 500 kV switchyard from a remote location (the control room).

b. Ratings

1) Westinghouse - SF_6 type

2) 3000 amps

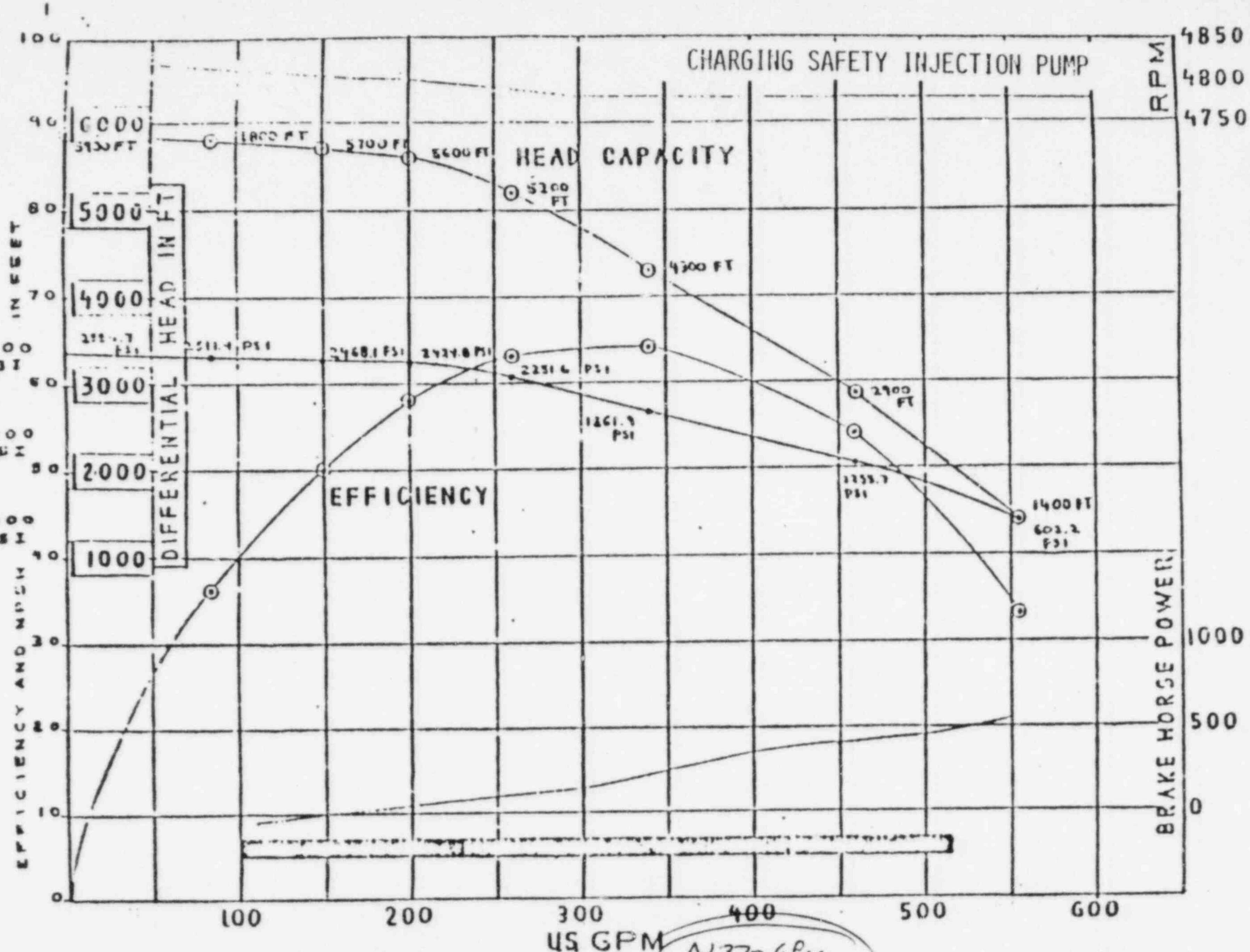
3) 1800 kV

04 c. More detailed information about these circuit breakers is available in section IV. C of this document and in Attachment 1 of this handout.

3. Load Interruptng Switches (1T60, 2T60)

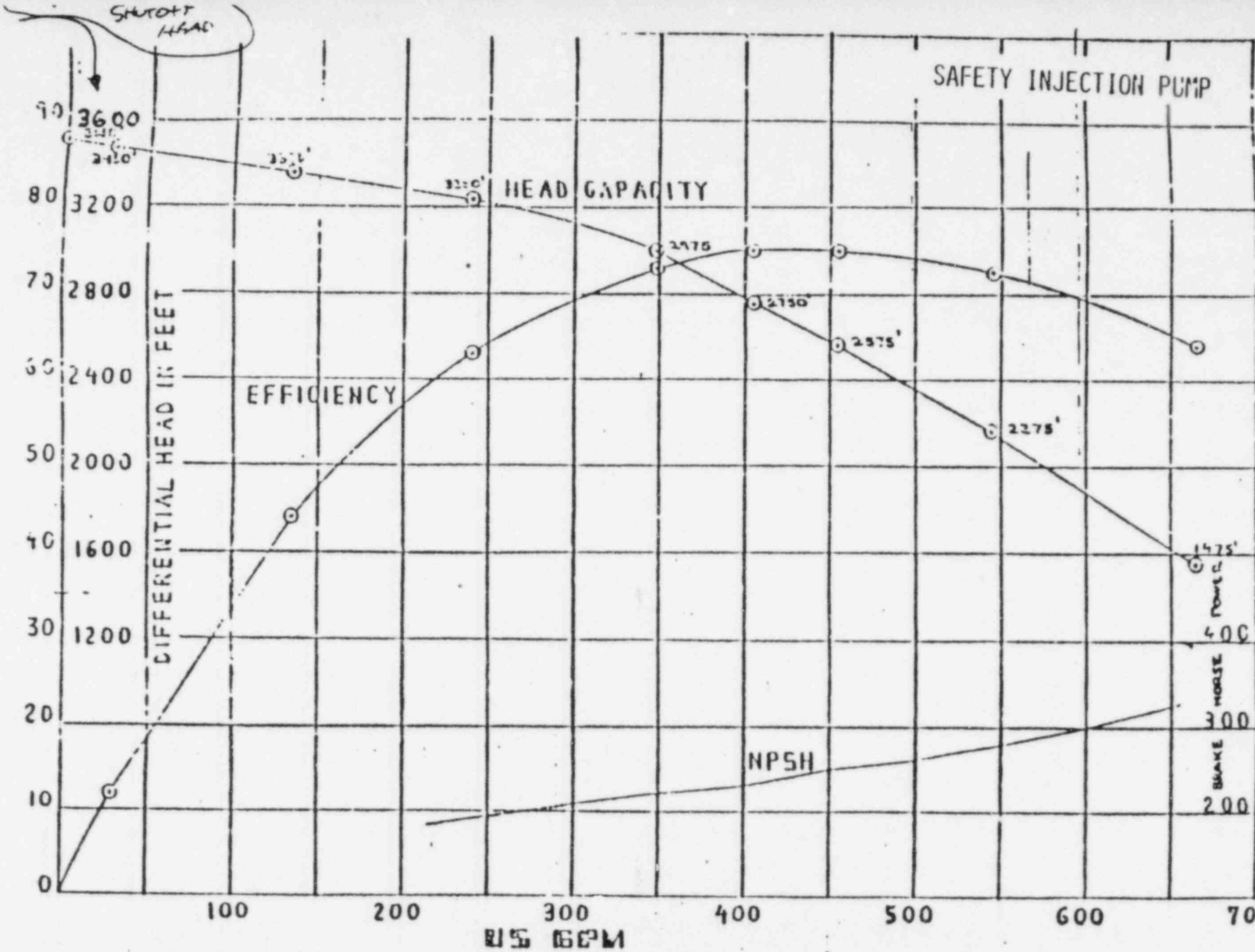
01 a. These switches will automatically open when a ground fault is detected on 13 kV Bus Sections 1 or 4. Thus isolating the 13 kV Bus Section from the 500 kV ring bus.

(Full Rx)



N 370 GPM

6.07


SHUTOFF HEAD
SAFETY INJECTION PUMP
HEAD CAPACITY
EFFICIENCY
NPSH
DIFFERENTIAL HEAD IN FEET
BRAKE HORSE POWER
US GPM
3600
3200
2800
2400
2000
1600
1200
90
80
70
60
50
40
30
20
10
0
100
200
300
400
500
600
700
400
300
200
2275'
1475'

6.08 a

SGWLC
LOGIC

6.10b

rods. The master cycler always transmits the zero count "GO" pulse to slave cyclers servicing group 1 rods. The three-count "GO" pulses always go to the group 2 slave cyclers. In this way the group of a bank are staggered in movement as they are moved.

Certainly, one question remaining concerns how the power cabinet knows which group of rods is to receive the current orders from the slave cycler since each slave can service several groups in a power cabinet. Remember that it was the bank overlap unit that determined which slaves were to receive "GO" pulses. Likewise, the BOU again helps out by determining the exact group in a power cabinet that will receive current orders. In the figure this signal has been labeled "Group Select" or "Multiplex" and is seen entering the power cabinets.

Bank Overlap Unit

The major function of the bank overlap unit is to ensure that the control bank rods move with proper overlap and in sequence. Remember that bank overlap is only used when the Bank Selector is in AUTO or MAN, since the other positions are for individual bank control. In the discussions of the slave and master cyclers, it has been shown that the bank overlap unit was responsible for directing the "GO" pulses and current orders so that the proper groups of rods would move.

To accomplish its functions, the BOU has three major sections: the counter, the thumbwheel switches, and the multiplexer (or sequencer).

The counter section has two major inputs. It receive zero count and three count pulses from the output of the master cycler. It also receives IN and OUT signals from the supervisory circuit. By counting the total number of pulses, the counter knows how many steps the rods have taken. The IN/OUT information allows the unit to add pulses for outward rod movement and subtract for inward rod motion.

The objective of proper <u>sequencing</u> is to see that when the control rods are on the bottom, they move out, beginning with bank A, and are followed by B, C and D in order. They also must stop traveling when they reach the top of their travel (228 steps). Inward rod motion is in the opposite sequence. The <u>overlap</u> function decides <u>when</u> in the sequence the next bank will move. The plant uses 100 step overlap. This means that control bank B will begin moving out when bank A reaches 128 steps withdrawn, bank C will begin moving when bank B reaches 128 steps withdrawn, and bank D will

10 b

start moving when bank C is 128 steps withdrawn. Figure 16 shows the bank overlap feature for rods moving in their proper sequence. Notice that one axis is Total Step Count. Bank overlap occurs at 128, 256, and 384 total steps. The other numbers represent the top of each bank's travel in total steps.

The bank overlap counter is continually calculating the total step count based upon the zero and three count pulses. The thumbwheel section monitors the counter and will both initiate the next bank's movement and stop a bank once it reaches 228 steps withdrawn. There are six thumbwheel switches, each with three adjustable thumbwheels used to establish the bank overlap and rod travel points. Switches S1, S3, and S5 are for overlap of control banks B, C and D respectively. Switches S2, S4, and S6 are the rod withdrawal limits (228 steps for each bank) for control of banks A, B and C. The rod withdrawal limit for bank D is <u>not</u> covered by thumbwheel switch and will be discussed later.

When the counter has reached the setpoint for a thumbwheel, this information is instantly sent to the multiplex section. Two singals emerge form this section, which have already mentioned. One signal tells the master cycler which slave cycler should receive "GO" pulses, such as when bank movement should begin. Conversely, it would tell the master cycler to cease "GO" pulses to a slave cycler when that bank has reached its travel limit. The other signal from multiplex goes to the correct power cabinet to "multiplex", or group select, the correct group within the cabinet.

As a result, if bank D rods are moving, the multiplex section of the BOU is directing the master cycler "GO" pulses to slave cyclers 1BD and 2BD, while also sending signals to power cabinets 1BD and 2BD to make sure only bank D rods will get the firing orders from the slave cyclers.

Realize that when an overlap condition exists, two banks are moving at one time, meaning that the master cycler "GO" pulses are going to all four slave cyclers and, therefore, all four power cabinets are being used.

The bank overlap unit has a digital display and three pushbuttons adjacent to the thumbwheels. The digital display always shows the total step count. Under normal conditions, this display should indicate a total count somewhere between zero and 612 steps. The RESET button would reset the bank overlap counter back to 0 counts, which can be verified by observing the digital display. This is <u>not</u> normally done. If a reset was performed while banks are

6.10d

The movable gripper latches now support the drive rod. The lift coil is now energized. This action results in the lift pole piece attracting and thus raising the lift armature, the movable gripper armature, the movable gripper arms, the drive rod, and attached control rod up approximately 5/8 inch, or 1 step. At this point, the stationary coil is again energized with maximum current. The stationary gripper arms move in and lift the drive rod 1/16 inch. The stationary gripper arms assume control rod support from the movable gripper arms.

The movable gripper coil is deenergized, and the movable gripper arms move out. The lift coil is deenergized, and the movable gripper latch assembly is lowered to its normal position. Finally, the stationary coil current is reduces to its hold value and the drive rod lowers 1/16 inch. This action completes the withdrawal sequence that raises the control rod 5/8 inch (one step).

The sequence to raise (or lower) the drive rod one step is accomplished in 780 m sec. The maximum rod speed is 72 steps per minute (SPM), or 45 inches per minute. By varying the time between steps, the control rod speed is varied between 72 and 8 SPM.

Reactor Control Unit (Figure 9)

The reactor control unit develops the electrical control signals to be sent to the Rod Control System to move the control rods when in automatic control. This circuit will not only determine the direction of rod travel but will also provide a rod speed signal. The majority of this circuit is located in a nonprotection section of the process control racks, with a small portion coming from power range Excore NIS. The figure provides a functional approach to circuit operation.

Before covering the details of this circuit, it is worthwhile to take an overview of the Reactor Control Unit. Keep in mind the overall objective of the Rod Control System when in the automatic mode of operation. The rods are positioned to keep T_{avg} on a program that is based upon plant power level. Having RCS average temperature increase with power helps to minimize the drop in secondary steam pressure. Turbine inpulse pressure is used as the input to develop the T_{avg} program. This is an accurate indication of

plant power level and normally power changes occur here first with nuclear power following steam demand. Auctioneered high RCS average temperature provides a feedback of the actual parameter being controlled, thereby allowing fine control of T_{avg} to its program. By comparing the program or reference temperature (created by turbine impulse pressure) with T_{avg}, a temperature error signal (T_{error}) can be generated. The greater the difference between the two, the greater the transient has been; and an increasing signal can be used to produce a faster rod movement to help correct the situation.

Another part of the reactor control unit is the power mismatch circuit, which utilizes Excore NIS power range inputs and turbine impulse pressure. The objective is to provide a means for fast response to power transients. It would be easy to build a reactor control unit with a circuit just to look at the rate of change of turbine power and use it to tell the rods to move faster. However, to add overall stabilty to the system, turbine power is compared to NIS nuclear power to determine more closely the power mismatch between primary and secondary. This stability aspect can be seen more clearly after the entire reactor control unit is discussed.

Thus there are two basic parts to the reactor control unit - the power mismatch circuit and the temperature error circuit. These two produce error signals which are combined into a signal called compensated temperature error. This signal determines rod speed and direction. A closer look at the individual components of the reactor control unit will provide better insight as to how automatic rod control is accomplished.

A single turbine impulse pressure transmitter (PT-505) is used to represent secondary power levels between 0 and 120 percent. The reference temperature (T_{ref}) for the program is developed by a special unit that takes the turbine power input and creates the program T_{avg} temperature band on the output. These values are currently 547°F and 571°F. The lag unit on the output provides some filtering but also puts a slight lag into signal response.

The actual RCS average temperature input comes from the autioneered high T_{avg} unit. This signal is filtered (lag) and then modified (lead/lag) to improve response. This compensates for the actual delays which occur in sensing reactor coolant temperature changes. The output is a signal representing a temperature between 530°F and 630°F.

Both the T_{avg} and T_{ref} signals then combine in a single summing unit. In mathematical terms the summer performs a $T_{ref} - T_{avg}$ operation. Depending oupon the input signals, the output of the summer could be positive, negative or zero. A positive signal translates to a need for rods to move out. The opposite movement is called for when a negative signal appears. This signal is called temperature error and is combined with the signal from the power mismatch circuit to produce a final output.

The two inputs to the power mismatch circuit represent nuclear power and turbine power in a range of 0 to 120 percent. A box labeled "Difference and Rate" in the figure first takes the difference between the two and then looks at the rate of change of the output. Mathematically, the difference box performs a $Q_t - Q_n$ operation and the rate could be represented by

$$\frac{d(Q_t - Q_n)}{dt},$$

or the rate of change of the difference. Realize, therefore, that an output will only be produced when there is a rate of change between nuclear power and turbine power. Conversely, no output will appear if the inputs are not changing with respect to each other. Even if there is a difference between the two, if it is steady state, there will be no output from the difference and rate box. Realize also that this signal representing the rate of change of the power mismatch can be positive or negative. An increasing nuclear power with respect to turbine power would produce a negative signal which translates to a desire for inward rod movement. Outward rod movement is desired when the output is positive, indicating that turbine power is increasing with respect to nuclear power.

The next circuit component, the non-linear gain unit, functionally performs two operations. As the name implies, it is a signal gain unit. When the rate of change between nuclear power and turbine power is small, the signal input is small. Therefore, the gain provided for the signal is small. For larger inputs the gain is increased. The second function is subtle but important. It converts a rate of change of power mismatch into an equvalent signal in degree farenheit (^{o}F). Notice that the gains described are $0.3^{o}F$/percent and $1.5^{o}F$/percent. The conversion to temperature representation is much more useful to the reactor control unit since temperature is the other input to the final summer. The signal here can also be positive or negative as described for the difference and rate box.

7.03

___ 5.15.6 CLEAR AND TAG the Auxiliary Feedwater Pumps

It may be desirable to use the AFW Pumps to maintain Steam Generator levels if so, C&T may be deferred until pumps are secured.

a. TAG the electrical power supplies for the motor driven pumps

b. TRIP Pump and TAG the local reset lever for the steam driven pump

___ 5.15.7 REDUCE the number of running RCP's to two in one of the following combinations

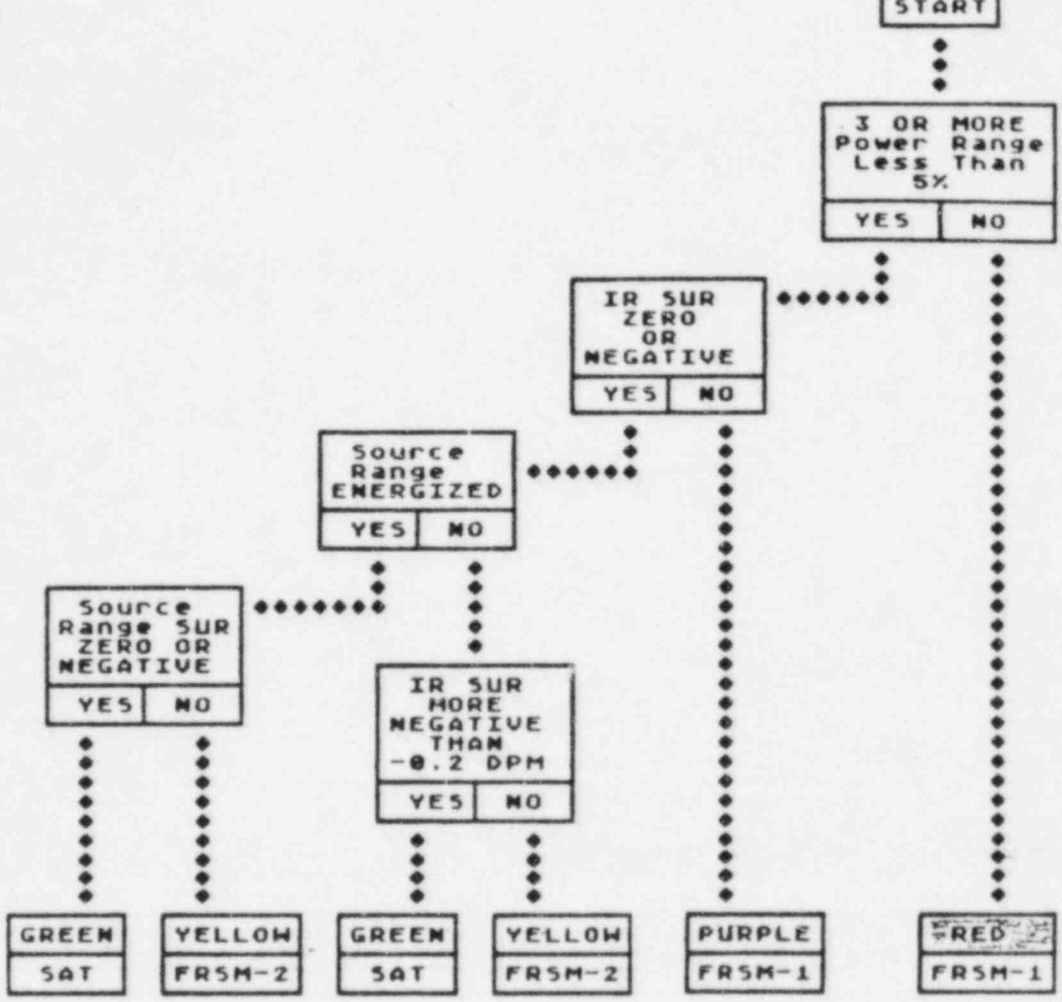
a. 21 RCP and 24 RCP
or
b. 22 RCP and 23 RCP

If Chemical addition is in progress to the RCS and more flow is desirable this step may be marked N/A. Operation of RCP's, other than specified, can result in uneven RCS temperatures among the loops and create conditions in the Steam Generators leading to inadvertent Safety Injections created by Steam Generator differential pressures.

MASTER

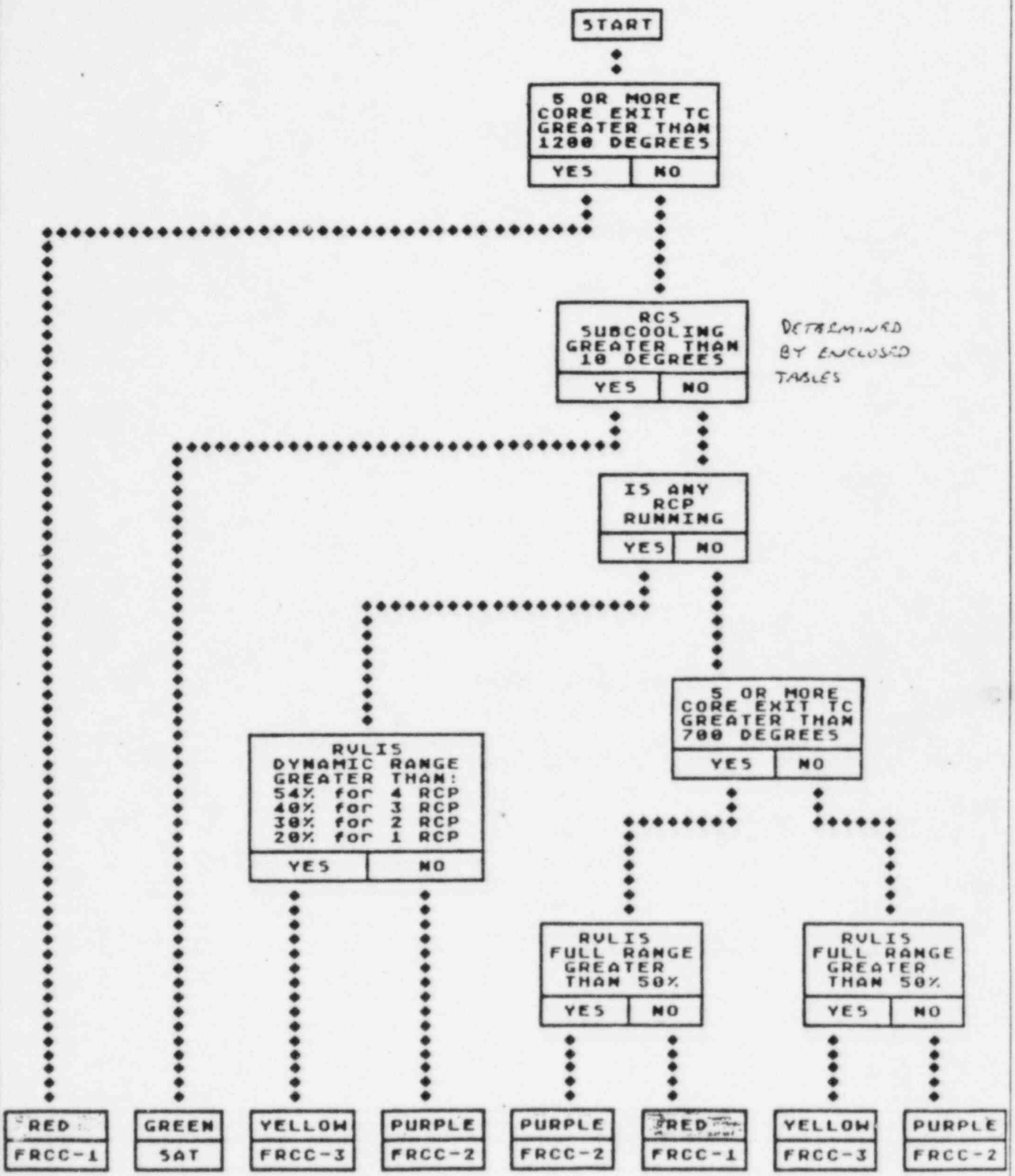
7.04

FIGURE 1
SHUTDOWN MARGIN STATUS TREE



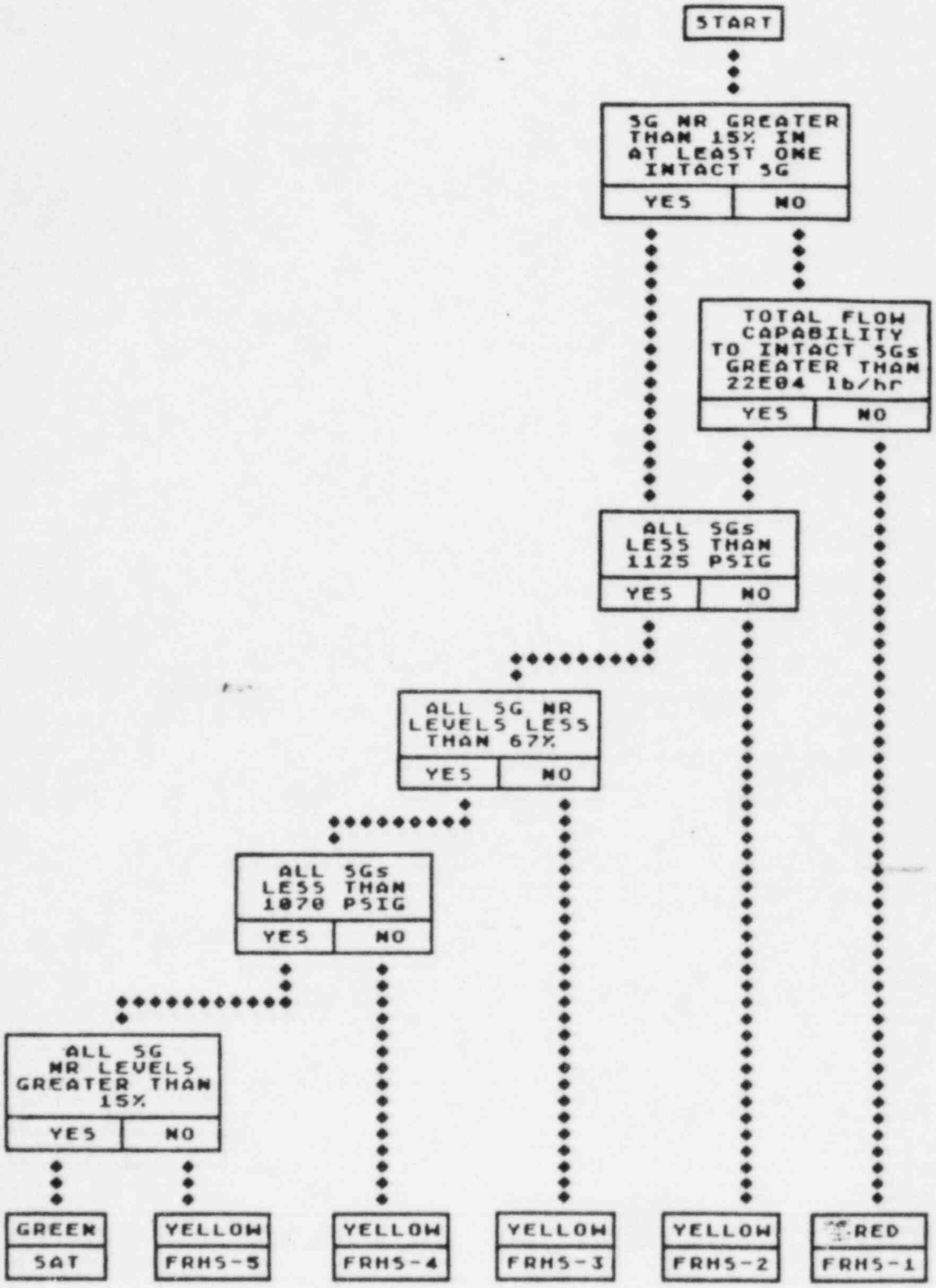
MASTER

FIGURE 2
CORE COOLING STATUS TREE



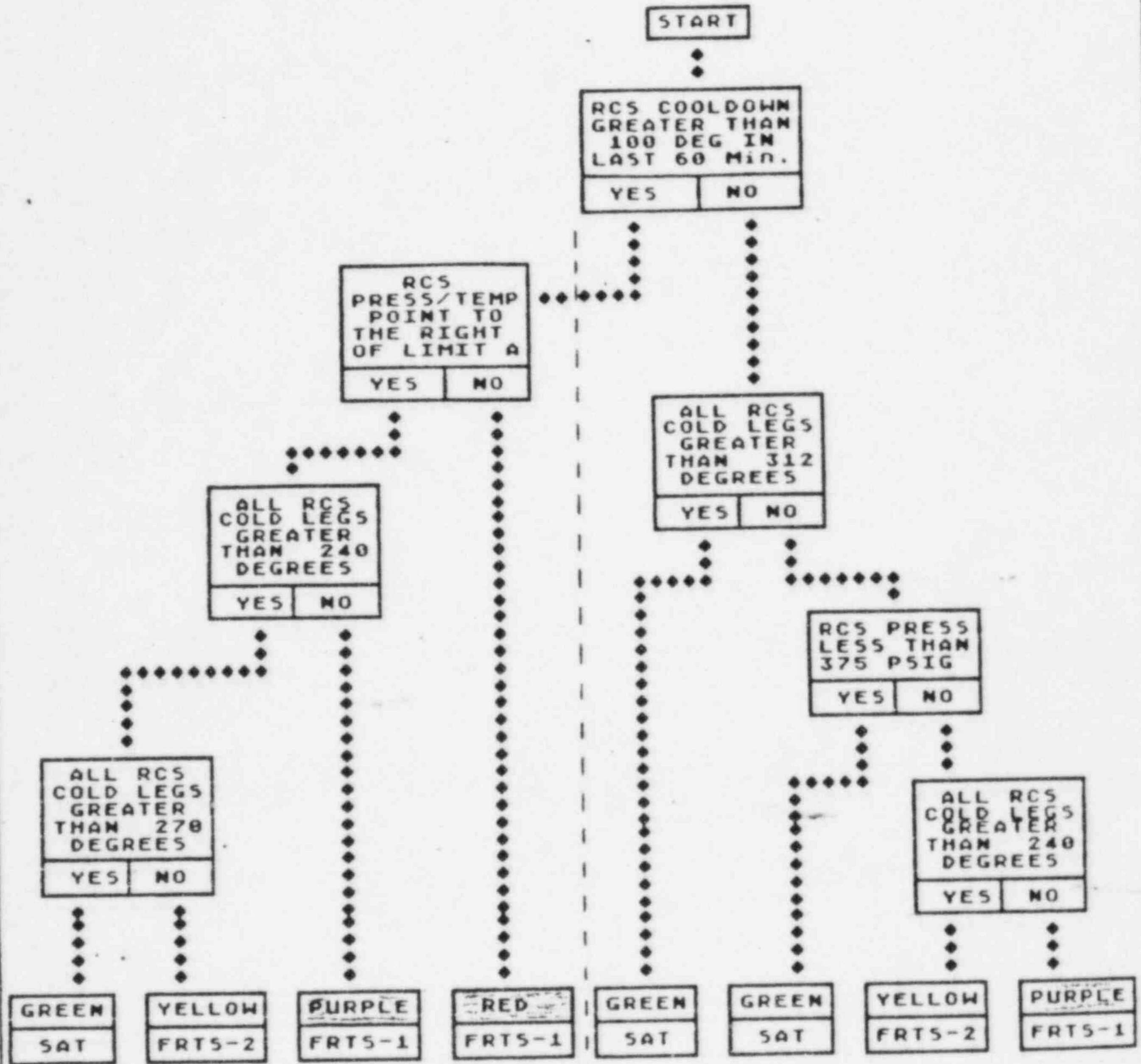
MASTER

FIGURE 3
HEAT SINK STATUS TREE



FLOW CAPABILI of (1) MOTOR DRIVEN AFP @ DISCHARGE PRESS. of 1200

FIGURE 4
THERMAL SHOCK STATUS TREE



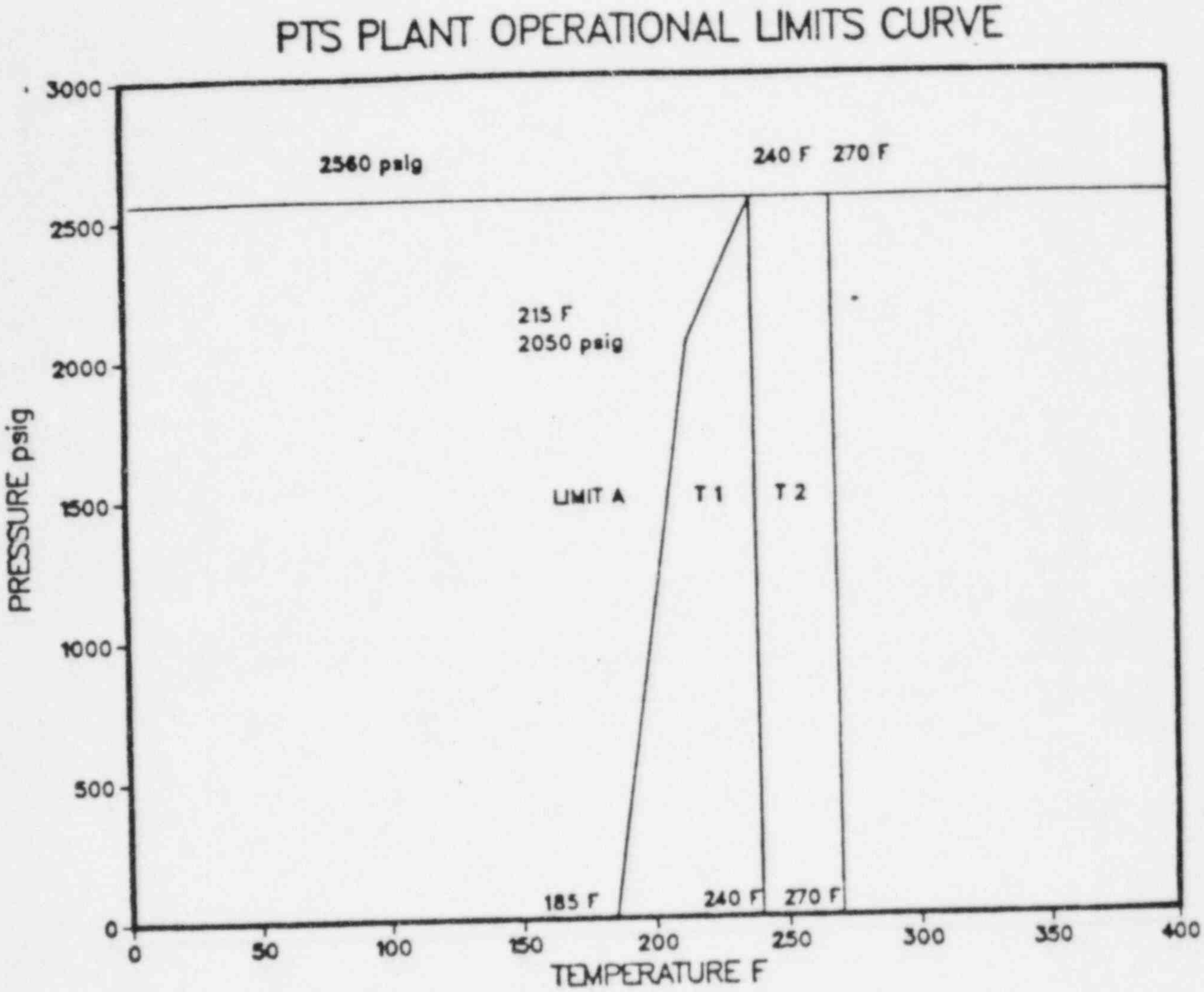
P.1
BnT
5-23

FOR EXCESSIVE COOLDOWN THERMAL STRESS

FOR CONSTANT TEMP AND PRESS ↑ PRESS. STRESS

MASTER

FIGURE 4A
THERMAL SHOCK LIMIT A CURVE



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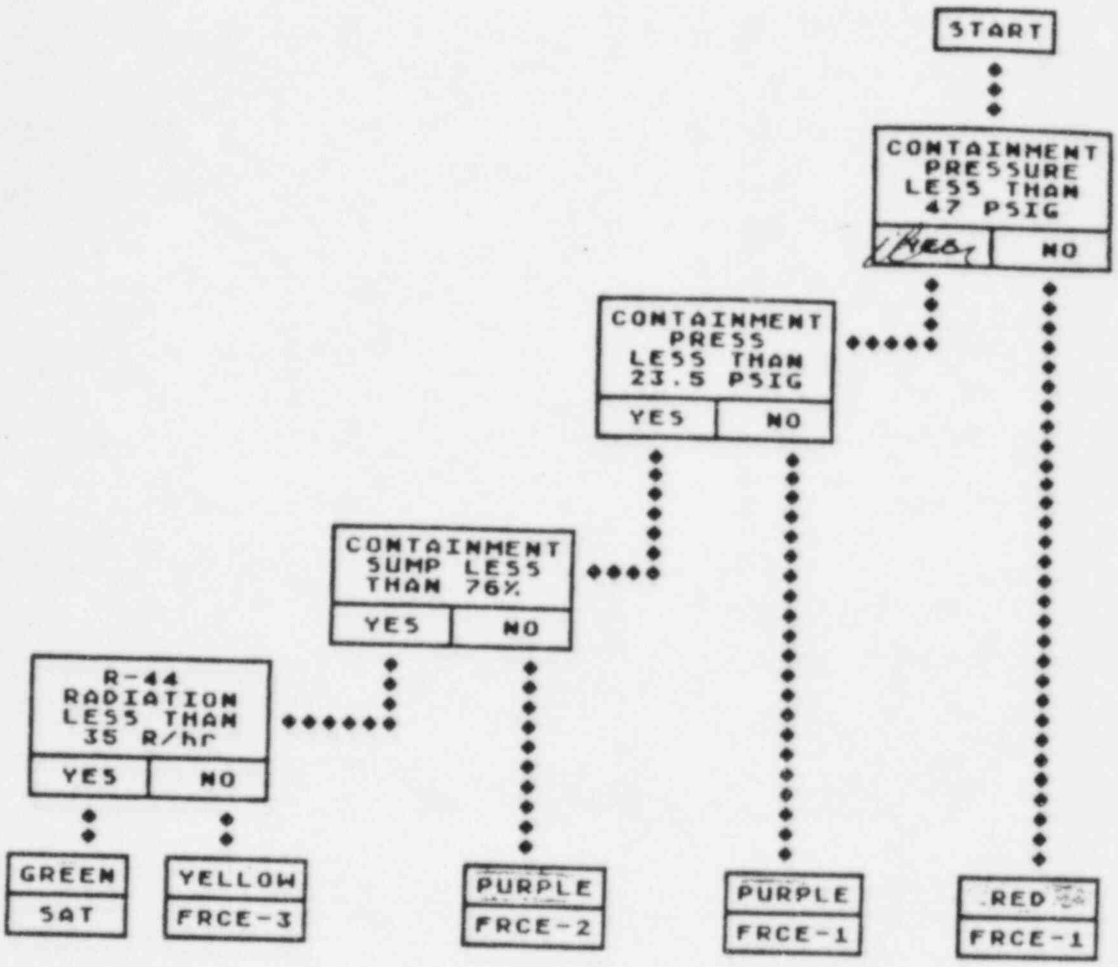
[illegible]

FIGURE 5
CONTAINMENT ENVIRONMENT STATUS TREE



MASTER

6.10.2 Dose limits for females of childbearing age:

7.08

It is recommended by the National Council on Radiation Protection and Measurements and shall be the policy of the station that during the gestation period, the maximum dose equivalent to the fetus from occupational exposure of the expectant mother shall not exceed 0.5 rem (500 millirem). Tl expectant mother shall notify her supervisor who shall infor the RPE of her pregnancy.

6.10.3 Administrative dose limits:

Administrative dose limits are established to ensure that regulatory and station limits are not exceeded during non-emergency conditions. The administrative limits shall b authorized by management and supervisory personnel in define blocks. Doses shall require higher management approval as a individual's accumulated exposure for the quarter approaches the regulatory limit.

The administrative dose limits that shall be used to control personnel radiation exposure during non-emergency conditions for personnel eighteen years of age and older are as follows

NOTE

If designated personnel are not available for approval, a Senior Shift Supervisor and the Shift Radiation Protection Technician may authorize the exposure.

100 mrem/Q +100	Current quarter exposure with no official documentation (may be increased after evaluation by Senior Supervisor-RP)	Technical Supervisor-RP
1000mrem/Q +100		Automatic authorization at the start of each quarter
2000mrem/Q +100	NRC Form 4 must be complete and individual must be 19 years old.	Senior Supervisor-RP and individual's Senior supervisor or equivalent
2500mrem/Q 3000mrem/Y 4000mrem/Y	No exposure extension is allowed for levels above 2500mrem/Q for planned operations.	Department head, Radiation Protection Engineer and the General Manager-Salem Operations
>5000mrem/Y	This level is considered an emergency exposure level and requires special authorization	Vice President-Nuclear

The Senior Supervisor-RP shall be responsible for establishing procedures to implement the administrative control limits.

7.09 a

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABl 3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
Three Loops	1 T_{avg}/operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops		15
OR, COINCIDENT WITH					
Steam Line Pressure-Low				1, 2, 3##	
Four Loops Operating	1 pressure/ loop	1 pressure in any 2 loops	1 pressure in any 3 loops		14*
Three Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops		15
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water level--High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2, 3	14*
6. SAFEGUARDS EQUIPMENT CONTROL SYSTEM (SEC)	3	2	3	1, 2, 3, 4	13
7. UNDERVOLTAGE, VITAL BUS					
a. Loss of Voltage	3	2	3	1, 2, 3	14*
b. Sustained Degraded Voltage	3	2	3	1, 2, 3	14*

7.09 b

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3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

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

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

b. Three separate and independent diesel generators with:

   1. Separate day tanks containing a minimum volume of 130 gallons of fuel, and

   2. A common fuel storage system consisting of two storage tanks, each containing a minimum volume of 20,000 gallons of fuel, and two fuel transfer pumps.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.2 within one hour and at least once per 8 hours thereafter; restore at least two offsite circuits and three diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.2 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN

* One inoperable fuel transfer pump is equivalent to one inoperable diesel generator.

7.10C

PART V

PRESSURIZER HEATER FAILURE

2.0 INITIAL CONDITIONS

2.1 Overhead Annunciators

2.1.1 REACTOR COOLANT LOW PRESS (E-11)

2.1.2 PRESSURIZER LOW PRESS (E-19)

2.1.3 REACTOR COOLANT LOW PRESS HEATERS ON (E-27)

2.2 Reactor Coolant - high pressure alarm (Console)

2.2.1 High Pressure Deviation Alarm (Console)

3.0 IMMEDIATE ACTIONS

3.1 Automatic

3.1.1 As pressurizer pressure decreases, a reactor trip will occur at 1865 psig and a safety injection at 1765 psig. REFER to EI I-4.3 Reactor Trip or EI I-4.0 Safety Injection Initiation as appropriate.

3.2 Manual

3.2.1 ATTEMPT to energize the Control Group Heaters by depressing its ON pushbutton.

3.2.2 VERIFY all Pressurizer Heaters are in AUTO.

3.2.3 VERIFY Pressurizer Pressure Controller is in AUTO.

3.2.4 ATTEMPT to energize either or both Groups of Backup Heaters.

3.2.5 If pressure is high and pressurizer heaters are energized, ATTEMPT to de-energize the heaters.

4.0 SUBSEQUENT ACTIONS	COMMENTS
4.1 Dispatch an operator to Local Panel Elev 78 Penetration. VERIFY Local Breaker position, RESET if Tripped before swap to Emer. Sup.	Tech Spec 3.4.4 Power operation may continue up to 72 hours, if only one group of heaters has failed.

MASTER

4.2 If all pressurizer heaters are de-energized and cannot be energized from their normal supply, PROCEED as follows:	The emergency vital power supply for pressurizer heaters are designed to maintain pressure during a natural circulation cooldown. These heaters will not be sufficient to maintain pressure with Reactor Coolant Pumps running.
4.2.1 TAKE manual control of pressurizer level.	
4.2.2 SLOWLY REDUCE reactor power IAW IOP-4 Power Operation and IOP-5 Hot Standby to Minimum Load while attempting to maintain pressure with pressurizer level.	By maintaining or raising pressurizer level during the power reduction, the pressure decay can be minimized.
4.2.3 When the reactor is shutdown and Tavg is less than 541°F, STOP the No. 11 and 13 RCPs.	This will stop Pressurizer Spray Valve bypass flow which will reduce the rate of depressurization.
4.2.4 If the heaters cannot be energized in 72 hours, PERFORM a normal cooldown IAW IOP-6 Hot Standby to Cold Shutdown.	Tech Spec 3.4.4
a. Should it become necessary to initiate pressurizer spray to reduce pressure during the cooldown, START, No. 13 RCP.	
b. INITIATE Aux Spray if Delta T is less than 320°F via CVCS.	

7.10e

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3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3.1, and all equipment hatches are closed and sealed.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

c. After each closing of a penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at Pa (47 psig) and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

*Except vents, drains, test connections, etc. which are (1) one inch nominal pipe diameter or less, (2) located inside the containment, and (3) locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed at least once per 92 days.

8.02

18. Operational Status Change (con'd)

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
E. Event that results (or should have resulted) in ECCS discharge as a result of a valid signal (manual or automatic)	1. ECCS automatic initiation set point reduced or manual initiation; OR 2. Lit logic lights for initiation on RP4; AND 3. Discharge to RCS is called for or verified by control console indication (flow, valve positions, tank levels, etc.)	ATT 1 UE
F. Event that results in manual or automatic actuation of Engineered Safety Features (ESF), including the Reactor Protection System (RPS)	1. As stated, except resulting from and part of a planned test Examples: a) SEC Mode Operation b) Rx trip c) SI (without water into RCS) d) Containment isolation	ATT 29 (50.72b - 4
G. Scheduled shutdown for testing, maintenance refueling	1. Any planned shutdown	ATT 32 Other
H. Derating caused by regulatory action	1. Upon official notification from site management or NRC	ATT 32 Other
I. Any reactivity anomaly	1. Disagreement with predicted steady state reactivity balance during power ops $\geq$ 1000 pcm; OR 2. Calculated SDM < required; OR 3. SUR $\geq$ 5.2 dpm; OR 4. Unplanned positive reactivity insertion > 500 pcm	ATT 13 Other
J. Any unplanned criticality	1. As stated	ATT 12 UE
K. Any event or condition requiring unit shutdown to comply with any T/S LCO	1. Shutdown initiated	ATT 1 UE

ATTACHMENT 29

Each step shall be initialed by the responsible individual when completed.

I. **Notifications**

___ SSS 1. Complete Section I of NRC Data Sheet (page 2 of this attachment).

___ SSS 2. Notify Operations Manager and confirm classification of event.

L. Fry (Office - 4523; Car - [609]342-5103; Beeper - [800]612-4532; Home - [609]678-7634)

or

OPS. ENGINEER IS BACKUP ⇒ L. Catalfomo (Office - 4522; Beeper - [800]612-4534; Home - [609]678-3176; Car - [302] 428-9084)

notified ______________ hrs on ______________
time date

___ SSS 3. If this attachment is being utilized as a result of a reactor trip notify the General Manager - Salem Operations of the event.

J. Zupko (Office - 4500; Car - [609]342-5036; Beeper - [609]342-5803; Home - [609]468-5527)

ASS'T. GM IS BACKUP ⇒ L. Miller (Office - 4497; Car - [609]342-5077; Beeper - [800]612-4531; Home - [609]769-1727)

___ SSS 4. Notify LAC Dispatcher of event (direct line, or [609] - 935-7300).

______________ notified at ______________ hrs on ______________
name time date

SSS 5. Notify Public Affairs Manager - Nuclear or Alternate with details of event:

Public Affairs Manager - Nuclear (Contact One)

- R. Silverio	Office: 4699 Home: 829-1546 Beeper: 342-5804
- W. Denman	Office: 4480 Home: 935-6349 Beeper: 342-5849
- B. Gorman	Office: 4480 Home: 228-1089 Beeper: 342-5851

notified ______________ hrs on ______________.
time date

___ SSS

6. Notify NRC Operations Center (ENS line, [202]951-0550, or [301]427-4056), [301]427-4259, or [301]492-8893 of the event **within 4 hours.** Use NRC Data Sheet to record additional information provided to the NRC.

____________________ notified at______ hrs on ________.
name time date

___ SSS

7. If this attachment is being utilized as a result of a reactor trip notify the NRC Resident Inspector **within 4 hours.**

J. Linville - (Office - 4479; Home - [609]243-4998)

R. Summers - (Office - 4479; Home - [609]848-6741)

II. Reporting (STA FUNCTION)

SSS — 1. Ensure that an Incident Report is prepared in accordance with AP-6.

SSS — 2. Forward this attachment, along with the Incident Report and any supporting documentation, to the Senior Operations Technical Supervisor.

SOTS — 3. Review Incident Report and any other available relevant information for correct classification of event and corrective action taken.

SOTS — 4. Contact the LER Coordinator and request written followup (**required 30 days after event**). Provide this attachment and any other supporting documentation received from the SSS.

LER — 5. Prepare Licensee Event Report.

LER number ________________

LER — 6. Follow-up action: response requested from _____________

Response request number ______________________

LER — 7. Prepare Licensee Event Report.

LER Number____________________

LER — 8. Return this attachment to the Operations Manager.

8.07a

ADMINISTRATIVE PROCEDURES
SALEM GENERATING STATION
ADMINISTRATIVE PROCEDURE NO. 13
TEMPORARY JUMPERS, PUMPS, AND LIFTED LEADS

1.0 PURPOSE

This procedure describes the program for controlling the use of temporary electrical or mechanical jumpers, pumps and the lifting of leads.

2.0 SCOPE

This procedure addresses temporary electrical/mechanical jumpers and lifted leads that remain in any system upon completion of a procedure or active troubleshooting. These jumpers/lifted leads are not to be confused with jumpers/lifted leads utilized while performing: active troubleshooting, calibration, testing, maintenance, or an approved procedure which requires installation and subsequent removal of the temporary jumpers/lifted leads.

This procedure also addresses all temporary pumps which are installed in the controlled access area or installed outside the controlled access are to carry radioactive materials that are intended to be removed from the system and not be part of an approved design change. The potential of a temporary pump to transfer hazardous materials which are present due to a spill or other such accident shall be addressed in the review of all temporary pump requests, regardless of whether radioactive materials are to be transported by the pumps or not.

3.0 REFERENCES

3.1 NRC Clarification of 10CFR50.59 Requirements for Jumpers/Lifted Leads and Procedures (AITS: F01004908)

3.2 INPO Good Practice OP-202, Temporary Bypass, Jumper, and Lifted Lead Control

3.3 AP-8, Design Change, Test and Experiment Program

3.4 AP-9, Maintenance Program

3.5 AP-11, Record Retention Program

4.0 DEFINITIONS

4.1 Mechanical Jumper: A piece of piping, hose, spoolpiece, blank or blind flange, or tubing which join two or more systems together or bypasses a component (s) within a system, thus altering the systems design or configuration.

4.2 Electrical Jumper: A wiring connection used in a component or circuit which bypasses a component within an electrical circuit, thus modifying the circuit design or configuration.

8.07 C

5.1.3 Indicates switchable equipment within the work area which is properly isolated, but which is unsafe to work on because of inadequate clearance to other energized equipment (see table 1).

TABLE 1
MINIMUM CLEARANCE TO ENERGIZED EQUIPMENT

phase-to-phase voltage (KV)	minimum clearance
4-13	H + 2'0"
26	H + 2'6"
69	H + 3'
138	H + 3'4"
230	H + 5'
345	H + 7'
500	H + 11'

Where "H" is equal to the body height with arms fully extended.

NOTE

Rubber gloves and sleeves shall be required to install barriers within 6 feet of energized equipment.

5.1.4 Identifies the person(s) for whom the equipment has been cleared and tagged.

5.2 Yellow Permissive Tag - serves the following purposes:

5.2.1 Distinctly marks electrical equipment that is safe for work and is used only in conjunction with red blocking tags on high voltage electrical equipment.

5.2.2 Identifies the person for whom the equipment has been cleared and tagged.

5.2.3 Gives the person named on the tag the authority to work on, operate, or adjust the equipment.

5.2.4 Prohibits operation of the equipment by any other person unless permission is granted by the person named on the yellow tag.

5.3 White Caution Tag - serves the following purposes:

5.3.1 Indicates, as part of a switching procedure, a position other than the normal operating position of electrical equipment.

5.3.2 Indicates an abnormal condition or limitation of the equipment. The tag shall identify the abnormal condition and may include special instructions of a temporary nature. **These instructions shall be complied with under all conditions and circumstances.**

8.07d

5.3.3 The white caution tag shall not be used on blocking points to isolate equipment from energy sources for the protection of personnel or equipment.

5.4 Workers Blocking Tag - serves the following purposes:

5.4.1 Identifies the mechanical or low voltage electrical blocking points between any circuit or equipment that is energized and the de-energized equipment upon which work is to be performed.

5.4.2 Blocks and prohibits the operation of equipment by any individual other than the worker named on the tag or persons directed by him/her.

5.4.3 Indicates that the equipment is tagged at the request of the person in charge (named on the tag) and that the worker (named on the tag) may operate the equipment.

5.4.4 This tag cannot be applied to a component which has any other safety tag.

5.4.5 This tag is authorized for use only by Electric Transmission and Distribution personnel as set forth in Section 8.19.

5.4.6 This tag is red with a yellow stripe.

6.0 DESCRIPTION AND USE OF TAGGING REQUESTS

6.1 All forms associated with the safety tagging system are contained in Operations Directive-8, "TRIS Tagging Operations". The following is a list and brief description of each form:

6.1.1 Tagging Request Form (OD-8-A-1) - This form shall be used whenever red blocking tags or yellow permissive tags are needed for the protection of personnel or equipment.

6.1.2 Addendum to Tagging Request (OD-8-A-2) - This form shall be used in conjunction with the Tagging Request Form when specific blocking points are requested to be tagged.

6.1.3 Group Tagging Request Authorization Sheet (OD-8-A-3) This form shall be used only when a tagging request is made under group tagging rules.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

8.08a

Note 2: Overpower $\Delta T \leq \Delta T_o[K_4 - K_5 \frac{\tau_3 S}{1+\tau_3 S} T - K_6 (T-T'') - f_2(\Delta I)]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T'' = Reference T_{avg} at RATED THERMAL POWER $\leq$ 577.9°F

K_4 = 1.080

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature

K_6 = 0.00119/°F for $T > T''$; $K_6 = 0$ for $T \leq T''$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator, Sec^{-1}.

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 4 percent.

000124

SALEM GENERATING STATION

OPERATIONS DEPARTMENT DOCUMENT APPROVAL COVER SHEET

Title: REMOVING, RETURNING TO SERVICE AND LOSS OF PROTECTIVE SYSTEM CHANNEL

No.: IV-10.3.1 Unit: 1 Rev.: 3

Remarks: 3 pages of text, 10 tables, last OTSC - P-1
Revised to incorporate output function - Axial Flux difference.

Safety Related Review (Ref. AD-13): S/R yes X no ____

Author's Checklist Completed: yes X

Author C. Bowden Date 6-25-84

SRO N M Ball Date 7-12-84

Ops. Eng. [illegible] Date 7-17-84

SOTS+ NA [illegible] Date ____

Ops. Mgr. [illegible] Date 7/18/84

QA** [illegible] 7-28-84 D C Perkins Date 7/30/84

SORC** S.K. Mille 84-99B Date 8-1-84

General Manager** JM Zupko [illegible] Date 8/1/84

* required for SPM documents only
\+ required for EOP validation acceptance only
** required for safety related documents and fire protection documents

MASTER

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OPERATING PROCEDURE
IV-10.3.1
REMOVING, RETURNING TO SERVICE
AND LOSS OF PROTECTIVE SYSTEM CHANNEL

1.0 PURPOSE

1.1 This procedure provides the instructions necessary for the following:

1.1.1 Operation of plant with Loss of Protective System Channel.

1.1.2 Removing and Returning to Service a Protection System Channel.

2.0 INITIAL CONDITIONS

2.1 The failed or out of service Protective Channel is required for plant operation per Technical Specification for the current mode of plant operation.

3.0 PRECAUTIONS

3.1 A failed or out of service Protective Channel will require complying with an "Action Statement" per Technical Specifications.

3.2 When maintenance or testing is performed on a Protective Channel, verify that no other channel in the logic is energized which together with the subject channel being energized will cause the Logic Train to initiate an inadvertant automatic protection system actuation.

4.0 ATTACHMENTS LIST

4.1 Tables

4.1.1 Table 1 - Nuclear Instrumentation-Power Range

4.1.2 Table 2 - Reactor Coolant System-RCP Flow

4.1.3 Table 3 - Reactor Coolant System-Loop Temperature

4.1.4 Table 4 - Reactor Coolant System-Pressurizer

4.1.5 Table 5 - Steam Generator-No. 11

4.1.6 Table 6 - Steam Generator-No. 12

4.1.7 Table 7 - Steam Generator-No. 13

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TABLE 1
NUCLEAR INSTRUMENTATION POWER RANGE

INOP DEV	BISTABLE SWITCH	PROTECTION RACK NO.	OUTPUT FUNCTION(s)	REMARKS
N-41	1BS-411C	2	OT Delta T, 2/4 Reactor Trip	
	1BS-411D	2	OT Delta T, 2/4 Auto. Turb. Runback and Block Auto and Manual Rod Withdrawal	
	Place the following switches in the indicated position:	NI Racks No. 81	Switches: 1. Upper Section Deviation Channel Defeat 2. Lower Section Deviation Channel Defeat 3. Rod Stop Bypass 4. Power Mismatch Bypass 5. Comparator Channel Defeat	1. Remove Control Power Fuses After switches on NI Rack 81 repositioned.
		Control Rack 26	6. Axial Flux difference	Test Position
N-42	1BS-421C	6	OT Delta T, 2/4 Reactor Trip	
	1BS-421D	6	OT Delta T, 2/4 Auto. Turb. Runback and Block Auto and Manual Rod Withdrawal	
	Place the following switches in the indicated position:	NI Rack No. 81	Switches: 1. Upper Section Deviation Channel Defeat 2. Lower Section Deviation Channel Defeat 3. Rod Stop Bypass 4. Power Mismatch Bypass 5. Comparator Channel Defeat	1. Remove Control Power Fuses After switches on NI Rack 81 repositioned.
		Control Rack 26	6. Axial Flux Difference	Test Position

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INOP DEV	BISTABLE SWITCH	PROTECTION RACK NO.	OUTPUT FUNCTION(s)	REMARKS
N-43	1BS-431C	13	OT Delta T, 2/4 Reactor Trip	
	1BS-431D	13	OT Delta T, 2/4 Auto. Turb. Runback and Block Auto and Manual Rod Withdrawal	
	Place the following switches in the indicated position:	NI Racks No. 81	Switches: 1. Upper Section Deviation Channel Defeat 2. Lower Section Deviation Channel Defeat 3. Rod Stop Bypass 4. Power Mismatch Bypass 5. Comparator Channel Defeat	1. Remove Control Power Fuses After switches on NI Rack 81 reposi-tioned.
		Control Rack 26	6. Axial Flux Difference	Test Position
N-44	1BS-441C	15	OT Delta T, 2/4 Reactor Trip	
	1BS-441D	15	OT Delta T, 2/4 Auto. Turb. Runback and Block Auto and Manual Rod Withdrawal	
	Place the following switches in the indicated position:	NI Rack No. 81	Switches: 1. Upper Section Deviation Channel Defeat 2. Lower Section Deviation Channel Defeat 3. Rod Stop Bypass 4. Power Mismatch Bypass 5. Comparator Channel Defeat	1. Remove Control Power Fuses After switches on NI Rack 81 reposi-tioned.
		Control Rack 26	6. Axial Flux Difference	Test Position

MASTER

8.08 b

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

* Excluding decay heat.

** Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

8.10a

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 40 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2230 ± 20 psig.

f. 1 GPM leakage at a Reactor Coolant System pressure of 2230 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by;

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory at least once per 12 hours.

8.11b

1. Primary Leakage

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Primary leak a) no PRESSURE BOUNDARY leakage b) 1 gpm UNIDENTIFIED leakage c) 10 gpm IDENTIFIED leakage d) 40 gpm CONTROLLED leakage e) RCS PRESS ISOL VALVES (See T/S)	1. Exceeding action statements of T/S LCO 3.4.6.2 (Unit 1) or 3.4.7.2 (Unit 2) See Tech Spec Action Requirements	ATT 1 UE
B. Primary leak > 50 gpm	1. One chg pmp cannot maintain level	ATT 2 A
C. Known LOCA greater than total makeup capacity	1. Low/decreasing PZR level with maximum charging flow	ATT 3 SAE

Refer to Section 5 prior to classification

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
D. PZR safety/PORV failure to reseat	1. PZR press >2200 psig & POPS not armed, or PZR press <375 psig & POPS armed; AND 2. PORV/safety valve tailpipe hi temp, or PRT temp, press, or level increasing	ATT 1 UE
E. Pipe cracks in stagnant borated water systems	1. Cracks in weld areas of safety related piping (as reported by Engineering or ISI/MIET) IE Information Notice - one hour	ATT 6 (50.72b-1H:

6. Radiological Releases

Note: Action levels listed are for valid RMS channel indications. The validity of the indication should be confirmed by sample analysis or other means as necessary.

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Accidental, unplanned, or uncontrolled gaseous release, that exceeds 2 times the applicable concentrations of the limits specified in Appendix B, Table II of Part 20, to unrestricted areas (averaged over 60 min)	1. R41C > 1.5E3 cpm for > 60 min; **OR** 2. R41B > 5.7E3 cpm increase in 60 min	ATT 29 (50.72b - 4Hr)
B. Accidental, unplanned, or uncontrolled liquid release	1. Any unplanned or uncontrolled liquid release outside the controlled access area.	ATT 29 (50.72b - 4Hr
C. Liquid release that exceeds T/S limits for $\geq$ 15 min	1. R18 alarm and no isolation **AND** Confirmed analysis of liquid waste effluent indicating discharge exceeding T/S limits. **OR** 2. Any R19 alarm and blowdown to 12(22) S/G blowdown tank	ATT 7 UE
D. Gaseous release that exceeds T/S limits for > 60 min	1. R41C > 1.1E5 for > 60 min; **OR** 2. R41B > 2.1E4 cpm increas in 60 min;	ATT 7 UE
E. Gaseous release that exceeds 10 times T/S limits for > 15 min	1. R41C > 1.1E6 cpm for > 15 min; **OR** 2. R41B > 5.3E4 cpm increase in 15 min; **OR** 3. R45B > 7.1E-3 uCi/cc for > 15 min	ATT 8 A

18. Operational Status Changes

Initiating Event/ Condition	Emergency Action Level	Notification/ Reporting
A. Any event or condition during operation that results in the condition of the plant, including principal safety barriers, being seriously degraded	1. As judged by the SSS/EDO	ATT 2 A
B. Any event or condition during operation that results in the plant being: 1) in an unanalyzed condition that significantly compromises plant safety; 2) in a condition outside the design basis; 3) in a condition not covered by operating or emergency procedures	Examples: a) accumulation of voids that could inhibit the ability to adequately remove heat from the reactor core b) voiding in instrument lines, resulting in erroneous indication, causing the operator to misunderstand the true condition of the plant	ATT 6 (50.72b -
C. Any event, found while shutdown, that had it been found during operation would have resulted in the plant, including principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety	1) As judged by SSS/EDO	ATT 29 (50.72b -
D. Any deviation from T/S or license condition in an emergency when action is needed to protect the public health and safety (action <u>must</u> be approved at least by a licensed SRO)	1. Action required because no action consistent with license and ./S can provide adequate or equivalent protection	ATT 6 (50.72b -

Attachment 4

ATTACHMENT 4

NRC Resolution of Facility Comments on SRO Exam

Question	Resolution
5.01.a	Subjective comment - considered in grading
5.01.b	Included
5.02	Included
5.04	Included
5.05.a	Included
5.05.b	Added for information
5.06.a	Included
5.06.b	Not changed
5.07	Included
5.08.a	Included
5.09.a	Subjective comment - considered in grading
5.09.b and c	Included
6.01.a	Included
6.01.b	Not changed
6.02	Not changed
6.03	Not changed
6.04.a	Not changed
6.04.b	Added for information
6.05.a	Corrected
6.05.c	Included
6.06.a	Added for information
6.06.b	Added "Service Water for MS-10"
6.07	Included
6.08.a	Included
6.09	Included
6.10.b	Included
6.10.d(3)	Included
7.03	Deleted question
7.04	Included
7.05	Included
7.08	Included
7.09	Included
7.10.c	No changed
7.10e	Included
8.01	Included
8.02	Accepted "STA, NRC, and Local"
8.04	Not changed
8.07.a and b	Not changed
8.07.c	Included
8.07.d	Included first part only
8.07.e	Changed answer to "None"
8.08.a and c	Included
8.08.b	Not Changed
8.10.a	Not changed
8.11.b	Included